14.12 MAIN STEAM LINE BREAK ACCIDENT

14.12.1 General

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

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 If the most reactive control element assembly (CEA) is assumed stuck in its fully withdrawn position after reactor trip, there is an increased possibility that the core will return to power and criticality.

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

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

14.12.2 Applicable Industrial and Regulatory Requirements

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

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

Although DNB and possible clad perforation following a steam pipe rupture are not necessarily unacceptable, the following analysis, in fact, shows that no DNB occurs for any rupture assuming the most reactive control element assembly stuck in its fully withdrawn position.

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

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

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

14.12.3 Method of Analysis

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

14.12.4 Inputs and Assumptions

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

14.12.5 Results

The MSLB event case initiated from HFP was simulated for 200 seconds using CESEC C89300mod5 with parameters that maximize the potential for Return to Power (R-T-P) or/and Return to Criticality (R-T-C). The limiting MSLB accident occurs with all RCPs running. The results of this case is per Table 14.12-4. This case shows a peak R-T-P of 18.50% and a peak reactivity of -0.069%$\Delta\rho$. The peak LHR and minimum DNBR did not violate their respective SAFDLs.

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

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

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

14.12.6 Radiological Consequences of a MSLB

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

14.12.6.1 Methods of Analysis

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

Based on 1 Rod:

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

where,

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

E_γ or β = Average Gamma or Beta Decay Energy of Xe-133 (Mev/Disintegration)

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

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

Based on 1 Rod:

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

where,

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

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

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

1. First Find Total Heat Generation Rate During Cooldown

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

where

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

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

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

2. Calculate Steam Release Rate

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

where

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

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

H_{AFW} = Enthalpy of AFW Flow

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

Whole Body Dose =

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

where,

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

N = Number Of Failed Rods

V_{RCS} = Minimum RCS Volume (GAL)

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

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

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

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

t = Time Interval (Min)

Thyroid Dose =

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

where,

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

Additional input values are obtained from Section 14.12.6-2.

14.12.6.2 Inputs and Assumptions

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

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

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

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

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

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

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

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

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

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

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

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

14.12.6.3 Results

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

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

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

14.12.7 Affected Plant Technical Specifications

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

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

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

14.12.8 Affected Plant Systems

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

14.12.9 Limiting Parameters for Reload Analysis

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

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

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

14.12.10 Specific References

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Parameter	Units	Cycle 19
Initial Core Power*	MWt	1.0
Initial Core Inlet Temperature	°F	532
Initial Pressurizer Pressure	psia	2172
Initial Steam Generator Pressure	psia	890
Initial Steam Generator Water Mass Inventory	lbm	123,685
RCS Flow Rate	gpm	197,000
Minimum CEA Worth Available at Trip (Shutdown Margin)	$\%\Delta\rho$	-4.0
Doppler Multiplier		1.4003
Moderate Cooldown Curve	$\%\Delta\rho$ vs temp.	See Figure 14.12-1
Effective MTC	$\times 10^{-4}\Delta\rho/°F$	-3.5
Inverse Boron Worth	$ppm/\%\Delta\rho$	-110.4
βFraction (including uncertainty)		0.005223
Min. MSL Stop Valve Closure Time	sec	4
SG Low Pressure Trip (MS + MF iso) (includes a 22 psi margin)	psia	478 psia

* Reactor coolant pump heat assumed to be zero.

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

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

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

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

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

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

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

Table 14.12-5 - "Fuel Fission Product Inventory"

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

(1) Assumes all rods have burnup of 51,000 MWD/MTU, maximum for three 18 month cycles, 4.05 w/o enrichment 2700 Mwt. (This inventory bounds the inventory associated with 4.5 w/o at 1500 mWt for Fort Calhoun Station.)

(2) Assumes 10% of isotopes released to gap as per Regulatory Guide 1.77

Table 14.12-6 - "Noble Gas Isotopic Parameters"

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

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

Radionuclide	Beta Skin DCF ($Rem\text{-}m^3/Ci\text{-}s$)	Whole Body Gamma DCF	β-DCF	γ-DCF
Kr-83m	0	5.02×10^{-6}	0	.000538
Kr-85m	4.64×10^{-2}	3.72×10^{-2}	4.79	3.99
Kr-85	4.25×10^{-2}	5.25×10^{-4}	4.38	.0563
Kr-87	3.08×10^{-1}	1.87×10^{-1}	31.8	20.0
Kr-88	7.50×10^{-2}	4.64×10^{-1}	7.74	49.7
Xe-133m	3.14×10^{-2}	8.00×10^{-3}	3.24	.857
Xe-133	9.69×10^{-3}	9.33×10^{-3}	1.00	1.00
Xe-135m	2.25×10^{-2}	9.92×10^{-2}	2.32	10.6
Xe-135	5.89×10^{-2}	5.72×10^{-2}	6.07	6.13
Xe-138	1.31×10^{-1}	2.81×10^{-1}	13.5	30.1

Table 14.12-8 - "Iodine Isotopic Parameters"

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

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

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

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

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

14.13 CONTROL ELEMENT ASSEMBLY EJECTION ACCIDENT

14.13.1 General

The CEA ejection accident is defined as the mechanical failure in the form of a complete circumferential rupture of a CEDM housing or nozzle on the reactor vessel head resulting in the ejection of a control rod. The consequence of this mechanical failure is a rapid reactivity insertion which when combined with an adverse power distribution may result in localized fuel damage. The CEA ejection accident was analyzed for Framatome ANP Richland, Inc. fuel (Reference 14.13-11) and reevaluated for Westinghouse fuel (Reference 14.13-10) for Cycle 20.

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

In the following analysis, it is assumed that a CEA is ejected instantaneously from the core, although no mechanism for such an event has been identified.

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

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

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

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

14.13.2 Method of Analysis

14.13.2.1 WestingHouse Fuel

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

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

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

In the hot spot analysis, the transient temperature distribution in a cross-section of a metal-clad uranium dioxide fuel rod, and the heat flux at the surface of the rod, is calculated, using as input, the nuclear power versus time and the local coolant conditions. The Zirconium-water reaction is explicitly represented, and all material properties are represented as functions of temperature.

A parabolic radial power distribution is used within the fuel rod. At hot full power conditions, the radial power distribution in the fuel pellet is represented by an inverted parabola which has been flattened so as to place the emphasis on the pellet centerline. This assumption causes the center of the pellet to heat up and maximizes the fraction of the fuel melt for these cases. At hot zero power conditions, the radial power distribution is an exaggerated parabolic shape placing the energy at the pellet perimeter. This causes an increase in the heat transfer to the cladding and maximizes the clad temperature increase effect.

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

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

14.13.2.2 Framatome ANP Richland, Inc. Fuel

The CEA ejection event was evaluated using the ANF-RELAP and XCOBRA-IIIC computer codes (Ref. 14.13-12). The ANF-RELAP code was used to model the salient system components and to calculate neutron power, fuel thermal response, and fluid conditions (such as coolant flow rates, temperatures and pressures). The thermal-hydraulic conditions from the ANF-RELAP calculation were used as the boundary conditions for the XCOBRA-IIIC code which was used to calculate MDNBRs.

The possibility for fuel failures was evaluated through the DNB and FCM analyses. The DNBR SAFDL was verified by using XCOBRA-IIIC to calculate the MDNBR for the operating conditions considered and then comparing this MDNBR to the HTP DNB correlation limit. The maximum centerline temperature of the peak-power pellet was compared to the FCM criterion to show that the FCM SAFDL is not exceeded. The possibility for fuel damage also was evaluated through an energy deposition analysis that was performed using the methodology described in Reference 14.13-13.

The analysis models used are described as follows. The Ft. Calhoun ANF-RELAP primary system model includes representations of the reactor vessel, pressurizer, hot leg piping, tube side of the SGs, primary coolant pumps and the RCS piping. The ANF-RELAP model explicity represents the two hot legs and four cold legs of the plant. The secondary system modeling contains the shell side of the SGs, main feedwater system, the auxiliary feedwater system, the main steam piping, the main steam isolation valves, the main steam safety valves and the turbine isolation valves.

The ANF-RELAP model calculates the plant transient system responses (for example, pressures, temperatures and flow rates) during the accident sequence. Using the core boundary conditions calculated with the ANF-RELAP at the time of peak core average surface heat flux, the XCOBRA-IIIC code is used to calculate the local conditions within the hot assembly. The minimum fuel rod DNBR is determined using the HTP DNB correlation and is compared against the correlation safety limit plus a mixed core penalty (Ref. 14.13-14). Fuel energy deposition calculations are performed using a three-dimensional core simulator. No credit is taken for the power-flattening eccects of Doppler or moderator feedback in the calculation of ejected rod worths and peaking factors.

14.13.3 Results

14.13.3.1 Westinghouse Fuel

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

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

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

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

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

Table 14.13-1 - "CEA Ejection Accident Assumptions for Westinghouse Fuel"

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

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

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

Table 14.13-2 - "CEA Ejection Accident Results For Westinghouse Fuel"

Average fuel pellet enthalpy limit	200 (cal/gm)

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

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

Fuel melting limit	<10%

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

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

Peak Zirconium-Water reaction limit	16 wt. %

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

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

14.13.3.2 Framatome ANP Richland, Inc.

The analysis addresses the following acceptance criteria from USNRC Regulatory Guide 1.77 (Ref. 14.13-5):

The radial average fuel pellet enthalpy at the hot spot must be ≤280 cal/gm.

The maximum system pressure will not exceed "Safety Limit C" as defined in Section III of the ASME Boiler and Pressure Vessel Code.

The radiological consequences will not exceed 25% of the limits in 10 CFR 100.

The initial system thermal-hydraulic analysis parameters assumed in the HFP and HZP system response calculations are shown in Table 14.13-3. The system transients were analyzed assuming that offsite power is available; the reactor coolant pumps comtinue to operate throughout the transient events. The pressurizer spray system, a function which minimizes the RCS pressure and MDNBR, is therefore assumed available.

For each of the four operating conditions, the ejection of only a single CEA with the maximum ejected worth was considered. Rod worths which bound the maximum ejected CEA worths provided by neutronics analysis were assumed.

The neutronics parameters assumed in the analysis are shown in Table 14.13-4. The HFP analysis cases used a DNB-limiting axial power shape selected from among the setpoint axials over the range of the DNB LCO with a $\pm$5% ASI uncertainty. For the HZP BOC analysis case, an axial power shape consistent with the PDIL was used. For the HZP EOC analysis case, a top-peaked power shape consistent with the post-ejection conditions was used.

The sequence of events for the limiting CEA ejection case (HFP BOC) system calculation is shown in Table 4.13-5. Figures 14.13-1 through 14.13-5 illustrate the behavior of key parameters during the transient. The CEA ejection results in an immediate reactor power increase, which quickly trips the reactor VHP condition and results in increased core fuel temperature and heat flux. The core power increase is assisted by the positive moderator temperature feedback and is opposed by the negative Doppler feedback. The RCS pressure does not rise sufficiently to open the pressurizer PORVs (this also is true for all four analysis cases and therefore the RCS over-pressurization safety criterion is met).

The results of the DNB analysis indicate that the MDNBRs are above the HTP correlation DNB safety limit (including the mixed-core penalty) for all four cases and therefore no fuel failures are predicted due to DNB considerations. The results of the FCM analysis indicate that the peak fuel cenerline temperatures for all four cases are below the melting limit and therefore no fuel failures are predicted due to FCM considerations. Since no fuel failures are predicted to occur, the radiological consequences safety criterion is met.

The results of the fuel energy deposition analysis indicate that the maximum total deposited enthalpies for all four cases are below 280 cal/gm. Therefore the enthalpy desposition safety criterion is met.

14.13.4 Radiological Consequences of a CEA Ejection Accident

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

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

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

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

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

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

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

   The melted fuel fraction was determined as follows:

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

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

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

5. The total primary-to-secondary leak rate is assumed to be 1 gpm.

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

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

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

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

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

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

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

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

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

Results

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

Table 14.3-3 - "System Thermal-Hydraulic Analysis Parameters, Framatome ANP Richland, Inc. Fuel"

Parameter	HFP Value	HZP Value
Initial Core Power (MW_t)	1530	10^{-3}
Initial Pressurizer Pressure (psia)	2100	2100
Initial Pressurizer Level (%)	60	48
Initial Total RCS Loop Flow (gpm)	198584	198584
Initial Core Inlet Temperature (°F) [a]	543	532
Variable High Power Trip Setpoint (% of Rated Thermal Power)	112	30
Pressurizer PORV Opening Setpoint Pressure (psia)	2350	2350

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

Table 14.13-4 - "Neutronics Analysis Parameters, Framatome ANP Richland, Inc. Fuel"

Parameter	HFP Value	HZP Value
Control Rod Worth, BOC (pcm)*	70.0	250.0
Control Rod Worth, EOC (pcm)*	70.0	300.0
Maximum Post-Ejection F_r, BOC	1.911	2.806
Maximum Post-Ejection F_r, EOC	1.706	2.665
Maximum Post-Ejection F_Q, BOC	2.259	4.404
Maximum Post-Ejection F_Q, EOC	2.121	6.711
Moderator Temperature Coefficient, BOC (pcm/°F)*	+5.0	+5.0
Moderator Temperature Coefficient, EOC (pcm/°F)*	-22.48	-4.0
Doppler Temperature Coefficient, BOC (pcm/°F)*	-1.004	-1.004
Doppler Temperature Coefficient, EOC (pcm/°F)*	-1.124	-1.124

* 1 pcm = 1×10^{-3} %$\Delta\rho$

Table 14.13-5 - "CEA Ejection Event, Framatome ANP Richland, Inc. Fuel, HFP BOC Sequence of Events"

Event	Time (sec)	Value
Ejection of a single CEA	0.0	---
Core power reaches VHP trip setpoint	0.105	112% RTP
Core power peaks	1.000	115.6% RTP
Scram rod insertion begins	1.005	---
Core average rod surface heat flux peaks	1.275	6.311 kW/ft
Maximum pressurizer pressure attained	4.200	2141 psia

Table 14.13-6 - "Activity Released from the Secondary System"

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

The activity released to the environment from the containment is presented in Table 14.13-7 (Ref. 14.13-6).

Table 14.13-7 - "Activity Released from the Containment"

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

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

Table 14.13-8 - "Resulting Doses"

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

14.13.5 Conclusions

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

The dose acceptance criteria is based on the recommendations of Standard Review Plan (NUREG-0800) Section 15.4.8, Appendix A, i.e., 75 rem thyroid and 6 rem whole-body. The calculated doses for the CEA ejection event are within the acceptance criteria. Specific results of radiological consequences are presented in Table 14.13-8.

14.13.6 References

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

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

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

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

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

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

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

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

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

14.13-10 EA-FC-00-030, "Cycle 20 CEA Ejection Verification - Westinghouse."

14.13-11 EMF-2488, Rev 0, "Ft. Calhoun Control Element Assembly Ejection Analysis," Siemens Power Corporation, November 2000.

14.13-12 ANF-89-151(P)(A), "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," Advanced Nuclear Fuels Company, May 1992.

14.13-13 XN-NF-78-44 (NP)(A), "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors," Exxon Nuclear Company, October 1983.

14.13-14 XN-NF-82-21(P)(A), Rev 1, "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," Exxon Nuclear Company, September 1983.

14.14 STEAM GENERATOR TUBE RUPTURE ACCIDENT

Since the performance of the FSAR analysis, the steam generator tube rupture accident has been evaluated using recent computer codes (Reference 6) which also addresses a release pathway through the steam driven auxiliary feedwater pump (FW-10).

14.14.1 General

The steam generator tube rupture accident is a penetration of the barrier between the reactor coolant system and the main steam system. The integrity of this barrier is significant from the radiological safety standpoint, as a leaking steam generator tube would allow transport of reactor coolant into the main steam system. Radioactivity contained in the reactor coolant would mix with shell side water in the affected steam generator. This radioactivity would be transported by steam to the turbine and then to the condenser, or directly to the condenser via the steam system dump and bypass valves. Noncondensible radioactive materials would be discharged through the condenser vacuum pumps to the atmosphere. FW-10 (Auxiliary Feedwater Pump) was considered as a release path also. Modification MR-FC-91-039 provided Operations with the ability to override CIAS from steam generator blowdown (SGBD) sampling isolation valves to draw a steam generator blowdown sample. This modification allows for establishment of an acceptable/analyzed blowdown sampling release path should plant conditions warrant the drawing of a blowdown sample. Appropriate AOPs/EOPs instruct the Operator to reroute the SGBD sampling discharge flow to a lineup leading to the radioactive waste disposal system.

Combustion Engineering's experience with nuclear steam generators indicates that the probability of complete severance of a tube is remote; a double-ended tube rupture has never occurred in a Combustion Engineering steam generator. The more probable modes of failure which result in smaller penetrations are those involving the occurrence of pinholes or small cracks in the tubes, and of cracks in the seal welds between the tubes and tube sheet.

Diagnosis of the accident is facilitated by radiation monitors in the blowdown sample lines from each steam generator and in the condenser vacuum pump discharge line. Additionally, the CIAS signal for HCV-2506A/B and HCV-2507A/B can be overriden for a period not to exceed two hours. This allows samples to be obtained from the steam generators to aid in the identification of the affected steam generator. These monitors initiate alarms in the control room and inform the operator of abnormal activity levels and that corrective action is required; the steam generator blowdown is automatically valved-off should a high activity level be reached.

The behavior of the systems varies depending upon the size of the rupture. For leak rates up to the capacity of the charging pumps in the CVCS, reactor coolant inventory can be maintained and an automatic reactor trip would not occur. The gaseous fission products would be released to atmosphere from the secondary system at the condenser vacuum pump discharge. Those fission products not discharged in this way would be retained by the main steam, feedwater and condensate systems.

For leaks that exceed the capacity of the charging pumps, the pressurizer water level and pressure decrease and a TM/LP reactor trip results. The turbine then trips and the steam system dump and bypass valves open. The steam generator water level indicators aid in detection of these larger leaks since the water inventory in the leaking steam generator may increase more rapidly than that of the intact steam generator following reactor trip.

The amount of radioactivity released increases with break size. For this analysis, an area equivalent to a double-ended break of one steam generator tube is assumed for the rupture size. At normal operating conditions, the leak rate through the double-ended rupture of one tube is greater than the maximum flow available from the charging pumps; the reactor coolant system pressure decreases and a low pressurizer pressure trip occurs. Following the reactor trip, the reactor coolant average temperature is reduced by exhausting steam through the steam dump and bypass system. The radioactivity exhausted through the steam dump and bypass system flows to the condenser where the noncondensible gaseous products are released to the atmosphere.

Based on guidance contained in the Emergency Operating Procedures (EOPs), which serve to minimize releases and off-site doses, the operator will use the steam dump and bypass valves to reduce the reactor coolant system hot leg temperature. Once the dump valves are closed, operation of the bypass valve (PCV-910) allows steam to pass to the condenser to reduce this temperature to 510°F. When the hot leg temperature is below 510°F, the affected steam generator is isolated to terminate the release source. This hot leg isolation temperature of 510°F bounds the minimum cold leg temperature requirements for adequate reactor coolant pump (RCP) net positive suction head (NPSH). This steam generator hot leg isolation temperature value is consistently used in the EOPs and this analysis, regardless of whether RCPs are available (i.e., off-site power is available). When the reactor coolant temperature is below 300°F. the operator can place the shutdown heat removal system into operation and isolate both steam generators.

14.14.2 Method of Analysis

The analysis of a steam generator tube rupture was performed using a digital computer simulation of the system. The simulation includes neutron kinetics with fuel and moderator temperature feedback, the effect of the shutdown group of CEA's and the reactor coolant and main steam systems including the pressurizer, steam generators, and steam dump and bypass valves. The method of analysis used provides radiological consequences results that bound operator actions that may be taken in accordance with the EOPs (Reference 7). This results from the analysis assumption that the operator will feed and steam both steam generators for cooldown, for the first two hours. Cooldown to 510°F would occur much sooner than this when using the EOPs.

The distribution of the fission gases and radioiodines throughout the secondary system is obtained from a digital simulation of the behavior of each isotope in each volume of the secondary system. The effect of gas solubility in the condenser hot well and iodine volatility in the steam generators and hot well is considered, in addition to moisture carryover from each steam generator.

The iodine content per unit mass of steam leaving each steam generator is assumed to be 10 percent of the iodine concentration in the steam generator liquid. The work of Styrikovich et al (Reference 1) indicates that this assumption is conservative. The partitioning of iodine in the condenser hot well is calculated as a function of the iodine concentration in the condenser hot well liquid. The partition coefficient is determined as described in References 2 and 3.

The effect of long-lived I-129 and stable I-127 is included when this determination is made.

Steam Generator Tube Rupture Radiological Consequences

Scenarios with and without Loss of Off-Site Power were modeled with the loss of off-site power case resulting in the most adverse radiological consequences.

The Reference 4 analysis methodology, as described in Reference 8 and approved by the NRC in Reference 9, resulted in the following sequence of events:

1. A tube rupture occurs in steam generator A (RC-2A). A similar sequence could be developed for steam generator B (RC-2B).

2. Steam generator A starts to fill with reactor coolant and reactor coolant pressure starts to drop.

3. Three charging pumps come on. It is conservatively assumed that all three charging pumps remain in operation following reactor trip to maximize the primary to secondary system pressure differential and consequently the leak rate and radiological consequences.

4. The reactor protective system initiates a scram in approximately 373 seconds (Reference 4). Thus the plant will stay on the line approximately 373 seconds after the rupture, discharging activity out the condenser vent. The condenser off gas radiation monitor (RM-057) will alarm. The steam generator A blowdown radiation monitor (RM-054A) will also alarm. A loss of off-site power is assumed to occur concurrent with the reactor trip.

5. Safety injection actuation signal (SIAS) and containment isolation signals (CIAS) actuate as result of reactor coolant pressure dropping below 1600 psia. As a result of SIAS and CIAS, the following occur:

    a. high pressure safety injection pumps start (maintaining the reactor coolant system primary pressure in the neighborhood of 1200-1400 psia).

    b. the reactor letdown is automatically secured (by CIAS and pressurizer level controls.

6. Initially the Main Steam Safety Valves (MSSVs) automatically lift on both steam generators discharging activity.

7. The operator notes that pressurizer level is normal and shuts down the high pressure safety injection pumps and charging pumps manually when the stop and throttle criteria are met to control pressurizer level and to prevent going solid (losing the bubble in the pressurizer).

8. The operator determines which steam generator is leaking within 30 minutes of the event initiation by use of the blowdown or level indicators. (Steam generator A is indicating a high water level while steam generator B is indicating a low water level). The operator closes the steam line isolation valve (HCV-1041A) on the damaged steam generator (A).

9. The operator opens both air assisted MSSVs (MS-291 and MS-292) and the atmospheric steam dump valve (HCV-1040) to discharge steam from the intact steam generator (B) and to cool the primary system hot leg temperature down to 510°F. Steaming both steam generators provides a bounding analysis with the largest EAB and LPZ doses for this event.

10. After a primary system hot leg temperature of 510°F is reached, the operator continues to reduce that temperature by releasing steam from the intact steam generator. This terminates the further release of activity from the damaged steam generator.

11. Initial radioactivity of steam generator coolant is assumed to be equal to 0.10 μCi/gm Dose Equivalent I-131 (T.S. 2.20).

12. The intact steam generator will use the initial radioisotopic steam generator concentration throughout the analysis.

13. Auxiliary feedwater pump FW-10 is considered as a release path. The drain lines from HCV-1041A and HCV-1042A were also included as release paths.

14. A nominal 1 gpm leak of reactor coolant into the intact steam generator is assumed throughout the accident.

15. Release rate calculations assume uniform flow rates and uniform mixing.

16. In accordance with T.S. 2.1.3, a reactor coolant DEC I-131 specific concentration of 60 µCI/gm will be used, while the activities at 1% failed fuel will be used for noble gas nuclides.

17. No credit is taken for reduction of reactor coolant activity due to possible dilution (e.g. dilution by charging or safety injection flow).

18. It will be assumed that noble gases in the steam generator are immediately released to the environment.

The above sequence of events assumes a SG isolation, RCS cooldown to 510°F and then a continued RCS cooldown to where the shutdown cooling system can be used.

During the first 30 minutes following the tube rupture, less than 16 percent of the reactor coolant inventory is transported to the main steam system. During this interval, approximately 26,498.1 lbs. of steam is vented to the atmosphere in order to remove heat from the reactor system.

Utilizing primary coolant fission product inventories as reported in Section 11.1 Fort Calhoun USAR and secondary activity levels commensurate with the highest allowable activity per the technical specification and a RCS leakrate of 1 gpm to the intact generator. RCS leakage to the damaged generator was calculated through the CESEC code documented in EA-FC-93-082 (Reference 4) and EA-FC-92-029 (Reference 5). The radiological releases for this incident were then calculated. The total two-hour releases accompanying a tube rupture are:

1. 99.43 dose equivalent curies of I-131

2. 306,653 dose equivalent curies of Xe-133

Applying a dispersion factor of 2.55×10^{-4} sec/m^3 for ground level (EAB), point source release, the integrated two hour whole body dose at the site boundary is 0.387 rem and the dose to the thyroid is 12.11 rem.

See Table 14.14-1 for LPZ total integrated doses.

14.14.3 Conclusions

The maximum off-site radiation doses resulting from a double-ended rupture of one steam generator tube, while operating with 1 percent defective fuel at 1500 MWt power level, are maintained within the guidelines of 10 CFR 100.11.

Table 14.14-1 - "Radiological Release for Steam Generator Rupture Without Offsite AC"

	EAB	LPZ	10CFR100.11 Limit*
Whole Body Dose (Rem)	0.387	0.00687	25
Thyroid Dose (Rem)	12.11	0.215	300

* Reference 9 approved limit with iodine spiking.

14.14.4 Section 14.14 References

1. Transfer of Iodine from Aqueous Solutions to Saturated Vapor, Styrikovich, M. A., Martynova, I. I., Kathovskaya K. Ya., Dubrovskii, I. Ya., and Smirnova, I. N., translated from Atomnaya Energiga Vol. 17 No. 1, July 1964.

2. A Program for Calculating the Partition Coefficient of Iodine Between Water and Air, Parsly, L. F., ORNL 68-7-3, July 5, 1968.

3. Partition Factors and Steam Release Calculation Methodology: CE Transient Analysis Methods for Fort Calhoun Station Unit No. 1, CENPSD-164-P, Combustion Engineering, Inc., 1981.

4. Bounding Radiological and Transient Consequences of a SGTR, EA-FC-93-082, 11/01/93.

5. "Chapter 14 (USAR) Radiological Consequences Update with Cycle 14 Revised Source Term," EA-FC-92-029, 1992.

6. Letter from OPPD (R.L. Andrews) to NRC (J.M. Taylor), dated April 10, 1987.

7. Response to Fort Calhoun Station Condition Report 199700976 Action Item 1 (including 10CFR50.59 evaluation), April 1998.

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

9. Letter NRC-93-0301 from NRC (S.D. Bloom) to OPPD (T.L. Patterson), dated November 2, 1993.

14.15 LOSS-OF-COOLANT ACCIDENT

14.15.1 General

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

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

The injection water for the high- and low-pressure injection systems is supplied from the borated safety injection and refueling water (SIRW) tank. This analysis assumes a minimum usable SIRW tank inventory of 250,000 gallons (Ref. 14.15-28). When the SIRW tank is nearly empty, water is recirculated from the containment sump, as described in Section 6.2.

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

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

A complete LBLOCA analysis for operation of the Fort Calhoun Station at 1500 MWT was performed for Cycle 20 by Framatome ANP Richland, Inc. (Ref. 14.15-34). A complete SBLOCA analysis was also performed for Cycle 20 (Ref. 14.15-35). The limiting break was based on a peak rod average burnup of 62,000 MWD/MTU and a radial peaking factor of 1.732. The steam generators are assumed to have 20% plugging in each generator.

For a postulated LOCA, a reactor trip is initiated when the pressurizer pressure-low setpoint is reached. A Safety Injection Actuation Signal (SIAS) is actuated by either a containment pressure high or a pressurizer pressure low-low signal. The consequences of the accident are limited in two ways:

1. Reactor trip and safety injection (of borated water) supplement void formation in causing a rapid reduction of the nuclear power to a residual level corresponding to the delayed fission product decay. For a postulated LBLOCA, credit for Control Element Assembly (CEA) insertion to keep the reactor subcritical post-LOCA is not taken since large break forces may degrade the trip function of the CEAs.

The requirement for post-LOCA subcriticality is met by maintaining a sufficiently high boron concentration in the SIRW tank.

2. Injection of borated water ensures sufficient flooding of the core to prevent excessive fuel temperatures.

Before the reactor trip occurs, the reactor is in an equilibrium condition, i.e., the heat generated in the core is being removed via the secondary system. After reactor trip and turbine trip, core heat and heat from the vessel and internals is transferred to the RCS fluid, and then to the containment and the secondary system dependant upon the break size.

The reactor coolant pumps (RCPs) are tripped at the initiation of the accident due to an assumed loss of offsite power or remain running until operator actions trip the pumps in accordance with the trip 2/leave 2 operating criteria (Ref. 14.15-20). The effects of the RCP coastdown are included in the blowdown analyses. Without a loss of offsite power, a Steam Generator Isolation Signal (SGIS), which occurs as a result of containment pressure-high, terminates normal feedwater flow by closing the main feedwater isolation valves. With the assumed loss of offsite power, main feedwater is lost with the coastdown of the electric motor driven main feedwater pumps. If offsite power is available, the steam is dumped to the condenser, although not credited in the analysis; if offsite power is not available, the steam is assumed to be dumped to the atmosphere via the Main Steam Safety Valves (MSSVs). Makeup to the steam generators is initiated by the auxiliary feedwater pumps if steam generator level falls below the auxiliary feedwater system actuation setpoint. If not terminated previously by High Pressure Safety Injection (HPSI) pump flow, the core uncovery transient is turned around when the RCS pressure falls below approximately 255 psia and the (SITs) inject borated water.

14.15.2 Performance Criteria for Emergency Core Cooling System

The reactor is designed to withstand the thermal effects caused by a LOCA including the double-ended severance of the largest RCS pipe. The reactor core and internals together with the Emergency Core Cooling System (ECCS) are designed so that the essential heat transfer geometry of the core is preserved following the accident. The ECCS, even when operating during the injection mode with the most severe single active failure, is designed to meet the requirements of 10 CFR 50.46.

"Acceptance Criteria for Emergency Core Cooling Systems for Light Water Power Reactors", (Reference 14.15-1).

These requirements are:

1. Peak clad temperature (PCT) does not exceed 2200°F.

2. The amount of cladding that chemically reacts with the coolant does not exceed 1% of the zircalloy cladding surrounding the fuel, excluding the cladding surrounding the plenum volume.

3. Oxidation of the cladding does not exceed 17% of the original cladding thickness, which precludes embrittlement problems.

4. Calculated changes in core geometry shall be such that the core remains amenable to cooling.

5. After initial operation of the ECCS, core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period required by the long-lived radioactivity remaining in the core.

14.15.3 Thermal Analysis Description

The analysis specified by 10 CFR 50.46 is presented in Sections 14.15.4 and 14.15.5 for large and small breaks, respectively. The results of the large and small break LOCA analyses, which are summarized in Tables 14.15-3 and 14.15-6, show compliance with the above Acceptance Criteria. The highest PCT calculated was for a double-ended cold leg guillotine (DECLG) break with a Moody discharge coefficient (C_D) of 1.0.

The large break analysis is based on the initial reactor conditions shown in Table 14.15-1. The analytical techniques used are in compliance with Appendix K of 10 CFR 50, and are described in Reference 14.15-3.

This document describes the major phenomena modeled, the interfaces among the computer codes, and the features of the codes that ensure compliance with the Acceptance Criteria.

The method of analysis to determine PCT is divided into two types of analysis: 1) LBLOCA, and 2) SBLOCA. The methods of analysis for the large and small break LOCAs are described below and results are given.

14.15.4 Large Break LOCA Analysis

14.15.4.1 Description of Analysis and Assumptions

A loss-of-coolant accident is defined as an instantaneous rupture of a RCS pipe ranging in cross-sectional area up to, and including, that of the largest pipe in the RCS. A LBLOCA event is typically described in 3 phases: blowdown, refill and reflood. The blowdown phase covers the period from the beginning of the transient until most of the reactor vessel fluid has exited the break and the reactor vessel pressure approaches containment pressure. Using Framatome ANP Richland, Inc. methodology, the blowdown phase typically lasts about 18 to 23 seconds, depending on the break size, and continues until the end-of-bypass (EOBY) when ECCS fluid no longer bypasses the core. The refill phase covers the period from the EOBY until time at which fluid from the ECCS has filled the lower downcomer and lower plenum up to the bottom of the heated length of the fuel, i.e., to the time of the Beginning-of-Core-Recovery (BOCREC). The reflood phase covers the time period from BOCREC until the core is quenched.

The blowdown phase is characterized by a sudden depressurization from operating pressure down to the saturation pressure of the core exit fluid. The RCS then depressurizes at a slower rate as the RCS fluid is expelled primarily by vaporization. A reactor trip signal occurs when either the pressurizer low pressure trip setpoint or containment vessel high pressure trip setpoint is reached. Typically, reactor trip and scram are conservatively neglected in the LOCA analysis, and reactor shutdown is accomplished by moderate feedback.

The limiting break has been shown by experience to occur in a cold leg pipe between the pump discharge and the vessel. An immediate flow reversal and stagnation occurs in the core due to flow out the break, which causes the fuel rods to pass through the critical heat flux (CHF), usually within 1 second following the break. Once the CHF has been reached in the analysis, rewet and pre-CHF regimes are precluded in the calculational methodology during blowdown, consistent with the requirements of 10 CFR 50 Appendix K (Ref. 14.15-1), and the fuel rods transfer heat to the coolant by way of the transition and film boiling modes of heat transfer. Since loss-of-offsite power is assumed to occur coincident with the LOCA, coastdown of the reactor coolant pumps begins coincident with the loss-of-offsite power. For short periods of time following the break and up to about 5 to 8 seconds into the transient, the pump head may be sufficient to resume positive flow through the core, which can provide significant cooling of the fuel rods.

A Safety Injection Actuation Signal (SIAS) is assumed to occur when the high containment pressure setpoint is reached. A start time delay is assumed for the diesel generators and SIS pumps due to the assumed loss-of-offsite power. Once the time delay criteria are met, the HPSI pumps begin injecting SI flow into the system. Once the delay time criteria are met and the system pressure at the injection location has fallen below the shut-off head of the LPSI pumps, low pressure SI flow is injected into the primary system. The worst single failure criterion is typically met by assuming either the loss of one LPSI pump or the loss of a diesel generator. When the system pressure falls below the SIT pressure, flow from the SITs is injected into the cold legs. ECCS flow is assumed to bypass the core and flow to the break until the EOBY is predicted to occur.

Following EOBY, ECCS flow fills the lower downcomer and lower plenum until the liquid level reaches the bottom of the core (BOCREC time). During this downcomer and lower plenum refill period, Framatome ANP Richland, Inc. evaluation models assume no net heat removal from the core. A small amount of heat is transferred from the hot high power fuel rods to cooler fuel rods and structure by radiation heat transfer.

Reflood begins at BOCREC time. ECCS fluid fills the downcomer to slightly above the bottom of the cold leg elevation and provides the driving head to move coolant through the core. Steam is generated as the mixture level moves up to the core and liquid is entrained and carried over to the steam generator. Steam binding occurs as the steam and entrained liquid flows through the intact and broken loop steam generators and RCPs. The RCPs are assumed to have locked rotors (per 10 CFR 50 Appendix K), which maximizes their resistance and reduces the core reflood rate. The fuel rods are eventually cooled and quenched by radiation and convective heat transfer as the quench front moves up the core. The reflood heat transfer rate is predicted through experimentally-determined heat transfer and carry-over rate fraction correlations. In accordance with the requirements of 10 CFR 50 Appendix K, a spectrum of break types (double-ended guillotine and split breaks) and sizes are analyzed.

Table 14.15-1 - "Fort Calhoun Large Break LOCA Analysis Input Parameters and Assumptions"

NSSS Power - 102% of 1500 MWt	1530 MWt
Peak Linear Heat Rate - at 102% of 1500 MWt	15.5 kW/ft
Radial Peaking Factor (F_R^T)	1.89
Maximum Allowable Peaking Factor (F_q)	2.529
Reactor Coolant System Pressure	2100 psia
Reactor Coolant System Flow Rate	198,584 gpm
Reactor Inlet Temperature	543°F *
Reactor Trip Signal	N/A
SI Signal	19.7 psia, High Containment Pressure
Safety Injection Tank Water Volume	877 ft^3/Tank
Safety Injection Tank Pressure	265.6 psia
Steam Generator Tube Plugging Level	20% (symmetric)

*Nominal hot full power inlet temperature, supports operation including ± 2°F for operating band and ± 2°F for measurement uncertainty.

14.15.4.2 Method of Large Break Analysis

The Framatome ANP Richland, Inc. SEM/PWR-98 LBLOCA evaluation model consists of a series of computer codes that are linked together to perform loss-of-coolant accident analyses to demonstrate plant and fuel design conformance to 10 CFR 50.46 criteria and 10 CFR 50 Appendix K requirements. Overall the methodology analyzes the transient chronologically in phases consisting of (1) initialization, (2) blowdown, (3) refill, and (4) reflood. Typically each phase consists of a calculation of system transient behavior and conditions that in turn provides boundary conditions on the detailed core or hot rod transient calculation. In addition, there are a number of codes that interface and automate the methodology. That is, those codes calculate extended input, provide initial or boundary condition data, transfer data from one step to another, and automate and link the calculation steps.

The initiation step includes reduction of initial plant geometrical data to input for the RELAP4-EM system calculation and reduction of fuel data for input to the RELAP4-EM core calculation. Core physics calculations, as described later in this section, are performed to provide reactor kinetics input. Core power distribution and axial power shape input are calculated using core physics methods. RODEX2 calculations are performed to determine exposure-dependent initial fuel rod condition inputs. Blockage input for the swelling and rupture calculation is determined. Initial operating conditions are defined and input. Steady state initialization calculations are performed with RELAP4-EM to determine that a correct steady state with regard to flow, initial pressure distributions, and loss coefficients is being computed by the code. In addition, the steady state energy balance for each core and steam generator volume is confirmed.

When the initial data have been input and steady state conditions have been achieved, analyses of the blowdown phase of the accident can be performed.

RELAP4-EM is used to do the blowdown calculation. The SEM/PWR-98 model combines the system and hot channel blowdown calculations into a single blowdown transient calculation. The output of the blowdown calculation is the information needed to determine the end-of-bypass time (TEOBY), system initial conditions for the beginning of refill, break mass and enthalpy versus time for the calculation of containment pressure, and core and hot rod temperatures at the end-of-bypass for input to the heatup calculations.

Following blowdown, several calculations are made to provide the input necessary to extend the calculations into the refill and reflood phases. An extended power calculation is performed using RELAP4-EM to compute the delayed neutron fission power. The total power is computed as the sum of the RELAP4-EM delayed neutron fission power, ANS actinide power, and 1.2 times the decay heat data from the 1971-73 Draft ANS decay heat standard. Extended ECCS flows are also calculated using RELAP4-EM to provide the necessary input to the refill and reflood calculations. Containment pressure vs time is calculated using the ICECON code and the blowdown mass and energy release data from RELAP4-EM. The ICECON code contains Framatome ANP Richland, Inc.'s version of the COMTEMPT code used for dry containment analyses and is part of a group of codes called RFPAC. The containment pressure is necessary input for the refill and reflood calculations.

System behavior during the refill phase of the LOCA is calculated using the PREFILL code. PREFILL is also part of RFPAC and integrates the ECCS flows to determine when the beginning of core reflood occurs. Possible loss of ECCS water due to the hot wall delay effect is calculated as is the time delay for ECCS water to flow through the downcomer to the lower plenum. Heat transfer from hot vessel surfaces to the ECCS water is determined to provide the initial subcooling of the ECCS water at the beginning of reflood. The PREFILL calculation provides the time of beginning of core reflood (BOCREC) and extended ECCS flow and initial conditions for the reflood calculations. The hot rod calculation during refill is performed using the TOODEE2 code. During refill, the core is assumed to heatup adiabatically.

The only mechanism considered for removing heat from the hot rod is radiation to cooler fuel rods and internal fuel assembly structures such as guide tubes. Fuel temperatures for the RELAP4-EM at the end-of-bypass time and the pin power distribution about the hot rod are used to develop input for the radiation model calculation.

The reflood phase of the LOCA transient begins at the time of BOCREC. The system calculation is performed for reflood using the REFLEX code which is also part of the RFPAC package. REFLEX computes reflood rate versus time as well as parameters necessary to calculate the heat transfer using the Fuel Cooling Test Facility (FCTF) reflood heat transfer correlations. The REFLEX code is based on a flow network where the pressure drop for flow of effluent from the core around the loops and out the break is balanced against the available gravitational driving head in the downcomer in a quasi-steady state calculation. Pressure loss penalties are applied to account for interaction of ECC water with steam flowing around the loops. The result of the calculation defines a net core flow or flooding rate. Carryover is calculated using the FCTF carryover correlation.

The hot rod calculation for the reflood phase is a continuation of the TOODEE2 refill calculation. Reflood cooling is calculated using the FCTF reflood heat transfer correlation as recently revised by SPC and approved by NRC. This calculation considers swelling and rupture, steam cooling for elevation above the rupture elevation at reflood rates below one inch per second, and metal/water reaction both inside (rupture node only) and outside the fuel rod cladding. The results of the TOODEE2 calculation are PCT and local maximum metal/water reaction, which can be compared directly to 10 CFR 50.46 criteria. A subsequent core-wide metal/water reaction calculation can be performed if the results from the hot rod indicate the 1% core-wide limit might be approached.

14.15.4.3 Results of Large Break Analysis

Calculations were performed for a spectrum of DECLG (C_D=0.4, 0.6, 0.8 and 1.0) and single-ended cold leg split (SECLS) breaks (C_D=0.8 and 1.0) at a peak LHGR of 15.5 kW/ft with the following bounding combinations of stored energy and axial shape:

1. Bounding BOC stored energy (where maximum densification occurs) with chopped cosine and axial shape.

2. Bounding BOC stored energy (where maximum densification occurs) combined with a bounding middle-of-cycle (MOC) axial shape, and

3. Bounding MOC stored energy combined with a bounding end-of-cycle (EOC) axial shape.

The bounding MOC and EOC axial shapes were determined from projected axial shapes for equilibrium and xenon oscillation transient conditions.

Sensitivity calculations were also performed to determine the limiting single-failure. Loss of a LPSI pump (NOLPSI) was determined to be more limiting than the loss of a diesel generator (NODIESEL).

The results of the calculations indicate that the 1.0 DECLG break with the EOC axial shape (and MOC stored energy) and loss-of-LPSI pump single-failure was the limiting case.

The time sequence of events for this break spectrum is given in Table 14.15-2 and the associated peak cladding temperatures (PCT) and hot spot metal reactions are summarized in Table 14.15-3.

Table 14.15-2 - "Sequence of Events for Fort Calhoun LBLOCA Limiting Case"

Event	Time (sec)
Analysis begins	0.00
Break opened	0.05
SIAS initiated	0.69
Broken loop SIT injection begins	5.05
Single intact loop SIT injection begins	14.84
Double intact loop SIT injection begins	14.84
Refill begins (EOBY)	18.84
Reflood begins (BOCREC)	30.36
HPSI and LPSI initiated	30.70
Fuel rupture occurs	44.16
Broken loop SIT discharge valve closes (SIS calculation)	48.37
Double intact loop SIT discharge valve closes (SIS calculation)	56.01
Single intact loop SIT discharge valve closes (SIS calculation)	55.91
PCT occurs	140.92

Table 14.15-3 - "Summary of Results for Fort Calhoun LBLOCA Limiting Case"

PCT	
Temperature	1905°F
Time	140.9 seconds
Elevation	10.04 ft
Hot Rod Burst	
Time	44.2 seconds
Elevation	9.0 ft
Channel Blockage Fraction	0.41
Metal-Water Reaction	
Local Maximum	3.13%
Elevation of Local Maximum	10.04 ft
Core Maximum	<1.0%

Figures 14.15-41 through 14.15-60 present the limiting LBLOCA case: C_D = 1.0, EOC axial shape, MOC stored energy, and loss of LPSI pump. In figures 14.15-42, 43 and 44, the acronyms DIL, SIL and BL stand for Double Intact Loop, Single Intact Loop, and Broken Loop, respectively. In the Framatome ANP Richland model, the Broken Loop refers to the cold leg that was assumed to include the break from the coolant system to containment. The Single Intact Loop is the other cold leg attached to the same steam generator as the "Broken" cold leg. The Double Intact Loop refers to the cold legs attached to the other steam generator (these two cold legs are lumped together in the model) .

14.15.5 Small Break LOCA Analysis

14.15.5.1 Event Description

The Framatome ANP Richland, Inc. SBLOCA evaluation model for event response of the plant and hot fuel rod used in this analysis (References 14.15-4 and 5, as modified by 10 CFR 50.56) consists of three computer codes; RODEXZ, ANF-RELAP, TOODEE2. This methodology has been reviewed and approved by the NRC to perform SBLOCA analyses.

The core power level utilized in the SBLOCA analysis was 102% of 1500 MWT, the licensed power. The peak linear heat rate and peaking factor used in the analyses are also given in Table 14.15-4. Additional assumptions for the analysis of Fort Calhoun Station were: operation at a total primary system flow rate of 198,584 gpm, a 20% steam generator tube plugging in each of the two steam generators, and a 30.9 second delay in delivery of pumped ECCS flow assuming loss of offsite power coincident with reactor trip. Eight of the ten(i.e., four per steam generator) MSSVs were assumed to be operable. They were assumed to be set at 3% above the Technical Specification setpoint value and require an additional 3% accumulation before being assumed fully open.

A spectrum of cold leg break sizes (0.022, 0.036, 0.049, 0.068, and 0.087 ft^2) was analyzed in order to determine the most limiting break size. These breaks were analyzed following the method presented in Section 14.15.5.2.

Table 14.15-4 "Input Parameters for Fort Calhoun SBLOCA Analysis"

Parameter	Analysis Value
Rated Thermal Power	1,500 MW_t*
Kinetic Parameters:	
Radial Peaking Factor (F_r)	1.732†
Total Peaking Factor (F_q)	2.529
Peak LHGR	15.5 kW/ft
Fuel:	
Cladding Outside Diameter	0.440 in.
Cladding Inside Diameter	0.384 in.
Cladding Thickness	0.028 in.
Pellet Outside Diameter	0.3770 in.
Pellet Density	95.35% of theoretical
Active Fuel Length	128 in.
RCS:	
Flow Rate	198,584 gpm
Pressure	2,100 psia‡
Vessel Inlet Coolant Temperature	543°F§
Steam Generators:	
Broken Loop Tube Plugging	20%
Intact Loop Tube Plugging	20%
HPSI System:	
Fluid Temperature	120°F
Delay Time	30.9 sec.**
SITs:	
Liquid Volume	815.4 ft^3
Pressure	255 psia
Fluid Temperature	120°F
AFW:	
Fluid Temperature	120°F
Delay Time	60.9 sec.

* 102% of this value was used in the analysis

† 1.890 including uncertainties

‡ Nominal pressurizer pressure, supports operation including +50, -25 psi for operating band and ±2°F for measurement uncertainty.

§ Nominal hot full power inlet temperature, supports operation including ±2°F for measurement uncertainty.

** The HPSI delay time was 12.0 seconds for the manual RCP trip sensitivity calculations.

14.15.5.2 Method of Small Break Analysis

The three Framatome ANP Richland, Inc. computer codes used in this analysis are:

1. The RODEX2 code was used to determine the initial fuel stored energy and gap conditions for the initialization of the system blowdown and hot rod response calculations.

2. The SPC version of RELAP5/MOD2 (ANF-RELAP) was used to model the primary system and secondary side of the steam generators throughout the event. The governing conservation equations for mass, energy and momentum transfer are used along with appropriate correlations consistent with Appendix K of 10 CFR 50.

3. The TOODEE2 code was employed to model the behavior of the hot rod during the entire event. TOODEE2 uses thermal-hydraulic boundary conditions from the ANF-RELAP system calculation.

Fuel rod temperatures corresponding to beginning-of-cycle (BOC) peak stored energy conditions were used to initialize ANF-RELAP. Cladding and fuel pellet dimensions, plenum gas inventory and composition, and effective plenum volume were taken at end-of-cycle (EOC) conditions, and were used to initialized TOODEE2. This conservative combination of BOC and EOC conditions bounds operation of the fuel over the entire span of the fuel cycle.

The RCS of the plant was nodalized in the ANF-RELAP model into control volumes interconnected by flow paths or "junctions." The model includes four SITs, a pressurizer, and two steam generators with both primary and secondary sides modeled. All of the loops were modeled explicitly to provide an accurate representation of the plant.

A steam generator tube plugging level of 20% per steam generator was assumed. The HPSI system flow was modeled to be evenly distributed to the four cold legs. LPSI flow was modeled but not initiated since the system pressure for this event did not fall below the shutoff head of the LPSI pumps. No credit was taken for the charging system flow.

The heat generation rate in the ANF-RELAP reactor core model was determined from point reactor kinetics equations with actinide and decay heating as prescribed by Appendix K.

The single failure criterion required by Appendix K was satisfied by assuming the loss of one emergency diesel generator (EDG), which resulted in the disabling of one HPSI pump, and the motor driven AFW pump. The swing HPSI pump was not credited, leaving only a single HPSI pump in operation. Initiation of the HPSI system was delayed by 30.9 seconds (i.e., 30 second HPSI delay + 0.9 second instrumentation delay) beyond the time of the SIAS. The maximum Technical Specification delay represents the time required for EDG startup, switching, and pump startup. The disabling of the motor driven AFW pump leaves only the turbine driven AFW pump available. The initiation of the turbine driven AFW pump was delayed 60.9 seconds (i.e., 60 second AFW delay + 0.9 second instrumentation delay) beyond the time of the Auxiliary Feedwater Actuation Signal indicating low steam generator level (15% wide range)

14.15.5.3 Results of Small Break Analysis

The SBLOCA analyses were performed with the assumptions contained in Table 14.15-4. The time sequence of events and results of key parameters for the 0.022, 0.036, 0.049, 0.068, and 0.087 ft^2 breaks analyzed are shown in Tables 14.15-5 and 14.15-6, respectively.

The 0.049 ft^2 break was shown to be the PCT limiting break size. The PCT for the 0.049 ft^2 break size was 1865°F. The 0.036 ft^2 break was shown to be the limiting local cladding oxidation break size. The maximum local cladding oxidation for the 0.036 ft^2 break size was calculated to be 2.57% of the total cladding thickness before oxidation.

The limiting core-wide metal reaction was calculated to be less that 1% of the maximum hypothetical amount for the active core as required by 10 CFR 50.46. These results indicate that a coolable geometry would be maintained during a SBLOCA event.

Transient plots for key system parameters are provided for the PCT limiting break size (0.049 ft^2) in Figures 14.15-61 through 14.15-68.

Table 14.15-5 - " Sequence of Events Fort Calhoun SBLOCA Analysis"

	Break Size (ft^2)				
	0.022	0.036	0.049	0.068	0.087
Event	Time (sec)				
Event initiation	0	0	0	0	0
Reactor trip signal and RCP trip	30	18	14	11	9
SIAS setpoint reached	43	26	19	15	13
HPSI injection starts	73	56	49	45	43
Loop seals clear					
Intact loop, cold leg 1A	---	---	---	---	---
Intact loop, cold leg 1B	~648	~406	~310	~226	~188
Broken loop, cold leg 2A (broken)	---	---	---	---	---
Broken loop, cold leg 2B (intact)	~640	~416	~310	~228	~190
Break uncovery	~654	~414	~312	~224	~176
Minimum primary system mass	2,746	1,904	1,394	920	648
SIT injection starts	---	---	~1,448	~918	~648
AFW flow starts	---	---	---	---	---
PCT occurs	3,158	2,553	1,451	921	650
End of calculation	4,000	3,000	2,500	1,500	1,000

Table 14.15-6 Break Spectrum Calculation Results for Fort Calhoun SBLOCA Analysis

	Break Size (ft^2)				
	0.022	0.036	0.049	0.068	0.087
Hot Rod Burst					
Time (sec)	---	2,182	1,255	863	640
Elevation (ft)	---	9.5	10.0	9.8	9.8
Channel Blockage Fraction (%)	---	68.5	68.8	69.5	70.2
Peak Cladding Temperature					
Temperature (F°)	1,064	1,720	1,865	1,659	1,554
Time (sec)	3,158	2,553	1,451	921	650
Elevation (ft)	10.0	10.3	9.5	9.5	9.5
Metal-Water Reaction					
Local Maximum (%)	0.04	2.57	2.54	1.03	0.59
Elevation of Local Maximum (ft)	10.0	10.3	9.5	9.8	9.8
Hot Rod Average (%)	0.00	0.18	0.35	0.15	0.07
Core Wide (%)	<1	<1	<1	<1	<1

14.15.6 Long Term Cooling Considerations (ECCS)

General

An evaluation of the post-LOCA Long Term Cooling ECCS performance by Westinghouse of Fort Calhoun station has demonstrated conformance with criterion (5) of 10 CFR Part 50.46(b) (Ref. 14.15-36). Procedures have been defined for utilizing the ECCS to remove decay heat and thereby maintain core temperatures at acceptable low values for an indefinite period of time.

Long term cooling is initiated when the core is reflooded after a LOCA and is continued until the plant is secured. The objective of long term cooling is to maintain the core temperature at an acceptably low value while removing decay heat for the extended period of time required by the long-lived radioactive isotopes remaining in the core. In satisfying this objective, the post-LOCA long term cooling requirements (as contained in the Emergency Operating Procedures) make provisions for maintaining core cooling and boric acid flushing by simultaneous hot and cold leg injection, or for initiating cooldown of the Reactor Coolant System (RCS) if the break is sufficiently small with natural circulation present, such that success of such operation is assured.

Within 8.5 hours of the start of the LOCA and if shutdown cooling has not been established, then simultaneous hot and cold leg injection is established in accordance with the Emergency Operating Procedures (EOPs). The EOPs are based on the NRC approved CE Emergency Procedure Guidelines and no distinction/classification between large and small breaks is made. The HPSI pumps discharge lines are realigned so that the total injection flow is split between the hot and cold legs. The hot side injection is achieved by injection in the RCS through the pressurizer auxiliary spray system. Boron precipitation/accumulation will not occur as long as all of the following conditions are satisfied:

1. Maximum Safety Injection Tank and Safety Injection Refueling Water Tank boron concentration does not exceed 2400 ppm.

2. Simultaneous Hot and Cold Leg Injection is initiated within 8.5 hours of the start of the LOCA if shutdown cooling cannot be established.

3. Both Hot and Cold Leg injection rates must be at or above 140 gpm (Ref. 14.15-6).

In the event that the auxiliary spray line becomes inoperable, the hot side injection is achieved by realigning the LPSI pump discharge to the shutdown cooling suction line in conjunction with the opening of a PORV. With the PORV open, the depressurization is sufficient to create a flushing flow either by allowing the RCS to refill or by establishing adequate hot side LPSI flow. Sufficient injection flow is provided to both cool the core and flush the reactor vessel for an indefinite period of time. This injection mode provides cooling for the RCS and prevents boric acid precipitation/accumulation in the vessel following a LOCA.

For the small break LOCA with the steam generator(s) available to remove decay heat, the cooldown of the RCS and long term decay heat removal is provided by natural or forced circulation cooling. For the small break LOCA, without heat removal through a steam generator, cooldown of the RCS and long term decay heat removal is provided by once-through-cooling, i.e., opening of the pressurizer power operated relief valves (PORVs) to release steam from the RCS. This action depressurizes and maintains the RCS pressure below the HPSI shutoff head, allowing the HPSI pumps to flush the core and accelerate refilling of the RCS.

14.15.6.1 Large Break LOCA Results

For a large break LOCA, boric acid precipitation/ accumulation is minimized by the core flushing flow which is provided by the simultaneous hot side and cold side injection from the HPSI pumps. The simultaneous hot side and cold side injection mode is initiated within 8.5 hours post-LOCA if shutdown cooling cannot be established in accordance with the EOPs.

Large break long term cooling without establishment of shutdown cooling is achieved for those break sizes for which simultaneous hot and cold side injection can both flush and cool the core. A break size of .005 ft^2 is the smallest for which the large depressurization maintains the RCS pressure sufficiently low to allow a HPSI pump to flush and cool the RCS. This process is also successful for small breaks as large as 0.015 ft^2.

The above description considered only the condition where off-site power is unavailable. With offsite power available, it is possible to more quickly cool down the RCS using the turbine bypass system and thereby initiate operation of the shutdown cooling system. However, opening of the PORVs with HPSI flow is sufficient to maintain decay heat removal for an indefinite period of time such that it is not necessary to initiate operation of the shutdown cooling system to assure continued heat removal.

14.15.6.2 Small Break LOCA Results

For a small break LOCA operator response is proceduralized in the LOCA and Functional Recovery EOPs. If less than 8.5 hours have elapsed since the start of the LOCA and Shutdown Cooling can be established, cooldown of the RCS is accomplished with the steam generators, if available. If the steam generators are not available, long term cooling is initiated by once-through-cooling by opening the PORVs. Opening of the PORVs results in cooling and reducing the RCS pressure sufficiently such that the HPSI pump refills and cools the RCS. The refilling and cooling of the RCS results in maintaining the boric acid concentration in the vessel well below the precipitation limit by dispersing the boron through the RCS by natural circulation.

The results of the analysis demonstrate that, for break sizes of 0.015 ft^2 or smaller, the RCS will refill and subsequently achieve a subcooled condition. The boric acid concentration in the vessel is also maintained well below the Westinghouse LOCA methodology (References 14.15-30 and 14.15-31) precipitation limit of 23.53 wt% as approved by the NRC, prior to refill.

14.15.7 Other LOCA Event Analyses

14.15.7.1 Loss of Coolant Accident During Shutdown

At any time after the Safety Injection Tanks (SITs) are isolated, the Reactor Coolant System pressure is 400 psia or less (Ref. 14.15-23), and the total stress in any component will be less than the total stress at the design pressure. Therefore, the possibility of a LOCA during shutdown cooling becomes even more remote than while at power.

The pressurizer pressure low signal (PPLS) has been bypassed below 1600 psia, and the safety injection tanks will be valved out when the system temperature and pressure reach 400°F and 400 psia respectively during shutdown. Using a maximum cooling rate of 75°F/hr, the above conditions are reached in less than 2 ½ hours. Less than 1 ½ hours later, when the system temperature and pressure reach 300°F and 250 psia, the system is placed in the shutdown cooling mode. In this mode, the coolant temperature is reduced from 300°F to 140°F in about 24 hours.

This shutdown procedure will usually occur only once per year or at most a few times per year. For each shutdown, a period of about 25 hours exists during which automatic initiation of the ECCS is not available, before refueling temperature is reached.

Bypassing of the PPLS does not bypass the containment pressure high signal (CPHS). The CPHS will still initiate safety injection and the automatic sequences involved. Furthermore, after the SITs are isolated, the operator will still be alerted to a LOCA by a combination of the following alarms and/or indications:

1. Low pressurizer level alarm
2. Low reactor coolant system pressure
3. High containment pressure alarm
4. Containment activity alarm
5. Containment temperature
6. Containment sump level alarm
7. Shutdown cooling temperature
8. Component cooling water temperature to and from the containment air cooling and ventilating system
9. Low volume control tank level alarm

For large breaks within the reactor coolant system, all of the above alarm indications will normally be present. As break size decreases, so will the number of indications and alarms. For the break size equal to or less than the capacity of one charging pump, the least number of indications and alarms that can be postulated to occur is three; namely, increases in: (1) containment activity, (2) temperature, and (3) temperature of the component cooling water to and from the containment atmosphere.

The containment air cooling and ventilating system can be used to reduce the high containment building pressure and to remove decay heat from the building if it becomes necessary to stop shutdown cooling. The capacity of the containment air cooling and ventilating system will be in excess of that required since the energy release will be far less than for a LOCA at full power; thus, no spray system backup should be required. Diesels will also be started to provide a standby source of power if outside power is lost during the accident.

An evaluation was performed to define the minimum Emergency Core Cooling (ECC) equipment operability requirements necessary to mitigate a LOCA during shutdown conditions (Ref. 14.15-32 and 14.15-33).

Minimum safety injection required to be operable during shutdown conditions are as follows:

<u>Mode 3 with RCS Pressure ≥1700 psia</u>

Safety Injection required is two LPSI and two HPSI trains aligned for automatic actuation. The HPSI and LPSI flow paths consist of piping, valves, and pumps that enable water to be injected into the Reactor Coolant System.

Four operable safety injection tanks (SITs) with isolation valves open.

Mode 3 with RCS Pressure <1700 psia and RCS Temperature ≥300°F

Safety injection required is one HPSI train aligned for automatic actuation. The exception is during startup (shutdown for at least 24 hours), one HPSI train is available for manual actuation below an RCS temperature and pressure of 450° F and 1700 psia. The HPSI pumps shall be disabled in accordance with Technical Specification 2.3 LTOP requirements. The HPSI flow path consists of piping, valves, and a pump that enables water to be injected into the Reactor Coolant System.

Four operable safety injection tanks (SITs) with isolation valves open. The SIT isolation valves may be closed when RCS pressure is less than 400 psia. The exception is during startup (shutdown for at least 24 hours), the SITs are required to be unisolated before reaching an RCS temperature and pressure of 515°F and 1700 psia.

14.15.7.2 Steam Generator Tube Failures in Conjunction with a Double-Ended Cold Leg Guillotine Break

An analysis was undertaken to determine how many steam generator tubes must fail, in conjunction with rupture of a reactor coolant system cold leg pipe, to cause steam binding sufficient to prevent emergency cooling water form rising above the midplane of the core.

The analysis showed there is no limit to the number of steam generator tubes that can rupture concurrent with a 24-inch double-ended cold leg break which will prevent water from rising to the core midplane. This is because the leaking steam generator will eventually discharge its contents to the primary system which will then blow down to the containment reducing the reactor coolant pressure to an acceptable value for refill. The nature of the refill inside the core barrel will depend upon the time at which the core pressure drops below that of the containment plus the static head of water in the downcomer annulus. Should the above condition exist shortly (about 1 minute) after the reactor coolant pressure drops to about 200 psig, then the SITs will contribute to the refill. However, should the primary system blowdown take longer, the refill of the barrel will be based on flow from the high and low pressure safety injection pumps.

14.15.7.3 Break Size Consistent With Charging Pump Capacity

Consideration has been given to the maximum reactor coolant system break size for which the charging pumps will make up the flow loss to the containment; so that a normal shutdown may occur.

As described in Section 9.2, there are three 40 gpm charging pumps. The number of charging pumps in operation and the corresponding maximum break area for which the charging pumps will make up the flow loss are given in Table 14.15-7. The discharge rate was determined from the orifice flow equation with a value of unity employed for the discharge coefficient.

Table 14.15-7 - Maximum Break Area Consistent with Charging Pump Capacity

No. of Pumps	Area (ft^2)	Equivalent Circular Diameter (in.)
1 (40 gpm)	1.37×10^{-4}	0.160
2 (80 gpm)	2.75×10^{-4}	0.224
3 (120 gpm)	4.12×10^{-4}	0.276

14.15.7.4 Core and Internals Integrity Analysis

The consequences of a LOCA on the reactor internal structure have been analyzed for reactor coolant system pipe breaks up to a double-ended rupture of a 32-inch pipe. Following a pipe rupture, two types of loading occur sequentially. The first is an impulse load of 15 to 30 milliseconds duration caused by rapid system depressurization from initial subcooled conditions to saturated conditions. This initial blowdown phase is followed by a two-phase fluid blowdown which persists for time periods varying up to several seconds, depending on the size of the postulated rupture.

In the early portion of the blowdown, acoustic waves propagate through the Reactor Coolant System. The WHAM code (Ref. 14.15-7) is used to calculate the pressure variations in the system following pipe rupture. WHAM calculates the impulse pressure loadings which the system is subjected to during passage of the pressure waves through the system.

For the saturated portion of the blowdown, the loadings on the reactor internals are associated with the fluid drag forces imposed by the high velocity two-phase fluid in its flow to the break location. The short term impulse forces are generally greater than the long term drag forces, except for the loads on some of the CEA shrouds in the case of a pipe rupture near the pressure vessel outlet nozzle.

The results of the blowdown force analyses for the reactor core support system are presented in Table 14.15-8, for a 32-inch hot leg break and for a 24-inch cold leg break, both for full power and zero power initial conditions.

Table 14.15-8 -Maximum Pressure Difference Across Core and Core Support Structure

	Pressure Difference, psi	
	32 in. Outlet Pipe Break	24 in. Outlet Pipe Break
Zero Power		
Lower Structure	54	40
Reactor Core	163	154
Upper Guide Structure	91	35
Full Power		
Lower Structure	29	30
Reactor Core	95	151
Upper Guide Structure	52	45

The maximum calculated stresses and deflections during blowdown for critical reactor components were found to be below allowable stresses (Ref. 14.15-19). Reference 14.15-19 lists the corresponding allowable values of stress, pressure, or deformation based on the design criteria specified in Section 3.2, along with the estimated values of stress, pressure, or deformation at which failure would occur. It is emphasized that the shutdown mechanism for large breaks is voiding of the core. It is not necessary for the CEAs to insert (see Section 1.5.5).

During an inlet pipe break, the core support barrel is subjected to time dependent axial loads and axially varying radial pressure differentials. Axial stresses in the core support barrel and shear and bending stresses in the lower support structure were evaluated, using conservative stress analysis methods. The peak of the axial pressure pulse, calculated from WHAM, was applied as a steady loading. The SEAL-SHELL computer program was used to assure that stresses and deformation are within design limits when the barrel is subjected to the peak of the time-dependent axially varying, radial pressure distribution.

During an outlet break, the core support barrel is subjected to a time dependent upward force and external radial pressure. Loads on the upper guide structure were evaluated by calculating the acceleration of the core under the time varying axial pressures (during the subcooled portion of the LOCA) and equating the kinetic energy of the core to the strain energy of the upper guide structure and core after impact. The strain energy is then related to stresses in the system. Bending stresses in the upper guide structure were evaluated during two-phase flow by using peak values for the axially varying pressure forces during this regime. The upper guide structure, modeled as a system of continuous and discrete elements, was subjected to a pressure time history, and axial stresses were evaluated.

The analyses which have been performed indicate that design limits are not exceeded even when dynamic effects are taken into account.

14.15.7.5 Reactor Vessel Thermal Shock

The effect of operation of the emergency core cooling system on the reactor vessel following a loss-of-coolant has been discussed in CE report A-68-9-1, "Thermal Shock Analysis on Reactor Vessel Due to Emergency Core Cooling System Operation", by W.H. Tupenny et al, March 15, 1968. This was submitted as Amendment 9 to the Maine Yankee License Application (AEC Docket No. 50-309). Additional information for this condition appears in CE report A-68-10-2, "Experimental Determination of Limiting Heat Transfer Coefficients During the Quenching of Thick Steel Plates in Water", by J.H. Simon, M.W. Davis and W.H. Tupenny, December 13, 1968. This report was placed in the public record in January, 1969. This work is discussed in Section 1.5.4.

14.15.7.6 Hydrogen Accumulation in Containment

Hydrogen accumulation in the containment is discussed in Section 14.17. The hydrogen produced by radiolysis is released to the containment building where it mixes with the steam-air atmosphere. In addition to the radiolytic hydrogen, the hydrogen produced by reaction of steam with the zircaloy cladding is considered.

To control hydrogen concentration, a purge of the containment building would be instituted when 3 percent (volume) hydrogen is reached. This value is below the flammability limit.

14.15.7.7 Reactor Operator Action

An evaluation has been performed to determine what actions the reactor operator would have to perform or may have to perform in the unlikely event of a design basis LOCA. Consideration was also given as to the allowable time periods within which the action would or might have to be performed. The results of the evaluation are contained in EOP-03.

In this evaluation it was assumed that all engineered safeguards and support systems functioned to fulfill the design objectives of each system. Certain additional actions may have to be performed in the unlikely event of a malfunction of portion(s) of certain system(s); however, this evaluation was performed to specifically identify reactor operator actions that will or may be required in the event of a loss-of-coolant accident.

This evaluation began by assuming a double-ended rupture of a 32-inch reactor coolant system pipe. This is followed by the response of the systems in the order and manner in which the various components of the systems are expected to function.

The results of this evaluation indicate that, in the event of a LOCA with the engineered safeguards and support systems functioning to fulfill design objectives, no control actions are required by the reactor operator for several hours following the accident.

14.15.8 Radiological Consequences of a LOCA

The ECCS, following a design basis LOCA (double-ended break), limits the clad temperature to well below the melting point and ensures that the reactor core remains intact and in a coolable geometry, minimizing the release of fission products to the environs. However, in order to demonstrate that the Fort Calhoun Station does not represent any undue radiological hazard to the public, the off-site doses and doses to Control Room personnel from a postulated LOCA have been calculated to assure that the radiological hazards are below the limits of 10 CFR Part 100 (Ref. 14.15-22) or International Commission on Radiological Protection Publication 30 (ICRP) Section 6.4 (Ref. 14.15-8) as appropriate.

14.15.8.1 Off-Site Radiological Consequences

The off-site doses from a postulated LOCA have been calculated based on the following assumptions and conditions (Ref. 14.15-25):

1. The reactor core equilibrium noble gas and iodine inventories are based on long-term operation at a power level of 1500 MWt.

2. 100 percent of the core noble gas inventory and 25 percent of the core iodine inventory are immediately available for release from the containment.

3. Of the iodine fission product inventory released to the containment, 91% is in the form of elemental iodine, 5% is in the form of particulate iodine, and 4% is in the form of organic iodine.

4. The following credit is taken for spray removal of halogens. Iodine Spray Removal:

Elemental	12.37 hr^{-1}
Particulate	5.244 hr^{-1} (DF$\leq$50)
	12.37 hr^{-1} (DF>50, up to 7218 sec after LOCA)

Recirculation Spray Phase

Elemental	16.94 hr^{-1}
Particulate	7.947 hr^{-1} (DF$\leq$50)
	0.977 hr^{-1} (DF>50, up to 7218 sec after LOCA)

5. The containment leak rate is 0.1 percent of the free volume for the first 24 hours, and 0.05 percent of the free volume for the remaining duration of the accident.

6. The Containment Air Cooling and Filtering System is not credited for the removal of halogen gases from the containment atmosphere.

7. Containment purge system is started when the hydrogen concentration in the containment reaches three (3) volume percent. Its operation is assumed to be initiated no sooner than 30 days following the accident, therefore, this source is not considered. (NOTE: THE RADIOLOGICAL CONSEQUENCES OF HYDROGEN PURGE SYSTEM ARE FULLY DISCUSSED IN SECTION 14.17 OF THE USAR).

8. The dispersion factor for the EAB is 2.56×10^{-4} sec/m^3, and the dispersion factor for LPZ outer boundary is conservatively assumed (based on a dispersion factor for 0 - 8 hours) as: 4.53×10^{-6} sec/m^3 (Ref. 14.15-24).

Period	x/Q (sec/m^3)
0-2 hr	2.51E-5
2-8 hr	7.29E-6
8-24 hr	4.83E-6
1-4 day	1.98E-6
4-30 day	5.49E-7

9. The composite of equilibrium core inventory of iodine and noble gas assumes an initial enrichment of 3.5% to 5% for Fort Calhoun Station (Reference 14.15-24).

10. Containment net free volume is $1.05 \times 10^5 ft^3$.

11. The leakage from ESF equipment (Ref. 14.15-35), i.e. high and low pressure SI pumps, etc, is assumed to be 4000 cc/hr, twice the maximum Technical Specification limit, beginning at the time of the recirculation actuation signal (20.4 minutes) and continuing for the duration of the accident (30 days). The volume of the containment sump is 314,033 gallons and contains 50% of core iodine inventory.

12. No filtering is credited for the ESF leakage.

13. There are three activity release pathways for radioactive material: 1) Containment leakage, 2) ESF/SIRWT leakage, and 3) Containment Vacuum Relief Line leakage.

Reactor Core Activity for Extended Burnup

The nuclide inventories (curies) in the reactor core used in determining the EAB and LPZ doses are shown below (from Reference 14.15-29).

Nuclide	Inventory (curies)
{I-I29}	1.39
[I-130]	4.44E5
I-130	8.34E5
I-131	4.08E7
I-132	5.97E7
I-133	8.47E7
I-134	9.47E7
[I-134]	8.11E6
I-135	8.04E7
I-136	3.78E7
[I-137]	3.98E7
[I-138]	2.00E7
Kr-83m	5.43E6
Kr-85m	1.15E7
Kr-85	4.35E5
Kr-87	2.32E7
Kr-88	3.25E7
Kr-89	4.09E7
Kr-90	4.40E7

Xe-131m	5.35E5
Xe-133	8.48E7
Xe-131m	2.64E6
Xe-135m	1.75E7
Xe-135	3.08E7
Xe-137	7.71E7
Xe-138	7.38E7
Br-82	1.16E5
Br-83	5.40E6
Br-85	1.15E7
[Br-87]	1.84E7
[Br-89]	1.24E7
[Br-90]	6.62E6

{ } Denotes tracer isotope
[ ] Denotes parent isotope only

Based on these assumptions, the resulting off-site doses are below the 10 CFR Part 100 limits:

EAB:	Pathway	Thyroid (rem)	Whole Body (rem)
	Vacuum Relief Line Leakage	0.0313	7E-5
	Containment Leakage	17.91	0.6997
	ESF/SIRWT Leakage	1.225	0.00648
	Total	19.17	0.706

LPZ:	Pathway	Thyroid (rem)	Whole Body (rem)
	Vacuum Relief Line Leakage	0.0307	7E-6
	Containment Leakage	3.189	0.1045
	ESF/SIRWT	0.6118	0.00295
	Total	3.84	0.107
	10 CFR 100	300.0	25.0

14.15.8.2 Post-LOCA Doses to Control Room Personnel

Control Room Radiation Shielding

Radiation shielding is provided for the control room envelope by concrete walls with a density of 145 pounds/ft^3 as follows (Ref. 14.15-9):

Control Room Wall	Thickness
North	2' 0"
South	1' 3"
East	1' 6"
West	1' 6"
Roof	1' 6"

In addition, as a result of a previous control room shielding review (Ref. 14.15-10), a 1 foot thick concrete wall is provided to shield personnel from containment spray pipes in the auxiliary building. Major penetrations in the bulk shielding include two door entrances in the east wall, two duct penetrations in the far western corner of the south wall, and two penetrations in the roof: one for the toilet exhaust and one for the elevator machine room exhaust.

Radiological Habitability Analyses

Radiological analyses have been performed to assure that the radiation doses to control room personnel do not exceed the limits of ICRP-30 (Reference 14.15-26) for postulated design basis accidents. The methodology, data, assumptions, and calculated results for each design basis accident is presented below.

Design parameters for the control room ventilation and emergency filtration system used in the radiological habitability analyses are as follows:

1. The emergency filtration system has a capacity of 2,000 cfm to filter an outside makeup air flow rate of 1,000 cfm $\pm$100 cfm plus recirculated control room air. The radiological analyses assume an outside makeup flow rate of 1,100 cfm with 900 cfm on recirculated air.

2. The emergency filtration system provides a filtration efficiency of 99 percent for both elemental and organic iodine.

3. An unfiltered inleakage of 8 scfm is assumed into the control room. This assumption is based on the test data of the control room ventilation system.

4. Iodine filtration of the recirculated air is assumed to be unavailable for 2 hours post accident to account for repair of the non-redundant recirculation duct isolation damper.

5. The normal operation mode unfiltered outside air intake flow rate of 1,000 cfm is isolated within 15 seconds upon actuation of the emergency filtration system.

Loss of Coolant Accident

The radiological consequences due to a design basis LOCA have been analyzed and include an evaluation of the radiological impact due to airborne radioactivity inside and outside the control room, as well as direct shine from contained radiation sources. Dose contributions from the following sources have been evaluated:

1. Airborne Radioactivity within the Control Room (Ref. 14.15-11)
2. Overhead Cloud Shine (Ref. 14.15-14)
3. Control Room Emergency Charcoal Absorbers (Ref. 14.15-15)
4. Piping containing Post-LOCA Radioactivity (Ref. 14.15-16)
5. Containment Shine (Ref. 14.15-17)
6. ESF Internal/Externa: Radiological Leakage (Ref. 14.15-18)

Each dose contributor is discussed in detail below.

Airborne Radioactivity within the Control Room

Control room doses due to airborne radioactivity releases from containment and engineered safety feature (ESF) leakage have been evaluated. The calculated doses are as follows (Ref. 14.15-11):

Pathway	Whole Body (rem)	Thyroid (rem)	Beta (rem)
Vacuum Relief Line Leakage	7E-5	0.374	5E-4
Containment Leakage			
Unfiltered activity	0.017	19.60	0.311
Filtered activity	1.411	9.286	25.34
ESF/SIRWT Leakage			
Unfiltered activity	1.14E-4	1.039	1.06E-3
Filtered activity	0.0113	1.429	0.104
Total	1.44	31.7	25.8

The doses have been calculated based on the following assumptions and conditions:

1. Airborne radioactivity releases are based on the same data and assumptions outlined previously for calculating the off-site doses.

2. The control room emergency iodine filtration system is initiated 44 seconds post-LOCA.

3. Credit for decay of fission products in containment during the duration of the accident is taken.

4. No credit is taken for the halogen gas removal abilities of the containment air cooling and filter units even though the redundant system should be effective in removing the elemental and particulate halogens.

5. The control room net free volume is 45,100 cubic feet.

6. Released via containment wall:

Period	x/Q (sec/m^3)
0-2 hr	4.87E-3
2-8 hr	4.19E-3
8-24 hr	2.11E-3
1-4 day	1.61E-3
4-30 day	1.35E-3

Released via Aux. Bldg stack:

Period	x/Q (sec/m^3)
0-2 hr	3.16E-3
2-8 hr	2.37E-3
8-24 hr	1.16E-3
1-4 day	8.93E-4
4-30 day	7.15E-4

Overhead Cloud Shine

The direct radiation shine dose to control room personnel from an overhead cloud of airborne radioactivity due to containment and ESF leakage was calculated (Ref. 14.15-14). The 30-day integrated radiation dose for bulk control room shielding (no penetrations) is considered to be 1.2 rem. This number was proved conservative by Reference 14-15.10, which is based on the following data and assumptions:

1. Airborne radioactivity releases are based on the same data and assumptions outlined previously for calculating airborne radioactivity within the Control Room.

2. Concrete shield wall thickness is assumed to be 1'6" for all walls and the roof. This is a conservative average based upon the actual dimensions.

3. Dose is based on an unshielded semi-hemispherical cloud model per Reg. Guide 1.4 (Ref. 14.15-27) with attenuation credit for 1'6" of concrete at a density of 145 pounds/ft^3.

4. The airborne cloud radioactivity concentration is based on meteorological dispersion factors for the centerline of the control room envelope which is about 21 meters from the containment in the north sector.

The direct radiation doses in localized areas of the control room due to the major penetrations have also been evaluated using a radiation dose of 1.2 rem through bulk control room shielding. These penetrations include the following:

- Two door entrances in the east wall.
- Two ventilation duct penetrations (12" and 16" diameter) in the far western corner of the south wall.
- Toilet exhaust penetration (12" X 12") in the roof above the mezzanine office area.
- Elevator machine room exhaust penetration (22" X 22") in the roof above the elevator machine room.

The radiation shine from the ventilation duct penetrations in the south wall has not been quantified since this impacts the mechanical equipment room, which is not habitable for continuous occupancy as discussed in Reference 14.15-12. Similarly, radiation shine through the elevator machine exhaust penetration has not been quantified since this area does not require post-accident occupancy.

Radiation shine from the control room doors was calculated as a function of distance into the control room (Ref. 14.15-14). Based on these results, at a distance of 8 feet a 3 rem integrated dose is calculated which when added to other doses is less than the 5 rem ICRP-30 limit.

Locations within 8 feet of the doors will not be occupied to any significant extent, and, therefore, will not present a radiation dose concern.

Control Room Emergency Charcoal Absorbers

The maximum calculated dose in the control room due to direct shine from the radioactivity buildup on the control room charcoal absorber is 0.043 rem gamma whole body (Ref. 14.15-15). Shielding credit is taken for the 1'3" thick south concrete wall.

Piping Containing Post-LOCA Radioactivity

The only significant dose contribution from piping sources is from containment spray piping at elevations 1032" and 1036'8" of the auxiliary building. The 30-day integrated dose at various control room locations has been calculated as follows (Ref. 14.15-16):

Location	Dose (rem)
Mechanical Equipment room	9.8
Toilet Room/Lunch Room	1.54
Auxiliary Control Panel Area	0.50
Main Control Board Area	0.008
Mezzanine Office	1.4

The calculated doses within the mechanical equipment room exceed ICRP-30; however, this area will be administratively controlled during post-accident conditions. The maximum dose for all other areas of the control room envelope is 1.54 rem. The most representative dose for the control room envelope is 0.008 rem and corresponds to the dose calculated for the main control board area, this value is used as input to the calculation of the total gamma whole body dose from significant direct shine contributors.

Containment Shine

The direct shine dose from the containment, which includes the dose from the airborne radioactivity in the containment atmosphere and the radioactivity buildup in the containment cooling and filtering system, was calculated to be 0.128 rem (Ref. 14.15-17). The iodine filters are accounted for in order to create a more bounding environment.

Summary of LOCA Doses

A summary of the radiological consequences due to a design basis LOCA is as follows:

	30-Day Post-LOCA Integrated Dose (rem)		
Dose Contribution	Gamma	Thyroid	Beta Skin
1. Airborne Radioactivity within Control Room	1.44	32.55	25.79
2. Overhead Cloud Shine	1.2	N/A	N/A
3. Control Room Emergency Charcoal Absorbers	0.043	N/A	N/A
4. Piping containing Post-LOCA Radioactivity (Main Control Board Area)	0.008*	N/A	N/A
5. Containment Shine	0.128	N/A	N/A
Total	2.82	32.55	25.79
ICRP-30 Dose Limits	5	50	50

* This dose applies to the main control board area of the control room. The maximum dose within the control room envelope occurs in the toilet/lunch room area and is calculated to be 3.5 rem.

Based upon the above results it can be seen that the calculated doses are within the limits of ICRP-30. The 5 rem gamma dose limit is met throughout the control room envelope except for the mechanical equipment room, which is not continuously occupied at any time. The area directly inside the two control room doors is also calculated to exceed 5 rem; however, these areas will not be occupied to a large extent during post-LOCA conditions.

14.15.9 Conclusions

The LOCA analysis demonstrates that the ECCS provides adequate core cooling, by keeping the core in a coolable geometry, over the entire spectrum of breaks, including a double-ended hot leg guillotine.

The results of radiological consequences show that the thyroid and whole body doses, using the conservative assumptions, are well within the limits of 10 CFR Part 100 at the EAB and LPZ, and well below the limits of ICRP-30 for Control Room Personnel.

14.15.10 Specific References

14.15-1 "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Reactors", 10CFR50.46 and Appendix K of 10CFR50.

14.15-2 Bordelon, F.M., Massie, H.W. and Zordan, T.A.; "Westinghouse ECCS Evaluation Model-Summary", WCAP-8339; July 1974.

14.15-3 "SEM/PWR-98: ECCS Evaluation Model for PWR LBLOCA Applications," EMF-2087 (P) (A), Revision 0, June 1999.

14.15-4 XN-NF-82-49 (P)(A), Revision 1, "Exxon Nuclear Company Evaluation Model EXEM PWR Small Break Model, " Exxon Nuclear Company, Inc., April 1989.

14.15-5 XN-NF-82-49 (P)(A), Supplement 1, Revision 1, "Exxon Nuclear Company Evaluation Model Revised EXEM PWR Small Break Model, Exxon Nuclear Company, Inc., December 1994.

14.15-6 Letter from M.J. Weber (Westinghouse) to R.L. Phelps (OPPD), "Long Term Core Cooling Evaluation", 91-CF-G-0082; December 17, 1991.

14.15-7 Fabric and Stanislav; "Early Blowdown Waterhammer Analysis for Loss of Fluid, Test Facility", 65-28-4, AEC Control No. AT-(10-1)-1165; June, 1965 (Revised April 1967).

14.15-8 OPPD LIC-00-0025, "Application for Amendment of Facility Operating License No. DPR-40," 04/14/00.

14.15-9 Stone & Webster Calculation No. 16472.26-UR(B)-006-0, 30 Day Integrated Dose Due to Cloud Surrounding the Control Room, 6/28/89.

14.15-10 OPPD Letter LIC-80-0166, December 31, 1980.

14.15-11 FC68012, Stone & Webster Calculation No. 08639-UR (B)-004, CCN No. 2 "Site Boundary and Control Room Submersion Doses Due to LOCA (TID 14844 Source Term, 8 SCGM Unfiltered Control Room In-Leakage, Containment Spray Iodine Scrubbing, No Containment Charcoal Filter Credit," 04/19/01

14.15-12 Control Room Habitability Evaluation for NUREG-0737, Item III.D.3.4, November, 1989.

14.15-13 Murphy, K.G., and Kampe, K.M.; "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Design Criteria 19"; 13th Air Cleaning Conference, August, 1985.

14.15-14 Stone & Webster Calculation No. 16472.26-UR(B)-016-0, 30 Day Integrated Gamma Dose Through Control Room Doors From LOCA Cloud Direct Shine; 6/28/89.

14.15-15 Stone & Webster Calculation No. 16472.26-UR(B)-015-0, 30 Day Integrated Dose (gamma) to Operators in the Control Room due to Direct Shine from the Control Room Intake/Recirculation Filters; 6/28/89.

14.15-16 Stone & Webster Calculation No. 16472.19-UR-001-0, 30 Day Post-LOCA Total Integrated Doses in the Control Room Due to Piping in the Auxiliary Building; 11/18/88.

14.15-17 Stone & Webster Calculation No. 16472.26-UR(B)-007-01, 30 Day Integrated Dose (gamma) to Operators in the Control Room due to Direct Shine from the Containment (airborne activity and carbon filters); 6/28/89.

14.15-18 Radiological Consequences of Internal and External ESF Equipment Leakage, FC05934; 5/26/92.

14.15-19 Maximum Stresses, Pressures and Deflections for Critical Internals Components; CENPSD-110-P.

14.15-20 CEN-268 Rev. 1, "Justification of Trip Two/Leave Two Reactor Coolant Pump Trip Strategy During Transients" May 1987.

14.15-21 "FC6805, Stone & Webster Calculation No. 08639-US (B)-003, "Iodine Removal Coefficents and Decontamination Factor-LOCA," 06/14/99.

14.15-22 Code of Federal Regulations, Title 10 Part 100.11 "Determination of Exclusion Areas Low Population Zone and Population Center Distance" Current Revision.

14.15-23 WCAP-12476 "Evaluation of LOCA during Modes 3&4 Operation for W NSSS," November 1991

14.15-24 FC6807, Stone & Webster Calculation No. 08639-WM (B)-002, "Atmospheric Dispersion Coefficients (X/Qs at the Control Room and TSC Air Intakes for Identified Design Basis Accidents' Release Points," 10/04/99.

14.15-25 Letter Westinghouse (R.G. Creighton) to OPPD (R. L. Phelps) 91CF*-G-0054 dated August 22, 1991.

14.15-26 ICRP-30 Methodology, "Limits for Intakes of Radionuclides by Workers".

14.15-27 U.S. Atomic Energy Commission Regulatory Guide 1.4 "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors" Rev. 2, June 1974.

14.15-28 Mixed Vendor Core Data List for Fort Calhoun Unit 1, EA-FC-90-004, R0.

14.15-29 Engineering Analysis EA-FC-90-111, Rev. 0, Shielding Calculation Revision - Phase I - Comparison Study.

14.15-30 Letter CLC-NS-309 from C.L. Caso (Westinghouse) to T.M. Novak (NRC), dated April 1, 1975.

14.15-31 Westinghouse Memorandum SE-SE-1200, from H.V. Julian to M. Harding and I. Ratsep, dated May 26, 1977.

14.15-32 Letter from J.M. Cleary (ABB-CE) to J.L. McManis (OPPD), "Recommended ECCS Equipment Operability Requirements when the Reactor is not Critical", ST-98-272, May 21, 1998 (Calculation FC06738).

14.15-33 Letter from J.M. Cleary (ABB-CE) to A.W. Richards (OPPD), "Relaxation of ECCS Equipment Requirements While Shutdown," ST-99-221, March 19, 1999 (Calculation FC06739)

14.15-34 EMF-2506, Revision 0, "Fort Calhoun Large Break LOCA/EDDS Analysis, " Siemens Power Corporation, December 2000.

14.15-35 EMF-2482, Revision 0, "Fort Calhoun Small Break Analysis," Siemens Power Corporation, December 2000.

14.15-36 01CF-G-004, CAB-01-57, Omaha Public Power District, Fort Calhoun, "Fort Calhoun Cycle 20 LOCA Reload Confirmation Transmittal," Westinghouse Electric Company, February 20, 2001.

14.15.11 General References

14.22 REACTOR COOLANT SYSTEM DEPRESSURIZATION INCIDENT

14.22.1 General

The RCS Depressurization event is characterized by a rapid decrease in the primary system pressure caused by either the inadvertent opening of both power operated relief valves (PORV's) or a single primary safety valve while operating at rated thermal power.

Following the initiation of the event, steam is discharged from the pressurizer steam space to the quench tank where it is condensed and stored. To compensate for the decreasing pressure, the water in the pressurizer flashes to steam and the proportional heaters increase the heat added to the water in the pressurizer in an attempt to maintain pressure. During this time, the pressurizer level also begins to decrease causing the letdown control valves to close and additional charging pumps to start in an attempt to maintain level. As the pressure continues to drop, the backup heaters energize to further assist in maintaining the primary pressure. A reactor trip is initiated by the TM/LP trip to prevent exceeding the DNBR SAFDL (Ref. 14.22-11).

In order to ensure that enough margin is built into the TM/LP trip, to prevent the DNBR SAFDL from being exceeded, a conservative pressure bias term for the TM/LP trip must be calculated. The pressure bias term accounts for the DNBR margin degradation, caused by the depressurization, between the time reactor trip conditions exist and the time of minimum DNBR. This time is primarily due to the signal processing delays in the TM/LP trip logic and the CEA clutch coil delay time.

14.22.2 Method of Analysis

The CESEC plant transient thermal-hydraulic code is used to simulate the overall response of the reactor coolant and steam systems during the transient (References 14.22-2, 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 14.22-6, 14.22-7, and 14.22-12). The DNBR safety limit includes a 2% mixed-core penalty (Reference 14.22-13).

Table 14.22-1 contains a list of the key input parameters for the CESEC plant simulation (Reference 14.22-9). The most negative moderator temperature coefficient (MTC) of reactivity is used to increase the coolant temperature feedback effects. This results on higher heat fluxes and thus, greater residual heat; thereby, minimizing DNBR. In order to maximize the negative reactivity feedback from the increasing fuel temperature, the negative doppler coefficient is used. The initial pressurizer pressure corresponds to the maximum allowed plus uncertainties. The charging pumps, the pressurizer proportional heaters, and the pressurizer backup heaters are assumed to be inoperable with the letdown valves open at the maximum flow. The higher initial pressure, the maximum letdown flow, and the inoperability of the pressurizer heaters and charging pumps, result in a faster rate of depressurization. These assumptions yield a lower transient minimum DNBR and a maximum pressure bias term (Reference 14.22-1).

Table 14.22-1 - " Key Parameters for the RCS Depressurization Event"

Parameter	Units	Value
Initial Core Power Level	MWt	1541.6
Core Inlet Coolant Temperature	°F	547
Pressurizer Pressure	psia	2172
Moderator Temperature Coefficient	$10^{-4}\Delta\rho/°F$	-3.5
RCS Flow Rate	gpm	230,000
Total Trip Delay Time (Processing plus CEA holding coil delay)	sec	1.4

14.22.3 Affected Plant Technical Specifications

The RCS depressurization event analysis uses inputs from the following Technical Specifications (Reference 14.22-10):

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

The results of the RCS depressurization event analysis are used as inputs to Technical Specification 1.3, Limiting Safety System Setting, Reactor Protective System.

14.22.4 Affected Plant Systems

For this event the affected plant systems are the reactor coolant system, the reactor protective system (TM/LP), and the reactivity control system.

14.22.5 Limiting Parameters for Reload Analysis

Reevaluation of the RCS depressurization incident is required when either of the following conditions exists.

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

Any changes to parameters and/or technical specifications must result in a DNBR that is 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 RCS depressurization event.

14.22.6 Results

The RCS Depressurization 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 14.22-8): The CESEC plant simulations from the Cycle 19 analysis of the RCS Depressurization incident (Reference 14.22-9) 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 14.22-8).

In addition to evaluating the DNB performance, the TM/LP LSSS settings for Cycle 19 were verified to be applicable for Cycle 20 (Reference 14.22-11) since the CESEC plant simulations were not required to be performed for Cycle 20. The Tm/LP LSSS settings are based on a Cycle 19 pressure bias value of 30.0 psia (Reference 14.22-9).

The sequence of events for the RCS Depressurization incident is presented in Table 14.22-2. Figures 14.22-1 through 14.22-4 shows the typical transient behavior of the core power, core average heat flux, reactor coolant system temperatures, and the reactor coolant system pressure.

The RCS Depressurization incident for Cycle 20 results in a minimum DNBR value that is greater that the HTP correlation 95/95 DNBR safety limit plus 2% mixed-core penalty (Reference 14.22-8).

Table 14.22-2 - "Sequence of Events for the RCS Depressurization Event"

Time (sec)	Event	Setpoint or Value
0.0	Inadvertent Opening of both Pressurizer Relief Valves	-----
7.77	Manual Trip	2079.37 psia
10.0	Time of Minimum DNBR	2054.41 psia

14.22.7 Conclusions

The analysis of this event shows that the minimum DNBRs calculated a greater than the minimum DNBR safety limit. Additionally, in Reference 14.22-11, the TM/LP LSSS settings for Cycle 19 were verified to be applicable for Cycle 20. The TM/LP LSSS settings are based on a Cycle 19 pressure bias value of 30.0 psia.

14.22.8 Specific References

14.22-1 OPPD-NA-8303-P, Rev. 04, Omaha Public Power District Reload Core Analysis Methodology "Transient and Accident Methods and Verification," Section 5.7.

14.22-2 "CESEC-Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," CENPD-107, CE Proprietary Report, April 1974.

14.22-3 "CESEC-Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," CENPD-107, Supplement 6, CE Nonproprietary Report, August 1979.

14.22-4 "CESEC-Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," December 1981, transmitted as Enclosure 1-P to LD-82-001, January 6, 1982.

14.22-5 Response to questions on CESEC, CEN-234(c)-P, Louisiana Power and Light Company, Waterford Unit 3, Docket 50-382, December 1982.

14.22-6 EMF-2062(P), Guidelines for PWR Safety Analysis, G104,035, "Inadvertent Opening of a Pressurizer Relief Valve (SRP 15.6.1)," June 1998.

14.22-7 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.

14.22-8 E-4350-595-1, "Ft. Calhoun Unit 1, Cycle 20: Non-LOCA Transient MDNBRs," December 2000.

14.22-9 "Cycle 19 RCS Depressurization Analysis," EA-FC-98-051, Rev. 0.

14.22-10 Fort Calhoun Operating License DPR-40 and Technical Specifications, including all amendments through Amendment 190, April 15, 1999.

14.22-11 EMF-2548, Revision 0, "Fort Calhoun Cycle 20 Statical Verification of LSSS and LOC Setpoints," Framatome ANP Richland, Inc. March 2001.

14.22-12 "EMF-92-153 (P)(A) and Supplement 1, "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," March 1994.

14.2-13 XN-NF-82-21 (P)(A), "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," Revision 1, September 1983.

APPENDIX A

QUALITY ASSURANCE PROGRAM

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1. INTRODUCTION

This Appendix describes the Omaha Public Power District's (OPPD) Quality Assurance Program for the operation of Fort Calhoun Station. The program is based on the criteria of Appendix B to 10CFR Part 50, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants"; General Design Criterion 3, Appendix A to 10CFR Part 50, "Fire Protection"; Subpart H of 10CFR Part 71, "Packaging and Transportation of Radioactive Material;" the applicable guidance provided in American National Standard, ANSI N18.7 "Administrative Controls and Quality Assurance for the Operations Phase of Nuclear Power Plants" and, ANSI N45.2, "Quality Assurance Program Requirements for Nuclear Power Plants," and its associated daughter standards; and, Regulatory Guide 1.120, Revision 1, "Fire Protection Guidelines for Nuclear Power Plants" and Appendix A to Branch Technical Position 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants." Attachment 1 details OPPD's specific commitments with respect to the ANSI N45.2 series, other industry QA standards, and associated NRC Regulatory Guides, including clarifications or alternatives used as a basis for OPPD's Quality Assurance Program and QA Plan.

The program is applied to: Critical Quality Elements (CQE) defined as those structures, systems, components, or items whose satisfactory performance is required to prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public; those fire protection systems and equipment used or installed in areas housing safety-related equipment, and other areas where an unsuppressed fire could potentially damage safety-related structures, systems or components; those activities affecting the components of radioactive material packaging for transport which are significant to safety; Limited Critical Quality Elements (Limited CQE) defined as those structures, systems, components or items whose satisfactory performance is required to prevent or mitigate the failure of those structures, systems, components or items identified as CQE.

2. ORGANIZATION

OPPD's organization for carrying out an effective operations phase Quality Assurance Program is shown in Figure A-1.

2.1 Vice President

The Vice President is the executive responsible for plant operations; formulation, implementation, and assessment of the effectiveness of the fire protection program; implementation and maintenance of the ALARA Radiation program; and packaging of radioactive material for transport. This executive as the Chief Nuclear Officer is responsible for approval, revision approval and overall implementation of the Quality Assurance Plan.

2.2 Division Manager

Division Manager is an upper level management position. When a Division Manager has responsibilities for fulfilling the Quality Assurance Program, the individual shall maintain sufficient authority and organizational freedom to implement the assigned requirements. Division Managers that are fulfilling quality assurance functions report directly to the executive serving as the Chief Nuclear Officer. The Quality Assurance Plan describes the responsibilities of each Division Manager.

2.3 Manager

Manager is divisional management position. When a Manager has responsibilities for fulfilling the Quality Assurance Program, the individual shall maintain sufficient authority and organizational freedom to implement the assigned requirements. Managers that are fulfilling quality assurance functions report directly to the appropriate Division Manager. The Quality Assurance Plan describes the responsibilities of each Manager.

Each Manager maintains a staff of qualified individuals to assist in the fulfillment of the requirements of the Quality Assurance Plan.

2.4 Plant Review Committee (PRC)

A committee composed of key management personnel designated as the Plant Review Committee (PRC) acts in an advisory capacity to the Manager-Fort Calhoun Station and serves in accordance with the Technical Specifications and the Plant Standing Orders.

2.5 Safety Audit and Review Committee (SARC)

A committee composed of highly qualified and experienced OPPD management personnel and consultants, designated as the Safety Audit and Review Committee (SARC), functions to provide independent review and audit of activities in accordance with the Technical Specifications and the SARC Charter. The SARC reports to and advises the Vice President on reviews and audits of the designated activities.

3. QA PROGRAM

3.1 Corporate Policy

The Omaha Public Power District (OPPD), as the owner and operator of Fort Calhoun Station, has established a Company policy to maintain and operate the facility with due regard for public and plant safety as prescribed by various regulatory requirements. Since there is a close correlation between safety and plant quality, the control of quality is a responsibility of every individual associated with station design, procurement, modification, maintenance, and operation.

The OPPD Quality Assurance Program delineates the established policy and quality requirements, and is implemented and applied to those activities as specified therein. It establishes the OPPD Quality Assurance Plan, sets forth the quality policies, defines the requirements, and specifies responsibilities within OPPD for implementing the program. Compliance with the QA Plan, as well as with the implementing procedures developed from it, is mandatory. Management gives full support to maintaining an effective quality program. Compliance with applicable requirements of the QA Plan is made a condition of contract for supporting companies.

The Vice President has overall responsibility and authority for the implementation of the Quality Assurance Program for Fort Calhoun Station. Revisions to the QA Plan are approved by the Vice President.

The CQE/Limited CQE structures, systems, and components controlled by the QA Program are identified in the CQE list for Fort Calhoun Station. Limited CQE items include consumables such as gaskets, packing and lubricants among many other off-the-shelf items. With respect to fire protection, the QA program is applicable to the following fire protection equipment in support of nuclear safety related equipment areas: fire water supply (pumps, main piping, and valves); fire suppression systems and hose racks; fire detection and alarm systems; fire area barriers and penetrations; emergency lighting that supports fire event credited manual actions; and breathing equipment intended for Fire Brigade use. The pertinent sections of the QA Plan are applied to the fire protection system to an extent consistent with safety. Therefore, Sections A.7, A.9, A.10, A.13 and A.14 of this program description are not applicable to the fire protection program.

The Quality Assurance Program applies to the procurement, maintenance, repair, and use of packaging for the transport of radioactive material. This shall include receptacles, wrappers, and their contents excluding fissile material and other radioactive material, but including absorbent material, spacing structures, thermal insulation, radiation shielding, devices for cooling and for absorbing mechanical shock, external fittings, neutron moderators, nonfissile neutron absorbers, and other supplementary equipment which has safety significance. All other activities (such as design, fabrication, assembly, and modification) are not covered by OPPD's QA Program and shall be satisfied by obtaining certifications from package suppliers that these activities were conducted in accordance with an NRC-approved QA Program. All transportation activities shall meet the requirements of 10 CFR 71 and Department of Transportation Regulations.

Any disputes which can not be resolved to the satisfaction of the Manager-Quality Assurance & Quality Control shall be brought before the Vice President via the Division Manager-Nuclear Assessments. The Vice President would then be responsible for resolving the dispute after considering all aspects of the issue.

Changes to OPPD's QA Program description are included in the update of the USAR.

3.2 QA Plan

The OPPD QA Plan requires that OPPD organizations and companies under contract to supply technical services or products for the plant comply with the following requirements:

a. The authority and duties of individuals and groups performing quality assurance functions are clearly established and delineated in writing. They have sufficient authority and organizational freedom to:

(1) identify quality problems

(2) recommend solutions for conditions adverse to quality

(3) verify implementation

b. An individual or group assigned responsibility for auditing that an activity has been correctly performed, is not directly responsible for performing the specific activity.

Copies of the QA Plan are issued in a controlled manner. A distribution list is maintained showing recipients of controlled QA Plan copies. Personnel signify the receipt of their copy of the QA Plan by signing and returning a receipt acknowledgment. Recommended changes to this plan are solicited and such recommendations are given due consideration by the Manager-Quality Assurance & Quality Control and OPPD management. Necessary revisions are prepared, reviewed for adequacy, approved and issued in a controlled manner. These revisions are also controlled by means of a receipt acknowledgment. Revisions are dated and identified with formal revision numbers as they are issued.

3.3 QA Program Procedures

The QA Plan requires that the various QA Program procedures be derived from approved QA policies by means of a review of these procedures, both prior to issuance and during audits of the activity prescribed by the procedure. Procedure reviews are accomplished in accordance with established procedures.

3.4 Training and Indoctrination

Personnel responsible for performing activities affecting quality are instructed as to the purpose, scope, and implementation of the QA Plan and QA Program manuals, instructions, and procedures by participation in training programs and on-the-job training.

The QA Plan requires that personnel performing activities affecting quality possess documented evidence that they are trained and qualified in the principles and techniques of the activity being performed. Procedures provide for training and qualification in the principles and techniques of the activity being performed, including:

(1) Nondestructive evaluation (NDE) personnel administration

(2) Auditor training and qualification

(3) Indoctrination and training of quality assurance personnel

Established procedures specify the training and qualification requirements. The QA auditing and surveillance programs provide assurance that the personnel are trained in the activity.

The scope, the objective, and the method of implementing the various indoctrination and training programs are prescribed in writing and records are maintained to verify the progress and success of the programs. This documentation is audited and the programs receive periodic reviews within the applicable divisions to verify their adequacy.

The indoctrination and training programs assure that the proficiency of personnel performing activities affecting quality is maintained by specifying retraining, re-examining, and/or recertifying in accordance with the specified requirements. The indoctrination and training programs provide for documenting the training sessions, describing the content, the date held, the attendees, and the results of any examinations conducted.

4. DESIGN CONTROL

The OPPD Quality Assurance Plan provides for several levels of design control for modification. OPPD design activities meet applicable QA Plan requirements for activities affecting quality. QA audits assure that OPPD's design control measures provide a clear definition of design interfaces, review and approval of designs, including changes or revisions, and that those performing design review activities are independent of those originating the design. The verification of engineering and design adequacy of the contractors' design documents is performed in accordance with an OPPD approved Quality Assurance program and procedures.

Requirements for OPPD design development and review are contained within the QA Plan. Administrative instructions for initiating, controlling and documenting modification of station equipment and facilities is provided. The Manager-Fort Calhoun Station is responsible for reviewing and approving designs prior to their implementation at Fort Calhoun Station. Utilization of the Plant Review Committee is governed by the station Operating Manual and plant Standing Orders for their review function.

If an unreviewed safety question is involved, the design is further reviewed by the Safety Audit and Review Committee as specified in the SARC Charter prior to submittal to the NRC for approval.

Procedures require an independent review of design documents for CQE/Limited CQE designs. Procedures assure that design characteristics can be controlled, inspected, and tested. Independent design review and verification activities are required by the QA Plan to be performed in accordance with approved procedures by appropriately qualified engineers for engineering calculations, specifications, and design drawings for items within the QA Program boundary.

The QA Plan and established procedures require that selected documents be reviewed to determine that they contain, as appropriate:

a. Applicable design bases, technical requirements, regulatory requirements, component and material identification, drawings specifications, codes and industry standards, tests and inspection requirements, and special process instructions for such activities as fabrication, cleaning, erection, packaging, handling, shipping, storage, and inspection;

b. Requirements that identify the documentation to be prepared, maintained, submitted, and made available to the purchaser for review and comment, such as drawings, specifications, procedures, inspection and test records, personnel and procedure qualifications, and chemical and physical test results on materials;

c. Requirements for the retention, control, and maintenance of documents and records for activities affecting quality.

The QA Plan and established procedures require that design adequacy be verified by systematic evaluation of the elements of the design with respect to requirements for design, safety, function, and quality. Verification may be accomplished by performing design reviews, by the use of alternate or simplified calculational methods, or by conducting a suitable test program. The verifying process shall be performed by individuals or groups other than those who performed the original design.

Detailed design or design changes involving Critical Quality Elements (CQE) that are performed by the Design Engineering-Nuclear Department are performed in accordance with approved procedures. Procedures require technical calculations and safety analyses be provided by the design engineer and describe how safety analyses and technical calculations are processed. Procedures provide design controls for compatibility of materials and accessibility for inservice inspection, maintenance, and repair.

Materials, parts, equipment and processes essential to CQE/Limited CQE and the fire protection program are required to be selected and reviewed for suitability of application. The methods of assurance of suitability are required to include independent design verification by individuals or groups competent in the applicable field of design and related nuclear power plant requirements.

The methods of selection and review are required to provide for (as applicable): reactor physics, stress, thermal, hydraulic and accident analyses; compatibility of materials; as low as practicable radiation levels; accessibility for in service inspection, maintenance and repair; test requirements and delineation of acceptance criteria for inspections and tests.

The QA Plan requires that measures be established to assure that applicable fire protection program guidelines and requirements are included in design and procurement documents prepared and that deviations there from are controlled. Field changes and design deviations that affect the intent of the modification shall be subject to the same level of controls, reviews, and approvals that were applicable to the original document. Quality standards are specified in the design documents such as appropriate fire protection codes and standards. Deviations or changes from these standards are individually approved. New designs and plant modifications, including fire protection systems, are reviewed by qualified personnel to assure inclusion of appropriate fire protection requirements. These reviews include items such as:

a. Reviews to verify adequacy of wiring isolation

b. Reviews to verify appropriate requirements for room isolation

c. Reviews to verify appropriate material is used

Materials, parts, and equipment for CQE/Limited CQE structures, systems, and components are procured in accordance with established procedures regardless of commercial or previous approval status. Procedures require an engineering and quality review of procurement documents for CQE and limited CQE items.

Procedures require that CQE/Limited CQE design changes affecting the design basis be processed through the design engineer and that an independent reviewer reviews the changes. The process is documented and retained as records.

The OPPD QA Plan requires that OPPD's manufacturers' and contractors' design activities meet applicable regulatory requirements for quality-related activities. The OPPD QA Plan requires verification that applicable regulatory requirements have been incorporated in activities affecting quality design review, audit, and surveillance of manufacturers and contractors. This assures that design input (applicable regulatory requirements and design bases as specified in the license application for safety related structures, systems, and components) for Fort Calhoun Station Unit No. 1 are correctly translated into design output documents (specifications, drawings, procedures, and instructions). QA audits assure that OPPD's manufacturers' or contractors' design control measures provide a clear definition of design interfaces, review and approval of initial design, including changes or revisions, and that those performing design review activities are independent of those originating the design.

The design activities of contractors for safety-related structures, systems, or components are required to comply with OPPD approved design development and control requirements.

5. PROCUREMENT DOCUMENT CONTROL

Appropriate requirements have been established by the OPPD Quality Assurance Plan to assure that procurement documentation is controlled and accurately reflects applicable regulatory requirements, design bases, and other appropriate requirements, such as industry codes and standards. Procurement documents and specifications require that bidders or suppliers submit for review by OPPD written quality assurance programs consistent with the importance and complexity of the materials, equipment, or service procured. Such vendors quality assurance programs are required to be consistent with pertinent provisions of Appendix B to 10CFR, Part 50, or Subpart H of 10CFR Part 71, as appropriate. OPPD satisfies these requirements as follows:

(1) Review of procurement documentation for CQE and limited CQE listed materials, equipment, and services is performed in accordance with established procedures which require OPPD personnel to review CQE and limited CQE procurement documents and document their review.

(2) Procurement documents for fire protection and radioactive material packaging materials, equipment, and services are reviewed, approved and documented by qualified personnel to verify the adequacy of fire protection and quality requirements. This review assures that fire protection requirements and quality requirements are correctly stated, inspectable and controllable; that there are adequate acceptance and rejection criteria; and that the procurement document has been properly prepared, reviewed, and approved.

(3) Procurement documents are reviewed to assure that the item is materially compatible with the environment in which it will be used and that applicable documentation is specified.

(4) Planned, periodic, and documented audits are performed by responsible OPPD personnel to provide assurance that the procurement activities of OPPD are being carried out in accordance with approved procedures.

The QA Plan and established procedures require that quality data be included in or appended to the procurement documents or engineering data attachments, as appropriate. The quality data prescribes as necessary:

(1) Quality requirements including use of procedures or instructions

(2) Requirements for a supplier quality program and documentation

(3) Requirements for documentary evidence of quality to be furnished by the supplier (e.g., test results, certification that specific requirements have been met, or traceability to the source)

(4) Access requirements for surveillance, inspection, and audits at the supplier's work site

The OPPD QA Plan requires that revisions or amendments which affect the safety or quality aspects of purchase orders, procurement documents or contracts be prepared, reviewed, and approved in the same manner as the original documents.

The procurement process for spares and replacement parts for Fort Calhoun Station, as required by the OPPD QA Plan and further delineated in established procedures, is more controlled than the original procurement process. The procurement process for FC1 occurred from 1967 to 1970; the 10CFR, Part 50, Appendix B, QA requirements were not invoked until 1971.

6. INSTRUCTIONS, PROCEDURES AND DRAWINGS

6.1 Written procedures and administrative policies shall be established, implemented and maintained that meet or exceed the minimum requirements of Sections 5.1 and 5.3 of ANSI N18.7-1972 and Appendix A of USNRC Regulatory Guide 1.33 except as provided in 6.2 and 6.3 below.

6.2 Each procedure as described in Section 6.1, and changes thereto, and any other procedure or procedure change that the Manager - Fort Calhoun Station determines to affect nuclear safety, shall be reviewed and approved as described below, prior to implementation.

6.2.1 Each procedure, or change thereto, shall be reviewed by a Qualified Reviewer (QR) who is knowledgeable in the functional area affected but is not the individual preparer. The QR may be from the same line-organization as the preparer. The QR shall render a determination in writing of whether or not cross-disciplinary review of a procedure, or change thereto is necessary. If necessary, such review shall be performed by appropriate personnel.

6.2.2 Each procedure, or change thereto, shall be reviewed by the Department Head designated by Administrative Controls Standing Orders as the responsible Department Head for that procedure, and the review shall include a determination of whether or not a 10 CFR 50.59 safety evaluation is required. If a 10 CFR 50.59 safety evaluation is not required, the procedure or change thereto, shall be approved by the responsible Department Head or the Manager-Fort Calhoun Station, prior to implementation. Administrative Controls Standing Orders, and the Fire Protection Program Plan shall be reviewed in accordance with the QA Program, Section 19, (6) and approved by the Manager-Fort Calhoun Station.

6.2.3 If the responsible Department Head determines that a procedure, or change thereto, requires a 10 CFR 50.59 safety evaluation, the responsible Department Head shall render a determination in writing of whether or not the procedure, or change thereto, involves an Unreviewed Safety Question (USQ) and shall forward the procedure, or change thereto with the associated safety evaluation to the PRC for review in accordance with the QA Program, Section 19, (6).a. If a USQ is involved, NRC approval is required prior to implementation of the procedure, or change.

6.2.4 Qualified Reviewers shall meet or exceed the respective qualifications for either Supervisors Requiring an AEC License, Professional-Technical Personnel, or Technical Support Personnel, as specified in ANSI N18.1 - 1971. Personnel recommended to be QRs shall be reviewed by the PRC and approved and designated as such by the PRC Chairman. The responsible Department Head shall ensure that a sufficient complement of QRs for their functional area is maintained in accordance with Administrative Controls Standing Orders.

6.2.5 Each procedure as specified by Section 6.1 shall be reviewed periodically as set forth in Administrative Controls Standing Orders.

6.2.6 Records documenting the activities performed under Section 6.2.1 through 6.2.4 shall be maintained in accordance with Quality Assurance Program, Section 18.

6.3 Temporary changes to procedures of Section 6.1 above may be made provided:

6.3.1 The intent of the original procedure is not altered.

6.3.2 The change is approved by two members of the plant supervisory staff, at least one of whom holds a Senior Reactor Operator's License.

6.3.3 The change is documented, reviewed by a Qualified Reviewer and approved by either the Manager - Fort Calhoun Station or the Department Head designated by Administrative Controls Standing Orders as the responsible Department Head for that procedure within 14 days of implementation.

6.4 Written procedures approved per Section 6.2 above shall be implemented which govern the selection of fuel assemblies to be placed in Region 2 of the spent fuel racks (Technical Specification 2.8). These procedures shall require an independent verification of initial enrichment requirements and fuel burn-up calculations for a fuel bundle to assure the "acceptance" criteria for placement in Region 2 are met. This independent verification shall be performed by individuals or groups other than those who performed the initial acceptance criteria assessment, but who may be from the same organization.

6.5 Written procedures shall be established and maintained for implementation of the Fire Protection Program.

6.6 Appropriate requirements have been established in the OPPD Quality Assurance Plan to assure that activities affecting quality are prescribed by documented instructions, procedures, or drawings; that they are accomplished in accordance with such documents, and are approved only when acceptance criteria are met. The responsibility for the development of the instructions, procedures, or drawings is delegated to the organization responsible for the activity; however, the developed instructions, procedures, and drawings are subject to OPPD QA audit. The Quality Assurance Plan contains the specific requirements pertaining to the instructions, procedures, and drawings associated with activities affecting quality.

6.6.1 The QA Plan requires that approved changes be promptly included where applicable into instructions, procedures, and drawings associated with the change. The OPPD QA Plan requires that changes be reviewed for their effect on present instructions, procedures, and/or drawings.

6.6.2 The OPPD QA Plan requires that procedures include a description of the sequence of activities or operation for fabrication, processing, assembly, inspection and test. Instructions indicate the operations or processes to be performed, type of characteristics to be measured or observed, the methods of examination, the applicable acceptance criteria and documentation requirements. The QA Plan also requires establishment of those inspections, tests, and holdpoints at which time conformance of parts, components, and subsystems to requirements will be verified.

6.6.3 OPPD personnel review such documentation to assure that it adequately reflects applicable quality requirements. In reviewing activities, OPPD personnel assure that instructions, procedures, and drawings contain appropriate quantitative (such as dimensions, tolerances, and samples) acceptance criteria for determining that important activities have been satisfactorily accomplished.

6.6.4 Inspections, tests, administrative controls, fire drills, and training that govern the fire protection program are prescribed by the QA Plan and established instructions, procedures, or drawings and are accomplished in accordance with these documents. Instructions and procedures for design, installation, inspection, test, maintenance, modification and administrative controls are reviewed in accordance with the established procedures to assure the proper inclusion of fire protection requirements.

7. DOCUMENT CONTROL

The OPPD QA Plan requires that documents, including changes, are reviewed for adequacy and approved for release by authorized personnel. These requirements also provide that contractors include, in their internal programs, measures to assure that changes to documents will be reviewed and approved by the same organization that performed the original review and approval. The OPPD QA Plan requires that changes to documents that have been reviewed and approved by OPPD organizations be reviewed and approved by the same OPPD organizations that performed the original review and approval. These requirements also provide that the documents are distributed to and used at the location where the prescribed activity is performed. The scope of these requirements apply to OPPD as well as to contractors and subcontractors.

The QA Plan requires a document control system that utilizes numbering of documents requiring control, predetermined distribution lists, and review and approval procedures. Controlled documents associated with Fort Calhoun Station have been controlled by document change transmittal letters instructing the recipient to remove and destroy obsolete or superseded pages. The QA Plan requires:

(1) maintenance of distribution lists

(2) use of receipt acknowledgments which indicate that superseded pages/documents are destroyed or marked as superseded

The Quality Assurance Plan requires that design engineering and procurement documentation, except for fire protection equipment, which consists of specifications, drawings, USAR material, instruction, procedures, reports, and changes thereto, and manufacturing and construction documents and records required for traceability, evidence of quality, and substantiation of the as-built configuration, be controlled.

Instructions, procedures, specifications, drawings, and procurement documents are controlled in accordance with the QA Plan and established division/department procedures.

A "Table of Contents" or "Index" system is used by OPPD departments to identify the current revision number of instructions, procedures, and procurement documents. The controlled copies are distributed to predetermined, responsible personnel, and a distribution list is maintained. Superseded documents are returned to the originator or destroyed as directed in the transmittal letter.

The QA Plan and established procedures identify those individuals or groups responsible for reviewing, approving, and issuing documents and revisions thereto.

8. <u>CONTROL OF PURCHASED MATERIAL, EQUIPMENT, AND SERVICES</u>

Measures utilized by OPPD to control purchased material, equipment, and services for an operating plant consist of individual and committee reviews, audits, and inspections. These measures are described in the OPPD Quality Assurance Plan and the established procurement procedures.

Potential manufacturers or contractors who are to be considered by OPPD or its prime contractors for the supply of items will normally be evaluated in advance of their use as an OPPD vendor. OPPD's evaluation of potential vendors is performed in accordance with established procedures. The evaluation involves the review of available historical data on manufacturers' or contractors' performance and capability; review of their quality assurance programs; or results of previous shop surveys and audits. Quality assurance program documents are required to be submitted with bids for CQE listed items. The manufacturer or contractor selected to supply the material, equipment, or services is approved by the Manager-Nuclear Procurement Services. If required, a pre-award survey at the supplier's facility is conducted before award of contract.

Documented, objective evidence such as certifications, chemical and physical analyses, inspection reports, test results, personnel and process qualification results, code stampings and nondestructive test reports are required to be evaluated by OPPD and suppliers or contractors. This verification will assure conformance to design requirements, drawings, specifications, codes, standards, regulatory requirements and other applicable criteria. These documents become a part of the quality verification records to be retained as a QA record in accordance with Section A.18.

Source inspection, when deemed necessary, is required by the applicable procurement document. The purchasing organization shall require that holdpoints be determined as necessary for this activity. Manufacturers are required to give sufficient notice of approaching holdpoints to allow scheduling of personnel.

Both in-process and final source inspections cover review of the quality verification documentation. An inspection document is used to establish the inspection sequence and for recording inspection results. This document also becomes part of the quality verification records. Provision is made for reporting deviations and nonconformances, if any; for recommending disposition and corrective action; for reinspection, if required; and for release for shipment, if appropriate. OPPD or its contractor may elect to participate in selected source inspections.

The OPPD QA Plan requires that manufacturers or contractors provide the quality verification documentation at the plant prior to the scheduled time of installation or use of the subject material and equipment. Audits will assure that the contractor is implementing a records management system. Delivered components will not be used until objective evidence of the quality verification package has been received unless there is a documented waiver.

Receiving inspection of purchased products is accomplished in accordance with established procedures. These procedures require that shipments delivered to the station be checked for shipping damage, agreement of actual count with the purchase order and packing slip, and agreement of the individual item identification with the purchase order and packing slip.

Procedures require that receiving inspection records be prepared for each purchase order requiring delivery of CQE, Limited CQE, fire protection and radioactive material packaging items to the station.

Receiving inspection records include a copy of the purchase order and material inspection records. Special instructions may be included for complex inspection requirements and tests to be performed at the plant or the supplier's work site as determined from the purchase order. Drawings and/or specification documents are included as appropriate. The inspector(s) perform(s) the receiving inspection in accordance with the above instructions and/or specifications.

The QA Plan requires that inspection records or certificates of conformance attesting to the quality of materials and equipment be submitted to OPPD for permanent retention. Such records are available for review during audits and are forwarded prior to or concurrent with material or equipment shipments to which they are related. In addition, prior to acceptance of material, the Nuclear Procurement Services Department is notified to verify that necessary documentation has been received.

Products intended for use as CQE/Limited CQE are inspected upon receipt in accordance with established procedures, which require a "nonconforming material" tag to be affixed to rejected material, and the material segregated in the receiving area to prevent inadvertent use. Accepted material is identified, and there are records traceable to the material indicating acceptance.

9. IDENTIFICATION AND CONTROL OF MATERIALS, PARTS, AND COMPONENTS

Appropriate requirements have been established by the OPPD Quality Assurance Plan to assure accurate identification traceability and control of materials, parts, and components so that the use of incorrect or defective material, parts, or components is prevented.

Material received at the storeroom for use as CQE/Limited CQE or as packaging for radioactive material for transport are identified to prevent the use of incorrect or defective material. The identification of the item is maintained by an appropriate code, letter, or number so that the identity of the material is maintained. Items shipped to the plant are normally identified by nameplate or other identification marking on the item. In those instances when it is not practical to provide identification markings on the individual items, identification information is provided in shipping paperwork that is transmitted with each shipment.

The traceability of materials is assured through the use of established procedures. The receiving inspection records contain the documentation needed for the traceability of the item. Those documents which are not included are referenced as to their location. The method of identification to be applied to purchased materials is specified as part of the purchase document. Codes and standards referenced in the purchase document have incorporated the appropriate marking method, such that the fit, function, or quality of the item is not affected. The correct identification of materials is verified and documented prior to release.

Contractors are required to utilize procedures which establish and document a system or method of identifying the material (e.g., physical marking, tagging, labeling, color code). This system clearly indicates whether materials are acceptable or unacceptable for further use, as required by the quality program. Material traceability is provided as specifically required by applicable codes; otherwise, material identification, either on the item or on records traceable to the item, are used, as appropriate. Where identification marking of an item is employed, the marking will be clear, understandable, and legible, and applied in such a manner as not to affect the function of the item. The identification and control measures provide for relating the item of production (batch, lot, components, part) at any stage, from materials receipt through fabrication, shipment, and installation to an applicable drawing, specification, or other technical document.

OPPD requires its suppliers to establish and implement a program for inspecting, marking, identifying and documenting material prior to use or storage. This program must be documented. Holdpoints are required where inspections must be made and certified complete before start of the next operation. Inspection of materials include the following:

(1) Verification that identification and markings are in accordance with applicable codes, standards, specifications, drawings, and purchase orders.

(2) Visual examination of materials and components for physical damage or contamination.

(3) Examination of quality verification records to assure that the material received was manufactured, tested and inspected prior to shipment in accordance with applicable requirements.

(4) Actual inspection, as required, of workmanship, configuration and other characteristics.

These inspections are documented and controlled. OPPD performs surveillance of vendor facilities as necessary to assure implementation of the program.

OPPD requires that contractors establish specific measures to assure compliance with approved procedures for identification and control of materials, parts, and components, including coatings and partially fabricated assemblies. OPPD verifies conformance by one or more of the following methods:

(1) Review and approval of contractors' quality assurance programs and procedures.

(2) Surveillance of selected manufacturing, fabrication, construction and installation activities by quality assurance personnel.

(3) Auditing:

  (a) of contractors for satisfactory performance of committed quality actions; and

  (b) of OPPD activities for adherence to quality requirements.

10. <u>CONTROL OF SPECIAL PROCESSES</u>

The QA Plan requires that written procedures and controls be prepared to assure that special processes, including welding, heat treating, protective coatings, and nondestructive testing are accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements. These procedures describe the operations to be performed, the sequence of operations, the characteristics involved (e.g., flow temperature, fitup, finish, hardness, and dimensions), the limits of these characteristics, process controls, measuring and testing equipment to be utilized, and documentation requirements.

Examination, tests, and inspections are conducted to verify conformance to the specified requirements.

Written procedures are required to cover training, examination, qualification, certification, and verification of personnel as well as the maintenance of required personnel records.

Compliance with these procedures is required for plant maintenance personnel, contractors, and vendors. Procedures for control of special processes are subject to review and approval by OPPD on an individual basis.

OPPD assures conformance with these requirements by:

(1) Review of procedures by the plant and QA & QC Department personnel for inclusion of special processes requiring control; definition of requirements for training, qualification and certification; conformance to applicable codes, standards, drawings, specifications, or other criteria.

(2) Audits to verify the adequacy of selected plant and vendor shop activities and the effectiveness of the special process procedures being implemented.

11. <u>INSPECTION</u>

OPPD will establish with its personnel and contractors a division of responsibility which will determine the services, structures, systems, components, and materials for which each is responsible. The organization having the responsibility for maintenance or repair of such items is also responsible to assure that adequate inspection is accomplished. OPPD's QA & QC Department, however, retains the responsibility and authority for review, approval, and surveillance or audit of the inspection procedures utilized by plant personnel or contractors.

OPPD QA & QC Department personnel are responsible for the inspection of work performed by OPPD maintenance personnel on nuclear safety related structures, systems, or components, and on radioactive material packaging. Activities affecting fire protection will be inspected by Quality Control personnel or other personnel who are independent of the individuals performing the activity being inspected to verify conformance with documented installation drawings and test procedures for accomplishing the activities. Inspection personnel will be knowledgeable in the design and installation requirements for fire protection to the extent necessary to perform the inspection.

The review and approval of a contractor's inspection program and procedures is accomplished as an integral part of OPPD's review of the organization's Quality Assurance/Quality Control programs. The QA & QC Department uses the following criteria in evaluating inspection methods proposed by plant OPPD personnel or organizations under contract to OPPD:

(1) Inspection procedures for functional groups such as procurement, project engineering, construction, and shop inspectors, must be described including measures to identify inspection and test status.

(2) Duties and responsibilities of personnel performing quality activities must be clearly established.

(3) Qualifications of personnel performing quality activities must be commensurate with their duties and responsibilities.

(4) Documentation methods for inspection activities of each group must be established (e.g., inspection forms, reports).

(5) Documentation control systems for identification and distributing inspection documents must be defined.

(6) Review and approval procedures for inspection documentation must be provided.

(7) Surveillance methods must be established to assure proper implementation of inspection procedures.

(8) Planning of inspection sequence activities by plant maintenance personnel or the contractors includes the type of characteristics to be measured, the methods of examination, and the criteria. OPPD will approve inspection holdpoints in the sequence.

The Manager-Fort Calhoun Station assures that the periodic inspections made by OPPD Personnel include:

(1) Periodic inspections of fire protection systems, breathing equipment and emergency lighting to assure the acceptable conditions of these items.

(2) Periodic inspections of materials subject to degradation such as fire stops, seals, and fire retardant coatings to assure that such items have not been damaged or deteriorated.

Inspection planning is utilized to assure conformance to procedures, drawings, specifications, codes, standards, and other documented instructions. Inspections are not to be performed by those individuals who performed the activity being inspected. Sufficient inspections are conducted to verify conformance particularly in areas rendered inaccessible by further processing. Process monitoring may be utilized in lieu of inspection in those cases where inspection is impossible, disadvantageous, or destructive. When required for adequate control, a combination of inspection and process monitoring is employed. Holdpoints verify (by review of inspection reports, visits to supplier shops, and plant surveillance) that inspections are being performed and documented by personnel in conformance with approved procedures.

The provisions which assure inspection is performed with the necessary drawings and specifications are covered in established procedures.

Modifications are inspected in accordance with established procedures. A plan for inspection and monitoring is developed and incorporated in planning documents of work segments, including designation of mandatory holdpoints. The inspection and monitoring plan is designed to verify conformance of work and products with the planning documents, applicable design documents, and specific quality standards and requirements. The plan provides for inspection and monitoring during critical stages in the progression of work and for inspection at the conclusion of each work segment.

Repairs and replacements are inspected in process or during receiving inspection in accordance with established procedures. Holdpoints for inspection or witnessing are specified in accordance with the plant Standing Orders.

OPPD inspectors are qualified and maintain their qualification by participation in the training and indoctrination delineated in Section A.3.4. OPPD QA & QC personnel performing nondestructive examination are trained and qualified in accordance with established procedures. Consultant and contractor inspectors performing inspection duties for OPPD are required to provide documentary evidence that they are qualified and that the certifications are current.

12. TEST CONTROL

The OPPD Quality Assurance Plan requires that OPPD personnel, contractors, and suppliers designate appropriate tests to be performed at specific stages of manufacturing, fabrication, construction, and operation. Conduct of tests are governed by written procedures which incorporate requirements and acceptance limits to assure that the structures, systems, and components tested will perform satisfactorily in service. Tests are conducted in accordance with these procedures and are properly documented.

OPPD assures that necessary tests are conducted by contractors performing maintenance or repair service for an operating plant. Such testing is performed in accordance with quality assurance and engineering test limits contained in applicable design documents. Test requirements and acceptance criteria are provided by the organization responsible for the specification of the item under test, unless otherwise designated. The entire test program covers required testing including, as appropriate, performance testing of production equipment, calibration testing of instruments, hydrostatic testing of pressure boundary components and surveillance testing.

Measures are established which assure that modifications, repairs, and replacements are tested in accordance with the original design and testing requirements or acceptance alternatives. Documentation of tests conducted is included in the completed design package, with the completed maintenance order by special procedure, or included in the receiving inspection packet.

Following modification, repair, or replacement, sufficient testing is performed to demonstrate that fire protection equipment in support of nuclear safety related equipment areas will perform satisfactorily in service and that design criteria are met. Written test procedures for installation tests are prepared by the responsible engineering group and incorporate the requirements and acceptance limits contained in applicable design documents.

Test procedures are evaluated for the following criteria and includes them where applicable:

(1) Requirements that prerequisites for the test have been met. Test prerequisites may include, but are not limited to, the following:

(a) calibrated instrumentation

(b) adequate and appropriate equipment

(c) trained, qualified and, as appropriate, licensed or certified personnel

(d) preparation, condition, and completeness of item to be tested

(e) suitable and, if required, controlled environmental conditions

(f) mandatory inspection holdpoints, where applicable, for witness by OPPD, contractor, or authorized inspector

(g) provisions for data collection and storage

(h) acceptance and rejection criteria

(i) methods of documenting or recording test data results

(2) Designation of specific test methods to adequately assess appropriate parameters.

(3) Designation of measuring and test equipment to be used.

(4) Specific environmental considerations.

(5) Measures to prevent damage to the item or system under test.

(6) Safety considerations.

(7) Documentation requirements.

Test results are evaluated to verify as applicable:

(1) Proper functioning of the system, structure, or component.

(2) Conformance to design specifications.

(3) Compliance with stated test requirements.

(4) That test results are within allowable limits.

(5) That recording and documentation is complete and accurate.

Audits by OPPD QA, vendor surveillance, and witness of specific tests serve to assure the functional adequacy of, and verify compliance with, the testing program.

13. CONTROL OF MEASURING AND TEST EQUIPMENT

The OPPD Quality Assurance Plan requires that organizations performing activities affecting quality involving measuring and test equipment have written procedures to govern these actions. The QA Plan requires that the standards used for calibration and accuracy verification of measuring and test equipment be traceable to the National Institute of Standards and Technology or other appropriate sources. In addition, only properly calibrated measuring and test equipment is used. A calibration program is established to which the tools, instruments, gauges, and other devices shall conform. Records of calibrations are maintained and the calibration equipment appropriately marked to indicate the date and acceptance of the calibration. Calibration activities being performed by OPPD personnel are in accordance with Standing Orders or other procedures. If a standards error exceeds the guaranteed accuracy, then the standard is replaced. Calibration standards are procedurally controlled to guarantee accuracy ratios consistent with industry standards.

When inspection and testing equipment is found to be out of calibration due to use or damage, or when out of limits at recalibration, items inspected, tested, or measured with that equipment since the latest valid calibration are considered as being potentially unacceptable. Resolution of these cases is determined on a case basis.

14. HANDLING, STORAGE AND SHIPPING

OPPD's QA Plan requires that instructions or guidance for plant handling, preservation, storage, and control of products are prepared and approved prior to arrival of the products at the plant. These procedures specify, as required, that special environmental facilities, such as inert gas, humidity control, or temperature controlled storage area are established prior to the receipt of the products. Contractors performing maintenance or repair services are required to provide procedures for the handling of products to prevent damage or deterioration. The procedures are reviewed and approved by OPPD.

To assure existence of the requirements for procedures in the procurement documents, OPPD verifies the inclusion during its review prior to authorization for document issuance. OPPD personnel procure, receive, store and handle CQE/Limited CQE and radioactive material packaging products, material, and components in accordance with established procedures.

15. INSPECTION, TEST AND OPERATING STATUS

OPPD's QA Plan requires that procedures be established to identify the inspection, test, and operating status of radioactive material packaging and safety-related structures, systems, and components. Identification of the inspection, test, and operating status of structures, systems, and components is provided in the surveillance test program. Schedules and methods for periodic testing of fire protection systems and components have been developed and documented by the Manager-Fort Calhoun Station. Fire protection equipment in support of nuclear safety related equipment areas is tested periodically to assure that the equipment will properly function and continue to meet the design criteria. Test results are documented, evaluated, and reviewed for acceptability.

The application and removal of inspection and welding stamps and status indicators are procedurally controlled, and nonconforming, inoperative, or malfunctioning structures, systems, or components are identified in accordance with established procedures. System completeness and acceptance at the end of a maintenance or repair phase are determined by:

(1) reviewing for adequacy, completeness, and conformance to quality assurance requirements for each system or component being accepted;

(2) performing surveillance and monitoring of the test activities associated with the approved test program;

(3) reviewing the test records to verify that test results comply with established requirements.

The suppliers' and contractors' inspection and test status of items are required to be maintained through the use of status indicators such as physical location, tags, markings, shop travelers, stamps, or inspection records. These measures provide for assuring that only items that have received the required inspections and tests are used in manufacturing and are released for shipment. The procedures for control of status indicators, including the authority for application and removal of tags, markings, labels or stamps are documented in approved manufacturing or quality assurance procedures.

16. NONCONFORMING MATERIALS, PARTS, OR COMPONENTS

The OPPD Quality Assurance Plan requires that measures be established to control the identification, documentation, segregation, and disposition of nonconforming material, parts, or components. The implementing instructions which fulfill these requirements are prescribed in established procedures. The QA Plan identifies those individuals or groups delegated the responsibility and authority for the disposition and approval of nonconforming items. Nonconforming items are controlled and identified in accordance with written procedures to prevent inadvertent use or installation. Control measures include tagging or marking and segregation when feasible. Control measures are maintained until the item has been removed from the plant site or corrective work has been completed and accepted.

Established procedures cover:

(1) Initiation of the documentation for material rejected at receiving inspection or in-plant activities including the tagging of nonconforming items. The documentation identifies the nonconforming item and describes the nonconformance.

(2) Assignment of disposition and/or corrective action responsibilities, including the inspection requirements and signature approval of the disposition.

(3) Control of correction work planning and acceptance.

Nonconformance reports are analyzed to detect adverse quality trends. Trending of nonconformances is conducted in accordance with procedures.

The OPPD Quality Assurance Plan requires that measures be taken and documented by contractors and suppliers to control the identification, documentation, segregation, and disposition of nonconforming material, parts, or components. These measures prevent inadvertent use or installation of defective components and are subject to review and approval by OPPD. Written procedures will be required for investigation of the nonconforming item, decisions on its disposition, and preparation of adequate reports. Procedures also control further processing, fabrication, delivery, or installation of items for which disposition is pending. Reports documenting actions taken on nonconforming items are made available to OPPD for evaluation. Departures from design specifications and drawing requirements that are dispositioned "use as-is" and "repair" are reported to affected organizations and OPPD management.

The effectiveness of nonconformance control procedures is assured by:

(1) Contractor quality assurance and manufacturing, fabrication, or construction personnel being involved in processing nonconforming reports.

(2) OPPD participation in dispositions and approvals.

(3) Document review at final inspection or shipping release and at receiving inspection by OPPD.

(4) Audits and/or Surveillances by OPPD and contract personnel.

17. CORRECTIVE ACTION

The QA Plan requires that measures be established to assure that conditions adverse to quality are promptly identified, reported, and corrected. Responsibility for performing corrective action is assigned to OPPD personnel and contractors and suppliers so that each will be alert to those conditions adverse to quality within his own area of responsibility. In the case of significant conditions adverse to quality, measures are taken to assure that the cause of the condition is determined and corrective action is implemented to preclude repetition. Corrective action procedures require thorough investigation and documentation of significant conditions adverse to quality. The cause and corrective action are reported in writing to the appropriate levels of management. The corrective action to be applied is subject to review and approval by the PRC. Corrective action followup and closeout procedures provide that corrective action commitments are implemented in a systematic and timely manner and are effective.

The effectiveness of the suppliers' or contractors' corrective action program is assessed during audits by the supplier, the contractor, and by OPPD. Stop work authority is exercised as required.

A quarterly report of internal deficiencies that have occurred is prepared and distributed to the management of OPPD organizations participating in the QA program.

18. QUALITY ASSURANCE RECORDS

18.1 The following records shall be retained for at least five years:

18.1.1 Records, and logs of facility operation covering time interval at each power level.

18.1.2 Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

18.1.3 Licensee Event Reports (LER).

18.1.4 Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

18.1.5 Records of reactor tests and experiments.

18.1.6 Records of changes made to Operating Procedures.

18.1.7 Records of annual physical inventory of all source material of record.

18.2 The following records shall be retained for the duration of the Facility Operating License:

18.2.1 Records of drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.

18.2.2 Records of new and irradiated fuel inventory, fuel transfers and assembly burn-up histories.

18.2.3 Records of facility radiation and contamination surveys.

18.2.4 Records of radiation exposure for all individuals entering radiation control areas.

18.2.5 Records of gaseous and liquid radioactive material released to the environs.

18.2.6 Records of transient or operational cycles for those facility components designed for a limited number of transients or cycles.

18.2.7 Records of training and qualification for current members of the plant staff.

18.2.8 Records of in-service inspections performed pursuant to these Technical Specifications.

18.2.9 Records of Quality Assurance activities required by the QA Manual.

18.2.10 Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

18.2.11 Records of meetings of the Plant Review Committee and the Safety Audit and Review Committee.

18.2.12 Records of Environmental Qualification of Electrical Equipment pursuant to 10 CFR 50.49.

18.2.13 Records of the service lives of all hydraulic and mechanical snubbers, including the date at which the service life commences and associated installation and maintenance records.

18.2.14 Records of analyses required by the Radiological Environmental Monitoring Program.

18.2.15 Records of reviews performed for changes made to the Offsite Dose Calculation Manual and the Process Control Program.

18.2.16 Records of radioactive shipments.

18.3 A complete record of the analysis employed in the selection of any fuel assembly to be placed in Region 2 of the spent fuel racks will be retained as long as that assembly remains in Region 2 (reference Technical Specifications 2.8 and 4.4).

18.4 OPPD's Quality Assurance Plan requires that OPPD and its contractors have a quality records system which provides documentary evidence of the performance of activities affecting quality. The requirements include that:

18.4.1 Records are maintained that show evidence of performance of activities affecting quality. Typical records to be maintained include quality assurance programs and plans, design data and studies, design review reports, specification procurement documents, procedures, inspection and test reports, material certifications, personnel certifications, test reports, audit reports, reports of nonconformances and corrective actions, as-built drawings, operating logs, calibration history, maintenance data, and failure and incident reports.

18.4.2 Inspection and test records, as a minimum, identify the date of the inspection or test, the inspector or data recorder, the type of observation, the results, the acceptability, and the action taken in connection with any nonconformances noted.

18.4.3 Records are protected against deterioration and damage.

18.4.4 Criteria are established for determining the classification of the record as well as the length of the retention period.

18.4.5 A method of identification and indexing of records for ease of retrievability is established.

18.4.6 Responsibility for record keeping during design, fabrication, construction, preoperational testing, and commercial operation is documented.

18.4.7 Method of transfer of records between organizations and ultimate transfer to OPPD shall be established.

18.5 Requirements and responsibilities for the handling, storage, and retention of records which furnish documentary evidence of quality are prescribed by established procedures. The records are accumulated and handled in a controlled manner in accordance with these procedures.

18.6 Records may be maintained on optical disks, as approved by NRC Generic Letter 88-18. Records maintained on optical disks are subject to the specific quality controls provided in the Generic Letter.

19. AUDITS

The OPPD QA Plan requires that planned and periodic audits be performed to verify compliance of activities affecting quality and to determine the effectiveness of the QA program. OPPD QA and Nuclear Procurement Services personnel perform such performance based audits on OPPD internal activities, contractors, suppliers, and others as necessary to provide an objective evaluation of the effectiveness of their programs; to determine that their programs are in compliance with established requirements, methods, and procedures; and to verify implementation of recommended corrective action.

The internal audit cycle for activities affecting quality and the fire protection program is promulgated in establishing procedures and is based on the safety importance of the activities being performed. An audit schedule is distributed on a calendar year basis and is updated as necessary to ensure coverage of status changes. If, in the opinion of the Manager-Quality Assurance & Quality Control, a given area requires added emphasis, the frequency of audits is increased until the situation is clarified.

The OPPD audits, both internal and external, are conducted in accordance with the established procedures. Consultants may be utilized by OPPD on audits as required. The QA Plan specifies that the auditing system used by OPPD, its contractors, and suppliers:

(1) utilizes an audit planning document which defines the organizations and activities to be audited and the frequency of audits;

(2) requires auditors to be familiar with the type of activities to be audited and have no direct responsibilities in the type of activities to be audited;

(3) provides auditing checklists or other objective guidelines to identify those activities which affect quality;

(4) requires examination of the essential characteristics of the quality activity examined;

(5) requires an audit report to be prepared and that it notes the extent of examination and deficiencies found.

Established procedures provide the means which assure that audits are performed in a thorough and professional manner. OPPD audits determine the existence of a control system and the deficiencies of that system, and the actual practice of the system. Audit checklists are used to ensure that audits include the objective evaluation of work areas, activities, processes and items and the review of documents and records.

Established procedures require that a formal report be prepared upon completion of each audit. The audit report identifies any deficiencies or nonconformances found during the audit, and recommended solutions.

(PRC) The Plant Review Committee shall be as prescribed below:

(1) Function

The Plant Review Committee shall function to advise the Manager - Fort Calhoun Station on all matters related to nuclear safety in accordance with USAR Section 12.5 and Plant Standing Orders.

(2) Composition

The official Plant Review Committee shall consist of at least six but not more than eleven members and shall be composed of the:

Chairman: Manager - Fort Calhoun Station

Members: The members shall be Department Heads or supervisory staff representing operations, maintenance, engineering, chemistry, radiation protection and other technical disciplines as determined by the Chairman.

All members shall be qualified to the applicable requirements of Technical Specification 5.3 prior to being appointed by the Chairman.

(3) Alternates

Alternate members shall be appointed in writing by the Plant Review Committee Chairman to serve on a temporary basis.

(4) Meeting Frequency

The Plant Review Committee shall meet at least once per calendar month and as convened by the Plant Review Committee Chairman.

(5) Quorum

A quorum of the Plant Review Committee shall consist of the Chairman or Alternate Chairman and a majority of members including alternates. At any one time, no more that a minority of the quorum shall consist of alternate members participating as voting members in Plant Review Committee activities.

(6) Responsibilities

The Plant Review Committee shall be responsible for:

a. Review of (1) Administrative Controls Standing Orders and changes thereto, (2) procedures required by Quality Assurance Program, Section 6 and requiring a 10 CFR 50.59 safety evaluation, and (3) proposed changes to procedures required by Quality Assurance Program, Section 6 and requiring a 10 CFR 50.59 safety evaluation;

b. Review of all proposed tests and experiments that affect nuclear safety.

c. Review of all proposed changes to the Technical Specifications.

d. Review of all proposed changes to the Core Operating Limits Report.

e. Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety.

f. Investigation of all violations of the Technical Specifications and shall prepare and forward a report covering evaluation and recommendations to prevent recurrence to the Vice President and to the Chairperson of the Safety Audit and Review Committee.

g. Review of facility operations to detect potential safety hazards.

h. Performance of special reviews and investigations and reports thereon as requested by the Chairperson of the Safety Audit and Review Committee.

i. Review of the Fire Protection Program Plan and shall submit changes to the Chairperson of the Safety Audit and Review Committee.

j. Review of all Reportable Events.

(7) Authority

The Plant Review Committee shall:

a. Recommend in writing to the Manager - Fort Calhoun Station approval or disapproval of items considered under PRC (6), a through e above.

b. Render determinations in writing with regard to whether or not each item considered under PRC (6), b through f above constitutes an unreviewed safety question.

c. Provide immediate written notification to the Vice President and the Chairperson of the Safety Audit and Review Committee of disagreement between the Plant Review Committee and the Manager - Fort Calhoun Station; however, the Manager - Fort Calhoun Station shall have responsibility for resolution of such disagreements pursuant to Technical Specifications 5.1.1.

(8) Records

The Plant Review Committee shall maintain written minutes of each meeting and copies shall be provided to the Vice President and Chairperson of the Safety Audit and Review Committee.

(SARC) The Safety Audit and Review Committee shall be as prescribed below.

(1) Function

The Safety Audit and Review Committee shall function to provide the independent review and audit of designated activities in the areas of:

a. nuclear power plant operation

b. nuclear engineering

c. chemistry and radiochemistry

d. metallurgy

e. instrumentation and control

f. radiological safety

g. mechanical and electrical engineering

h. quality assurance

i. fire protection

(2) Composition

The Safety Audit and Review Committee shall be composed of:

Chairperson:	Member as appointed by the Vice President
Member:	Vice President
Member:	Division Manager - Nuclear Assessments
Member:	Division Manager - Engineering & Operations Support
Member:	Manager - Fort Calhoun Station
Member:	Other qualified OPPD personnel and/or consultants as required and as determined by the SARC Chairperson

(3) Alternates

Alternate members shall be appointed in writing by the Chairperson of the Safety Audit and Review Committee to serve on a temporary basis; however, no more than two alternates may participate in the Safety Audit and Review Committee activities at any one time.

(4) Consultants

Consultants shall be utilized as determined by the Safety Audit and Review Committee Chairperson to provide expert advice to the Safety Audit and Review Committee.

(5) Meeting Frequency

The Safety Audit and Review Committee shall meet at least once every six months.

(6) Quorum

A quorum of the Safety Audit and Review Committee shall consist of the Chairperson or his designated alternate and a majority of the Safety Audit and Review Committee members including alternates. No more than a minority of the quorum shall have line responsibility for the operation of the nuclear plant.

(7) Review

The Safety Audit and Review Committee shall review:

a. The safety evaluations for 1) procedures, equipment or systems and 2) tests or experiments completed under the provision of section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in section 50.59, 10 CFR.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59.

d. Proposed changes to Technical Specifications and Facility Operating License DPR-40.

e. Violations of applicable statutes, codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.

g. All Licensee Event Reports required by 10 CFR 50.73.

h. Any indication of unanticipated deficiency in some aspect of design or operation of safety related structures, systems or components.

i. Reports and meeting minutes of the Plant Review Committee.

The Chairperson of the Safety Audit and Review Committee (SARC) may designate subgroups, special working committees, or audit teams as he deems necessary in order to carry out the responsibilities of the SARC. These subgroups, committees, or audit teams will perform the SARC responsibilities and report on their activities for review at the next regularly scheduled SARC meeting following any group's action.

(8) Audit

Audits of facility activities shall be performed under the cognizance of the Safety Audit and Review Committee. These audits shall encompass:

a. The conformance of facility operation to provisions contained within the Technical Specifications and applicable license conditions.

b. The training and qualifications of the facility staff.

c. Actions taken to correct deficiencies occurring in facility equipment, structures, systems, components or method of operation that affect nuclear safety.

d. The performance of activities required by the Quality Assurance Program to meet the criteria of Appendix B, 10 CFR Part 50.

e. The Radiological Effluent Program including the Radiological Environmental Monitoring Program and the results thereof, Offsite Dose Calculation Manual and implementing procedures, and the Process Control Program for the solidification of radioactive waste.

f. The fire protection and loss prevention program utilizing either qualified off-site licensee personnel or an outside fire protection consultant.

g. Any other area of facility operation considered appropriate by the Safety Audit and Review Committee of the Vice President.

(9) Authority

The Safety Audit and Review Committee shall report to and advise the Vice President on those areas of responsibility specified in Sections SARC (7) and (8) above.

(10) Records

Records of Safety Audit and Review Committee activities shall be prepared, approved and distributed as indicated below:

a. Minutes of each Safety Audit and Review Committee meeting shall be prepared, approved and forwarded to the Vice President within 30 days following each meeting.

b. Reports of reviews encompassed by SARC (7) e through i above shall be prepared, approved and forwarded to the Vice President within 30 days following completion of the review.

c. Audit reports encompassed by SARC (8) above shall be forwarded to the Vice President and to the responsible management positions designated by the Safety Audit and Review Committee within 30 days after completion of the audit.

Figure A-1- "Nuclear Quality Assurance Program Organization for Fort Calhoun Station"

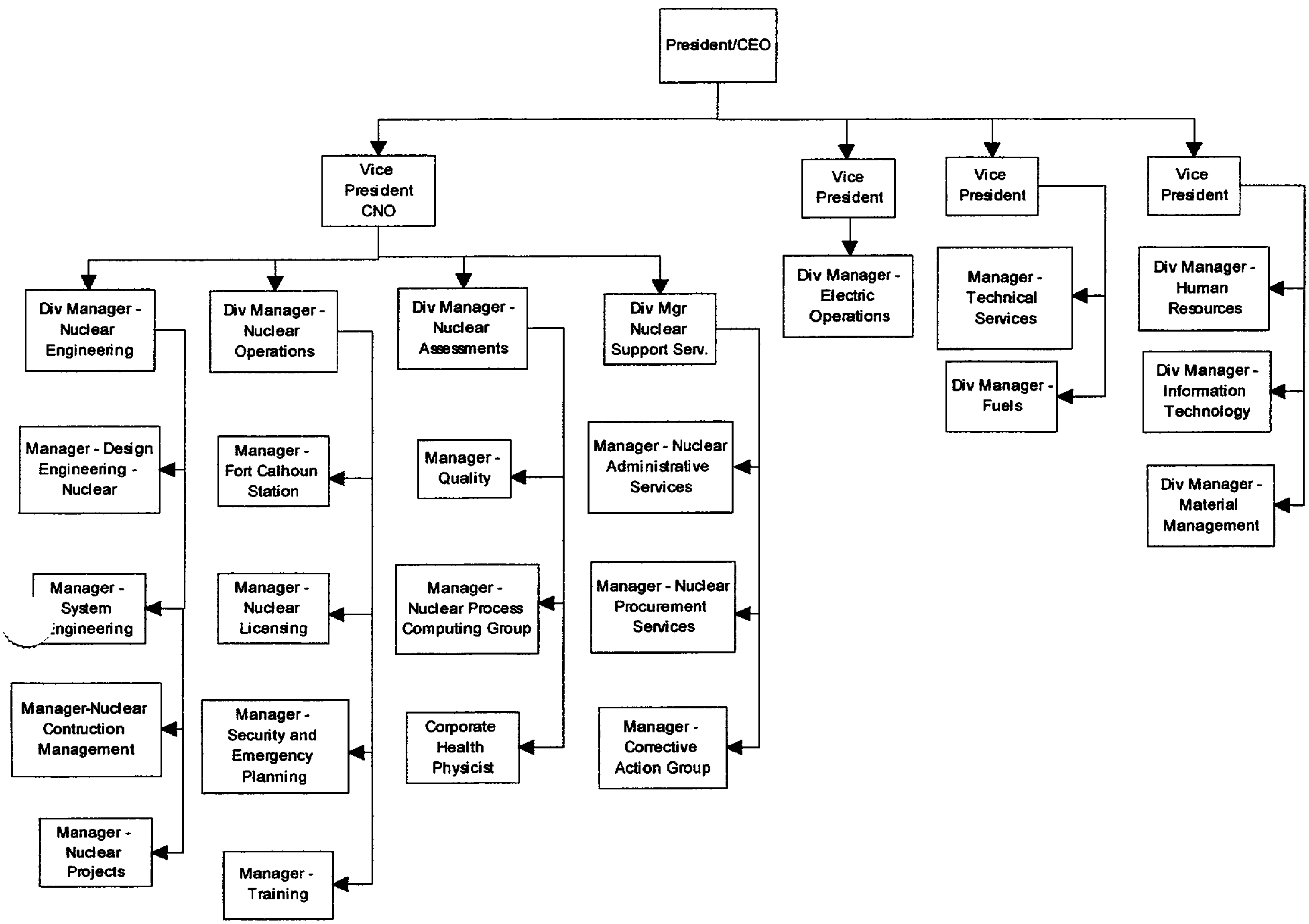
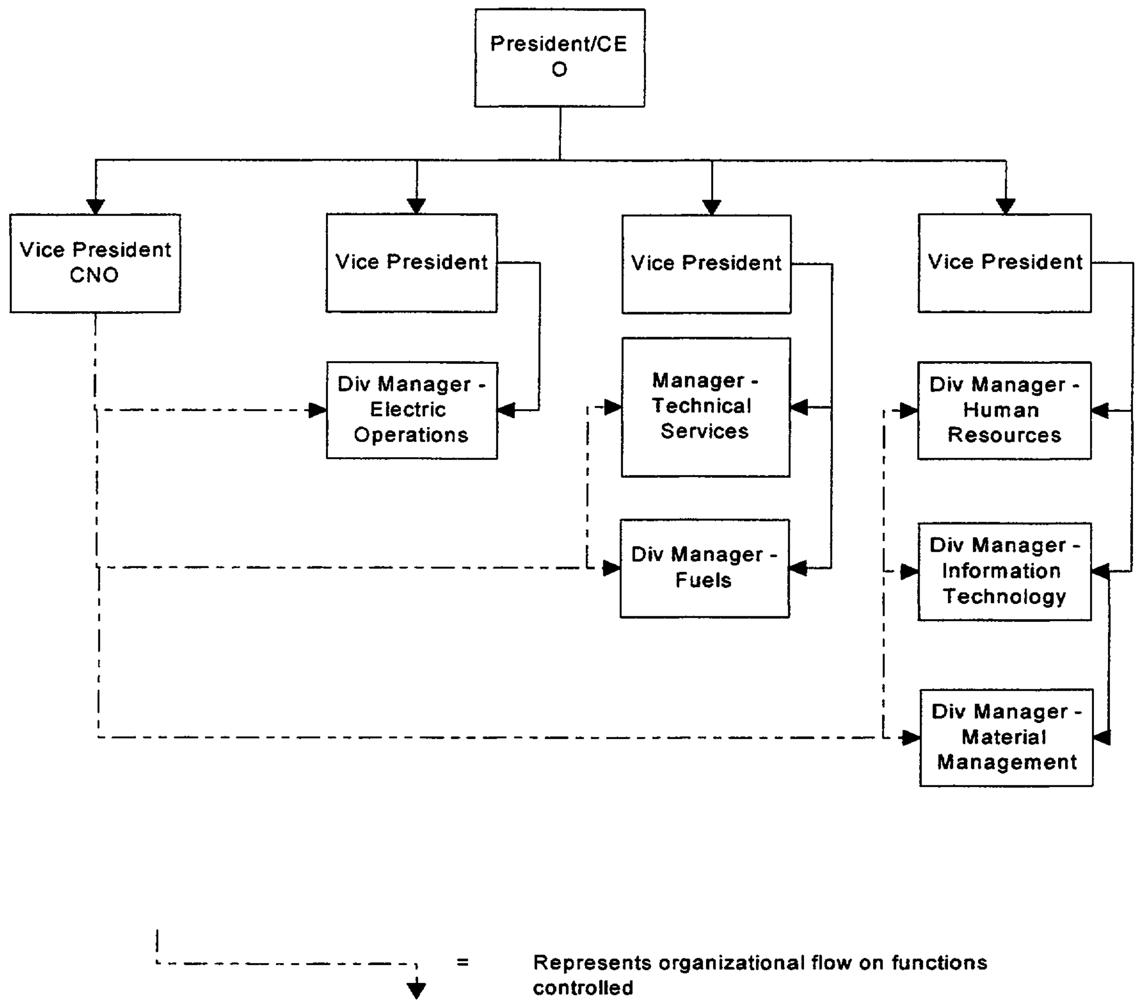


Figure A-2- "Nuclear Quality Assurance Program Organization for Non-Nuclear Divisions performing functions controlled by the Quality Assurance Program"



Attachment 1 - "Industry Standards and Associated Regulatory Guides Used as a Basis for the OPPD QA Program"

Introduction

This attachment identifies the specific industry standards and NRC Regulatory Guides which form the base of OPPD's QA Program as described in the USAR Appendix, and as delineated in the QA Plan. A position statement is provided for each standard/Regulatory Guide which describes the nature and extent of OPPD's commitments, including any alternatives used or exceptions taken. OPPD interprets the verbs (shall, should and may) used in industry standards to mean the following:

- Shall: Indicates a requirement.
- Should: Indicates a recommendation.
- May: Indicates permission or an option.

Where the below listed standards make reference to other documents to be included as a part of the referencing standard, it is assumed that the reference is to the specific standard(s) identified and described in this attachment, or as defined in other applicable District commitment documents. In the development of the QA Plan, requirements of construction based standards were incorporated to the extent that they are applicable to operations phase activities, retaining the basic quality assurance controls, but not necessarily encompassing the specific implementation associated with that control or measure delineated for the construction phase.

Commitments

A. Standard: ANSI N45.2-1977, "Quality Assurance Program Requirements for Nuclear Facilities"

Regulatory Guide: RG 1.28, Revision 2, "Quality Assurance Program Requirements (Design and Construction)"

Position: OPPD's QA Program and QA Plan comply with the requirements of this standard and Regulatory Guide to the extent that these requirements apply to activities affecting quality performed in the operations phase of Fort Calhoun Station.

B. Standard: ANSI N18.7-1976," Administrative Controls and Quality Assurance for the Operations Phase of Nuclear Power Plants"

Regulatory Guide: RG 1.33, Revision 2, "Quality Assurance Program Requirements (Operations)"

Positions: OPPD's QA Program and QA Plan comply with the applicable requirements of this standard and Regulatory Guide with the following alternatives or exceptions:

1. Preoperational testing of Fort Calhoun Station is completed, therefore, these requirements are implemented only to the extent required by the Station Technical Specifications and as applicable to preoperational testing associated with station modification activities and post-maintenance testing.

2. OPPD's audit program is a four-tiered program consisting of (1) regularly scheduled internal and external audits conducted on a 3 year cycle by the Quality Assurance and Quality Control and Nuclear Procurement Services Departments, (2) regularly scheduled QA surveillances conducted by the Quality Assurance & Quality Control Department, (3) scheduled audits performed by, or under the cognizance of, the Safety Audit and Review Committee in accordance with the Station Technical Specifications, and (4) a management review to determine the adequacy and effectiveness of OPPD's Quality Assurance Program performed under the auspices of the Safety Audit and Review Committee.

3. Audits performed under the cognizance of the Safety Audit and Review Committee are conducted at the following frequencies:

   a. The conformance of facility operation to provisions contained within the Technical Specifications and applicable license conditions at least once per two years.

   b. The training and qualifications of the facility staff at least once per two years.

   c. Actions taken to correct deficiencies occurring in facility equipment, structures, systems, components or method of operation that affect nuclear safety at least once per two years.

   d. The performance of activities required by the Quality Assurance Program to meet the criteria of Appendix B, 10 CFR Part 50, at least once per two years.

e. The Radiological Effluent Program including the Radiological Environmental Monitoring Program and the results thereof, the Offsite Dose Calculation Manual and implementing procedures, and the Process Control Program for the solidifications of radioactive waste at least once per two years.

f. An independent fire protection and loss prevention inspection and audit shall be performed annually utilizing either qualified off-site licensee personnel or an outside fire protection firm.

g. An inspection and audit of the fire protection and loss prevention program by an outside qualified fire consultant shall be performed at intervals no greater than three years.

4. OPPD uses a dynamic procedure review process instead of a static two year review cycle to prevent the use of outdated or inappropriate documents. This dynamic process ensures applicable procedures and instructions are reviewed for possible revision upon the identification of new or revised source material.

5. Written procedures and administrative policies affecting Fort Calhoun Station are also controlled by requirements contained in the Administrative Controls section of the FCS Technical Specifications.

C. Standard: ANSI N45.2.1-1973, "Cleaning of Fluid Systems and Associated Components During the Construction Phase of Nuclear Power Plants"

Regulatory Guide: RG 1.37, Revision 0, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water Cooled Nuclear Power Plants"

Position: OPPD's QA Program and QA Plan comply with the applicable requirements of this standard and Regulatory Guide to the extent that these requirements apply to activities affecting quality performed in the operations phase of the Fort Calhoun Station as delineated in OPPD's QA Plan.

D. Standard: ANSI N45.2.2-1972, "Packaging, Shipping, Receiving, Storage and Handling of Items for Nuclear Power Plants (During the Construction Phase)"

Regulatory Guide: RG 1.38, Revision 2, "Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage and Handling of Items for Water Cooled Nuclear Power Plants"

Position: OPPD's QA Program and QA Plan comply with the requirements of this standard and Regulatory Guide to the extent that these requirements apply to activities affecting quality performed in the operations phase of Fort Calhoun Station.

E. Standard: ANSI N45.2.3-1973, "Housekeeping During the Construction Phase of Nuclear Power Plants"

Regulatory Guide: RG 1.39, Revision 2, "Housekeeping Requirements for Water Cooled Nuclear Power Plants"

Position: OPPD's QA Program and QA Plan comply with the applicable requirements of this standard and Regulatory Guide to the extent that these requirements apply to activities affecting quality performed in the operation phase of the Fort Calhoun Station. The Fort Calhoun Station is divided into zones for security, storage, fire protection, and radiation protection and since zones designated by ANSI 45.2.3 are primarily designed for control of units during construction, additional zones have not been established for housekeeping purposes. Limitations are applied on eating, drinking, and smoking in specified areas at the Fort Calhoun Station. Housekeeping practices and controls have been established for the control of activities that can affect the quality of CQE related systems and components.

F. Standard: ANSI N45.2.4-1972, "Installation, Inspection and Testing Requirements for Instrumentation and Electric Equipment During the Construction of Nuclear Power Generating Stations"

Regulatory Guide: RG 1.30, Revision 0, "Quality Assurance Requirements for the Installation, Inspection and Testing of Instrumentation and Electrical Equipment"

Position: The applicable requirements of this standard and Regulatory Guide are implemented for modification activities which meet or exceed original plant specifications and manufacturer's recommendations, as described in the QA Plan.

G. Standard: ANSI N45.2.5-1974, "Supplementary Quality Assurance Requirements for Installation, Inspection and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants"

Regulatory Guide: RG 1.94, Revision 1, "Quality Assurance Requirements for Installation, Inspection and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants"

Position: The applicable requirements of this standard and Regulatory Guide are implemented for modification activities which meet or exceed original plant specifications and manufacturer's recommendations, as described in the QA Plan.

H. Standard: ANSI N45.2.6-1978, "Qualifications of Inspection Examination, and Testing Personnel for Nuclear Power Plants"

Regulatory Guide: RG 1.58, Revision 1, "Qualification of Nuclear Power Plant Inspection, Examination and Testing Personnel"

Position: OPPD's QA Program and QA Plan comply with the applicable requirements of this standard and Regulatory Guide with the following alternatives or expections:

1. OPPD and contractor inspectors performing Quality Control inspections are certified in accordance with this standard; however these certification requirements are not applied to personnel performing operational surveillance testing and inspection in accordance with the Technical Specifications, to investigative inspections or to the conduct of preliminary inspections for purpose of planning corrective or improvement actions, or to the surveillance of plant operations to verify compliance with procedures. Certification of inspectors for nondestructive examinations is accomplished in accordance with SNT-TC-1A guidelines.

I. Standard: ANSI N45.2.8-1975, "Supplementary Quality Assurance Requirements for Installation, Inspection and Testing of Mechanical Equipment and Systems for the Construction Phase of Nuclear Power Plants"

Regulatory Guide: RG 1.116, Revision 0, "Quality Assurance Requirements for Installation Inspection and Testing of Mechanical Equipment and Systems"

Position: The applicable requirements of this standard and Regulatory Guide are implemented for modification activities which meet or exceed original plant specifications and manufacturer's recommendations, as described in the QA Plan.

J. Standard: ANSI N45.2.9-1974, "Requirements for Collection, Storage and Maintenance of Quality Assurance Records for Nuclear Power Plants"

Regulatory Guide: RG 1.88, Revision 2, "Collection, Storage, and Maintenance of Nuclear Power Quality Assurance Records"

Position: OPPD's QA Program and QA Plan comply with the applicable requirements of this standard and Regulatory Guide with the following alternatives or exceptions:

1. The Fort Calhoun Station File Room meets the criteria of NUREG-0800, Standard Review Plan, Part 17.1, Acceptance Criteria 17.4, Alternative (3); ANSI N45.2.9-1979; NFPA 232; and will withstand a maximum wind velocity of 110 miles per hour.

2. Fire rated file cabinets used for interim record storage meet a one hour or greatest fire rating.

K. Standard: ANSI N45.2.10-1973, "Quality Assurance Terms and Definitions"

Regulatory Guide: RG 1.74, Revision 0, "Quality Assurance Terms and Definitions"

Position: OPPD's QA Program and QA Plan comply with the general terms and definitions of this standard.

L. Standard: ANSI N45.2.11-1974, "Quality Assurance Requirements for the Design of Nuclear Power Plants"

Regulatory Guide: RG 1.64, Revision 2, "Quality Assurance Requirements for the Design of Nuclear Power Plants"

Position: OPPD's QA Program and QA Plan comply with the requirements of this standard and Regulatory Guide to the extent that these requirements apply to activities affecting safety in the operations phase of Fort Calhoun Station. Additionally, in unique circumstances which occur in a detailed specialty field (such as reactor physics, seismic, stress analysis, etc.) where the only technically qualified individual within the licensee's organization available to perform analysis verification is the immediate supervisor, such review will be allowed when:

1. Justification allowing the review is documented and approved in advance by the Division Manager - Engineering and Operations Support and

2. All other Regulatory Guide 1.64, Revision 2, independence criteria are met.

M. Standard: ANSI N45.2.12-1977, "Requirements for Auditing of Quality Assurance Programs for Nuclear Power Plants"

Regulatory Guide: RG 1.144, Revision 1, "Auditing of Quality Assurance Programs for Nuclear Power Plants"

Position: OPPD's QA Program and QA Plan comply with the requirements of this standard and Regulatory Guide to the extent that these requirements apply to activities affecting safety performed in the operations phase of Fort Calhoun Station"

N. Standard: ANSI N45.2.13-1976, "Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants"

Regulatory Guide: RG 1.123, Revision 1, "Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants"

Position: OPPD's QA Program and QA Plan comply with the requirements of this standard and Regulatory Guide to the extent that these requirements apply to activities affecting safety performed in the operations phase of Fort Calhoun Station.

O. Standard: ANSI N45.2.23-1978, "Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants"

Regulatory Guide: RG 1.146, Revision 0, "Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants"

Position: OPPD's QA Program and QA Plan comply with the requirements of this standard and Regulatory Guide.

P. Standard: ANSI N18.1-1971, "Selection and Training of Nuclear Power Plant Personnel"

Regulatory Guide: RG 1.8, Revision 1, "Personnel Selection and Training"

Position: OPPD's QA Program and QA Plan comply with the applicable requirements of this standard and Regulatory Guide with the following alternatives or exceptions:

1. Qualification requirements for the Supervisor-Radiation Protection and for the Shift Technical Advisor are in accordance with the Fort Calhoun Station Technical Specifications.

2. Qualification requirements for the Manager-Operations, as described in this standard, shall be met by the FCS Supervisor-Operations.

Q. Standard: ANSI N101.4-1972, "Quality Assurance for Protective Coatings Applied to Nuclear Facilities"

Regulatory Guide: RG 1.54, Revision 0, "Quality Assurance Requirements for Protective Coatings Applied to Water Cooled Nuclear Power Plants"

Positions: The applicable requirements of this standard and Regulatory Guide are implemented for modification activities which meet or exceed original plant specifications and manufacturer's recommendations, as described in the QA Plan.