Omaha Public Power District
444 South 16th Street Mall
Omaha, Nebraska 68102-2247

February 7, 2001
LIC-01-0010

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station P1-137
Washington, D.C. 20555

References: See Attachment F

SUBJECT: Application for Amendment of Operating License

Pursuant to 10 CFR 50.90, 50.91, and 50.4, Omaha Public Power District (OPPD) is submitting this "Application for Amendment of Operating License" to revise the Fort Calhoun Station Unit No. 1 Technical Specifications. In addition, this submittal will revise the Fort Calhoun Station Unit No. 1 accident source term, pursuant to 10 CFR 50.67, used in the design basis radiological consequences analyses. Finally, this submittal will satisfy OPPD commitments provided to the NRC in references 5, 6, 7, 8, and 9.

OPPD proposes to amend the following sections of the Fort Calhoun Station Unit No. 1 Technical Specifications (TS): section 2.8 to add requirements to place the control room ventilation system in filtered air mode during refueling operations in the reactor containment building or at the spent fuel pool, place a spent fuel pool area radiation monitor in operation during refueling operations at the spent fuel pool, and delete the specification that requires a ventilation isolation actuation signal (VIAS) and two radiation monitors to be operable during refueling operations; sections 2.3 and 3.6 to increase the volume of trisodium phosphate (TSP) from 110 cubic feet to 126 cubic feet in the reactor containment building and replace the word undisturbed with representative; section 3.16 to include both internal and external leakage in the limit specified for the residual heat removal (RHR) system leakage, to add a requirement to perform an internal leakage test, and to correct the title. Finally, OPPD proposes to credit the alternative source term (AST) for the design basis site boundary and control room dose analyses.

Attachment C contains a mark-up reflecting the requested Technical Specification changes.
Attachment D provides the Discussion, Justification, and No Significant Hazards Consideration.
Attachment E provides the Site Boundary and Control Room Dose Analyses.
Attachment F provides References.

A001

OPPD respectfully requests 60 days to implement the proposed technical specifications following NRC approval. If you have additional questions, or require further information, please contact me or members of my staff.

Sincerely,

W. G. Gates

W. G. Gates
Vice President

WGG/dls

Attachments

c: E. W. Merschoff, NRC Regional Administrator, Region IV
L. R. Wharton, NRC Project Manager
W. C. Walker, NRC Senior Resident Inspector
B. E. Casari, Director - Environmental Health Division, State of Nebraska
Winston & Strawn

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
Omaha Public Power District	)	Docket No. 50-285
(Fort Calhoun Station	)	
Unit No. 1)	)	

APPLICATION FOR AMENDMENT
OF
FACILITY OPERATING LICENSE

Pursuant to Section 50.90 of the regulations of the U. S. Nuclear Regulatory Commission ("the Commission"), Omaha Public Power District, holder of Facility Operating License No. DPR-40, herewith requests that Technical Specifications set forth in Appendix A of the Facility Operating License be amended to: add requirements to place the control room ventilation system in the filtered air mode during refueling operations in the reactor containment building or at the spent fuel pool, place a spent fuel pool area radiation monitor in operation during refueling operations at the spent fuel pool, delete a specification that requires a ventilation isolation actuation signal (VIAS) and two radiation monitors to be operable during refueling operations, increase the volume of trisodium phosphate (TSP) in the reactor containment building, replace the word undisturbed with representative, modify a surveillance requirement to include both internal and external leakage for the residual heat removal (RHR) system leakage, modify a surveillance requirement to add an internal leakage test on the RHR system, correct a TS section title, and credit the alternative source term (AST) for the design basis site boundary and control room dose analyses.

The proposed changes to the Technical Specifications are provided in Attachment C of this Application. A Discussion, Justification, and No Significant Hazards Consideration, which demonstrates the proposed changes do not involve significant hazards, is appended in Attachment D. Attachment E provides the Site Boundary and Control Room Dose Analyses. Attachment F provides References. The proposed changes to Appendix A, Technical Specifications of the Facility Operating License, would not authorize any change in the types or any increase in the amounts of effluents or any change in the authorized power level of the facility.

WHEREFORE, Applicant respectfully requests that Appendix A of the Facility Operating License be amended hereto as Attachment C.

A copy of this Application, including its attachments, has been submitted to the Director - Nebraska State Division of Environmental Health, as required by 10 CFR 50.91.

OMAHA PUBLIC POWER DISTRICT

W. G. Gates

W. G. Gates
Vice President

STATE OF NEBRASKA)
) ss
COUNTY OF DOUGLAS)

Subscribed and sworn to me, a Notary Public in and for the State of Nebraska on this

7th day of February 2001

Theresa Peterson

Notary Public

GENERAL NOTARY-State of Nebraska
THERESA PETERSEN
My Comm. Exp. Feb. 17, 2001

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
Omaha Public Power District	)	Docket No. 50-285
(Fort Calhoun Station	)	
Unit No. 1)	)	

AFFIDAVIT

W. G. Gates, being duly sworn, hereby deposes and says that he is the Vice President in charge of all nuclear activities of the Omaha Public Power District; that he is duly authorized to sign and file with the Nuclear Regulatory Commission the attached information concerning the Application for Amendment of the Facility Operating License dated February 6, 2001, regarding the addition of requirements to place the control room ventilation system in the filtered air mode during refueling operations in the reactor containment building or at the spent fuel pool, placing a spent fuel pool area radiation monitor in operation during refueling operations at the spent fuel pool, deletion of a specification that requires a ventilation isolation actuation signal (VIAS) and two radiation monitors to be operable during refueling operations, increasing the volume of trisodium phosphate (TSP) in the reactor containment building, replacing the word undisturbed with representative, modification of a surveillance requirement to include both internal and external leakage for the residual heat removal (RHR) system leakage, modification of a surveillance requirement to add an internal leakage test on the RHR system, crediting of the alternative source term (AST) for the design basis site boundary and control room dose analyses, correcting a TS section title; that he is familiar with the content thereof; and that the matters set forth therein are true and correct to the best of his knowledge, information, and belief.

W. G. Gates

W. G. Gates
Vice President

STATE OF NEBRASKA)
) ss
COUNTY OF DOUGLAS)

Subscribed and sworn to me, a Notary Public in and for the State of Nebraska on this 7th day of February 2001

Theresa Petersen
Notary Public

GENERAL NOTARY-State of Nebraska
THERESA PETERSEN
My Comm. Exp. Feb. 17, 2001

LIC-01-0010
Attachment C
Requested Changes to Technical Specifications Set Forth in Appendix A of the Facility Operating License No. DPR-40

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.3 Emergency Core Cooling System (Continued)

(3) Protection Against Low Temperature Overpressurization

The following limiting conditions shall be applied during scheduled heatups and cooldowns. Disabling of the HPSI pumps need not be required if the RCS is vented through at least a 0.94 square inch or larger vent.

Whenever the reactor coolant system cold leg temperature is below 385°F, at least one (1) HPSI pump shall be disabled.

Whenever the reactor coolant system cold leg temperature is below 320°F, at least two (2) HPSI pumps shall be disabled.

Whenever the reactor coolant system cold leg temperature is below 270°F, all three (3) HPSI pumps shall be disabled.

In the event that no charging pumps are operable when the reactor coolant system cold leg temperature is below 270°F, a single HPSI pump may be made operable and utilized for boric acid injection to the core, with flow rate restricted to no greater than 120 gpm.

(4) Trisodium Phosphate (TSP) Dodecahydrate

During operating Modes 1 and 2, the TSP baskets shall contain ≥ ~~110~~ 126 ft^3 of active TSP.

a. With the above TSP requirements not within limits, the TSP shall be restored within 72 hours.

b. With Specification 2.3(4)a required action and completion time not met, the plant shall be in hot shutdown within the next 6 hours and cold shutdown within the following 36 hours.

Basis

The normal procedure for starting the reactor is to first heat the reactor coolant to near operating temperature by running the reactor coolant pumps. The reactor is then made critical. The energy stored in the reactor coolant during the approach to criticality is substantially equal to that during power operation and therefore all engineered safety features and auxiliary cooling systems are required to be fully operable.

2.0 LIMITING CONDITIONS FOR OPERATION
2.3 Emergency Core Cooling System (Continued)

With respect to the core cooling function, there is functional redundancy over most of the range of break sizes.[(3)(4)]

The LOCA analysis confirms adequate core cooling for the break spectrum up to and including the 32 inch double-ended break assuming the safety injection capability which most adversely affects accident consequences and are defined as follows. The entire contents of all four safety injection tanks are assumed to be available for emergency core cooling, but the contents of one of the tanks is assumed to be lost through the reactor coolant system. In addition, of the three high-pressure safety injection pumps and the two low-pressure safety injection pumps, for both large break analysis and small break analysis it is assumed that one high pressure pump and one low pressure pump operate [(5)]; and also that 25% of their combined discharge rate is lost from the reactor coolant system out of the break. The transient hot spot fuel clad temperatures for the break sizes considered are shown in USAR Section 14.

The restriction on HPSI pump operability at low temperatures, in combination with the PORV setpoints ensure that the reactor vessel pressure-temperature limits would not be exceeded in the case of an inadvertent actuation of the operable HPSI and charging pumps.

Removal of the reactor vessel head, one pressurizer safety valve, or one PORV provides sufficient expansion volume to limit any of the design basis pressure transients. Thus, no additional relief capacity is required.

Technical Specification 2.2(1) specifies that, when fuel is in the reactor, at least one flow path shall be provided for boric acid injection to the core. Should boric acid injection become necessary, and no charging pumps are operable, operation of a single HPSI pump would provide the required flow path. The HPSI pump flow rate must be restricted to that of three charging pumps in order to minimize the consequences of a mass addition transient while at low temperatures.

Trisodium Phosphate (TSP) dodecahydrate is required to adjust the pH of the recirculation water to ≥ 7.0 after a loss of coolant accident (LOCA). This pH value is necessary to prevent significant amounts of iodine, released from fuel failures and dissolved in the recirculation water, from converting to a volatile form and evolving into the containment atmosphere. Higher levels of airborne iodine in containment may increase the releases of radionuclides and the consequences of the accident. A pH of ≥ 7.0 is also necessary to prevent stress corrosion cracking (SCC) of austenitic stainless steel components in containment. SCC increases the probability of failure of components.

Radiation levels in containment following a LOCA may cause the generation of hydrochloric and nitric acids from radiolysis of cable insulation and sump water. TSP will neutralize these acids.

The required amount of TSP is represented in a volume quantity converted from the Reference 7 mass quantity using the manufactured density. Verification of this amount during surveillance testing utilizes the measured volume.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.8 Refueling
2.8.2 Refueling Operations - Containment

2.8.2(4) Control Room Ventilation System

Applicability

Applies to operation of the control room ventilation system during CORE ALTERATIONS and REFUELING OPERATIONS inside containment.

Objective

To minimize the consequences of a fuel handling accident to the control room staff.

Specification

The control room ventilation system shall be IN OPERATION and in the Filtered Air mode.

Required Actions

(1) If the control room ventilation system is not IN OPERATION or not in the Filtered Air mode, immediately suspend CORE ALTERATIONS and REFUELING OPERATIONS.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.8 Refueling
2.8.3 Refueling Operations - Spent Fuel Pool

2.8.3(5) ~~Ventilation Isolation Actuation Signal (VIAS)~~ Control Room Ventilation System

Applicability

~~Applies to operation of the Ventilation Isolation Actuation Signal (VIAS) during REFUELING OPERATIONS in the spent fuel pool.~~ Applies to operation of the control room ventilation system during REFUELING OPERATIONS in the spent fuel pool area. The provisions of Specification 2.0.1 for Limiting Conditions for Operation are not applicable.

Objective

~~To minimize the consequences of an accident occurring during REFUELING OPERATIONS in the spent fuel pool that could affect public health and safety.~~ To minimize the consequences of a fuel handling accident to the control room staff.

Specification

~~VIAS including manual actuation capability shall be OPERABLE with two gaseous radiation monitors on the auxiliary building exhaust stack OPERABLE, and supplied by independent power supplies.~~

(1) The control room ventilation system shall be IN OPERATION and in the Filtered Air mode.

(2) A spent fuel pool area radiation monitor shall be IN OPERATION.

Required Actions

(1) ~~With less than two gaseous radiation monitors on the auxiliary building exhaust stack OPERABLE or VIAS manual actuation capability inoperable, immediately suspend REFUELING OPERATIONS.~~ If the control room ventilation system is not IN OPERATION or not in Filtered Air mode, immediately suspend REFUELING OPERATIONS.

(2) If a spent fuel pool area radiation monitor is not IN OPERATION, immediately suspend REFUELING OPERATIONS.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.8 Refueling

Bases (Continued)

2.8.2(4) Control Room Ventilation System

Operating the control room ventilation system in the Filtered Air mode is a conservative measure to reduce control room operator exposure. This allows the radiological consequences analysis for a fuel handling accident to credit the Filtered Air mode at the time of the accident. When "immediately" is used as a completion time, the required action should be pursued without delay and in a controlled manner. Suspension of CORE ALTERATIONS and REFUELING OPERATIONS shall not preclude completion of movement of a component to a safe, conservative position.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.8 Refueling

Bases (Continued)

2.8.3(4) Spent Fuel Pool Area Ventilation (Continued)

The provisions of Specification 2.0.1 for Limiting Conditions for Operations are not applicable. If moving fuel assemblies while in MODES 4 or 5, LCO 2.0.1 would not specify any actions. If moving fuel assemblies in MODES 1, 2, or 3, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

2.8.3(5) ~~Ventilation Isolation Actuation Signal (VIAS)~~ Control Room Ventilation System

~~A Ventilation Isolation Actuation Signal (VIAS) is initiated by a Safety Injection Actuation Signal (SIAS), a Containment Spray Actuation Signal (CSAS) or a Containment Radiation High Signal (CRHS). During REFUELING OPERATIONS, only the CRHS is required to respond to a fuel handling or reactivity accident. The requirements of this specification are met when the Containment/Auxiliary Building Stack Swing Monitor (RM-052) and the Auxiliary Building Stack Radiation Monitor (RM-062) are OPERABLE, monitoring the Auxiliary Building exhaust stack, powered from independent 480-VAC buses and capable of actuating both the A and B trains of VIAS. When the RCS is below 300°F, the 480-VAC buses may be supplied by a single 4160-VAC power source. Above 300°F, Specification 2.7 requires both 4160-VAC buses to be operable. In addition, one manual actuation channel is required to be OPERABLE. (Note, the Offsite Dose Calculation Manual may have additional requirements/restrictions concerning operation of these monitors.)~~

~~In the event that one of the above radiation monitors becomes inoperable, or both are OPERABLE but RM-052 is not monitoring the exhaust stack, or VIAS manual actuation capability is inoperable, REFUELING OPERATIONS must be suspended thus precluding the possibility of a fuel handling accident. The doses calculated at the exclusion area boundary (EAB) and low population zone (LPZ) for a fuel handling accident in the spent fuel pool are well within 10 CFR 100 limits using conservative assumptions i.e., all rods in a single assembly fail with no credit taken for iodine filtration by VA-66.~~

~~VIAS aligns the control room air filtration system to the filtered air makeup mode, which prevents significant radionuclides from entering the control room. VIAS also initiates other actions, such as opening of the air supply and exhaust dampers in the safety injection pump rooms in preparation for safety injection pump operation. These other functions are not required to mitigate the consequences of a fuel handling accident, and therefore are not required to be OPERABLE.~~

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.8 Refueling

Bases (Continued)

2.8.3(5) ~~Ventilation Isolation Actuation Signal (VIAS) (continued)~~

~~When conducting REFUELING OPERATIONS in the spent fuel pool during MODES 1 and 2, LCO 2.15 is also applicable to VIAS. The allowable bypass condition for inoperable CRHS during MODES 1 and 2 is to close the containment pressure relief, air sample, and purge system valves. This is justified because a SIAS or CSAS will still initiate a VIAS. Since SIAS and CSAS would not initiate in response to a fuel handling accident, both the actions of this specification and 2.15 must be followed when the CRHS is inoperable in MODES 1 or 2 and REFUELING OPERATIONS are being conducted in the spent fuel pool.~~

Operating the control room ventilation system in the Filtered Air mode and requiring a radiation monitor to be IN OPERATION are conservative measures to reduce control room operator exposure. This allows the radiological consequences analysis for a fuel handling accident to credit the Filtered Air mode at the time of the accident.

Radiation monitoring will assure operators are alerted if a radiological incident occurs. This specification can be satisfied by using a permanent spent fuel pool area radiation monitor or a portable area radiation monitor.

~~When VIAS is inoperable REFUELING OPERATIONS in the spent fuel pool are immediately suspended. This effectively precludes a fuel handling accident from occurring.~~ When "immediately" is used as a completion time, the required action should be pursued without delay and in a controlled manner. Suspension of REFUELING OPERATIONS shall not preclude completion of movement of a component to a safe, conservative position.

References

(1) USAR Section 9.5
(2) USAR Section 14.18

3.0 **SURVEILLANCE REQUIREMENTS**

3.6 Safety Injection and Containment Cooling Systems Tests

Applicability

Applies to the safety injection system, the containment spray system, the containment cooling system and air filtration system inside the containment.

Objective

To verify that the subject systems will respond promptly and perform their intended functions, if required.

Specifications

(1) Safety Injection System

System tests shall be performed on a refueling frequency. A test safety feature actuation signal will be applied to initiate operation of the system. The safety injection and shutdown cooling system pump motors may be de-energized for this portion of the test.

A second overlapping test will be considered satisfactory if control board indication and visual observations indicate all components have received the safety feature actuation signal in the proper sequence and timing (i.e., the appropriate pump breakers shall have opened and closed, and all valves shall have completed their travel).

(2) Containment Spray System

a. System tests shall be performed on a refueling frequency. The test shall be performed with the isolation valves in the spray supply lines at the containment blocked closed. Operation of the system is initiated by tripping the normal actuation instrumentation.

b. At least every ten years the spray nozzles shall be verified to be open.

c. The test will be considered satisfactory if:

(i) Visual observations indicate that at least 264 nozzles per spray header have operated satisfactorily.

(ii) No more than one nozzle per spray header is missing.

d. ~~Undisturbed~~ Representative samples of Trisodium Phosphate Dodecahydrate (TSP) that have been exposed to the same environmental conditions as that in the mesh baskets shall be tested on a refueling frequency by:

3.0 **SURVEILLANCE REQUIREMENTS**

3.6 Safety Injection and Containment Cooling Systems Tests (continued)

(i) Verifying that the TSP baskets contain ≥~~110~~ 126 ft^3 of granular trisodium phosphate dodecahydrate.

(ii) Verifying that a sample from the TSP baskets provides adequate pH upward adjustment of the recirculation water.

3.0 **SURVEILLANCE REQUIREMENTS**
3.6 Safety Injection and Containment Cooling Systems Tests (Continued)

Operation of the system for 10 hours every month will demonstrate operability of the filters and adsorbers system and remove excessive moisture build-up on the adsorbers.

Demonstration of the automatic initiation capability will assure system availability.

Periodic determination of the volume of TSP in containment must be performed due to the possibility of leaking valves and components in the containment building that could cause dissolution of the TSP during normal operation. A refueling frequency shall be utilized to visually determine that ≥ ~~110~~ 126 ft^3 of TSP is contained in the TSP baskets. This requirement ensures that there is an adequate quantity of TSP to adjust the pH of the post-LOCA sump solution to a value ≥ 7.0.

The periodic verification is required on a refueling frequency. Operating experience has shown this surveillance frequency acceptable due to margin in the volume of TSP placed in the containment building.

Testing must be performed to ensure the solubility and buffering ability of the TSP after exposure to the containment environment. A representative sample of 1.80 - 1.83 grams of TSP from one of the baskets in containment is submerged in 0.99 - 1.01 liters of water at a boron concentration of 2445 - 2465 ppm. At a standard temperature of 115 -125°F, without agitation, the solution should be left to stand for 4 hours. The liquid is then decanted and mixed, the temperature adjusted to 75 - 79°F and the pH measured. At this point, the pH must be ≥ 7.0. The representative sample weight is based on the minimum required TSP weight of ~~5,788~~ 6672 lbs_m which, at a manufactured density of at least 53.0 lb_m/ft^3, corresponds to the minimum volume of ~~110~~ 126 ft^3, and maximum possible post-LOCA sump volume of 375,143 gallons, normalized to buffer a 1.0 liter sample. The boron concentration of the test water is representative of the maximum possible boron concentration corresponding to the maximum possible post-LOCA sump volume. The post-LOCA sump volume originates from the Reactor Coolant System (RCS), the Safety Injection Refueling Water Tank (SIRWT), the Safety Injection Tanks (SITs) and the Boric Acid Storage Tanks (BASTs). The maximum post-LOCA sump boron concentration is based on a cumulative boron concentration in the RCS, SIRWT, SITs and BASTs of 2445 ppm. Agitation of the test solution is prohibited~~,~~ since an adequate standard for the agitation intensity cannot be specified. The test time of 4 hours is necessary to allow time for the dissolved TSP to naturally diffuse through the sample solution. In the post-LOCA containment sump, rapid mixing would occur, significantly decreasing the actual amount of time before the required pH is achieved. This would ensure achieving a pH ≥ 7.0 by the onset of recirculation after a LOCA.

References

(1) USAR, Section 6.2
(2) USAR, Section 6.3
(3) USAR, Section 14.16
(4) USAR, Section 6.4

3.0 SURVEILLANCE REQUIREMENTS

3.16 ~~Recirculation~~ Residual Heat Removal System Integrity Testing

Applicability

Applies to determination of the integrity of the residual heat removal (RHR) system and associated components.

Objective

To verify that the leakage from the residual heat removal system components is within acceptable limits.

Specifications

(1) a. The portion of the shutdown cooling system that is outside the containment, and the piping between the containment spray pump suction and discharge isolation valves, shall be examined for leakage at a pressure no less than 250 psig. This shall be performed on a refueling frequency.

b. Piping from valves HCV-383-3 and HCV-383-4 to the suction isolation valves of the low pressure safety injection pumps and containment spray pumps and to the high pressure safety injection pumps shall be examined for leakage at a pressure no less that 82 psig. This shall be performed at the testing frequency specified in (1)a. above.

c. The portion of the high pressure safety injection (HPSI) system that is located outside the containment and downstream of the HPSI pumps shall be examined for leakage when subjected to the discharge pressure of a HPSI pump operating in the minimum recirculation mode. This test shall be performed at the frequency specified in (1)a. above. The leakage contribution from this section shall be the observed leakage from this piping at the test pressure multiplied by the square root of the ratio 1500/P, where P is the test discharge pressure (in psig) of the operating HPSI pump.

d. An internal leakage test shall be performed on a refueling frequency. The test shall measure and quantify the leakage to the safety injection refueling water tank (SIRWT) from applicable water leakage paths.

~~d.~~ e. Visual inspection of the system's components shall be performed at the frequency specified in (1)a. above to uncover any significant external leakage to atmosphere (including leakage from valve stems, flanges, and pump seals). The leakage shall be measured by collection and weighing or by any other equivalent method.

(2) a. The sum of leakages from section (1)a, (1)b, ~~and~~ (1)c, and (1)d above shall not exceed ~~1243~~ 3800 cc/hour.

b. Repairs shall be made as required to maintain leakage within the acceptable limits.

3.0 **SURVEILLANCE REQUIREMENTS**

3.16 Residual Heat Removal System Integrity Testing (Continued)

Basis

The limiting ~~external~~ leakage rate to atmosphere ~~rate~~ from the RHR system (~~1243~~ 3800 cc/hour) is based upon a plant specific leak rate analysis for RHR system components operating after a design basis accident.

The test pressures for sections 3.16(1)a and 3.16(1)b, and the pressure correction factors in sections 3.16(1)c give adequate margins over the highest pressures within the lines after a design basis accident.(1)

A RHR system leakage of ~~1243~~ 3800 cc/hr will limit off-site exposures due to leakage to insignificant levels relative to those calculated for direct leakage from the containment in the design basis accident. The safety injection system pump rooms are equipped with individual charcoal filters which are placed into operation by means of switches in the control room. The radiation detectors in the auxiliary building exhaust duct are used to detect high radiation level. The ~~1243~~ 3800 cc/hour leak rate is sufficiently high to allow for reasonable leakage through the pump seals ~~are~~ and valve packings, and yet small enough to be readily handled by the pumps and radioactive waste system. Leakage to the safety injection system pump room sumps will be returned to the spent regenerant tanks.(2) Additional makeup water to the containment sump inventory can be readily accommodated via the charging pumps from either the ~~SIRW tank~~ safety injection refueling water tank (SIRWT) or the concentrated boric acid storage tanks.

The analysis for the loss of coolant accident assumed a total (internal and external) leakage from all RHR systems sources of 3800 cc/hour. The internal leakage would leak back into the water remaining in the SIRWT.

References

(1) USAR, Section 9.3
(2) USAR, Section 6.2

LIC-01-0010

Attachment D

Discussion, Justification, and No Significant Hazards Consideration

DISCUSSION, JUSTIFICATION, AND NO SIGNIFICANT HAZARDS CONSIDERATION

I. DISCUSSION

As a holder of an operating license issued prior to January 10, 1997, and in accordance with 10 CFR 50.67, Fort Calhoun Station (FCS) is voluntarily replacing the accident source term used in all of its design basis site boundary and control room dose analyses by the Alternative Source Term (AST). The methodology/scenarios used in the existing design basis accident analyses, which are discussed in the FCS Updated Safety Analysis Report (USAR), are being updated to reflect the guidance provided in Regulatory Guide (RG) 1.183 (Reference 2). Some of the design basis accident analyses utilize pre-NUREG 0800 assumptions. The updated analyses reflect the results of a design basis verification and reconstitution effort that was initiated by Omaha Public Power District (OPPD) to support a total upgrade of the radiological accident analyses. Also included in this effort is the use of updated site boundary and control room atmospheric dispersion factors.

The site boundary and control room dose analyses for the following design basis accidents have undergone a change in design basis as discussed above:

1. Loss of Coolant Accident (LOCA)
2. Fuel Handling Accident in the Spent Fuel Pool (FHA in Spent Fuel Pool)
3. Fuel Handling Accident in the Reactor Containment Building (FHA in Containment)
4. Seized Reactor Coolant Pump Rotor Accident (SRA)
5. Control Rod Ejection Accident (CREA)
6. Main Steam Line Break (MSLB)
7. Steam Generator Tube Rupture (SGTR)
8. Gaseous Waste Decay Tank Failure (GWDTF)
9. Liquid Waste Tank Failure - Airborne releases (LWTF)

The Heavy Load Drop (HLD) Event, although not a FCS design basis accident, was also re-analyzed to maintain consistency with other accidents.

At FCS, the MSLB, SGTR, GWDTF and LWTF are not impacted by implementation of the AST, as there is no accident-initiated fuel damage associated with the events. However, to maintain consistency in design basis, these analyses have also been revised. In addition, the MSLB and SGTR analyses were revised to maintain consistency and incorporate related guidance provided in RG 1.183.

With this application, FCS proposes full implementation of the AST as defined in RG 1.183, Section 1.2.1 and permitted in 10 CFR 50.67, "Accident Source Term" (Reference 4).

The Site Boundary and Control Room Dose Analyses presented in Attachment E contains the analyses acceptance criteria, background information on computer codes, inventory source terms, dispersion factors, dose calculation methodology, accident re-analyses, summary results and conclusions.

Further, the following clarifications are provided to address source term implementation considerations of RG 1.183 that are not explicitly stated in Attachment E:

1. Impact Upon Equipment Environmental Qualification

   In RG 1.183 (Section C, paragraph 1.3.5), the following is stated: "*The NRC staff is assessing the effect of increased cesium releases on EQ doses to determine whether licensee action is warranted. Until such time as this generic issue is resolved, licensees may use either the AST or the TID 14844 assumptions for performing the required EQ analysis.*" Consistent with this guidance, no further evaluation of this issue is presented in support of implementing the AST for FCS. The existing equipment qualification analyses, which are based upon the Technical Information Document (TID) 14844 (Reference 3) source term, are considered acceptable.

2. Impact Upon Emergency Planning Radiological Assessment Methodology

   This application of the AST for FCS replaces the existing design basis source term with a source term developed as defined in RG 1.183. The Emergency Assessment of Gaseous and Liquid Effluents (EAGLE) model that is employed for emergency planning radiological assessments includes definitions of source terms for various design basis accidents. Calculation results from EAGLE are used in various emergency preparedness processes. The basis of the existing source term definitions in the EAGLE calculations will be evaluated on or before September 30, 2001 to determine: 1) the manner in which the source terms used in emergency preparedness activities rely upon the design basis event source term definition and 2) what specific changes may be warranted in the emergency preparedness source terms and their detailed usage.

3. Impact of Increased Particulate Loading on Containment Fan Cooler Units

   The impact of increased particulate loading on the containment fan cooler units (e.g., fans, cooling coils, and drains) has been addressed for Indian Point 2 in the report AEB-99-01 (Reference 12). This report compared the amount of aerosols removed by the fan coolers with the amount of steam condensed by the fan coolers. If the amount of aerosols removed were small compared with the amount of water condensed, then it could be concluded that the aerosols may be washed off the fan cooler coils and have a small impact on fan cooler performance. It was concluded that for Indian Point 2, both radioactive and non-radioactive aerosols would have a small impact on fan cooler performance because ample water would be available to wash down the fan cooler coils. Parameters supporting this conclusion have been evaluated and determined to be comparable to FCS.

4. Technical Support Center (TSC) Habitability Evaluation

   During the TSC habitability analysis using AST, it was identified that the post LOCA doses could exceed 5.0 rem Total Effective Dose Equivalent (TEDE) using the existing TSC HVAC system due to 30-day occupancy assumptions and design airflow in the TSC. A re-analysis and design evaluation of the TSC HVAC has been initiated to resolve this discrepancy, which is not related to implementation of the associated AST Technical Specifications changes. Appropriate administrative measures assure TSC personnel will not exceed 5.0 rem TEDE until a permanent resolution is implemented. The approach and schedule for the permanent resolution will be provided by July 31, 2001.

5. Radiolysis Products

   OPPD has evaluated the potential for the development of radiolysis products created during a LOCA event. The result of this evaluation is that an additional amount of approximately 16 cubic feet of trisodium phosphate (TSP) is required to maintain a containment sump pH equal to or greater than 7.0. This amount will be added to the containment sump baskets during the 2001 refueling outage.

6. Post Accident Access

   A comparison was performed between TID-14844 and AST core inventories to determine the affect on vital area access as required by NUREG-0737 (Reference 13). This comparison showed that the ASTs were lower for a majority of the isotopes when compared with TID-14844. The vital area access dose calculations are conservative and include a safety factor of 10 percent. Based on a preliminary evaluation, the doses will be bounded by current calculations. As an additional effort, OPPD will review, on or before September 30, 2001, the current vital area access dose calculations with respect to AST to ensure the results of the preliminary evaluation are valid.

7. Other Design Basis Not Affected

   The AST change has been determined to have no affect on post accident sampling capability, accident monitoring instrumentation, or leakage control.

To maintain the conditions and assumptions utilized by these analyses, OPPD proposes to revise the Fort Calhoun Station Unit No. 1 Technical Specifications (TS) as follows:

1. Add TS 2.8.2(4) to require the control room ventilation system to be in operation and in the Filtered Air mode during core alterations and refueling operations in the reactor containment building. Add Bases 2.8.2(4) to describe the basis for placing the control room ventilation system in the Filtered Air mode.

2. Revise TS 2.8.3(5) to delete the requirement for the ventilation isolation actuation signal (VIAS) to be operable with two radiation monitors operable, require the control room ventilation system to be in operation and in the Filtered Air mode, and require a spent fuel pool area radiation monitor to be in operation during refueling operations in the spent fuel pool area. Modify Bases 2.8.3(5) to describe the basis for placing the control room ventilation system in the Filtered Air mode and requiring a spent fuel pool area radiation monitor to be in operation.

3. Revise TS 3.16 to require performance of an internal leakage test, correct the title, and limit total RHR leakage to 3800 cc/hour. Modify Basis 3.16 to describe the basis for specifying the new limit for RHR system leakage.

4. Revise TSs 2.3(4) to increase the minimum amount of trisodium phosphate (TSP) from 110 ft^3 to 126 ft^3, and modify Basis 2.3 to describe the basis for specifying this revised limit.

5. Revise TSs 3.6(2) to increase the minimum amount of trisodium phosphate (TSP) from 110 ft^3 to 126 ft^3, and replace the word undisturbed with representative. Modify Basis 3.6 to describe the basis for specifying the revised TSP limit.

II. JUSTIFICATION

This TS amendment application results from the re-analysis of the design basis radiological consequences for LOCA, FHA (in containment and spent fuel pool), CREA, MSLB, SGTR, SRA, GWDTF, LWTF, and HLD (in containment). These analyses incorporated the results of the FCS design verification process and the features of the AST, including the TEDE analysis methodology and modeling of plant systems and equipment operation that influence the events. The calculated radiological consequences were compared with the revised limits provided in 10 CFR 50.67(b)(2) (Reference 4), as clarified per the additional guidance in RG 1.183 (Reference 2). Dose calculations were performed for the exclusion area boundary (EAB) for the worst 2-hour period, and for the low population zone (LPZ) and control room for the duration of the accident (30 days). All the radiological consequence calculations for the AST were performed by Stone & Webster Engineering Corporation (SWEC) with the PERC2 computer code in accordance with 10 CFR 50 Appendix B quality assurance procedures, and a summary report of these calculations can be found in Attachment E. The computer codes used to support this application are safety related Category 1 and have been previously used in industry submittals.

Explanation of SWEC calculations is provided in attachment E. The results of these analyses and the dose acceptance criteria that apply for implementing the AST are provided in Table 1. As this table shows, the resultant doses are below the regulatory limits listed in RG 1.183. The resultant EAB and LPZ doses are significantly below the regulatory limits for all of the accident scenarios except the HLD assessment. The resultant control room doses are less than 60 percent of the limits for all of the accident scenarios except the SRA and the LOCA assessments.

The HLD analyses were performed with pool decontamination factors based on approximately eleven feet and twenty-three feet of water above the fuel assemblies. Only the results from the eleven-foot case are shown in Table 1 since this case yields the most conservative dose results. Performing two heavy load drop analyses was a result of the FCS design verification process, which determined that some heavy loads are moved above the reactor cavity prior to flooding the cavity. The current analysis was reviewed and determined to be bounding. The control room and offsite doses for SGTR, MSLB, SRA and CREA were based on conservative steaming rates used in the radiological consequences calculations.

The control room dose result for the SRA analysis is due to additional conservative assumptions and methods, such as the assumption of a seven-hour delay for control room emergency ventilation actuation following an SRA (Attachment E).

The LOCA analysis was performed without crediting the containment charcoal filters. As explained in Attachment E, the AST calculations were performed using 38 scfm unfiltered in-leakage. This in-leakage is a conservative value based on extensive tracer gas testing performed at FCS (Reference 14), which determined that the actual control room unfiltered in-leakage is approximately 8 scfm. These conservative assumptions result in doses which are below the regulatory limits.

The proposed change to TS 3.16 is necessary to ensure that the total RHR leakage assumed in the re-analysis for LOCA is periodically verified within specifications.

The proposed change the volume of TSP in containment (TSs 2.3 and 3.6) is necessary to ensure the post-LOCA pH of the recirculation water is equal to or greater than seven. Radiation levels in containment following a LOCA may cause the generation of hydrochloric and nitric acids from radiolysis of cable insulation and sump water. TSP will neutralize these acids.

The change to replace the word undisturbed with representative (TS 3.6) is to clarify the meaning of the specification and to be consistent with the Standard TSs (STS), NUREG-1433. The word representative means typical example and the word undisturbed means not to interfere with. The word representative clearly describes the TSP sample that has been exposed to the same environmental conditions as that in the mesh baskets in containment.

10 CFR 50.36 requires, in part, that if an operating restriction is an initial condition of a design basis accident (DBA), then a Limiting Condition for Operation (LCO) should be established. This requirement is the justification for the proposed changes to TSs 2.8.2 and 2.8.3.

The alternative source term developed, as defined in RG 1.183, has been incorporated into the re-analysis of radiological effects on the design basis accidents for FCS. This represents a full implementation of the alternative source term in which the RG 1.183 source term will become the licensing basis source term for assessment of design basis events. The analysis results from the reanalyzed events meet all of the acceptance criteria as specified in 10 CFR 50.67 and RG 1.183.

Table 1 Alternative Source Term Dose Results (rem)

Accident	Control Room		EAB		LPZ	
	Resultant Dose	Reg. Limit	Resultant Dose	Reg. Limit	Resultant Dose	Reg. Limit
Control Rod Ejection Accident (CREA)	3.00	5.00	2.00	6.30	0.50	6.30
Main Steam Line Break (MSLB)	2.50	5.00	1.50	2.50	0.50	2.50
Steam Generator Tube Rupture (SGTR)	1.50	5.00	1.50	2.50	0.50	2.50
Seized Rotor Accident (SRA)	4.70	5.00	0.50	2.50	0.50	2.50
Fuel Handling Accident (FHA) in Containment	0.50	5.00	1.50	6.30	0.50	6.30
Fuel Handling Accident (FHA) in Spent Fuel Pool	0.50	5.00	1.50	6.30	0.50	6.30
Gaseous Waste Decay Tank Failure (GWDTF)	0.04	5.00	0.14	0.50	0.01	0.50
Liquid Waste Tank Failure (LWTF)	0.32	5.00	0.08	0.50	0.01	0.50
Heavy Load Drop (HLD) in Containment[1]	2.00	5.00	5.00	6.30	0.50	6.30
Loss of Coolant Accident (LOCA)	4.50	5.00	2.50	25.00	0.50	25.00

[1] The Heavy Load Drop in Containment is not considered part of the FCS design bases (Reference 11) but was reanalyzed for completeness.

III. NO SIGNIFICANT HAZARDS CONSIDERATION

The standards used to arrive at a determination that a request for amendment involves no significant hazards consideration are included in the Commission's Regulations, 10 CFR 50.92, which state that the operation of the facility in accordance with the proposed amendments would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any previously evaluated; or (3) involve a significant reduction in a margin of safety.

The proposed amendments have been reviewed with respect to these three factors and it has been determined that the proposed changes do not involve a significant hazard because:

1. The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed changes to FCS TS modify requirements to: place the control room ventilation system in operation and in filtered air mode during refueling operations in the containment or spent fuel pool, place a spent fuel pool area radiation monitor in operation during refueling operations at the spent fuel pool, delete a specification that requires a ventilation isolation actuation signal (VIAS) and two radiation monitors to be operable, increase the volume of trisodium phosphate (TSP) in the reactor containment building, include both internal and external leakage for the residual heat removal (RHR) system leakage test, perform an internal leakage test on the RHR system, and credit the alternative source term (AST) for the design basis site boundary and control room dose analyses. These TS changes do not impact operation of other equipment or systems important to safety. The proposed TS changes reflect the parameters used in the radiological consequences calculations described in Attachment E.

   The current TS 3.16 limits RHR system leakage to 1243 cc/hour from external sources and does not provide a limit for leakage from internal sources due to valve seat back leakage to the safety injection refueling water tank (SIRWT) or require an internal leakage test to be performed. The re-analysis for LOCA assumed a total leakage from all RHR sources of 3800 cc/hour. The internal leakage would leak back into the water remaining in the SIRWT. While it appears the allowable leakage is being increased, the limit is more inclusive, and therefore, more conservative than the current leakage limit. The internal leakage test performed on the RHR system will measure and quantify the back leakage into the SIRWT.

   The proposed changes to TSs 2.3 and 3.6 are necessary to ensure the post-LOCA pH of the recirculation water is equal to or greater than 7.0. Radiation levels in containment following a LOCA may cause the generation of hydrochloric and nitric acids from radiolysis of cable insulation and sump water. TSP will neutralize these acids. The radiolysis analysis performed demonstrates that the sump pH will be greater than or equal to 7.0 post design basis accident (DBA), which meets the intent of RG 1.183

regarding iodine revolatization. Therefore, there is no increase in the probability or consequences of an accident previously evaluated due to radiolysis concerns.

The proposed change to TS 2.8.2(4) requires the control room ventilation system to be in operation and in the Filtered Air mode. This is a conservative action to reduce control room operator exposure. This action is credited in the fuel handling accident analysis. 10 CFR 50.36 requires, in part, that if an operating restriction is an initial condition of a DBA, then a Limiting Condition for Operation (LCO) should be established. Therefore, this action, which will reduce operator exposure, will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change to TS 2.8.3(5) will delete the requirement for the ventilation isolation actuation signal (VIAS) to be operable with two radiation monitors operable, and require the control room ventilation system to be in operation and in the Filtered Air mode and a spent fuel pool area radiation monitor to be in operation during refueling operations in the spent fuel pool. The current basis for TS 2.8.3(5) is to ensure the control room ventilation system is operated in Filtered Air mode upon receipt of a VIAS. The proposed change will require the control room ventilation system placed in the Filtered Air mode during refueling operations, thereby eliminating the need for the VIAS to be operable. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The changes proposed do not affect the precursors for accidents or transients analyzed in Chapter 14 of the FCS USAR. Therefore, there is no increase in the probability of accidents previously evaluated. The probability remains the same since the accident analyses performed and discussed in the basis for the TS changes, involve no change to a system, component or structure that affects initiating events for any USAR Chapter 14 accident evaluated. A re-analysis of USAR Chapter 14 events was conducted with respect to radiological consequences. This re-analysis was performed in accordance with current accepted methodology, and consequences were expressed in terms of TEDE dose. The current methodology is no longer exactly comparable to the previous methods used for dose consequences. The previous dose calculations analyzed the dose consequences to thyroid and whole body as a result of postulated DBA events. The previous dose calculations were shown to be well below the regulatory limits of 10 CFR 100.11 (25 percent) with respect to thyroid and whole body dose. The current accepted NRC methodology, as described in 10 CFR 50.67, specifies new dose acceptance criteria in terms of TEDE dose. The revised analyses for all evaluated DBA events meet the applicable TEDE dose acceptance criteria (specified also in RG 1.183) for alternative source term implementation. The most current analyses do not credit several engineered safeguards features (ESF) filtration systems as the previous analyses did, and hence, are more conservative in that aspect. If a comparison is performed between the previous calculations (thyroid and whole body dose) and revised analyses TEDE results (per method shown in footnote 7 of RG 1.183), a slight increase in dose consequences is exhibited but is not significant, and the TEDE results are below regulatory acceptance criteria.

The changes proposed do not increase the probability of an accident previously evaluated. Because of the new regulatory requirements related to AST implementation, the dose consequences, if compared to previous ones, are only slightly increased (using guidance in footnote 7 of RG 1.183). However, the dose consequences of the revised analyses are below the AST regulatory acceptance criteria.

2. The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The implementation of the proposed changes does not create the possibility of an accident of a different type than was previously evaluated in the USAR. The proposed changes to FCS TS modify requirements to: place the control room ventilation system in operation and in filtered air mode during refueling operations in the containment or spent fuel pool, place a spent fuel pool area radiation monitor in operation during refueling operations at the spent fuel pool, delete a specification that requires a ventilation isolation actuation signal (VIAS) and two radiation monitors to be operable, increase the volume of trisodium phosphate (TSP) in the reactor containment building, include both internal and external leakage for the residual heat removal (RHR) system leakage test, perform an internal leakage test on the RHR system, and credit the alternative source term (AST) for the design basis site boundary and control room dose analyses

   The changes proposed do not change how DBA events were postulated nor do the changes themselves initiate a new kind of accident with a unique set of conditions. The changes proposed were based on a complete re-analysis of offsite and control room operator doses, where the system requirements being revised were not credited in the calculations. The revised analyses are consistent with the regulatory guidance established in RG 1.183. The revised analyses utilize the most current understanding of source term timing and chemical forms as a more appropriate mitigation technique. Not crediting filtration systems and only crediting natural forces is conservative from the aspect of dose consequences. Through this re-analysis, no new accident initiator or failure mode was identified.

3. The proposed changes will not involve a significant reduction in the margin of safety.

   The implementation of the proposed changes does not reduce the margin of safety. The radiological analyses results, with the proposed changes, remain within the regulatory acceptance criteria (10 CFR 50 Appendix A, 10 CFR 50.67) utilizing the TEDE dose acceptance criteria directed in RG 1.183. These criteria have been developed for application to analyses performed with alternative source terms. These acceptance criteria have been developed for the purpose of use in design basis accident analyses such that meeting these limits demonstrates adequate protection of public health and safety.

An acceptable margin of safety is inherent in these licensing limits. Therefore, there is no significant reduction in the margin of safety as a result of the proposed changes.

Therefore, based on the above, OPPD's position is that these proposed amendment changes do not involve a significant hazard as defined by 10 CFR 50.92. Also, since there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite, and there is no significant increase in individual or cumulative occupational radiation exposure, the proposed changes will not result in a condition which significantly alters the impact of the FCS on the environment. Thus, the proposed changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c) (9), and pursuant to 10 CFR 51.22 (b), no environmental assessment need be prepared.

LIC-01-0010
Attachment E
Site Boundary and Control Room Dose Analyses

IMPLEMENTATION OF ALTERNATIVE SOURCE TERMS

SITE BOUNDARY & CONTROL ROOM DOSE ANALYSES
for
FORT CALHOUN STATION

Prepared for
OMAHA PUBLIC POWER DISTRICT

January 2001

Prepared by
Stone & Webster
A Shaw Group Company

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1 INTRODUCTION

As a holder of an operating license, issued prior to January 10, 1997, and in accordance with 10CFR50.67 (Reference 1), Fort Calhoun Station (FCS) is voluntarily revising/replacing the accident source term used in all of its design basis site boundary and control room dose analyses by the Alternative Source Term (AST).

The methodology / scenarios used in the existing design basis accident analyses discussed in the FCS UFSAR, (some of which utilize pre-NUREG 0800 assumptions) are being updated to reflect the guidance provided in Regulatory Guide 1.183 (Reference 2). In addition, the updated analyses reflect the results of a design basis verification/re-constitution effort that was initiated by the licensee to support a total upgrade on the radiological accident analyses. Included in this verification process were the results of tracer gas testing performed to quantify control room unfiltered inleakage. Also included is the use of updated site boundary (Exclusion Area Boundary and Low Population Zone), and control room atmospheric dispersion factors.

The site boundary and control room dose analyses for the following design basis accidents have undergone a change in design basis as discussed above:

1. Loss of Coolant Accident (LOCA)
2. Fuel Handling Accident in the Fuel Pool (FHA in Fuel Pool)
3. Fuel Handling Accident in the Containment (FHA in Containment)
4. Seized Rotor Accident (SRA)
5. Control Rod Ejection Accident (CREA)
6. Main Steam Line Break (MSLB)
7. Steam Generator Tube Rupture (SGTR)
8. Gaseous Waste Decay Tank Failure (GWDTF)
9. Liquid Waste Tank Failure – Airborne releases (LWTF)

In addition, the Heavy Load Drop Event (HLD) was re-analyzed to maintain consistency in the radiological consequence analyses and incorporate related guidance provided in Regulatory Guide 1.183.

Note that at FCS, the MSLB, SGTR, GWDTF and LWTF are not impacted by implementation of the AST, as there is no accident initiated fuel damage associated with these events. However, to maintain consistency in design basis, and in the case of the MSLB & SGTR, to incorporate related guidance provided in RG 1.183, these analyses have also been revised.

With this application, FCS proposes a full implementation of the AST as defined in RG 1.183, Section 1.2.1.

2 REGULATORY APPROACH

2.1 Exceptions to Regulatory Guide 1.183

Except as noted, the updated FCS accident analyses follow the guidance provided in Regulatory Guide 1.183:

- The site boundary and control room breathing rates "traditionally acceptable" to NRC in accident analyses, were rounded "up" from their traditional values when presented in RG 1.183. The FCS accident analyses which were initiated prior to the release of RG 1.183, utilize the "traditional" breathing rates which had been noted in DG 1081(Draft Guide to RG 1.183). The impact on the dose analyses due to usage of the traditional breathing rates, (instead of those noted in RG 1.183), is negligible.

- To account for fuel conditions outside the bounds of RG 1.183, conservative estimates of FCS specific fuel gap fractions are utilized (in lieu of values noted in Table 3, RG 1.183) for non-LOCA events.

- Except as noted, assumptions regarding the occurrence and timing of a Loss of Offsite Power (LOOP) are in accordance with RG 1.183 and are selected with the intent of maximizing the doses. For the following reasons, a LOOP is not assumed with the FHAs, the HLD, the LWTF and the GWDTF. Per NRC Information Notice 93-17 (Reference 3), the need to evaluate a design basis event assuming a simultaneous/subsequent LOOP is based on the cause/effect relationship between the two events (an example illustrated in Reference 3 is that a LOCA results in a turbine trip and a loss of power generation to the grid, thus causing grid instability and a LOOP a few seconds later, i.e., a reactor trip could result in a LOOP). Reference 3 concludes that plant design should reflect all credible sequences of the LOCA/LOOP, but states that a sequence of a LOCA and an unrelated LOOP is of very low probability and is not a concern. The accidents listed above (i.e.; the FHAs, the HLD, LWTF & GWDTF) cannot cause a LOOP. Consequently, following the logic sequence discussed in Reference 3 relative to the LOCA/LOOP, these analyses do not address the potential effects of a LOOP.

- RG 1.183 does not address the HLD, LWTR or GWDTF. The accident scenarios utilized for these analyses reflect other guidance and/or site specific models.

2.2 Dose Acceptance Criteria

FCS has utilized the following acceptance criteria for the updated site boundary and control room dose analyses:

The acceptance criteria for the ***EAB and LPZ Dose*** is based on 10 CFR Part 50 § 50.67, and Section 4.4 Table 6 of Regulatory Guide 1.183:

(i) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, should not receive a radiation dose in excess of the accident specific total effective dose equivalent (TEDE) value noted in Reference 2, Table 6.

(ii) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), should not receive a radiation dose in excess of the accident specific TEDE value noted in Reference 2, Table 6.

Note :

1. The acceptance criteria utilized for the HLD event is that noted in RG 1.183 for the FHA.

2. The acceptance criteria utilized for the GWDTF is 500 mrem per BTP ETSB 11-5 (Reference 4).

3. The acceptance criteria utilized for the LWTF (airborne releases) is also 500 mrem. This value was selected for the LWTF as it represents a radwaste system failure, and is therefore considered similar to the GWDTF. It is noted that NUREG 0800 has eliminated the LWTF (airborne releases) from Chapter 15. However, since the LWTF (airborne releases) is reported in the current FCS UFSAR, it is considered a part of FCS licensing basis.

The acceptance criteria for the ***Control Room Dose*** is based on10 CFR Part 50 § 50.67:

Adequate radiation protection is provided to permit occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident.

3 COMPUTER CODES

The QA Category 1 Stone & Webster computer codes utilized to support this application are listed below:

1. Industry Computer Code SCALE 4.3, "Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation for Workstations And Personal Computers," Control Module SAS2, Version 3.1, developed by ORNL (S&W Program NU-230, V05, L03).

2. S&W Proprietary Computer Program ACTIVITY2, "Fission Products in a Nuclear Reactor", NU-014, V01, L03.

3. S&W Proprietary Computer Program IONEXCHANGER, NU-009, V01, L02.

4. S&W Proprietary Computer Program EN-113, "Atmospheric Dispersion Factors", V06, L08.

5. Industry Computer Code ARCON96, "Atmospheric Relative Concentrations in Building Wakes" developed by PNL (S&W Program EN-292, V00, L00).

6. S&W Proprietary Computer Code, PERC2, "Passive Evolutionary Regulatory Consequence Code", NU-226, V00, L01.

7. S&W Proprietary Computer Code, SWNAUA, "Aerosol Behavior in Condensing Atmosphere", NU-185, V02, L0.

8. S&W Computer Code, SW-QADCGGP, "A Combinatorial Geometry Version of QAD-5A", NU-222, V00, L02.

The above computer codes have been used extensively by S&W to support nuclear power plant design.

4 RADIATION SOURCE TERMS

4.1 Core Inventory

The inventory of fission products in the FCS reactor core is based on maximum full-power operation of the core at a power level equal to the current licensed rated thermal power including a 2% instrument error per Regulatory Guide 1.49 (Reference 5), and current licensed values of fuel enrichment and burnup.

The FCS equilibrium core inventory is calculated using computer code ORIGEN-S. The ORIGEN-S calculation is performed by utilizing the Control Module SAS2H of the ORNL SCALE 4.3 computer code package. SCALE 4.3 is a modular code system developed by Oak Ridge National Laboratory. SCALE 4.3 has been qualified by Stone & Webster for QA Category I use. The SAS2H control module provides a sequence to calculate the nuclide inventory in a fuel assembly by calling various neutron cross section treatment modules and the exponential matrix point-depletion module ORIGEN-S. SAS2H-ORIGENS calculates the time-dependent neutron flux and the buildup of fissile trans-uranium nuclides. It accounts for all major nuclear interactions including fission, activation, and various neutron absorption reactions. It calculates the neutron-activated products, the actinides and the fission products in a reactor core.

The FCS reactor core consists of 133 fuel assemblies with various Uranium-235 enrichments. The core average U-235 enrichment is 4.327% by weight. To obtain a conservative "composite" core inventory for the purposes of radiological accident analyses, the radionuclide inventory for 3.5%, 4%, and 5% average enriched cores are calculated. The highest activity for each isotope for the above three enrichments is chosen to represent the inventory of that isotope in the "composite" core.

The equilibrium core inventory is calculated based on plant operation at 102% of the power level (i.e., at 1530 MW_{th}), and assuming an 18-month fuel cycle. The equilibrium core at the end of a fuel cycle is assumed to consist of fuel assemblies with three different burnups, i.e., approximately 1/3 of the core is subjected to one fuel cycle, 1/3 of the core to two fuel cycles and 1/3 of the core to three fuel cycles. Minor variations in fuel irradiation time and duration of refueling outages will have a slight impact on the estimated inventory of long-lived isotopes in the core. However, these inventory changes will have an insignificant impact on the radiological consequences of postulated accidents.

The core inventory developed by ORIGEN-S using the above methodology includes over 800 isotopes. The FCS equilibrium core inventory of dose significant isotopes relative to LWR accidents is presented in Table 4.1-1.

4.2 Coolant Inventory

Stone & Webster QA Category I Proprietary computer code, ACTIVITY2, is used to calculate the design basis primary coolant activity concentrations for FCS based on the core inventory developed above using ORIGEN-S. The source terms for the primary coolant activity include the leakage from failed fuel and the decay of parent and second parent. The depletion terms of the primary coolant activity include radioactive decay, purification of the letdown flow and neutron absorption when the coolant passes the reactor core. The nuclear library includes 3rd order decay chains of approximately 200 isotopes.

Stone & Webster QA Category I Proprietary computer code, IONEXCHANGER, is used to calculate the design basis halogen and remainder activity concentrations in the secondary side liquid. The source terms for the secondary side activity include the primary-to-secondary leakage in steam generators and the decay products of parent and second parent. The depletion terms of the secondary side liquid activity include radioactive decay, and purification due to the steam generator blowdown flow.

The noble gas concentrations in the secondary steam are calculated by dividing the appearance rate (μCi/sec) by the steam flow rate (g/sec). The noble gas appearance rate in the steam generator steam space includes the primary-to-secondary leak contribution and the noble gas generation due to decay of halogens in the SG liquid. The activity concentrations of the other isotopes in the steam are determined by the SG liquid concentrations and the partition coefficients recommended in NUREG 0017, Rev 1.

The primary coolant technical specification activities are based on 1.0 μCi/gm Dose Equivalent I-131 for iodines and 100 / E_{avg} μCi/gm for non-iodine nuclides that make up >95% of the gross primary coolant activity with half-lives greater than 15 minutes. In addition:

- Isotopic compositions are based on the design reactor coolant equilibrium concentrations at 1% defective fuel.
- Iodine concentrations in the coolant are based on thyroid dose conversion factors for I-131, I-132, I-133, I-134, and I-135 obtained from TID-14844 (Reference 6).
- Average beta and gamma energies per disintegration are based on References 7, 8, and 29.

The technical specification iodine dose equivalent I-131 concentrations per nuclide are calculated with the following equation:

$$DEI131(i) = \frac{C(i) \times CT_{tot}}{\sum\{F(i) \times C(i)\}}$$

Where:

DCF(i) = TID-14844 Thyroid Dose Conversion Factor per Nuclide (Rem/Ci)
F(i) = $DCF(i) / DCF_{I\text{-}131}$
C(i) = reactor coolant equilibrium iodine concentration per nuclide
CT_{tot} = reactor coolant total (DE I131) technical specification iodine concentrations.

The pre-accident iodine spike, CT_{tot} is 60 μCi/gm (transient Tech Spec limit for full power operation) or 60 times the reactor coolant total iodine technical specification concentration.

The accident generated iodine spike activities are based on an accident dependent multiplier, times the equilibrium iodine appearance rate. The equilibrium appearance rates are conservatively calculated based on the technical specification reactor coolant activities, along with the maximum design letdown rate, maximum technical specification allowed leakage, and an assumed ion-exchanger iodine efficiency of 100%.

The secondary liquid technical specification concentration CT_{tot} is 0.1 µCi/gm, where C(i) is the design secondary coolant equilibrium concentrations per nuclide.

The technical specification non-iodine concentrations are calculated with the following equation:

$$\text{Non-Iodine Act(j) } \mu\text{Ci/gm} = \frac{100 \times C(j)}{\sum\{C(j) \times [(E_\beta(j) + E_\gamma(j)]\}}$$

Where:

$E_\beta(j)$	=	average beta energy for isotope j (MeV/disintegration)
$E\gamma(j)$	=	average gamma energy for isotope j (MeV/disintegration)
C(j)	=	reactor coolant equilibrium Non-Iodine Concentration per nuclide
100	=	Normalization factor [(µCi/gm) (MeV/disintegration)]

The noble gas and halogen primary and secondary coolant Technical Specification Activity Concentrations are presented in Table 4.2-1. The pre-accident iodine spike concentrations and the equilibrium iodine appearance rates (utilized to develop accident initiated iodine spike values), are presented in Table 4.2-2.

4.3 Gap Fractions for Non-LOCA Events

Table 3 in Regulatory Guide 1.183, specifies the fraction of Fission Product Inventory in the Gap to be used for non-LOCA accidents. The footnote identifies that the applicability of Table 3 is limited to LWR fuel with peak burnups of 62 GWD/MTU "provided that the maximum linear hear generation rate does not exceed 6.3 kW/ft peak rod average power for burnups exceeding 54 GWD/MTU." FCS utilizes a few high burnup fuel assemblies in high flux regions to decrease peaking ratios; these rods are driven to linear heat generation rates slightly in excess of 6.3 kW/ft.

The ANSI/ANS Standard 5.4 "Method for Calculating the Fractional Release of Volatile Fission Products from Oxide Fuel" (Reference 9) provides a conservative methodology for estimating gap activities. For isotopes with half-lives shorter than 1 month (iodines and noble gases excluding I-129 and Kr-85), the calculated gap activities do not depend strongly on prior fuel operating temperatures and power operation. An equilibrium release fraction is calculated using the peak fuel rod's average temperatures plus 200°F (at a linear heat generation rate of 8 kW/ft) for the fuel burnup periods beyond 54 GWD/MTU, to determine the worst release fraction using the equilibrium equation:

$$F = 3\left[\frac{1}{\sqrt{\mu}}\coth(\sqrt{\mu}) - \frac{1}{\mu}\right]$$

$$\mu = \frac{\lambda}{D'}$$

where:

λ	=	isotopic decay constant, sec^{-1}
D'	=	diffusion coefficient corrected for burnup and temperature

The 200°F margin and the higher than expected linear heat generation rate are used to encompass uncertainties in fuel assemblies' design configuration and variations in fuel management schemes.

A simplified and conservative approach is utilized for isotopes with half-lives in excess of a year; an incremental fuel release is calculated for the time period above 54 GWD/MTU (assuming constant power operation) and *added* to the RG 1.183 gap fractions to estimate the impact of higher power operation on high burnup rods.

For the fuel assemblies analyzed, calculation of the equilibrium release fraction for short half-life isotopes (< 1 month) yields gap fractions approximately ½ that of RG 1.183 indicating that the RG 1.183 numbers encompass the higher heat generation rate and temperatures used to envelope other vendors' fuel. The approach utilized for the long-lived isotopes is inherently conservative in that it has the RG 1.183 values as the lower limit. The Gap Release fractions predicted by this simplified approach is approximately 45% greater than the corresponding RG 1.183 values.

Since the simplified approach does not explicitly address power operation and temperature distributions in the fuel prior to 54 GWD/MTU, additional margin is added to the results of this assessment. Based on engineering judgement, a factor of 2 margin is applied to the RG 1.183 gap release fractions for non-LOCA events in order to address operation at higher power levels of several high burnup rods at FCS.

Except as noted, the following table provides the gap fractions utilized in the FCS non-LOCA analyses.

Nuclide Group	Reg.Guide 1.183 Gap Fraction for Non-LOCA events	Assumed FCS Gap Fraction for Non-LOCA events
I-131	0.08	0.16
Kr-85	0.10	0.20
Other Noble Gases	0.05	0.10
Other Halogens	0.05	0.10
Alkali Metals	0.12	0.24

Notes:

- RG 1.183 does not specifically address a Heavy Load Drop Event. The FCS HLD event conservatively assumes that the entire core is damaged. In RG 1.183, the gap fractions associated with the LOCA reflect "core average" since the entire core is postulated to be damaged. Therefore, the fraction of Core Inventory in the Fuel Gap specified in RG 1.183 for the Large Break LOCA is deemed applicable for the HLD.
  Noble gases : 5%
  Halogens : 5%
  Alkali Metals :5%

- In accordance with RG 1.183, the gap fraction associated with the Control Rod Ejection accident is as follows:
  Noble gases : 10%
  Halogens : 10%

TABLE 4.1-1
FCS Equilibrium Core inventory (Power Level : 1530 MWth)

ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)	ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)
AG-110			6.00E+06	PU-239			1.67E+04
	PARENT:	AG-110M	1.43E+05		PARENT:	NP-239	8.42E+08
AG-110M			1.43E+05		2ND PARENT:	AM-239	1.49E-01
AG-111			2.39E+06	PU-240			2.14E+04
	PARENT:	AG-111M	2.39E+06		PARENT:	NP-240	1.65E+06
	GRAND PARENT:	PD-111	2.39E+06		2ND PARENT:	NP-240M	0.00E+00
AG-112			1.10E+06	PU-241			5.40E+06
	PARENT:	PD-112	1.09E+06		PARENT:	CM-245	1.09E+01
AM-241			5.86E+03		2ND PARENT:	CF-249	3.11E-04
	PARENT:	PU-241	5.40E+06	PU-242			8.25E+01
	2ND PARENT:	CM-241	6.60E-01		PARENT:	AM-242	3.03E+06
AS-76			7.58E+02		GRAND PARENT:	AM-242M	3.76E+02
BA-137M			4.57E+06	RB-86			6.31E+04
	PARENT:	CS-137	4.80E+06		PARENT:	RB-86M	5.21E+03
	GRAND PARENT:	XE-137	7.71E+07	RB-88			3.33E+07
BA-139			7.59E+07		PARENT:	KR-88	3.25E+07
	PARENT:	CS-139	7.43E+07		GRAND PARENT:	BR-88	1.81E+07
	GRAND PARENT:	XE-139	5.60E+07	RB-89			4.37E+07
BA-140			7.59E+07		PARENT:	KR-89	4.09E+07
	PARENT:	CS-140	6.69E+07		GRAND PARENT:	BR-89	1.24E+07
	GRAND PARENT:	XE-140	3.96E+07	RB-90			4.05E+07
BA-142			6.62E+07		PARENT:	KR-90	4.40E+07
	PARENT:	CS-142	3.01E+07		GRAND PARENT:	BR-90	6.62E+06
	GRAND PARENT:	XE-142	5.78E+06		2ND PARENT:	RB-90M	1.24E+07
BR-82			1.16E+05	RB-90M			1.24E+07
	PARENT:	BR-82M	9.80E+04		PARENT:	KR-90	4.40E+07
BR-83			5.40E+06		GRAND PARENT:	BR-90	6.62E+06
	PARENT:	SE-83M	2.73E+06	RH-103M			6.39E+07
	2ND PARENT:	SE-83	2.51E+06		PARENT:	RU-103	6.41E+07
BR-85			1.15E+07		GRAND PARENT:	TC-103	6.42E+07
	PARENT:	SE-85	4.77E+06	RH-105			4.05E+07
CD-115			3.30E+05		PARENT:	RH-105M	1.24E+07
	PARENT:	AG-115	2.32E+05		GRAND PARENT:	RU-105	4.37E+07
	GRAND PARENT:	PD-115	2.93E+05		2ND PARENT:	RU-105	4.37E+07
	2ND PARENT:	AG-115M	9.69E+04	RH-105M			1.24E+07
CD-115M			1.51E+04		PARENT:	RU-105	4.37E+07
	PARENT:	AG-115	2.32E+05		GRAND PARENT:	TC-105	4.31E+07
	GRAND PARENT:	PD-115	2.93E+05	RH-106			2.37E+07
CE-141			7.00E+07		PARENT:	RU-106	2.15E+07
	PARENT:	LA-141	6.94E+07		GRAND PARENT:	TC-106	3.07E+07
	GRAND PARENT:	BA-141	6.88E+07	RN-220			1.19E-01
CE-143			6.63E+07		PARENT:	RA-224	1.19E-01
	PARENT:	LA-143	6.58E+07		GRAND PARENT:	TH-228	1.19E-01
	GRAND PARENT:	BA-143	5.75E+07	RU-103			6.41E+07
CE-144			5.24E+07		PARENT:	TC-103	6.42E+07
	PARENT:	LA-144	5.87E+07		GRAND PARENT:	MO-103	6.31E+07
	GRAND PARENT:	BA-144	4.58E+07	RU-106			2.15E+07
CM-242			1.74E+06		PARENT:	TC-106	3.07E+07
	PARENT:	AM-242	3.03E+06		GRAND PARENT:	MO-106	2.03E+07

TABLE 4.1-1
FCS Equilibrium Core inventory (Power Level : 1530 MWth)

ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)	ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)
	GRAND PARENT:	AM-242M	3.76E+02	SB-122			3.58E+04
CM-244			1.37E+05		PARENT:	SB-122M	3.58E+03
	PARENT:	AM-244	4.76E+06	SB-124			2.75E+04
CS-132			1.39E+03		PARENT:	SB-124M	6.21E+02
CS-134			6.06E+06	SB-125			3.30E+05
	PARENT:	CS-134M	1.46E+06		PARENT:	SN-125	2.02E+05
CS-134M			1.46E+06		GRAND PARENT:	IN-125	3.33E+05
CS-135M			1.41E+06		2ND PARENT:	SN-125M	6.10E+05
CS-136			1.97E+06	SB-127			3.50E+06
CS-137			4.80E+06		PARENT:	SN-127	1.41E+06
	PARENT:	XE-137	7.71E+07		GRAND PARENT:	IN-127	7.17E+05
	GRAND PARENT:	I-137	3.98E+07		2ND PARENT:	SN-127M	1.90E+06
CS-138			7.93E+07	SB-129			1.31E+07
	PARENT:	XE-138	7.38E+07		PARENT:	SN-129	5.09E+06
	GRAND PARENT:	I-138	2.00E+07		GRAND PARENT:	IN-129	1.46E+06
CS-139			7.43E+07		2ND PARENT:	SN-129M	4.94E+06
	PARENT:	XE-139	5.60E+07	SB-130			4.35E+06
	GRAND PARENT:	I-139	1.03E+07	SB-130M			1.85E+07
CS-140			6.69E+07		PARENT:	SN-130	1.39E+07
	PARENT:	XE-140	3.96E+07	SB-131			3.25E+07
	GRAND PARENT:	I-140	2.58E+06		PARENT:	SN-131	1.18E+07
EU-154			2.62E+05		GRAND PARENT:	IN-131	4.55E+05
EU-155			1.16E+05	SB-132			1.92E+07
	PARENT:	SM-155	1.49E+06		PARENT:	SN-132	9.40E+06
EU-156			8.45E+06		GRAND PARENT:	IN-132	1.20E+05
	PARENT:	SM-156	9.32E+05	SB-132M			1.87E+07
EU-157			9.40E+05	SB-133			2.76E+07
	PARENT:	SM-157	5.83E+05		PARENT:	SN-133	2.57E+06
EU-158			3.40E+05	SE-83			2.51E+06
EU-159			1.71E+05		PARENT:	AS-83	3.41E+06
GA-72			6.69E+02		GRAND PARENT:	GE-83	5.73E+05
	PARENT:	ZN-72	6.66E+02	SM-153			1.71E+07
GD-159			2.24E+05		PARENT:	PM-153	3.70E+06
	PARENT:	EU-159	1.71E+05	SN-121			3.28E+05
GE-77			2.92E+04		PARENT:	IN-121M	3.05E+05
	PARENT:	GE-77M	7.87E+04		GRAND PARENT:	CD-121	2.99E+05
	GRAND PARENT:	GA-77	7.68E+04		2ND PARENT:	IN-121	2.87E+04
	2ND PARENT:	GA-77	7.68E+04	SN-123			2.58E+04
H-3			2.12E+04		PARENT:	IN-123	2.65E+05
HO-166			2.61E+03	SN-125			2.02E+05
	PARENT:	DY-166	1.38E+02		PARENT:	IN-125	3.33E+05
I-129			1.39E+00	SN-127			1.41E+06
	PARENT:	TE-129	1.24E+07		PARENT:	IN-127	7.17E+05
	GRAND PARENT:	TE-129M	2.51E+06	SR-89			4.54E+07
	2ND PARENT:	TE-129M	2.51E+06		PARENT:	RB-89	4.37E+07
I-130			8.34E+05		GRAND PARENT:	KR-89	4.09E+07
	PARENT:	I-130M	4.44E+05	SR-90			3.82E+06
I-131			4.08E+07		PARENT:	RB-90	4.05E+07
	PARENT:	TE-131	3.44E+07		GRAND PARENT:	KR-90	4.40E+07
	GRAND PARENT:	TE-131M	8.06E+06		2ND PARENT:	RB-90M	1.24E+07
	2ND PARENT:	TE-131M	8.06E+06	SR-91			5.59E+07

TABLE 4.1-1
FCS Equilibrium Core inventory (Power Level : 1530 MWth)

ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)	ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)
I-132			5.97E+07		PARENT:	RB-91	5.25E+07
	PARENT:	TE-132	5.86E+07		GRAND PARENT:	KR-91	3.02E+07
	GRAND PARENT:	SB-132	1.92E+07	SR-92			5.84E+07
I-133			8.47E+07		PARENT:	RB-92	4.61E+07
	PARENT:	TE-133	4.65E+07		GRAND PARENT:	KR-92	1.59E+07
	GRAND PARENT:	SB-133	2.76E+07	SR-93			6.47E+07
	2ND PARENT:	TE-133M	3.83E+07		PARENT:	RB-93	3.72E+07
I-134			9.47E+07		GRAND PARENT:	KR-93	5.32E+06
	PARENT:	TE-134	7.75E+07	SR-94			6.39E+07
	GRAND PARENT:	SB-134	5.10E+06		PARENT:	RB-94	1.90E+07
	2ND PARENT:	I-134M	8.11E+06		GRAND PARENT:	KR-94	2.40E+06
I-135			8.04E+07	TB-160			3.42E+04
	PARENT:	TE-135	4.07E+07	TC-99M			6.81E+07
	GRAND PARENT:	SB-135	2.24E+06		PARENT:	MO-99	7.70E+07
I-136			3.78E+07		GRAND PARENT:	NB-99	4.52E+07
	PARENT:	TE-136	1.86E+07	TC-101			6.94E+07
	GRAND PARENT:	SB-136	3.50E+05		PARENT:	MO-101	6.94E+07
IN-115M			3.30E+05		GRAND PARENT:	NB-101	6.59E+07
	PARENT:	CD-115	3.30E+05	TC-104			5.23E+07
KR-83M			5.43E+06		PARENT:	MO-104	4.99E+07
	PARENT:	BR-83	5.40E+06		GRAND PARENT:	NB-104	1.90E+07
	GRAND PARENT:	SE-83M	2.73E+06	TC-105			4.31E+07
KR-85			4.35E+05		PARENT:	MO-105	3.63E+07
	PARENT:	KR-85M	1.15E+07	TE-127			3.44E+06
	GRAND PARENT:	BR-85	1.15E+07		PARENT:	TE-127M	5.66E+05
	2ND PARENT:	BR-85	1.15E+07		GRAND PARENT:	SB-127	3.50E+06
KR-85M			1.15E+07		2ND PARENT:	SB-127	3.50E+06
	PARENT:	BR-85	1.15E+07	TE-127M			5.66E+05
	GRAND PARENT:	SE-85	4.77E+06		PARENT:	SB-127	3.50E+06
KR-87			2.32E+07		GRAND PARENT:	SN-127	1.41E+06
	PARENT:	BR-87	1.84E+07	TE-129			1.24E+07
	GRAND PARENT:	SE-87	6.81E+06		PARENT:	TE-129M	2.51E+06
KR-88			3.25E+07		GRAND PARENT:	SB-129	1.31E+07
	PARENT:	BR-88	1.81E+07		2ND PARENT:	SB-129	1.31E+07
	GRAND PARENT:	SE-88	3.56E+06	TE-129M			2.51E+06
KR-89			4.09E+07		PARENT:	SB-129	1.31E+07
	PARENT:	BR-89	1.24E+07		GRAND PARENT:	SN-129	5.09E+06
	GRAND PARENT:	SE-89	1.25E+06	TE-131			3.44E+07
KR-90			4.40E+07		PARENT:	SB-131	3.25E+07
	PARENT:	BR-90	6.62E+06		GRAND PARENT:	SN-131	1.18E+07
LA-140			7.78E+07		2ND PARENT:	TE-131M	8.06E+06
	PARENT:	BA-140	7.59E+07	TE-131M			8.06E+06
	GRAND PARENT:	CS-140	6.69E+07		PARENT:	SB-131	3.25E+07
LA-141			6.94E+07		GRAND PARENT:	SN-131	1.18E+07
	PARENT:	BA-141	6.88E+07	TE-132			5.86E+07
	GRAND PARENT:	CS-141	5.12E+07		PARENT:	SB-132	1.92E+07
LA-142			6.84E+07		GRAND PARENT:	SN-132	9.40E+06
	PARENT:	BA-142	6.62E+07	TE-133			4.65E+07
	GRAND PARENT:	CS-142	3.01E+07		PARENT:	TE-133M	3.83E+07
LA-143			6.58E+07		GRAND PARENT:	SB-133	2.76E+07
	PARENT:	BA-143	5.75E+07		2ND PARENT:	SB-133	2.76E+07

TABLE 4.1-1
FCS Equilibrium Core inventory (Power Level : 1530 MWth)

ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)
	GRAND PARENT:	CS-143	1.56E+07
MO-99			7.70E+07
	PARENT:	NB-99M	3.08E+07
	GRAND PARENT:	ZR-99	6.99E+07
	2ND PARENT:	NB-99	4.52E+07
MO-101			6.94E+07
	PARENT:	NB-101	6.59E+07
	GRAND PARENT:	ZR-101	3.96E+07
NB-95			7.34E+07
	PARENT:	ZR-95	7.32E+07
	GRAND PARENT:	Y-95	7.08E+07
	2ND PARENT:	NB-95M	8.38E+05
NB-95M			8.38E+05
	PARENT:	ZR-95	7.32E+07
	GRAND PARENT:	Y-95	7.08E+07
NB-97			6.81E+07
	PARENT:	NB-97M	6.43E+07
	GRAND PARENT:	ZR-97	6.78E+07
	2ND PARENT:	ZR-97	6.78E+07
NB-97M			6.43E+07
	PARENT:	ZR-97	6.78E+07
	GRAND PARENT:	Y-97	5.56E+07
ND-147			2.78E+07
	PARENT:	PR-147	2.77E+07
	GRAND PARENT:	CE-147	2.64E+07
NP-239			8.42E+08
	PARENT:	AM-243	9.63E+02
	GRAND PARENT:	PU-243	1.37E+07
PD-109			1.47E+07
	PARENT:	RH-109	1.25E+07
	GRAND PARENT:	RU-109	1.08E+07
	2ND PARENT:	PD-109M	7.55E+04
PM-147			8.38E+06
	PARENT:	ND-147	2.78E+07
	GRAND PARENT:	PR-147	2.77E+07
PM-148			6.73E+06
	PARENT:	PM-148M	1.31E+06
PM-148M			1.31E+06
PM-149			2.31E+07
	PARENT:	ND-149	1.56E+07
	GRAND PARENT:	PR-149	1.46E+07
PM-151			8.08E+06
	PARENT:	ND-151	8.00E+06
	GRAND PARENT:	PR-151	4.79E+06
PR-142			2.25E+06
PR-143			6.48E+07
	PARENT:	CE-143	6.63E+07
	GRAND PARENT:	LA-143	6.58E+07
PR-144			5.27E+07
	PARENT:	CE-144	5.24E+07
	GRAND PARENT:	LA-144	5.87E+07
TE-133M			3.83E+07
	PARENT:	SB-133	2.76E+07
	GRAND PARENT:	SN-133	2.57E+06
TE-134			7.75E+07
	PARENT:	SB-134	5.10E+06
	GRAND PARENT:	SN-134	4.32E+05
TH-228			1.19E-01
	GRAND PARENT:	RA-228	0.00E+00
XE-131M			5.35E+05
	PARENT:	I-131	4.08E+07
	GRAND PARENT:	TE-131M	8.06E+06
XE-133			8.48E+07
	PARENT:	I-133	8.47E+07
	GRAND PARENT:	TE-133M	3.83E+07
	2ND PARENT:	XE-133M	2.64E+06
XE-133M			2.64E+06
	PARENT:	I-133	8.47E+07
	GRAND PARENT:	TE-133M	3.83E+07
XE-135			3.08E+07
	PARENT:	I-135	8.04E+07
	GRAND PARENT:	TE-135	4.07E+07
	2ND PARENT:	XE-135M	1.75E+07
XE-135M			1.75E+07
	PARENT:	I-135	8.04E+07
	GRAND PARENT:	TE-135	4.07E+07
XE-137			7.71E+07
	PARENT:	I-137	3.98E+07
	GRAND PARENT:	TE-137	6.10E+06
XE-138			7.38E+07
	PARENT:	I-138	2.00E+07
	GRAND PARENT:	TE-138	1.48E+06
Y-90			3.92E+06
	PARENT:	SR-90	3.82E+06
	GRAND PARENT:	RB-90	4.05E+07
	2ND PARENT:	Y-90M	1.94E+02
Y-91			5.76E+07
	PARENT:	SR-91	5.59E+07
	GRAND PARENT:	RB-91	5.25E+07
	2ND PARENT:	Y-91M	3.25E+07
Y-91M			3.25E+07
	PARENT:	SR-91	5.59E+07
	GRAND PARENT:	RB-91	5.25E+07
Y-92			5.88E+07
	PARENT:	SR-92	5.84E+07
	GRAND PARENT:	RB-92	4.61E+07
Y-93			4.39E+07
	PARENT:	SR-93	6.47E+07
	GRAND PARENT:	RB-93	3.72E+07
Y-94			6.88E+07
	PARENT:	SR-94	6.39E+07
	GRAND PARENT:	RB-94	1.90E+07

TABLE 4.1-1
FCS Equilibrium Core inventory (Power Level : 1530 MWth)

ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)	ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)
	2ND PARENT:	PR-144M	7.35E+05	Y-95			7.08E+07
PU-238			1.14E+05		PARENT:	SR-95	5.73E+07
	GRAND PARENT:	CM-238	0.00E+00		GRAND PARENT:	RB-95	9.12E+06
	2ND PARENT:	NP-238	1.36E+07	ZR-95			7.32E+07
					PARENT:	Y-95	7.08E+07
					GRAND PARENT:	SR-95	5.73E+07
				ZR-97			6.78E+07
					PARENT:	Y-97	5.56E+07
					GRAND PARENT:	SR-97	2.15E+07

TABLE 4.2-1
Primary and Secondary Coolant
Technical Specification Iodine and Noble Gas Concentrations

Nuclide	Primary Coolant (µCi/gm)	Secondary Coolant (µCi/gm)
I-131	6.91E-01	7.46E-02
I-132	2.27E-01	1.24E-02
I-133	9.52E-01	8.30E-02
I-134	1.17E-01	1.76E-03
I-135	5.00E-01	2.96E-02
Kr-83m	3.51E-01	
Kr-85m	1.26E+00	
Kr-85	1.01E+02	
Kr-87	8.33E-01	
Kr-88	2.37E+00	
Xe-131m	3.68E+00	
Xe-133m	3.22E+00	
Xe-133	2.40E+02	
Xe-135m	7.20E-01	
Xe-135	9.01E+00	

TABLE 4.2-2
Primary Coolant Pre-Accident Iodine Spike Concentrations and Equilibrium Iodine Appearance Rates

Nuclide	Pre-Accident Iodine Spike Activity Concen. (µCi/gm)	Activity Appearance Rates (µCi/sec)
I-131	41.4	5.57E+03
I-132	13.6	4.31E+03
I-133	57.1	8.70E+03
I-134	7	4.32E+03
I-135	30	5.88E+03

5 ACCIDENT ATMOSPHERIC DISPERSION FACTORS (X/Q)

5.1 Site Boundary Atmospheric Dispersion Factors

Normalized atmospheric dispersion (X/Q) values are calculated at the FCS Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) for post accident gaseous releases from the Containment Wall, Auxiliary Building Stack, Auxiliary Building Fresh Air Intake, MSSV/ADV Stacks, Radwaste Processing Building Ventilation Discharge Nozzle, Room 81 Pressure Relief Domes, the Condenser Evacuation Discharge, and the Turbine Discharge Point of the Turbine Driven Auxiliary Feedwater Pump.

The applicable methodology is identified in U.S. Nuclear Regulatory Commission Regulatory Guide 1.145 (U.S. NRC, 1982, Reference 10). The methodology is implemented using the Stone & Webster QA Category I proprietary computer "Atmospheric Dispersion Factors" (EN-113) using a continuous temporally representative 5-year period of hourly data from Fort Calhoun's meteorological tower (i.e., January 1, 1994 through December 31, 1998).

The Regulatory Guide 1.145 methodology for ground level sources is as follows:

$$X/Q_1 = \{(u)[(\pi)(\sigma_y)(\sigma_z) + (A/2)]\}^{-1}$$
$$X/Q_2 = [(u)(3\pi)(\sigma_y)(\sigma_z)]^{-1}$$
$$X/Q_3 = [(u)(\pi)(\sigma_y)(\sigma_z)]^{-1}$$

where:

- σ_y = $(M)(\sigma_y)$; with M representing the meander factor in RG 1.145.
- X/Q = the relative concentration (sec/m^3)
- σ_y, σ_z = horizontal and vertical dispersion coefficients, respectively, based on stability class and horizontal downwind distance (m)
- u = wind speed at the 10-meter elevation (m/sec)
- x = downwind distance (m)
- A = cross-sectional building area (m^2)

X/Q_1 and X/Q_2 values are calculated by EN-113 and the higher value selected. This value is then compared to the X/Q_3 value calculated by EN-113, and the smaller value is then selected as the appropriate value.

The EAB distances used in the calculation for each of the 16 downwind sectors are derived from a drawing of the site boundary. An LPZ distance of 9 miles (14,484 meters) is used. The EAB X/Q values for the Containment Wall and the Aux. Bldg. Stack are conservatively based on the EAB distances from the outer edge of the Containment Wall as the Auxiliary Building Stack is attached to the north wall of the Containment Building. These EAB X/Q values are also applicable to the Auxiliary Building Fresh Air Intake given the proximity of the vent to the Containment Building.

The same EAB distances are used for the MSSV/ADV Stacks and Room 81 Pressure Relief Domes as they are very close to each other relative to the EAB distances. These distances are derived by conservatively choosing the shorter distance to the EAB in each direction relative to the MSSV/ADV Stacks or Room 81 Pressure Relief Domes. This set of EAB distances is also considered to be representative of those for the Condenser Evacuation Discharge and the Turbine Discharge Point of the Turbine Driven Auxiliary Feedwater Pump. A separate set of EAB distances is used for the Radwaste Processing Building Ventilation Discharge Nozzle as it is somewhat farther removed from the other release points.

One LPZ distance in all directions is used for all release points given the magnitude of this distance (14,484 meters) relative to the separation of the release point locations.

The EAB distances for each of the 22.5° sectors are derived from the site boundary drawing by considering a 45° sector centered on each 22.5° sector as described in Regulatory Guide 1.145, Regulatory Position C.1.2. The Containment Building cross-sectional area relative to plant grade is used in the calculation for the Containment Wall and the Auxiliary Building Stack releases. No building wake effect (i.e., A = 0) is conservatively assumed for the other release points. The NRC Regulatory Guide 1.111 (U.S. NRC, 1977, Reference 11) "plain" terrain recirculation factors are used in the calculation of the annual average X/Q values.

The following assumptions are made for these calculations:

- The EAB distances on which the Containment Wall and Aux. Building Stack X/Q values are based are conservatively determined from the outer edge of the Containment Building,

- The Containment Wall and Aux. Building Stack EAB X/Q values are applicable to the Aux. Building Fresh Air Intake given the proximity of the vent to the Containment Building.

- The EAB distances for the MSSV/ADV Stacks or Room 81 Pressure Relief Domes are derived by conservatively choosing the shorter distance to the EAB in each direction relative to the MSSV/ADV Stacks or Room 81 Pressure Relief Domes.

- Containment Building wake effect is conservatively not used for the MSSV/ADV Stacks, Room 81 Pressure Relief Domes, Radwaste Processing Building Ventilation Discharge Nozzle, Condenser Evacuation Discharge release and the Turbine Discharge Point of the Turbine Driven Auxiliary Feedwater Pump.

- The NRC Regulatory Guide 1.111 "plain" terrain recirculation factors are used in the calculation of the annual average X/Q values.

The highest EAB & LPZ X/Q values from among all 22.5° downwind sectors for each release/receptor combination and accident period are summarized in Table 5.1-1. The 0.5%

sector dependent X/Q values are presented with parenthesis indicating worst case downwind sector.

5.2 Control Room Atmospheric Dispersion Factors

The control room intake X/Q values for the six releases are calculated using the latest version of the "Atmospheric Relative CONcentrations in Building Wakes" (ARCON96) methodology (Ramsdell, 1997, Reference 12). Stone & Webster has qualified computer code ARCON96 for QA Category I use. Input data consist of: hourly on-site meteorological data; release characteristics such as release height, stack radius, stack exit velocity, and stack flow rate; the building area affecting the release; and various receptor parameters such as its distance and direction from the release to the control room air intake and intake height.

This methodology has the ability to evaluate ground-level, vent, and elevated stack releases and treats building wake effects and stable plume meander effects when applicable. A mixed mode approach is used in the case of vent releases to determine if the release should be treated as ground-level or elevated. This methodology is also able to evaluate area source releases, (as in the case of multiple vents spread over a roof top), using the virtual point source technique where initial values of the dispersion coefficients are assigned based on the size of the area source. The various averaging time period X/Q values are calculated directly from running averages of the hourly X/Q values.

A continuous temporally representative 5-year period of hourly data from Fort Calhoun's meteorological tower (i.e., January 1, 1994 through December 31, 1998) is used in this calculation. Each hour of data, at a minimum, has a validated wind speed and direction at the 10-meter level and a temperature difference between the 60- and 10-meter levels.

All releases are conservatively treated as ground-level as there are no releases at this site that are high enough to escape the aerodynamic effects of the plant buildings (i.e., 2.5 times Containment Building height, U.S. NRC, 1982). In addition, the stack/vent release flows are not necessarily maintained throughout the accident period. The containment building area assumed to have an effect on the dispersion of the applicable releases is the portion above the auxiliary building roof. Only the Containment Wall and Auxiliary Building Stack releases are considered to be effected by the containment building wake effect. All other releases do not consider building wake effect as there is little or no interference from buildings either upwind or downwind of the releases given the release/receptor trajectories.

The specific release-receptor combinations for which X/Q values are calculated are as follows:

1. Containment Wall to Control Room Air Intake
2. Aux. Building Stack to Control Room Air Intake
3. Aux. Building Fresh Air Intake Vent to Control Room Air Intake
4. MSSV/ADV Discharge Stacks to Control Room Air Intake
5. Radwaste Processing Building Ventilation Discharge Nozzle to Control Room Air Intake
6. Room 81 Pressure Relief Domes to Control Room Air Intake

7. Condenser Evacuation Discharge to Control Room Intake
8. Turbine Discharge Point of the Turbine Driven Auxiliary Feedwater Pump.

The following assumptions are made for these calculations:

- The plume centerline from each release is conservatively transported directly over the control room air intake

- The containment building area having an effect on the dispersion of the applicable releases is that portion above the auxiliary building roof. Only the Containment Wall and Aux. Building Stack releases are considered to be effected by the containment building wake effect. All other releases do not consider building wake effect,

- The MSSV/ADV releases are from the centroid of a rectangle encompassing the discharge stacks,

- The Room 81 Pressure Relief Dome releases are from the centroid of a rectangle encompassing the four relief domes,

- The ARCON96 default wind direction range of 90° centered on the direction that transports the gaseous effluents from the release points to either of the intakes is used in the calculation unless a wider range is indicated by the Murphy & Campe S/D ratio (Murphy & Campe, 1974, Reference 13). The ARCON default calm wind speed value of 0.5 m/sec is also used in this calculation,

- The ARCON96 default values for surface roughness length (i.e., 0.10 meter), representative of the topography in the vicinity of the Fort Calhoun Plant, and sector averaging constant (4.0) are used in the calculation,

- All releases are conservatively treated as ground level releases as there are no release conditions that merit categorization as an elevated release (i.e., 2.5 times Containment Building height) at this site.

The X/Qs values for all release-receptor combinations are summarized in Table 5.2-1.

TABLE 5.1-1
Fort Calhoun Site Boundary Atmospheric Dispersion Factors (sec/m^3)

Exclusion Area Boundary [1]

Release Point	Averaging Period 0-2 hr
Containment Wall/ Aux. Bldg. Stack/ Aux. Bldg. Fresh Air Intake	2.56E-4 (E)
MSSV/ADV Stacks/ Room 81 Press. Relief Domes/ Condenser Evacuation Discharge/ Turbine Discharge Point of the Turbine Driven Auxiliary Feedwater Pump.	2.56E-4 (E)
Radwaste Processing Bldg. Ventilation Discharge Nozzle	2.46E-4 (ESE)

Note 1: An EAB X/Q value of 2.56E-4 sec/m^3 is used for all release points.

Low Population Zone

Release Point	Averaging Period 0-2 hr	0-8 hr	8-24 hr	1-4 day	4-30 day
All Releases	2.51E-5(NW)	7.29E-6(NW)	4.83E-6 (NW)	1.98E-6(NW)	5.49E-7(NW)

TABLE 5.2-1
Fort Calhoun Control Room Atmospheric Dispersion Factors (sec/m^3)

	Averaging Period				
Release/Receptor Combination	**0-2 hr**	**2-8 hr**	**8-24 hr**	**1-4 d**	**4-30 d**
Containment Wall/CR Air Intake	4.87E-03	4.19E-03	2.11E-03	1.61E-03	1.35E-03
Aux. Bldg. Stack/CR Air Intake	3.16E-03	2.37E-03	1.16E-03	8.93E-04	7.15E-04
Aux. Bldg. Air Intake/CR Air Intake	3.12E-03	2.21E-03	9.58E-04	6.88E-04	4.61E-04
MSSVs/ADVs/CR Air Intake	5.06E-03	4.46E-03	2.08E-03	1.59E-03	1.34E-03
Radwaste Nozzle/CR Air Intake	1.05E-03	9.04E-04	4.02E-04	2.84E-04	2.27E-04
Room 81 Domes/CR Air Intake	5.92E-03	4.79E-03	2.36E-03	1.73E-03	1.49E-03
Cond. Evac. Disch/CR Air Intake	2.04E-03	1.54E-03	7.12E-04	4.62E-04	3.36E-04
Turbine Discharge of the Turbine Driven AFW Pump/CR Intake	4.73E-03	3.75E-03	1.88E-03	1.36E-03	1.17E-03

6 DOSE CALCULATION METHODOLOGY

A S&W proprietary computer program, PERC2, is used to calculate the Committed Effective Dose Equivalent (CEDE) from inhalation and the Deep Dose Equivalent (DDE) from submersion due to halogens and noble gases at the offsite locations and in the control room. The CEDE is calculated using the ICRP-30 dose conversion factors. The committed doses to other organs due to inhalation of halogens and noble gas daughters are also calculated. PERC2 is a multiple compartment activity transport code with the dose model consistent with the regulatory guidance. The decay and daughter build-up during the activity transport among compartments and the various cleanup mechanisms are included.

The PERC2 activity transport model, first calculates the integrated activity (using a closed form integration solution) at the offsite locations and in the control room air region, and then calculates the cumulative doses as described below:

Committed Effective Dose Equivalent (CEDE) Inhalation Dose - The dose conversion factors by internal organ type are applied to the activity in the air space of the control room, or at the EAB/LPZ. The exposure is adjusted by the appropriate respiration rate and occupancy factors for the CR dose at each integration interval as follows:

$$Dh(j) = A(j) \times h(j) \times C2 \times C3 \times CB \times CO$$

Where:

Dh(j)	=	Committed Effective Dose Equivalent (rem) from isotope j
A(j)	=	Integrated Activity (Ci-s/m^3)
h(j)	=	Isotope j Committed Effective Dose Equivalent (CEDE) dose conversion factor (mrem/pCi) based on Federal Guidance Report No.11, Sept. 1988 (Reference 14)
C2	=	Unit conversion of $1x10^{12}$ pCi/Ci
C3	=	Unit conversion of $1x10^{-3}$ rem/mrem
CB	=	Breathing rate (m^3/s)
CO	=	Occupancy factor

Deep Dose Equivalent (DDE) from External Exposure - According to the guidance provided in Section 4.1.4 and Section 4.2.7 of RG 1.183, the Effective Dose Equivalent (EDE) may be used in lieu of DDE in determining the contribution of external dose to the TEDE if the whole body is irradiated uniformly. The EDE in the control room is based on a finite cloud model that addresses buildup and attenuation in air. The dose equation is based on the assumption that the dose point is at the center of a hemisphere of the same volume as the control room. The dose rate at that point is calculated as the sum of typical differential shell elements at a radius R. The equation utilizes, the integrated activity in the control room air space, the photon energy release rates per energy group from activity airborne in the control room, and the ANSI/ANS 6.1.1-1991 "neutron and gamma-ray flux-to-dose-rate factors" (Reference 15).

The Deep Dose Equivalent at the EAB and LPZ locations is very conservatively calculated using the semi-infinite cloud model outlined in TID-24190, Section 7-5.2, Equation 7.36, (Reference 16) where 1 rad is assumed 1 rem.

$\gamma D\infty(x,y,0)$ rad = $0.25\ E\gamma_{BAR}\ \psi(x,y,0)$

$E\gamma_{BAR}$	=	average gamma released per disintegration (Mev/dis)
$\psi(x,y,0)$	=	concentration time integral (Ci-sec/m^3)
0.25	=	$[\ 1.11 \times 1.6\text{x}10^{-6} \times 3.7\text{x}10^{10}\ ]\ /\ [\ 1293 \times 100 \times 2\ ]$

where:

1.11	= ratio of electron densities per gm of tissue to per gm of air
$1.6\text{x}10^{-6}$ (erg/Mev)	= number of ergs per Mev
$3.7\text{x}10^{10}$ (dis/sec-Ci)	= disintegration rate per curie
1293 (g/m^3)	= density of air at S.T.P.
100	= ergs per gram per rad
2	= factor for converting an infinite to a semi-infinite cloud

7 RADIOLOGICAL ACCIDENT REANALYSES

As discussed in Section 1, the methodology / scenarios used in the existing design basis accident analyses discussed in the FCS UFSAR, (some of which utilize pre-NUREG 0800 assumptions) are being updated to reflect the guidance provided in Regulatory Guide 1.183. In addition, the updated analyses reflect the results of a design basis verification/re-constitution effort that was initiated by the licensee to support a total upgrade on the radiological accident analyses. Included in this verification process were the results of tracer gas testing performed to quantify control room unfiltered inleakage. Also included is the use of updated site boundary (Exclusion Area Boundary and Low Population Zone), and control room atmospheric dispersion factors.

The site boundary and control room dose analyses for the following design basis accidents have undergone a change in design basis as discussed above:

1. Loss of Coolant Accident (LOCA)
2. Fuel Handling Accident in the Fuel Pool (FHA in Fuel Pool)
3. Fuel Handling Accident in the Containment (FHA in Containment)
4. Seized Rotor Accident (SRA)
5. Control Rod Ejection Accident (CREA)
6. Main Steam Line Break (MSLB)
7. Steam Generator Tube Rupture (SGTR)
8. Gaseous Waste Decay Tank Failure (GWDTF)
9. Liquid Waste Tank Failure – Airborne releases (LWTF)

In addition, the Heavy Load Drop Event (HLD) was re-analyzed to maintain consistency in the radiological consequence analyses and incorporate related guidance provided in Regulatory Guide 1.183.

Note that at FCS, the MSLB, SGTR, GWDTF and LWTF are not impacted by implementation of the AST, as there is no accident initiated fuel damage associated with these events. However, to maintain consistency in design basis, and in the case of the MSLB & SGTR, to incorporate related guidance provided in RG 1.183, these analyses have also been revised.

The worst 2-hour period dose at the EAB, and the dose at the LPZ for the duration of the release, is calculated for each of these events based on postulated airborne radioactivity releases. This represents the post accident dose to the public due to inhalation and submersion for each of these events. Offsite breathing rates used are as follows: 0-8 hr (3.47-E04 m^3/sec), 8-24 hr (1.75E-04 m^3/sec), 1-30 days (2.32E-04 m^3/sec). Due to distance/plant shielding, the dose contribution at the EAB/LPZ due to direct shine from contained sources is considered negligible for all the accidents.

The 0 to 30-day dose to an operator in the control room due to airborne radioactivity releases is developed for all of the referenced design basis accidents. This represents the post accident dose to the operator due to inhalation and submersion. The CR shielding design is based on the LOCA which represents the worst case DBA relative to radioactivity releases. The direct shine dose due to contained sources/external cloud is included in the CR doses reported for the LOCA.

7.1 Control Room Design / Operation / Transport Model

The Fort Calhoun control room (CR) is modeled as a single region. Isotopic concentrations in areas outside the control room envelope are assumed to be comparable to the isotopic concentrations at the control room intake locations. The FCS control room is designed to operate at 1/8 w.g during both normal operation as well as accident mode. The control room post-accident ventilation model corresponds to a "single intake" design whereby on receipt of any one of several post accident signals (Safety Injection Actuation Signal [SIAS], Containment Spray Actuation Signal [CSAS], Containment Atmosphere Radiation High Signal [CRHS], Containment Pressure High Signal [CPHS], Pressurizer Pressure Low Signal [PPLS]), the control room ventilation system switches automatically from a normal unfiltered intake of 1000 cfm to a filtered intake.

For those events that address a Loss of Offsite Power (LOOP) the model considers the most unfavorable time following the accident. To address the LOOP, the automatic initiation of the CR emergency system is delayed by 44 seconds to take into account the following: 14 seconds for the diesel generator to become fully operational (including sequencing delays), 15 seconds for the damper re-alignment, and 15 seconds for the emergency fans to come up to speed.

In the emergency ventilation mode, the control room has both intake and recirculation filtration at 1000 cfm each, and an assumed unfiltered inleakage of 38 cfm. Based on tracer gas testing, the measured inleakage into the CR is 38 cfm of which it is estimated that the unfiltered inleakage is 8 cfm. (Reference 17). The CR filter has an efficiency of 99% for all iodines. The CR is equipped with double vestibule doors; therefore, per SRP 6.4 (Reference 18), it is assumed that there is no unfiltered inleakage due to egress/ingress.

Due to single failure considerations (the CR emergency ventilation recirculation flow damper is not redundant) and in accordance with the damper repair alternative discussed in SRP 6.4, Appendix A, (Reference 19), the CR emergency recirculation filtration is assumed to be unavailable for the first 2 hours (120 mins) after the event, for all automatic initiation scenarios following accidents assumed to occur during power operations. For these events, two emergency ventilation scenarios are considered to evaluate the control room design for habitability. It is postulated that since the same fan supports both the CR intake and recirculation flow, the intake flow rate may be different from its design value when there is no recirculation flow. The two scenarios analyzed to bound the issue are as follows: Scenario (a) during the first 120 minutes when there is no recirculation, the intake flow is assumed to be at its design value of 1000 cfm and Scenario (b) during the first 120 minutes when there is no recirculation the intake flow rate is assumed to be the sum of the design intake and recirculation flow rate, i.e., 2000 cfm.

The remaining events are treated as follows:

- The SRA conservatively assumes that the referenced 2-hour period (during which there is no CR recirculation filtration) occurs after the CR is manually put into emergency ventilation, 7 hours after the event.

- Per plant procedures, fuel movement in the fuel pool or containment, as well as heavy load movement above the reactor cavity cannot be initiated prior to placing the CR on emergency ventilation mode. Consequently, the above automatic initiation of CR emergency ventilation scenarios are not applicable to the FHAs in containment / Fuel pool area, as well as the HLD event.

- No credit is taken for the CR emergency ventilation for the GWDTF and the LWTF; therefore the above automatic initiation of CR emergency ventilation scenarios are not applicable.

A 10% margin is applied on all CR ventilation flows. Table 7.1-1 lists key assumptions / parameters associated with FCS control room design.

7.2 Loss of Coolant Accident (LOCA)

RG 1.183 identifies the large break LOCA as the design basis case of the spectrum of break sizes for evaluating performance of release mitigation systems / containment and facility siting relative to radiological consequences.

FCS has identified three activity release paths following a LOCA: (a) Containment Leakage, (b) ESF System Leakage (including Safety Injection Refueling Water Tank (SIRWT) back leakage), and (c) Containment Vacuum Relief Line Release. At FCS, containment purge for hydrogen control occurs after 30 days following a LOCA. Consequently, the dose impact of a containment purge is not included in the dose assessment.

Except as noted in Section 2, this assessment follows the guidance provided in RG 1.183 for the LOCA. Table 7.2-1 lists some of the key assumptions / parameters utilized to develop the radiological consequences following a LOCA.

Doses due to Submersion and Inhalation

A S&W proprietary computer program, PERC2, is used to calculate the control room and site boundary dose due to airborne radioactivity releases following a LOCA. PERC2 is a QA Category I code. It utilizes an exact solution analytical computational process that addresses radionuclide progeny, time dependent releases, transport rates between regions and deposition of radionuclide concentrations in sumps, walls and filters.

Containment Vacuum Relief Line Release

It is assumed that the containment vacuum release line is operational at the initiation of the LOCA and that the release is terminated as part of containment isolation. The entire RCS inventory, assumed to be at Technical specification levels, is released to the containment at T = 0 hours. It is conservatively assumed that 100% of the volatiles are instantaneously and homogeneously mixed in containment atmosphere. Containment pressurization (due to the RCS mass and energy release), combined with the relief line cross-sectional area, results in a 600 scfm release of containment atmosphere to the environment over a period of 5 seconds (i.e., prior to containment isolation). Since the release is isolated within 5 seconds after the LOCA, i.e., before the onset of the gap phase release assumed to be at 30 seconds, no fuel damage releases are postulated. The chemical form of the iodine released from the RCS is assumed to be 97% elemental and 3% organic.

Containment leakage.

The inventory of fission products in the reactor core available for release via containment leakage following a LOCA is based on Table 4.1-1 which represents a conservative equilibrium reactor core inventory of dose significant isotopes, assuming maximum full power operation at 1.02 times the current licensed thermal power, and taking into consideration fuel enrichment and burnup.

The fission products released from the fuel are assumed to mix instantaneously and homogeneously throughout the free air volume of the primary containment as it is released from the core. Fission product cleanup following a LOCA is accomplished by the containment spray system. Mixing of the "effectively" sprayed volume of containment with the unsprayed volume of the containment by the containment ventilation system aids in the cleanup. In order to quantify the effectiveness of the containment spray system, both the volume fraction of containment that is sprayed and the mixing rate between the sprayed and unsprayed volumes are quantified.

Effectively Sprayed Volume Fraction of Containment

The sprayed volume fraction of the containment is determined by superimposing the spray patterns onto the containment arrangement drawings. The sprayed volume is the volume of the unblocked spray patterns. The spray patterns are based on the nozzle manufacturer's laboratory tests at atmospheric conditions. The patterns have been compressed to account for the higher density atmosphere that exists during the DBA. The effectively sprayed volume is determined by the method described in the buoyancy mixing analysis description (Stone & Webster Engineering Corp. 2000, proprietary methodology previously submitted to NRC via References 20 and 30). The effectively sprayed volume fraction is 0.694.

Removal of Particulates by Sprays

The particulates are effectively removed from the containment atmosphere by the containment spray system. The particulate removal rate is calculated with Stone & Webster's proprietary SWNAUA Computer Program SWNAUA. The SWNAUA Program is a modified version of the NAUA/MOD4 Computer Program (Bunz et al 1982, Reference 21).

There are many aerosol mechanics phenomena that promote the depletion of aerosols from the containment atmosphere. These include the natural phenomena gravitational settling, diffusional plate-out, and diffusiophoresis. The particulate removal calculation only takes credit for diffusiophoresis and the removal effectiveness of sprays. However, agglomeration of the aerosol is considered. If natural removal phenomena were considered in this study, the effectiveness of spray removal would have been slightly reduced but the total removal effectiveness by all removal mechanisms would have increased.

The spray model in SWNAUA evaluates the particulate removal efficiency for each particle size in the aerosol by the following mechanisms: inertial impaction, interception, and Brownian diffusion. The aerosol removal constant due to spray is presented in NUREG-0772 (Ref 22) as:

$$\lambda_{spray} = \frac{3\, F_m\, h\, \varepsilon}{4\, R_{sp}\, \rho_w\, V} \, x \, \frac{v_{spray} - v_{sed}}{v_{spray}}$$

where:

λ_{spray} = Particulate removal constant for spray,
F_m = Spray mass flow rate,
h = Spray fall height,
ε = Collision efficiency,
R_{sp} = Spray droplet radius,
ρ_w = Density of the spray droplet,
V = Volume of containment,
v_{spray} = Velocity of the spray droplets, and
v_{sed} = Aerosol sedimentation velocity.

The plant parameters for Fort Calhoun are provided below:

Plant Parameters for Fission Product Cleanup Calculations

Parameter	Value
Sprayed Containment Volume	2.28×10^{10} cm^3
Fall Height	2,134 cm
Spray Flow Rate	1,885 gpm (Injection Phase) 3,100 gpm (Recirculation Phase)

The collision efficiency is divided into three contributing mechanisms as described in BMI-2104 (Battelle Columbus Laboratories 1984, Reference 23):

$$\varepsilon = \varepsilon_i + \varepsilon_r + \varepsilon_d$$

Where:

ε_i = Efficiency due to inertial impaction,
ε_r = Efficiency due interception, and
ε_d = Efficiency due to Brownian diffusion.

For viscous flow around the spray droplet, the inertial impaction efficiency is given in NUREG-0772 (Nuclear Regulatory Commission 1981):

$$\varepsilon_i = \frac{1}{\left[1 + \frac{0.75 \ln(2\,Stk)}{Stk - 1.214}\right]^2}$$

The critical Stokes number, Stk, for viscous flow is 1.214; for Stk below this value, the model assumes the efficiency of inertial impaction is 0.0. The Stk is calculated from BMI-2104 (Battelle Columbus Laboratories 1984):

$$Stk = \frac{2\,\rho_p\, r^2\, C_c\,(v_{spray} - v_{sed})}{9\,\mu\, R_{sp}}$$

where:

r = Aerosol particle radius,
ρ_p = Aerosol density,
C_c = Cunningham slip correction factor,
μ = Gas viscosity.

For droplet sizes typical of nuclear plant spray systems, the data of Walton and Woolcock (1960) (Reference 24), show that collision efficiency will be closer to that predicted for potential flow around the droplet. Calvert (1970) (Reference 25) fit this data with the expression:

$$\varepsilon_i = \left(\frac{Stk}{Stk + 0.7}\right)^2$$

The collision efficiency predicted by this equation is always higher than that predicted by the viscous expression given above. Calvert's fit is employed in this calculation.

As for the remaining constituents of the collision efficiency, the spray model employs an interception efficiency of the form:

$$\varepsilon_r \cong \frac{3}{2}\left(\frac{r}{R_{sp}}\right)^2 x \left(1 - \frac{1}{3}\frac{r}{R_{sp}}\right)$$

which is a conservative approximation of the expression given by BMI-2104 (Battelle Columbus Laboratories 1984). The efficiency due to Brownian motion is also taken from this report:

$$\varepsilon_d = 3.5\, Pe^{-2/3}$$

where:

Pe	=	Peclet number
	=	$2v_{spray}R_{sp}/D_B$,
D_B	=	Aerosol diffusion coefficient
	=	$k_{Boltz}TB$ (Fuchs 1964, p. 181, Reference 26),
k_{Boltz}	=	Boltzmann constant
	=	$1.3804 \quad 10^{-16}$ erg/K.

The aerosol mobility, B, is given by Fuchs (1964, p. 27):

$$B = \frac{C_c}{6\pi\,\mu\,r}$$

In most cases, the overall collision efficiency is dominated by inertial impaction, but for small aerosols, Brownian diffusion may become dominant. The collision efficiency due to inertial impaction increases as the aerosol size is increased, whereas that due to Brownian diffusion increases as the aerosol size decreases.

A single spray droplet radius of 900μ is used in the analysis, although the model has the capability of handling a distribution of up to 20 droplet radii. In any event, the spray removal efficiency is determined for each aerosol size bin.

<u>Removal of Particulates by Diffusiophoresis</u>

During diffusiophoresis, particulate matter is entrained in the steam as it flows to the condensation surfaces. In this calculation, steam is assumed to condense only on the spray droplets and on the particulate matter. No credit is taken for steam condensation on heat sinks. The diffusiophoresis model in the SWNAUA computer code is the same as that in the NAUA/MOD4 computer code. The removal rate of particulate by sprays and diffusiophoresis combined is given in Figure 7.2-1.

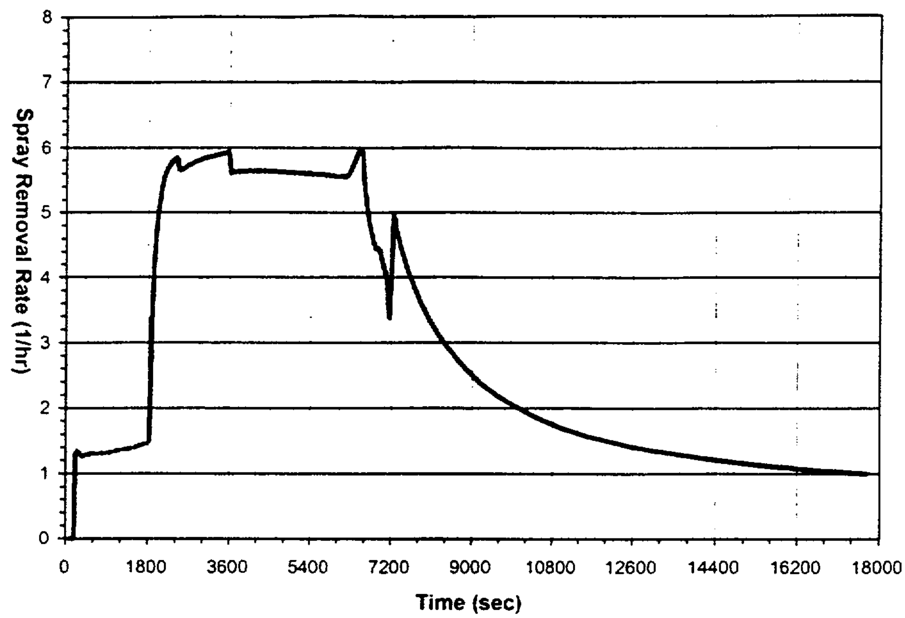


Figure 7.2-1: Particulate Removal Rate by Sprays and Diffusiophoresis

Elemental Iodine Removal by Sprays

Since the calculated particulate iodine removal coefficients are less than the elemental removal model of Standard Review Plan 6.5.2 (Nuclear Regulatory Commission 1988, Reference 27), it is assumed that the elemental removal rates are limited to the particulate removal rates. That is, it is assumed that the elemental form plates out onto the particulate form and is therefore removed at the same rate.

Mixing by the Containment Ventilation System

The mixing calculation shows that the containment ventilation system mixes the effectively sprayed and unsprayed volume of the containment at a higher rate than can be justified by natural convection. The methodology utilized to develop the mixing lambda is based on forced circulation by the containment fan coolers and is discussed in Reference 30. This rate, 4.84 unsprayed volumes per hour, is therefore used in the dose calculations.

Radiological Transport Model

As indicated previously, the fission products released from the fuel are assumed to mix instantaneously and homogeneously throughout the free air volume of the primary containment as it is released from the core. Two fuel release phases are considered for DBA analyses: (a) the *gap release*, which begins 30 seconds after the LOCA and continues for 30 mins and

(b) the *early In-Vessel release* phase which begins 30 minutes into the accident and continues for 1.3 hours.

The core inventory release fractions, by radionuclide groups, for the gap and early in-vessel damage are as follows:

Group	Gap Release Phase	Early In-Vessel Release Phase
Noble gas	0.05	0.95
Halogens	0.05	0.35
Alkali Metals	0.05	0.25
Tellurium Group	-	0.05
Ba, Sr	-	0.02
Noble Metals	-	0.0025
Cerium Group	-	0.0005
Lanthanides	-	0.0002

Elements in each Radionuclide Group released to the containment following a LOCA is assumed to be as follows (note that the groupings were expanded from that in RG 1.183 to address isotopes in the core with similar characteristics; the added isotopes are in bold font):

Noble gases:	Xe, Kr, **Rn, H**
Halogens:	I, Br
Alkali Metals:	Cs Rb
Tellurium Grp:	Te, Sb, Se, **Sn, In, Ge, Ga, Cd, As, Ag**
Ba,Sr:	Ba, Sr, **Ra**
Noble Metals:	Ru, Rh, Pd, Mo, Tc, Co
Cerium Grp:	Ce, Pu, Np, **Th, U, Pa, Cf, Ac**
Lanthanides:	La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am, **Gd, Ho, Tb, Dy**

Since the FCS sump pH is controlled to values of 7 and greater, the chemical form of the radioidine released from the fuel is assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodine. With the exception of noble gases, elemental and organic iodine, all fission products released are assumed to be in particulate form.

The activity released from the core during each release phase is modeled as increasing in a linear fashion over the duration of the phase. The release into the containment is assumed to terminate at the end of the early in-vessel phase, approximately 1.8 hours after the LOCA.

As discussed earlier, the activity transport model takes credit for aerosol/iodine removal via containment sprays. It considers mixing between the sprayed and unsprayed regions of the containment, reduction in airborne radioactivity in the containment by concentration dependent aerosol spray removal lambdas and isotopic in-growth due to decay. It is conservatively assumed that the sprays remove the elemental iodine at the same rate as the aerosol. Since the spray removal coefficients are based on calculated time dependent airborne aerosol mass, there is no restriction on the DF for particulate iodine. The maximum DF for elemental iodine is based on SRP 6.5.2 and is limited to a DF of 200. For FCS, this DF value is not reached for either the particulate or the elemental iodine before spray credit is stopped.

Credit for aerosol and elemental iodine removal via sprays is taken starting at T=185 seconds (spray initiation time based on minimum ESF case) and continued upto approximately T=5 hours after the LOCA. The "effectively" sprayed volume (69.4% of the containment free volume) utilized in this analysis is based on the sum of the sprayed volumes above and below the operating floor, as well as the unsprayed volume above the operating floor which is very highly mixed with the sprayed volume above the operating floor. A correction factor is applied to the spray removal coefficients within the sprayed region to consider the finite mixing rates between the sprayed and unsprayed region above the operating floor.

Mixing between the sprayed and unsprayed regions of the containment is assumed for the duration of the accident. The mixing rate between the "effectively" sprayed and unsprayed region is based on minimum fan cooler air flows below the operating floor, and is 4.84 per hour unsprayed region.

Trisodium Phosphate (TSP) located in baskets is used to maintain the sump pH greater than 7.0. Long-term production of acids (HCl and HNO_3), by irradiation is included in determining the required mass of TSP. Long-term retention of iodine in sump liquids is strongly dependent on the sump pH. The analysis does not address iodine re-evolution as a sump pH of $\geq$ 7 is achieved prior to the recirculation phase.

Radioactivity is assumed to leak from both the sprayed and unsprayed region to the environment through cracks and penetrations in the containment wall / steel liner, at the containment technical specification leak rate for the first day, and half that leakage rate for the remaining 29 days.

ESF / SIRWT Leakage

With the exception of noble gases, all the fission products released from the core in the gap and early in-vessel release phases are assumed to be instantaneously and homogeneously mixed in the primary containment sump water at the time of release from the fuel. A minimum sump volume of 314,033 gallons is utilized in this analysis. With the exception of iodine, all radioactive materials in the recirculating liquid are assumed to be retained in the liquid phase. The subsequent environmental radioactivity release is discussed below:

- ESF leakage: Equipment carrying sump fluids and located outside containment are postulated to leak at twice the expected value into the auxiliary building. ESF leakge is expected starting at initiation of the recirculation mode which at FCS is 20.4 minutes (start time based on maximum ESF; note that due to the long term nature of this release, minor variations in the start time of this release will not significantly impact the resultant doses). Since the temperature of the recirculation fluid is less than 212°F, 10% of the halogens associated with this leakage become airborne and are exhausted (without mixing and without holdup) to the environment via a release point in the Auxiliary Building with the most unfavorable dispersion characteristics relative to the control room intake (i.e., the Auxiliary Building Vent Stack). The chemical form of the iodine released from the sump water is 97% elemental and 3% organic. No credit is taken for the ESF filter system.

- SIRWT Back-leakage: Sump water back-leakage into the SIRWT (located in the Aux. Bldg.) is postulated to occur at twice the expected leakrate, and be released into the auxiliary building atmosphere via the SIRWT vent, starting at T=20.4 minutes (see ESF leakage above). Since the temperature of the fluid is less than 212°F, 10% of the halogens associated with this leakage become airborne and are exhausted (without mixing and without holdup) to the environment via a release point in the Auxiliary Building with the most unfavorable dispersion characteristics relative to the control room intake (i.e., the Auxiliary Building Vent Stack). No credit is taken for the ESF filter system. The SIRWT leakage activity is assumed to be released to the environment at the same release point as ESF leakage. As noted for the ESF leakage, the chemical form of the iodine released due to SIRWT leakage is 97% elemental and 3% organic

Due to their similar characteristics, the ESF and SIRWT leakage are modeled together as one release. The combined ESF and SIRWT leakage is set by Tech Spec to 3800 cc/hr. The analysis uses 7600 cc/hr as the ESF/SIRWT leakage rate (includes factor of 2).

Accident Specific Control Room Model Assumptions – Inhalation and Submersion

Due to the rapid pressure transient expected following a LOCA, the signal to initiate the CR emergency ventilation following a LOCA is assumed to occur at T=0 hours. The analysis assumes a LOOP at T=0 hours. However, the impact of a LOOP at the most unfavorable time following the accident, is also assessed. To address the concern that with AST, the activity release from the containment is at its maximum at approximately 1.8 hours post LOCA, the impact of a LOOP on the CR ventilation System at T=1.8 hours (approx) is conservatively "added" to the calculated airborne doses in the Control Room based on a LOOP at T=0 hours. The increase in the CR airborne doses due to the above conservative approach will be small. During the 44-second period after T=1.8 hours when the CR emergency ventilation system is assumed inoperative, no credit is taken for CR pressurization and an unfiltered inleakage of half the flow required to maintain pressurization, (i.e., 500 cfm) is assumed. As discussed earlier, due to single failure of the recirculation damper, the emergency recirculation filtration system is assumed to be unavailable for the first 2 hours after the event. The remaining CR parameters utilized in this model is discussed in Section 7.1.

***Control Room Dose due to Direct Shine from the External Cloud and Contained Sources*:**

The dose contribution in the control room due to direct shine from the external cloud and from contained sources (for both bulk shielding and through penetrations), is addressed. The external cloud contribution includes containment leakage, ESF leakage and SIRWT leakage. The contained sources include shine from the Containment Structure, ESF piping (SI-301R), control room HVAC filters, and the in-containment recirculation filters. FCS does not take credit for the In-Containment recirculation filters for accident analyses, but intends to retain the filter for normal operation purposes. Consequently, no credit is taken for fission product removal via the in-Containment recirculation filters when calculating the CR airborne dose; however, from a direct shine perspective, it is conservatively assumed that this filter will be operating post LOCA and will therefore act as a radiation source.

CR doses due to shine from contained sources is calculated at following locations: Main Control Board, Auxiliary Panel, Near South Wall Penetrations @ El 1036, Mezzanine office, Control Room Doorway, and Machine Room @ Mezzanine level. The main control board and the auxiliary panel represent the general access areas in the Control Room. The remaining four locations are low occupancy / less frequented areas which are specifically evaluated as they represent the worst-case locations in the Control Room relative to direct shine due to proximity to penetrations or to localized sources. The total time that an operator could spend in one or all of the referenced four locations, is conservatively estimated at less than 30% of the total time spent daily in the CR. The above "occupancy adjustments" are utilized to determine the maximum 30-day integrated dose in Control Room. The maximum control room operator dose following a LOCA is presented in Section 8.

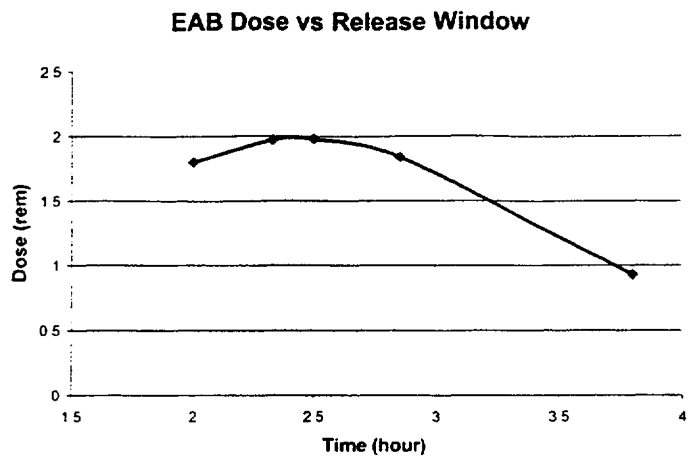
Site Boundary Dose Assessment

To find "worst-case 2-hour release window", several assessments are made with the environmental release beginning at 0, 20, 30, 50 and 108 minutes after a LOCA utilizing the full complement of nuclides. These start points for the "worst-case 2-hour window" are chosen for the following reasons:

T= 0 [Reference point]
T= 20 min [Near end of gap release, just before early-in-vessel release]
T= 30 min [Encompasses total early-in-vessel release]
T= 50 min [gives a window with an endpoint just after the early-in-vessel release]
T=108 min [provides a window that starts immediately after the early-in-vessel release]

The 0-2 hr EAB Atmospheric Dispersion Factor is utilized for all cases.

The 2-hour EAB doses from various "2-hour release windows" is tabulated and plotted. The abscissa is the endpoint of the release period while the ordinate is the 2-hour TEDE dose from Containment leakage for each window. From the graph below, the maximum dose occurs with a 2-hour release that ends at roughly 2.5 hours.



The EAB and LPZ dose following a LOCA is presented in Section 8.

7.3 Fuel Handling Accident in the Fuel Pool (FHA in Fuel Pool)

A S&W proprietary computer program, PERC2, is used to calculate the control room and site boundary dose due to airborne radioactivity releases following a FHA in the fuel pool.

Except as noted in Section 2, this assessment follows the guidance provided in RG 1.183 for the FHA. Table 7.3-1 lists some of the key assumptions / parameters utilized to develop the radiological consequences following a FHA in the fuel pool.

By FCS procedure, fuel handling activities in the Fuel Pool Area cannot be initiated until 72 hours after reactor shut down. It is postulated that the accident results in the damage of one (1) fuel assembly thus releasing all of the fuel gap activity associated with that assembly. As discussed in Section 4.3, the gap fractions utilized for Non-LOCA analyses at FCS is twice that recommended by RG 1.183. A radial peaking factor of 1.8 is applied to the activity release. The activity (consisting of noble gases, halogens, and alkali metals) is released in a "puff" to the fuel pool which has a minimum of 23 ft of water above the damaged fuel assembly.

The radioiodine released from the fuel gap is assumed to be 95% CsI, 4.85% elemental, and 0.15% organic. Due to the acidic nature of the water in the fuel pool (pH less than 7), the CsI is assumed to immediately disassociate, thus changing the chemical form of iodine in the water to 99.85% elemental and 0.15% organic. Based on decontamination factors of 500 and 1 for the elemental and organic iodines, respectively, the chemical form of the iodines above the pool is 57% elemental and 43% organic.

Noble gas and unscrubbed iodines rise to the water surface where they are mixed in the available air space. All of the alkali metals released from the gap are retained in the pool.

The fuel pool area is located in the Auxiliary building. The activity associated with a FHA in the fuel pool is collected by the fuel pool area ventilation system and released, unfiltered, to the environment, via the Auxiliary Building Vent Stack. Since there is no means of isolating, the fuel pool area, all of the airborne activity resulting from the FHA is exhausted out of the auxiliary building in a period of 2 hours. The closest opening in the Auxiliary building to the control room intake is the Auxiliary Building Fresh Air Intake. However, the Auxiliary building Vent Stack X/Q's are used as they bound that of the Auxiliary Building Fresh Air Intake.

Since the event is based on a 2-hour release, the worst 2-hour period for the EAB is the 0 to 2-hour period.

Accident Specific Control Room Model Assumptions

As discussed in Section 7.1, the 2-hour delay associated with manual alignment / repair of the recirculation damper is not applicable for this event as the CR is already aligned in the emergency mode prior to fuel movement. In addition, as discussed in Section 2, the analyses does not address a concurrent / subsequent LOOP. The remaining CR parameters utilized in this model is discussed in Section 7.1

The EAB, LPZ and Control room dose following a FHA in the Fuel Pool is presented in Section 8.

7.4 Fuel Handling Accident in the Containment (FHA in Containment)

A S&W proprietary computer program, PERC2, is used to calculate the control room and site boundary dose due to airborne radioactivity releases following a FHA in the Containment.

Except as noted in Section 2, this assessment follows the guidance provided in RG 1.183 for the FHA. Table 7.3-1 lists some of the key assumptions / parameters utilized to develop the radiological consequences following a FHA in the Containment.

By FCS procedure, fuel handling activities in the Containment cannot be initiated until 72 hours after reactor shut down. It is postulated that the accident results in the damage of one (1) fuel assembly thus releasing all of the fuel gap activity associated with that assembly. As discussed in Section 4.3, the gap fractions utilized for Non-LOCA analyses at FCS is twice that recommended by RG 1.183. A radial peaking factor of 1.8 is applied to the activity release. The activity (consisting of noble gases, halogens, and alkali metals) is released in a "puff" to the reactor cavity which has a minimum of 23 ft of water above the damaged fuel assembly.

The radioiodine released from the fuel gap is assumed to be 95% CsI, 4.85% elemental, and 0.15% organic. Due to the acidic nature of the water in the reactor cavity (pH less than 7), the CsI will immediately disassociate, thus changing the chemical form of iodine in the water to 99.85% elemental and 0.15% organic. Based on decontamination factors of 500 and 1 for the elemental and organic iodines, respectively, the chemical form of the iodines above the reactor cavity is 57% elemental and 43% organic.

Noble gas and unscrubbed iodines rise to the water surface where they are mixed in the available air space. All of the alkali metals released from the gap are retained in the reactor cavity water. Since the containment is assumed to be open, and there is no means of isolating the accident release, all of the airborne activity resulting from the FHA is exhausted out of the containment in a period of 2 hours.

The containment purge exhaust flow is operative during fuel movement in containment. This exhaust flow is released to the environment via the Aux Building Vent Stack. However, since the containment is "open", containment releases could occur from anywhere along the containment wall (e.g., via the equipment or personnel hatch). Because the location of the release is unknown, the worst case dispersion factors (x/Qs) are used in this analysis, i.e. those associated with the containment wall.

Since the event is based on a 2-hour release, the worst 2-hour period for the EAB is the 0 to 2-hour period.

Accident Specific Control Room Model Assumptions

As discussed in Section 7.1, the 2-hour delay associated with manual alignment / repair of the recirculation damper is not applicable for this event as the CR is already aligned in the emergency mode prior to fuel movement. In addition, as discussed in Section 2, the analyses does not address a concurrent / subsequent LOOP. The remaining CR parameters utilized in this model is discussed in Section 7.1

The EAB, LPZ and Control Room dose following a FHA in the Containment is presented in Section 8.

7.5 Heavy Load Drop Event (HLD)

A S&W proprietary computer program, PERC2, is used to calculate the control room and site boundary dose due to airborne radioactivity releases following a HLD event.

Except as noted in Section 2, this assessment follows the guidance provided in RG 1.183 for the FHA in Containment. As discussed in Section 4.3, the gap fractions are based on the guidance provided in RG 1.183 for the LOCA. Table 7.5-1 lists some of the key assumptions / parameters utilized to develop the radiological consequences following a HLD event.

This analysis assumes that heavy load movement in containment cannot be initiated until 72 hours after reactor shut down. Per plant procedures, with the exception of containment purge flow (50,000 cfm), containment closure is in effect during heavy load movement. A reduced containment purge rate (5000 cfm) is assumed during heavy load movement above the reactor cavity when the water level in the cavity is less than 23 ft.

Two scenarios are assessed, one with a minimum water level of 23 ft of water in the reactor cavity, and one with a minimum water level of 11.15 ft (water 1 ft below the reactor vessel flange). Redundant safety related radiation monitors that sample the containment atmosphere and the Containment/Aux. bldg. vent stack flow will isolate the containment purge system on a high radiation signal. Consequently, the postulated radioactivity release to the environment following a HLD Event in the containment building is assumed to be terminated upon the automatic isolation of the containment purge valves on receipt of a the high radiation signal from the most limiting radiation monitor.

The duration of the release following a HLD event is based on the sum of the sample transit time, monitor response time, damper closure time, plus the time required to purge all remnant activity from the ventilation duct/vent stack outside containment, after containment isolation.

The sample transit time, based on the limiting monitor, is estimated to be 1.02 minutes. The monitor response time is dependent on the monitor response, high alarm setpoint, and the activity levels in the sample. This response time is approximately 2 seconds for both scenarios. The containment purge damper closure time is established at 5 seconds.

To account for the activity left in the ductwork/vent stack outside the containment isolation valves after the isolation valves are closed, the PERC2 model assumes full purging for the time it would take to purge the duct/stack and release all of the activity. The extended purge time is calculated assuming slug flow, using the containment and Aux. Bldg. flow rates, and the total volume in the duct and stack. The time to purge the ductwork/vent stack is calculated to be 4.8 seconds for Scenario 1 (23 ft of water) and 35.3 seconds for Scenario 2 (11.15 ft of water).

The HLD event in containment conservatively assumes that all the fuel assemblies are damaged thus releasing all of the fuel gap activity. Since the entire core is assumed to be damaged, this analysis utilizes the gap fractions applicable to the LOCA as they represent "core average". A radial peaking factor is not utilized since the event impacts all the assemblies. The activity (consisting of noble gases, halogens, and alkali metals) is released in a "puff" to the flooded reactor cavity.

The radioiodine released from the fuel gap is 95% CsI, 4.85% elemental, and 0.15% organic. Due to the acidic nature of the water in the reactor cavity (pH less than 7), the CsI will immediately disassociate, thus changing the chemical form of iodine in the water to 99.85% elemental and 0.15% organic. Based on RG 1.183 decontamination factors of 500 and 1 for the elemental and organic iodines, respectively, the chemical form of the iodines above the cavity is 57% elemental and 43% organic for the case that has 23 ft of water in the cavity. The iodine DF ($\cong$ 20) for the case with 11.15 ft of water in the reactor cavity is based on methodology outlined by G. Burley, 1971 (Reference 28).

Noble gas and unscrubbed iodines rise to the water surface where they are mixed in the available containment air space. All of the alkali metals released from the gap are retained in the pool. Based on the assumption that the containment recirculation fans are in operation (minimum flow of approximately 190,000 cfm to 190,500 cfm; note that minor variations in this flowrate will not impact the conclusions of this evaluation relative to mixing) prior to heavy load movement, and demonstration of mixing (based on forced convection in containment and location of the purge exhaust registers vs the containment recirculation flow registers), the airborne noble gas and iodine activity is mixed in 50% of containment volume, then collected by the containment purge system and released, unfiltered, to the environment via the Auxiliary Building Stack. The release is terminated when isolation of containment purge is completed and all remnant activity in the duct/vent stack has been released.

Since the release is terminated within minutes of the event due to the isolation function provided by the radiation monitors, the worst 2-hour period for the EAB is the 0 to 2-hour period.

Accident Specific Control Room Model Assumptions

As discussed in Section 7.1, the 2-hour delay associated with manual alignment / repair of the recirculation damper is not applicable for this event as the CR is already aligned in the emergency mode prior to heavy load movement above the fuel. In addition, as discussed in Section 2, the analyses does not address a concurrent / subsequent LOOP. The remaining CR parameters utilized in this model is discussed in Section 7.1

The EAB, LPZ and Control Room dose following a HLD event is presented in Section 8.

7.6 Seized Rotor Accident (SRA)

A S&W proprietary computer program, PERC2, is used to calculate the control room and site boundary dose due to airborne radioactivity releases following a SRA.

Except as noted in Section 2, this assessment follows the guidance provided in RG 1.183 for the SRA. Table 7.6-1 lists some of the key assumptions / parameters utilized to develop the radiological consequences following a SRA.

A FCS Seized Rotor Accident results in 1% failed fuel and a release of the associated gap activity. The gap activity (consisting of noble gases, halogens and alkali metals) are instantaneously and homogeneously mixed in the reactor coolant system and transmitted to the secondary side via primary to secondary steam generator (SG) tube leakage assumed to be at the tech spec value of 1 gpm (@STP. As discussed in Section 4.3, the gap activity in the failed fuel are FCS specific values, and reflect FCS specific fuel conditions which are outside the bounds of RG 1.183. The conservative gap fractions utilized are twice that recommended by RG 1.183.

A radial peaking factor of 1.8 is applied to the activity release. The chemical form of the iodines in the gap are assumed to be 95% CsI, 4.85% elemental and 0.15% organic. At FCS, the SG tubes remain covered for the duration of the event; therefore, the gap iodines are assumed to have a partition coefficient of 100 in the SG. The iodine releases from the SG are assumed to be 97% elemental and 3% organic. The gap noble gases are released freely to the environment without retention in the SG whereas the particulates are carried over in accordance with the design basis SG moisture carryover fraction.

The condenser is assumed unavailable due to a coincident loss of offsite power. Consequently, the radioactivity release resulting from a SRA is discharged to the environment from both steam generators via the MSSVs and the ADVs. *(Note that a portion of this steam is released via the turbine exhaust of the turbine driven AFW Pump. However, the atmospheric dispersion factor of this release point is bounded by that of the MSSVs/ADVs. Consequently, the dose analyses conservatively assumes that all of the steam is discharged via the MSSVs/ADVs.).* The SG releases continue for 8 hours, at which time shutdown cooling is initiated via operation of the RHR system and environmental releases are terminated.

The accident assessment addresses steam releases from the MSSVs/ADVs for two events: a 2-hour event and an 8-hour event. For the 2-hour event, a 75°F/hr cooldown is conservatively assumed in the RCS to maximize the releases earlier on in the event. For the 8-hour event, the steam release rate used is such that the shutdown cooling temperature of 300°F is reached at 8 hours. In summary, the 2-hour event is intended to maximize releases early on in the event, and the 8-hour event is intended to maximize the total steam (i.e., radioactivity) released due to a SRA. Tables 7.6-2 and 7.6-3 provide the estimated steam release per SG, as a function of time, for the 2-hour and 8-hour events, respectively.

The worst 2-hour EAB dose is developed by evaluating both shutdown sequences. Based on engineering judgement, it is determined that the "worst" 2-hour EAB dose following a SRA will occur either during the T = 0 to 2-hour period for the 2-hour event case, or the T= 6 to 8-hour period for the 8-hour event case. The start times to determine the "worst-case 2-hour window" is chosen based on the fact that the noble gas activity release rate is at its highest level at the onset of the event, while the iodine and particulate activity release rate from the steam generator liquid peaks when the secondary system releases end at 8 hours. Regardless of the start time of the 2-hour window analyzed, the 0-2 hr X/Q is utilized.

The 8-hour event maximizes the total steam (and therefore, radioactivity) release to the environment. Therefore the 8-hour event is used to determine the 30-day control room and the LPZ dose.

The activity associated with the release of secondary steam/liquid, and primary to secondary leakage of normal operation RCS, (both at Tech Spec levels) via the MSSVs/ADVs are insignificant compared to the failed fuel release and are therefore not included in this assessment.

Accident Specific Control Room Model Assumptions

The Seized Rotor Accident does not initiate any signal which could automatically start the control room emergency ventilation. However, current EOPs require HP radiation surveillance in the control room, following a LOOP, or on receipt of a high radiation level alarm from the safety related condenser offgas monitor. If the survey indicates that the radiation levels in the control room are in excess of that expected during normal operation, the CR emergency ventilation system is manually activated. The assessment conservatively assumes that the CR emergency ventilation is manually initiated 7 hours after the accident. It is expected that following a SRA (with or w/o a LOOP), the CR emergency ventilation will be initiated well within this 7-hour period.

As discussed earlier, this assessment assumes a LOOP at T=0 hours. Since this scenario is based on manual initiation of the CR emergency system at T=7 hours, no additional delays in its actuation (i.e., the postulated 44 second delay due to a postulated LOOP) is addressed.

As discussed in Section 7.1, due to single failure of a CR damper, emergency recirculation filtration is not available for the first 2 hours (120 minutes) after a DBA for all automatic initiation sequences. However, since there is no specific signal that indicates that a Seized Rotor Event has occurred, it is conservatively assumed that filtered emergency recirculation is not available until 2 hours after the initiation of the emergency ventilation system, i.e., credit for filtered recirculation is taken after T=9 hours into the event. The remaining CR parameters utilized in this model is discussed in Section 7.1.

The EAB, LPZ and Control room dose following a SRA is presented in Section 8.

7.7 Control Rod Ejection Accident (CREA)

A S&W proprietary computer program, PERC2, is used to calculate the control room and site boundary dose due to airborne radioactivity releases following a CREA.

Except as noted in Section 2, this assessment follows the guidance provided in RG 1.183 for the CREA. Table 7.7-1 lists some of the key assumptions / parameters utilized to develop the radiological consequences following the CREA.

In accordance with guidance provided in RG 1.183, two independent release paths to the environment are analyzed: first, via *containment leakage* of the fission products released due to the event from the primary system to containment, *assuming that the containment pathway is the only one available*; and second, via releases from the *secondary system*, outside containment, following primary-to-secondary leakage in the steam generators, assuming that *the latter pathway is the only one available.*

The actual doses resulting from a postulated CREA would be a composite of doses resulting from portions of the release going out via the containment building and, portions via the secondary system. If regulatory compliance to dose limits can be demonstrated for each of the scenarios, the dose consequence of a scenario that is a combination of the two will be encompassed by the more restrictive of the two analyzed scenarios.

The FCS CREA analysis evaluates the following two scenarios. Loss of Offsite Power is assumed at T=0 hours.

Scenario 1 : The failed/melted fuel resulting from a postulated CREA is released into the RCS, which is released in its entirety into the containment via the ruptured control rod drive mechanism housing, is mixed in the free volume of the containment, and then released at containment technical specification leak rate. Environmental releases are assumed to occur via the containment wall.

Scenario 2: The failed/melted fuel resulting from a postulated CREA is released into the RCS which is then transmitted to the secondary side via steam generator tube leakage. The condenser is assumed to be unavailable due to a loss of offsite power. Environmental releases occur from both steam generators via the MSSVs and the ADVs.

A CREA at FCS will result in 10% failed fuel and 1% melted fuel. A peaking factor of 1.8 is applied to the release. In accordance with RG 1.183, the gap activity is assumed to be composed of 10% of the core noble gas and 10% of the core halogens associated with the percentage of fuel that has failed. Depending on the release pathway, the composition of the melted fuel is varied. For the containment leakage pathway, the melted fuel activity released is assumed to be composed of 100% of the core noble gas and 25% of the core halogens associated with the percentage of fuel that has melted. For the Secondary System Release pathway the melted fuel activity released is composed of 100% of the core noble gas and 50% of the core halogens associated with the percentage of fuel that has melted

The chemical composition of the iodine in the gap/melted fuel is assumed to be 95% CsI, 4.85% elemental and 0.15% organic. However, because the sump pH is not controlled following a CREA, it is conservatively assumed that the iodine released via the containment leakage pathway has the same composition as the iodine released via the secondary system release pathway; i.e.; it is assumed that for both scenarios, 97% of all halogens available for release to the environment are elemental, while the remaining 3% is organic.

Scenario 1: Transport From Containment

The failed / melted fuel activity released due to a CREA into the RCS is assumed to be instantaneously released into the containment where it mixes homogeneously in the containment free volume. The containment is assumed to leak at the technical specification leak rate of 0.001 day^{-1} for the first 24 hours and at half that value for the remaining 29 days after the event. Except for decay, no credit is taken for depleting the halogen (or noble gas) concentrations airborne in the containment.

Scenario 2: Transport From Secondary System

The failed / melted fuel activity released due to a CREA into the RCS is assumed to be instantaneously and homogeneously mixed in the reactor coolant system and transmitted to the secondary side via primary to secondary steam generator (SG) tube leakage assumed to be at the tech spec value of 1 gpm (@STP). At FCS, the SG tubes remain covered for the duration of the event; therefore, the gap iodines have a partition coefficient of 100 in the SG. The gap noble gases are released freely to the environment without retention in the SG.

The condenser is assumed unavailable due to a coincident loss of offsite power. Consequently, the radioactivity release resulting from a CREA is discharged to the environment from both steam generators via the MSSVs and the ADVs. *(Note that a portion of this steam is released via the turbine exhaust of the turbine driven AFW Pump. However, the atmospheric dispersion factor of this release point is bounded by that of the MSSVs/ADVs. Consequently, the dose analyses conservatively assumes that all of the steam is discharged via the MSSVs/ADVs.).* The SG releases continue for 8 hours, at which time shutdown cooling is initiated via operation of the RHR system and environmental releases are terminated.

The accident assessment addresses steam releases from the MSSVs/ADVs for two events: a 2-hour event and an 8-hour event. For the 2-hour event, a 75°F/hr cooldown is conservatively assumed in the RCS to maximize the releases earlier on in the event. For the 8-hour event, the steam release rate used is such that the shutdown cooling temperature of 300°F is reached at 8 hours. In summary, the 2-hour event is intended to maximize releases early on in the event, and the 8-hour event is intended to maximize the total steam (i.e., radioactivity) released due to a CREA. Tables 7.6-2 and 7.6-3 provide the estimated steam release per SG, as a function of time, for the 2-hour and 8-hour events, respectively.

The worst 2-hour EAB dose is developed by evaluating both shutdown sequences. Based on engineering judgement, it is determined that the "worst" 2-hour EAB dose following a CREA will occur either during the T = 0 to 2-hour period for the 2-hour event case, or the T= 6 to 8-hour period for the 8-hour event case. The start times to determine the "worst-case 2-hour window" is chosen based on the fact that the noble gas activity release rate is at its highest level at the onset of the event, while the iodine and particulate activity release rate from the steam generator liquid peaks when the secondary system releases end at 8 hours. Regardless of the start time of the 2-hour window analyzed, the 0-2 hr X/Q is utilized.

The 8-hour event maximizes the total steam (and therefore, radioactivity) release to the environment. Therefore the 8-hour event is used to determine the 30-day control room and the LPZ dose.

The activity associated with the release of secondary steam/liquid, and primary to secondary leakage of normal operation RCS, (both at Tech Spec levels) via the MSSVs/ADVs are insignificant compared to the failed fuel release, and are therefore not included in this assessment.

Accident Specific Control Room Model Assumptions

The CREA will result in a SIAS signal, 38 seconds into the event, which will result in the initiation of the CR emergency ventilation. An additional delay of 44 seconds is assumed to account for a coincident LOOP. As discussed earlier, due to single failure of the recirculation damper, the emergency recirculation filtration system is assumed to be unavailable for the first 2 hours after the event. The remaining CR parameters utilized in this model is discussed in Section 7.1

The EAB, LPZ and Control room dose following a CREA is presented in Section 8.

7.8 Main Steam Line Break (MSLB)

A S&W proprietary computer program, PERC2, is used to calculate the control room and site boundary dose due to airborne radioactivity releases following a MSLB.

Except as noted in Section 2, this assessment follows the guidance provided in RG 1.183 for the CREA. Table 7.8-1 lists some of the key assumptions / parameters utilized to develop the radiological consequences following the MSLB.

A FCS MSLB results in a SI actuation signal within 13.6 seconds after the event with the faulted steam generator being postulated to dry out by 136 seconds after the accident. Based on an assumption of a simultaneous Loss of Offsite Power, the condenser is assumed to be unavailable, and the ADVs of the intact steam generators are used to cool down the reactor until the RHR system starts shutdown cooling after the primary side temperature drops to 300°F. The elevated iodine activity in the RCS due to a postulated pre-accident or concurrent iodine spike as well as the Tech. Spec. noble gas activity leak to the faulted/intact steam generators and are released to the environment from the break point (Room 81 Dome), and from the ADVs. The SG releases continue until shutdown cooling is initiated via operation of the RHR system and environmental releases are terminated. *(Note that a portion of this steam is released via the turbine exhaust of the turbine driven AFW Pump. However, the atmospheric dispersion factor of this release point is bounded by that of the MSSVs/ADVs and the Room 81 Dome. Consequently, the dose analyses conservatively assumes that all of the steam is discharged via the MSSVs/ADVs and the Room 81 Dome.).*

Since at FCS, there is no postulated fuel damage associated with this accident, the main radiation source is the activity in the primary coolant system. Two spiking cases are addressed: a pre-accident iodine spike and a concurrent iodine spike.

a. Pre-accident spike - the initial primary coolant iodine activity is assumed to be 60 μCi/gm of DE I-131 which is the transient Technical Specification limit for full power operation. The initial primary coolant noble gas activity is assumed to be at Tech Spec levels.

b. Concurrent spike - the initial primary coolant iodine activity is assumed to be at Technical Specification of 1 μCi/gm DE I-131 (equilibrium Technical Specification limit for full power operation). Immediately following the accident the iodine appearance rate from the fuel to the primary coolant is assumed to increase to 500 times the equilibrium appearance rate corresponding to the 1 μCi/gm DE I-131 coolant concentration. The duration of the assumed spike is 8 hours. The initial primary coolant noble gas activity is assumed to be at Tech Spec levels.

The initial secondary coolant iodine activity is the Technical Specification limit of 0.1 μCi/gm DE I-131.

Following a MSLB, the primary and secondary reactor coolant activity is released to the environment via two pathways.

Affected Steam Generator

The first release path is via the affected Steam Generator's main steam line at the postulated break point. The affected steam generator is assumed to steam dry within 136 seconds of the MSLB, releasing all of the iodine in the secondary coolant that was initially contained in the steam generator. The secondary steam initially contained in the affected steam generator is also released; however, this contribution is not included in this analysis since the associated radioactivity is insignificant compared to the other contributions. The primary to secondary leakage is limited to 1 gpm at STP by Technical Specification. To maximize the dose consequence, it is assumed that the entire 1 gpm leak occurs in the affected steam generator. All iodine and noble gas activities in the referenced tube leakage are released directly to the environment without hold-up or decontamination. The primary to secondary leakage continues until the temperature of the RCS reaches 212°F.

Intact Steam Generator

The second release path is via the plant ADVs from the one (1) remaining intact steam generator which is used to cool the reactor and the primary system. The iodine activity in the intact SG liquid is released to the environment in proportion to the steaming rate and the partition factor. The steam releases from the ADVs address a 2-hour event and an 8-hour event. For a 2-hour event, a 75°F/hr cooldown was conservatively assumed in RCS. The resulting higher steam release rate is used to determine the EAB dose during the first 2 hours of the accident. For an 8-hour event, a cooldown was assumed such that a shutdown cooling temperature of 300°F was reached at 8 hours into the event. The resulting larger total steam release is used to determine the LPZ and control room doses. In the 2-hour steam release event, RCS reaches 212°F, at 4.94 hours after the accident. In the 8-hour steam release event, RCS reaches 212°F, at 10.94 hours after the accident. Tables 7.8-2 and 7.8-3 provide the estimated steam release from the intact SG, as a function of time, for the 2-hour and 8-hour events, respectively.

Accident Specific Control Room Model Assumptions

The MSLB will result in a SIAS signal, 14 seconds into the event, which will result in the initiation of the CR emergency ventilation. An additional delay of 44 seconds is assumed to account for a coincident LOOP. As discussed earlier, due to single failure of the recirculation damper, the emergency recirculation filtration system is assumed to be unavailable for the first 2 hours after the event. The remaining CR parameters utilized in this model is discussed in Section 7.1

The EAB, LPZ and Control room dose following a MSLB is presented in Section 8.

7.9 Steam Generator Tube Rupture (SGTR)

A S&W proprietary computer program, PERC2, is used to calculate the control room and site boundary dose due to airborne radioactivity releases following a SGTR.

Except as noted in Section 2, this assessment follows the guidance provided in RG 1.183 for the SGTR. Table 7.9-1 lists some of the key assumptions / parameters utilized to develop the radiological consequences following the SGTR.

The FCS SGTR results in a reactor trip and a simultaneous loss of offsite power within 412 seconds after the event. Due to the tube rupture the primary coolant with elevated iodine concentrations (pre-accident or concurrent iodine spike) flows to the faulted steam generator and the associated activities are released to the environment via secondary side steam releases. Before the reactor trip, the activities are released from the air ejector. After the reactor trip the steam release is via the MSSVs/ADVs. *(Note that a portion of this steam is released via the turbine exhaust of the turbine driven AFW Pump. However, the atmospheric dispersion factor of this release point is bounded by that of the MSSVs/ADVs. Consequently, the dose analyses conservatively assumes that all of the steam is discharged via the MSSVs/ADVs.).*

The spiking primary coolant activities leaked into the intact steam generator at the maximum allowable primary-to-secondary leakage value are also released to the environment via secondary steam releases.

Since at FCS, there is no postulated fuel damage associated with this accident, the main radiation source is the activity in the primary coolant system. Two spiking cases are addressed: a pre-accident iodine spike and a concurrent iodine spike.

a. Pre-accident spike - the initial primary coolant iodine activity is assumed to be 60 μCi/gm of DE I-131 which is the transient Technical Specification limit for full power operation. The initial primary coolant noble gas activity is assumed to be at Tech Spec levels.

b. Concurrent spike - the initial primary coolant iodine activity is assumed to be at Technical Specification of 1 μCi/gm DE I-131 (equilibrium Technical Specification limit for full power operation). Immediately following the accident, the iodine appearance rate from the fuel to the primary coolant is assumed to increase to 335 times the equilibrium appearance rate corresponding to the 1 μCi/gm DE I-131 coolant concentration. The duration of the assumed spike is 8 hours. The initial primary coolant noble gas activity is assumed to be at Tech Spec levels.

The initial secondary side liquid and steam activity is relatively small and its contribution to the total dose is negligible compared to that contributed by the rupture flow and is therefore not considered in this assessment.

Faulted SG Release

A postulated SGTR at FCS will result in a large amount of primary coolant being released to the faulted steam generator via the break location with a significant portion of it flashed to the steam space. The noble gases in the entire break flow and the iodine in the flashed flow are assumed immediately available for release from the steam generator without retention. The iodine in the non-flashed portion of the break flow mixes uniformly with the steam generator liquid mass and is released into the steam space in proportion to the steaming rate and partition factor. Before the reactor trip at 412 seconds, the activities in the steam are released to the environment from main condenser air ejector. All steam noble gases and organic iodine are released directly to the environment. Only a portion of the elemental iodine carried with the steam is partitioned to the air ejector and released to the environment. The rest is partitioned to the condensate, returned to both steam generators and assumed to be available for future steaming release. After the reactor trip, the break flow continues until the primary system is fully depressurized. No credit is taken for the condenser, since, to maximize the dose, a LOOP is assumed to occur simultaneously with the reactor trip. The steam is released from the MSSVs/ADVs. All activity releases from the faulted steam generator cease when it is isolated at 120 minutes after the accident. Table 7.9-2 provides the break flow of primary coolant into the faulted SG as a function of time. Additional information of steam releases from the faulted SG as a function of time is provided in Tables 7.9-3 and 7.9-4.

Intact SG release

The activity release from the intact steam generator is due to normal primary-to-secondary leakage and steam release from the secondary side. The Primary to Secondary leak rate is assumed to be at the maximum Tech Spec allowable value. All of the iodine activity in the referenced leakage is assumed to mix uniformly with the steam generator liquid and released in proportion to the steaming rate and the partition factor. Before the reactor trip at 412 seconds, the main steam is released from the air ejector / condenser. After the reactor trip, the steam is released from the MSSVs/ADVs. The reactor coolant noble gases that enter the intact steam generator are released directly to the environment without holdup. The steam release from the intact steam generator continues until initiation of shutdown cooling 8 hours after the accident. Steam releases from MSSVs/ADVs include a 2-hour event and an 8-hour event. For a 2-hour event, a 75°F/hr cooldown was conservatively assumed in RCS. The resulting higher steam release rate is used to determine the EAB dose. For an 8-hour event, a cooldown was assumed such that a shutdown cooling temperature of 300°F was reached at 8 hours into the event. The resulting larger total steam release is used to determine the LPZ and control room doses. To satisfy the maximum "2-hour window" EAB dose criteria, the 8-hour release rates with 2-hour EAB χ/Q values are also analyzed to ensure that the maximum 2-hour dose is calculated. Tables 7.9-3 and 7.9-5 provide the estimated steam release from the intact SG, as a function of time, for the 2-hour and 8-hour events, respectively.

Accident Specific Control Room Model Assumptions

The SGTR will result in a SIAS signal, 426 seconds into the event, which will result in the initiation of the CR emergency ventilation. An additional delay of 44 seconds is assumed to account for a coincident LOOP. As discussed earlier, due to single failure of the recirculation damper, the emergency recirculation filtration system is assumed to be unavailable for the first 2 hours after the event. The remaining CR parameters utilized in this model is discussed in Section 7.1

The EAB, LPZ and Control room dose following a SGTR is presented in Section 8.

7.10 Gaseous Waste Decay Tank Failure (GWDTF)

A S&W proprietary computer program, PERC2, is used to calculate the control room and site boundary dose due to airborne radioactivity releases following a GWDTF.

As noted in Section 2, RG 1.183 does not address a GWDTF. This assessment follows the guidance provided in BTP ETSB 11-5. Table 7.10-1 lists some of the key assumptions / parameters utilized to develop the radiological consequences following the GWDTF.

The activity available for release following a WGDTF is based on the guidance provided in BTP ESTB 11-5. The reactor is assumed to have operated at full power (including 2% instrument error) with 1% defective fuel, and a cold shutdown is assumed to be conducted at the end of the equilibrium fuel cycle. Immediately following shutdown, all noble gases are assumed to have been removed from the Reactor Coolant System (RCS) and transferred to the gas decay tank that is assumed to fail. No credit is taken for radiological decay during the transfer of RCS activity to the decay tank and all noble gas inventories are assumed to be available for release. The calculated noble gas activity in the Waste Gas Decay Tank (WGDT) is essentially the total design noble gas inventory in the RCS and it is conservative because no degassing and no Volume Control Tank (VCT) purging is assumed. The WDDT inventory is presented in Table 7.10-2. The WGDT activity is instantaneously released as a "PUFF", to the environment via the Auxiliary Building Stack.

Accident Specific Control Room Model Assumptions

The FCS control room ventilation system is assumed to be in the normal operation mode for this assessment; i.e., the Control Room ventilation normal intake flow (1000 cfm ± 10%) enters the Control Room unfiltered. The remaining CR parameters utilized in this model is discussed in Section 7.1

The EAB, LPZ and Control room dose following a WGDTF is presented in Section 8.

7.11 Liquid Waste Tank Failure – Airborne Releases (LWTF)

A S&W proprietary computer program, PERC2, is used to calculate the control room and site boundary dose due to airborne radioactivity releases following a LWTF.

As noted in Section 2, RG 1.183 does not address a LWTF. The accident scenario is FCS specific and represents a very conservative model. Table 7.10-1 lists some of the key assumptions / parameters utilized to develop the radiological consequences following the LWTF.

The maximum liquid radionuclide inventory in the Radioactive Waste Processing Building is the activity accumulated in the Filtration and Ion-Exchangers (FIX), which is located in Room 506. This room is seismic, curbed, and steel lined and coated to form a seismic resistant sump, which will retain the total inventory of the liquid radioactivity in an unlikely event of a rupture of a FIX tank. The liquid inventory will then be pumped back to the existing tanks and thus no liquid will be released to the environment. Evaluation of other liquid waste tanks have indicated that, their contents will also not be released to the environment.

However, the following conservative scenario is postulated in the current UFSAR to address a potential worst-case airborne release following a LWTF. This scenario is considered FCS licensing basis. A fraction of the halogen activity accumulated in the resin of the FIX is assumed released into the water upon tank rupture and a portion of the activity in the water becomes airborne and is released into the environment. All noble gases generated due to decay of halogens in the FIX tank are also available for release after tank rupture.

The following are the bases used in determining the airborne activity release for a rupture of a FIX tank.

The Filtration and Ion Exchange system is used to treat the high level liquid waste generated in FCS. The influent to the FIX is the contents of the Waste Holdup Tanks, and the effluent from the FIX is discharged to the Monitor Tanks. The worst-case liquid source in the Holdup Tanks is untreated primary coolant waste. The Holdup Tanks also receive reactor coolant waste via the letdown system, which has been treated by purification demineralizers. Other input streams to the Holdup Tanks have lesser activity concentrations. Therefore, a conservative feed stream source to the FIX is the weighted-average activity of untreated primary coolant and the Letdown Purification Demineralizer treated primary coolant.

Design RCS equilibrium activity concentrations (i.e., 1% defective fuel) is accumulated in the FIX at the maximum design flow rate (50 gpm) for sufficient time until the equilibrium activity is achieved. The activity inventory in the FIX is calculated taking into consideration decay and daughter buildup.

Upon the tank rupture, 10% of the halogen inventory on the ion exchanger resin is assumed instantaneously and non-mechanistically transferred to water. In addition, 10% of the halogen inventory in the water is assumed airborne and released to the atmosphere. Also, 100% of the noble gases generated by decay of halogens in the FIX tank are also instantaneously released to the atmosphere. The activity available for release from a LWTF is presented in Table 7.10-2. Upon tank failure, the LWT activity is instantaneously released as a "PUFF", to the environment via the Radwaste Building Exhaust Nozzle.

Accident Specific Control Room Model Assumptions

The FCS control room ventilation system is assumed to be in the normal operation mode for this assessment; i.e., the Control Room ventilation normal intake flow (1000 cfm ± 10%) enters the Control Room unfiltered. The remaining CR parameters utilized in this model is discussed in Section 7.1

The EAB, LPZ and Control room dose following a LWTF is presented in Section 8.

TABLE 7.1-1
Analysis Assumptions & Key Parameter Values
FCS Control Room

Control Room Parameters	
Free Volume	45,100 ft^3
Unfiltered Normal Operation Intake	1000 ± 10%
Emergency Intake Rate	1000 cfm ± 10%
Emergency Recirculation Rate	1000 cfm ± 10%
Emergency Intake Filter Efficiency	99% (iodine & particulates)
Emergency Recirculation Filter Efficiency	99% (iodine & particulates)
Unfiltered Inleakage	38 cfm
Occupancy Factors	0-24 hr (1.0)
	1 - 4 d (0.6)
	4-30 d (0.4)
Operator Breathing Rate	0-30 d (3.47E-04 m^3/sec)
Operator Action to Repair Recirc Damper	2 hours after accident
Emergency Intake Rate during Recirc Damper Repair Period	1000 cfm ± 10% to 2000 cfm± 10%

Delay in Initiation of Control Room Emergency Ventilation due to LOOP

Diesel Generator start up /sequencing	14 seconds
CR Damper Realignment	15 seconds
CR Emergency Fan Ramp UpTime	15 seconds
Total	44 seconds

TABLE 7.2-1
Analysis Assumptions & Key Parameter Values
Loss of Coolant Accident

Containment Leakage Parameters	
Power Level	1530 MWth
Free Volume	1.05E+6 ft^3
Sprayed Fraction	69.4%
Spray Period	185 sec to 5 hr
Mixing Rate	4.84 Unsprayed vol/hr
Containment Leakrate (0–24 hr)	0.1% vol fractions per day
Containment Leakrate (1-30 day)	0.05% vol fractions per day
Maximum DF for Elemental Iodine	200
Sump/Recirculation Spray pH	≥ 7
Fuel Activity Release Fractions	Per Reg. Guide 1.183
Fuel Release Timing (gap)	Onset: 30 sec Duration: 0.5 hr
Fuel Release Timing (Early-In-Vessel)	Onset: 0.5 hr Duration: 1.3 hr
Chemical Form of Iodine released	4.85% elemental 95% particulate 0.15% organic
Spray Removal Constants	Figure 7.2-1
Core Activity	Table 4.1-1
Release Point	Containment Outer Wall
ECCS/SIRWT Leakage Parameters	
Sump Volume (minimum)	314,033 gallons
Combined ESF and SIRWT Leakrate	7,600 cc/hr (2×Tech Spec)
Leakage Period	20.4 min – 30 days
Iodine Release Fraction	0.1
Chemical Form of Iodine Released	97% elemental; 3% organic
Release Point	Auxiliary Building Vent Stack
Containment Vacuum Relief Parameters	
Primary Coolant Tech Spec Activity	Table 4.2-1
Chemical Form of Iodine Released	97% elemental; 3% organic
Containment Vacuum Relief (0-5 sec)	10 scfs
Release Point	Auxiliary Building Vent Stack

CR emergency Ventilation : Initiation Signal/Timing

Initiation time (signal) assumed to be 0 sec (SIAS / CSAS/ CPHS/ PPLS)

TABLE 7.3-1
Analysis Assumptions & Key Parameter Values
Fuel Handling Accident in Fuel Pool Area or Containment

Power Level	1530 MWth
Number of Damaged Fuel Assemblies	1
Total Number of Fuel Assemblies	133
Decay Time Prior to Fuel Movement	72 hours
Radial Peaking Factor	1.8
Fraction of Core Inventory in gap	I-131 (16%) Kr-85 (20%) Other Noble Gases (10%) Other Halides (10%) Alkali Metals (24%)
Equilibrium Core Activity	Table 4.1-1
Iodine Form of gap release before scrubbing	99.85% elemental 0.15% Organic
Scrubbing Decontamination Factors	Elemental Iodine (500) Organic Iodine (1) Noble Gas (1) Particulates (∞)
Rate of Release from Fuel	PUFF
Environmental Release Rate	All airborne activity in a 2-hour period
<u>Environmental Release Points</u>	
Accident in Fuel Pool Area	Auxiliary Building Vent Stack
Accident in Containment	Containment wall

CR Emergency Ventilation: Initiation Signal/Timing
By procedure CR emergency ventilation placed in operation prior to fuel movement.

TABLE 7.5-1
Analysis Assumptions & Key Parameter Values
Heavy Load Drop Accident in Containment

Power Level	1530 MWth
Volume of Containment	1.05E+6 ft^3
Mixing in Containment free Volume	50%
Percentage of Damaged Fuel Assemblies	100%
Total Number of Fuel Assemblies	133
Decay Time Prior to Fuel Movement	72 hours
Fraction of Core Inventory in gap	Noble Gases (5%) Halogens (5%) Alkali Metals (5%)
Equilibrium Core Inventory	Table 4.1-1
Iodine Form of gap release before scrubbing	99.85% elemental 0.15% Organic
Scrubbing Decontamination Factors	
Case 1: (23 ft of water)	Elemental Iodine (500) Organic Iodine (1) Noble Gas (1) Particulates (∞)
Case 2: (11.15 ft of water)	Elemental Iodine (20) Organic Iodine (1) Noble Gas (1) Particulates (∞)
Rate of Release from Fuel	PUFF
Rate of Environmental Release	
Case1: (containment purge rate)	50,000 cfm
Case2: (containment purge rate)	5,000 cfm
Stack Flow (without cont. purge)	72,500 cfm
Environmental Release Point	Auxiliary Building Vent Stack
Release Termination	Purge valve isolated on high radiation signal
Release Termination Time	
Case 1	73 seconds
Case 2	103.5 seconds

CR emergency Ventilation : Initiation Signal/Timing
By procedure CR emergency ventilation placed in operation prior to heavy load movement

TABLE 7.6-1
Analysis Assumptions & Key Parameter Values
Seized Rotor Accident

Power Level	1530 MWth
Minimum Reactor Coolant Mass	264,900 lbm
Primary to Secondary SG tube leakage	1 gpm @ stp
Melted Fuel Percentage	0%
Failed Fuel Percentage	1%
Equilibrium Core Activity	Table 4.1-1
Radial Peaking Factor	1.8
Fraction of Core Inventory in Fuel gap	I-131 (16%) Kr-85 (20%) Other Noble Gases (10%) Other Halides (10%) Alkali Metals (24%)
Iodine Chemical Form in Gap	4.85% elemental 95% CsI 0.15% Organic
Secondary Side Parameters	
Minimum Post-Accident SG Liquid Mass	57,808 lbm per SG
Iodine Species released to Environment	97% elemental; 3% organic
Iodine Partition Coefficient in SGs	100 (all tubes submerged)
Particulate Carry-Over Fraction in SGs	0.0025
Steam Releases per SG	Tables 7.6-2 and 7.6-3
Termination of releases to and from SGs	8 hours
Fraction of Noble Gas Released	1.0 (Released to Environ without holdup)
Environmental Release Point	MSSVs/ADVs

CR emergency Ventilation : Initiation Signal/Timing

Control Room Emergency Ventilation is Initiated by Operator Action: 7 hours after accident

Control Room Emergency Filtered Recirculation Initiation: 9 hours after accident

TABLE 7.6-2
Cumulative Steam Releases (lbm) per Steam Generator
Control Rod Ejection and Seized Rotor Accident: 2-Hour Event

Time (sec)	MSSV Release (lbm)	ADV Release (lbm)
0	0	0
900	47973	0
1800	69242	0
2700	69242	26940
3600	69242	52696
4500	69242	77365
5400	69242	101267
6300	69242	124845
7200	69242	147951

TABLE 7.6-3
Cumulative Steam Releases (lbm) per Steam Generator
Control Rod Ejection and Seized Rotor Accident: 8-Hour Event

Time (sec)	MSSV Release (lbm)	ADV Release (lbm)
0	0	0
900	47973	0
1800	69242	0
2700	69242	22650
3600	69242	44212
4500	69242	64730
5400	69242	84547
6300	69242	104069
7200	69242	123146
8100	69242	141877
9000	69242	160328
9900	69242	178542
10800	69242	196756
11700	69242	214752
12600	69242	232735
13500	69242	250690
14400	69242	268632
15300	69242	286320
16200	69242	304007
17100	69242	321482
18000	69242	338934
18900	69242	356231
19800	69242	373506
20700	69242	390781
21600	69242	407894
22500	69242	425008
23400	69242	442121
24300	69242	459221
25200	69242	476321
26100	69242	493101
27000	69242	509881
27900	69242	526661
28800	69242	543427

TABLE 7.7-1
Analysis Assumptions & Key Parameter Values
Control Rod Ejection Accident

Containment Pathway Parameters	
Power Level	1530 MWth
Free Volume	1.05E+6 ft^3
Containment Leakrate (0 –24 hr)	0.1% vol fractions per day
Containment Leakrate(1-30 day)	0.05% vol fractions per day
Failed Fuel Percentage	10%
Percentage of Core Inventory in Fuel Gap	10% (noble gases & halogens)
Melted Fuel Percentage	1%
Percentage of Core Inventory in melted fuel released to Containment Atmosphere	100% Noble Gas; 25% Halogens
Chemical Form of Iodine in Failed/Melted fuel	4.85% elemental; 95% CsI 0.15% organic
Radial Peaking Factor	1.8
Core Activity Release Timing	PUFF
Form of Failed/Melted Iodine in the Containment Atmosphere	97% elemental; 3% organic
Equilibrium Core Activity	Table 4.1-1
Termination of Containment Release	30 days
Environmental Release Point	Containment wall
Secondary Side Pathway Parameters	
Reactor Coolant Mass	264,900 lbm
Primary-to-Secondary Leakrate	1 gpm @stp
Fraction of Failed/Melted Fuel	Same as Containment Pathway
Percentage of Core Inventory in melted fuel released to Reactor Coolant	100% Noble Gas; 50% Halogens
Iodine Species released to Environment	97% elemental; 3% organic
Iodine Partition Coefficient	100 (all tubes submerged)
Fraction of Noble Gas Released	1.0 (Released to Environ without holdup)
Minimum Post-Accident SG Liquid Mass	57,808 lbm per SG
Steam Releases per SG	Table 7.6-2 nad Table 7.6-3
Termination of Release from SGs	8 hours
Environmental Release Point	MSSVs/ADVs
CR emergency Ventilation : Initiation Signal/Timing	
Initiation time (signal)	38 sec (SIAS)

TABLE 7.8-1
Analysis Assumptions & Key Parameter Values
Main Steam Line Break

Power Level	1530 MWth
Reactor Coolant Mass	264,900 lbm
Leakrate to Affected Steam Generator	1 gpm @ stp
Leakrate to Intact Steam Generator	0 gpm
Failed/Melted Fuel Percentage	0%
RCS Tech Spec Iodine Concentration	Table 4.2-1 (1μCi/gm DE-131)
RCS Tech Spec Noble Gas Concentration	Table 4.2-1 ($100/E_{BAR}$)
RCS Equilibrium Iodine Appearance Rates	Table 4.2-2 (1μCi/gm DE-131)
Pre-Accident Iodine Spike Activity	Table 4.2-2 (60μCi/gm DE-131)
Accident Initiated Spike Appearance Rate	500 times equilibrium
Duration of Accident Initiated Spike	8 hours
Secondary System Release Parameters	
Iodine Species released to Environment	97% elemental; 3% organic
Tech Spec Activity in SG liquid	Table 4.2-1 (0.1μCi/gm DE-131)
Iodine Partition Coefficient in Intact SG	100 (all tubes submerged)
Fraction of Noble Gas Released from Intact SG	1.0 (Released to Environ without holdup)
Fraction of Iodine Released form Faulted SG	1.0 (Released to Environ without holdup)
Fraction of Noble Gas Released from faulted SG	1.0 (Released to Environ without holdup)
Minimum Post-Accident SG Liquid Mass	57,808 lbm (Intact SG only)
Maximum Liquid in each SG	125,707 lbm
Steam Releases from Intact SG	Table 7.8-2 and 7.8-3
Dryout of Affected SG	136 seconds
Secondary Fluid released from Faulted SG	159,346 lbm
Termination of release (1 gpm leak) : Faulted SG	Case 1 (4.94 hours) Case 2 (10.94 hours)
Termination of release from Intact SG	8 hours
Release Point: Faulted SG	Room 81 Domes in the Auxiliary. Bldg.
Release Point : Intact SG	ADVs
CR emergency Ventilation : Initiation Signal/Timing	
Initiation time (signal)	14 sec (SIAS)

TABLE 7.8-2
Integrated Steam Releases (lbm) from Intact Steam Generator
Main Steam Line Break: 2-Hour Event

Time (sec)	ADV Release (lbm)
0	0
1800	0
2700	53880
3600	105392
4500	154730
5400	202533
6300	249690
7200	295901

TABLE 7.8-3
Integrated Steam Releases (lbm) from Intact Steam Generator
Main Steam Line Break: 8-Hour Event

Time (sec)	ADV Release (lbm)
0	0
1800	0
2700	45301
3600	88425
4500	129459
5400	169094
6300	208138
7200	246291
8100	283755
9000	320657
9900	357085
10800	393513
11700	429505
12600	465471
13500	501380
14400	537263
15300	572639
16200	608014
17100	642963
18000	677886
18900	712463
19800	747013
20700	781562
21600	815789
22500	850015
23400	884243
24300	918443
25200	952642
26100	986203
27000	1019761
27900	1053322
28800	1086854

TABLE 7.9-1
Analysis Assumptions & Key Parameter Values
Steam Generator Tube Rupture

Power Level	1530 MWth
Reactor Coolant Mass	264,900 lbm
Break Flow to Affected Steam Generator	Table 7.9-2
Time of Reactor Trip	412 sec
Termination of Release to Affected SG	2 hours
Amount of Break Flow that Flashes	0-15min (15%) 15-60 min (5%) 1 – 2 hr (2%)
Leakrate to Intact Steam Generator	1 gpm @ stp
Failed/Melted Fuel Percentage	0%
RCS Tech Spec Iodine Concentration	Table 4.2-1 (1μCi/gm DE-131)
RCS Tech Spec Noble Gas Concentration	Table 4.2-1 (100/E_{BAR})
RCS Equilibrium Iodine Appearance Rates	Table 4.2-2 (1μCi/gm DE-131)
Pre-Accident Iodine Spike Activity	Table 4.2-2 (60μCi/gm DE-131)
Accident Initiated Spike Appearance Rate	335 times equilibrium
Duration of Accident Initiated Spike	8 hours
Secondary System Release Parameters	
Intact SG Liquid Mass (min)	57,808 lbm
Faulted SG Liquid Mass (min)	77,947 lbm
Initial Mass in Steam Generators	155,894 lbm
Form of All Iodine Released to the Environment via Steam Generators	97% elemental; 3% organic
Iodine Partition Coefficient (unflashed portion)	100 (all tubes submerged)
Fraction of Iodine Released (flashed portion)	1.0 (Released to Environ without holdup)
Fraction of Noble Gas Released from either SG	1.0 (Released to Environ without holdup)
Partition Factor in Condenser AEJ	2000 elemental iodine 1 organic iodine
Steam Flowrate to Condenser	0-412 sec (937 lbs/s per SG)
Steam Releases via MSSV/ADVs	Table 7.9-3, Table 7.9-4 and Table 7.9-5
Termination of Release from SGs	8 hours
Environmental Release Points	0-412 sec (Condenser Evac. Discharge) 412 sec – 8 hr (MSSVs/ADVs)
CR emergency Ventilation : Initiation Signal/Timing	
Initiation time (signal)	426 sec (SIAS)

TABLE 7.9-2
Break Flow from Primary Coolant to Faulted Steam Generator Steam Generator Tube Rupture

Time (sec)	Integrated Break Flow (lb)
0	0
200	12010
500	26260
800	34290
1100	44780
1400	55880
1800	70080
2000	77880
2500	97040
3000	118950
3500	142520
4000	167510
4500	192930
5000	219370
5500	246600
6000	275060
6500	303570
7000	332630
7200	344190

TABLE 7.9-3
Cumulative Steam Releases (lbm) from Faulted & Intact Steam Generator
Steam Generator Tube Rupture: 2-Hour Event

Time (sec)	MSSV Release (lbm)	ADV Release (lbm)
0	0	0
412	0	0
900	26012	0
1800	47281	0
2700	47281	26940
3600	47281	52696
4500	47281	77365
5400	47281	101267
6300	47281	124845
7200	47281	147951

TABLE 7.9-4
Cumulative Steam Releases (lbm) from Faulted Steam Generator
Steam Generator Tube Rupture: 8-Hour Event

Time (sec)	MSSV Release (lbm)	ADV Release (lbm)
0	0	0
412	0	0
900	26012	0
1800	47281	0
2700	47281	22650
3600	47281	44212
4500	47281	64730
5400	47281	84547
6300	47281	104069
7200	47281	123146
28800	47281	123146

TABLE 7.9-5
Cumulative Steam Releases (lbm) from Intact Steam Generator
Steam Generator Tube Rupture: 8-Hour Event

Time (sec)	MSSV Release (lbm)	ADV Release (lbm)
0	0	0
412	0	0
900	26012	0
1800	47281	0
2700	47281	22650
3600	47281	44212
4500	47281	64730
5400	47281	84547
6300	47281	104069
7200	47281	123146
8100	47281	160610
9000	47281	197511
9900	47281	233940
10800	47281	270368
11700	47281	306360
12600	47281	342326
13500	47281	378235
14400	47281	414118
15300	47281	449494
16200	47281	484869
17100	47281	519818
18000	47281	554741
18900	47281	589318
19800	47281	623868
20700	47281	658417
21600	47281	692644
22500	47281	726870
23400	47281	761098
24300	47281	795298
25200	47281	829497
26100	47281	863058
27000	47281	896616
27900	47281	930177
28800	47281	963709

TABLE 7.10-1
Analysis Assumptions & Key Parameter Values
Waste Gas Decay Tank and Liquid Waste Tank Failure

Waste Gas Tank Accident Parameters	
Inventory: (Noble Gas)	Table 7.10-2
Environmental Release	PUFF
Release Point	Auxiliary Building Stack
Liquid Radwaste Tank Accident Parameters	
Inventory:	Table 7.10-2
Environmental Release	PUFF
Release Point	Radwaste Building Exhaust Nozzle

CR emergency Ventilation : Initiation Signal/Timing
Not Credited

TABLE 7.10-2
Liquid Waste Tank and Waste Gas Decay Tank Activity

Nuclide	LWT Activity (Ci)	WGDT Activity (Ci)
KR 83M	7.32E-01	3.87E+01
KR 85M	7.37E-04	1.39E+02
KR 85	2.18E-06	1.11E+04
KR 87	1.25E-04	9.19E+01
KR 88	0	2.62E+02
KR 89	0	7.35E+00
XE131M	2.81E+01	4.06E+02
XE133M	1.13E+01	3.55E+02
XE133	3.88E+02	2.65E+04
XE135M	1.00E+01	7.95E+01
XE135	6.51E+01	9.94E+02
XE137	0	1.66E+01
XE138	0	5.81E+01
BR 83	7.32E-03	
BR 84	7.74E-04	
BR 85	7.37E-06	
BR 87	1.25E-06	
I129	6.54E-06	
I130	1.81E-02	
I131	2.62E+01	
I132	1.18E+00	
I133	3.89E+00	
I134	2.09E-02	
I135	6.51E-01	
I136	5.67E-06	

8 SUMMARY OF RESULTS: CONTROL ROOM / SITE BOUNDARY DOSES

The accidents listed below have been analyzed for dose consequences at the site boundary and control room.

1. Loss of Coolant Accident (LOCA)
2. Fuel Handling Accident in the Fuel Pool (FHA in Fuel Pool)
3. Fuel Handling Accident in the Containment (FHA in Containment)
4. Heavy Load Drop Event (HLD)
5. Seized Rotor Accident (SRA)
6. Control Rod Ejection Accident (CREA)
7. Main Steam Line Break (MSLB)
8. Steam Generator Tube Rupture (SGTR)
9. Gaseous Waste Decay Tank Failure (GWDTF)
10. Liquid Waste Tank Failure – Airborne releases (LWTF)

In accordance with RG 1.183, the "worst 2-hour period" dose at the EAB, and the dose at the LPZ "for the duration of the release" is presented in Table 8.1-1. These dose values represent the post accident dose to the public due to inhalation and submersion for each of these events. Due to distance/plant shielding, the dose contribution at the EAB/LPZ due to direct shine from contained sources is considered negligible for all the accidents. The associated regulatory limit as discussed in Section 2 is also presented.

Per regulatory guidance, the CR dose is integrated over 30 days. The calculated doses address the fact that for events with a duration less than 30 days, the CR dose needs to include the remnant radioactivity within the CR envelope after the event has terminated. Except as noted, the 30-day integrated dose to the control room operator, due to inhalation and submersion, is presented in Table 8.1-2 for all of the referenced design basis accidents.

The CR shielding design is based on the LOCA which represents the worst case DBA relative to radioactivity releases. The dose contribution due to direct shine from post LOCA contained sources/external cloud is identified and included in the CR doses reported for the LOCA.

TABLE 8.1-1
Exclusion Area Boundary and Low Population Doses (TEDE)

Accident	EAB Dose (rem) [2,4,5]	LPZ Dose (rem) [3,5]	SB Reg. Limit (rem)
LOCA	2.50	0.50	25.00
Fuel Handling Accident in Spent Fuel Pool Area	1.50	0.50	6.30
Fuel Handling Accident in Containment	1.50	0.50	6.30
Heavy Load Drop in Containment[1]	3.5 (5.0)	0.5 (0.5)	6.30
Seized Rotor	0.50	0.50	2.50
Control Rod Ejection Accident	2.00	0.50	6.30
Main Steam Line Break	0.50 (PIS) 1.50 (AIS)	0.50 (PIS) 0.50 (GIS)	25 (PIS) 2.5(AIS)
Steam Generator Tube Rupture	1.50 (PIS) 1.50 (AIS)	0.50 (PIS) 0.50 (AIS)	25 (PIS) 2.5 (AIS)
Gas Waste Decay Tank Failure	0.14	0.01	0.50
Liquid Waste Tank Failure	0.08	0.01	0.50

NOTES:

1 Heavy Load Drop calculations performed with 23 feet of water and 11.15 feet of water in the reactor cavity. Results for the 11.15 feet calculation are shown in parenthesis.
2 EAB Doses are based on the worst 2-hour period following the onset of the event.
3 LPZ Doses are based on the duration of the release.
4 Except as noted, the maximum 2 hr dose period for the EAB dose for each of the accidents is the 0 to 2 hrs time period.
- LOCA : 0.5 to 2.5 hr
- CREA : 6 to 8 hr
- SRA: 6 to 8 hr
- MSLB (AIS model) : 8 to 10 hr.

5. Except for the following accidents all doses are rounded up to the nearest 0.5 Rem: WGDTF and LWTF.

TABLE 8.1-2
30 Day Integrated Control Room Doses (TEDE)

Accident	Control Room Operator	
	Dose (rem)[3]	Reg. Limit (rem)
LOCA[2]	4.50 (2.0)	5.00
Fuel Handling Accident in Spent Fuel Pool Area	0.50	5.00
Fuel Handling Accident in Containment	0.50	5.00
Heavy Load Drop in Containment[1]	1.5 (2.0)	5.00
Seized Rotor Accident	4.70	5.00
Control Rod Ejection Accident	3.00	5.00
Main Steam Line Break	2.50	5.00
Steam Generator Tube Rupture	1.50	5.00
Gas Waste Decay Tank Failure	0.04	5.00
Liquid Waste Tank Failure	0.32	5.00

NOTES:

1 Heavy Load Drop calculations performed with 23 feet of water and 11.15 feet of water in the reactor cavity. Results for the 11.15 feet calculation is shown in parenthesis.

2 Portion shown in parenthesis for the LOCA represents that portion of the total dose of 4.5 Rem that is the contribution of direct shine from contained sources/external cloud.

3 Except for the following accidents all doses are rounded up to the nearest 0.5 Rem: SRA, WGDTF and LWTF.

9 CONCLUSIONS

The alternative source term as defined in Regulatory Guide 1.183 has been incorporated into the FCS site boundary and control room dose re-analyses discussed herein. The estimated FCS dose consequences for all design basis events addressed in RG 1.183, meet the acceptance criteria specified in 10CFR50.67 and RG 1.183. This represents a full implementation of the alternative source terms in which the RG 1.183 source term will become the licensing basis for FCS.

10 REFERENCES

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Attachment F

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