S & W Radiological Consequences Evaluation
(Non Proprietary)

Implementation of GSI-191 Containment Sump Modification

Note: In this attachment, some of the titles describing components and parameters are different than the titles given in the remainder of the LAR. The following table defines the equivalence of titles.

Component/Parameter Title in S & W Radiological Evaluation	**Component/Parameter Title In OPPD's LAR**
Containment Recirculation Fan Coolers (CRFC)	Containment Air Cooling and Filtering System (CACFS)
Containment Vacuum Relief Line	Containment Pressure Relief Line
CRFC filters	CACF filters or the HEPA filters
CRFC filtered flow	CACF flow
CRFC unfiltered flow	CAC flow

IMPLEMENTATION OF GSI-191
CONTAINMENT SUMP MODIFICATION

SITE BOUNDARY & CONTROL ROOM
DOSE CONSEQUENCES

LOSS-OF-COOLANT ACCIDENT

FORT CALHOUN STATION

Prepared for

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1 INTRODUCTION

Amendment No. 201 to the FCS Operating license replaced the radiological source term used in the design basis site boundary and control room dose consequence analyses with Alternative Source Terms (AST) pursuant to 10CFR50.67, SRP 15.0.1, and Regulatory Guide 1.183. (References 1, 2, 3 and 4)

This chapter addresses the potential impact on the estimated dose consequences at the Exclusion Area Boundary (EAB), Low Population Zone (LPZ), Control Room (CR) and Technical Support Center (TSC) due to plant modifications incorporated as part of the resolution to GSI-191 and associated sump water management initiative.

Design changes associated with the above modification that will potentially impact the reported dose consequences of the FCS Loss of Coolant Accident (LOCA) are listed below:

- Elimination of the use of containment sprays
- Utilization of the HEPA filters in the Containment Recirculation Fan Coolers (CRFC) for post-LOCA fission product removal

The radiological consequences of the remaining design basis accidents are not impacted by this application.

2 REGULATORY APPROACH

2.1.1 Exceptions to Regulatory Guide 1.183

The LOCA dose consequence analysis summarized herein continues to reflect the following licensing basis exceptions to Regulatory Guide 1.183:

- Utilizes the "traditional" breathing rates which had been noted in DG 1081 (Draft Guide to Regulatory Guide1.183). The impact on the dose analyses due to usage of the "traditional" breathing rates, (instead of those noted in Regulatory Guide 1.183 which essentially represent the "rounded up" values for the "traditional" breathing rates), is negligible.

- 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 when developing the dose consequences in the Technical Support Center. The dose impact of a LOOP on the initiation of the TSC ventilation system is not included since the TSC ventilation system is manually initiated and is effective 30 minutes after the accident. Per NRC Information Notice 93-17 (Reference 5), 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 5 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 5 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 (the quoted example is a LOOP that occurs 2 minutes after the LOCA) is of very low probability and is not a concern. Based on the above it can be concluded that the probability of a "related" LOOP occurring 30 minutes after the event is very low and therefore need not be addressed.

2.1.2 Changes to Current Licensing Basis

Provided below is the summary of changes to the current licensing basis proposed by this application.

- The current licensing basis LOCA dose consequence analysis credits containment sprays to remove airborne activity in containment. No credit is taken for elemental iodine plateout on wetted surfaces.

  The updated analysis assumes that sprays are not initiated following LOCA. Instead the HEPA filters associated with the Containment Recirculation Fan Coolers (CRFCs) are credited for particulate fission product removal, and plateout on the wetted surfaces is credited for lowering the elemental halogen airborne activity concentrations in containment.

2.2 Dose Acceptance Criteria

In accordance with current licensing basis, the acceptance criteria for the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) doses for the LOCA are based on 10CFR Part 50 § 50.67 and Section 4.4 Table 6 of Regulatory Guide 1.183 (also noted in Table 1 of SRP 15.0.1):

(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, would not receive a radiation dose in excess of 25 rem TEDE

(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), would not receive a radiation dose in excess of 25 rem TEDE

The acceptance criterion for the Control Room Dose is based on 10CFR 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.

The acceptance criterion for the TSC Dose is based on:

NUREG 0737, Supplement 1, § 8.2.1.f (Reference 16) which requires that plant design "assure that radiation exposure to any person working in the TSC would not exceed 5 rem whole body or its dose equivalent to any organ". The 5 Rem whole body (or its dose equivalent to any organ) guideline is the same guideline specified for a Control Room operator in 10 CFR 50, Appendix A, GDC19. (Reference 17) Since 10 CFR 50 § 50.67 supersedes GDC 19 when licensing a facility using alternative source terms for evaluating design basis accidents, the updated guideline for the control room in 10 CFR 50 § 50.67 is applied to the TSC for purposes of demonstrating habitability.

3 COMPUTER CODES

The QA Category 1 computer codes utilized in the dose consequence analyses that support this application are listed below. The referenced computer codes have been used extensively to support nuclear power plant design and are a part of FCS current licensing basis:

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, V04, L03).

2. S&W Proprietary Computer Program ACTIVITY2, "Fission Products in a Nuclear Reactor", NU-014, V01, L04.

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

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

5. S&W Proprietary Computer Code, PERC2, "Passive Evolutionary Regulatory Consequence Code", NU-226, V00, L02.

6. 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

This modification does not impact the isotopic core inventory utilized to determine dose consequences.

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 of 1530 MWt including a 2% instrument error per Regulatory Guide 1.49 (Reference 6), and current licensed values of fuel enrichment and burnup.

The methodology used to develop the licensing basis core inventory, and the associated isotopic listing, was presented in Section 4.1 of Attachment E of Reference 7, and approved by NRC via Reference 4.

The isotopic listing has since been updated to reflect minor changes to design input. The changes were determined to be minimal with insignificant impact on the design basis dose consequence analyses.

The updated design basis core activity of isotopes significant to dose consequences is used in this evaluation and is presented herein in Table 4.1-1.

4.2 Coolant Inventory

This modification does not impact the Technical Specification primary coolant concentrations utilized to determine dose consequences.

The methodology used to develop the Technical Specification primary coolant concentrations, and the associated isotopic listing, was presented in Section 4.2 of Attachment E of Reference 7, and approved by NRC via Reference 4.

The isotopic listing has since been updated to reflect minor changes to design input. The changes were determined to be minimal with insignificant impact on the design basis dose consequence analyses.

The updated noble gas and halogen primary coolant Technical Specification Activity Concentrations is used in this evaluation and is presented herein in Table 4.2-1.

5 ACCIDENT ATMOSPHERIC DISPERSION FACTORS (X/Q)

5.1 Site Boundary Atmospheric Dispersion Factors

The Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) atmospheric dispersion factors (χ/Q) for FCS remain unchanged by this application and are consistent with current licensing basis. The highest EAB & LPZ χ/Q values from among all 22.5° downwind sectors for each release/receptor combination and accident period applicable to the LOCA are summarized in Table 5.1-1. The 0.5% sector dependent χ/Q values are presented with parenthesis indicating worst case downwind sector.

5.2 Control Room Atmospheric Dispersion Factors

The control room atmospheric dispersion factors (χ/Q) for FCS remain unchanged by this application and are consistent with current licensing basis. The χ/Q values for all release-receptor combinations applicable to the LOCA are summarized in Table 5.2-1.

5.3 Technical Support Center Atmospheric Dispersion Factors

The TSC atmospheric dispersion factors (χ/Q) for FCS remain unchanged by this application and are consistent with current licensing basis. The χ/Q values for all release-receptor combinations applicable to the LOCA are summarized in Table 5.3-1.

6 DOSE CALCULATION METHODOLOGY

The dose calculation methodology is similar to that outlined in Section 6 of Attachment E of Reference 7 and is current licensing basis. As noted in Reference 7, 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/TSC air region, and then calculates the cumulative doses as described below:

<u>Committed Effective Dose Equivalent (CEDE) Inhalation Dose</u> - The dose conversion factors by isotope are applied to the activity in the air space of the control room/TSC, or at the EAB/LPZ. The exposure is adjusted by the appropriate respiration rate and occupancy factors for the CR/TSC 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 10)
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 CR/TSC 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/TSC. 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/TSC, and the ANSI/ANS 6.1.1-1991 "neutron and gamma-ray flux-to-dose-rate factors" (Reference 11).

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 12) where 1 rad is assumed 1 rem.

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

$E\gamma_{BAR}$	=	average gamma released per disintegration (Mev/dis)
$\psi(x,y,0)$	=	concentration time integral (Ci-sec/m^3)
0.25	=	$[\,1.11 \times 1.6x10^{-6} \times 3.7x10^{10}\,] / [\,1293 \times 100 \times 2\,]$

where:

1.11	= ratio of electron densities per gm of tissue to per gm of air
$1.6x10^{-6}$ (erg/Mev)	= number of ergs per Mev
$3.7x10^{10}$ (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

The design modifications proposed by this application will potentially impact the reported dose consequences of the FCS LOCA. The radiological consequences of the remaining design basis accidents are not impacted by this application.

7.1 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.

Except as noted in Section 2, this assessment follows the guidance provided in RG 1.183 for the LOCA. Table 7.1-1 lists some of the key assumptions / parameters utilized to develop the radiological consequences following a LOCA.

Doses due to Submersion and Inhalation

In accordance with current licensing basis, S&W proprietary computer program, PERC2, is used to calculate the site boundary, control room and TSC 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

In accordance with current licensing basis, 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) via the Auxiliary Building Vent Stack. 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.

In accordance with Regulatory Guide 1.183 and current licensing basis, 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: (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 are 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.

In accordance with Reference 2 and current licensing basis, 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.

The activity transport model takes credit for aerosol removal by the HEPA filters in the containment recirculation fan coolers (CRFCs), and credits plateout on the surface of the water in the sump for elemental iodine removal. The maximum DF for elemental iodine is based on SRP 6.5.2 (Reference 8) and is limited to a DF of 200; there is no regulatory based maximum DF for aerosol removal via use of HEPA filters.

<u>CRFC HEPA Filter Efficiency</u>

The existing CRFC HEPA filter/filter housing configuration at FCS does not support the use of dioctyl-phthalate (DOP) testing for bypass leakage. Thus, the existing HEPA installation and maintenance inspection procedures, in conjunction with additional requirements relative to filter replacement, and a calculation that estimates the maximum potential bypass leakage following a DBA (with a conservative safety factor of 2 applied to bound filter performance) is utilized to support the 50% filter efficiency used in the dose consequence analysis.

Examination of the equipment drawings, fan performance curves and existing maintenance and surveillance procedures indicate that the potential for bypass leakage would be primarily dominated by leakage through cracks in the gasketing material or through holes in the filter media itself. To establish the potential gap due to gasket deterioration, consideration is given to the gasket compression associated with the clamping mechanisms. Estimates of maximum bypass leakage due to gasket deterioration is made assuming maximum credible failures of the gasket (to establish worst case crack leakage), and the equivalent size of a hole (orifice) in filter media is calculated to confirm that a hole of this size in the filter media can be detected by visual inspection, since visual identification of damage to the filter media currently triggers a replacement of the individual filter elements. New replacement requirements for the entire HEPA filter train include a) a pressure drop of 2 inches w.c. or projected to exceed a 2 inch w.c. pressure drop during the next operating cycle, and b) a limit of a maximum of 10 years service life.

<u>Elemental Iodine Plateout</u>

SRP 6.5.2, which allows elemental iodine plateout on wetted surfaces, requires either condensation or sprays to provide "new or fresh" surfaces for absorption of elemental iodine as the liquid flows down the walls. Since with a "no spray scenario", neither condensation, nor

sprays can be demonstrated on the walls, the surface of the pool water is utilized as the renewable wet surface. The issue of surface renewal for the pool water has to be addressed only upto the time a sump water pH of 7 or greater can be established since, the neutral or basic pH would drive the chemical equilibrium to have very little elemental iodine present in solution or at the surface, hence pool deposition would occur unhindered. Prior to that time frame, credit for surface renewal is taken based on the following:

- The ECCS injection water (single phase) exiting the break location is at a lower temperature than the pool water (denser) and will enter the pool as a stream and thus entrain the surface pool water setting up convection and momentum driven currents that would promote surface renewal.
- Structurally the containment is set up to direct water to the sump thereby avoiding water holdups and isolated water sections. This works to extend the area of influence for the convective and momentum driven flow even though obstructions exist.

To conservatively address the issue of surface renewal across the pool surface, a reduced surface area (half the calculated value) is utilized until the recirculation phase for core injection is initiated.

Iodine re-evolution

Long-term retention of iodine in sump liquids is strongly dependent on the sump pH. Per SRP 6.5.2, II.1.g, relative to iodine re-evolution, "long term iodine retention maybe assumed only when the equilibrium sump pH after mixing and dilution with the primary coolant and ECCS injection is above 7". Sodium tetraborate 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 sodium tetraborate. The analysis does not address iodine re-evolution as the ultimate sump pH of $\geq$ 7 is achieved in accordance with timing requirements of NUREG/CR-5732.

Containment Mixing

Mixing between regions above and below the operating floor is assumed for the duration of the accident and is based on the minimum CRFC flow rate, and the maximum containment free volumes above and below the operating floor. As noted previously in Reference 7, at FCS, the CRFC intake locations are above the operating floor, and only 23% of the total flow is directed below the operating level.

Radioactivity is assumed to leak from the area below and above the operating floor 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 twenty-nine 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. Consistent with current licensing basis, 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: In accordance with current licensing basis, 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 postulated to be at 20.4 minutes. Recirculation starts when the SWIRT inventory becomes low. The initiation time listed is conservative since it is based on maximum ESF and continues to assume initiation of the containment spray system (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 and the TSC intakes, i.e., the Auxiliary Building Vent Stack and the Auxiliary Building Fresh Air Intake, respectively. 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: In accordance with current licensing basis, 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 and the TSC intakes, i.e., the Auxiliary Building Vent Stack and the Auxiliary Building Fresh Air Intake, respectively. 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).

7.2 Control Room Design / Operation / Transport Model

The FCS control room design and operation outlined in Reference 7 and accepted by NRC via Reference 4 remains unchanged by this application and is summarized below.

The FCS 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 location which services both the normal and accident mode of operation. 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 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 13). 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 14), 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 15), 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.

A 10% margin is applied on all CR ventilation flows. Table 7.2-1 lists key assumptions / parameters associated with FCS control room design.

Accident Specific Control Room Model Assumptions – Inhalation and Submersion

The LOCA specific control room model assumptions outlined in Reference 7 and accepted by NRC via Reference 4 remains unchanged by this application and is summarized below.

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.

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

The post-LOCA control room direct shine dose model assumptions outlined in Reference 7 and accepted by NRC via Reference 4 remains unchanged by this application and is summarized below.

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 CRFC filters.

In accordance with current licensing basis, 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.

7.3 TSC Design / Operation / Transport Model

In accordance with current licensing basis, the FCS TSC is modeled as a single region. Isotopic concentrations in areas outside the TSC envelope are assumed to be comparable to the isotopic concentrations at the TSC intake location which services both the normal and accident mode of operation. The TSC is designed to operate at 1/8 w.g during accident mode. During normal operation the unfiltered fresh air intake rate is 8500 cfm. The TSC is in filtered ventilation and recirculation mode (via manual operator action) by 30 minutes after LOCA. As discussed in Section 2, a LOOP is not considered for the TSC dose analysis. In the post-accident mode, the TSC has 1000 cfm filtered intake, 2000 cfm filtered recirculation and 150 cfm unfiltered inleakage. The efficiency of the accident intake and recirculation filters is 95% for particulates, elemental iodine, and organic iodine. The TSC is equipped with double vestibule doors; therefore, per SPP 6.4, there is no unfiltered inleakage due to egress/ingress. A 10% margin is applied on all TSC ventilation flows. Table 7.3-1 lists key assumptions / parameters associated with the TSC design.

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

The dose contribution in the TSC 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, control room HVAC filters, and the CRFC filters.

The maximum TSC operator dose following a LOCA is presented in Section 8.

Site Boundary Dose Assessment

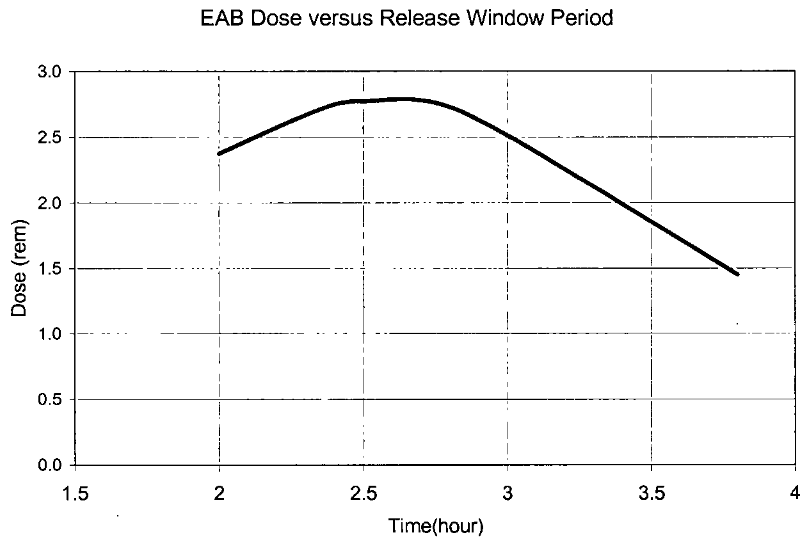
In accordance with current licensing basis, to find the "worst-case 2-hour release window", several assessments are made with the environmental release beginning at 0, 20, 30, 50 and 108 minutes after a LOCA utilizing the full complement of nuclides. These start points for the "worst-case 2-hour window" 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 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.

8 SUMMARY OF RESULTS: CONTROL ROOM / TSC / SITE BOUNDARY DOSES

The estimated doses to the public and to the plant operator presented in this section reflect the design changes associated with this modification that will potentially impact the reported dose consequences of a Loss of Coolant Accident, i.e.,

- Elimination of the use of containment sprays
- Utilization of the HEPA filters in the Containment Recirculation Fan Coolers for post-LOCA fission product removal

The radiological consequences of the remaining design basis accidents are not impacted by this application.

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. These dose values represent the post accident dose to the public due to inhalation and submersion. Due to distance/plant shielding, the dose contribution at the EAB/LPZ due to direct shine from contained sources is considered negligible. The associated regulatory limit as discussed in Section 2 is also presented.

Per regulatory guidance, the CR and TSC dose is integrated over 30 days and presented in Table 8-2. The values reported include the dose contribution due to direct shine from post LOCA contained sources and the external cloud.

9 CONCLUSIONS

The radiological analyses and evaluations developed in support of this application and summarized herein demonstrate that the plant modifications incorporated in support of GSI implementation relative to minimization of sump debris transport and sump screen blockage and the associated sump water management initiative will not impact compliance with applicable regulatory requirements.

The LOCA has been evaluated using the guidance provided in Regulatory Guide 1.183. As demonstrated in Tables 8-1 ad 8-2, the estimated dose consequences at the EAB, LPZ, Control Room remain within the acceptance criteria of 10CFR50.67 as supplemented by Regulatory Guide 1.183 and SRP 15.0.1. In addition, as noted in Table 8-2, the assessment also demonstrates that the dose consequences in the TSC remain compliant with regulatory guidance provided in Supplement 1 of NUREG-0737.

10 REFERENCES

1. 10CFR50.67, "Accident Source Term".

2. Regulatory Guide 1.183, Revision 0, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors", July 2000.

3. NUREG-0800, Standard Review Plan Section 15.0.1, "Radiological Consequence Analyses using Alternative Source Terms," Revision 0.

4. NRC Letter dated December 5, 2001, Issuing Amendment No. 201 to the Facility Operating License No. DPR-40 for the Fort Calhoun Station Unit No. 1.

5. NRC Information notice 93-17, Revision 1, "Safety Systems Response to Loss of Coolant and Loss of Offsite Power" March 25, 1994 (original issue March 8, 1993).

6. Regulatory Guide 1.49, Revision 1, "Power Levels of Nuclear Power Plants".

7. OPPD (W. Gates) to NRC (Document Control Desk), dated February 7, 2001 (LIC-01-0010).

8. NUREG 0800, Standard Review Plan Section 6.5.2, "Containment Spray as a Fission Product Cleanup System", Revision 2.

9. NUREG/CR-5732, "Iodine Chemical Forms in LWR Severe Accidents – Final Report," April 1992.

10. EPA-520/1-88-020, September 1988, Federal Guidance Report No.11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion".

11. ANSI/ANS 6.1.1-1991, "Neutron and Gamma-ray Fluence-to-dose Factors".

12. TID-24190, Air Resources Laboratories, "Meteorology and Atomic Energy", July 1968.

13. FCS Engineering Analysis EA-FC-03-040, Revision 0, "Tracer Gas Testing of the Control Room Envelope".

14. NUREG 0800, Standard Review Plan Section 6.4, "Control Room Habitability System" Revision 2.

15. NUREG 0800, Standard Review Plan Section 6.4, Appendix A, "Acceptance Criteria for Valve or Damper Repair Alternative" Revision 2.

16. NUREG-0737, Supplement 1, "Clarification of TMI Action Plan Requirements – Requirements for Emergency Response Capability," December 17, 1982.

17. 10CFR50 Appendix A, GDC 19, "Control Room".

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

ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)	ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)
AG-110			6.84E+06	PU-240			2.35E+04
	PARENT	AG-110M	1.71E+05		PARENT	NP-240	1.71E+06
AG-110M			1.71E+05	PU-241			6.06E+06
AG-111			2.51E+06		PARENT	CM-245	1.60E+05
	PARENT	AG-111M	2.52E+06		GRAND PARENT	AM-245	2.83E-05
	GRAND PARENT	PD-111	2.51E+06		2ND PARENT	CF-249	7.58E-04
AG-112			1.15E+06	PU-242			9.98E+01
	PARENT	PD-112	1.14E+06		PARENT	AM-242	3.48E+06
AM-241			7.28E+03		GRAND PARENT	AM-242M	4.70E+02
	PARENT	PU-241	6.06E+06	RB-86			6.93E+04
	2ND PARENT	CM-241	8.06E-01		PARENT	RB-86M	5.71E+03
AS-76			8.34E+02	RB-88			3.25E+07
BA-137M			5.01E+06		PARENT	KR-88	3.17E+07
	PARENT	CS-137	5.27E+06		GRAND PARENT	BR-88	1.76E+07
	GRAND PARENT	XE-137	7.69E+07	RB-89			4.26E+07
BA-139			7.55E+07		PARENT	KR-89	3.98E+07
	PARENT	CS-139	7.39E+07		GRAND PARENT	BR-89	1.21E+07
	GRAND PARENT	XE-139	5.55E+07	RB-90			3.95E+07
BA-140			7.56E+07		PARENT	KR-90	4.28E+07
	PARENT	CS-140	6.65E+07		GRAND PARENT	BR-90	6.49E+06
	GRAND PARENT	XE-140	3.91E+07		2ND PARENT	RB-90M	1.21E+07
BA-142			6.57E+07	RB-90M			1.21E+07
	PARENT	CS-142	2.98E+07		PARENT	KR-90	4.28E+07
	GRAND PARENT	XE-142	5.77E+06		GRAND PARENT	BR-90	6.49E+06
BR-82			1.26E+05	RH-103M			6.52E+07
	PARENT	BR-82M	1.08E+05		PARENT	RU-103	6.53E+07
BR-83			5.31E+06		GRAND PARENT	TC-103	6.53E+07
	PARENT	SE-83M	2.68E+06	RH-105			4.19E+07
	2ND PARENT	SE-83	2.47E+06		PARENT	RH-105M	1.28E+07
BR-85			1.12E+07		GRAND PARENT	RU-105	4.52E+07
	PARENT	SE-85	4.66E+06		2ND PARENT	RU-105	4.52E+07
CD-115			3.41E+05	RH-105M			1.28E+07
	PARENT	AG-115	2.39E+05		PARENT	RU-105	4.52E+07
	GRAND PARENT	PD-115	3.01E+05		GRAND PARENT	TC-105	4.46E+07
	2ND PARENT	AG-115M	9.97E+04	RH-106			2.59E+07
CD-115M			1.58E+04		PARENT	RU-106	2.36E+07
	PARENT	AG-115	2.39E+05		GRAND PARENT	TC-106	3.21E+07
	GRAND PARENT	PD-115	3.01E+05	RN-220			1.58E-01
CE-141			6.97E+07		PARENT	RA-224	1.58E-01
	PARENT	LA-141	6.91E+07		GRAND PARENT	TH-228	1.57E-01
	GRAND PARENT	BA-141	6.85E+07	RU-103			6.53E+07
CE-143			6.57E+07		PARENT	TC-103	6.53E+07
	PARENT	LA-143	6.52E+07		GRAND PARENT	MO-103	6.42E+07

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

ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)	ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)
	GRAND PARENT	BA-143	5.70E+07	RU-106			2.36E+07
CE-144			5.38E+07		PARENT	TC-106	3.21E+07
	PARENT	LA-144	5.81E+07		GRAND PARENT	MO-106	2.13E+07
	GRAND PARENT	BA-144	4.51E+07	SB-122			4.03E+04
CM-242			2.06E+06		PARENT	SB-122M	4.03E+03
	PARENT	AM-242	3.48E+06	SB-124			3.13E+04
	GRAND PARENT	AM-242M	4.70E+02		PARENT	SB-124M	6.80E+02
CM-244			1.99E+05	SB-125			3.62E+05
	PARENT	AM-244	6.18E+06		PARENT	SN-125	2.05E+05
CS-132			1.56E+03		GRAND PARENT	IN-125	3.39E+05
CS-134			7.02E+06		2ND PARENT	SN-125M	6.24E+05
	PARENT	CS-134M	1.59E+06	SB-127			3.58E+06
CS-134M			1.59E+06		PARENT	SN-127	1.44E+06
CS-135M			1.64E+06		GRAND PARENT	IN-127	7.32E+05
CS-136			2.19E+06		2ND PARENT	SN-127M	1.95E+06
CS-137			5.27E+06	SB-129			1.32E+07
	PARENT	XE-137	7.69E+07		PARENT	SN-129	5.16E+06
	GRAND PARENT	I-137	3.97E+07		GRAND PARENT	IN-129	1.47E+06
CS-138			7.90E+07		2ND PARENT	SN-129M	4.93E+06
	PARENT	XE-138	7.33E+07	SB-130			4.39E+06
	GRAND PARENT	I-138	2.00E+07	SB-130M			1.84E+07
CS-139			7.39E+07		PARENT	SN-130	1.39E+07
	PARENT	XE-139	5.55E+07	SB-131			3.24E+07
	GRAND PARENT	I-139	1.02E+07		PARENT	SN-131	1.18E+07
CS-140			6.65E+07		GRAND PARENT	IN-131	4.60E+05
	PARENT	XE-140	3.91E+07	SB-132			1.92E+07
	GRAND PARENT	I-140	2.57E+06		PARENT	SN-132	9.48E+06
EU-154			3.06E+05		GRAND PARENT	IN-132	1.22E+05
EU-155			1.33E+05	SB-132M			1.87E+07
	PARENT	SM-155	1.55E+06	SB-133			2.74E+07
EU-156			9.60E+06		PARENT	SN-133	2.59E+06
	PARENT	SM-156	9.74E+05	SE-83			2.47E+06
EU-157			1.02E+06		PARENT	AS-83	3.35E+06
	PARENT	SM-157	6.12E+05		GRAND PARENT	GE-83	5.67E+05
EU-158			3.58E+05	SM-153			1.84E+07
EU-159			1.81E+05		PARENT	PM-153	3.79E+06
GA-72			6.81E+02	SN-121			3.34E+05
	PARENT	ZN-72	6.79E+02		PARENT	IN-121M	3.11E+05
GD-159			2.45E+05		GRAND PARENT	CD-121	3.05E+05
	PARENT	EU-159	1.81E+05		2ND PARENT	IN-121	2.95E+04
GE-77			2.89E+04	SN-123			2.63E+04
	PARENT	GE-77M	7.77E+04		PARENT	IN-123	2.71E+05
	GRAND PARENT	GA-77	7.58E+04	SN-125			2.05E+05
	2ND PARENT	GA-77	7.58E+04		PARENT	IN-125	3.39E+05

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

ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)	ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)
HO-166			3.40E+03	SN-127			1.44E+06
	PARENT	DY-166	1.47E+02		PARENT	IN-127	7.32E+05
I-129			1.54E+00	SR-89			4.42E+07
	PARENT	TE-129	1.26E+07		PARENT	RB-89	4.26E+07
	GRAND PARENT	TE-129M	2.54E+06		GRAND PARENT	KR-89	3.98E+07
	2ND PARENT	TE-129M	2.54E+06	SR-90			4.12E+06
I-130			9.28E+05		PARENT	RB-90	3.95E+07
	PARENT	I-130M	4.93E+05		GRAND PARENT	KR-90	4.28E+07
I-131			4.08E+07		2ND PARENT	RB-90M	1.21E+07
	PARENT	TE-131	3.44E+07	SR-91			5.48E+07
	GRAND PARENT	TE-131M	8.18E+06		PARENT	RB-91	5.14E+07
	2ND PARENT	TE-131M	8.18E+06		GRAND PARENT	KR-91	2.94E+07
I-132			5.97E+07	SR-92			5.74E+07
	PARENT	TE-132	5.87E+07		PARENT	RB-92	4.51E+07
	GRAND PARENT	SB-132	1.92E+07		GRAND PARENT	KR-92	1.56E+07
I-133			8.45E+07	SR-93			6.38E+07
	PARENT	TE-133	4.63E+07		PARENT	RB-93	3.65E+07
	GRAND PARENT	SB-133	2.74E+07		GRAND PARENT	KR-93	5.22E+06
	2ND PARENT	TE-133M	3.81E+07	SR-94			6.31E+07
I-134			9.44E+07		PARENT	RB-94	1.88E+07
	PARENT	TE-134	7.69E+07		GRAND PARENT	KR-94	2.37E+06
	GRAND PARENT	SB-134	5.06E+06	TB-160			4.12E+04
	2ND PARENT	I-134M	8.26E+06	TC-99M			6.80E+07
I-135			8.02E+07		PARENT	MO-99	7.69E+07
	PARENT	TE-135	4.07E+07		GRAND PARENT	NB-99	4.51E+07
	GRAND PARENT	SB-135	2.26E+06	TC-101			6.95E+07
I-136			3.76E+07		PARENT	MO-101	6.95E+07
	PARENT	TE-136	1.85E+07		GRAND PARENT	NB-101	6.60E+07
	GRAND PARENT	SB-136	3.56E+05	TC-104			5.38E+07
IN-115M			3.41E+05		PARENT	MO-104	5.12E+07
	PARENT	CD-115	3.41E+05		GRAND PARENT	NB-104	1.96E+07
KR-83M			5.34E+06	TC-105			4.46E+07
	PARENT	BR-83	5.31E+06		PARENT	MO-105	3.76E+07
	GRAND PARENT	SE-83M	2.68E+06	TE-127			3.53E+06
KR-85			4.69E+05		PARENT	TE-127M	5.87E+05
	PARENT	KR-85M	1.13E+07		GRAND PARENT	SB-127	3.58E+06
	GRAND PARENT	BR-85	1.12E+07		2ND PARENT	SB-127	3.58E+06
	2ND PARENT	BR-85	1.12E+07	TE-127M			5.87E+05
KR-85M			1.13E+07		PARENT	SB-127	3.58E+06
	PARENT	BR-85	1.12E+07		GRAND PARENT	SN-127	1.44E+06
	GRAND PARENT	SE-85	4.66E+06	TE-129			1.26E+07
KR-87			2.27E+07		PARENT	TE-129M	2.54E+06
	PARENT	BR-87	1.80E+07		GRAND PARENT	SB-129	1.32E+07
	GRAND PARENT	SE-87	6.65E+06		2ND PARENT	SB-129	1.32E+07

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

ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)	ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)
KR-88			3.17E+07	TE-129M			2.54E+06
	PARENT	BR-88	1.76E+07		PARENT	SB-129	1.32E+07
	GRAND PARENT	SE-88	3.49E+06		GRAND PARENT	SN-129	5.16E+06
KR-89			3.98E+07	TE-131			3.44E+07
	PARENT	BR-89	1.21E+07		PARENT	SB-131	3.24E+07
	GRAND PARENT	SE-89	1.22E+06		GRAND PARENT	SN-131	1.18E+07
KR-90			4.28E+07		2ND PARENT	TE-131M	8.18E+06
	PARENT	BR-90	6.49E+06	TE-131M			8.18E+06
LA-140			7.75E+07		PARENT	SB-131	3.24E+07
	PARENT	BA-140	7.56E+07		GRAND PARENT	SN-131	1.18E+07
	GRAND PARENT	CS-140	6.65E+07	TE-132			5.87E+07
LA-141			6.91E+07		PARENT	SB-132	1.92E+07
	PARENT	BA-141	6.85E+07		GRAND PARENT	SN-132	9.48E+06
	GRAND PARENT	CS-141	5.09E+07	TE-133			4.63E+07
LA-142			6.78E+07		PARENT	TE-133M	3.81E+07
	PARENT	BA-142	6.57E+07		GRAND PARENT	SB-133	2.74E+07
	GRAND PARENT	CS-142	2.98E+07		2ND PARENT	SB-133	2.74E+07
LA-143			6.52E+07	TE-133M			3.81E+07
	PARENT	BA-143	5.70E+07		PARENT	SB-133	2.74E+07
	GRAND PARENT	CS-143	1.54E+07		GRAND PARENT	SN-133	2.59E+06
MO-99			7.69E+07	TE-134			7.69E+07
	PARENT	NB-99M	3.07E+07		PARENT	SB-134	5.06E+06
	GRAND PARENT	ZR-99	6.97E+07		GRAND PARENT	SN-134	4.42E+05
	2ND PARENT	NB-99	4.51E+07	TH-228			1.57E-01
MO-101			6.95E+07	XE-131M			5.44E+05
	PARENT	NB-101	6.60E+07		PARENT	I-131	4.08E+07
	GRAND PARENT	ZR-101	3.96E+07		GRAND PARENT	TE-131M	8.18E+06
NB-95			7.28E+07	XE-133			8.46E+07
	PARENT	ZR-95	7.25E+07		PARENT	I-133	8.45E+07
	GRAND PARENT	Y-95	7.01E+07		GRAND PARENT	TE-133M	3.81E+07
	2ND PARENT	NB-95M	8.29E+05		2ND PARENT	XE-133M	2.65E+06
NB-95M			8.29E+05	XE-133M			2.65E+06
	PARENT	ZR-95	7.25E+07		PARENT	I-133	8.45E+07
	GRAND PARENT	Y-95	7.01E+07		GRAND PARENT	TE-133M	3.81E+07
NB-97			6.79E+07	XE-135			3.15E+07
	PARENT	NB-97M	6.40E+07		PARENT	I-135	8.02E+07
	GRAND PARENT	ZR-97	6.74E+07		GRAND PARENT	TE-135	4.07E+07
	2ND PARENT	ZR-97	6.74E+07		2ND PARENT	XE-135M	1.76E+07
NB-97M			6.40E+07	XE-135M			1.76E+07
	PARENT	ZR-97	6.74E+07		PARENT	I-135	8.02E+07
	GRAND PARENT	Y-97	5.52E+07		GRAND PARENT	TE-135	4.07E+07
ND-147			2.78E+07	XE-137			7.69E+07
	PARENT	PR-147	2.76E+07		PARENT	I-137	3.97E+07
	GRAND PARENT	CE-147	2.62E+07		GRAND PARENT	TE-137	6.13E+06

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

ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)	ISOTOPE	PARENT RELATIONSHIP	PARENT ISOTOPE	ACTIVITY (CURIES)
NP-239			8.64E+08	XE-138			7.33E+07
	PARENT	AM-243	1.25E+03		PARENT	I-138	2.00E+07
	GRAND PARENT	PU-243	1.64E+07		GRAND PARENT	TE-138	1.50E+06
	2ND PARENT	U-239	8.66E+08	Y-90			4.24E+06
PD-109			1.57E+07		PARENT	SR-90	4.12E+06
	PARENT	RH-109	1.31E+07		GRAND PARENT	RB-90	3.95E+07
	GRAND PARENT	RU-109	1.14E+07		2ND PARENT	Y-90M	2.11E+02
	2ND PARENT	PD-109M	8.44E+04	Y-91			5.64E+07
PM-147			8.72E+06		PARENT	SR-91	5.48E+07
	PARENT	ND-147	2.78E+07		GRAND PARENT	RB-91	5.14E+07
	GRAND PARENT	PR-147	2.76E+07		2ND PARENT	Y-91M	3.18E+07
PM-148			6.95E+06	Y-91M			3.18E+07
	PARENT	PM-148M	1.36E+06		PARENT	SR-91	5.48E+07
PM-148M			1.36E+06		GRAND PARENT	RB-91	5.14E+07
PM-149			2.35E+07	Y-92			5.78E+07
	PARENT	ND-149	1.57E+07		PARENT	SR-92	5.74E+07
	GRAND PARENT	PR-149	1.46E+07		GRAND PARENT	RB-92	4.51E+07
PM-151			8.22E+06	Y-93			4.33E+07
	PARENT	ND-151	8.14E+06		PARENT	SR-93	6.38E+07
	GRAND PARENT	PR-151	4.87E+06		GRAND PARENT	RB-93	3.65E+07
PR-142			2.48E+06	Y-94			6.79E+07
PR-143			6.43E+07		PARENT	SR-94	6.31E+07
	PARENT	CE-143	6.57E+07		GRAND PARENT	RB-94	1.88E+07
	GRAND PARENT	LA-143	6.52E+07	Y-95			7.01E+07
PR-144			5.40E+07		PARENT	SR-95	5.66E+07
	PARENT	CE-144	5.38E+07		GRAND PARENT	RB-95	9.04E+06
	GRAND PARENT	LA-144	5.81E+07	ZR-95			7.25E+07
	2ND PARENT	PR-144M	7.55E+05		PARENT	Y-95	7.01E+07
PU-238			1.40E+05		GRAND PARENT	SR-95	5.66E+07
	2ND PARENT	NP-238	1.54E+07	ZR-97			6.74E+07
PU-239			1.80E+04		PARENT	Y-97	5.52E+07
	PARENT	NP-239	8.64E+08		GRAND PARENT	SR-97	2.12E+07
	GRAND PARENT	U-239	8.66E+08				
	2ND PARENT	AM-239	1.77E-01				

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

Nuclide	Primary Coolant (μCi/gm)
I-131	6.94E-01
I-132	2.24E-01
I-133	9.45E-01
I-134	1.14E-01
I-135	4.93E-01
KR-83M	3.59E-01
KR-85M	1.34E+00
KR-85	7.18E+01
KR-87	8.69E-01
KR-88	2.50E+00
XE-131M	3.94E+00
XE-133M	3.49E+00
XE-133	2.53E+02
XE-135M	7.79E-01
XE-135	1.07E+01

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

Exclusion Area Boundary

Release Point	Averaging Period 0-2 hr
Containment Wall/ Aux. Bldg. Stack/ Aux. Bldg. Fresh Air Intake	2.56E-4 (E)

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

TABLE 5.3-1
Fort Calhoun Technical Support Center 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/TSC Air Intake	1.93E-03	1.67E-03	8.52E-04	6.26E-04	5.09E-04
Aux. Bldg. Stack/TSC Air Intake	1.61E-03	1.29E-03	6.50E-04	4.83E-04	3.96E-04
Aux. Bldg. Air Intake/TSC Air Intake	3.44E-03	2.96E-03	1.33E-03	9.86E-04	7.89E-04

TABLE 7.1-1
Analysis Assumptions & Key Parameter Values
Loss of Coolant Accident

Core Power Level	1530 MWt
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
Core Activity	Table 4.1-1
Containment Leakage Parameters	
Containment Mixing Free Volume:	
- Above the operating floor at El. 1045'	608,400 ft^3
- Below the operating floor at El. 1045'	460,100 ft^3
CRFC Filtered Flow rate	86,500 cfm per unit
CRFC Unfiltered Flow rate	52,000 cfm per unit
Duration of CRFC Flow rate	30 days
CRFC Intake Suction Location	Above the operating floor
CRFC Flow Returned Above El. 1045':	73% to Operating level 4% to Dome
CRFC Flow Returned Below El. 1045':	10% to Basement level 13% to Intermediate level
CRFC HEPA Filter Efficiency	50% for aerosols 0% for noble gas, elemental/organic iodine
Containment Leakrate (0–24 hr)	0.1% weight fractions per day
Containment Leakrate (1-30 day)	0.05% weight fractions per day
Release Point	Containment Outer Wall
Maximum DF for Elemental Iodine	200
Sump Water pH	≥ 7
Mass Transfer Coef. (Kw) used for elemental iodine deposition rate	4.9 meters/hr
Wetted surface area of cont. floor for elemental plateout	6000 ft^2 (covered in 30 minutes)
Assumed wetted surface area prior to Recirc. Initiation.	3000 ft^2 (0-3.25 hr)
Min. / Max. Recirc. Initiation Time	20.4 min / 3.25 hr

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	
- For Control Room Dose	Auxiliary Building Vent Stack
- For TSC Dose	Auxiliary Building Fresh Air Intake

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

TABLE 7.2-1
Analysis Assumptions & Key Parameter Values
FCS Control Room

Control Room Parameters	
Free Volume	45,100 ft^3
Unfiltered Normal Operation Intake	1000 cfm ± 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%

CR Emergency Ventilation: Initiation Signal/Timing

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

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 Up Time	15 seconds
Total	44 seconds

TABLE 7.3-1
Analysis Assumptions & Key Parameter Values
FCS Technical Support Center

TSC Parameters	
Free Volume	36,036 ft^3
Unfiltered Normal Operation Intake	8500 cfm ± 10%
Emergency Intake Rate	1000 cfm ± 10%
Emergency Recirculation Rate	2000 cfm ± 10%
Emergency Intake Filter Efficiency	95% (iodine & particulates)
Emergency Recirculation Filter Efficiency	95% (iodine & particulates)
Unfiltered Inleakage	150 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)

TSC Emergency Ventilation: Initiation Signal/Timing
In emergency mode at T=30 minutes (via operator action)

TABLE 8-1
Exclusion Area Boundary and Low Population Doses (TEDE)

Accident	EAB Dose (rem) [1,3]	LPZ Dose (rem) [2,3]	SB Reg. Limit (rem)
LOCA	3.0	0.50	25.00

NOTES:

1. EAB Dose is based on the worst 2-hour period following the onset of the event. The maximum 2 hr dose period for the LOCA EAB dose is 0.5 to 2.5 hr.
2. LPZ Doses are based on the duration of the release.
3. The EAB and LPZ doses are rounded up to the nearest 0.5 Rem.

TABLE 8-2
30 Day Integrated Control Room and TSC Doses (TEDE)

Operator Location	LOCA	
	Dose (rem) [1]	Reg. Limit (rem)
Control Room	4.7 (1.7)	5.00
Technical Support Center	4.3 (0.3)	5.00

NOTES:

1. Portion shown in parenthesis represents that portion of the total dose that is the contribution of direct shine from contained sources/external cloud.

Affidavit

For

AREVA NP, Inc.

Proprietary Information

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

1. My name is Gayle F. Elliott. I am Manager, Product Licensing, for AREVA NP Inc. and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by AREVA NP to determine whether certain AREVA NP information is proprietary. I am familiar with the policies established by AREVA NP to ensure the proper application of these criteria.

3. I am familiar with the AREVA NP information contained in report 77-9051353P-001, "Summary of FCS Containment Analysis Without Containment Spray," dated July 2007, and referred to herein as "Document." Information contained in this Document has been classified by AREVA NP as proprietary in accordance with the policies established by AREVA NP for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by AREVA NP and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is

requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by AREVA NP to determine whether information should be classified as proprietary:

(a) The information reveals details of AREVA NP's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for AREVA NP.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for AREVA NP in product optimization or marketability.

(e) The information is vital to a competitive advantage held by AREVA NP, would be helpful to competitors to AREVA NP, and would likely cause substantial harm to the competitive position of AREVA NP.

The information in the Document is considered proprietary for the reasons set forth in paragraphs 6(b) and 6(c) above.

7. In accordance with AREVA NP's policies governing the protection and control of information, proprietary information contained in this Document have been made available, on a limited basis, to others outside AREVA NP only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. AREVA NP policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 18th day of July, 2007.

Danita R. Kidd

Danita R. Kidd
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 12/31/08

Danita R. Kidd
NOTARY PUBLIC
Commonwealth of VA
Comm. Expires: 12-31-08

Reg # 205569