Mr. S. K. Gambhir
Division Manager - Nuclear Operations
Omaha Public Power District
Fort Calhoun Station FC-2-4 Adm.
Post Office Box 399
Hwy. 75 - North of Fort Calhoun
Fort Calhoun, NE 68023-0399

December 5, 2001

SUBJECT: FORT CALHOUN STATION, UNIT NO. 1 - ISSUANCE OF AMENDMENT (TAC NO. MB1221)

Dear Mr. Gambhir:

The Commission has issued the enclosed Amendment No. 201 to Facility Operating License No. DPR-40 for the Fort Calhoun Station, Unit No. 1. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated February 7, 2001, as supplemented by letters dated October 17 and November 2, 2001.

The amendment replaces the current accident source term used in the design basis radiological analyses for control room habitability with an alternative source term (AST) pursuant to 10 CFR 50.67, "Accident Source Term." Omaha Public Power District (OPPD) has requested a full implementation of the AST. OPPD is also using this amendment request to address TS and the associated TS Bases changes related to this request. In addition, OPPD has stated that all future revisions of the Appendix E analyses to the "Implementation of Alternative Source Terms for Fort Calhoun Report," dated January 2001, will utilize an overall effective decontamination factor for Iodine 200.

A copy of the related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/

Alan B. Wang, Project Manager, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-285

Enclosures: 1. Amendment No. 201 to DPR-40
2. Safety Evaluation

cc w/encls: See next page

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Omaha Public Power District
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ADAMS ACCESSION NUMBER: **ML013030027**
Amendment: ML013400297

*See previous concurrence

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DISTRIBUTION FOR FORT CALHOUN LICENSE AMENDMENT NO. 201

Dated: December 5, 2001

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OMAHA PUBLIC POWER DISTRICT

DOCKET NO. 50-285

FORT CALHOUN STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 201
License No. DPR-40

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the Omaha Public Power District (the licensee) dated February 7, 2001, as supplemented by letters dated October 17 and November 2, 2001, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, Facility Operating License No. DPR-40 is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B. of Facility Operating License No. DPR-40 is hereby amended to read as follows:

   B. Technical Specifications

      The Technical Specifications contained in Appendix A, as revised through Amendment No. , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of its date of issuance and shall be implemented within 60 days from the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Stephen Dembek, Chief, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: December 5, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 201

FACILITY OPERATING LICENSE NO. DPR-40

DOCKET NO. 50-285

Replace the following pages of Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

REMOVE	INSERT
2-22	2-22
2-23a	2-23a
2-39d	2-39d
2-39i	2-39i
2-39q	2-39q
2-39r	2-39r
2-39s	2-39s
2-39t	---
3-54	3-54
3-54a	3-54a
3-57a	3-57a
3-84	3-84
3-85	3-85

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 201 TO FACILITY OPERATING LICENSE NO. DPR-40

OMAHA PUBLIC POWER DISTRICT

FORT CALHOUN STATION, UNIT NO. 1

DOCKET NO. 50-285

1.0 INTRODUCTION

By application dated February 7, 2001, as supplemented by letters dated October 17 and November 2, 2001, Omaha Public Power District (OPPD) requested changes to the Technical Specifications (TSs) (Appendix A to Facility Operating License No. DPR-40) for the Fort Calhoun Station (FCS), Unit No. 1. The requested changes would replace the current accident source term used in design basis radiological analyses for control room habitability with an alternative source term (AST) pursuant to 10 CFR 50.67, "Accident Source Term." OPPD has requested a full implementation of the AST. OPPD is also using this amendment request to address TS changes related to this request.

The supplemental letters dated October 17 and November 2, 2001, provided additional information that clarified the application. These letters did not expand the scope of the application as originally noticed, nor did they change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on May 2, 2001, (66 FR 22031).

In December 1999, the NRC issued a new regulation, 10 CFR 50.67, "Accident Source Term," which provided a mechanism for licensed power reactors to voluntarily replace the traditional accident source term used in their design basis accident (DBA) analyses with ASTs. Regulatory guidance for the implementation of these ASTs is provided in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." Section 50.67 requires a licensee seeking to use an AST to apply for a license amendment and requires that the application contain an evaluation of the consequences of DBAs. This amendment request addresses these requirements in proposing to replace the existing accident source term used in design basis radiological consequence analyses with an AST. OPPD re-analyzed the offsite and control room dose consequences for the DBAs previously analyzed in the FCS Updated Safety Analysis Report (USAR). Although several of the re-analyzed accidents are not impacted by the implementation of an AST, OPPD has updated the analyses in the interest of design basis consistency. OPPD also updated the site boundary and control room atmospheric dispersion factors.

Associated with these efforts, OPPD has proposed the following TS changes:

- 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 and spent fuel pool (SFP).

- Delete the existing requirement that the ventilation isolation actuation signal be operable with two radiation monitors during fuel movement in the SFP.

- Require performance of an internal leakage test on the residual heat removal (RHR) system with an acceptance criterion that leakage be limited to 3800 cc/hour.

- Increase the minimum amount of trisodium phosphate from 110 ft^3 to 126 ft^3.

- Make conforming changes to TS Bases.

2.0 EVALUATION

The staff has reviewed the changes proposed by OPPD, as described in their submittal of February 7, 2001, with additional information submitted by letter dated October 17, 2001. The staff reviewed analysis inputs, methods, and results described in the OPPD submittal. The following sections of this safety evaluation (SE) provide the results of the staff's review of OPPD's AST analyses. Table 1 tabulates the analysis inputs and assumptions found acceptable to the staff. Table 2 tabulates the results of OPPD's docketed analyses. Although the staff did some confirmatory analyses, the staff's approval of this amendment is based on the information docketed by OPPD and on the staff's finding that the methods, inputs, and assumptions used in the OPPD AST analyses are acceptable.

2.1 Atmospheric Dispersion Factors (χ/Q)

OPPD re-analyzed the χ/Q values for the exclusion area boundary (EAB), low population zone (LPZ), and the control room. The revised values are tabulated in Table 1 of this SE. The EAB and LPZ values were calculated using a proprietary Stone and Webster computer code that implements the methodology identified in RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants." Terrain recirculation factors used in the calculation of annual average χ/Q values were obtained from RG 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors." Meteorological data collected on the FCS meteorological tower between January 1, 1994 and December 31, 1998 (i.e., five years) were used in this assessment. The distances to the EAB were determined for each of the 16 downwind sectors using the 45 degree sector approach of RG 1.145. Several release points were considered for the EAB:

- Containment surface (wall)
- Auxiliary building stack
- Auxiliary building fresh air intake (backflow)
- Main steam safety valve and atmospheric dump valve (MSSV/ADV)
- Radwaste processing building ventilation exhaust

- Room 81 pressure relief domes
- Condenser evacuation discharge
- Turbine-driven auxiliary feedwater pump turbine exhaust

OPPD conservatively grouped these release points based on configuration similarities, performed calculations on the subset and then selected the most limiting value for use for all release points. For the LPZ, a single distance was assumed in all directions given the magnitude of the distance relative to the separation of release point locations. Control room intake χ/Q values were calculated for release points identified above using the NRC-sponsored ARCON96 computer code as described in NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes." Meteorological data collected on the FCS meteorological tower between January 1, 1994 and December 31, 1998 (i.e., five years) were used in this assessment. All releases were treated conservatively as ground level releases. Building wake was considered only for the containment wall and auxiliary building stack release points, as there is little or no interference from buildings along the remaining release-intake combinations. Since there are multiple MSSVs, ADVs and pressure relief domes, the release-to-intake distances were based on the centroid of rectangles encompassing the discharge stacks or the 4 pressure relief domes.

Based on its review of the information provided by the licensee, the staff finds that OPPD used analysis methods and assumptions acceptable to the staff in determining the χ/Q values. The staff compared the proposed EAB χ/Q values to the value generated by the staff during original licensing and found them to be comparable (2.56E-4 sec/m^3 versus 2.7E-4 sec/m^3). With regard to the control room values, the staff reviewed the use of the methodology and found it to be consistent with current staff positions on the use of ARCON96. The staff performed a qualitative review of the code results and deemed them to be reasonable. The staff finds the revised χ/Q values are acceptable.

2.2 Accident Radiation Source Terms

2.2.1 Core Inventory

The inventory of fission products in the reactor core is based on the maximum full power operation of the core at the rated thermal power level. An instrument uncertainty correction of 2 percent was applied in accordance with RG 1.49, "Power Levels of Nuclear Power Plants." OPPD developed the core inventory using the SAS2H and ORIGEN-S modules of the NRC-sponsored SCALE code package. The determination of core inventory is dependent on the level of fuel enrichment. Since the FCS core contains fuel assemblies with different levels of enrichment, core inventory calculations were performed for enrichments of 3.5, 4.0, and 4.5 percent. The highest activity for each isotope for the three enrichments was chosen to represent the inventory of that isotope in the composite core. An 18 month fuel cycle and a 3 region core management approach is assumed. The staff finds this approach to be consistent with current regulatory guidance and, therefore, acceptable.

2.2.2 Coolant Inventory

OPPD recalculated the specific activity of the reactor coolant and the secondary coolant to reflect the changes in core inventory. These values were determined using proprietary computer codes that calculate the isotopic activities as a function of the applicable production and depletion processes including, fuel leakage, primary-to-secondary leakage, radioactive decay, neutron activation, and in-growth of decay products. OPPD normalized the reactor and secondary coolant iodine activities to the respective dose equivalent I-131 TS. Reactor coolant noble gases were normalized to the associated reactor coolant TS. Secondary noble gas concentrations are not assessed since noble gases are not retained in the steam generator bulk water. The staff finds this approach to be consistent with current regulatory guidance and, therefore, acceptable.

2.2.3 Loss-of-Coolant Accident (LOCA) Fission Product Release Characteristics

For the LOCA, the fission products are projected to be released from the fuel to the reactor coolant and then to the containment in distinct phases. Two phases are addressed in DBAs. The gap release phase begins 30 seconds after the LOCA starts and continues for 30 minutes. The early in-vessel release phase begins at 30 minutes and continues for 1.3 hours. Table 2.2.3-1 tabulates the fraction of each of several nuclide groups that is released in each release phase. The inventory in each release phase is assumed to be released at a constant rate over the duration of the phase and starting at the onset of the phase. The staff finds this data to be consistent with regulatory guidance and, therefore, acceptable.

Table 2.2.3-1 Release Fractions as a Function of Release Period

RADIONUCLIDE GROUP	**GAP RELEASE** *(0.5 Hours)*	**EARLY IN-VESSEL** *(1.3 Hours)*
Noble Gases (Xe, Kr)*	0.05	0.95
Halogens (I, Br)	0.05	0.35
Alkaline Metals (Cs, Rb)	0.05	0.25
Tellurium Group (Te, Sb, Se)	0	0.05
Ba, Sr	0	0.02
Noble Metals (Ru, Rh, Pd, Mo, Tc, Co)	0	0.0025
Cerium Group (Ce, Pu, Np)	0	0.0005
Lanthanides (La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am)	0	0.0002

*Example elements in each group. OPPD identified additional elements in their submittal.

2.2.4 Gap Fractions for Events other than LOCA

The majority of the fission products in a reactor core are retained in the ceramic fuel pellet. A fraction of this activity migrates through the pellet and escapes to the volume between the fuel pellet and the fuel rod as a function of temperature and fuel burnup. The activity is retained in this gap region, hence the term "gap fraction." RG 1.183 tabulates gap fractions considered to be acceptable, given certain pre-conditions. One of these pre-conditions specifies that the maximum peak rod average burnup cannot exceed 62 gigawatt-day per metric ton uranium (GWD/MTU), the current NRC-approved burnup limit. For fuel with burnups exceeding 54 GWD/MTU, the maximum linear heat generation rate cannot exceed 6.3 kilowatt per feet (kW/ft). FCS utilizes a few high burnup fuel assemblies in high neutron flux regions to decrease peaking ratios; these rods are driven to linear heat generation rates slightly in excess of 6.3 kW/ft.

OPPD proposed a FCS-specific set of gap fractions. These gap fractions were determined using ANSI/ANS-5.4, "American National Standard Method for Calculating the Fractional Release of Volatile Fission Products." Using this methodology, OPPD determined an equilibrium release fraction using the peak fuel rod's average temperature at a linear heat generation rate of 8 kW/ft plus 200°F. OPPD stated that they used a higher than expected linear heat generation rate and the 200 degree margin to envelope analysis uncertainties. OPPD determined that the gap fractions for short-life isotopes were approximately one-half of the corresponding values tabulated in RG 1.183. The OPPD gap release fractions for the longer-lived isotopes (e.g., Cs137) were approximately 45 percent greater than the corresponding RG 1.183 values. Since the proposed approach does not explicitly address power operation and temperature distributions prior to 54 GWD/MTU, OPPD elected to use gap fractions that are a factor of 2 greater than the RG 1.183 values.

The staff has reviewed the gap fractions proposed by OPPD and the methodology used to develop them. The staff believes that there is reasonable assurance that the radiation doses analyzed using the gap fractions proposed by OPPD will bound the radiation doses resulting from an actual event. The staff bases this conclusion on the site-specific OPPD analysis, including: (1) the nearly factor of four margin in the short-lived isotopes, which are the predominant contributors to offsite dose for DBAs; (2) the conservative application of a radial peaking factor based on the fuel assembly in the highest power position to all assemblies projected to be damaged; (3) the conservatively assumed higher linear heat generation rate and the additional 200 degree margin; and (4) the general acceptability and conservatism of the ANSI/ANS-5.4 method. As such, the proposed values in Table 2.2.4-1 are acceptable for use at FCS.

Table 2.2.4-1 Release Fractions as a Function of Release Period

RADIONUCLIDE GROUP	RG 1.183 Gap Fraction	FCS Gap Fraction
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

2.3 Control Room Modeling

The FCS control room is modeled as a single region. The control room is designed to operate at 1/8" water gauge during normal and accident operations. The FCS control room operates in the isolated zone, filtered pressurization and filtered recirculation mode. The control room analysis assumes the following:

- Before the event, the control room ventilation system works in an unfiltered recirculation mode. The control room air handling units recirculate and condition the control room atmosphere. These units draw in unfiltered outside air at 1000 cfm through 2 redundant isolation dampers. This intake serves as normal makeup and serves to maintain a slight positive pressure during normal operations.

- Upon receipt of a safety injection actuation signal, a containment spray actuation signal, a containment atmosphere radiation high signal, a containment pressure high signal, or a pressurizer pressure low signal, the normal control room unfiltered intake dampers close. The recirculation/intake filter trains are placed into service and and filtered intake dampers open. Recirculation/intake fans draw 1000 cfm from the control room as recirculation flow and 1000 cfm as filtered makeup. The filter media has an efficiency of 99 percent for all iodine species.

- The damper that allows recirculation flow between the control room and the suction of the filtered intake fans does not meet single failure criteria. Since the same fan is used to draw outside air and recirculation flow, failure of the recirculation damper to open could result in greater than 1000 cfm outside air being drawn in, while preventing the cleanup of the control room atmosphere. As a compensatory action, OPPD assumes that there is no filtered recirculation for the first two hours of the event, allowing time for damper repair. Since the radioactive release rate is not constant over this two hour period, OPPD assesses the control room dose assuming a filtered intake flow rate of 1000 cfm and 2000 cfm. The staff confirmatory calculations indicated that the higher filtered intake did not significantly increase the doses and in some cases, reduced the doses.

- OPPD has previously conducted integrated control room infiltration testing and has assumed an unfiltered inleakage rate of 38 cfm in the control room analyses. Since the FCS control room design includes double vestibule doors, OPPD assumes no unfiltered inleakage due to egress and ingress. This assumption is consistent with the guidance of Section 6.4 of the Standard Review Plan.

- In the event of a loss-of-offsite power (LOOP) in conjunction with a DBA, the control room isolation would be delayed until the emergency diesel generator (EDG) re-energized emergency buses. OPPD conservatively assumes that the control room isolation is delayed 44 seconds. (This delay is added to the accident-specific initiation times.)

2.4 Analyzed Accidents

The following are the review of various DBAs with some accident-specific considerations addressed in the discussion.

2.4.1 Large Break Loss-of-Coolant Accident (LBLOCA) Radiological Consequences

The objective of analyzing the radiological consequences of an LBLOCA is to evaluate the performance of various plant safety systems intended to mitigate the postulated release of radioactive materials from the plant to the environment. OPPD assumes an abrupt failure of a main reactor coolant pipe and core damage occurs due to multiple failures. Fission products from the damaged fuel are released into the reactor coolant system (RCS) and then into the containment. With the LBLOCA, it is anticipated that the initial fission product released to the containment will last 30 seconds and will release all of the radioactive materials dissolved or suspended in the RCS liquid. The gap inventory release phase begins with the onset of fuel cladding failure and is assumed to continue to 30 minutes. As the core continues to degrade, the gap inventory release phase ends and the in-vessel release phase commences. This phase continues for 1.3 hours. Table 2.2.3-1 tabulates the isotopic releases from the core to the containment. The inventory in each release phase is assumed to be released at a constant rate over the duration of the phase and starting at the onset of the phase.

Fission products are released from the core into the containment and are assumed to mix instantaneously and homogeneously throughout the free volume of the containment. The release into the containment is assumed to terminate at the end of the early in-vessel phase, approximately 1.8 hours after the LBLOCA. OPPD assumes that the iodine released to the containment are of the following species: 95 percent CsI, 4.85 percent elemental iodine, and 0.15 percent organic forms. This iodine speciation is acceptable if the containment sump pH is maintained at a value of 7.0 or higher. This is accomplished at FCS by trisodium phosphate (TSP) baskets located in the containment sump area. As part of this application, OPPD has proposed to increase the amount of TSP provided at FCS.

Once dispersed in the containment, the release to the environment is assumed to occur through three pathways:

  - A release from containment vacuum relief line.

- Leakage of containment atmosphere (i.e., design leakage).
- Leakage from the emergency core cooling systems (ECCS) that recirculate containment sump water outside containment (including backleakage releases via the safety injection refueling water tank).

At FCS, hydrogen control containment purging is not needed until after the thirty day exposure period. Since this purge is an event for which planning and analysis can be performed, the dose impact of a containment purge is not addressed as part of the DBA LOCA.

Release from Containment Vacuum Relief Line

The containment vacuum release line is assumed to be in operation at the start of the LBLOCA, providing a path for release to the environment. This line is isolated as a result of a containment isolation signal, which is projected to occur prior to 5 seconds post-accident. Since the onset of fission product releases from the fuel occurs at 30 seconds, this pathway is isolated prior to fuel damage occurring. OPPD conservatively assumes that the entire RCS radionuclide inventory is released to the containment at T=0 and that 100 percent of the volatile nuclides are instantaneously and homogeneously mixed in the containment atmosphere. Based on the rate of containment pressurization and the vacuum release line flow characteristics, OPPD estimates that release flow rate will be 600 cfm during this period. Since fuel damage and containment sprays will not have started in this period, OPPD assumes that the chemical form of the iodine released from the RCS is 97 percent elemental and 3 percent organic. The staff finds these assumptions to be consistent with applicable regulatory guidance and therefore, acceptable.

Leakage of Containment Atmosphere

During an LBLOCA, the FCS containment atmosphere is sprayed by the containment spray system. This system is automatically started by high containment pressure and is assumed to be fully operable between 185 seconds and approximately 5 hours following the LBLOCA. The containment spray is effective at removing particulate aerosols and elemental iodine from the containment atmosphere. Because of the configuration of equipment and internal structures within the containment, the sprays do not cover the full containment volume. OPPD modeled the containment as being comprised of two regions–sprayed and unsprayed. Based on evaluations of spray nozzle coverage and containment arrangement, OPPD projects that 69.4 percent of the containment free volume is sprayed. Forced circulation by the containment fan coolers and natural convection establishes mixing between the sprayed and unsprayed region. OPPD first proposed a mixing rate of 4.84 unsprayed volumes per hour in their response to Generic Letter 99-02 dated July 28, 2000, and supplemented on December 1, 2000. Although the staff did not accept selected aspects of the methodology used to develop this mixing rate, the staff found the site-specific numeric value to be acceptable for use at FCS in that review. The containment leaks at a rate of 0.1 percent volume per day for the first 24 hours and 0.05 percent volume per day for days 1 through 30. This leakage occurs from both the sprayed and unsprayed regions.

OPPD assumed that containment sprays were effective for particulates and elemental iodine. No credit for spray removal was assumed for noble gases or for organic forms of iodine. The effectiveness of the sprays for fission product scrubbing is represented by the spray removal rate (often referred to as spray coefficients or spray lambda, λ). For particulate fission products, OPPD proposed spray removal rates determined using the Stone and Webster proprietary computer program SWNAUA, as an alternative to the methods identified in SRP Section 6.5.2 and RG 1.183. The removal rates provided by SWNAUA are based on calculated time-dependent aerosol airborne concentrations. In its review of the proposed spray removal rates, the staff reviewed the brief description of the code model provided by OPPD, reviewed the analysis inputs and assumptions, evaluated the reasonableness of the estimated removal rates, and reviewed the use of the estimated removal rates in the radiological analyses. The staff did not review the SWNAUA code on a generic use basis, but rather, reviewed the FCS plant-specific removal rate values for acceptability. Although there are several aerosol phenomena that promote the depletion of aerosols from the containment, the OPPD particulate removal calculation only takes credit for diffusiophoresis and the removal effectiveness of the sprays. Aerosol agglomeration was considered. OPPD asserts that if the natural removal phenomena were included, the total removal effectiveness would increase even though the effectiveness of the spray removal would be slightly reduced. OPPD assumed a single spray droplet radius of 900 micron. In practice, a large distribution of spray droplet sizes would result, most of which would have radii less than 900 micron. Since the efficiency of spray removal increases with decreasing spray droplet radii, the single droplet size is considered to be conservative. Although a single spray droplet size was used, a distribution of aerosol sizes were considered and the spray removal efficiency was determined for each aerosol size bin.

In the diffusiophoresis phenomena, particulate matter is entrained in steam as it flows towards condensation surfaces. The RCS blowdown to the containment creates large quantities of steam that condense on containment spray droplets and on containment internal surfaces. In the current approach, OPPD has conservatively assumed that the steam condenses only on the spray droplets and aerosol particulates. No credit was taken for steam condensation on heat sinks. OPPD presented the removal rate of particulates by sprays and diffusiophoresis versus time after the accident by means of a graph (Figure 7.2-1 of the February 7, 2001, submittal). Values range from about 1.4 hr^{-1} at 1800 seconds post-accident, about 5.8 hr^{-1} from about 1,800 to 6,600 seconds with short spikes to 6 hr^{-1}, and tapering off to about 1 hr^{-1} at 18,000 seconds.

The SWNAUA code was not used to determine the elemental iodine spray removal. Instead, OPPD took the conservative, non-mechanistic, position that the spray removal rate for the elemental iodine would be assumed to be equal to that determined for the particulate spray removal. OPPD correctly noted that the spray models provided in SRP Section 6.5.2 provides elemental iodine removal rates greater than those for particulates. The staff performed a confirmatory calculation and determined that the calculated elemental iodine removal rate was greater than the particulate removal rates shown in Figure 7.2-1 of the February 7, 2001, submittal. Consistent with the guidance of SRP 6.5.2, OPPD established a maximum decontamination factor (DF) of 200 for elemental iodine. This DF is not reached prior to the assumed termination of containment spray removal credit. Since the particulate spray removal coefficients are based on calculated time dependent airborne aerosol masses, there is no restriction on particulate iodine DFs.

In their submittal, OPPD stated the assumption that the elemental iodine plates out onto the particulate form and is therefore removed at the same rate. This mechanism is not supported by any empirical data and is not acceptable to the staff. In discussions with the staff, OPPD clarified this statement as a simplifying assumption to explain their conservative, non-mechanistic, use of the particulate removal rates for the elemental iodine removal. The staff agrees that the use of the particulate removal rates for elemental iodine is conservative, but not on the basis of the offending statement, as noted above. Subject to this clarification, the staff finds that OPPD's modeling of containment spray removal is acceptable, as are the removal rates shown in Figure 7.2-1 of the February 7, 2001, submittal for use at FCS. However, the staff notes that this plant-specific approval does not constitute approval of the generic use of the SWNAUA code for design basis analyses by other facilities.

Leakage from Systems that Recirculate Containment Sump Water Outside Containment

During the progression of an LBLOCA, some fission products released from the fuel will be carried to the containment sump via spillage from the RCS or by transport of activity in the containment atmosphere to the sump by the containment spray system and by natural processes such as deposition and plateout. During the initial phases of an LBLOCA, safety injection and containment spray systems draw water from the safety injection refueling water tank (SIRWT). At a certain point in the response (T=20.4 minutes), these systems start to draw water from the containment sump instead. This recirculation flow causes contaminated water to be circulated through piping and components outside of the containment where small amounts of system leakage could provide a path for the release of fission products to the environment. Although the SIRWT is isolated during recirculation, design leakage through valving provides a pathway for backleakage of the containment sump water to the SIRWT. The SIRWT is located in and vented to the auxiliary building. Since these pathways represent a bypass of the containment, the dose consequences must be considered. Since the transport for both of these pathways are comparable and the associated releases are ultimately through the auxiliary building vent stack, the leakage can be summed and both paths assessed as one.

For the purposes of assessing the consequences of leakage via these pathways, OPPD assumes that all of the radioiodines are instantaneously moved to the containment sump as they are released from the fuel/RCS. Noble gases are assumed to remain in the containment atmosphere. The remaining radionuclides in Table 2.2.3-1 are aerosols or particulates. Since radionuclides of this chemical form are not assumed to become airborne on release from the ECCS, they are not included in the ECCS source term. This source term assumption is conservative in that all of the radioiodines released from the fuel are credited in both the containment atmosphere and containment sump. In a mechanistic treatment, the relocation of the radioiodines in the containment atmosphere would occur over time and deplete the radioiodine concentration in the containment atmosphere.

OPPD assumes that the leakage is two times the TS criterion, or 7600 cc/hour. Since the temperature of this fluid is less than 212°F, OPPD assumes that 10 percent of the entrained iodine activity is released to the atmosphere of the surrounding auxiliary building. OPPD assumes that this activity is exhausted without holdup, mixing, or filtration via the auxiliary building vent stack. OPPD selected this release point since it has the most unfavorable dispersion characteristics relative to the control room intake. OPPD conservatively assumes that the chemical form of the iodine released is 97 percent elemental and 3 percent organic.

These assumptions are consistent with applicable regulatory guidance and therefore, are acceptable.

Accident-Specific Control Room Assumptions

Due to the rapid containment pressure transient expected following an LBLOCA, the signal to actuate control room emergency ventilation is assumed to occur at T=0. Although a LOOP is assumed to occur at T=0, OPPD also considered a LOOP occurring at about T=1.8 hours, the time of maximum activity in the containment atmosphere. For this latter case, OPPD assumed that for the 44 second period in which the ventilation system is considered inoperable, the unfiltered inleakage was assumed to be one half of the flow necessary to maintain pressurization (i.e., 500 cfm). These assumptions are consistent with staff guidance and, therefore, acceptable. OPPD also considered the dose to control room personnel due to direct radiation shine from the external cloud and from contained sources, including the containment structure. Doses were calculated for several locations within the control complex.

Offsite Dose Calculations

OPPD determined the worst two-hour dose at the EAB by computing the dose for the following two-hour periods:

0-120m	Reference point
20-140m	Near end of gap release, start of in-vessel release
30-150m	Encompasses early in-vessel phase
50-170m	Window with endpoint just after early in-vessel phase
108-228m	Window starting after early in-vessel phase

The resulting doses were plotted and the maximum dose determined (release ending at 2.5 hours). The staff compared the sampling intervals identified above against possible changes in analysis parameters. The only time-variant analysis parameters that could affect EAB dose are the containment spray lambdas, the actuation and termination of containment sprays, and the initiation of recirculation. The containment sprays initiate at about 185 seconds and continue until 5 hours. Since the maximum containment activity occurs at the end of the early-in-vessel phase at about 1.8 hours, doses would not increase with the termination of containment sprays. The containment spray lambdas increase rapidly at 30 minutes and decrease gradually starting at about 120 minutes. Both of these transitions are bounded by the time intervals used. Initiation of containment sump recirculation occurs at 20.4 minutes. This also is bounded by the time intervals used.

LBLOCA Conclusion

Details on the assumptions found acceptable to the staff are presented in Table 1. The resulting doses are presented in Table 2. The doses for the postulated LBLOCA were found to be acceptable.

2.4.2 Fuel Handling Accident (FHA) Radiological Consequences

This accident analysis postulates that a spent fuel assembly is dropped during refueling, damaging all of the rods in the assembly. This accident could happen inside the containment or in the SFP area. OPPD evaluated the consequences of a FHA in either location. All of the gap inventory in the damaged rods is assumed to be released from the fuel. As discussed in Section 2.2.4 of this SE, OPPD assumed that 16 percent of the I-131 inventory of the core was in the fuel rod gap, along with 20 percent of the Kr-85, and 10 percent of all other iodines, noble gases, and 24 percent of alkali metals. The core inventory is based on a decay period of 72 hours after reactor shutdown and a radial peaking factor of 1.8.

OPPD assumed the iodine species fractions for the fuel release to be 95 percent cesium iodine, 4.85 percent elemental, and 0.15 percent organic. Due to the acidic nature of the water in the fuel pool and reactor cavity, it is assumed that the cesium iodide immediately dissociates into elemental iodine in the fuel pool and cavity water. The resulting percentages are then 99.85 percent elemental and 0.15 percent organic. OPPD assumed a DF of 500 for elemental iodine and a DF of 1.0 for noble gases and organic iodides. RG 1.183, while identifying a DF of 500 for elemental iodine, stated that the effective pool DF was 200 (calculated value rounded down). The staff expected licensees to use the effective value in DBA analyses. Given the conservative assumptions regarding gap fractions and the margin available between the calculated doses and the acceptance criterion, OPPD's analyses need not be redone for this application. By letter dated October 17, 2001, OPPD committed to use an effective pool DF of 200 for elemental iodine in future re-analyses.

For the FHA in the fuel pool area, OPPD assumes that the release from the pool is collected by the fuel pool ventilation system and released, unfiltered, via the auxiliary building vent stack. All of the airborne activity is assumed to be released in two hours. For the FHA in the containment, OPPD does not assume collection or filtration of the release since the containment may be open during the event. All of the airborne activity is assumed to be released in two hours. Although the containment purge exhaust flow, which is directed to the auxiliary building vent stack, is operable during the event, OPPD has used χ/Q values associated with a containment surface release as these values are more limiting.

Since the control room is aligned in the emergency mode prior to fuel movement, the actuation delay associated with LOOP is not applicable and the two-hour delay associated with the recirculation damper is not applicable. 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. Details on the assumptions found acceptable to the staff are presented in Table 1. The resulting doses are presented in Table 2. The doses for the postulated FHA in the SFP area and the containment were found to be acceptable.

2.4.3 Heavy Load Drop (HLD) Radiological Consequences

This accident analysis postulates that all of the fuel assemblies in the core are damaged by the dropping of a heavy load over the core in the containment. This event is not specifically addressed in RG 1.183. However, this analysis follows the RG 1.183 guidance for an FHA, except as discussed below. All of the gap inventory in the damaged rods is assumed to be

released from the fuel. Since the entire core is involved, OPPD used the LOCA gap fractions specified in RG 1.183 and did not include a radial peaking factor. These assumptions are acceptable in that the gap fractions for the non-LOCA events are predicated on the assembly having the highest burnup and the highest power. Since the entire core is involved, this weighting is not necessary. Similarly, the radial peaking factor weighting of the average core inventory is not needed.

The analysis assumes that the heavy load movement does not occur until 72 hours after shutdown. During this period, containment closure is set by plant procedures except for the containment purge exhaust flow. The flow is assumed to be 50,000 cfm when the reactor cavity level is greater than 23 feet and 5,000 cfm otherwise. Redundant safety-related monitors isolate the containment purge system on a high radiation signal, terminating the release. Delays in radiation monitor and purge system response were evaluated. OPPD assumes a sample transport time of 1.02 minute, a monitor response time of 2 seconds, and damper closure time of 5 seconds. The OPPD analysis also considers the activity left in the duct following isolation by including the duct purge time in the isolation delay. For the 5,000 cfm case, the purge time is about 35.3 seconds and about 4.8 seconds for the 50,000 cfm case. Both estimates are based on the assumption of plug flow in the duct. The resulting release termination times for the two cases are 73 seconds and 103.5 seconds.

OPPD considered two cases with regard to reactor cavity water level. In the first case, the cavity was assumed to be at normal water level and an elemental iodine DF of 500 and an organic DF of 1 were assumed. As noted in the discussion for the FHA, the staff expected licensees to use an effective DF value of 200 in DBA analyses. Given the conservative assumptions regarding the amount of fuel damage, the staff will not require that the analysis be redone. By letter dated October 17, 2001, OPPD committed to use an effective DF of 200 in future re-analyses. For the second case, the water level was assumed to be one foot below the reactor vessel flange which provides 11.15 feet of water cover. Due to the reduced water coverage, OPPD used an iodine DF of 20, as determined using the methodology provided in the staff report, "Evaluation of Fission Product Release and Transport for a Fuel Handling Accident."

OPPD assumed that the released activity mixes with 50 percent of the containment volume prior ro release. OPPD based this assumption on the high flow rate of the containment recirculation fans and the relative location of the containment recirculation ventilation registers and those of the purge exhaust. Since the control room is aligned in the emergency mode prior to fuel movement, the actuation delay associated with LOOP is not applicable and the two-hour delay associated with the recirculation damper is not applicable. 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. Details on the assumptions found acceptable to the staff are presented in Table 1. The resulting doses are presented in Table 2. The doses for the postulated HLD in the containment were found to be acceptable.

2.4.4 <u>Seized Rotor Accident (SRA) Radiological Consequences</u>

For this accident a reactor coolant pump rotor is assumed to seize instantaneously, causing a rapid reduction in the flow through the affected RCS loop. A reactor trip will occur, shutting down the reactor. The flow imbalance creates localized temperature and pressure changes in

the core. If severe enough, these differences may lead to localized boiling and fuel damage. The radiological consequences are due to leakage of the contaminated reactor coolant to the steam generators (SG) and from there, to the environment. A LOOP is conservatively assumed to occur when the reactor trips, rendering the main condenser unavailable. With the main condenser unavailable, the plant is cooled down by releases of steam to the environment via ADV and/or the MSSV.

A SRA at FCS is projected to result in 1.0 percent failed fuel and the release of the associated gap activity. This release is assumed to be instantaneously and homogeneously mixed in the reactor coolant system and transported to the secondary side via SG tube leakage assumed to be at the TS value of 1 gpm at standard temperature and pressure (STP). As discussed in Section 2.2.4 of this SE, OPPD assumed that 16 percent of the I-131 inventory of the core was in the fuel rod gap, along with 20 percent of the Kr-85, and 10 percent of all other iodines, noble gases, and 24 percent of alkali metals. The core inventory is based on a decay period of 72 hours after reactor shutdown and a radial peaking factor of 1.8. OPPD assumed the iodine species fractions for the fuel release to be 95 percent cesium iodine, 4.85 percent elemental, and 0.15 percent organic. OPPD assumes an iodine partitioning factor of 100 in the SGs. The iodine releases from the SG are assumed to be 97 percent elemental and 3 percent organic. Noble gases are released to the environment without holdup in the SG. Particulates are carried over in accordance with the design basis moisture carryover fraction of 0.0025.

The releases from the SG will continue for 8 hours, at which time shutdown cooling is initiated with the RHR system and the environmental releases are terminated. The accident analysis considers two events–a 2-hour event with a 75 degree/hour cooldown rate and an 8-hour event based on reaching 300 degrees in 8 hours. The effect of the two cases is to maximize the release in the two-hour period and the total steam release in the longer period. OPPD asserts that the limiting 2-hour dose will occur at 0-2 hours or at 6-8 hours. The noble gases release rate will be highest at the onset of the event, while the iodine and particulate release rate peaks when the secondary system releases end at 8 hours. For the control room and LPZ doses, the 8-hour event releases are considered.

The SRA does not initiate any signal which could automatically start the control room emergency ventilation. For this reason, OPPD assumes manual initiation is delayed for seven hours, based on procedurally required radiation surveys in the control room. Consistent with the discussion in Section 2.3, the recirculation filtration is conservatively not credited for the first two hours following the initiation of emergency ventilation (i.e., nine hours). Details on the assumptions found acceptable to the staff are presented in Table 1. The resulting doses are presented in Table 2. The doses for the postulated SRA were found to be acceptable.

2.4.5 Control Rod Ejection Accident (CREA) Radiological Consequences

This accident analysis postulates the mechanical failure of a control rod drive mechanism pressure housing that results in the ejection of a rod cluster control assembly and drive shaft. Localized damage to fuel cladding and a limited amount of fuel melt are projected. This failure breeches the reactor pressure vessel head resulting in a LOCA to the containment. A reactor trip will occur. From a radiological analysis standpoint, the CREA is similar to a small break LOCA, but with lesser fission product releases.

The release to the environment is assumed to occur through two separate pathways:

- Release of containment atmosphere (i.e., design leakage)
- Release of RCS inventory via primary-to-secondary leakage through SGs

While the actual doses from a CREA would be a composite of the two pathways, an acceptable dose from each pathway, modeled as if it were the only pathway, would demonstrate that the composite dose would also be acceptable.

OPPD assumed that 10 percent of the fuel rods fail releasing the fission product inventory in the fuel rod gap. It is assumed that 10 percent of the core inventory of iodines and noble gases is in the fuel rod gap. A radial peaking factor of 1.8 was applied. In addition, localized heating is assumed to cause 1 percent of the fuel to melt, causing 100 percent of the noble gases and 25 percent of the iodines contained in the melted fuel to be released to the containment. For the secondary release case, 100 percent of the noble gases and 50 percent of the iodines contained in the melted fuel are assumed to be released to the secondary. Because the containment sump pH is not controlled for a CREA, OPPD conservatively assumed that the chemical form of the iodine released to the environment would be 97 percent elemental and 3 percent organic. These assumptions are consistent with RG 1.183 and are acceptable.

For the containment leakage case, the fission products released from the fuel are assumed to be instantaneously and homogeneously mixed in the containment free volume. The containment is projected to leak at its design leakage of 0.1 percent of its contents by weight per day for the first 24 hours and then at 0.05 percent for the remainder of the 30-day accident duration. This leakage is not collected and enters the environment without processing.

For the secondary release case, the fission products released from the fuel are assumed to be instantaneously and homogeneously mixed in the reactor coolant system and transported to the secondary side via primary-to-secondary leakage at the TS value of 1 gpm at STP. At FCS, the SG tubes remain covered for the duration of the event. An iodine partition factor of 100 was assumed in the SGs. There is no mitigation of the released noble gases. A LOOP is conservatively assumed to occur at T=0, rendering the main condenser unavailable. With the main condenser unavailable, the plant is cooled down by releases of steam to the environment via ADVs and/or the MSSVs. Releases from the SGs will continue for eight hours, at which time shutdown cooling is initiated with the RHR system and the environmental releases are terminated. The accident analysis considers two events—a 2-hour event with a 75 degree/hour cooldown rate, and an 8-hour event based on reaching 300 degrees in 8 hours. The effect of the two cases is to maximize the release in the 2-hour period and the total steam release in the longer period. OPPD asserts that the limiting 2-hour dose will occur at 0-2 hours or 6-8 hours. The noble gas release rate will be highest at the onset of the event, while the iodine and particulate release rate peaks when the secondary releases end at 8 hours. The worst 2-hour χ/Q is used for either case when determining the EAB dose. For the control room and LPZ doses, the 8-hour event releases are considered.

The CREA will cause a safety injection actuation signal (SIAS) at 38 seconds into the event. An additional delay of 44 seconds prior to initiation of control room emergency ventilation is assumed to account for a coincident LOOP. Consistent with the discussion in Section 2.3, the

recirculation filtration is not credited for the first two hours following initiation of emergency ventilation.

Details on the assumptions found acceptable to the staff are presented in Table 1. The resulting doses are presented in Table 2. The estimated doses for the postulated CREA were found to be acceptable.

2.4.6 Main Steam Line Break (MSLB) Radiological Consequences

The accident considered is the complete severance of a main steam line outside containment. The radiological consequences of a break outside containment will bound those results from a break inside containment. Thus, only the break outside is considered with regard to dose. The faulted SG will rapidly depressurize and release the initial contents of the SG to the environment. The rapid secondary depressurization causes an SIAS within 13.6 seconds post break. A reactor trip occurs. OPPD assumes that the faulted SG boils dry in 136 seconds, releasing the entire liquid inventory and dissolved radioiodines through the faulted steam line to the environment. A LOOP is conservatively assumed to occur when the reactor trips. This LOOP renders the main condenser unavailable. With the main condenser unavailable, the plant is cooled down by releases of steam to the environment via ADVs and/or MSSVs on the unaffected SG until RHR cooling is started after the primary temperature drops to 300°F.

No fuel damage is postulated to occur because of the MSLB. OPPD assumes the initial iodine inventory in the RCS and SG to be at the maximum concentrations permitted by TS, 1.0 μCi/g and 0.1 μCi/g, respectively. Two iodine spiking cases are considered. The first assumes that an iodine spike occurred just before the event and the RCS iodine inventory is at 60 μCi/gm dose equivalent I-131. The second case assumes that the event initiates an iodine spike. Iodine is released from the fuel to the RCS at a rate 500 times the normal iodine appearance rate for 8 hours.

The faulted SG is assumed to blow down dry within 136 seconds of the MSLB, releasing all of the iodine initially in the secondary coolant. Although secondary steam would also be released, its contribution to dose is negligible and is not calculated. All radionuclides transferred from the RCS via the primary-to secondary leakage of 1.0 gpm are released to the environment, without holdup, via the faulted SG. The release from the faulted SG continues until the RCS temperature is lower than 212°F. The plant is cooled down by the release of steam from the intact SG. The accident analysis considers two events—a two-hour event with a 75-degree per hour cooldown rate, and an 8-hour event based on reaching 300 degrees in 8 hours. Radioactivity releases from the intact SG terminate at this time. The plant is cooled down further using the RHR system. The RCS reaches 212°F about 5 hours after the event for the first case, and about 11 hours after the event for the second case.

The rate at which iodine activity in the unaffected SG is released to the environment is based on the steaming rate and the iodine partition factor of 100. Since OPPD assumed that all primary-to-secondary leakage is into the affected SG, only the initial activity in the SG is available for release.

The MSLB will cause an SIAS at 14 seconds into the event. An additional delay of 44 seconds prior to initiation of control room emergency ventilation is assumed to account for a coincident LOOP. Consistent with the discussion in Section 2.3, the recirculation filtration is not credited for the first two hours following initiation of emergency ventilation. Details on the assumptions found acceptable to the staff are presented in Table 1. The resulting doses are presented in Table 2. The estimated doses for the postulated MSLB were found to be acceptable.

2.4.7 Steam Generator Tube Rupture (SGTR) Radiological Consequences

The accident considered is the double-ended rupture of a single SG tube. The radiological consequences of this event are caused by the transfer of radioactive reactor coolant to the secondary side of the SG and the subsequent release of radioactive materials to the environment. OPPD assumes that the primary-to-secondary break flow following a SGTR results in a depressurization of the RCS and a reactor trip at about 412 seconds after the event. A LOOP is conservatively assumed to occur when the reactor trips. With the main condenser unavailable due to the LOOP, the plant is cooled down by releases of steam to the environment via ADVs and/or MSSVs. Releases prior to the trip are via the main condenser air ejector. No fuel damage is postulated to occur as of result of a SGTR. OPPD assumes the initial iodine inventory in the RCS and SG to be at the maximum concentrations permitted by TSs, 1.0 μCi/g and 0.1 μCi/g, respectively. Two iodine-spiking cases are considered. The first assumes that an iodine spike occurred just before the event and the RCS iodine inventory is at 60 μCi/gm dose equivalent I-131. The second case assumes the event initiates an iodine spike. Iodine is released from the fuel to the RCS at a rate 335 times the normal iodine appearance rate for 8 hours. The initial secondary side liquid and steam activity is relatively small, and its contribution to the total dose is negligible and is not calculated.

Releases Prior to Reactor Trip

Before the reactor trip at 412 seconds, activity in the steam from both SGs is released to the environment from the main condenser air ejector. All steam noble gases and organic iodine are released to the environment. OPPD assumed an elemental iodine partitioning factor of 2000 in the main condenser and air ejector. Although the staff has not established guidance on partitioning in the main condenser and air ejector, it finds this value acceptable for FCS, given (1) the relatively short duration of the release, (2) the relative magnitude of the break flow during this period in comparison to the 2-hour break flow, (3) the duration of the coincident iodine spike, and (4) that the main condenser and air ejector are operating in a normal manner during this brief period.

Ruptured SG

In the ruptured SG, a large amount of primary coolant is released via the tube break until the primary system is fully depressurized. A significant portion of this break flow flashes to vapor when it encounters the lower pressure of the secondary side and enters the SG steam space. All of the noble gases in the break flow and those iodines in the flashed flow are assumed to be immediately available for release without retention. The iodine in the non-flashed break flow mixes uniformly in the SG bulk water and is subsequently released into the steam space and to the environment at a rate based on the steaming rate for the SG and the iodine partition factor.

At FCS, the steam generator tubes remain covered in this transient. All activity releases from the faulted SG cease when it is isolated at 120 minutes after the accident.

Intact SG

The activity from the intact SG is due to normal primary-to-secondary leakage at the maximum value allowed by TSs and the steam release from the SG. All of the iodine activity in the primary-to-secondary leakage is assumed to mix uniformly with the SG bulk water and is released in proportion to the steaming rate for the SG and an iodine partition factor of 100. At FCS, the steam generator tubes remain covered in this transient. Noble gases are released without holdup. Releases from the intact SG will continue for 8 hours, at which time shutdown cooling is initiated with the RHR system and the environmental releases are terminated. The accident analysis considers two events—a 2-hour event with a 75-degree per hour cooldown rate, and an 8-hour event based on reaching 300 degrees in 8 hours. For the control room and LPZ doses, the 8-hour event releases are considered.

The SGTR will cause an SIAS at 426 seconds into the event. An additional delay of 44 seconds prior to initiation of control room emergency ventilation is assumed to account for a coincident LOOP. Consistent with the discussion in Section 2.3, the recirculation filtration is conservatively not credited for the first two hours following initiation of emergency ventilation. Details on the assumptions found acceptable to the staff are presented in Table 1. The resulting doses are presented in Table 2. The estimated doses for the postulated SGTR were found to be acceptable.

2.4.8 Waste Gas Decay Tank (WGDT) Failure Radiological Consequences

The OPPD licensing basis includes analyses of the radiological consequences of a rupture of a WGDT. A WGDT is used to store processed radiogases removed from the RCS to allow for radioactive decay before the controlled release to the environment. RG 1.183 does not address this particular DBA. Guidance is provided in Branch Technical Position ETSB 11-5.

The WGDT is assumed to contain the gaseous radioactive materials (e.g., fission products and RCS activation products) resulting from operation at 100 percent for a prolonged period with 1.0 percent failed fuel. Prior to the event, it is assumed that the FCS is placed into cold shutdown, and that all noble gases are transferred from the RCS to the WGDT as a part of that shutdown, thereby maximizing the tank's contents. RCS degassing and volume control tank (VCT) purging is not performed, and radioactive decay is not assumed. The WGDT activity is released instantaneously as a puff release to the environment via the auxiliary building stack.

The analysis for this accident assumes that the control room remains in its normal alignment (1,000 cfm normal intake flow) for the duration of the event. Details on the assumptions found acceptable to the staff are presented in Table 1. The resulting doses are presented in Table 2. The estimated doses for the postulated WGDT failure were found to be acceptable.

2.4.9 Liquid Waste Tank (LWT) Failure Radiological Consequences

The OPPD licensing basis includes analyses of the radiological consequences of an airborne

release associated with a LWT failure. RG 1.183 does not address this particular DBA. The postulated accident sequence is FCS-specific and intended to be conservative. Liquid waste tanks are used to store liquid wastes prior to and following processing. For this particular sequence, OPPD has opted to evaluate the airborne releases associated with a filtration and ion-exchanger (FIX) vessel failure. The FIX system is used to treat high-level liquid waste generated in FCS. The influent for the FIX is the contents of the waste holdup tanks. Effluent from the FIX is discharged to the monitor tanks. The FIX tanks are located in a room that is constructed to control leakage from any FIX. Should this occur, the spillage would be pumped to another waste storage tank.

In this accident sequence, a fraction of the halogen activity in the resin of the FIX is assumed to be released into the water upon tank rupture. A portion of the activity in the water becomes airborne and is released to the environment.

The worst-case input to the waste holdup tanks is untreated primary coolant waste. Since the tanks also receive RCS letdown waste that has been treated by the purification demineralizers, the feed stream to the FIX is the weighted-average of the treated and untreated primary coolant. The RCS activity is assumed to correspond to 1.0 percent failed fuel at the maximum design flow of 50 gpm for a sufficient period for equilibrium activity to be achieved. Upon tank rupture, 10 percent of the halogen inventory in the FIX is assumed to be instantaneously and non-mechanistically transferred to the water. Of this transferred inventory, 10 percent of the halogens are assumed to become airborne and are released to the environment as a puff release via the radwaste building exhaust nozzle. All of the noble gases created from decay of the halogens are assumed to become airborne.

The analysis for this accident assumes that the control room remains in its normal alignment (1,000 cfm normal intake flow) for the duration of the event. Details on the assumptions found acceptable to the staff are presented in Table 1. The resulting doses are presented in Table 2. The estimated doses for the postulated LWT failure were found to be acceptable.

2.5 Other Radiological Analyses

2.5.1 Environmental Qualification (EQ) Impact

Consistent with the guidance in Paragraphs 1.3.5 and 1.3.6 of RG 1.183, OPPD did not analyze the impact of the AST on the integrated radiation doses to equipment subject to environmental qualification criteria. This guidance calls for an assessment only if assumptions or inputs of the EQ analyses are affected by the plant modification associated with the AST implementation. The guidance states that the licensees need not assess the effect of increased cesium on the sump integrated doses pending the resolution of a generic safety issue (GSI). OPPD has proposed no plant modifications that would warrant the re-analysis. Additionally, the GSI has not been resolved at this time. If the NRC should determine that all licensees need to address the increased cesium issue, a generic communication will be issued, as appropriate.

2.5.2 Impact on Emergency Planning Radiological Assessment Methodology

OPPD has stated that evaluations will be performed to determine the impact of the AST on the current emergency radiological assessment methods. The staff supports the use of the

updated data in such assessments. Changes to these methods will need to be evaluated pursuant to the change provisions of 10 CFR 50.54(q) to determine whether further staff review is necessary.

2.5.3 Impact of Increased Particulates on Containment Fan Cooler Units

The AST increases the amount of aerosol particulates. Such an increase could have the potential for degrading the performance of containment fan cooler units. As noted by OPPD, the NRC staff previously considered this impact in its review of the AST implementation at Indian Point Unit 2, and concluded that the impact would be small because ample water would be available in the containment atmosphere to wash down the fan cooler coils. OPPD performed a scoping analysis and determined that for the parameters supporting this conclusion, the FCS values are comparable to those used in the Indian Point evaluation.

2.5.4 Technical Support Center (TSC) Habitability Evaluation

OPPD considered the impact of the AST on the habitability of the FCS TSC and originally determined that the post-LOCA dose could exceed 5 rem total effective dose equivalent (TEDE) after about 28 days. Guidance regarding the habitability of the TSC called for the projected dose to be less than 5 rem TEDE for 30 days. Since the submittal was made, OPPD took additional action to resolve this concern and the estimated dose now meets the acceptance criteria. OPPD updated their submittal by letters dated October 17 and November 2, 2001. Since the habitability acceptance criterion is now met, the proposed implementation of the AST will not adversely impact the habitability of the TSC.

2.5.5 Post-Accident Personnel Dose Impacts

OPPD performed a scoping analysis comparing the previous TID14844 and AST core inventories to determine the impact, if any, on vital area access, as required by NUREG-0737. This scoping analysis showed that the AST inventory was lower for a majority of the isotopes in comparison to TID14844. In addition, the FCS vital access dose calculations include a safety factor of 10 percent. Based on this scoping analysis, OPPD concluded that the AST doses will be bounded by the current TID14844 calculations. OPPD stated, during a November 27, 2001, telephone conversation with the NRC staff, that this conclusion has since been confirmed. An AST benchmarking study reported in SECY-98-154 reported that results of analyses based on TID14844 would be limiting for periods up to one to four months, after which time the AST results would become limiting. The staff finds OPPD's evaluation of post-accident dose impacts to be acceptable, based on (1) the results of OPPD's scoping analysis and subsequent confirmation of those results, (2) the conclusion of the staff's benchmarking study, and (3) the fact that emergency vital area accesses will occur within the first one-to-two weeks post-accident when the original source term is more limiting.

2.6 Proposed TS Changes

The letter dated February 7, 2001, requested several changes to the TSs. Our evaluation of these changes is provided below.

2.6.1 TSs 2.3 and 3.6

The proposed changes to TS 2.3, "Emergency Core Cooling System," and TS 3.6, "Safety Injection and Containment Cooling Systems Tests," consist of increasing the amount of TSP in the containment sump from 110 cubic feet to 126 cubic feet. TSP is used to control sump pH after a LOCA at a value equal to or higher than 7. Increasing the amount of TSP is required to account for the hydrochloric and nitric acids formed in the containment by radioactivity released from a damaged core. The licensee performed calculations which indicated that the proposed increase of TSP will be sufficient to counteract the effect of the containment acids and maintain the sump water alkaline.

Maintaining sump water alkaline is needed for preventing dissolved radioactive iodine from being released to the containment atmosphere during the recirculation phase of the containment sprays. Most of the iodine leaves the damaged core in an ionic form which is readily dissolved in the sump water. However, in an acidic environment, some of it becomes converted into elemental form, which is much less soluble. This causes its re-evolution. An additional reason for maintaining high pH in sump water is to minimize corrosion damage of the stainless steel components, which could experience stress corrosion cracking in low pH water.

After a LOCA, the containment sump is mostly filled with water coming from the primary system containing boric acid; its pH is, therefore, acidic. One method for making it basic is to place a sufficient amount of TSP in the containment sump baskets. When the containment is filled with water, the TSP dissolves from the containment sump baskets and buffers the water in the containment at a value higher than 7. The amount of TSP should be proportional to the amounts of acids in the containment water. The current amount of TSP in the Fort Calhoun plant was calculated assuming the presence of boric acid only. However, recent LOCA analyses have indicated that there are additional sources of containment acids which were not accounted for previously. These acids come from two sources: hydrochloric acid (HCL) produced by radiolytic decomposition of cable jacketing and nitric acid (HNO_3) synthesized in the radiation field existing in the containment. The licensee calculated the amounts of these acids using the methodology described in NUREG/CR-5950, "Iodine Evolution and pH Control." It determined that during a LOCA event at the Fort Calhoun plant, 535.5 mol-g of HCl and 470.6 mol-g of HNO_3 are formed. Both of these acids are strong acids and can be assumed to be completely ionized. The licensee made an assumption that one mol-g of TSP neutralized one mol-g of acid. Although this assumption is not derived from the exact mechanism of buffering by TSP, it is still very conservative. The resulting amount of the additional TSP was 88 pounds, which corresponds to approximately 16 ft^3 TSP. The staff reviewed the licensee's calculation and performed an independent evaluation. The staff found that the licensee conservatively determined the additional amount of TSP needed for neutralizing the effect of hydrochloric and nitric acids on the sump pH.

The staff evaluated the licensee's proposed modification of the TS, addressing the increase of the amount of TSP in the sump, which was needed to account for the production of hydrochloric and nitric acids in the containment sump after a LBLOCA. Based on the results of this evaluation, the staff concludes that the amount of TSP determined by the licensee will ensure that the sump pH in the post-LOCA environment will remain in an alkaline range and is consistent with the assumptions made in the revised analysis.

2.6.2 TS 2.8

OPPD proposed changes to TS 2.8.2(4) and 2.8.3(5) and the associated Bases. For TS 2.8.2(4) and 2.8.3(5), requirements would be added to place the control room ventilation system in operation in the filtered air mode during refueling operations in the reactor containment building or SFP. For TS 2.8.3(5), a SFP area radiation monitor would be placed in operation, and the specification that requires a ventilation isolation actuation signal (VIAS) with two radiation monitors to be operable during refueling operations in the SFP area would be deleted. These changes are necessary to ensure that the initial conditions assumed in the revised fuel handling and heavy load drop accident analyses would be met. The analyses for both of these accidents assume that the control room ventilation system is in the filtered air mode prior to the event.

2.6.2.1 TS 2.8.2(4) Refueling Operations - Containment

During core alterations and refueling operations in the containment, the proposed change to TS 2.8.2(4) adds a requirement for the control room ventilation system to be in operation and in the filtered exhaust air mode. The licensee stated that this is a conservative action to reduce control room operator exposure to the effects of a fuel handling accident in containment. This is an additional restriction to the TS that will help reduce control room operator exposure if a fuel handling accident were to occur. Therefore, the staff finds the proposed change to be acceptable.

2.6.2.2 TS 2.8.3(5) Refueling Operations - SFP

Currently, a VIAS is initiated by an 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. This requirement is met when the containment/auxiliary building stack swing monitor and the auxiliary building stack radiation monitor are operable. In addition, one manual actuation channel is required to be operable.

The current basis for TS 2.8.3(5) is to ensure the control room ventilation system is operated in the filtered air mode upon receipt of a VIAS. Other functions of the VIAS are to initiate the closure of the containment pressure relief, air sample, and purge system valves, if open. This action prevents release of significant radionuclides from the containment to the environment. The containment penetrations providing direct access to the environment are required to be closed, or capable of being closed by an operable VIAS in accordance with TS 2.8.2(1). 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. The above functions, except for control room ventilation operation in the filtered mode, are not required to mitigate the consequences of a fuel handling accident in the SFP area, as this area is outside of containment; therefore, they are not required to be operable.

The proposed change to TS 2.8.3(5) will delete the requirement for the VIAS to be operable

with two radiation monitors operable, and add a requirement for the control room ventilation system to be in operation and in the filtered air mode and a SFP area radiation monitor to be in operation during refueling operations in the SFP area. As stated above, since the other functions of the VIAS are not required for a fuel handling accident in the SFP area, the requirement for the VIAS and for the two radiation monitors that provide a signal for the activation of VIAS to be operable can be eliminated.

Based on the staff's review of the information provided in the licensee's submittal and the staff's review of the FCS USAR, the staff concludes that reasonable assurance of safety will be maintained if the requested changes are granted. Therefore, the staff finds it acceptable for the licensee to place the control room ventilation system in the filtered air mode, and to delete the specification that requires a VIAS and two radiation monitors to be operable, during refueling operations in the SFP area. These changes are consistent with the associated revised accident analysis.

2.6.3 TS 3.16

OPPD has proposed changes to TS 3.16, "Residual Heat Removal System Integrity Testing," and the associated Bases to: (1) require performance of an internal leakage test on the RHR system, and (2) increase the acceptance criterion for the sum of leakages addressed by this TS from 1243 to 3800 cc/hour.

The requirement for testing of internal leakage in the RHR assesses backleakage from the contaminated systems to the SIRWT, a possible pathway for containment sump water leakage to the environment that is not currently addressed by the TS. The DBA LOCA analysis considers the dose consequences from leakage from systems that re-circulate containment sump water outside of the containment during a LOCA. The updated LOCA analyses conservatively assume the initial leakage to be 3,800 cc/hour. The proposed changes assure that the initial conditions assumed in the accident analyses will be met. The increase in consequences associated with the increase leakage criterion was found to be acceptable in Section 2.4.1 of this SE. Therefore, the proposed changes are acceptable.

2.7 Change to Bases Sections

Bases Sections 2.3, 2.8.2(4), 2.8.3(5), 3.16, and 3.6 have been revised to reflect the proposed TS changes. The staff has reviewed these Bases sections and has no objections to the proposed Bases changes.

2.8 Conclusion

This licensing action is considered to be a full implementation of the AST. With the approval of this amendment, the AST, the TEDE criteria, and the analysis methods, assumptions and inputs become the licensing basis for the assessment of radiological consequences of DBAs at FCS. All future radiological analyses done to show compliance with regulatory requirements shall use this approved licensing basis.

The staff reviewed the assumptions, inputs, and methods used by OPPD to assess the radiological impacts of the proposed changes. The staff finds that OPPD used analysis methods and assumptions consistent with the conservative guidance of RG 1.183, with the exceptions discussed and accepted earlier in this SE. The staff compared the doses estimated by OPPD to the applicable acceptance criteria and, where applicable, to the results estimated by the staff in its confirmatory calculations. The staff finds, with reasonable assurance, that the licensee’s estimates of the total offsite and control room doses due to DBAs will comply with the requirements of 10 CFR 50.67 and the guidance of RG 1.183.

The staff finds reasonable assurance that FCS will continue to provide sufficient safety margins with adequate defense-in-depth to address unanticipated events and to compensate for uncertainties in accident progression and in analysis assumptions and parameters. The staff concludes that the proposed AST implementation and the associated TS changes are acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Nebraska State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (66 FR 22031). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental

impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Attachments: Table 1 - DBA Analysis Assumptions
Table 2 - Analysis Results

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Date: December 5, 2001

TABLE 1

DBA ANALYSIS ASSUMPTIONS

Assumptions Common to One or More Analyses

Reactor power, MWt	1530
RCS mass, lb	264,900
RCS specific activity, µCi/gm dose equivalent I-131	1.0
RCS to secondary leak rate, gal/min	1.0
Iodine spike appearance rate (does not include multiplier), uCi/sec	
I-131	5570
I-132	4310
I-133	8700
I-134	4320
I-135	5888
Iodine spike duration, hrs	8
Control room volume, ft^3	45,100
Normal ventilation makeup flow, cfm	1000
Emergency ventilation system	
Filtered air makeup, cfm	1000/2000
Filtered recirculation, cfm	1000
Unfiltered inleakage, cfm	38
Filter efficiency, iodine, percent	99
Filter efficiency, particulate, percent	99
Control room switchover from normal to emergency after receipt of SI signal, seconds	varies
Control room emergency ventilation delay during LOOP, seconds	44
Time to repair recirculation damper, hours	2
Control room breathing rate, m^3/sec	3.47E-4
Control room occupancy factors	
0-24 hours	1.0
1-4 days	0.6
4-30 days	0.4

Offsite χ/Q, sec/m3

Period	0-2 hr	0-8 hr	8-24 hr	1-4 d	4-30 d
EAB	2.56E-4				
LPZ	2.51E-5	7.29E-6	4.83E-6	1.98E-6	5.49E-7

Control room χ/Q, x 10^{-3} sec/m^3

Period	0-2 hr	2-8 hr	8-24 hr	1-4 d	4-30 d
Containment Wall	4.87	4.19	2.11	1.61	1.35
Aux. Bldg. Stack	3.16	2.37	1.16	0.893	0.715
Aux. Bldg. Intake	3.12	2.21	0.958	0.688	0.461
MSSV/ADVs	5.06	4.46	2.08	1.59	1.34
Radwaste Nozzle	1.05	0.904	0.402	0.284	0.227
Room 81 Dome	5.92	4.79	2.36	1.73	1.49
Cond. Evac. Disch.	2.04	1.54	0.712	0.462	0.336
TDAFW Disch.	4.73	3.75	1.88	1.36	1.17

Assumptions for LOCA Analyses

Core Inventory		
Calculated by SAS2H/ORIGEN		
Core release fractions and timing	Table 2.2.3-1 this SER	
Iodine species fraction	Atmosphere	Sump
Particulate/aerosol	95	0
Elemental	4.85	97
Organic	0.15	3
Containment volume, ft^3		
Sprayed		728,700
Unsprayed		321,300
Total		1.05E+6
Duration of release, days		30
Containment mixing rate, unsprayed volumes per hour		4.84
Containment spray lambda, hr^{-1}		See SE Text
Aerosol radius, cm		1E-7 to 0.01
Aerosol size bins		100
	30-1830s	1830-6510s
Aerosol injection rate, g/sec	6.84	56.99
Mean geometric radius, cm	7.5E-6	4.0E-5
Geometric standard deviation	1.56	1.46
Aerosol Density, g/cc	3.7	4.6
Spray fall height, cm		2134

Spray flow rate, gpm	
Injection phase	1885
Recirculation phase	3100
Spray droplet size, micron	900
Containment spray decontamination factor (DF)	
Elemental spray cutoff	200
Particulate spray reduction	n/a
Containment spray timings	
spray initiation, sec	185
Elemental spray cutoff, hrs	DF not reached
Spray terminated, hrs	5.0
Containment release, percent/day	
0-24 hours	0.1
24-720 hours	0.05
Containment vacuum relief flow rate 0-5 seconds, scfs	10
ECCS/SIRWT leak rate, cc/hr	7600
ECCS/SIRWT release start, min	20.4
ECCS/SIRWT release end, days	30
Containment sump volume, gals	314,033

Containment surface χ/Q used for containment leakage

Auxiliary building vent stack χ/Q used for containment vacuum release and ECCS/SIRWT

Control room HVAC switchover, sec	0

Assumptions for Fuel Handling Accident Analysis

Number of fuel assemblies in core	133
Number of damaged assemblies	1
Radial peaking factor	1.8
Decay period, hours	72
Fuel rod gap fractions	
I-131	0.16
Kr-85	0.20
Alkali metals	0.24
All others	0.10
Iodine species fractions	
Elemental	0.9985
Organic	0.0015

Particulates	none
Water depth, ft	
Case 1	11.15
Case 2	23
Iodine pool scrubbing factor, effective	
Case 1	20
Case 2	200
Release rate from fuel	puff
Release duration to environment, hours	2
Control room HVAC switchover, sec	0
Containment surface χ/Q used for containment FHA	
Auxiliary building vent stack χ/Q used for fuel pool area FHA	

Assumptions for Heavy Load Drop Analysis

Number of fuel assemblies in core	133
Number of damaged assemblies	133
Radial peaking factor	1.0
Decay period, hours	72
Fuel rod gap fractions	
Iodines	0.05
Noble gases	0.05
Alkali metals	0.05
Iodine species fractions	
Elemental	0.9985
Organic	0.0015
Particulates	none
Containment volume, ft^3	1.05E+6
Containment mixing, percent	50
Water depth, ft	
Case 1	11.15
Case 2	23

Iodine pool scrubbing factor, effective	
Case 1	20
Case 2	200
Release rate from fuel	puff
Release to environment, cfm	
Case 1	50,000
Case 2	5,000
Stack flow (without containment purge), cfm	72,500
Release terminates	
Case 1	73
Case 2	103.5
Control room HVAC switchover, sec	0
Auxiliary building vent stack χ/Q used for fuel pool area FHA	

Assumptions for Seized Rotor Accident Analyses

Reactor coolant mass, lbm	264,900
Primary-to-secondary SG tube leakage, gpm @STP	1.0
Melted fuel fraction	0
Failed fuel fraction	0.01
Fuel rod gap fractions	
I-131	0.16
Kr-85	0.20
Alkali metals	0.24
All others	0.10
Iodine species fractions in gap	
Elemental	0.0485
Organic	0.0015
Particulates	0.95
Radial peaking factor	1.8
Steam generator mass (each), lbm	57,808
Steam releases	submittal; Tables 7.6-2, 7.6-3
Release duration, hours	8
Iodine partition coefficient	100
Particulate carryover fraction	0.0025

Iodine species fractions in release	
Elemental	0.97
Organic	0.03
Particulates	0.0
Control room ventilation switchover, hours	7
Control room recirculation initiated, hours	9
MSSV/ADV χ/Q used for steam generator releases	

Assumptions for Control Rod Ejection Accident Analyses

Radial peaking factor		1.8
Fraction of rods that exceed DNB		0.10
Gap fraction, all nuclide groups		0.10
Fraction of rods that experience melt		0.01
Melt release fraction		
Noble gases		1.0
Iodine (containment release)		0.25
Iodine (secondary release)		0.50
Iodine species fraction	Containment	SG
Particulate/aerosol	0.95	0.0
Elemental	0.0485	0.97
Organic	0.0015	0.03
Fuel release timing		puff
Containment volume, ft^3		1.05E+6
Containment release, percent/day		
0-24 hours		0.1
24-720 hours		0.05
Duration of containment release, days		30
Reactor coolant mass, lbm		264,900
Steam generator mass (each), lbm		57,808
Primary-to-secondary SG tube leakage, gpm @STP		1.0
Steam releases		submittal; Tables 7.6-2, 7.6-3
Release duration, hours		8
Iodine partition coefficient		100

Control room ventilation switchover, sec	38
Containment surface χ/Q used for containment leakage	
MSSV/ADV χ/Q used for steam generator releases	

Assumptions for Main Steam Line Break Analyses

RCS activity	
T/S RCS concentrations, µCi/gm	submittal; Table 4.2-1
Spike appearance rate, µCi/sec	submittal; Table 4.2-2
Spike multiplier	500
Spike duration, hours	8
Pre-incident iodine spike case, µCi/gm	submittal; Table 4.2-2
Fuel damage	none
Primary-to-secondary leakage, gpm @ STP	
Faulted SG	1.0
Intact SG	0.0
Faulted SG blowdown (100 percent)	
Duration, sec	136
Mass released, lbm	159,346
Steam generator mass @, lbm	
Minimum	57,808
Maximum	125,707
Pre-event steam generator activity	submittal; Table 4.2-1
Iodine partition coefficient (intact SG)	0.01
Iodine species fraction	
Elemental	0.97
Organic	0.03
Particulate	0.0
Noble gas holdup	none
Release termination, hours	
Case 1 (fast cooldown)	4.94
Case 2 (prolonged cooldown)	10.94
Control room HVAC switchover, sec	14
Room 81 Domes χ/Q used for faulted SG releases	
MSSV/ADV χ/Q used for intact SG releases	

Assumptions for Steam Generator Tube Rupture Analyses

RCS activity	
T/S RCS concentrations, µCi/gm	submittal; Table 4.2-1
Spike appearance rate, µCi/sec	submittal; Table 4.2-2
Spike multiplier	335
Spike duration, hours	8
Pre-incident iodine spike case, µCi/gm	submittal; Table 4.2-2
Fuel damage	none
Reactor trip, sec	412
Break flow	submittal; Table 7.9-2
Termination of break flow, hours	2.0
Break flow flash fraction	
0-15 min	0.15
15-60 min	0.05
1-2 hrs	0.02
Primary-to-secondary leakage (intact SG), gpm @STP	1.0
Steam generator mass @, lbm	
Intact	57,808
Ruptured	77,947
Initial (all SG)	155,894
Pre-event steam generator activity	submittal; Table 4.2-1
Iodine partition coefficient	
Unflashed portion	0.01
Flashed portion	1.0
Air ejector / condenser	2000
Iodine species fraction	
Elemental	0.97
Organic	0.03
Particulate	0.0
Steam release via condenser	
Timing, sec	0-412
Mass flow rate, lbm/sec	937
Steam release via MSSV/ADV	
Termination, hours	8
Masses	submittal; Tables 7.9-3, 7.9-4, 7.9-5
Noble gas holdup	none
Control room HVAC switchover, sec	426

SJAE χ/Q used for SJAE releases
MSSV/ADV χ/Q used for SG releases

Assumptions for Waste Gas Tank Accident Analyses

Inventory	submittal; Table 7.10-2
Failed fuel fraction	0.01
RCS noble gases degassed to WGDT immediately following SD	
Release duration	Puff
Auxiliary building stack χ/Q used	
Control room HVAC switchover, sec	not credited

Assumptions for Liquid Waste Tank Accident Analyses

Inventory	submittal; Table 7.10-2
Source	Untreated & treated primary coolant
Failed fuel fraction	0.01
Flow rate to FIX, gpm	50
Release duration	Puff
Fraction of halogens released from resin to sump	0.1
Fraction of halogens released from sump to building atmosphere	0.1
Radwaste building nozzle χ/Q used	
Control room HVAC switchover, sec	not credited

TABLE 2

ANALYSIS RESULTS

	rem, TEDE	rem, TEDE	rem, TEDE
Loss of coolant accident (25)	2.5	0.5	4.5
Fuel handling accident (6.3)			
Pool area	1.5	0.5	1.5
Containment	1.5	0.5	1.5
Heavy drop in containment (6.3)			
23 feet water over	3.5	0.5	1.5
11 feet water over	5.0	0.5	2.0
Seized rotor accident (2.5)	0.5	0.5	4.7
Control rod ejection accident (6.3)	2.0	0.5	3.0
Main steam line break			
Pre-incident spike (25)	0.5	0.5	2.5
Co-incident spike (2.5)	1.5	0.5	2.5
Steam generator tube rupture			
Pre-incident spike (25)	1.5	0.5	1.5
Co-incident spike (2.5	1.5	0.5	1.5
Waste gas decay tank failure (0.5)	0.14	0.01	0.04
Liquid waste tank failure (0.5)	0.08	0.01	0.32

*With exception of WGDT failure/LWT failure, doses were rounded up to nearest 0.5 rem
( ) = Dose acceptance criterion for EAB and LPZ
Control room acceptance criterion is <5 rem TEDE