November 27, 2006

Mr. J. V. Parrish
Chief Executive Officer
Energy Northwest
P.O. Box 968 (Mail Drop 1023)
Richland, WA 99352-0968

SUBJECT: COLUMBIA GENERATING STATION - ISSUANCE OF AMENDMENT RE: ALTERNATIVE SOURCE TERM (TAC NO. MC4570)

Dear Mr. Parrish:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 199 to Facility Operating License No. NPF-21 for the Columbia Generating Station (CGS). The amendment consists of changes to the Technical Specifications and Final Safety Analysis Report in response to your application dated September 30, 2004, as supplemented by letters dated March 16, September 29, 2005, and March 21, August 7, August 24, and September 11, 2006.

The proposed change will replace the current accident source term used in design-basis radiological analyses with an alternative source term pursuant to Title 10 of the *Code of Federal Regulations,* Section 50.67 (10 CFR 50.67), "Accident source term." Energy Northwest updated the current loss-of-coolant accident, the control rod drop accident, the fuel-handling accident, the main steam line break analyses, and proposed revisions to several technical specifications with an exception. That exception is the Technical Information Document (TID)-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," which will continue to be used as the radiation dose basis for equipment qualification, and radiation zone maps/shielding calculations.

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 by D. Terao for/

Brian J. Benney, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-397

Enclosures:
1. Amendment No. 199 to NPF-21
2. Safety Evaluation

cc w/encls: See next page

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ADAMS Accession Nos.: Pkg ML062610429 (Amendment ML062610440, TS Pg ML063280053)

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ENERGY NORTHWEST

DOCKET NO. 50-397

COLUMBIA GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 199
License No. NPF-21

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

   A. The application for amendment by Energy Northwest (licensee) dated September 30, 2004, as supplemented by letters dated March 16, and September 29, 2005, and March 21, August 7, August 24, and September 11, 2006, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's 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 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, the license is amended by changes to the Technical Specifications and Final Safety Analysis Report as indicated in this license amendment. Paragraph 2.C.(2) of Facility Operating License No. NPF-21 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 199 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance and shall be implemented within 120 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

David Terao, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: November 27, 2006

ATTACHMENT TO LICENSE AMENDMENT NO. 199

FACILITY OPERATING LICENSE NO. NPF-21

DOCKET NO. 50-397

Replace the following pages of the 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. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE	INSERT
ii	ii
iii	iii
1.1-2	1.1-2
1.1-3	1.1-3
3.1.7-1, 3.1.7-2, and 3.1.7-3	3.1.7-1, 3.1.7-2, and 3.1.7-3
3.3.6.1-7	3.3.6.1-7
3.3.6.2-4	3.3.6.2-4
3.3.7.1-2 through 3.3.7.1-5	3.3.7.1-2 through 3.3.7.1-4
3.6.1.3-8 through 3.6.1.3-9	3.6.1.3-8 through 3.6.1.3-9
3.6.4.1-1	3.6.4.1-1
3.6.4.1-2	3.6.4.1-2
3.6.4.1-3	--------
3.6.4.2-1 through 3.6.4.2-4	3.6.4.2-1 through 3.6.4.2-3
3.6.4.3-1	3.6.4.3-1
3.6.4.3-2	3.6.4.3-2
3.6.4.3-3	-------
3.7.3-1 through 3.7.3-4	3.7.3-1 through 3.7.3-3
3.7.4-1	3.7.4-1
3.7.4-2	3.7.4-2
3.7.4-3	-------
3.8.2-1 through 3.8.2-4	3.8.2-1 through 3.8.2-3
3.8.5-1	3.8.5-1
3.8.5-2	3.8.5-2
3.8.8-1	3.8.8-1
3.8.8-2	3.8.8-2
3.9.7-1	3.9.7-1
------	3.9.10-1
5.5-7	5.5-7

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 199 TO FACILITY OPERATING LICENSE NO. NPF-21

ENERGY NORTHWEST

COLUMBIA GENERATING STATION

DOCKET NO. 50-397

1.0 INTRODUCTION

By application dated September 30, 2004 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML042930316), as supplemented by letters dated March 16 and September 29, 2005, and March 21 and September 11, 2006 (ADAMS Accession No. ML052850270, ML050900256, ML060900602 and ML062620329, respectively), Energy Northwest (EN/licensee) requested changes to the Technical Specifications (TSs) (Appendix A to Facility Operating License No. NPF-21) and Final Safety Analysis Report for the Columbia Generating Station (CGS). The supplements dated March 16 and September 29, 2005, and March 21 and September 11, 2006, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published on October 26, 2004 (69 FR 62472), in the *Federal Register*.

The requested change would replace the current accident source term used in design-basis radiological analyses with an alternative source term (AST) pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.67, "Accident source term." Energy Northwest updated the current loss-of-coolant accident (LOCA), the control rod drop accident (CRDA), the fuel handling accident (FHA), and the main steam line break (MSLB) analyses. The requested change would revise the licensing and design basis to reflect the application of full scope AST methodology consistent with the guidance provided in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (with the exception that the Technical Information Document, (TID)-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," will continue to be used as the radiation dose basis of equipment qualification (EQ)) and the associated TS changes. Changes to the TSs are analyzed in this safety evaluation (SE).

The AST methodology allows the licensee to revise the accident source term used in the design-basis radiological consequence analysis. To implement the AST, the requirements for being in MODE 4 requires changes to TS 3.1.7, "Standby Liquid Control System." The request also includes a requirement to maintain the pH of the suppression pool above 7 for a period of 30 days following a LOCA in order to minimize the amount of radioactive iodine released. In addition, the licensee takes credit for drywell spray removal of iodine.

2.0 BACKGROUND/REGULATORY EVALUATION

In December 1999, the U.S. Nuclear Regulatory Commission (NRC) issued a new regulation, 10 CFR 50.67, "Accident Source Term," which provided a mechanism for licensed power reactors to replace the traditional accident source term, used in their design-basis accident (DBA) analyses, with an AST. Regulatory guidance for the implementation of these ASTs is provided in RG 1.183. A licensee seeking to use an AST is required by 10 CFR 50.67 to apply for a license amendment. An evaluation of the consequences of affected DBAs is required to be included with the submittal. EN's application addresses these requirements in proposing to use the AST described in RG 1.183 as the source term in the evaluation of the radiological consequences of the design-basis LOCA, MSLB, CRDA, and FHA at the CGS. As part of the implementation of the AST, the total effective dose equivalent (TEDE) acceptance criteria of 10 CFR 50.67(b)(2) replaces the previous whole body and thyroid dose guidelines of 10 CFR 100.11 and 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 19 as the CGS licensing basis for the DBA LOCA, CRDA, FHA, and the MSLB. Specifically, 10 CFR 50.67(b)(2) states that the NRC may issue the amendment only if the applicant's analysis demonstrates with reasonable assurance that:

(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 0.25 sievert (Sv) (25 rem) total effective dose equivalent (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 0.25 Sv (25 rem) TEDE.

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

The NRC requirement and guidance documents that are applicable to the NRC staff's review of CGS's license amendment request include:

Part 50 of 10 CFR, Appendix A, GDC 26, requires that each reactor have two independent reactivity control systems of a different design while GDC 29 requires that the reactivity control system be capable of accomplishing its safety function in the event of anticipated operational occurrences.

Implementation of an AST involves re-analyzing DBAs according to 10 CFR 50.67, "Accident source term," and applying for a license amendment under 10 CFR 50.90.

GDC 17, "Electric power systems," of 10 CFR, Part 50, Appendix A requires that nuclear power plants have onsite and offsite electric power systems to permit the functioning of structures, systems, and components (SSCs) that are important to safety. The onsite system is required to have sufficient independence, redundancy, and testability to perform its safety function, assuming a single failure. The offsite power system must be supplied by two physically independent circuits that are designed and located so as to minimize, to the extent practical, the likelihood of their simultaneous failure under an operating and postulated accident, and environmental conditions. In addition, this criterion requires provisions to minimize the probability of losing electric power from the remaining electric power supplies as a result of loss of power from the unit, the offsite transmission network, or the onsite power supplies.

GDC 18, "Inspection and testing of electric power systems," requires that electric power systems that are important to safety must be designed to permit appropriate periodic inspection and testing.

Section 50.36 of 10 CFR, "Technical specifications," provides the content required in a licensee's TSs. Specifically, 10 CFR 50.36(c)(3) requires that the TS include surveillance requirements.

Section 50.49 of 10 CFR, "Environmental qualification of electric equipment important to safety for nuclear power plants." Compliance with this rule requires that the safety-related electrical equipment which are relied upon to remain functional during and following the design-basis events be qualified for an accident (harsh) environment. This provides assurance that the equipment needed in the event of an accident will perform its intended function.

NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," which provides a more realistic source term than the TID-14844 source term. This NUREG provides estimates of accident source term that were more physically based and that could be applied to boiling-water reactors (BWRs) and pressurized-water reactors. These source terms are characterized by the composition and the magnitude of the radioactive material, the chemical and physical properties of the material, and the timing of the release to the containment. Equipment dose calculations performed with this NUREG source term were lower than doses calculated with the TID-14844 source term during the gap release and early in-vessel release phases of core degradation.

RG 1.183 also provides guidance to the licensee of operating power reactors on acceptable applications of MSLBs. This RG states that the licensees may use either the AST or the TID-14844 assumptions for performing the required EQ analyses. It further states that no plant modifications are required to address the impact of the difference in source term characteristics (i.e., AST vs. TID 14844) on EQ doses.

NUREG-0933, Issue 187, "The Potential Impact of Postulated Cesium Concentration on Equipment Qualification." The issue states that the Sandia National Laboratories' report, "Evaluation of Radiological Consequences of Design Basis Accidents at Operating Reactors Using the Revised Source Term," dated September 28, 1998, showed that: (1) for equipment exposed to the containment atmosphere, the TID-14844 source term and the gap and in-vessel releases in the AST produced similar integrated doses, and (2) for equipment exposed to suppression pool water, the integrated doses calculated with the AST remain enveloped by those calculated with TID-14844 for the first 145 days post accident for a BWR, including the 30 percent versus 1 percent release of cesium. It was concluded that there was no clear basis to backfit the requirement to modify the design-basis for EQ to adopt the AST. There would be no discernible risk reduction associated with such a requirement. Longer term equipment operability issues associated with severe fuel damage accidents, (with which the AST is associated) could also be addressed under accident management or plant recovery actions as necessary.

NUREG-0800, "Standard Review Plan," Section 15.7.4, provides guidance to the NRC staff for the review and evaluation of system design features and plant procedures provided for the mitigation of the radiological consequences of postulated FHAs.

Standard Review Plan, Section 15.0.1, "Radiological Consequence Analyses Using Alternate Source Term," issued in July 2000, also provides new guidance on acceptable applications of alternative source terms.

Technical Specification Task Force Traveler (TSTF)-51, Revision 2 approved by the NRC on October 13, 1999, provides for the relaxation of some TS requirements during refueling after a sufficient decay period has occurred.

3.0 TECHNICAL EVALUATION

3.1 Accident Dose Calculations

The NRC staff reviewed the technical analyses related to the radiological consequences of DBAs that EN performed in support of this proposed license amendment. EN provided information regarding these analyses in the September 30, 2004, submittal, as supplemented in letters dated on September 29, 2005, and March 21, 2006. The NRC staff met with EN on December 6 through 7, 2005, and held teleconferences on January 5, 2006, January 31, 2006, February 12, 2006, May 9, 2006, and May 25, 2006.

The NRC staff reviewed the assumptions, inputs, and methods used by EN to assess these impacts. The NRC staff did independent calculations to confirm the conservatism of the licensee's analyses. However, the findings of this SE are based on the descriptions of the analyses and other supporting information submitted by EN.

In accordance with the guidance in RG 1.183, a licensee is not required to re-analyze all DBAs for the purpose of the application, just those affected by the proposed changes. EN considered the following DBA events, which the NRC staff considers applicable:

- Loss-of-coolant accident
- Fuel handling accident
- Main steam line break
- Control rod drop accident

The technical evaluation of these events is described below.

3.1.1 Loss-of-Coolant Accident

The objective of analyzing the radiological consequences of a LOCA is to evaluate the design of various plant safety systems. These safety systems are intended to mitigate the postulated release of radioactive materials from the plant to the environment in the event that the emergency core cooling system (ECCS) is not effective in preventing core damage. A LOCA is a failure of the reactor coolant system (RCS) that results in the loss of reactor coolant that, if not mitigated, could result in fuel damage, including a core melt. The primary coolant blows down through the break to the drywell, depressurizing the RCS. As the pressure builds in the drywell, steam and other gases expand into the wetwell. Passing through the suppression pool water, the steam is condensed, thereby reducing the pressure in the wetwell and drywell. A reactor trip occurs and the ECCS actuates to remove fuel decay heat. Thermodynamic analyses, performed using a spectrum of RCS break sizes, show that the ECCS and other plant safety features are effective in preventing significant fuel damage. Nonetheless, the radiological consequence portion of the LOCA analysis assumes that ECCS is not effective and that substantial fuel damage occurs. Appendix A of RG 1.183 identifies acceptable radiological analysis assumptions for a LOCA. The source term and release pathways related to the LOCA are discussed below.

3.1.1.1 Source Term

EN projected the core inventory of fission products using the ORIGEN 2 computer code. The resulting core inventories of dose-significant radionuclides were tabulated in Table 4.4-4 to Attachment 1 of Enclosure 1 of the September 30, 2004, submittal. These inventories are based upon an adjusted plant-specific pre-1995 ORIGEN 2 run. The three adjustments were: (1) a scale factor to bound the power level to 3556 megawatts thermal (Mwt), (2) a correction to increase selected krypton values (based on comparisons to other core inventory tables), and (3) an increase in the activity of longer-lived isotopes. The assumed power level of 3556 Mwt is the licensed power increased by 2 percent to account for measurement uncertainties. The ORIGEN 2 computer code is acceptable to the NRC staff for estimating the core inventory. The core inventory used excluded two cobalt nuclides from the RADTRAD inventory file of 60 nuclides. It also added 8 additional nuclides for a total of 66 nuclides.

Because the list of radionuclides is slightly different from the standard default nuclides in RADTRAD, the NRC staff asked EN to confirm that the most conservative radionuclides were used to determine the source for the CGS shielding studies for the shine doses from external sources to the control room (CR), and inhalation doses both for offsite and CR locations. EN provided this confirmation in the March 21, 2006, supplement.

3.1.1.2 Release Pathways

The release to the environment is assumed to occur through the following pathways:

- Design leakage of primary containment atmosphere.
- Design leakage through main steam line isolation valves (MSIVs).
- Design leakage from ECCS piping and components that recirculate suppression pool water outside of the primary containment.

Under the previous TID-14844 source term assumption of instantaneous core damage, the initial blowdown would also include all of the released fission products, a fraction of which would be retained by the suppression pool water. Under the AST, a substantial fraction of the fission product release occurs after the initial blowdown is complete. As such, EN did not credit any reduction in fission products transferred to the wetwell air space by suppression pool scrubbing, assuming instead a well-mixed wetwell air space and drywell after 2 hours.

EN assumes that a portion of the fission products released from the reactor pressure vessel will be removed by drywell sprays. The sprays are assumed to be initiated at 15 minutes and turned off after 1 day.

3.1.1.2.1 Containment Leakage Pathway

The drywell and wetwell are projected to leak at their design leakage of 0.5 percent of their atmospheric contents by weight per day for the first 24 hours, and 0.25 percent of their atmospheric contents by weight for the remainder of the 30-day accident duration. Leakage from the drywell and wetwell will collect in the free volume of the secondary containment and be released to the environment via ventilation system exhaust or leakage. Following a LOCA, the standby gas treatment system (SGTS) fans start and draw down the secondary containment to create a negative pressure with reference to the environment. The SGTS exhaust is processed through high-efficiency particulate air filter media before being released to the environment. EN states that, prior to the completion of the secondary containment drawdown, the containment leakage is assumed to go directly to the environment. After the 20-minute drawdown period, filtration of the leakage is credited; however, no credit is taken for the holdup in secondary containment.

3.1.1.2.2 Secondary Containment Bypass Leakages

Two sources of containment leakage that bypass secondary containment are MSIV leakage and miscellaneous leakages. The models for these leakages are discussed below. A new limit of 16 standard cubic feet per hour (scfh) per valve, or 64 scfh for four steam lines, at a test pressure of 39.7 pounds per square foot absolute (psia) (25 pounds per square inch gauge (psig)) is proposed in the TS change submitted with the license amendment request (LAR). Since the TS allowable leakage is assessed in units of scfh, and the steam lines are not at standard conditions of temperature and pressure, EN adjusted the assigned flow rates appropriately for the assumed accident conditions.

Credit was taken for natural deposition within the main steam lines. The main steam lines are seismically qualified up to the turbine stop valves. The main steam line piping between the two condensate filter demineralizers is also credited for natural deposition. MSIV leakage was reduced by a factor of two at 24 hours. No credit was taken for the main steam line leakage control system. The operability requirements for this system would be removed as part of the proposed TS changes.

To accommodate a postulated single failure of an MSIV to close, credit for natural deposition was taken for only three of the four steam lines. For the three credited lines, natural deposition was calculated according to AEB-98-03, "Assessment of Radiological Consequences for the Perry Pilot Plant Application Using the Revised (NUREG-1465) Source Term," dated December 9, 1998. A modified Bixler approach for gaseous iodine removal was used. The Bixler model is taken from NUREG/CR-6605, "RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation," dated April 1998. The Bixler model was modified by adopting the AEB-98-03 well-mixed flow expression for gaseous iodine removal. Proposed credit for organic iodine removal is discussed below.

The second source of bypass leakage, miscellaneous leakage paths, was assumed to equal the proposed TS limit of 0.04 percent primary containment volume per day at peak accident pressure. The supporting LOCA analysis was based on this limit for the first 24 hours. Consistent with the treatment of MSIV leakage, this leakage value was reduced by a factor of 2 at 24 hours.

The NRC staff finds that the bypass models discussed above are consistent with the broad guidance in Appendix A of RG 1.183 and the previously accepted methodology provided in AEB-98-03 and is, therefore, acceptable to the NRC staff.

3.1.1.2.2.1 Main Steam Deposition of Organic Iodine

Calculation, NE-02-04-05, "Columbia Offsite and Control Room Doses for LOCA Using AST and NRC Methods," proposes the use of a "Modified Bixler Model for Organic Iodine Removal." During a December 6 and 7, 2005, meeting on the EN AST submittal, the NRC staff discussed this methodology with EN. The NRC staff has determined from the docket information and the discussion of this docketed information that credit for organic deposition cannot be granted at this time, because the removal mechanisms for the deposition of organic iodine are not well understood.

The NRC staff performed a sensitivity study of the impact of crediting organic iodine deposition in the LOCA main steam line deposition analysis. Based upon this study, the NRC staff determined that the assumption of organic iodine deposition, as credited in this analysis, has very little effect on the overall analysis conclusions. This determination is based on plant-specific parameters and is applicable only to CGS. Therefore, the NRC staff has reasonable assurance that without this credit the dose acceptance criterion for the LOCA will be met.

3.1.1.2.3 Engineered Safeguards Features (ESF) Leakage

During the progression of a LOCA, some fission products released from the fuel will be carried to the suppression pool via spillage from the RCS. Post-LOCA, the suppression pool is a source of water for the ESF systems. Since portions of these systems are located outside the primary containment, potential leakages from these systems are evaluated as a radiation exposure pathway. For the purposes of assessing the consequences of leakage from the ESF systems, EN conservatively assumes that all of the radioiodines released from the fuel are instantaneously moved to the suppression pool. This source term assumption is conservative, in that all of the radioiodine released from the fuel is available for both primary containment atmosphere leakage and the ESF system leakage. EN assumes that 10 percent of the iodine in the ECCS leakage becomes airborne and is available for release as 97 percent elemental and 3 percent organic iodine. The release continues for 30 days. The NRC staff finds these assumptions to be consistent with the guidance of RG 1.183 and, therefore, acceptable.

Two sources of potential ESF leakage were included in the release model. The first is ESF system leakage directly into secondary containment. The current design-basis assumes a value of 1 gallon per minute (gpm). Consistent with RG 1.183, this value was increased by a factor of 2. Leakage was assumed to start at 15 minutes after the event.

The second source of potential ESF leakage is into the condensate storage tanks (CSTs). During the operation of high-pressure core spray or reactor core isolation cooling systems aligned to the suppression pool, radiological impact of leakage into the CSTs through the CSTs suctions and test returns has been evaluated.

EN determined the dose contribution from the CSTs to the CR and offsite locations to be approximately 1 percent and 2 percent of the total dose, respectively. EN judged that these doses are not significant and did not include them in the total dose from a LOCA. Based upon EN's analysis for the CST pathways, the NRC staff agrees that they are not significant to the conclusion that 10 CFR 50.67(b)(2) regulatory criteria are met for the current analysis. However, in the EN LOCA calculation, LM-0646, Revision 1, EN states that they conform to RG 1.183, Appendix A, Regulatory Position 5.2. This regulatory position considers the impact of design leakage through valves isolating ESF recirculation systems from tanks vented to the atmosphere. Should circumstances change (for example an increase in the operational or assumed leakage to the CSTs) such that the dose from these pathways either increases or becomes significant to the conclusions derived from the total dose from a postulated LOCA, the analyses should be updated to include the CST dose pathway.

3.1.1.2.2.2 Main Steam Isolation Leakage Control System

The original design function of the main steam isolation valve leakage control (MSIVLC) system was to minimize the release of fission products via the main steam lines that could potentially bypass containment and the SGTS after a LOCA. The MSIVLC system performed this function by directing MSIV leakage to the SGTS. This leakage was directed to the SGTS by a blower that served to maintain the pressure in the steam lines negative, with respect to atmosphere. The routing of this leakage to the SGTS provided for filtration of MSIV leakage and its exhaust via the plant stack.

The MSIV leakage in the AST LOCA dose model is assumed to flow directly to the environment without credit for SGTS filtration. Additionally, the MSIV leakage was analyzed as a release from the turbine generator building exhaust, and this provides a conservative χ/Q compared to a release via the plant stack. EN is planning to deactivate the MSIVLC system during the implementation of the approved AST LAR.

3.1.1.3 Secondary Containment Drawdown

Containment integrity is ensured in part by TS 3.6.4.1, “Secondary Containment” and its associated surveillance requirements. SR 3.6.4.1.4 demonstrates that the SGT has the capability of drawing down the secondary containment to negative 0.25 inches w.g. in a two minute period of time. Changes which reduce containment integrity would be detected by increases in draw down time in successive testing until ultimately the test would fail. SR 3.6.4.1.5 demonstrates that the SGT is capable of maintaining a negative pressure in the secondary containment for a period of time at a reduced SGT fan flow rate of 2240 cfm. An increase in inleakage above the fan flow rate would cause the test to fail and indicate that integrity had degraded. The licensee states in its current TS Bases document “[t]he internal pressure of the SGT system boundary [secondary containment] is maintained at a negative pressure of 0.25 inches water gauge when the system is in operation, which represents the internal pressure required to ensure zero exfiltration of air from the building using the 95%

meteorological data." SR 3.6.4.1.5 confirms that the SGT can maintain the negative pressure of 0.25 inches w.g. for an extended period with reduced SGT system flow under benign test conditions and provides reasonable assurance that the negative pressure can continue to be maintained under adverse conditions when the SGT system would be operating at its full design flow of 4800 cfm.

The licensee submitted an analysis based on a GOTHIC model of the secondary containment to evaluate the draw down time under accident conditions considering the effects of adverse temperature and wind conditions. The NRC staff did not perform a comprehensive review of this analysis although it noted that the licensee's analysis: (1) considered heat transfer from the primary to secondary containment, (2) assumed adiabatic boundary conditions for the surface of the secondary containment structure exposed to the outside environment, (3) considered secondary containment inleakage, (4) considered heat loads generated within the secondary containment, (5) considered fan performance characteristics in evaluating the depressurization of the secondary containment, (6) considered the assumption of loss of offsite power coincident with a LOCA in a manner consistent with SRP 6.2.3 "Secondary Containment Functional Design." The analysis predicted a worse case draw down time of 16 minutes. The licensee increased the draw down time to 20 minutes for use in the LOCA analysis.

The draw down time established through the surveillance programs is 2 minutes based on SR 3.6.4.1.4. The NRC staff finds that the use of 20 minutes as the draw down time in the LOCA analysis is acceptable because it considers the potential increase in draw down time due to accident and meteorological conditions and because it is more conservative than that determined through draw down testing.

3.1.1.4 Reactor Building Volume

The reactor building has free volume for dilution, but EN effectively did not credit it for holdup. For modeling purposes, the SGTS was assumed to have a flow rate of 5,000 cubic feet per minute (cfm). Throughout the accident, EN assumed the reactor building volume to be approximately equal to the assumed volumetric flow from the SGTS in 1 minute. Therefore, a small amount of delay is credited by the licensee.

The NRC staff performed an assessment to determine the impact of the small delay and of not modeling the range of SGTS flow rates allowable by TSs. The methodology used by the NRC staff did not credit mixing or holdup, and used design SGTS flow rates with uncertainty. The NRC staff results did not differ from the EN results significantly, therefore, the NRC staff concludes that the results obtained by the licensee are acceptable.

After 20 minutes, the SGTS filter efficiency for all forms of iodine and for particulates is 99 percent. The filter efficiency is reduced to an effective value of 98 percent based on a filter bypass of 50 cfm.

3.1.1.5 Control Room

EN evaluated the dose to the operators in the CR. Both CR remote intakes are normally open. Following a design-basis LOCA, the CR emergency filtration (CREF) system is automatically actuated by a high drywell pressure, low-low reactor water level or high radiation reactor building exhaust. The CR local intake is automatically secured and the CR pressurization process begins. Both trains of the CREF system receive a start signal and one or both start, depending on whether a single failure of one train was postulated.

The CR volume models the intake of activity from the environment for the purpose of calculating the dose to the CR operators. For the licensing basis case, one CREF train was assumed to fail at time zero, leaving one train operating at 800 cfm. The assumed CREF filter efficiencies were 95 percent for the gaseous iodine species and 99 percent for the particulates. The unfiltered inleakage for the single-CREF train scenario was 50 cfm. The CR exit flow rate is the sum of filtered and unfiltered incoming flow rates.

From a single-failure perspective, the assumption of a single failure in the CREF system was conservative since this failure was analyzed as occurring simultaneously with the postulated single failure of an MSIV to close. The dose consequences associated with a single failure of an MSIV to close, bound the consequences associated with a single failure of the CREF and the two failures are independent. Nonetheless, for conservatism, the mitigation of the LOCA with credit for only one CREF train is presented as the licensing basis case.

EN evaluated two additional cases. In these cases, both CREF trains were assumed to start as designed. In the first case, EN assumed that the CR operator secured one of the two trains 8 hours after the start of the accident. In the second case, EN assumed that both trains operated for the 30-day duration of the accident. The two-train filtered intake flow rate of 1300 cfm and an unfiltered inleakage of 75 cfm were used for these cases. EN determined that the CR dose calculated for both of these scenarios is bounded by the single-train licensing basis case discussed above. Securing a CREF train (when two trains are in operation following a design-basis LOCA) before 8 hours could increase the dose to the operator. To preclude this undesirable operator action, the appropriate plant procedure(s) will be revised to prohibit the securing of a CREF train within the first 10 hours of the design-basis LOCA.

The doses calculated in this AST evaluation are based on the limiting combinations of unfiltered leakages and filtered intake flows coupled with conservatively selected χ/Q s.

3.1.1.6 LOCA Review

The NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological impacts of the proposed changes. The assumptions and parameters are in Table 1 of this SE. Based upon the information provided by EN, the NRC staff concludes that the licensee used analysis methods and assumptions consistent with the guidance of RG 1.183. The NRC staff compared the radiation doses estimated by the licensee to the 10 CFR 50.67(b)(2) acceptance criteria and to the results estimated by the NRC staff in its confirmatory calculations. The NRC staff finds, with reasonable assurance, that the licensee's estimates of the exclusion area boundary (EAB), low-population zone (LPZ), and CR doses for the LOCA will continue to comply with the criteria.

3.1.2 Fuel Handling Accident

This accident analysis postulates that a spent fuel assembly is dropped during refueling. Three cases were evaluated. A drop of a fuel assembly in the reactor vessel cavity over the reactor core, and in the fuel transfer area (between the reactor vessel and the spent fuel pool) or over the spent fuel pool.

The drop of a fuel assembly in the reactor vessel cavity over the reactor core during refueling is the DBA worst case. At this location, the maximum drop (free fall distance) is approximately 34 feet and fuel pin damage is postulated to occur to both the dropped assembly and to some portion of those assemblies impacted in the reactor core.

The extent of damage for both cases is calculated based on the free fall distance and the resulting kinetic energy of the dropped assembly. This drop is postulated to damage 250 pins (based upon a fuel assembly with an 8 x 8 fuel pin array).

The fission product inventory in the core is largely contained in the fuel pellets that are enclosed in the fuel rod clad. However, the volatile constituents of this inventory will migrate from the pellets to the gap between the pellets and the fuel rod cladding. The fission product inventory in the fuel rod gap of the damaged fuel rods is assumed to be instantaneously released because of the accident. Fission products released from the damaged fuel are decontaminated by passage through the pool water, depending on their physical and chemical form. The fission products released from the pool are assumed to be released to the environment without credit for reactor building filtration, holdup or dilution. The CR was modeled without taking credit for automatic system actuation and the no credit is taken for the reactor building or the SGTS. Therefore, the normal outside air makeup flow of 1100 cfm continues for the duration of the event and no credit is taken for CREF filters.

The NRC staff questioned why EN did not factor the impact of ingress and egress as stated in RG 1.183 into the total unfiltered inleakage into the CR and whether the value of 1100 cfm was appropriate. EN's response, dated March 21, 2006, to NRC Question 19, stated that the 10 cfm for ingress and egress, and increasing the 1100 cfm by 50 percent would not impact the radiological dose results by greater than 0.9 percent. EN also presented the results of a study that calculated the dose at a very large unfiltered inflow rate of 100,000 cfm to demonstrate that the dose results are relatively insensitive to these large values of unfiltered inleakage.

The licensee assumed that 10 cfm for ingress and egress should not be included and that the normal intake value of 1100 cfm is an appropriate assumption for future analysis. The NRC staff did not use this assumption to perform its own calculations, and the overall analysis conclusions are not impacted up to a value of 100,000 cfm. Based upon the results of the EN calculation using 100,000 cfm unfiltered inleakage, the NRC staff has reasonable assurance that the dose acceptance criterion for the FHA is met.

3.1.2.1 Fuel Transfer Area or Spent Fuel Pool Drop

EN's evaluation of a second case involving a drop in the fuel transfer area (between the reactor vessel and the spent fuel pool) or over the spent fuel pool. EN stated that the postulated activity released would be substantially lower than the limiting case described above based on the following:

The maximum credible drop height is 17 inches. After a drop height of 17 inches, the kinetic energy available to cause fuel damage is substantially reduced. The number of pins damaged in the design-basis drop would bound the number of pins damaged in a drop elsewhere as the drop height is significantly greater in the licensing basis case. The TS minimum required water depth over the point of fuel assembly impact is approximately 22 feet, just 1 foot lower than the 23 feet, upon which a decontamination factor (DF) of 200 is based. The difference in water height is approximately 1 percent for normal water level conditions (22 feet, 9 inches) and a maximum difference of approximately 4 percent for the minimum TS water level (22 feet).

The drop height of 17 inches is limited by procedural controls. In accordance with Licensee Controlled Specification (LCS) 1.9.1, the top of active fuel in an assembly must be maintained at least 7 feet, 6 inches below the TS minimum required water level of 22 feet. Based on the comparable water depth available for decontamination and the difference in the postulated drop distances, EN concluded that the consequences of an FHA over the reactor cavity bound those for an FHA over the transfer area or over the spent fuel pool.

In a EN response, dated March 21, 2006, to NRC Question 14, EN provided further clarification of the second case. A rod drop analysis was performed to provide a quantitative assessment of the source term. A drop height of 4 feet was analyzed and the resulting number of rods failing was calculated to be 83 rods. The water height above the release point (assuming the assembly is laying flat) is 22 feet - 5.5 inches (assembly width) = 21 feet, 6.5 inches. Based on the comparable water depth available for decontamination and the difference in the postulated drop distances, EN concluded that the consequences of an FHA over the reactor cavity bound those for an FHA over the transfer area or over the spent fuel pool spent fuel pool.

3.1.2.2 Technical Specifications

EN proposed changing TSs 3.6.4.1, "Secondary Containment," 3.6.4.2, "Secondary Containment Isolation Valves," and 3.6.4.3, "Standby Gas Treatment System," by deleting "During movement of irradiated fuel assemblies in the secondary containment, During CORE ALTERATIONS" from the applicability statements. EN also proposed that footnote (b) be deleted from TS Table 3.3.6.2-1. EN justified the changes by stating that secondary containment is not credited for the mitigation of the FHA. The need to ensure the operability of this system during core fuel handling activities or core alterations is no longer necessary.

The NRC staff requested that EN provide confirmation that the analyzed configuration provided the most bounding atmospheric dispersion factors for all possible release paths (since containment was not credited). In an EN response, dated March 21, 2006, to NRC Question 22, EN provided this confirmation. EN stated that, since the analyzed source is closest to the local intake, no other point on the reactor building produces a higher χ/Q. Based upon this limiting χ/Q and the summary that follows, the NRC staff finds these changes acceptable.

3.1.2.3 FHA Review

The NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological impacts of the proposed changes. The assumptions found acceptable to the NRC staff are presented in Table 1 of this SE. Based upon the information provided by EN, the NRC staff finds that the licensee used analysis methods and assumptions consistent with the guidance of RG 1.183, except were discussed and accepted above. The NRC staff compared the radiation doses estimated by the licensee to the 10 CFR 50.67(b)(2) acceptance criteria and to the results estimated by the NRC staff in its confirmatory calculations. The NRC staff finds, with reasonable assurance, that the licensee's estimates of the EAB, LPZ, and CR doses for the FHA will continue to comply with these criteria.

3.1.3 Main Steam Line Break (MSLB)

The postulated MSLB accident is a double-ended break of one main steam line outside the primary containment. The assumed displacement of the pipe ends permits a maximum blowdown rate. The mass of coolant released is the amount in the steam line and connecting lines at the time of the break, plus the amount passing through the MSIVs prior to closure (6 seconds[1]). A total of 130,000 pounds mass (lbm) of blowdown (105,000 lbm of liquid and 25,000 lbm of steam) is released as documented in the current licensing basis. The quantity of blowdown is not affected by the application of the AST methodology to this event.

The release of steam to the environment resulting from the MSLB is assumed to be an instantaneous ground level puff. EN stated that the methodology used to establish the puff transit time and the normalized concentration as a function of distance traveled is consistent with RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants." The initial volume of the puff is established by the amount of steam released by the MSLB and by flashing a portion of the entrained liquid. The volume of the puff was calculated to be approximately 5.9E4 m^3.

3.1.3.1 Source Term

The fission product inventory available for release was based on the maximum equilibrium reactor coolant dose equivalent iodine 131 (DEI-131) concentration of 0.2 microcuries per gram (µCi/gm). This is the limit specified in TS Limiting Condition for Operation 3.4.8. In addition to the maximum equilibrium case, RG 1.183 specifies a pre-accident iodine spiking case. To account for iodine spiking, the equilibrium level of DEI-131 was increased by a factor of 20 to achieve a spiking concentration of 4.0 µCi/gm. No fuel damage was postulated for the MSLB.

The activity (in the terms of DEI-131) in the mass of the initial liquid blowdown was assumed to be released to the atmosphere instantaneously, as a ground level release, and no credit was taken for plateout, holdup, or dilution within facility buildings. For example, the DEI-131 total activity release for the iodine spiking case is 4 µCi/gm x 105,000 lbm (mass of the initial liquid blowdown) x 454 gm/lbm /1 E6 µCi/Ci =191 Ci.

The NRC staff questioned why EN did not factor in the iodine that would be contained in the 25,000 lbm of steam. EN's response, dated March 21, 2006, to NRC Question 26, stated that neglecting the iodine in the 25,000 lbm of steam was more than compensated by other conservative assumptions. Although the NRC staff believes the steam activity should be added into the total activity released, the NRC staff's confirmatory analysis showed the licensee's results were acceptable. This acceptance is based on the fact that even if this activity were to be added into the calculation the radiological doses still meet the acceptance criteria.

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[1] In an EN response, dated March 21, 2006, to NRC Question 2, EN stated that 6 seconds is conservative, relative to the maximum closure time of 5 seconds, as currently specified in Technical Specification Surveillance Requirement 3.6.1.3.6. The 6-second closure time assumption was only used for the purpose of the radiological analysis and no change to the MSIV closure time is requested.

3.1.3.2 Mitigation

The only mitigative action credited for the MSLB event was the termination of the release upon the automatic closure of the MSIVs. The CR ventilation was assumed to remain in the normal mode. The local air intake is used for analyzing dispersion. There is no accident signal credited to start emergency CR ventilation. No credit was taken for operator actions. The MSIV isolation actuates on a high-flow signal. The CR ventilation normal intake flow is assumed unfiltered and constant. EN's design-basis MSLB dose calculation results are the same as if evaluated at the CR intake and, therefore, are independent of the mitigating effects of the CR structure.

3.1.3.3 Radiological Transport Modeling

The release of steam resulting from the MSLB (through blowout panels in the steam tunnel) was assumed to be an instantaneous ground level puff. The release point was assumed to be blowout panel A. It was assumed that the plume translates directly to the local CR intake that is closest to the assumed release location.

3.1.3.4 MSLB Review

The NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological impacts of the proposed changes. The assumptions and parameters found acceptable to the NRC staff are presented in Table 1. Based upon the information provided by EN, the NRC staff finds that the licensee used analysis methods and assumptions consistent with the guidance of RG 1.183. The NRC staff compared the radiation doses estimated by the licensee to the applicable acceptance criteria and to the results estimated by the NRC staff in its confirmatory calculations. The NRC staff finds, with reasonable assurance, that the licensee's estimates of the EAB, LPZ, and CR doses for the MSLB will continue to comply with these criteria.

3.1.4 Control Rod Drop Accident

The postulated CRDA involves the rapid removal of a highest worth control rod resulting in a reactivity excursion. Consistent with the DBA, 1.8 percent of the fuel pins in the full core are postulated to be damaged, with melting occurring in 0.77 percent of the damaged rods (i.e., 0.014 percent of the core). A core average radial peaking factor of 1.7 was assumed in the analysis.

The CRDA is terminated by the average power range monitors (APRM) high-flux scram signal. The activity released from the damaged fuel that reaches the turbine and condenser is released from the turbine building at ground level at a rate of 1 percent condenser volume per day for a period of 24 hours. No credit is taken for turbine building holdup or dilution. The analysis assumptions for the transport, reduction, and release of the radioactive material from the fuel and the reactor coolant are consistent with RG 1.183.

3.1.4.1 Source Term

The source term used for the CRDA analysis was composed of releases from melted fuel and the gap activity from the fuel pins postulated to be damaged. This initial amount of activity was released into the reactor coolant at time zero.

3.1.4.2 Mitigation

The CRDA is terminated by the APRM high-flux scram signal. Partitioning of the initial activity released during its transport from the RCS to the condenser and ultimately to the environment was credited. Radioactive decay during the holdup in the turbine and condenser was also credited.

No other mitigation of the radiological release was credited. No credit for dilution or holdup in the turbine building was assumed. The CR ventilation was conservatively assumed to remain in its normal mode. There was no accident signal credited to start emergency CR ventilation. No credit was taken for operator actions. CR ventilation normal intake flow was unfiltered.

3.1.4.3 Radiological Transport Modeling

The radiological release model for the CRDA was developed consistent with RG 1.183. A ground level release was modeled from the turbine building at a rate of 1 percent condenser volume per day over a period of 24 hours. During normal operations, flow is through the local CR intake combined with flow from the remote intakes. The intake of the released radionuclides into the CR is based on a volumetric flow rate of 1100 cfm of unfiltered air through only the local intake. This assumption is conservative, because no manual action for CR isolation was credited for the entire 24-hour period.

The NRC staff questioned why EN did not factor the impact of ingress and egress, as stated in RG 1.183, into the total unfiltered inleakage into the CR and whether the value of 1,100 cfm value was appropriate. In an EN response, dated March 21, 2006, to NRC Question 19, EN stated that the 10 cfm for ingress and egress and increasing the 1,100 cfm by 50 percent would not impact the radiological dose results by greater than 2.5 percent. EN also presented the results of a study that calculated the dose at a very large unfiltered inflow rate of 100,000 cfm to demonstrate that the dose results are relatively insensitive to these large values of unfiltered inleakage.

The licensee assumed that 10 cfm for ingress and egress should not be included and that the normal intake value of 1100 cfm is an appropriate assumption for future analysis. The NRC staff did not use this assumption to perform its own calculations, and the overall analysis conclusions are not impacted up to a value of 100,000 cfm. Based upon the results of the EN calculation using 100,000 cfm unfiltered inleakage, the NRC staff has reasonable assurance that the dose acceptance criterion for the CRDA is met.

3.1.4.4 CRDA Review

The NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological impacts of the proposed changes. The assumptions found to be acceptable to the NRC staff are presented in Table 1 of this SE. Based upon the information provided by EN, the NRC staff finds that the licensee used analysis methods and assumptions consistent with the guidance of RG 1.183 except were discussed and accepted above. The NRC staff compared the radiation doses estimated by the licensee to the 10 CFR 50.67(b)(2) acceptance criteria and to the results estimated by the NRC staff in its confirmatory calculations. The NRC staff finds, with reasonable assurance, that the licensee's estimates of the EAB, LPZ, and CR doses for the CRDA will continue to comply with these criteria.

3.2 Atmospheric Dispersion Estimates

The licensee calculated new atmospheric dispersion factors (χ/Q values) for use in evaluating the radiological consequences of DBAs on the CR, EAB, and LPZ dose assessments. The licensee used the ARCON96 and PAVAN atmospheric dispersion computer models to calculate χ/Q values for the LOCA, the CRDA, and the FHA. In addition, the licensee used the instantaneous puff release methodology discussed in RG 1.194 to calculate χ/Q values for the MSLB accident to the CR and the PAVAN computer model to calculate the EAB and LPZ χ/Q values. The resulting set of CR, EAB, and LPZ χ/Q values represent a change from those currently presented in the CGS Final Safety Analysis Report.

3.2.1 Meteorological Data

The licensee generated the new LOCA, CRDA, and FHA CR and all offsite χ/Q values for this license amendment request, using meteorological data collected at the CGS site during the period 1996–1999. The licensee provided these data in the form of hourly data files (for input into the ARCON96 computer code) and a joint wind speed, wind direction, and atmospheric stability frequency distribution (for input into the PAVAN computer code) in Attachment 5, Item 11 to its letter dated September 30, 2004.

The licensee stated that the CGS onsite meteorological measurement program meets the guidelines of RG 1.23, "Onsite Meteorological Programs," with regard to tower location, construction and instrument locations, and instrument accuracies for air temperature, wind speed, and wind direction. The delta temperature measurement accuracy meets the guidance in RG 1.97, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident." Wind measurements at CGS were taken at 10 meters and 75 meters above ground level and the vertical temperature difference was measured between the 75-meter and 10-meter levels. The resulting combined data recovery rate of wind speed, wind direction, and stability exceeded the RG 1.23 goal of 90 percent for the 4-year period, although the recovery rates of atmospheric stability data in 1996 and 75-meter wind speed data in 1999 were in the upper 80 percentiles.

The NRC staff performed a quality review of the 1996–1999 hourly meteorological database provided by the licensee using the methodology described in NUREG-0917, "Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data." Further review was performed using computer spreadsheets.

With regard to atmospheric stability, the time of occurrence and duration of reported stability conditions were generally consistent with expected meteorological conditions (e.g., neutral and slightly stable conditions predominated during the year with stable and neutral conditions occurring at night and unstable and neutral conditions occurring during the day). However, the

NRC staff noted a few apparent anomalies in the data. As discussed in Enclosure 1 of the licensee's March 21, 2006, response to an NRC request for additional information (RAI), the licensee described several problem areas in the measurement program between 1996 and 1999. The program was upgraded in late 2001, but a need for further modifications was noted. In 2001 and 2002, the licensee performed a total review of the meteorological data process and the meteorological system. As a result of the review, numerous corrective actions were established and are being implemented. With regard to the 1996-1999 data used in the calculations for this license amendment request, the licensee and a certified meteorologist compared meteorological data collected at CGS with meteorological data collected independently at neighboring Hanford, Washington over the same period. The licensee's study determined that there was good agreement between the two data sources. In addition, 4-year average χ/Q values generated from both data sources matched up well, with the values generated using the CGS data being more conservative than the 4-year average χ/Q values generated using Hanford data.

During 1996-1999, wind speed and wind direction were reasonably similar between the two measurement heights and from year to year. Winds were predominately from the northwest, south, and southwest at both levels. A comparison of joint frequency distributions derived by the NRC staff from the ARCON96 hourly data with the joint frequency distributions developed by the licensee for input into the PAVAN atmospheric dispersion model showed good agreement.

3.2.2 Control Room Atmospheric Dispersion Factors

The licensee used the guidance provided in RG 1.194 to generate the new CR atmospheric dispersion factors. The licensee calculated new CR χ/Q values for all of the postulated release scenarios, other than for the MSLB event, using the ARCON96 computer code (NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes"). RG 1.194 states that ARCON96 is an acceptable methodology for assessing CR χ/Q values for use in DBA radiological analyses. The licensee executed ARCON96 using the 1996–1999 hourly 10-meter level wind speed and wind direction data from the CGS onsite meteorological tower. Stability class was based on the temperature difference data between the 75-meter and 10-meter levels on the onsite meteorological tower. The licensee modeled the roofline and condensate storage tank release locations as point sources and reactor building wall, reactor building vehicle air lock door (King Kong or KK door), and turbine building release locations as diffuse sources in a manner generally consistent with guidance provided in RG 1.194. Because the heights of all of the release locations are less than 2½ times the height of adjacent buildings, they were modeled using the ARCON96 ground-level release option in accordance with RG 1.194.

The NRC staff evaluated the applicability of the ARCON96 model and concluded that there are no unusual siting, building arrangements, release characterization, source-receptor configuration, meteorological regimes, or terrain conditions that preclude use of the ARCON96 model for the CGS site. The NRC staff qualitatively reviewed the inputs to the ARCON96 and found them generally consistent with site configuration drawings and site practice. The NRC staff also performed a random check of the ARCON96 calculations and obtained results that were similar to the licensee's estimates.

The licensee used the instantaneous puff release methodology described in RG 1.194 to calculate the CR χ/Q value for the MSLB accident as a ground-level hemispherical puff. The NRC staff qualitatively reviewed the licensee's inputs, made estimates using the RG 1.194 equation to estimate effective puff relative concentration and obtained results that were similar to the licensee's estimates.

CGS is served by three CR air intakes: a local intake and two remote intakes. The local intake is a louver located on the west side of the radwaste building wall at a height 26.5 meters above the ground and adjacent to the CR. Due to the proximity of the local intake to the CR, the licensee modeled unfiltered inleakage as though it entered the CR at the same location as the local intake. The two remote intakes are on opposite sides of the reactor building complex at ground level, separated by a distance of more than 200 meters. The licensee modeled effective χ/Q values for filtered releases from the roofline source, KK doors, reactor building walls, and turbine building by weighting the respective unfiltered ARCON96 generated χ/Q values using guidance provided in RG 1.194 for several CR intake flow rate combinations that varied as a function of time among the three intakes. The effective χ/Q values selected for use in the dose assessment described above were based upon an assumed constant flow rate of 150 cfm into the local intake and a concurrent flow of either 325 or 575 cfm into each of the two remote intakes. Flow into the two remote intakes was assumed to be equal, but vary as a function of time. The NRC staff performed a check of the weighting calculations used to generate the effective χ/Q values and obtained results that were similar to the licensee's estimates. However, the NRC staff notes that the effective χ/Q values are a function of the inflow rate into each CR intake. Therefore, approval of the effective χ/Q values referenced in Table 1 below is predicated on these flow rates. A change in the flow rates could impact the estimated dose to the CR operators.

To generate the secondary containment bypass χ/Q values, the licensee calculated arithmetic averages of the KK door and reactor building wall secondary containment bypass χ/Q values. The NRC staff has determined, from the docketed information, that the licensee has not adequately justified the arithmetic averaging of atmospheric dispersion factors for secondary containment bypass. There is uncertainty in the pathways for secondary containment bypass and for this reason the NRC staff does not accept averaging these atmospheric dispersion factors. The NRC staff performed a sensitivity study of the averaging as compared to using the worst case χ/Q values. Based upon this study, the NRC staff has reasonable assurance that without this averaging, as credited in this analysis, the dose acceptance criterion for the LOCA will be met.

3.2.3 Offsite Atmospheric Dispersion Factors

The licensee used RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," and the PAVAN computer code to calculate χ/Q values for the EAB and LPZ. Since all of the postulated release locations are less than 2½ times the height of adjacent structures, the licensee used the PAVAN ground-level release mode, inputting an EAB distance of 1,950 meters, LPZ distance of 4,827 meters, reactor building height of 69.8 meters, and reactor building cross-sectional area of 2,861 square meters. The licensee's meteorological input to PAVAN consisted of a joint frequency distribution of wind speed, wind direction, and atmospheric stability data for the 1996–1999 period. Wind speed and direction data from the meteorological tower's 10-meter level were used. Stability class was based on the temperature difference data between the 75-meter and 10-meter levels on the onsite meteorological tower. The licensee combined 4-yearly joint-frequency distribution data files to generate a single file representing the 4-year period, based upon 11 wind speed categories, with the calm category distributed separately from other 11 wind speed categories. The licensee then generated χ/Q values using both desert and Pasquill-Gifford sigmas, performed calculations as recommended in RG 1.145, and selected the highest resultant χ/Q values for use in the dose assessment. The NRC staff qualitatively reviewed the inputs to the licensee's computer runs and found them generally consistent with site configuration drawings and NRC staff practice. The NRC staff also generated comparative estimates using the PAVAN code and obtained results similar to the licensee's results.

3.2.4 Secondary Containment Drawdown - Meteorology

RG 1.183 states that the effect of high winds on the ability of the secondary containment to maintain a negative pressure should be evaluated on an individual case basis. The wind speed to be assumed is the 1-hour average value that is exceeded only 5 percent of the total number of hours in the data set. Ambient temperatures used in these assessments should be the 1-hour average value that is exceeded only 5 percent or 95 percent of the total hours in the data set, whichever is conservative for the intended use (e.g., if high temperatures are limiting, those exceeded only 5 percent of the time should be used). The licensee estimated the CGS wind speed that is exceeded only 5 percent of the time to be 17.2 miles per hour and the air temperatures that are exceeded only 5 and 95 percent of the time to be 86 °F and 28 °F, respectively. The NRC staff found these estimates reasonable when compared with estimates that the NRC staff generated from the 1996-1999 onsite wind data measured at the 10-meter level and climatic temperature data (Weather Data Viewer, Version 3, 2005 American Society of Heating, Refrigerating and Air-Conditioning Engineers, Inc., Atlanta, Georgia) for Pasco, Washington, which is approximately 18 miles southeast of the CGS site.

3.2.5 Atmospheric Dispersion Summary

Based on the review analysis discussed above, the NRC staff has concluded that the 1996-1999 meteorological data measured at the CGS site provide an acceptable basis for making atmospheric dispersion estimates for use in the DBA dose assessments performed in support of this specific license amendment request.

The NRC staff has reviewed the licensee's assessments of CR post-accident dispersion conditions generated from the licensee's meteorological data and atmospheric dispersion modeling. On the basis of this review, the NRC staff concludes that the resulting CR χ/Q values referenced in Table 1 (of this SE) are acceptable for use in the dose assessment described above. However, use of the effective χ/Q values is predicated on the flow rates as discussed in Section 3.5.1.2. Thus, the NRC staff notes that if the flow rates are subject to change, the effective χ/Q values would need to be reassessed and estimation of the CR operator doses could be impacted. Further, since the licensee has not adequately justified the arithmetic averaging of atmospheric dispersion factors for secondary containment bypass, use of these χ/Q values is specific to this dose assessment only. These averaged secondary containment bypass χ/Q values should not be used for any other applications.

The NRC staff also reviewed the licensee's assessments of EAB and LPZ post-accident dispersion conditions generated from the licensee's meteorological data and atmospheric dispersion modeling. The resulting EAB and LPZ χ/Q values are referenced in Table 1 of this SE. On the basis of the review described above, the NRC staff concludes that the χ/Q values for DBA releases to the CGS EAB and LPZ are acceptable for use in the dose assessments performed in support of this LAR.

3.3 Shutdown Requirements

Energy Northwest requested a change to TS 3.1.7 which adds the requirement that the reactor be in MODE 4 within 36 hours if the required action and associated completion time are not met. Current TS 3.1.7 states that the reactor be in MODE 3 within 12 hours if the required action and associated completion time are not met. The additional MODE 4 requirement to TS 3.1.7 does not affect the current requirements for reactor shutdown based on SLCS availability. The proposed change to TS 3.1.7 provides additional conservatism by increasing the shutdown requirements and enforcing MODE 4 operation if TS conditions are not met while in MODE 3.

The NRC staff finds the addition of MODE 4 requirements to limiting condition for operation (LCO) 3.1.7 is acceptable because it does not impact the reactivity control function of the SLCS and the SLCS will still meet GDC 26 and 29.

3.4 pH Controls

The NRC staff reviewed the amendment related to maintaining the suppression pool pH above 7 for a period of 30 days in accordance with RG 1.183, "Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors." Specifically the NRC staff reviewed the licensee's assumptions and methodology for preventing iodine re-evolution in a postulated LOCA.

In NUREG-1465 the NRC staff concluded that iodine entering the containment from the reactor coolant system during an accident would be composed of at least 95 percent cesium iodide (CsI). Upon dissolution in the suppression pool the predominant form of iodine would be the iodide ion (I^-). The radiation-induced conversion of iodide in water into elemental iodine (I_2) is strongly dependent on the pH. Without pH control, a large fraction of the iodine dissolved in water in the ionic form will be converted to elemental iodine and released into the containment atmosphere. If the pH is maintained above 7, less than 1 percent of the dissolved iodine will be converted to elemental iodine. Since the pH of the suppression pool is not controlled under normal conditions, elemental iodine may be released during a postulated LOCA when acids lower the suppression pool pH.

To prevent the release of elemental iodine during a LOCA an alkaline chemical capable of buffering the pH at a value above 7 must be added to the suppression pool. The CGS submittal proposes to do this by injecting sodium pentaborate from the SLCS during a LOCA. The analysis assumes that the sodium pentaborate solution is injected and mixed in the suppression pool within 8 hours of the onset of a LOCA.

The licensee's calculations consider the effects of acids and bases created during a LOCA. The sump pH is affected by the generation of hydrochloric acid from the irradiation of electrical cables and nitric acid from the irradiation of water and air. Calculations for the amount of hydrochloric and nitric acids generated are based on guidelines in NUREG-5950, "Iodine Evolution and pH Control." The licensee uses the STARpH 1.04 code to model the introduction of acid to the sump environment.

The NRC staff reviewed the licensee's methodology, assumptions, and calculations for determining the 30-day post-accident pH value and found the evaluation acceptable. The NRC staff independently verified through calculations that the sump pH is maintained above 7 for the duration of the 30-day period.

3.5 Iodine Removal Coefficients

In Standard Review Plan (SRP) Section 6.5.2, "Containment Spray as a Fission Product Cleanup System, "the effectiveness of the containment spray system can be estimated by considering the chemical and physical processes that can occur during an accident in which the system operates. The concentration of iodine in the spray solution is an important factor in determining the effectiveness of the sprays against elemental iodine vapor during an accident.

The elemental iodine and particulate spray removal coefficients (λ_s and λ_p, respectively) represent the rate at which elemental iodine vapor and particulate fission products are removed from the containment atmosphere by the spray system. The licensee assumes that the value for λ_s is equal to the calculated λ_p value of 6.20 hr^{-1}. The licensee performed calculations to confirm that this assumption was conservative. A DF of 121 was used. This DF value is conservative relative to the value of 200 allowed by the SRP.

The NRC staff independently verified through calculations that the particulate removal coefficient and the decontamination factor calculated for elemental iodine are appropriate. Therefore, the NRC staff finds that these values are acceptable for the fission product cleanup model.

The NRC staff reviewed the licensee's assumptions to minimize iodine re-evolution as presented in the re-analysis of the radiological consequences for a LOCA. The assumptions are appropriate and consistent with the methods accepted by the NRC staff for the calculation of post-accident containment sump pH. In addition, the NRC staff independently verified that the post-accident containment sump pH will be maintained above 7 for 30 days following a LOCA.

The NRC staff also verified the licensee's iodine spray removal and particulate removal coefficients. In addition, the NRC staff determined that the licensee's calculated decontamination factor for elemental iodine will be reached within 30-days of the beginning of the accident. These values are consistent with the guidance in SRP 6.5.2 and, therefore, the NRC staff concludes that these values are acceptable for the fission product cleanup model.

3.6 Equipment Qualification

The licensee has elected to retain TID-14844 assumptions for performing the required EQ analyses. The radiation doses used for the EQ analyses at current licensed core power level in support of AST evaluations.

The equipment exposed to the containment atmosphere, the TID-14844 source term and the gap and in-vessel releases in the AST produced similar integrated doses. For the equipment exposed to sump water, the integrated doses calculated with the AST exceeded those calculated with TID-14844 after 145 days for a BWR, because of the 30 percent vs. 1 percent release of cesium according to NUREG-1465. The continued use of the TID-14844 source term provides integrated doses for equipment, which envelop those that would be calculated using AST. Therefore, following implementation of AST, CGS will continue to meet 10 CFR 50.49 by using a radiation environment associated with the most severe DBA.

The NRC staff considered the applicability of the revised source terms to operating reactors and determined that the current analytical approach based on the TID-14844 source term would continue to be adequate to protect public health and safety, and that operating reactors licensed under this approach would not be required to reanalyze accidents using the revised source term.

Based on the above information, the NRC staff finds the licensee's position to retain the TID-14844 assumptions for performing the required EQ analyses is acceptable.

4.0 TECHNICAL SPECIFICATION CHANGES

The proposed changes to the TSs are contained in Attachment 1, pages 5-15 of the September 30, 2004, submittal and its supplements. The TS changes that are supported by the radiological design-basis analyses discussed in Section 3.1 are summarized below:

4.1 Table of Contents

Technical Specification	Description
1.1	"Definitions"
Table 3.3.6.2-1	"Secondary Containment Isolation Instrumentation"
Table 3.3.7.1-1	"Control Room Emergency Filtration (CREF) System Instrumentation"
3.6.1.3	"Primary Containment Isolation Valves (PCIVs)"
3.6.1.8	"Main Steam Isolation Valve Leakage Control (MSLC) System"
3.6.4.1	"Secondary Containment"
3.6.4.2	"Secondary Containment Isolation Valves (SCIVs)"
3.6.4.3	"Standby Gas Treatment (SGT) System"
3.7.3	"Control Room Emergency Filtration (CREF) System"
3.9.7	"Reactor Pressure Vessel (RPV) Water Level - New Fuel or Control Rods"
3.9.10	"Decay Time"

The NRC staff confirmed that the proposed changes were consistent with the methodology used in the revised AST methodology. Since the analysis using this AST methodology confirms that the applicable regulatory criteria are met, these changes are acceptable from a radiological dose consequence perspective. The NRC staff has provided additional details regarding the acceptance of individual specifications as stated in Section 3.0 entitled, "Technical Evaluation," unless otherwise stated.

Deleted section 3.6.1.8, "Main Steam Isolation Valve Leakage Control (MSLC) System," and added section 3.9.10, "Decay Time."

The NRC staff considers changes to the Table of Contents as editorial and is acceptable.

4.2 TS 1.1, "Definitions"

Revised the definition for DOSE EQUIVALENT 1-131 by replacing the word "thyroid" with "Total Effective Dose Equivalent (TEDE)" and replacing the references to dose conversion factors from TID-14844, RG 1.109, and ICRP-30, with a reference to Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1988.

The NRC staff reviewed this change and finds this change consistent with regulatory guidance and, therefore, acceptable.

4.3 TS 3.1.7, "Standby Liquid Control (SLC) System"

Added MODE 3 to the applicability statement and added the requirement to be in MODE 4 within 36 hours if a required action was not met.

This change is needed to support the use of the SLC system for buffering suppression pool pH as assumed in the LOCA analysis performed in support of this AST license amendment request.

The NRC staff reviewed the SLC system with respect to SLC roles in delivery of sodium pentaborate to the suppression pool for pH control. The control of pH in the suppression pool is required to mitigate the consequences of a design-basis accident in which fuel is damaged. As such, the new role being assigned to the SLC is a safety-related role. The licensee stated that the SLC is categorized in the final safety analysis report as a special safety system with a design function to mitigate the Anticipated Transient Without Scram (ATWS) event.

The NRC staff reviewed the licensee's submittals and the responses to RAI's on the use of the SLC system for the safety-related function. From the licensee statements, the NRC staff has concluded the following:

Although the SLC system is not a safety-related system, the SLC system as designed and installed is a high-quality system that provides reasonable assurance that the sodium pentaborate will be injected into the core upon activation, specifically:

1. The system components required for reactivity control and new suppression pool pH control functions are seismically qualified.

2. The system is provided with emergency power from the emergency diesel generators.
3. The system is subject to American Society of Mechanical Engineers Section XI, Inservice inspection requirements as required by 10 CFR 50.55a, codes and standards.
4. The system is within the scope of the 10 CFR 50.65 Maintenance Rule.
5. Most components (pumps, squib valves, etc.) are redundant in parallel trains powered from different electrical busses. The exceptions are the containment isolation check valves. This is discussed below under single-failure review.
6. Emergency Operating Procedures (EOP's) direct the activation of the SLC following a LOCA when reactor water level cannot be maintained above the top of active fuel. Manual initiation of SLC is also directed in the severe accident management guidelines (SAMGs), which are entered when adequate core cooling cannot be maintained.
7. Training will be provided on the new SLC injection function as part of operator re-qualification training and EOP and SAMG training.

The NRC staff considered components that could be subject to single failure. The licensee identified two components, the containment isolation check valves on the injection line as not being redundant. The containment isolation valves are Borg-Warner Lift Check Valves, Model 76797-1 valves, mounted in the injection line. In the periodic inspections and testing of these valves, CGS has not experienced any failures of these valves to open on demand. A review of the industry databases, EPIX and NPRDS, was performed and no failures of check valves of this type and manufacture failing to open were identified. Although acknowledging that a single failure to open of one of the two check valves could prevent SLC injection, the NRC staff has determined that the potential for failure is very low based on the quality as established by its procurement, periodic testing, inspection, and historical performance of the component. The NRC staff finds that the use of a single penetration of the containment with the identified check valves as described by the licensee to be acceptable.

The NRC staff considered the transport of the sodium pentaborate from the reactor vessel to the suppression pool. The SLC system injects the sodium pentaborate to the reactor vessel. The transport of reactor vessel contents, including the sodium pentaborate, to the suppression pool is by flow through the break (assumed to be a large recirculation pipe break) to the drains that feed the suppression pool. Core Spray systems and low-pressure coolant injection (LPCI) systems are used to maintain water level and assure core cooling after a LOCA accident.

Using the LPCI for suppression pool cooling also provides mixing. The NRC staff concluded that there would be mixing and transport at some rate and that it was reasonable to assume the concentration of sodium pentaborate in the core would equalize with the concentration in the suppression pool within an acceptable time after SLC injection. As a consequence, there would be sufficient pH control to deter and prevent iodine re-evolution.

The specific changes being made to TS Section 3.1.7 are the extensions of applicability to MODE 3 and an additional Required Action for Condition C to require Mode 4 if completion times are not met. The extension of applicability to Mode 3 provides the capability of injecting SLC during hot standby. This would not be necessary for the ATWS function of SLC, but is reasonable for the LOCA pH control function. Clarifying the action and response time is appropriate for this action. On the basis of the above discussion, the NRC staff finds these changes acceptable.

Further, the NRC staff finds the addition of MODE 4 requirements to LCO 3.1.7 is acceptable because it does not impact the reactivity control function of the SLC system, and the SLC system will still meet GDC 26 and 29.

4.4 Table 3.3.6.1-1, "Primary Containment Isolation Instrumentation"

Added MODE 3 to the applicable mode column for item K, "SLC System Initiation."

The NRC staff reviewed this change and noted that the change is needed to facilitate operation in MODE 3. The NRC staff considers this change to be editorial in nature and, therefore, acceptable.

4.5 Table 3.3.6.2-1, "Secondary Containment Isolation Instrumentation"

Deleted footnote (b) and corrected the spelling of "Function" in footnote (c).

The NRC staff reviewed the proposed change that after the implementation of the AST, footnote (b) would only apply to movement of "recently" irradiated fuel. Since the addition of TS 3.9.10 places a restriction that prevents the moment of "recently" irradiated fuel (i.e., fuel that has not decayed for a minimum of 24 hours after a shutdown), the NRC staff finds that the footnote is not needed and the proposed change is acceptable. The NRC staff also finds the spelling correction acceptable.

Although the secondary containment is not credited in the design-basis dose analysis, the licensee still has the obligation to confine, contain, process, and manage radiation releases caused by an accident or an abnormal event. Adoption of TSTF-51 requires a shutdown plan based on the recommendations of NUMARC 91-06, which should include isolating the secondary containment when necessary. CGS addressed their shutdown program in the letter dated September 29, 2005.

4.6 TS 3.3.7.1, "Control Room Emergency Filtration (CREF) System Instrumentation"

Deleted Actions E and F. Deleted "or radiation monitoring" and "as applicable" from Note 2 of the Surveillance Requirements (SR) section.

The NRC staff reviewed the proposed change and noted that Action E is invoked by Table 3.3.7.1-1, Item 4 for Main Control Room Instrumentation, which is being deleted as discussed below. Action F is in turn invoked by failure to complete Action E in the required times. The NRC staff concurs that removing the radiation monitoring words from the SR Note is consistent with this other TS change and, therefore, is acceptable.

4.7 Table 3.3.7.1-1, "Control Room Emergency Filtration (CREF) System Instrumentation"

Deleted footnote (b). Deleted item 4, "Main Control Room Ventilation Radiation Monitor."

The NRC staff reviewed the proposed change and noted that the CREF system was not credited in the AST FHA. Thus, the removal of footnote (b) is reasonable in that a TS control on this instrumentation under the conditions of a footnote (b) is not required. Also, the NRC staff noted that the CREF system is not credited in any design-basis analysis other than the LOCA. Activation of the CREF system for the LOCA is controlled by a high-drywell pressure, low-low reactor water level or high-radiation reactor building exhaust signals per Items 1, 2, and 3 in Table 3.3.7.1-1. With the removal of Item 4, the Main Control Room Ventilation Monitor instrumentation, there is still sufficient assurance that the CREF will be actuated on a LOCA to perform its safety function. The NRC staff finds these proposed changes acceptable.

4.8 TS 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)"

Deleted footnote I associated with SR 3.6.1.3.6. Revised SR 3.6.1.3.10 to increase the allowable limit for secondary containment bypass leakage from 0.74 scfh to 0.04 percent primary containment volume/day.

Revised SR 3.6.1.3.11 to increase the allowable main steam isolation valve (MSIV) leakage limit from 11.5 scfh per valve to 16.0 scfh per valve when tested at greater than or equal to 25.0 psig.

The NRC staff reviewed the proposed changes and agrees that the footnote I is no longer relevant and may be deleted. The licensee has shown that the relaxation of containment bypass leakage and MSIV leakage is acceptable in terms of the results for design-basis analyses, as discussed in Section 3.1.1.2.2 of this SE. As such the NRC staff finds the proposed changes are acceptable.

4.9 TS 3.6.1.8, "Main Steam Isolation Valve Leakage Control (MSLC) System"

Deleted entire TS.

The NRC staff reviewed the proposed change and agrees that no credit is being taken for this system in the mitigation of the doses of a design-basis accident. As such, there is no requirement for TS controls of this system based on 10 CFR 50.36. The NRC staff finds the proposed change to be acceptable, therefore, the TS requirements can be removed.

4.10 TS 3.6.4.1, "Secondary Containment"

Changes are proposed to the following three sections of this TS:

1) Deleted "During movement of irradiated fuel assemblies in the secondary containment, During CORE ALTERATIONS," from the applicability statement.

2) Deleted the portions of Action C related to fuel movement and core alterations. As a result of these deletions, Action C.3 became C.1. Additionally, the Limiting Condition for Operation (LCO) 3.0.3 note provided in Action C was deleted.

3) Revised SR 3.6.4.1.1 to change the minimum required containment vacuum from greater than or equal to 0.25 inch of vacuum water gauge to greater than 0.0 inch of vacuum water gauge. Deleted SR 3.6.4.1.4. Revised the existing SR 3.6.4.1.5 to change the maximum allowed standby gas treatment (SGT) subsystem flow rate from less than or equal to 2240 cubic feet per minute (cfm) to a secondary containment inleakage flow rate of less than or equal to 2430 cfm. Due to the deletion of SR 3.6.4.1.4, SR 3.6.4.1.5 is renumbered as SR 3.6.4.1.4.

For item 1), the NRC staff reviewed the proposed change to this TS, which establishes the operability requirements for secondary containment. The NRC staff noted that the secondary containment is not credited for the mitigation of the AST FHA once fuel has decayed to a sufficient level. The new TS to be added, TS 3.9.10, establishes a decay time requirement that prevents movement of fuel until it is sufficiently decayed. TSTF-51 provides operational flexibility during refueling by allowing some TS to be removed once fuel has decayed sufficiently. Although relaxing the TS, adoption of TSTF-51 requires the development of a shutdown administrative program consistent with NUMARC 91-06 that assures releases from an FHA will be contained, processed, and filtered to limit doses even lower than required by regulations. Establishing secondary containment is required to achieve the goals of the shutdown administrative program. The licensee addressed his shutdown administrative program in the letter dated September 29, 2005, as a response to an RAI. The NRC staff finds this proposed change to be acceptable.

For item 2), the NRC staff confirmed that changing Action C.3 to C.1 is editorial. The LCO 3.0.3 note associated with Action C is no longer required. The NRC staff finds this proposed change to be acceptable.

For item 3), the licensee has withdrawn the three changes to the SRs in a letter dated September 11, 2006, and also proposed changes to the TS bases. The NRC staff did not review the proposed bases changes contained in the September 11, 2006, since the licensee should use their bases change program to effect changes to their TS bases.

4.11 TS 3.6.4.2, "Secondary Containment Isolation Valves (SCIVs)"

Changes are proposed to the following two sections of this TS:

1) Deleted "During movement of irradiated fuel assemblies in the secondary containment, During CORE ALTERATIONS," from the applicability statement.

2) Deleted the portions of Action D related to fuel movement and core alterations. As a result of these deletions, Action D.3 became D.1. Additionally, the LCO 3.0.3 note provided in Action D was deleted.

The licensee states that this TS establishes the operability requirements for SCIVs. Since secondary containment is not credited for the mitigation of the AST FHA, the need to ensure the operability of the SCIVs during core alterations or fuel handling activities is no longer necessary. This change, combined with the addition of TS 3.9.10, is consistent with the scope and intent of TSTF-51. Changing Action D.3 to D.1 is editorial. The LCO 3.0.3 note associated with Action D is no longer required.

The NRC staff reviewed this proposed change that after the implementation of the AST, "During movement of irradiated fuel assemblies in the secondary containment" would only apply to movement of "recently" irradiated fuel. Since the addition of TS 3.9.10 places a restriction that prevents the moment of "recently" irradiated fuel (i.e., fuel that has not decayed for a minimum of 24 hours after shutdown), the NRC staff finds that the statement is not needed and the proposed change is acceptable. The NRC staff also finds the statement, "During CORE ALTERATIONS" is not needed since the dose results of core alterations is bounded by the FHA analysis. Thus, deleting this phrase is also acceptable.

Although the secondary containment is not credited in the design-basis dose analysis, the licensee still has the obligation to confine, contain, process, and manage radiation releases caused by an accident or an abnormal event. Adoption of TSTF-51 requires a shutdown plan based on the recommendations of NUMARC 91-06, which should include isolating the secondary containment when necessary. CGS addressed their shutdown program in the letter dated September 29, 2005.

4.12 TS 3.6.4.3, "Standby Gas Treatment (SGT) System"

Changes are proposed to the following three sections of this TS:

1) Deleted "During movement of irradiated fuel assemblies in the secondary containment, During CORE ALTERATIONS," from the applicability statement.

2) Deleted the portions of Actions C and E related to fuel movement and core alterations. As a result of these deletions, Action C.2.3 became C.2, and E.3 became E.1. Additionally, the LCO 3.0.3 notes provided in Actions C and E were deleted.

3) Revised SR 3.6.4.3.3 to add the phrase "and reaches greater than or equal to 4800 cfm within 2 minutes."

For item 1), the NRC staff reviewed the proposed change to this TS, which establishes the operability requirements for the SGT system. The NRC staff noted that the secondary containment is not credited for the mitigation of the AST FHA once fuel has decayed to a sufficient level. The new TS to be added, TS 3.9.10, establishes a decay time requirement that prevents movement of fuel until it is sufficiently decayed. TSTF-51 provides operational flexibility during refueling by allowing some TSs to be removed once fuel has decayed sufficiently. Although relaxing the TS, adoption of TSTF-51 requires the development of a shutdown administrative program consistent with NUMARC 91-06 that assures releases from an FHA will be contained, processed, and filtered to limit dose even lower than required by regulations. Operation of the SGT system may be required to achieve the goals of the shutdown administrative program. The licensee addressed his shutdown administrative program in the letter dated September 29, 2005, as a response to an RAI.

For item 2), the NRC staff confirmed that changing Actions C.2.3 to C.2 and E.3 to E.1 is editorial and that the LCO 3.0.3 notes associated with Actions C and E are no longer required.

For item 3), the licensee withdrew the proposed change in the letter dated September 11, 2006, in connection with other items being withdrawn affecting the secondary containment and SGT operation.

4.13 TS 3.7.3, "Control Room Emergency Filtration (CREF) System"

Changes are proposed to the following two sections of this TS:

1) Deleted "During movement of irradiated fuel assemblies in the secondary containment, During CORE ALTERATIONS," from the applicability statement.

2) Deleted the portions of Actions D and F related to fuel movement and core alterations. As a result of these deletions, Actions D.2.3 became D.2 and F.3 became F.1. Additionally, the LCO 3.0.3 notes provided in Actions D and F were deleted.

The licensee stated that this TS establishes the operability requirements for the CREF system. Since CREF is not credited for the mitigation of the AST FHA, the need to ensure the operability of the CREF system during core alterations or fuel handling activities is no longer necessary. This change, combined with the addition of TS 3.9.10, is consistent with the scope and intent of TSTF- 51. Changing Actions D.2.3 to D.2 and F.3 to F.1 are editorial. The LCO 3.0.3 notes associated with Actions D and F are no longer needed.

The NRC staff reviewed the proposed change and concurred with the licensee’s determination that the CREF system is not credited for mitigation of an AST FHA and thus TS controls on its operation during refueling are not required. The other changes are editorial or clarifications. The NRC staff finds that the proposed changes are acceptable.

4.14 TS 3.7.4, "Control Room Air Conditioning (AC) System"

Changes are proposed to the following two sections of this TS:

1) Deleted "During movement of irradiated fuel assemblies in the secondary containment, During CORE ALTERATIONS," from the applicability statement.

2) Deleted the portions of Actions C and E related to fuel movement and core alterations. As a result of these deletions, Actions C.2.3 became C.2, and E.3 became E.1. Additionally, the LCO 3.0.3 note provided in Actions C and E was deleted.

The licensee states that this TS establishes the operability requirements for the CR air-conditioning system. The CR air-conditioning system provides temperature control for the

CR following isolation of the CR. Since CR isolation is not credited for the mitigation of the AST FHA, the operability of the CR air-conditioning system during core alterations or fuel handling activities is no longer necessary. This change, combined with the addition of TS 3.9.10, is consistent with the scope and intent of TSTF-51. Changing Actions C.2.3 to C.2 and E.3 to E.1 are editorial. The LCO 3.0.3 note associated with Actions C and E is no longer needed.

The NRC staff reviewed the proposed change and concurred with the licensee's determination that the CR air-conditioning system is not credited for mitigation of an AST FHA and, thus, TS controls on its operation during refueling are not required. The other changes are editorial or clarifications. The NRC staff finds that the proposed changes are acceptable.

4.15 TSs Sections 3.8.2, 3.8.5, and 3.8.8

4.15.1 TS 3.8.2, "AC Sources - Shutdown"

Changes are proposed to the following two sections of this TS:

1) Deleted "During movement of irradiated fuel assemblies in the secondary containment" from the applicability statement.

2) Deleted the portions of Actions A and B related to core alterations and fuel movement. As a result of these deletions, Actions A.2.3 became A.2.1, A.2.4 became A.2.2, B.3 became B.1 and B.4 became B.2. Additionally, the LCO 3.0.3 note provided for the actions was deleted.

4.15.2 TS 3.8.5, "DC Sources - Shutdown"

Changes are proposed to the following two sections of this TS:

1) Deleted "During movement of irradiated fuel assemblies in the secondary containment" from the applicability statement.

2) Deleted the portions of Action A related to core alterations and fuel movement. As a result of these deletions, Actions A.2.3 became A.2.1 and A.2.4 became A.2.2. Additionally, the LCO 3.0.3 note provided for this action was deleted.

4.15.3 TS 3.8.8, 'Distribution Systems - Shutdown"

Changes are proposed to the following two sections of this TS:

1) Deleted "During movement of irradiated fuel assemblies in the secondary containment," from the applicability statement.

2) Deleted the portions of Action A related to core alterations and fuel movement. As a result of these deletions, Actions A.2.3 became A.2.1, A.2.4 became A.2.2 and A.2.5 became A.2.3. Additionally, the LCO 3.0.3 note provided for this action was deleted.

4.15.4 TSs Sections 3.8.2, 3.8.5 and 3.8.8 Discussion

TSs 3.8.2, 3.8.5, and 3.8.8 (which are currently applicable in Modes 4, 5, and during movement of irradiated fuel assemblies in the secondary containment) required, in part, immediate suspension of movement of irradiated fuel in secondary containment when both offsite preferred sources, redundant safety-related electric onsite power sources, or redundant safety-related distribution systems are no longer operable. The proposed change would delete "During movement of irradiated fuel assemblies in the secondary containment" from the applicability statement and the portions of Action A related to core alteration and fuel movement. The proposed change would add new TS 3.9.10 to ensure compliance with the decay time assumption used in the AST FHA analysis. This new TS requires a 24-hour decay time before in-vessel fuel movement can commence. A new surveillance requirement is provided to verify compliance with the required decay time prior to the movement of irradiated fuel. The proposed TS changes would allow, without TS restrictions, the movement of irradiated fuel assemblies that have decayed at least 24 hours when there is no offsite power, when there is no onsite power, or when there is no alternating current (AC) and direct current (DC) electric power through the electric distribution system to safety-related loads.

TSTF-51 guidelines for systems removed from service during movement of irradiated fuel that has decayed for 2 days or more, and during core alternations state that:

> "During fuel handling/core alternations, ventilation system and radiation monitor availability (as defined in NUMARC 91-06) should be assessed, with respect to filtration and monitoring of releases from the fuel. Following shutdown, radioactivity in the fuel decays away fairly rapidly. The basis of the Technical Specification operability amendment is the reduction in doses due to such decay. The goal of maintaining ventilation system and radiation monitor availability is to reduce doses even further below that provided by the natural decay. A single normal or contingency method to promptly close primary or secondary containment penetration should also be developed. Such prompt methods need not completely block the penetrations or be capable of resisting pressure. The purpose of the "prompt" methods mentioned above is to enable ventilation systems to draw the release from a postulated fuel handling accident in the proper direction such that it can be treated and monitored."

The NRC staff requested that the licensee justify the inoperability of equipment when the movement of irradiated fuel assemblies that have decayed at least 24 hours, specifically the availability of equipment needed to maintain plant shutdown (as described above in TSTF-51), for monitoring and maintaining the plant status, or to mitigate events postulated during an FHA. In response to the NRC staff's request, in a letter dated September 29, 2005, the licensee states that the plant will be in Mode 5 during periods when refueling-related movement of irradiated fuel in the reactor vessel are taking place. During Mode 5, selected AC sources, DC sources, and associated electrical distribution systems will be required to be operable by the TS. If one of these required electrical systems, structures, or components become inoperable, the existing TS Required Actions to immediately restore the required offsite power circuit, the required diesel generator, the required DC electrical power subsystem, and the required AC and DC electrical power distribution subsystems have not been altered by the proposed change. The operability requirements for those systems required to maintain the plant in a safe shutdown condition and maintain adequate decay heat removal are also not affected by the proposed changes to TS 3.8.2, 3.8.5, and 3.8.8. The existing refueling-related TS requirements in Section 3.9 of the TS regarding reactivity control and the operation of the shutdown cooling mode for the residual heat removal systems are not relaxed or altered.

The licensee also stated that the proposed changes are acceptable because the loss of an electrical SSC will not prevent the plant's ability to mitigate the consequences of a postulated FHA, as required, to ensure the consequences are within the regulatory acceptance limits. As documented in the AST FHA analysis, no credit for secondary containment, the SGTS, the CREF system, or any other electrically powered mitigation system is required or assumed. In addition, the loss of TS-required electrical SSCs do not increase the probability of an FHA. The refueling bridge/crane used to move fuel during refueling is fail safe (i.e., would fail as-is) and is typically powered by non-safety power sources. The loss of power to this equipment would effectively result in the suspension of fuel handling activities. The movement of fuel could not be resumed until power was restored. The licensee is committed to maintaining radiological dose as low as is reasonably achievable and is committed to managing outage risk by maintaining an appropriate defense-in-depth during outage activities. Plant Procedure Manual (PPM) 1.16.8A, "Outage Risk Management," mandates the development of a Shutdown Safety Plan. This procedure implements the guidance in NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management." The shutdown safety plan developed for each outage addresses the defense-in-depth for each of the key safety functions identified in NUMARC 91-06 and establishes contingency plans as appropriate to manage risk. Containment is one of the key safety functions identified in NUMARC 91-06 and is explicitly addressed in the CGS Shutdown Safety Plan.

Consistent with the reviewer's note included in TSTF-51, Revision 2 (which is the quoted text provided in the NRC staff's RAI), the licensee will revise PPM 1.16.8 A to require an assessment of the availability of the SGT ventilation system and the reactor building vent exhaust plenum radiation monitor with respect to the filtration and monitoring, respectively, of a postulated release from the fuel during the time periods when refueling-related fuel handling activities are taking place. This procedure will also be revised to require a contingency plan to promptly close the secondary containment if an FHA with radiological consequences was to occur during these fuel handling activities.

The licensee will continue to require and provide local area radiation monitoring during the movement of irradiated fuel that will serve to promptly detect any significant release of radioactive gases due to an FHA and prompt an appropriate response to an event of this type. The same area radiation monitors used to monitor the radiological conditions on the refueling floor during refueling activities will continue to be used after the approval and implementation of the requested changes. These monitors include: the spent fuel pool area radiation monitor, ARM-RE-2, and the refueling bridge/crane local alarming area radiation monitor. The spent fuel pool radiation monitoring instrumentation is required to be OPERABLE by the current licensee controlled specification (LCS) 1.3.7.5. This LCS is applicable "When fuel is stored in the Spent Fuel Pool." If this monitor was to become inoperable during fuel movement, the compensatory measure required by the LCS is to immediately provide portable continuous monitoring in the same vicinity. The setpoint for the spent fuel pool radiation monitor is required to be less than or equal to 20 mR/hr by the LCS. If a radiation condition occurs that exceeds the setpoint, a "Refueling Floor Area Rad High" alarm is annunciated in the CR.

The Annunciator Response Procedure (ARP) 4.602.A5 requires the suspension of all work activities (i.e., the suspension of any fuel movement activities) on the refueling floor and an evacuation of the area. This ARP also refers the operator to the Abnormal Condition Procedures ABN-RAD-HIGH and ABN-FUEL-HAND. In addition to the ARP's reference to these Abnormal Condition Procedures, the alarming of the refueling bridge/crane radiation monitor is a specified entry condition to the ABN-FUEL-HAND procedure. The licensee will revise ABN-FUEL-HAND procedure to direct the operator to implement the contingency plans of the Safe Shutdown Plan for closing secondary containment and placing a train of the SGT and the reactor building vent exhaust plenum monitor in service. This procedure direction will be applicable during outage-related refueling activities. The operation of the refueling bridge/crane local alarming area monitor is a procedurally-required prerequisite for refueling bridge operation. These requirements are contained in the System Operating Procedure 2.14.1, "Refueling Bridge Operation," and Fuel Handling and Refuel Activities Procedure 6.3.23, "Handling Irradiated Fuel in the Spent Fuel Pool." Procedure 2.14.1 requires the establishment of direct communication between the refueling bridge and the CR and verification that the reactor has been subcritical for at least 24 hours. During the performance of core alternations, this procedure requires the licensee to maintain continuous communication between the refueling bridge and the CR. Procedure 6.3.23 requires immediate notification of the CR in the event of fuel damage. As stated above, an alarming refueling bridge area monitor is an entry condition for the ABN-FUEL-HAND procedure.

The licensee has opted for continued use of the TID-14844 source term, which provides integrated doses for equipment enveloping those that would be calculated using AST as the current licensing basis. The NRC staff concurs with the licensee's determination that the source term associated with environmental qualification of equipment will remain consistent with 10 CFR 50.49. The NRC staff also finds that the licensee has provided adequate justification for the requested TS changes and, therefore, the proposed changes are acceptable.

4.16 TS 3.9.7, "Reactor Pressure Vessel (RPV) Water Level - New Fuel or Control Rods"

Changes are proposed to the following two sections of this TS:

> Increased the required water level above the top of irradiated fuel assemblies seated within the RPV in the LCO from 22' to 23'. Similarly, increased 22' to 23' in SR 3.9.7.1.

The NRC staff reviewed this change and determined that the change establishes operational requirements consistent with assumptions of the AST FHA analysis and is consistent with the requirements of 10 CFR 50.36. As such, the NRC staff finds this change to be acceptable.

4.17 TS 3.9.10, "Decay Time"

New TS 3.9.10 is proposed to ensure compliance with the decay time assumption used in the AST FHA analysis.

The NRC staff reviewed the proposed addition of a new "Decay Time" specification. This new TS requires a 24-hour decay time before in-vessel fuel movement can commence. A new SR is provided to verify compliance with the required decay time prior to the movement of irradiated fuel. Criterion 2 of 10 CFR 50.36(c)(2)(ii)(B) in part, specifies an operating restriction that is an initial condition of a design-basis analysis as an item that must have a supporting LCO. A 24-hour decay time is assumed in the development of the source term used in the AST FHA analysis. The NRC staff finds that specifying the decay time in the TS with an appropriate SR meets the requirements of the Commissions regulation and, thus, is acceptable.

4.18 TS 5.5.7, "Ventilation Filter Test Program (VFTP)"

> Revised the acceptable SGT system flow rates from a range of 4012 to 4902 cfm to a range 4320 to 5280 cfm in parts a, b, and d of this program description.

The licensee stated that the new GOTHIC model for the secondary containment draw down analysis credits a SGT system flow rate of 4800 cfm. The new 4800 cfm value for SGT system flow rate has been evaluated to ensure 99 percent filter efficiency credit in the design-basis analyses. No changes to plant equipment or equipment setpoints are required. The proposed SGT system flow rate for filter test purposes is 4320 to 5280 cfm (i.e., 4800 ± 10 percent). This flow range complies with American National Standards Institute Standard N510-1989, "Testing of Nuclear Air Treatment Systems."

The NRC staff reviewed this change, considers it to be conservative, and finds that it is acceptable because it requires better system performance.

5.0 REGULATORY COMMITMENTS

The licensee made the following regulatory commitments:

1. PPM 1.16.8A will be revised to require an assessment of the availability of the SGT ventilation system and the reactor building vent exhaust plenum radiation monitor with respect to the filtration and monitoring, respectively, of a postulated release from the fuel during the time periods when refueling related fuel handling activities are taking place.

2. PPM 1.16.8A will also be revised to require a contingency plan promptly to close the secondary containment if an FHA with radiological consequences was to occur during these fuel handling activities.

3. The licensee will continue to require and provide local area radiation monitoring during the movement of irradiated fuel that will serve promptly to detect any significant release of radioactive gases due to an FHA and prompt an appropriate response to an event of this type. The same area radiation monitors used to monitor the radiological conditions on the refueling floor during refueling activities will continue to be used after the approval and implementation of the requested TS changes. These monitors include: the spent fuel pool area radiation monitor, ARM-RE-2, and the refueling bridge/crane local alarming area radiation monitor.

4. The ABN-FUEL-HAND procedure will be revised to direct the operator to implement the contingency of the Safe Shutdown Plan for closing secondary containment and placing a train of SGT and the reactor building vent exhaust plenum monitor in service. This procedure direction will be applicable during outage related refueling activities.

The above compensatory measures have been entered as regulatory commitments in the licensee's Commitment Management System, which complies with Nuclear Energy Institute's Document 99-04, "Guidelines for Managing NRC Commitment Changes." The NRC staff has reviewed the compensatory measures and how they will be controlled, and finds that the licensee's commitments, combined with the existing procedural requirements and administrative controls, provide an appropriate level of defense-in-depth to manage risk during outage activities. Therefore, the NRC staff finds that the proposed TS changes acceptable.

6.0 STATE CONSULTATION

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

7.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the 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 published on October 26, 2004 (69 FR 62472). 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.

8.0 CONCLUSION

As described above, the NRC staff reviewed the assumptions, inputs, and methods used by EN to assess the radiological impacts of the proposed license amendment at CGS. The NRC staff finds that analysis methods and assumptions consistent with the conservative regulatory requirements and guidance identified in Section 2.0 above were used. The NRC staff compared the doses estimated by EN to the applicable criteria identified in Section 2.0. The NRC staff finds, with reasonable assurance, that the licensee's estimates of the EAB, LPZ, and CR doses will continue to comply with these criteria. Therefore, the proposed license amendment is acceptable, with regard to the radiological consequences of postulated DBAs.

This licensing action is considered a full implementation of the AST. With this approval, the previous accident source term in the CGS design-basis is superseded by the AST proposed by EN. The previous offsite and CR accident dose criteria, expressed in terms of whole body, thyroid, and skin doses, are superseded by the TEDE criteria of 10 CFR 50.67 or fractions thereof, as defined in RG 1.183. All future radiological analyses performed to demonstrate compliance with regulatory requirements shall address all characteristics of the AST and the TEDE criteria as described in the CGS design basis.

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.

9.0 REFERENCES

NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," February 1995.

NUREG/CR-5732, "Iodine Chemical Forms in LWR [Light-Water Reactor] Severe Accidents," January 1992.

NUREG/CR-5950, "Iodine Evolution and pH Control," October 1992.

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," SRP 6.5.2, "Containment Spray as a Fission Product Cleanup System," Revision 2, 1988.

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

"Safety Evaluation for Proposed Changes to the Grand Gulf Technical Specifications Implementing Alternate Source Term for Grand Gulf Nuclear Station," January 23, 2001 (ADAMS Accession No. ML010250120).

Principal Contributors: D. Nguyen, M. Yoder, T. Ford, E. Forrest

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Table 1

COLUMBIA GENERATING STATION ACCIDENT ANALYSIS PARAMETERS AND ASSUMPTIONS

General

Reactor power (3486 x 1.02), megawatts thermal	3,556
Core Inventory	Submittal Table 4.4-4
Dose conversion factors	FGR11/FGR12
Breathing rates, offsite, m^3/s	
0-8 hours	3.5E-4
8-24 hours	1.8E-4
>24 hours	2.3E-4
Breathing rate, control room, m^3/s	3.5E-4
Unfiltered inleakage due to ingress and egress, cfm	10
Control room unfiltered infiltration in emergency pressurization mode, cfm[2]	
One train in service	50
Both trains in service	75
Control room normal intake flow, cfm	1100[3]
Control room filtered pressurization minimum flowrate, cfm	
One train in service	800
Both trains in service	1300
Control room filtered recirculation, cfm	none
Control room volume, ft^3	214,000
Control room intake filter efficiency, %	
Particulates	99
Elemental and Organic iodine	95
Noble Gases	0
Control room occupancy factor	
0-24 hrs	1.0
1-4 days	0.6
4-30 days	0.4

[2] Includes 10 cfm unfiltered inleakage for ingress and egress. The 10 cfm unfiltered inleakage is added to the measured unfiltered inleakage to determine the total unfiltered inleakage.

[3] For the FHA and CRDA, emergency pressurization is not credited. The total unfiltered inleakage is assumed to be 100,000 cfm based upon a sensitivity study performed by the licensee. For the MSLB, the dose is taken at the CR intake and is not dependent upon the flow into the CR.

Loss-of-Coolant Accident (LOCA)

Containment Leakage Source

Onset of gap release phase, min	2.0

Core release fractions and timing–Containment atmosphere

Duration, hrs	0.5	1.5
Noble Gases:	0.05	0.95
Halogens:	0.05	0.25
Alkali Metals:	0.05	0.20
Tellurium:	0.00	0.05
Strontium:	0.00	0.02
Barium:	0.00	0.02
Noble Metals:	0.00	0.0025
Cerium Group:	0.00	0.0005
Lanthanides:	0.00	0.0002

Iodine species distribution	
Aerosol	0.95
Elemental	0.0485
Organic	0.0015
Primary containment volume, ft^3	
Drywell	200,540
Suppression pool air space	144,184
Suppression pool water volume	137,262
Containment leakrate, %/day	
0- 24 hours	0.5
Greater than 24 hours	0.25
Standby Gas Treatment System (SGTS) Filter Effective Efficiency, %	
Before drawdown (all species)	0
After drawdown (all species except noble gases)[4]	98
After drawdown (noble gases)	0
Secondary containment volumetric flow rate bypassing SGTS filters, cfm	
After drawdown	50
Secondary containment bypass, (% primary containment volume/day)	0.04
Secondary containment volume[5], ft^3	5,000
SGTS drawdown time, min	20

[4] The SGTS filter efficiency for all forms of iodine and for particulates is 99 percent. A filter bypass of 50 cfm was also assumed. This reduces the effective filter efficiency to a value of 98 percent.

[5] The SGTS was assumed to have a flow rate of 5,000 cfm. The volume is artificially set to SGTS flow rate volume exhausted in 1 minute to represent no mixing or holdup credit in the secondary containment.

Drywell natural deposition	
Particulate	None
Elemental	None
Control room isolation delay, minutes	0
Spray Initiation Time, minutes	15

Aerosol Drywell Spray Removal Rates

Time, hr	Removal Rate, 1/hr
0	0.00
0.25	6.20
2.44	0.62
24.0	0.00

Main Steam Line Isolation Valves (MSIV) Leakage

MSIV technical specification leak rate[6] at test pressure of ≥25 psig, scfh	
One line	16
Total	64
Normal Steam line (and steam) temperature, °F	544.0

Engineered Safety Feature (ESF) Leakage

Iodine species, %	
Particulate/aerosol	0
Elemental	97
Organic	3
Iodine flash fraction	0.1
SGTS charcoal filtration	Credited
ESF estimated leakage into secondary containment, gpm	1[7]
Leakage to CST, gpm	0.24

Atmospheric Dispersion Factors χ/Q values, sec/m^3

Control Room	Submittal Table 4.4-2 and 4.4-3 Amended Submittal[8]
Offsite	Submittal Table 4.3-5

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[6] MSIV leakage is reduced 50 percent after 24 hours.

[7] Both the ESF and CST leakage are doubled per Regulatory Guide 1.183. Values not corrected for accident conditions such as the ESF leakage, are taken to be at accident conditions.

[8] Table 4.3-4 of the original submittal is amended by the licensee's response to the RAI question number 33, dated March 21, 2006.

Fuel Handling Accident

Peaking factor	1.7
Fuel rods damaged, rods[9]	250
Decay period, hrs	24
Pool decontamination factor	
Iodine	200
Noble Gases	1
Particulate	Infinite
Fraction of core in gap	
I-131	0.08
Kr-85	0.10
Other iodines	0.05
Other noble gases	0.05
Alkali Metals[10]	0.12
Release period, hr	2
Release location	
SGTS operating	SGT stack
No SGTS	RB vent stack
Control Room Emergency Filtration (CREF) initiation	Not Credited
Control room normal unfiltered intake, cfm	1,100
Water Depth, ft	$\geq$23
Atmospheric Dispersion Factors χ/Q values, sec/m^3	Submittal Table 4.4-2 and 4.4-3

Main Steam Line Break

MSIV closure time, sec	6
Reactor coolant system pressure, psia	1,060
Reactor coolant system temperature, degrees F	552
Reactor coolant activity, μCi/gm dose equivalent I-131	
Normal	0.2
Spike	4.0

[9] Based upon fuel assembly with an 8x8 fuel pin array, but applied to all fuel types.

[10] Cesium and rubidium are present and were considered, but were not included in the calculation because the DF for particulate is assumed to be infinite.

Radioactivity release rate to environment	Instantaneous
Control Room Occupancy Factor	1
Control room CREF initiation	Not Credited
Atmospheric Dispersion Factors χ/Q values, sec/m^3	Submittal Table 4.5-1

Control Rod Drop Accident

Peaking factor	1.7
Fraction of core Inventory in gap	
Noble gases	0.1
Iodine	0.1
Other Halogens	0.05
Alkali Metals	0.12
Percentage of core with damaged rod, %	1.8
Percentage of damaged rods that fail,	0.77
Melted fuel release fraction to vessel	
Noble gases	1.0
Iodine	0.5
Br	0.3
Alkali Metals	0.25
Tellurium Metals	0.05
Ba, Sr	0.02
Noble metals	0.0025
Ce	0.0005
La	0.0002
Fraction of activity released to vessel that enters main condenser	
Noble gases	1.0
Iodine	0.1
others	0.01
Fraction of activity released from main condenser	
Noble gases	1.0
Iodine	0.1
others	0.01
Main condenser (plus LP turbine) free volume, ft^3	144,000
Release rate from main condenser, %/day	1
Release duration, hours	24
Control room CREF initiation	Not Credited
Atmospheric Dispersion Factors χ/Q values, sec/m^3	Submittal Table 4.6-1

cc:

Mr. W. Scott Oxenford (Mail Drop PE04)
Vice President, Technical Services
Energy Northwest
P.O. Box 968
Richland, WA 99352-0968

Mr. Albert E. Mouncer (Mail Drop PE01)
Vice President, Corporate Services/
General Counsel/CFO
Energy Northwest
P.O. Box 968
Richland, WA 99352-0968

Chairman
Energy Facility Site Evaluation Council
P.O. Box 43172
Olympia, WA 98504-3172

Mr. Douglas W. Coleman (Mail Drop PE20)
Manager, Regulatory Programs
Energy Northwest
P.O. Box 968
Richland, WA 99352-0968

Mr. Gregory V. Cullen (Mail Drop PE20)
Supervisor, Licensing
Energy Northwest
P.O. Box 968
Richland, WA 99352-0968

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011-4005

Chairman
Benton County Board of Commissioners
P.O. Box 190
Prosser, WA 99350-0190

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
P.O. Box 69
Richland, WA 99352-0069

Mr. Dale K. Atkinson (Mail Drop PE08)
Vice President, Nuclear Generation
Energy Northwest
P.O. Box 968
Richland, WA 99352-0968

Mr. William A. Horin, Esq.
Winston & Strawn
1700 K Street, N.W.
Washington, DC 20006-3817

Mr. Matt Steuerwalt
Executive Policy Division
Office of the Governor
P.O. Box 43113
Olympia, WA 98504-3113

Ms. Lynn Albin
Washington State Department of Health
P.O. Box 7827
Olympia, WA 98504-7827

Technical Services Branch Chief
FEMA Region X
130 228th Street, S.W.
Bothell, WA 98201-9796

Ms. Cheryl M. Whitcomb (Mail Drop PE03)
Vice President, Organizational
Performance & Staffing/CKO
Energy Northwest
P.O. Box 968
Richland, WA 99352-0968

Assistant Director
Nuclear Safety and Energy Siting Division
Oregon Department of Energy
625 Marion Street, NE
Salem, OR 97301-3742

Mr. J. V. Parrish
Chief Executive Officer
Energy Northwest
Mail Drop 1023
P.O. Box 968
Richland, WA 99352-0968