ENERGY NORTHWEST

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December 3, 2001
GO2-01-156

Docket No. 50-397

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Gentlemen:

Subject: **COLUMBIA GENERATING STATION, OPERATING LICENSE NPF-21 LICENSE AMENDMENT REQUEST -- ALTERNATIVE SOURCE TERM**

References:
1) Letter GO2-96-199, dated October 15, 1996, PR Bemis (Washington Public Power Supply System) to NRC, "Request for Amendment to Secondary Containment and Standby Gas Treatment System Technical Specifications"

2) Letter GO2-99-133, dated July 16, 1999, RL Webring (Energy Northwest) to NRC, "Withdrawal of Request for Amendment to Secondary Containment and Standby Gas Treatment System Technical Specifications"

3) Letter GO2-01-116, dated August 16, 2001, RL Webring (Energy Northwest) to NRC, "Resubmittal Plan - Request for Amendment to Secondary Containment and Standby Gas Treatment System Technical Specifications"

In accordance with the Code of Federal Regulations, Title 10, Parts 2.101, 50.67, 50.59 and 50.90, Energy Northwest hereby submits a request for amendment to the Columbia Generating Station Operating License. Specifically, we are requesting a revision to the licensing and design bases to reflect application of alternative source term methodology, with the exception that TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," will continue to be used as the radiation dose basis for equipment qualification.

On December 23, 1999, the NRC published new regulation 10 CFR 50.67, "Accident Source Term," in the Federal Register. This regulation provides a mechanism for licensed power reactors to replace the traditional source term used in design-basis accident analyses with an alternative source term. The direction provided in 10 CFR 50.67 is that licensees who seek to revise their current accident source term in design basis radiological consequences analyses should apply for a license amendment under 10 CFR 50.90.

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Regulatory guidance for the implementation of the alternative source term is provided in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 2000. This regulatory guide provides guidance to licensees of operating nuclear plants on acceptable applications of alternative source terms. The use of an alternative source term changes only the regulatory assumptions regarding the analytical treatment of the design basis accidents.

The alternative source term analyses for Columbia Generating Station were performed following the guidance in accordance with Regulatory Guide 1.183 and Standard Review Plan Section 15.0.1, "Radiological Consequences Analyses Using Alternative Source Terms." The analyses covered the control rod drop, fuel handling, loss of coolant, and main steam line break accident scenarios.

The proposed changes to the current licensing basis for Columbia Generating Station that are justified by the alternative source term analyses include:

- Revisions to several Technical Specifications and associated Bases to reflect implementation of alternative source term methodology.
- Revisions to main steam leakage control system Technical Specifications and associated Bases to reflect the proposed deactivation of the system.
- Revisions to standby gas treatment system Technical Specifications and associated Bases to resolve a Justification for Continued Operation (JCO) regarding the establishment of secondary containment vacuum under adverse environmental conditions (References 1, 2 and 3).
- Revisions to standby gas treatment system Technical Specifications and associated Bases to increase allowable secondary containment bypass leakage.
- Revisions to several Technical Specifications and associated Bases to reflect that secondary containment and the standby gas treatment system are no longer required during movement of irradiated fuel assemblies or during core alterations.
- Revision to the bounding radiological analysis for the loss of coolant accident to reflect that it is an inadequate core cooling accident that degrades to core damage, rather than the double-ended guillotine break of the recirculation system pump suction piping.
- Resolution of a previously-reported Unreviewed Safety Question (USQ) pertaining to increased unfiltered control room in-leakage into the control room envelope.
- Development of new offsite and control room atmospheric dispersion factors (X/Qs) that were calculated using site-specific meteorology data collected between 1985 and 1989.
- Use of the standby liquid control system to buffer suppression pool pH to prevent iodine re-evolution during a postulated radiological release.
- Use of the residual heat removal drywell spray system post-loss of coolant accident to wash inorganic iodine and particulates from the drywell atmosphere to the suppression pool.

Additional information is attached to this letter to complete the amendment request. Attachment 1 provides a description and the basis for the acceptability of the proposed changes associated with alternative source term methodology. Attachment 2 consists of a comparison table reflecting Regulatory Guide 1.183 requirements. Attachment 3 provides a description and the basis for acceptability of the proposed changes associated with the Technical Specifications and the design basis analyses issues pertaining to secondary containment draw-down and unfiltered control room in-leakage (JCO and USQ respectively). Attachment 4 consists of the 10 CFR 50.92 evaluation (no significant hazards consideration). Attachment 5 contains the environmental considerations evaluation. Attachment 6 contains the marked-up pages of the Technical Specifications showing the proposed changes. Attachment 7 consists of the typed Technical Specification pages, as they would be revised by this amendment request. Attachment 8 consists of a marked-up copy of the Technical Specification Bases associated with this proposed change.

Energy Northwest has concluded that changes proposed in this letter do not result in a significant hazards consideration. The changes proposed in this letter have also been evaluated using the identification criteria for licensing and regulatory actions requiring an environmental assessment as specified in 10 CFR 51.21. The proposed amendment meets the eligibility criteria for a categorical exclusion as set forth in 10 CFR 51.22. Therefore, an environmental assessment of the proposed change is not required.

This amendment request has been approved by the Columbia Generating Station Plant Operations Committee and reviewed by the Energy Northwest Corporate Nuclear Safety Review Board. In accordance with 10 CFR 50.91, the State of Washington has been provided a copy of this letter. The amendment request is also consistent with submittals associated with application of alternative source term that have been previously provided to the staff by the Nuclear Management Company for the Duane Arnold Energy Center (Letter NG-00-1589, dated October 19, 2000, Attachment 4) and the Carolina Power & Light Company for Brunswick Steam Electric Plant Unit Nos. 1 and 2 (Letter BSEP 01-0063/TSC-2001-04, dated August 1, 2001, Enclosure 2).

We plan to discuss our proposed implementation schedule with the staff at a later date. Should you have any questions or desire additional information pertaining to this matter, please call RN Sherman at (509) 377-8616.

Respectfully,

RL Webring

RL Webring
Vice President, Operations Support/PIO
Mail Drop PE08

Attachments

cc: EW Merschoff - NRC RIV
JS Cushing - NRC NRR
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JO Luce - EFSEC
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STATE OF WASHINGTON)
)
)
COUNTY OF BENTON)

Subject: Operating License NPF-21
Request for Amendment
Alternative Source Term

I, RL Webring, being duly sworn, subscribe to and say that I am the Vice President, Operations Support/PIO for ENERGY NORTHWEST, the applicant herein; that I have the full authority to execute this oath; that I have reviewed the foregoing; and that to the best of my knowledge, information, and belief the statements made in it are true.

DATE 12/3/ , 2001

RL Webring
Vice President, Operations Support/PIO

On this date personally appeared before me RL Webring, to me known to be the individual who executed the foregoing instrument, and acknowledged that he signed the same as his free act and deed for the uses and purposes herein mentioned.

GIVEN under my hand and seal this 3rd day of December 2001

LORI A. WALLI
COMMISSION EXPIRES
NOTARY
PUBLIC
3-29-05
STATE OF WASHINGTON

Lori A. Walli
Notary Public in and for the
STATE OF WASHINGTON

Residing at Richland

My Commission Expires 3-29-05

Proposed Alternative Source Term Changes

1.0 Introduction

Energy Northwest has performed radiological consequence analyses of the four design basis accidents (DBAs) that result in offsite exposure. These analyses were performed to support full scope implementation of Alternative Source Terms (AST) described in NRC Regulatory Guide 1.183,"Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors," (Reference 1).

The AST analyses have been performed in accordance with the guidance in Regulatory Guide 1.183, and Standard Review Plan Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," (Reference 2). One exception from the Regulatory Guide was taken; the atmospheric dispersion model used in the main steam line break consequence analysis. This exception is discussed in more detail in Section 2.3. The conformance with the guidance contained in Regulatory Guide 1.183 is summarized in Attachment 2. The implementation consisted of the following steps:

- Identification of the alternative source term based on plant-specific analysis of the fission product inventory,
- Calculation of the release fractions for the four boiling water reactor DBAs,
- Analysis of the atmospheric dispersion for the radiological propagation pathways,
- Calculation of deposition and removal mechanisms, and
- Calculation of offsite and control room personnel Total Effective Dose Equivalent (TEDE) doses.

2.0 Evaluation

2.1 Scope

2.1.1 Accident Radiological Consequence Analyses

The following accident analyses documented in the Columbia Generating Station Final Safety Analysis Report (FSAR) were addressed using methods and input assumptions consistent with the AST [Figure 1 and Energy Northwest Calculation NE-02-01-13 (Reference 3)]:

- FSAR Section 15.4.9, Control Rod Drop Accident
- FSAR Section 15.6.4, Steam System Piping Break Outside Containment

- FSAR Section 15.6.5, Loss of Coolant Accidents (Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary) –Inside Containment

- FSAR Section 15.7.4, Fuel Handling Accident

The analyses were based on current operating conditions and the proposed changes related to secondary containment draw-down analysis. The results demonstrate compliance with 10 CFR 50.67 and GDC 19 of 10 CFR 50 Appendix A.

2.1.2 NUREG-0737, Item II.B.2

Energy Northwest has determined that continued compliance will be maintained with NUREG-0737, Item II.B.2, "Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems Which May be Used in Post-Accident Operations." The decision is based, in part, on the review of an NRC memorandum from JE Rosenthal to AC Thadani that addressed the potential impact of cesium concentration on equipment qualification (Reference 4).

2.2 Method of Evaluation

2.2.1 Fission Product Inventory

The ORIGEN code (Reference 5) was used in the calculation of the plant specific fission product source term inventories at the original rated power. The results of this original calculation have since been corrected for power uprate to 3486 MWth and further adjusted to 102% (3556 MWth) in support of the AST evaluations. The inventory results were also corrected to increase the impact of krypton and to increase the longer-lived isotopes.

2.2.2 Radiological Consequence

New calculations were prepared for the simulation of the radionuclide release, transport, removal, and dose estimates associated with the postulated accidents listed in Section 2.1.1.

The STARDOSE computer code, revision 0, January 1997 (Reference 6), was used for the dose calculations. The RADTRAD computer code (Reference 7), version 3.02, was also used in this task as a check of the STARDOSE results. The RADTRAD and STARDOSE programs are radiological consequence analysis codes used to estimate post-accident doses at plant offsite locations and in the control room. The RADTRAD code is publicly available and is used by the NRC staff in safety reviews.

Although the STARDOSE code is proprietary property of Polestar Applied Technology, Inc, the NRC has previously reviewed results obtained from the application of this code.

Offsite atmospheric dispersion factors (X/Qs) were calculated with the PAVAN computer code (Reference 8). The PAVAN code calculates the relative concentration at a receptor location from an accidental release of radioactivity into the environment per the guidance in Regulatory Guide 1.145 (Reference 9). The PAVAN code has been validated in accordance with Energy Northwest procedures. The code has been used by the NRC staff in safety reviews.

Control room atmospheric dispersion factors (X/Qs) were calculated with the ARCON96 computer code (Reference 10). The application of the code was consistent with the guidance provided in References 11 and 12. The ARCON96 code calculates relative concentrations in plumes from nuclear power plants at control room air intakes in the vicinity of the release point (Figures 2 through 5 show the layout of intakes and release points). The code has been used by the NRC staff in safety reviews.

The MicroShield code, version 5.03 (Reference 13), was used in the determination of the control room dose from "shine." MicroShield is a point kernel integration code used for general purpose gamma shielding analysis. Although it is not an NRC approved code, MicroShield has been used in safety-related applications by many nuclear plants in the United States. The code has been used to support licensing submittals that have been accepted by the NRC. Validation of the MicroShield code was also undertaken with the QADMOD code (Reference 14).

In addition to the calculation tools described above, the radiological consequence analyses made use of hand calculations and spreadsheets, supported by appropriate references, to determine inputs and outputs such as plant specific source terms, filter efficiency determinations, and suppression pool pH analyses.

2.3 Inputs and Assumptions

2.3.1 Accident Radiological Consequence Analyses

<u>Release Mode</u>

The accident analyses were performed for a core inventory based on 3556 MWth (102% of the rated power of 3486 MWth) in accordance with Regulatory Guide 1.49 (Reference 15). The reactor core inventory for the analyses was based on an assumed fuel irradiation time of 1000 effective

full power days (EFPD) that develops "equilibrium" activities in the fuel. Most fission products reach equilibrium within a three-year period. The calculated short-term inventories are approximately proportional to core thermal power. The inventories of the very long-lived isotopes, that did not approach equilibrium, can also be assumed to increase proportionally if the fuel irradiation time remains within the original basis.

The release source term is developed using a 66 isotope subset. In addition, Energy Northwest used barium and strontium release fractions of 2%. A 5% release fraction for barium and strontium if they were grouped with the Tellurium (Te) group in Table 5 of Regulatory Guide 1.183 is inconsistent with the 2% from Table 1. We consider Table 5 to be incorrect. The 66 radionuclides used in the analyses include the 60 identified as being potentially important contributors to TEDE in NUREG/CR-4691 (Reference 16) [two cobalt isotopes (58 and 60) that have a minor impact were deleted]. Four noble gas isotopes from the NRC-issued Technical Information Document (TID)-14844 (Reference 17), three other short-lived noble gas isotopes, and Ba137m were added to the subset for a total of 66. This difference to the guidance in Regulatory Guide 1.183 was proposed because the STARDOSE code incorporates a 66 radionuclide subset. Energy Northwest determined that this represented a negligible difference from the results based on the 60 radionuclide subset in Regulatory Guide 1.183. This postulated set of radionuclides available for release represents a change in the Columbia Generating Station design and licensing bases for the radiological consequence analysis.

Transport Mode and Meteorological System Design

Meteorological conditions, such as wind speed, wind direction, and stratified atmospheric temperature, are sensed by the meteorological tower instrumentation and recorded in the control room. Indicated meteorological conditions are used to calculate doses downwind due to a radiation release. Wind speed and direction are monitored by separate channels at the 33-ft and 245-ft elevations. Primary and backup channels provide the air temperature difference between 33-ft and 245-ft elevations. The instrumentation is subject to periodic testing to demonstrate continued operability.

The meteorological tower is located less than 0.5 mile west of the plant site, with its base at 455-ft mean sea level, and consists of a 240-ft structure with a 5-ft extension mast. The tower is triangular in shape and of open lattice construction to minimize tower interference with meteorological measurements.

The siting of the instrumentation with respect to the meteorological tower and surrounding vegetation is very good. For example, the instrumentation at the 33-ft level is on booms pointed in the direction of the mean wind and are more than twice the tower width from the tower. The instrumentation at the 245-ft level is on a mast directly above the tower structure; therefore, it is not impacted by any tower wake effects. The base of the tower has been maintained as natural vegetation. The area around the tower is open terrain with no natural or man-made obstructions that might distort the data being collected. The meteorological monitoring program for plant operation and instrument surveillance requirements are discussed in further detail in Columbia Generating Station FSAR Sections 2.2.2.1, 2.3.3.2 and 7.5.1.6.2 respectively.

Atmospheric dispersion coefficients were calculated, for the identified release paths, based on site-specific meteorology data collected between January 1985 through December 1989. The dispersion coefficients developed represent a change to those used in the current FSAR analyses. The values in the FSAR were based on Regulatory Guide 1.3 results from the PAVAN code. The Regulatory Guide 1.145 results from PAVAN were used for the offsite X/Qs for the AST analyses.

Testing was performed on the control room emergency filtration system at the nominal flow rate of 1000 cfm per train to measure the unfiltered leakage into the control room envelope. The parameters for unfiltered leakage into the control room used as input to the consequence analyses bound the measured plant data. Although there are variations in the nominal and unfiltered leakage rates assumed in the four accident analyses, the exposures resulting from the LOCA and the corresponding unfiltered in-leakage rates establish the limit for the allowable unfiltered in-leakage. The non-LOCA transients used a range of values up to 350 cfm (2 trains) and 200 cfm (1 train). However, the LOCA source term event used a maximum of 250 cfm for 2 trains and 125 cfm for a single train.

The standard breathing rates used for control room personnel dose assessments and for the offsite personnel are shown in Table 1. Control room occupancy factors used are also included in Table 1. The values for breathing rates and occupancy factors are consistent with Regulatory Guide 1.183.

Removal Mode

Removal mechanisms are included in the applicable event-specific discussions.

2.3.1.1 LOCA Inputs and Assumptions

The key inputs used in this analysis are included in Tables 2 through 4. These inputs and assumptions are grouped into three main categories (release, transport, and removal).

LOCA Release Inputs

The LOCA analysis assumes the primary-to-secondary containment leakage rate at the limit of 0.5%/day specified in the technical specifications. As discussed in Appendix A, Section 3.7, of Reference 1, the primary containment leakage is reduced by 50% after 24 hours, based on the post-LOCA drywell pressure history. In addition to the primary containment leakage, 0.04%/day is assumed to bypass the secondary containment and is unfiltered released to the atmosphere. In addition to this secondary containment bypass leakage, the analysis assumes the technical specification maximum allowable 46 scfh (11.5 scfh per steam line) main steam isolation valve (MSIV) leakage to the environment.

The analysis assumes a leak rate of two gpm into secondary containment from the engineered safety feature systems. Ten percent of the activity in the leakage is assumed to become airborne. This leak rate is twice the plant leakage accepted in the Columbia Generating Station original SER limit (Reference 18) and is consistent with Regulatory Guide 1.183. The engineered safety feature leakage rate is assumed to begin approximately 15 minutes following the accident, with the actuation of the drywell sprays, and to continue throughout the 30-day duration of the postulated accident. Prior to 15 minutes there is no engineered safety feature recirculation (hence no leakage) assumed since an emergency core cooling system failure is an implicit assumption of the core damage leading to the AST.

Regulatory Guide 1.183 accident isotopic release specification allows deposition of iodine in the suppression pool. Essentially all of the iodine is assumed to remain in solution as long as the pool pH is maintained above 7. The Columbia Generating Station emergency operating procedures direct operators, upon detection of

symptoms indicating that core damage is occurring (e.g., primary containment high radiation), to manually initiate the standby liquid control system. Although no credit is taken for any operator action during the first 10 minutes of an event, the analysis includes the assumptions: 1) borated solution injection is initiated following the accident; and 2) approximately 4000 lb of sodium pentaborate or equivalent is delivered into the suppression pool. The calculation results demonstrate the buffering effect of the boron solution maintains the suppression pool pH above 7 for the 30-day duration of the postulated LOCA. Maintaining suppression pool pH above 7, as an assumption in support of radiological consequence analysis, represents a change to the Columbia Generating Station design and licensing basis.

<u>LOCA Transport Inputs</u>

At the beginning of the event, a loss of offsite power is assumed which results in the loss of reactor building ventilation that maintains secondary containment at a negative pressure with respect to the outside atmosphere. A conservatively-assumed 10-minute period allows delays for the emergency diesel generators to start and load and for the standby gas treatment system to draw the secondary containment pressure down to 0.25 inches of vacuum water gauge. This 10-minute period is sufficiently conservative to bound actual system performance and adverse environmental conditions. Consistent with the guidance of Appendix A of Regulatory Guide 1.183, the analysis assumes that the primary containment leakage that bypasses secondary containment and the engineered safety feature leakage are released directly to the environment, unfiltered, at ground level. To maximize the calculated post-accident doses, the ground level reactor building releases were assumed to discharge from the building location closest to the control room air intake. Following the 10-minute period assumed for the secondary containment draw-down, the analysis assumes that the secondary containment releases are ground level releases that are treated by the standby gas treatment system.

If the main steam lines and the main condenser were to remain intact, the MSIV leakage would eventually collect in the main condenser. However, the analysis assumes that only the main steam lines between the MSIVs remain intact. The analysis also assumes that one of the four outboard MSIVs fails to close. Because of the undefined condition at the turbine stop valves (e.g., the possibility of the valves being stuck open and piping failed

beyond), there may be an opportunity for significant free convection of outside air into that portion of the line. Although the main steam lines from the outboard MSIVs to the turbine stop valves remain intact, that portion of the piping is ignored and only the portion between the MSIVs in the three lines in which both valves close is available for retention.

The control room is automatically isolated and the control room emergency filtration (CREF) system is automatically initiated upon a receipt of the reactor building discharge high radiation signal, low RPV water level, or high pressure in the primary containment. In the analysis, the accident activity was assumed to enter the control room for the first 30 minutes of the LOCA at a nominal CREF filtered ventilation flow rate of 1800 cfm with unfiltered leakage of 250 cfm. After 30 minutes, credit is taken for the operator action of isolating one CREF train, reducing the filtered flowrate to 900 cfm and unfiltered leakage to 125 cfm.

<u>LOCA Removal Inputs</u>

The activity released from the core is reduced by spraying the drywell. Deposition mechanisms in the main steam lines as well as air exhaust filtration systems in the reactor building also reduce releases. Spraying the drywell and the deposition removal mechanisms are characteristics of the AST methodology and represent a change in the plant design and licensing basis.

The spray removal coefficient was calculated using the model in Standard Review Plan Section 6.5.2. Main steam line pipe deposition for the three lines that isolate was modeled using the RADTRAD code with the Brockmann - Bixler pipe deposition model. The AEB-98-03 model confirmed the conservatism of the Brockmann - Bixler model (Reference 19).

A filter efficiency of 99%, with 50 cfm bypass was used in the analysis for the standby gas treatment system. For the CREF system, the filter efficiency of 95% was used for elemental and organic iodines, while 99% efficiency was assumed for particulate. The standby gas treatment system charcoal filter efficiency of 99% for elemental and organic iodines is consistent with Regulatory Guide 1.52 (Reference 20). The filter efficiencies are consistent with the Columbia Generating Station Technical Specifications.

2.3.1.2 MSLB Accident and Assumptions

The key inputs used in this analysis are included in Table 5.

The postulated accident assumes a double-ended break of one main steam line outside the primary containment with displacement of the pipe ends that permits maximum blowdown rates. The break mass released includes that amount in the steam line and connecting lines at the time of the break, plus the amount that passes through the valves prior to closure. The mass used in the analysis bounds this calculated release. The analysis assumes MSIV isolation in 6.0 seconds, which is longer than the maximum time allowed by the technical specifications for instrument response time and valve stroke time. Two activity release cases corresponding to the pre-accident spike and maximum equilibrium concentration allowed by technical specifications of 4 μCi/gm and 0.2 μCi/gm dose equivalent I-131 respectively were assumed. These released activity assumptions are consistent with Regulatory Guide 1.183.

The analysis assumes an instantaneous ground level release. There are three control room intakes. Two of these are remote; located away from the power block (400 feet or more from the MSLB release location). The third is the local intake, which is contiguous with the radwaste building. This is the control room intake closest to the MSLB release location and it is assumed that the bubble translates directly to this local control room intake. This minimizes the effect of plume rise and associated dilution. The contaminated air flow into the control room was assumed to be from unfiltered inleakage only. The bubble diameter for the primary transport pathway is a maximum of 200 feet. The bubble will only transit across the local control room intake.

The plume dilution calculation addresses the effects of plume buoyancy and air entrainment in a conservative approach that is a departure from the traditional assumption of an undiluted steam plume transiting horizontally across the control room intake with no rise. The calculation uses plant parameters for the MSLB accident (e.g., mass of liquid-steam mixture released, timing of release, temperature of the liquid-steam mixture, noble gas and iodine concentration in the release) to obtain the initial conditions of the released steam plume. The steam plume is treated as a bubble with a given transit time up to and across the control room intake. This is followed by an evaluation of the bubble rise due to the equilibrium between the buoyancy force (resulting from the

density difference between ambient air and hot steam) and the drag force resulting from the friction between the bubble mass and the surrounding air. The dilution effect was quantified as a dilution factor of 0.1. Energy Northwest has determined that this model is supported by other, more sophisticated buoyancy models for puff releases.

2.3.1.3 FHA Inputs and Assumptions

The key inputs used in this analysis are included in Table 6.

The postulated fuel handling accident involves the drop of a fuel assembly on top of the reactor core during refueling operations. The analysis assumes that 0.528% of the fuel pins in the full core are damaged. A radial peaking factor of 2.0 was assumed in the analysis in addition to the source term corrections discussed in Section 2.3.1. A post-shutdown 24-hour decay period was used to determine the release activity inventory. This assumption is consistent with plant procedures, but is conservative when compared to plant refueling outage history. The analysis assumes that gap activity in the affected rods was released instantaneously into the water in the reactor well. The analysis assumes the fuel bundle is dropped 34 feet, but assumes a water depth of only 22 feet above the assemblies seated in the reactor pressure vessel. The decontamination provided by the 22 feet is equivalent to the 23 feet discussed in Regulatory Guide 1.183 and is consistent with the limits in the Technical Specifications. With an assumed decontamination factor (DF) of 500 applied to elemental iodine and a DF of 1 applied to organic iodine, the expected iodine speciation of 57% elemental and 43% organic is derived. This is consistent with Regulatory Guide 1.183. The parameters bound a similar event that might occur in the spent fuel pool.

In accordance with Regulatory Guide 1.183, the analysis assumes that the activity in the reactor building environment is released within two hours from the reactor building as a ground release with no further credit for reactor building holdup or dilution, or standby gas treatment system operation.

The analysis assumes that the CREF system and control room isolation are initiated within five minutes after the accident. Following the initiation, the plume release was assumed to enter the control room at the filtered rate of 1800 cfm and an unfiltered in-leakage rate of 350 cfm. After 30 minutes, one train of the filtration system is secured and the flow rates drop to 900 cfm

filtered and 200 unfiltered. As discussed in Section 2.3.1, these unfiltered leakage rates are higher than those assumed in the LOCA analyses. Filter efficiencies for CREF are listed in Table 4.

2.3.1.4 CRDA Input and Assumptions

The key inputs used in this analysis are included in Table 7.

The plant design basis control rod drop accident (CRDA) involves the rapid removal of a highest worth control rod resulting in a reactivity excursion that encompasses the consequences of any other postulated CRDA. The core performance analysis shows that the energy deposition that results from this event is inadequate to damage fuel pellets or cladding. However, for the dose consequence analysis, we assume about 1.8% of the fuel pins in the full core were damaged, with melting occurring in 0.77 % of the damaged rods (e.g., 0.014% of the core). A core average radial peaking factor of 1.50 was assumed in the analysis.

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% per day for a period of 24 hours. No credit is taken for turbine building holdup or dilution.

The analysis takes no credit for the filtration of the control room intake air system, assuming an unfiltered intake of 1800 cfm for the 24 hour duration of the event.

The analysis assumptions for the transport, reduction, and release of the radioactive material from the fuel and the reactor coolant are consistent with the guidance provided in Appendix C of Regulatory Guide 1.183.

3.0 Results

3.1 Evaluation Results

3.1.1 Accident Radiological Consequence Analyses

The postulated accident radiological consequence analyses were reviewed and updated for AST implementation impact.

3.1.1.1 LOCA

The radiological consequences of the design basis LOCA were analyzed using the STARDOSE code and the inputs and assumptions discussed in Section 2.3.1.1 of this report. The RADTRAD check calculation confirms the STARDOSE results. The post-accident doses are the result of four distinct activity releases:

<u>Primary to secondary containment leakage</u>: This leakage is directly (e.g., untreated) released into the environment until the secondary containment draw-down is complete and it is filtered by the standby gas treatment system.

<u>Primary leakage, secondary containment bypass</u>. This portion of the primary leakage bypasses the secondary containment and is released directly into the environment.

<u>ESF system leakage into the secondary containment</u>. This leakage starts after the secondary containment draw-down is complete and; therefore, is filtered by the standby gas treatment system.

<u>MSIV leakage from the primary containment into the environment or turbine building</u>. The MSIV leakage is released, undiluted and unfiltered (except that removal in the intact steam lines is considered).

The postulated exposure to the control room occupants includes terms for:

- In-leakage internal cloud immersion and inhalation contribution from the primary containment, secondary containment bypass, ESF, and MSIV leakage releases (major contribution).

- External cloud contribution from the primary containment, secondary containment bypass, ESF, and MSIV leakage releases. This term takes credit for control room structural shielding (minor contribution).

- A direct dose contribution from the secondary containment contained accident activity. This term takes credit for both reactor building and control room/ structural shielding (minor contribution).

The LOCA control room dose corresponds to an assumed unfiltered in-leakage rate of 250 cfm that is reduced to 125 cfm 30 minutes post-LOCA. In addition, the total control room dose includes a "shine" contribution from the reactor building. Table 8 presents the results of the LOCA radiological consequence analysis. As indicated, the exclusion area boundary (EAB), the low population zone (LPZ) and control room calculated doses are within the regulatory limits after AST implementation.

3.1.1.2 MSLB

The radiological consequences of the design basis MSLB accident were analyzed using the STARDOSE code for the control room dose and the inputs and assumptions discussed in Section 2.3.1.2. The offsite dose consequence results were obtained using the current FSAR postulated exposures adjusted for the new X/Q values and the dose conversion factors in Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," (Reference 21). Two activity release cases corresponding to the reactor coolant pre-accident spike and maximum equilibrium concentration allowed by Technical Specifications of 4 μCi/gm and 0.2 μCi/gm dose equivalent I-131 respectively were analyzed.

The MSLB accident control room dose presented in Table 9 corresponds to an unfiltered in-leakage rate of 300 cfm, reduced to 150 cfm, after 30 minutes. Four different cases of unfiltered in-leakage rates were evaluated to demonstrate the sensitivity of the analyses to the unfiltered in-leakage. The maximum dose occurs for high initial in-leakage, with reduced rate for sweep-out after the puff passes.

Table 9 presents the results of the MSLB accident radiological consequence analysis. As indicated, the control room, EAB, and LPZ calculated doses are within the regulatory limits after AST implementation.

3.1.1.3 FHA

The radiological consequences of the design basis fuel handling accident were analyzed using the STARDOSE code and the inputs and assumptions discussed in Section 2.3.1.3. The results of the FHA radiological consequence analysis for offsite and control room receptors are provided in Table 10. As indicated, both the

offsite (EAB and LPZ) and the control room occupant calculated doses are within the regulatory limits after AST implementation.

3.1.1.4 CRDA

The radiological consequences of the design basis CRDA were analyzed using the STARDOSE code and the inputs and assumptions discussed in Section 2.3.1.4. The results of the CRDA radiological consequence analysis for offsite and control room receptors are provided in Table 11. As indicated, both the offsite and control room calculated doses are within the regulatory limits after AST implementation.

3.1.2 Atmospheric Dispersion Factors

The X/Q values are summarized in Tables 12 and 13. Ground level release X/Q values for the control room are taken from the ARCON96 results and are itemized within Table 12. As shown in Figures 2 through 5, there are two remote intakes and one local intake for the control room. On receipt of reactor building exhaust high activity, low reactor water level, or high primary containment pressure signal, the one local intake will isolate. Ground level release X/Q values for the EAB and LPZ locations are taken from the PAVAN results and are itemized within Table 13.

3.1.3 Post-Accident Suppression Pool Water Chemistry Management

The re-evolution of elemental iodine from the suppression pool is strongly dependent on pool pH. The analysis assumed that the borated solution was injected within hours of the onset of a DBA LOCA (the results are not sensitive to the time of initiation). The modeling of the Columbia Generating Station containment cabling indicated the production of hydrochloric acid. However, the analysis demonstrated that the acid added from radiolysis of water [HNO_3] and radiolysis of cable [HCl] is not enough to bring the minimum pool pH during the 30 days post-LOCA below 8.0. As this result is well above 7.0, this satisfies the conditions for minimizing the re-evolution of elemental iodine. The suppression pool pH, as a function of time following the LOCA, is presented in Table 14.

3.2 Evaluation Conclusions

As shown in Tables 8 through 11, the plant accident radiological consequence analyses demonstrate that the post-accident offsite and control room doses can be maintained within regulatory limits following AST implementation. Furthermore, Energy Northwest has determined that continued compliance with NUREG-0737, Item II.B.2, will be maintained.

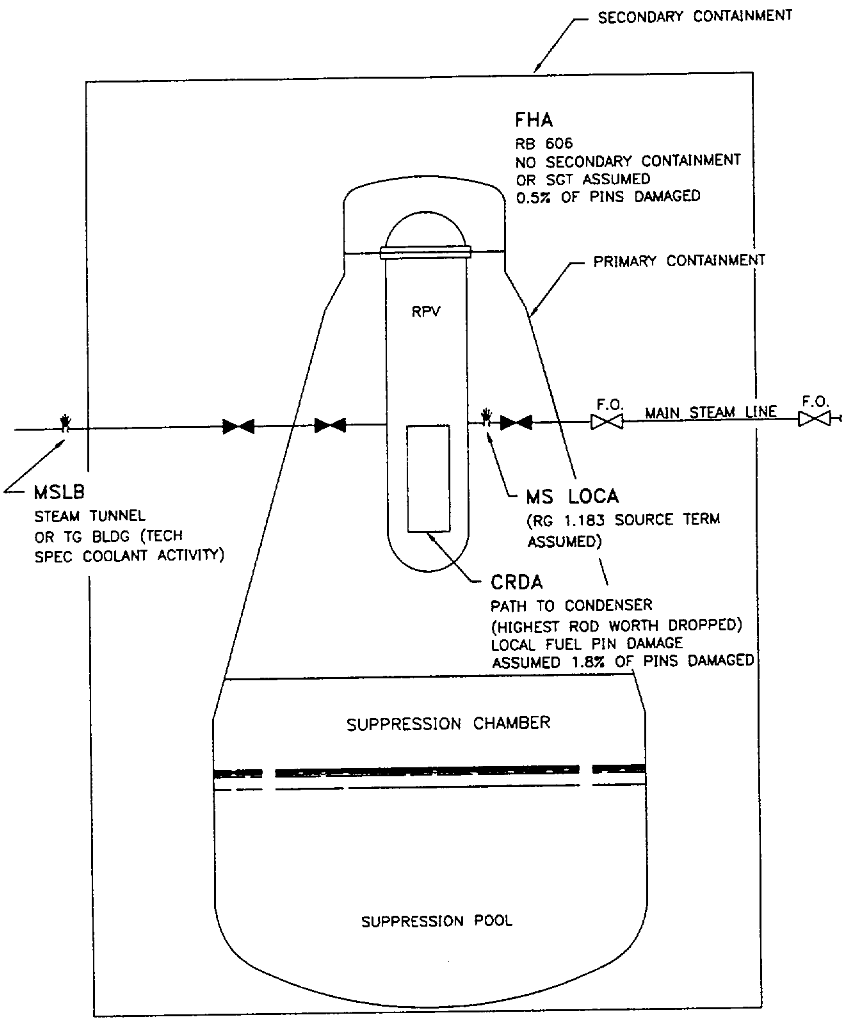
4.0. Summary

Implementation of the AST as the plant radiological consequence analyses licensing basis requires a license amendment pursuant to the requirements of 10 CFR 50.67. The analyses demonstrate the offsite and control room post-accident doses remain within the regulatory limits.

Implementation of the AST provides the basis for several changes to the licensing and design bases of Columbia Generating Station. In the consequence analyses, no credit was taken for secondary containment integrity until 10 minutes following the event. This increase in the time allowed for standby gas treatment to restore secondary containment to a negative pressure resolves the unreviewed safety question and subsequent justification for continued operation regarding the secondary containment draw-down time.

In addition, the allowable value for secondary containment bypass leakage was increased while maintaining personnel exposures below the established reference values in 10 CFR 50.67. In the dose consequence analyses for the control room occupants, the assumed unfiltered leakage was increased to a value that bounds the measured data. Further evaluation of the analyses performed in support of the AST implementation supported the conclusion that exposures to onsite and offsite receptors would remain below the values referenced in 10 CFR 50.67 without the operation of the main steam leakage control system.

Figures and Tables



ALTERNATE SOURCE TERM (AST) OBJECTIVES

MSLC DEACTIVATION
JCO RESOLUTION – SECONDARY CONTAINMENT
USQ RESOLUTION – CONTROL ROOM IN LEAKAGE

AST AFFECTED ACCIDENTS

DBA SOURCE TERM ACCIDENT (MS LOCA)
CONTROL ROD DROP ACCIDENT (CRDA)
MAIN STEAM LINE BREAK (MSLB)
FUEL HANDLING ACCIDENT (FHA)

Figure 1 Alternative Source Term Objectives and Affected Accidents

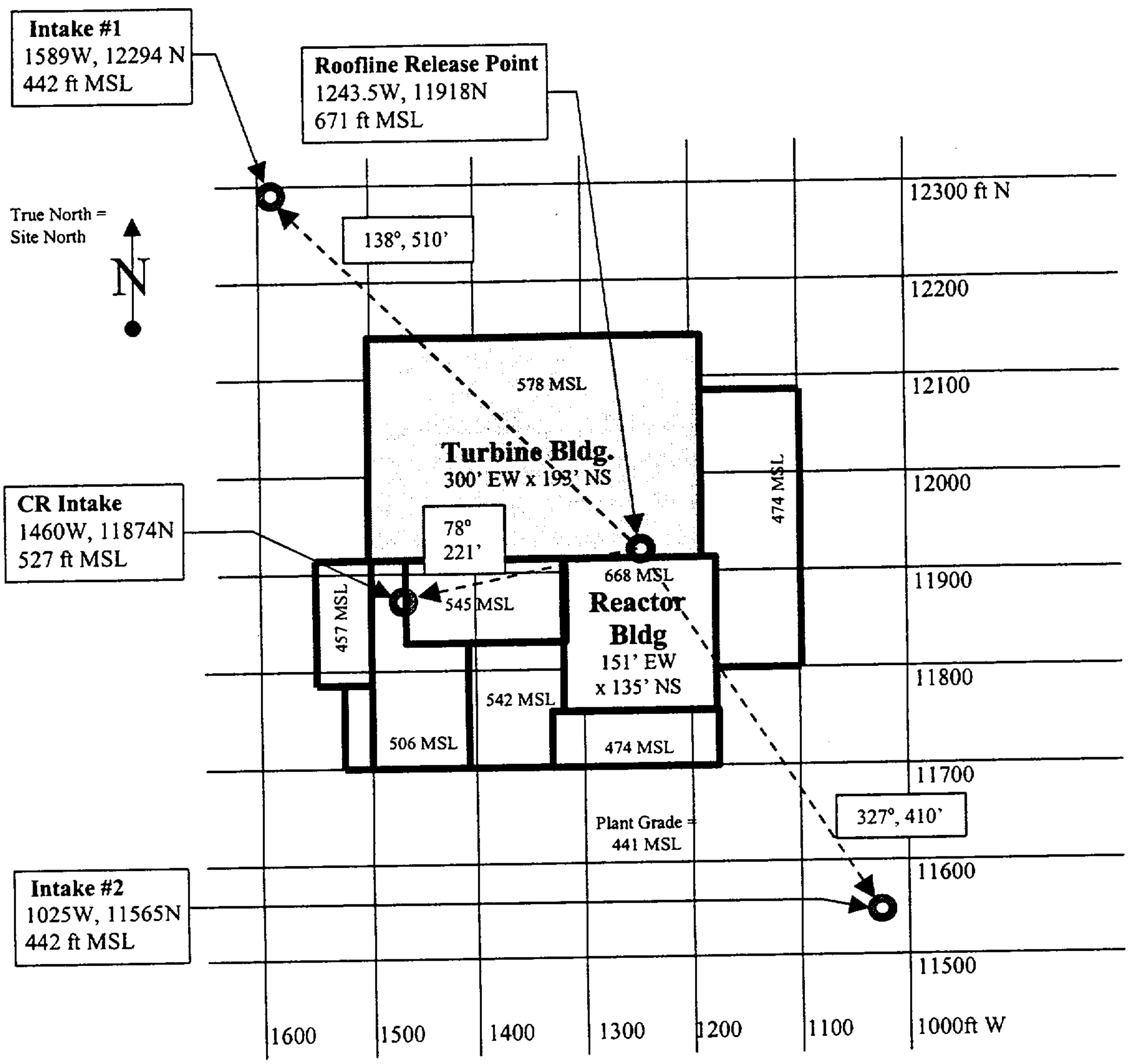


Figure 2 Schematic Layout of Point Release from the Reactor Building Roofline

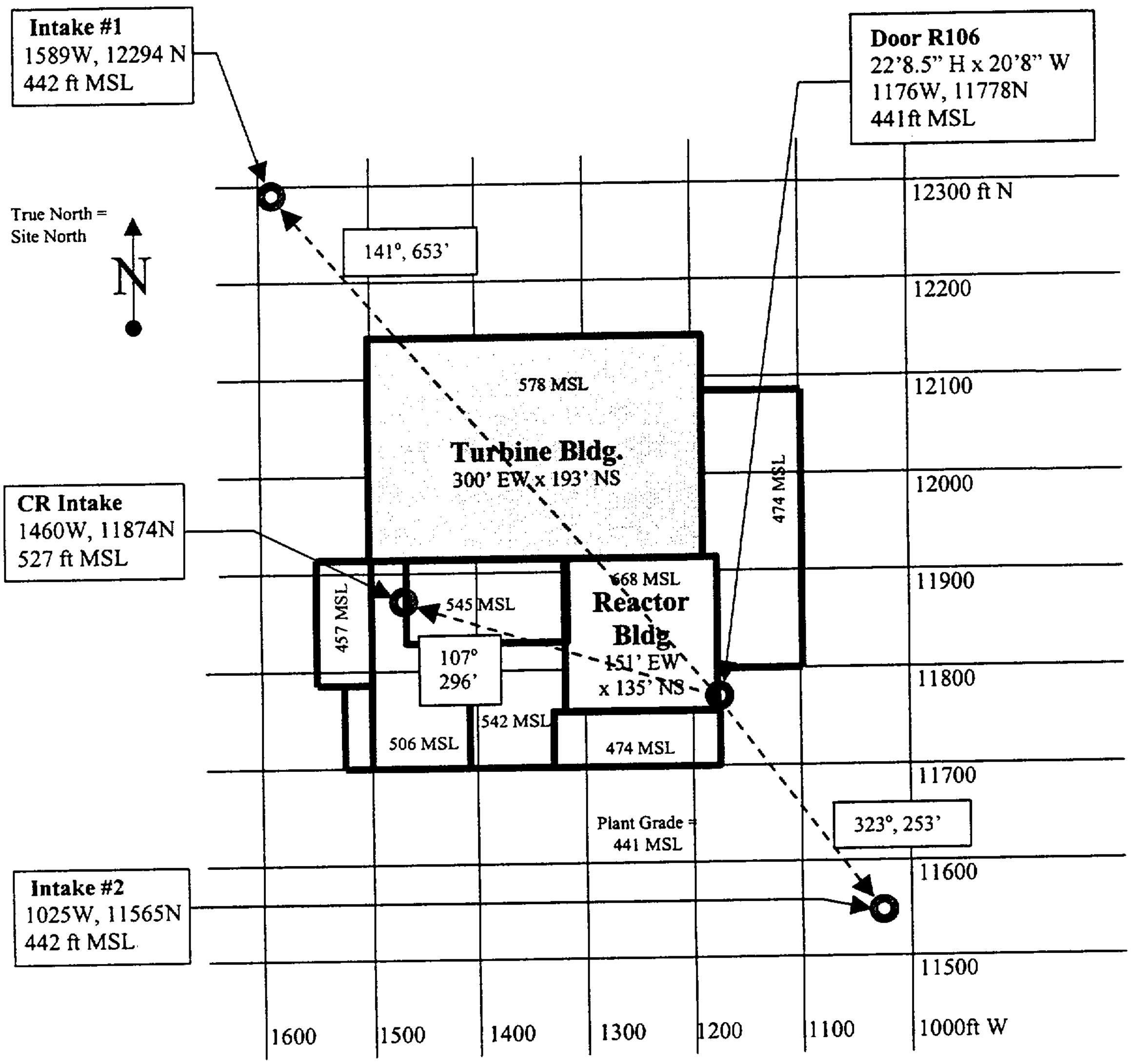


Figure 3 Schematic Layout of Diffuse Release from Reactor Building Door R106 (Railroad/Truck Bay Access)

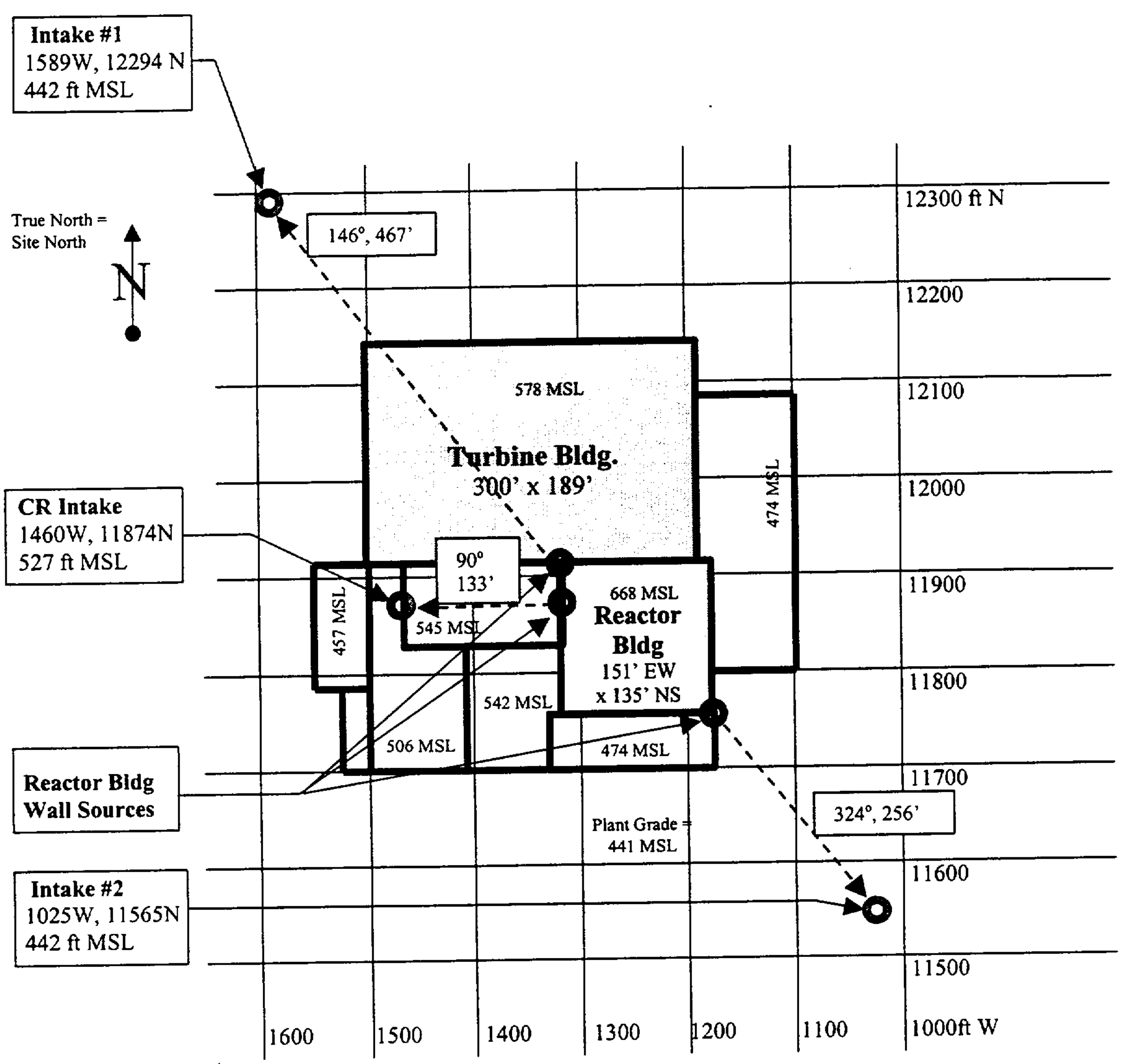


Figure 4 Schematic Layout of Diffuse Release from Reactor Building Walls

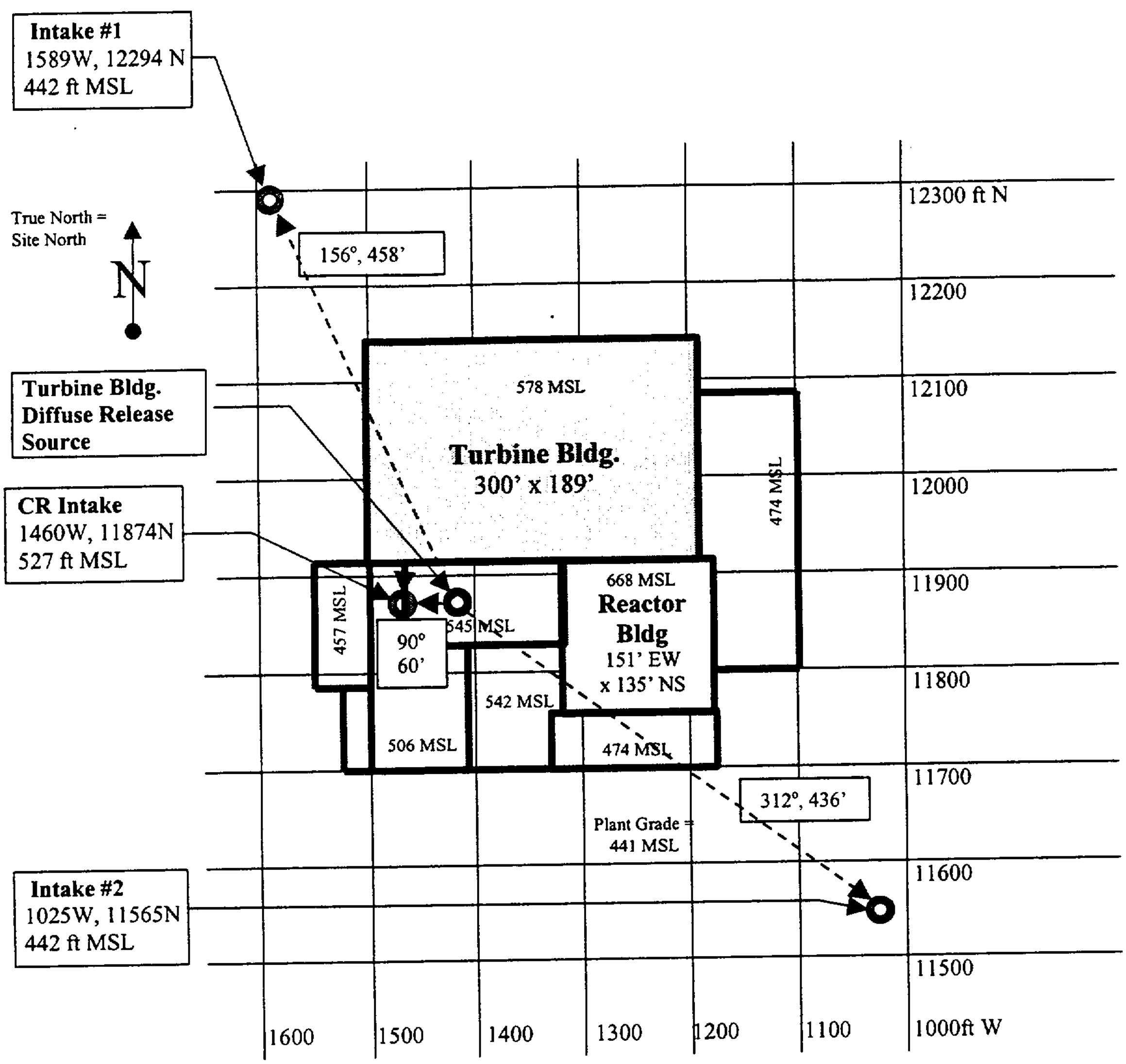


Figure 5 Schematic Layout of Diffuse Release from Turbine Building Exhaust System

Table 1: Personnel Dose Inputs	
Input/Assumption	**Value**
Onsite Breathing Rate	3.5E-04 m^3/sec
Offsite Breathing Rate	0-8 hours: 3.5E-04 m^3/sec* 8-24 hours: 1.8E-04 m^3/sec 1-30 days: 2.3E-04 m^3/sec
Control Room Occupancy Factors	0-1 day: 1.0 1-4 days: 0.6 4-30 days: 0.4

* A rate of 3.47E-4 m^3/sec was used for MSLB

Table 2: Key LOCA Analysis Inputs and Assumptions	
Release Inputs	
Input/Assumption	**Value**
Fission Products Core Inventory	Plant-Specific ORIGEN2
Fission Products Release Fractions	Regulatory Guide 1.183 Table 1 **Table 1** **BWR Core Inventory Fraction Released Into Containment** Group / Gap Release Phase / Early In-vessel Phase / Total Noble Gases 0.05 0.95 1.0 Halogens 0.05 0.25 0.3 Alkali Metals 0.05 0.20 0.25 Tellurium Metals 0.00 0.05 0.05 Ba, Sr 0.00 0.02 0.02 Noble Metals 0.00 0.0025 0.0025 Cerium Group 0.00 0.0005 0.0005 Lanthanides 0.00 0.0002 0.0002
Fission Products Form	Barium and strontium have release fractions of 2%. The nuclides include 58 of the 60 identified in NUREG/CR-4691, plus four additional noble gas isotopes from TID-14844, plus three other short-lived noble gas isotopes, plus Ba137m, for a total of 66.
Fission Products Timing	Reg.Guide 1.183 **Table 4** **LOCA Release Phases** Phase / PWRs Onset / PWRs Duration / BWRs Onset / BWRs Duration Gap Release 30 sec 0.5 hr 2 min 0.5 hr Early In-Vessel 0.5 hr 1.3 hr 0.5 hr 1.5 hr
Primary Containment Leak Rate	0.5%/day for 24 hours, 0.25%/day afterwards
Primary Containment Leakage, Secondary Containment Bypass (SCB)	0.04%/day for 24 hours, 0.02%/day afterwards
ESF Systems Leak Rate	2 gpm
Total MSIV leak rate	46 scfh for 24 hours, 23 scfh afterwards
SLC system sodium pentaborate Inventory	4000 lb_m

Table 3: Key LOCA Analysis Inputs and Assumptions	
Transport Inputs	
Input/Assumption	Value
Reactor Building Ground Release Location	Figures 2, 3 and 4
Turbine Building Ground MSIV Release Location	Figure 5
CREF system initiation (auto)	Before release occurs
One train of CREF secured (manual)	30 minutes
CREF Air Intake Flow Rate	1800 cfm (2 trains) 900 cfm (1 train)
Control Room Unfiltered In-leakage Rate	250 cfm (2 trains) 125 cfm (1 train)

Table 4: Key LOCA Analysis Inputs and Assumptions,		
Removal Inputs		
Input/Assumption	Value	
Aerosol DW Spray Removal Rates	Time	Rate
	0 – 0.25 hr	0
	0.25 – 2.44 hr	6.2/hr
	2.44 – 24 hr	0.62/hr
	24 – 720 hr	0
Main Steam Lines Deposition	Brockmann - Bixler Pipe Deposition Model	
SGT Flow Rate	5000 cfm	
SGT Filter Efficiency	99%, with 50 cfm bypass	
CREF Filter Iodine Efficiency	95% for the gaseous iodine species 99% for the particulates	

Table 5: Key MSLB Accident Analysis Inputs and Assumptions	
Input/Assumption	**Value**
Mass Release	130,000 lb_m
MSIV isolation time	6 seconds
Pre-Accident Spike Iodine Concentration	4 µCi/gm
Maximum Equilibrium Iodine Concentration	0.2 µCi/gm
Ground level release distance from release point to control room air intake	approximately 200 ft
Control Room Unfiltered Air Inleakage	300/150 cfm

Table 6: Key FHA Analysis Inputs and Assumptions	
Input/Assumption	**Value**
% Full Core Damaged	0.528 %
Radial Peaking Factor	2
Fuel Decay Period	24 hours
Suppression Pool Water Iodine Decontamination Factor	A filter efficiency of 0.998 is used to account for the water DF of 500 applied to the elemental iodine, while the organic iodine is not scrubbed. By applying the DF of 500 to the elemental iodine (which is assumed to make up 99.85% of the release from the gap) and applying a DF of one to the organic iodine (which is assumed to be the remainder of the iodine release from the gap), the expected iodine speciation of 57% elemental and 43% organic is obtained.
Release Period	2 hours
Reactor Building Ground Release Location	Figure 2
CREF system initiation (auto)	5 minutes
One train of CREF secured (manual)	30 minutes
CREF Outside Air Intake Flow Rate	1800 cfm (2 trains) 900 cfm (1 train)
Control Room Unfiltered In-leakage Rate	350 cfm (2 trains) 200 cfm (1 train)

Table 7: Key CRDA Analysis Inputs and Assumptions	
Input/Assumption	**Value**
Percentage of full core damaged	1.8%
Percent of damaged fuel with melt	0.77%
Radial Peaking Factor	1.50
Condenser Leak Rate	1%/day
Release Period	24 hours
Turbine Building Ground Release Location	Figure 5
Control Room Unfiltered In-leakage Rate	1800 cfm

Table 8: LOCA Radiological Consequence Analysis			
Location	**Duration**	**TEDE (rem)**	**Regulatory LIMIT TEDE (rem)**
Control room	30 day	4.39	5
EAB	Maximum, 2 hour	2.32	25
LPZ	30 day	2.97	25

Table 9: MSLB Accident Radiological Consequence Analysis				
LOCATION	Duration	4 µCi/gm dose equivalent I-131 TEDE (rem)	0.2 µCi/gm dose equivalent I-131 TEDE (rem)	Regulatory Limit TEDE (rem)
Control room	30-day integrated dose	0.28	0.014	5.0
EAB	Worst 2-hour integrated dose	0.71	0.035	25
LPZ	30-day integrated dose	0.20	0.01	25

Table 10: FHA Radiological Consequence Analysis			
Location	**Duration**	**TEDE (rem)**	**Regulatory Limit TEDE (rem)**
Control room	30 day	1.37	5
EAB	Maximum, 2 hour	1.01	6.3
LPZ	30 day	0.28	6.3

Table 11: CRDA Radiological Consequence Analysis			
Location	**Duration**	**TEDE (rem)**	**Regulatory Limit TEDE (rem)**
Control room	30 day	0.655	5
EAB	Maximum, 2 hour	0.022	6.3
LPZ	30 day	0.023	6.3

Table 12 Average control room χ/Q values for the different Release and Intake combinations using ARCON96						
Time Period	Filtered χ/Q (s/m³)			Unfiltered χ/Q (s/m³)		
	Turbine Building	**SCN Bypass**	**SGT Roofline**	**Turbine Building**	**SCN Bypass**	**SGT Roofline**
0 - 0.5 hrs	7.03E-04	4.13E-04	2.41E-04	3.17E-03	6.80E-04	5.79E-04
0.5 - 2 hrs	6.13E-04	4.03E-04	2.29E-04	3.17E-03	6.80E-04	5.79E-04
2 - 8 hrs	4.72E-04	1.70E-04	9.47E-05	2.40E-03	4.92E-04	2.86E-04
8 - 24 hrs	2.42E-04	8.21E-05	5.17E-05	1.19E-03	2.53E-04	1.79E-04
1 - 4 days	1.67E-04	5.77E-05	3.97E-05	7.91E-04	1.45E-04	1.25E-04
4-30 days	1.26E-04	4.87E-05	3.05E-05	5.87E-04	1.19E-04	9.15E-05

Table 13
χ/Q (s/m^3) values using PAVAN for the EAB and LPZ

Time Period	EAB χ/Q (s/m^3)	LPZ χ/Q (s/m^3)
0 - 2 hrs	1.8 E-4	
0 - 8 hrs		5.04 E-5
8 - 24 hrs		3.76 E-5
1 - 4 d		1.99 E-5
4 - 30 d		7.97 E-6

Table 14
Suppression Pool pH results

Time Period	pH
1h	8.0
2h	8.0
5h	8.6
12h	8.6
1d	8.5
3d	8.5
10d	8.4
20d	8.3
30d	8.3

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Table 1. Conformance with Regulatory Guide 1.183 Main Sections

RG Sec	RG Position	Columbia Analysis	Comments
3.1	The inventory of fission products in the reactor core and available for release to the containment should be based on the maximum full power operation of the core with, as a minimum, current licensed values for fuel enrichment, fuel burnup, and an assumed core power equal to the current licensed rated thermal power times the ECCS evaluation uncertainty. The period of irradiation should be of sufficient duration to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values. The core inventory should be determined using an appropriate isotope generation and depletion computer code such as ORIGEN 2 (Ref. 17) or ORIGEN-ARP (Ref. 18). Core inventory factors (Ci/MWt) provided in TID14844 and used in some analysis computer codes were derived for low burnup, low enrichment fuel and should not be used with higher burnup and higher enrichment fuels.	Conforms	ORIGEN 2-based. Long-lived isotopes adjusted for 24-month cycle. Power level used = 3556 MW(t) to account for two percent uncertainty (3486 x 1.02 = 3556).
3.1	For the DBA LOCA, all fuel assemblies in the core are assumed to be affected and the core average inventory should be used.	Conforms	
3.2	The core inventory release fractions, by radionuclide groups, for the gap release and early in-vessel damage phases for DBA LOCAs are listed in Table 1 for BWRs and Table 2 for PWRs. These fractions are applied to the equilibrium core inventory described in Regulatory Position 3.1. **Table 1** BWR Core Inventory Fraction **Released Into Containment** **Group** / **Gap Release Phase** / **Early In-Vessel Phase** / **Total** Noble Gases 0.05 0.95 1.0 Halogens 0.05 0.25 0.3 Alkali Metals 0.05 0.20 0.25 Tellurim Metals 0.00 0.05 0.05 Ba, Sr 0.00 0.02 0.02	Conforms	The fractions from Table 1 are used.

Table 1. Conformance with Regulatory Guide 1.183 Main Sections

RG Sec	RG Position	Columbia Analysis	Comments
	Noble Metals 0.00 0.0025 0.0025 Cerium Group 0.00 0.0005 0.0005 Lanthanides 0.00 0.0002 0.0002		
3.2	For non-LOCA events, the fractions of the core inventory assumed to be in the gap for the various radionuclides are given in Table 3. The release fractions from Table 3 are used in conjunction with the fission product inventory calculated with the maximum core radial peaking factor. Table 3 **Non-LOCA Fraction of Fission Product Inventory in Gap** **Group** **Fraction** I-131 0.08 Kr-85 0.10 Other Noble Gases 0.05 Other Halogens 0.05 Alkali Metals 0.12	Conforms	
3.3	Table 4 tabulates the onset and duration of each sequential release phase for DBA LOCAs at PWRs and BWRs. The specified onset is the time following the initiation of the accident (i.e., time = 0). The early in-vessel phase immediately follows the gap release phase. The activity released from the core during each release phase should be modeled as increasing in a linear fashion over the duration of the phase.	Conforms	The BWR durations from Table 4 are used. LOCA – modeled in a linear fashion. Non-LOCA -- instantaneous release.

Table 1. Conformance with Regulatory Guide 1.183 Main Sections

RG Sec	RG Position	Columbia Analysis	Comments
	Table 4 **LOCA Release Phases** PWRs: Phase / Onset / Duration — Gap Release: 30 sec / 0.5 hr; Early In-Vessel: 0.5 hr / 1.3 hr BWRs: Phase / Onset / Duration — Gap Release: 2 min / 0.5 hr; Early In-Vessel: 0.5 hr / 1.5 hr		
3.4	Table 5 lists the elements in each radionuclide group that should be considered in design basis analyses. **Table 5** **Radionuclide Groups** **Group** — **Elements** Noble Gases — Xe, Kr Halogens — I, Br Alkali Metals — Cs, Rb Tellurium Group — Te, Sb, Se, Ba, Sr Noble Metals — Ru, Rh, Pd, Mo, Tc, Co Lanthanides — La, Zr, Nd, Eu, Nb, Pm, Pr Sm, Y, Cm, Am Cerium — Ce, Pu, Np	Conforms	The regulatory guide is inconsistent between Tables 1 and 5. Barium and strontium have release fractions lower than the Te group, (see Item 3.2), and these fractions are used in lieu of the five percent release for the Te group. The nuclides used for Columbia are the 60 identified as being potentially important contributors to TEDE in NUREG/CR-4691 (MACCS User's Guide) [less the two cobalt isotopes which have a minor impact] plus four additional noble gas isotopes from TID-14844, plus three other short-lived noble gas isotopes, plus Ba137m for a total of 66.
3.5	Of the radioiodine released from the reactor coolant system (RCS) to the containment in a postulated accident, 95 percent of the iodine released should be assumed to be cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. This includes releases from the gap and the fuel pellets. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate	Conforms	See Appendix A, Section 2, for more information.

Table 1. Conformance with Regulatory Guide 1.183 Main Sections

RG Sec	RG Position	Columbia Analysis	Comments
	form. The same chemical form is assumed in releases from fuel pins in FHAs and from releases from the fuel pins through the RCS in DBAs other than FHAs or LOCAs. However, the transport of these iodine species following release from the fuel may affect these assumed fractions. The accident-specific appendices to this regulatory guide provide additional details.		
	The dose calculations should determine the TEDE. TEDE is the sum of the committed effective dose equivalent (CEDE) from inhalation and the deep dose equivalent (DDE) from external exposure. The calculation of these two components of the TEDE should consider all radionuclides, including progeny from the decay of parent radionuclides, that are significant with regard to dose consequences and the released radioactivity.	Conforms	TEDE calculated. Significant progeny included.
3.6	The amount of fuel damage caused by non-LOCA design basis events should be analyzed to determine, for the case resulting in the highest radioactivity release, the fraction of the fuel that reaches or exceeds the initiation temperature of fuel melt and the fraction of fuel elements for which the fuel clad is breached. Although the NRC staff has traditionally relied upon the departure from nucleate boiling ratio (DNBR) as a fuel damage criterion, licensees may propose other methods to the NRC staff, such as those based upon enthalpy deposition, for estimating fuel damage for the purpose of establishing radioactivity releases.	Conforms	Enthalpy deposition postulated for CRDA. Mechanical damage for FHA.
4.1.2	The exposure-to-CEDE factors for inhalation of radioactive material should be derived from the data provided in ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers" (Ref. 19). Table 2.1 of Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion" (Ref. 20), provides tables of conversion factors acceptable to the NRC staff. The factors in the column headed "effective" yield doses corresponding to the CEDE.	Conforms	Federal Guidance Report 11 dose conversion factors (DCFs) are used.
4.1.3	For the first 8 hours, the breathing rate of persons offsite should be assumed to be 3.5×10^{-4} cubic meters per second. From 8 to 24 hours following the accident, the breathing rate should be assumed to be 1.8×10^{-4} cubic meters per second. After that and until the end of the accident, the rate should be assumed to be 2.3×10^{-4} cubic meters per second.	Conforms	
4.1.4	The DDE should be calculated assuming submergence in semi-infinite cloud assumptions	Conforms	External exposure DCFs taken from

Table 1. Conformance with Regulatory Guide 1.183 Main Sections

RG Sec	RG Position	Columbia Analysis	Comments
	with appropriate credit for attenuation by body tissue. The DDE is nominally equivalent to the effective dose equivalent (EDE) from external exposure if the whole body is irradiated uniformly. Since this is a reasonable assumption for submergence exposure situations, EDE may be used in lieu of DDE in determining the contribution of external dose to the TEDE. Table III.1 of Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water, and Soil" (Ref. 21), provides external EDE conversion factors acceptable to the NRC staff. The factors in the column headed "effective" yield doses corresponding to the EDE.		TACT5 (NUREG/CR-4691) except where Federal Guidance Report 11 provided alternative values. These values compared to Federal Guidance Report 12-based values from NUREG/CR-6604. The combination of NUREG/CR-4691 and FGR 11 conservative compared to FRG-12.
4.1.5	The TEDE should be determined for the most limiting person at the EAB. The maximum EAB TEDE for any two-hour period following the start of the radioactivity release should be determined and used in determining compliance with the dose criteria in 10 CFR 50.67. The maximum two-hour TEDE should be determined by calculating the postulated dose for a series of small time increments and performing a "sliding" sum over the increments for successive two-hour periods. The maximum TEDE obtained is submitted. The time increments should appropriately reflect the progression of the accident to capture the peak dose interval between the start of the event and the end of radioactivity release (see also Table 6).	Conforms	
4.1.6	TEDE should be determined for the most limiting receptor at the outer boundary of the low population zone (LPZ) and should be used in determining compliance with the dose criteria in 10 CFR 50.67.	Conforms	
4.1.7	No correction should be made for depletion of the effluent plume by deposition on the ground.	Conforms	
4.2.1	The TEDE analysis should consider all sources of radiation that will cause exposure to control room personnel. The applicable sources will vary from facility to facility, but typically will include: • Contamination of the control room atmosphere by the intake or infiltration of the radioactive material contained in the radioactive plume released from the facility, • Contamination of the control room atmosphere by the intake or infiltration of airborne radioactive material from areas and structures adjacent to the control room envelope,	Conforms	First two items included in combination of filtered make-up and conservative overstatement of measured unfiltered inleakage. Last three items shown to be negligible.

Table 1. Conformance with Regulatory Guide 1.183 Main Sections

RG Sec	RG Position	Columbia Analysis	Comments
	• Radiation shine from the external radioactive plume released from the facility, • Radiation shine from radioactive material in the reactor containment, • Radiation shine from radioactive material in systems and components inside or external to the control room envelope, e.g., radioactive material buildup in recirculation filters.		
4.2.2	The radioactive material releases and radiation levels used in the control room dose analysis should be determined using the same source term, transport, and release assumptions used for determining the EAB and the LPZ TEDE values, unless these assumptions would result in non-conservative results for the control room.	Conforms	The source term, transport, and release methodology is the same for both the control room and offsite locations.
4.2.3	The models used to transport radioactive material into and through the control room, and the shielding models used to determine radiation dose rates from external sources, should be structured to provide suitably conservative estimates of the exposure to control room personnel.	Conforms	
4.2.4	Credit for engineered safety features that mitigate airborne radioactive material within the control room may be assumed. Such features may include control room isolation or pressurization, or intake or recirculation filtration. Refer to Section 6.5.1, "ESF Atmospheric Cleanup System," of the SRP (Ref. 3) and Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Post-accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants" (Ref. 25), for guidance.	Conforms	Pressurization and intake filtration are credited.
4.2.5	Credit should generally not be taken for the use of personal protective equipment or prophylactic drugs. Deviations may be considered on a case-by-case basis.	Conforms	Such credits are not taken.
4.2.6	The dose receptor for these analyses is the hypothetical maximum exposed individual who is present in the control room for 100% of the time during the first 24 hours after the event, 60% of the time between 1 and 4 days, and 40% of the time from 4 days to 30 days. For the duration of the event, the breathing rate of this individual should be assumed to be 3.5×10^{-4} cubic meters per second.	Conforms	
4.2.7	Control room doses should be calculated using dose conversion factors identified in	Conforms	The equation given is utilized for

Table 1. Conformance with Regulatory Guide 1.183 Main Sections

RG Sec	RG Position	Columbia Analysis	Comments
	Regulatory Position 4.1 above for use in offsite dose analyses. The DDE from photons may be corrected for the difference between finite cloud geometry in the control room and the semi-infinite cloud assumption used in calculating the dose conversion factors. The following expression may be used to correct the semi-infinite cloud dose, DDE_{∞}, to a finite cloud dose, DDE_{finite}, where the control room is modeled as a hemisphere that has a volume, V, in cubic feet, equivalent to that of the control room (Ref. 22). $$DDE_{finite} = \frac{DDE_{\infty} V^{0.338}}{1173}$$		finite cloud correction when calculating external doses due to the airborne activity inside the control room.
4.3	The guidance provided in Regulatory Positions 4.1 and 4.2 should be used, as applicable, in re-assessing the radiological analyses identified in Regulatory Position 1.3.1, such as those in NUREG-0737 (Ref. 2). Design envelope source terms provided in NUREG-0737 should be updated for consistency with the AST. In general, radiation exposures to plant personnel identified in Regulatory Position 1.3.1 should be expressed in terms of TEDE. Integrated radiation exposure of plant equipment should be determined using the guidance of Appendix I of this guide.	Conforms	A qualitative assessment of the regulatory positions on source term indicate that with no new operator actions required in areas such as ECCS pump rooms, doses would be lower than currently reported.
5.1.1	The evaluations required by 10 CFR 50.67 are re-analyses of the design basis safety analyses and evaluations required by 10 CFR 50.34; they are considered to be a significant input to the evaluations required by 10 CFR 50.92 or 10 CFR 50.59. These analyses should be prepared, reviewed, and maintained in accordance with quality assurance programs that comply with Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50.	Conforms	
5.1.2	Credit may be taken for accident mitigation features that are classified as safety-related, are required to be operable by technical specifications, are powered by emergency power sources, and are either automatically actuated or, in limited cases, have actuation requirements explicitly addressed in emergency operating procedures. The single active component failure that results in the most limiting radiological consequences should be assumed. Assumptions regarding the occurrence and timing of a loss of offsite power should be selected with the objective of maximizing the postulated radiological consequences.	Conforms	RHR drywell sprays and SLC systems are required by technical specifications, are powered by emergency power, and have actuation requirements explicitly addressed in emergency operating procedures.

Table 1. Conformance with Regulatory Guide 1.183 Main Sections

RG Sec	RG Position	Columbia Analysis	Comments
5.1.3	The numeric values that are chosen as inputs to the analyses required by 10 CFR 50.67 should be selected with the objective of determining a conservative postulated dose. In some instances, a particular parameter may be conservative in one portion of an analysis but be nonconservative in another portion of the same analysis.	Conforms	
5.1.4	Licensees should ensure that analysis assumptions and methods are compatible with the AST and the TEDE criteria.	Conforms	
5.3	Atmospheric dispersion values (X/Q) for the EAB, the LPZ, and the control room that were approved by the staff during initial facility licensing or in subsequent licensing proceedings may be used in performing the radiological analyses identified by this guide. Methodologies that have been used for determining X/Q values are documented in Regulatory Guides 1.3 and 1.4, Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," and the paper, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19" (Refs. 6, 7, 22, and 28). The methodology of the NRC computer code ARCON96 (Ref 26) is generally acceptable to the NRC staff for use in determining control room X/Q values.	Conforms	Dispersion values included in submittal. Determination consistent with Reg Guide 1.145 for offsite. ARCON96 used to determine control room values. The MSLB X/Q is calculated for both a puff and a trapped release case. The puff release X/Q credits buoyancy. The trapped release X/Q is calculated with ARCON96. See item 4.3 MSLB section below.

Table 2. Conformance with Regulatory Guide 1.183 Appendix A (LOCA)

App Sec	RG Position	Columbia Analysis	Comments
1	Acceptable assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide.	Conforms	See Main Sections 3.1 to 3.4 for more information.
2	If the sump or suppression pool pH is controlled at values of 7 or greater, the chemical form of radioiodine released to the containment should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. Iodine species, including those from iodine re-evolution, for sump or suppression pool pH values	Conforms	The stated distributions of iodine chemical forms are used. An evaluation has been done to demonstrate pH > 7.

Table 2. Conformance with Regulatory Guide 1.183 Appendix A (LOCA)

App Sec	RG Position	Columbia Analysis	Comments
	less than 7 will be evaluated on a case-by-case basis. Evaluations of pH should consider the effect of acids and bases created during the LOCA event, e.g., radiolysis products. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form.		
3.1	The radioactivity released from the fuel should be assumed to mix instantaneously and homogeneously throughout the free air volume of the primary containment in PWRs or the drywell in BWRs as it is released. This distribution should be adjusted if there are internal compartments that have limited ventilation exchange. The suppression pool free air volume may be included provided there is a mechanism to ensure mixing between the drywell to the wetwell. The release into the containment or drywell should be assumed to terminate at the end of the early in-vessel phase.	Conforms	Flow from the drywell to the wetwell has been ignored prior to the assumed core quench at two hours. Ignoring this flow is conservative since by remaining in the drywell, it contributes to MSIV leakage. For several minutes after the core quench, flow from the drywell to the wetwell would be expected to occur, and approximately half of the drywell contents would be expected to be purged into the wetwell at this time. Beyond the end of this purge flow, the drywell and wetwell gas spaces are assumed to be well-mixed.
3.2	Reduction in airborne radioactivity in the containment by natural deposition within the containment may be credited. Acceptable models for removal of iodine and aerosols are described in Chapter 6.5.2, "Containment Spray as a Fission Product Cleanup System," of the Standard Review Plan (SRP), NUREG-0800 (Ref. A-1) and in NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments" (Ref. A-2). The latter model is incorporated into the analysis code RADTRAD (Ref. A-3).	Conforms	No credit taken for natural deposition.
3.3	Reduction in airborne radioactivity in the containment by containment spray systems that have been designed and are maintained in accordance with Chapter 6.5.2 of the SRP (Ref. A-1) may be credited. Acceptable models for the removal of iodine and aerosols are described in Chapter 6.5.2 of the SRP and NUREG/CR-5966, "A Simplified Model of Aerosol Removal by Containment Sprays"1 (Ref. A-4). This simplified model is	Conforms	SRP model used. Drywell congestion explicitly addressed by reduced spray flow and fall height credit. Elemental iodine assumed to be removed at the same rate as

Table 2. Conformance with Regulatory Guide 1.183 Appendix A (LOCA)

App Sec	RG Position	Columbia Analysis	Comments
	incorporated into the analysis code RADTRAD (Refs. A-1 to A-3).		particulate.
3.3	The evaluation of the containment sprays should address areas within the primary containment that are not covered by the spray drops. The mixing rate attributed to natural convection between sprayed and unsprayed regions of the containment building, provided that adequate flow exists between these regions, is assumed to be two turnovers of the unsprayed regions per hour, unless other rates are justified. The containment building atmosphere may be considered a single, well-mixed volume if the spray covers at least 90% of the volume and if adequate mixing of unsprayed compartments can be shown.	Conforms	Drywell assumed to be well-mixed based on the fact that the drywell is sufficiently small and the spray flowrate is sufficiently large (i.e., the ratio of spray flow to volume sprayed is 20-40 times larger for the Columbia drywell than for a typical sprayed region of a PWR) that mixing by momentum exchange alone (between the droplets and the atmosphere) will keep the drywell well-mixed; i.e., natural convection will play no noticeable role.
3.3	The SRP sets forth a maximum decontamination factor (DF) for elemental iodine based on the maximum iodine activity in the primary containment atmosphere when the sprays actuate, divided by the activity of iodine remaining at some time after decontamination. The SRP also states that the particulate iodine removal rate should be reduced by a factor of 10 when a DF of 50 is reached. The reduction in the removal rate is not required if the removal rate is based on the calculated time-dependent airborne aerosol mass. There is no specified maximum DF for aerosol removal by sprays. The maximum activity to be used in determining the DF is defined as the iodine activity in the columns labeled "Total" in Tables 1 and 2 of this guide multiplied by 0.05 for elemental iodine and by 0.95 for particulate iodine (i.e., aerosol treated as particulate in SRP methodology).	Conforms	The SRP spray lambda is calculated per the SRP method. A reduction of 10 is taken when 98% of the particulate has been removed.
3.4	Reduction in airborne radioactivity in the containment by in-containment recirculation filter systems may be credited if these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. A-5 and A-6). The filter media loading caused by the increased aerosol release associated with the revised source term should be addressed.	Not applicable	
3.5	Reduction in airborne radioactivity in the containment by suppression pool scrubbing in BWRs should generally not be credited. However, the staff may consider such reduction on an individual case basis. The evaluation should consider the relative timing of the	Conforms	Pool scrubbing not credited.

Table 2. Conformance with Regulatory Guide 1.183 Appendix A (LOCA)

App Sec	RG Position	Columbia Analysis	Comments
	blowdown and the fission product release from the fuel, the force driving the release through the pool, and the potential for any bypass of the suppression pool (Ref. 7). Analyses should consider iodine re-evolution if the suppression pool liquid pH is not maintained greater than 7.		
3.6	Reduction in airborne radioactivity in the containment by retention in ice condensers, or other engineering safety features not addressed above, should be evaluated on an individual case basis. See Section 6.5.4 of the SRP (Ref. A-1).	Not applicable	
3.7	The primary containment (i.e., drywell for Mark I and II containment designs) should be assumed to leak at the peak pressure technical specification leak rate for the first 24 hours. For PWRs, the leak rate may be reduced after the first 24 hours to 50% of the technical specification leak rate. For BWRs, leakage may be reduced after the first 24 hours, if supported by plant configuration and analyses, to a value not less than 50% of the technical specification leak rate. Leakage from subatmospheric containments is assumed to terminate when the containment is brought to and maintained at a subatmospheric condition as defined by technical specifications.	Conforms.	Leakage reduced after 24 hours based upon reduced containment pressure. Primary containment pressure not brought subatmospheric.
3.7	For BWRs with Mark III containments, the leakage from the drywell into the primary containment should be based on the steaming rate of the heated reactor core, with no credit for core debris relocation. This leakage should be assumed during the two-hour period between the initial blowdown and termination of the fuel radioactivity release (gap and early in-vessel release phases). After two hours, the radioactivity is assumed to be uniformly distributed throughout the drywell and the primary containment.	Not applicable	
3.8	If the primary containment is routinely purged during power operations, releases via the purge system prior to containment isolation should be analyzed and the resulting doses summed with the postulated doses from other release paths. The purge release evaluation should assume that 100% of the radionuclide inventory in the reactor coolant system liquid is released to the containment at the initiation of the LOCA. This inventory should be based on the technical specification reactor coolant system equilibrium activity. Iodine spikes need not be considered. If the purge system is not isolated before the onset of the gap release phase, the release fractions associated with the gap release and early in-vessel phases should be considered as applicable.	Not applicable	

Table 2. Conformance with Regulatory Guide 1.183 Appendix A (LOCA)

App Sec	RG Position	Columbia Analysis	Comments
4.1	Leakage from the primary containment should be considered to be collected, processed by engineered safety feature (ESF) filters, if any, and released to the environment via the secondary containment exhaust system during periods in which the secondary containment has a negative pressure as defined in technical specifications. Credit for an elevated release should be assumed only if the point of physical release is more than two and one-half times the height of any adjacent structure.	Conforms	No elevated release point at Columbia and no credit taken for elevated release.
4.2	Leakage from the primary containment is assumed to be released directly to the environment as a ground-level release during any period in which the secondary containment does not have a negative pressure as defined in technical specifications.	Conforms	Assumed ground-level release directly to environment, unfiltered until secondary containment reaches technical specification pressure. Then secondary containment bypass leakage release and filtered release through standby gas treatment, both as ground level release.
4.3	The effect of high wind speeds 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% 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% or 95% of the total numbers of hours in the data set, whichever is conservative for the intended use (e.g., if high temperatures are limiting, use those exceeded only 5%).	Conforms	Met data and dispersion values are part of this submittal.
4.4	Credit for dilution in the secondary containment may be allowed when adequate means to cause mixing can be demonstrated. Otherwise, the leakage from the primary containment should be assumed to be transported directly to exhaust systems without mixing. Credit for mixing, if found to be appropriate, should generally be limited to 50%. This evaluation should consider the magnitude of the containment leakage in relation to contiguous building volume or exhaust rate, the location of exhaust plenums relative to projected release locations, the recirculation ventilation systems, and internal walls and floors that impede stream flow between the release and the exhaust.	Conforms	40% of the reactor building volume is credited for dilution.
4.5	Primary containment leakage that bypasses the secondary containment should be	Conforms	Bypass leakage rates included in

Table 2. Conformance with Regulatory Guide 1.183 Appendix A (LOCA)

App Sec	RG Position	Columbia Analysis	Comments
	evaluated at the bypass leak rate incorporated in the technical specifications. If the bypass leakage is through water, e.g., via a filled piping run that is maintained full, credit for retention of iodine and aerosols may be considered on a case-by-case basis. Similarly, deposition of aerosol radioactivity in gas-filled lines may be considered on a case-by-case basis.		submittal.
4.6	Reduction in the amount of radioactive material released from the secondary containment because of ESF filter systems may be taken into account provided that these systems meet the guidance of Regulatory Guide 1.52 (Ref. A-5) and Generic Letter 99-02 (Ref. A-6).	Conforms	
5.1	With the exception of noble gases, all the fission products released from the fuel to the containment (as defined in Tables 1 and 2 of this guide) should be assumed to instantaneously and homogeneously mix in the primary containment sump water (in PWRs) or suppression pool (in BWRs) at the time of release from the core. In lieu of this deterministic approach, suitably conservative mechanistic models for the transport of airborne activity in containment to the sump water may be used. Note that many of the parameters that make spray and deposition models conservative with regard to containment airborne leakage are nonconservative with regard to the buildup of sump activity.	Conforms	Fission products mixed into suppression pool during release.
5.2	The leakage should be taken as two times the sum of the simultaneous leakage from all components in the ESF recirculation systems above which the technical specifications, or licensee commitments to item III.D.1.1 of NUREG-0737 (Ref. A-8), would require declaring such systems inoperable. The leakage should be assumed to start at the earliest time the recirculation flow occurs in these systems and end at the latest time the releases from these systems are terminated. Consideration should also be given to design leakage through valves isolating ESF recirculation systems from tanks vented to atmosphere, e.g., emergency core cooling system (ECCS) pump miniflow return to the refueling water storage tank.	Conforms	ESF leakage is assumed to begin at the time drywell sprays are started. ESF leakage to the CST has been evaluated and shown to have a negligible contribution to dose. Surveillance for leakage is a plant program and analysis used 2 gpm, twice the 1 gpm assumed ESF leakage.
5.3	With the exception of iodine, all radioactive materials in the recirculating liquid should be assumed to be retained in the liquid phase.	Conforms	
5.4	If the temperature of the leakage exceeds 212°F, the fraction of total iodine in the liquid	Not applicable	

Table 2. Conformance with Regulatory Guide 1.183 Appendix A (LOCA)

App Sec	RG Position	Columbia Analysis	Comments
	that becomes airborne should be assumed equal to the fraction of the leakage that flashes to vapor. This flash fraction, FF, should be determined using a constant enthalpy, h, process, based on the maximum time-dependent temperature of the sump water circulating outside the containment: $$FF = \frac{h_{f_1} - h_{f_2}}{h_{fg}}$$ Where: h_{f1} is the enthalpy of liquid at system design temperature and pressure; h_{f2} is the enthalpy of liquid at saturation conditions (14.7 psia, 212°F); and h_{fg} is the heat of vaporization at 212°F.		
5.5	If the temperature of the leakage is less than 212°F or the calculated flash fraction is less than 10%, the amount of iodine that becomes airborne should be assumed to be 10% of the total iodine activity in the leaked fluid, unless a smaller amount can be justified based on the actual sump pH history and area ventilation rates.	Conforms	A release fraction of 10% is assumed.
5.6	The radioiodine that is postulated to be available for release to the environment is assumed to be 97% elemental and 3% organic. Reduction in release activity by dilution or holdup within buildings, or by ESF ventilation filtration systems, may be credited where applicable. Filter systems used in these applications should be evaluated against the guidance of Regulatory Guide 1.52 (Ref. A-5) and Generic Letter 99-02 (Ref. A-6).	Conforms	Credit is taken for holdup and dilution of ESF leakage in reactor building and for release through SGTS filters in the same way as containment leakage. Filter systems comply with Reg Guide 1.52 and GL 99-02.
6.1	For the purpose of this analysis, the activity available for release via MSIV leakage should be assumed to be that activity determined to be in the drywell for evaluating containment leakage (see Regulatory Position 3). No credit should be assumed for activity reduction by the steam separators or by iodine partitioning in the reactor vessel.	Conforms	
6.2	All the MSIVs should be assumed to leak at the maximum leak rate above which the technical specifications would require declaring the MSIVs inoperable. The leakage should be assumed to continue for the duration of the accident. Postulated leakage may be reduced after the first 24 hours, if supported by site-specific analyses, to a value not less than 50% of the maximum leak rate.	Conforms	

Table 2. Conformance with Regulatory Guide 1.183 Appendix A (LOCA)

App Sec	RG Position	Columbia Analysis	Comments
6.3	Reduction of the amount of released radioactivity by deposition and plateout on steam system piping upstream of the outboard MSIVs may be credited, but the amount of reduction in concentration allowed will be evaluated on an individual case basis. Generally, the model should be based on the assumption of well-mixed volumes, but other models such as slug flow may be used if justified.	Conforms	Conservative plug-flow treatment from RADTRAD employed. This result compared with AEB-98-03 in which well-mixed conditions are assumed. RADTRAD model more limiting.
6.4	In the absence of collection and treatment of releases by ESFs such as the MSIV leakage control system, or as described in paragraph 6.5 below, the MSIV leakage should be assumed to be released to the environment as an unprocessed, ground- level release. Holdup and dilution in the turbine building should not be assumed.	Conforms	MSIV leakage unprocessed, ground level release.
6.5	A reduction in MSIV releases that is due to holdup and deposition in main steam piping downstream of the MSIVs and in the main condenser, including the treatment of air ejector effluent by offgas systems, may be credited if the components and piping systems used in the release path are capable of performing their safety function during and following a safe shutdown earthquake (SSE). The amount of reduction allowed will be evaluated on an individual case basis. References A-9 and A-10 provide guidance on acceptable models.	Conforms	No credit taken for qualified steam lines beyond outboard MSIVs.
7.0	The radiological consequences from post-LOCA primary containment purging as a combustible gas or pressure control measure should be analyzed. If the installed containment purging capabilities are maintained for purposes of severe accident management and are not credited in any design basis analysis, radiological consequences need not be evaluated. If the primary containment purging is required within 30 days of the LOCA, the results of this analysis should be combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA. Reduction in the amount of radioactive material released via ESF filter systems may be taken into account provided that these systems meet the guidance in Regulatory Guide 1.52 (Ref. A-5) and Generic Letter 99-02 (Ref. A-6).	Conforms	No purge assumed.

Table 3. Conformance with Regulatory Guide 1.183 Appendix B (FHA)

App Sec	RG Position	Columbia Analysis	Comments
1	Acceptable assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide.	Conforms.	.
1.1	The number of fuel rods damaged during the accident should be based on a conservative analysis that considers the most limiting case. This analysis should consider parameters such as the weight of the dropped heavy load or the weight of a dropped fuel assembly (plus any attached handling grapples), the height of the drop, and the compression, torsion, and shear stresses on the irradiated fuel rods. Damage to adjacent fuel assemblies, if applicable (e.g., events over the reactor vessel), should be considered.	Conforms	Very conservative estimate of failed pins used.
1.2	The fission product release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached. All the gap activity in the damaged rods is assumed to be instantaneously released. Radionuclides that should be considered include xenons, kryptons, halogens, cesiums, and rubidiums.	Conforms	Cesium and rubidium not included because DF assumed to be infinite (see response to Section 3 below).
1.3	The chemical form of radioiodine released from the fuel to the spent fuel pool should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. The CsI released from the fuel is assumed to completely dissociate in the pool water. Because of the low pH of the pool water, the iodine re-evolves as elemental iodine. This is assumed to occur instantaneously. The NRC staff will consider, on a case-by-case basis, justifiable mechanistic treatment of the iodine release from the pool.	Conforms	All iodine added to pool assumed to dissociate.
2	If the depth of water above the damaged fuel is 23 feet or greater, the decontamination factors for the elemental and organic species are 500 and 1, respectively, giving an overall effective decontamination factor of 200 (i.e., 99.5% of the total iodine released from the damaged rods is retained by the water). This difference in decontamination factors for elemental (99.85%) and organic iodine (0.15%) species results in the iodine above the water being composed of 57% elemental and 43% organic species. If the depth of water is not 23 feet, the decontamination factor will have to be determined on a case-by-case method (Ref. B-1).	Conforms	DF of 500 applied to elemental iodine. DF of 1 applied to organic iodine. This results in speciation after decontamination of 57% elemental and 43% organic.
3	The retention of noble gases in the water in the fuel pool or reactor cavity is negligible (i.e., decontamination factor of 1). Particulate radionuclides are assumed to be retained by the water in the fuel pool or reactor cavity (i.e., infinite decontamination factor).	Conforms	
4.1	The radioactive material that escapes from the fuel pool to the fuel building is assumed to be released to the environment over a 2-hour time period.	Conforms	

Table 3. Conformance with Regulatory Guide 1.183 Appendix B (FHA)

App Sec	RG Position	Columbia Analysis	Comments
4.2	A reduction in the amount of radioactive material released from the fuel pool by engineered safety feature (ESF) filter systems may be taken into account provided these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. B-2, B-3). Delays in radiation detection, actuation of the ESF filtration system, or diversion of ventilation flow to the ESF filtration system(21) should be determined and accounted for in the radioactivity release analyses.	Conforms	No credit being taken for filtration from reactor building.
4.3	The radioactivity release from the fuel pool should be assumed to be drawn into the ESF filtration system without mixing or dilution in the fuel building. If mixing can be demonstrated, credit for mixing and dilution may be considered on a case-by-case basis. This evaluation should consider the magnitude of the building volume and exhaust rate, the potential for bypass to the environment, the location of exhaust plenums relative to the surface of the pool, recirculation ventilation systems, and internal walls and floors that impede stream flow between the surface of the pool and the exhaust plenums.	Conforms	Two-hour release to the environment assumed per Section 4.1.
5.1	If the containment is isolated during fuel handling operations, no radiological consequences need to be analyzed.	Not applicable	Containment not isolated.
5.2	If the containment is open during fuel handling operations, but designed to automatically isolate in the event of a fuel handling accident, the release duration should be based on delays in radiation detection and completion of containment isolation. If it can be shown that containment isolation occurs before radioactivity is released to the environment, no radiological consequences need to be analyzed.	Not applicable	Containment not isolated.
5.3	If the containment is open during fuel handling operations (e.g., personnel air lock or equipment hatch is open), the radioactive material that escapes from the reactor cavity pool to the containment is released to the environment over a 2-hour time period.	Conforms	
5.4	A reduction in the amount of radioactive material released from the containment by ESF filter systems may be taken into account provided that these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. B-2 and B-3). Delays in radiation detection, actuation of the ESF filtration system, or diversion of ventilation flow to the ESF filtration system should be determined and accounted for in the radioactivity release analyses.	Conforms	No credit being taken for filtration of release from reactor building.

Table 3. Conformance with Regulatory Guide 1.183 Appendix B (FHA)

App Sec	RG Position	Columbia Analysis	Comments
5.5	Credit for dilution or mixing of the activity released from the reactor cavity by natural or forced convection inside the containment may be considered on a case-by-case basis. Such credit is generally limited to 50% of the containment free volume. This evaluation should consider the magnitude of the containment volume and exhaust rate, the potential for bypass to the environment, the location of exhaust plenums relative to the surface of the reactor cavity, recirculation ventilation systems, and internal walls and floors that impede stream flow between the surface of the reactor cavity and the exhaust plenums.	Conforms	Two-hour release to the environment assumed per Section 4.1.

Table 4. Conformance with Regulatory Guide 1.183 Appendix C (CRDA)

App Sec	RG Position	Columbia Analysis	Comments
1	Assumptions acceptable to the NRC staff regarding core inventory are provided in Regulatory Position 3 of this guide. For the rod drop accident, the release from the breached fuel is based on the estimate of the number of fuel rods breached and the assumption that 10% of the core inventory of the noble gases and iodines is in the fuel gap. The release attributed to fuel melting is based on the fraction of the fuel that reaches or exceeds the initiation temperature for fuel melting and on the assumption that 100% of the noble gases and 50% of the iodines contained in that fraction are released to the reactor coolant.	Conforms	100% of the noble gases and 50% of the iodines released from melted fuel. Other releases also based on Regulatory Position 3 of main report.
2	If no or minimal fuel damage is postulated for the limiting event, the released activity should be the maximum coolant activity (typically 4 μCi/gm DE I-131) allowed by the technical specifications.	Conforms	Substantial fuel damage is postulated. Coolant activity neglected.
3.1	The activity released from the fuel from either the gap or from fuel pellets is assumed to be instantaneously mixed in the reactor coolant within the pressure vessel.	Conforms	
3.2	Credit should not be assumed for partitioning in the pressure vessel or for removal by the steam separators.	Conforms	
3.3	Of the activity released from the reactor coolant within the pressure vessel, 100% of the noble gases, 10% of the iodine, and 1% of the remaining radionuclides are assumed to	Conforms	

Table 4. Conformance with Regulatory Guide 1.183 Appendix C (CRDA)

App Sec	RG Position	Columbia Analysis	Comments
	reach the turbine and condensers.		
3.4	Of the activity that reaches the turbine and condenser, 100% of the noble gases, 10% of the iodine, and 1% of the particulate radionuclides are available for release to the environment. The turbine and condensers leak to the atmosphere as a ground- level release at a rate of 1% per day for a period of 24 hours, at which time the leakage is assumed to terminate. No credit should be assumed for dilution or holdup within the turbine building. Radioactive decay during holdup in the turbine and condenser may be assumed.	Conforms	Release rate of 1% per day for 24 hours. Decay assumed in condenser.
3.5	In lieu of the transport assumptions provided in paragraphs 3.2 through 3.4 above, a more mechanistic analysis may be used on a case-by-case basis. Such analyses account for the quantity of contaminated steam carried from the pressure vessel to the turbine and condensers based on a review of the minimum transport time from the pressure vessel to the first main steam isolation (MSIV) and considers MSIV closure time.	Not applicable	
3.6	The iodine species released from the reactor coolant within the pressure vessel should be assumed to be 95% CsI as an aerosol, 4.85% elemental, and 0.15% organic. The release from the turbine and condenser should be assumed to be 97% elemental and 3% organic.	Conforms	Release to environment assumed to be 97% elemental, 3% organic.

Table 5. Conformance with Regulatory Guide 1.183 Appendix D (MSLB)

App Sec	RG Position	Columbia Analysis	Comments
1	Assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide. The release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached.	Conforms	No fuel damage, release estimate based on coolant activity.
2	If no or minimal fuel damage is postulated for the limiting event, the released activity should be the maximum coolant activity allowed by technical specification. The iodine concentration in the primary coolant is assumed to correspond to the following two cases in the nuclear steam supply system vendor's standard technical specifications.	Conforms	4 uCi/gm consistent with spiking Tech Spec.
2.1	The concentration that is the maximum value (typically 4.0 μCi/gm DE I-131) permitted	Conforms	See previous.

Table 5. Conformance with Regulatory Guide 1.183 Appendix D (MSLB)

App Sec	RG Position	Columbia Analysis	Comments
	and corresponds to the conditions of an assumed pre-accident spike, and		
2.2	The concentration that is the maximum equilibrium value (typically 0.2 μCi/gm DE I-131) permitted for continued full power operation.	Conforms	See previous.
3	The activity released from the fuel should be assumed to mix instantaneously and homogeneously in the reactor coolant. Noble gases should be assumed to enter the steam phase instantaneously.	Conforms	
4.1	The main steam line isolation valves (MSIV) should be assumed to close in the maximum time allowed by technical specifications.	Conforms	The 6 sec assumed in analysis is longer than the Tech Spec max closing time of 5 sec.
4.2	The total mass of coolant released should be assumed to be that amount in the steam line and connecting lines at the time of the break plus the amount that passes through the valves prior to closure.	Conforms	
4.3	All the radioactivity in the released coolant should be assumed to be released to the atmosphere instantaneously as a ground-level release. No credit should be assumed for plateout, holdup, or dilution within facility buildings.	Does not conform	Two cases analyzed: 1. Instantaneous release, but with buoyant rise of released steam. 2. Confined blowdown, ground-level release over two hours. No plateout credited.
4.4	The iodine species released from the main steam line should be assumed to be 95% CsI as an aerosol, 4.85% elemental, and 0.15% organic.	Conforms	

Design Basis Analysis Issue Resolution, Technical Specification Changes, and Technical Specification Bases Changes – Description and Basis

1.0 Resolution of Design Basis Analysis Issues

1.1 JCO -- Secondary Containment Draw-down

This proposed amendment request resolves an issue regarding the establishment of secondary containment vacuum under adverse environmental conditions and supersedes previous submittals reported to the staff in licensee event reports[1] and also as an unreviewed safety question.[2]

The original requirement for standby gas treatment system performance is that the system was designed to reestablish secondary containment to a negative pressure of 0.25-inch vacuum water gauge within 120 seconds of initiation after the loss of coolant accident event. Since this requirement could not be accomplished under the loss of coolant accident conditions when combined with the loss of offsite power event under adverse weather conditions, we submitted Revision 0 of a Justification for Continued Operation to the NRC on September 29, 1989.

On January 3, 1990, the staff responded to our September 29, 1989, letter. That response stated that we had provided sufficient justification to allow continued operation. On February 16, 1990, we sent a letter to the staff that discussed a program plan for resolution of this issue. The plan included preparation of a secondary containment model to determine the wind and temperature conditions for which a defined secondary containment draw-down could be obtained and the required licensing document changes.

On December 22, 1992, we issued another letter to the staff, which discussed changes for the resolution of the secondary containment issue that was presented in the February 16, 1990, letter.

On October 15, 1996, we submitted the revised design basis and a request for amendment to secondary containment and standby gas treatment system Technical Specifications to the staff.[3] During the course of the staff's review of

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1 LER 88-023-00 and LER 88-023-01, "Technical Specification Violation of Secondary Containment to Outside Differential Pressure Caused by Design due to Programmatic Errors"

LER 89-040-00 and LER 89-040-01, "Standby Gas Treatment System Capability not within License Basis Consideration for Secondary Containment Performance under Certain Conditions due to Design"

2 Letter GO2-89-176, dated September 29, 1989, GC Sorensen (Washington Public Power Supply System) to NRC, "Unreviewed Safety Question Regarding Standby Gas Treatment"

3 Letter GO2-96-199 dated October 15, 1996, PR Bemis (Washington Public Power Supply System) to NRC, "Request for Amendment to Secondary Containment and Standby Gas Treatment System Technical Specifications"

our amendment request, we responded to three Requests for Additional Information in letters dated December 4, 1997, April 12, 1999 and June 10, 1999.

On July 16, 1999 we withdrew the amendment request due to the discovery of a non-conservative error in the methodology for computing the volumetric expansion that was used to determine the containment release concentration, and committed to resubmit the amendment request in its entirety.[4]

The Justification for Continued Operation is still in effect; though it has been revised several times as new computer programs were developed to perform refined calculations. Revision 5 of the Justification for Continued Operation was provided to the staff during a February 6, 1995, meeting on post accident containment response.[5]

No equipment changes are required for either secondary containment or the standby gas treatment system. This issue is being resolved by the proposed changes to the secondary containment and standby gas treatment system Technical Specifications and application of alternative source term methodology. These changes are necessary to reflect new standby gas treatment draw-down criteria and flow rates that were determined by calculations based upon actual system operation.

1.2 USQ -- Unfiltered Control Room In-leakage

This proposed amendment request also resolves a design basis analysis issue pertaining to increased unfiltered control room in-leakage into the control room envelope that was reported to the staff in licensee event reports.[6]

During September 8 through 11, 2000, a series of special tests, using a tracer gas decay methodology, were performed to determine the total in-leakage into the control room and the associated impact on control room dose. These tests were performed in support of this proposed amendment request.

On November 29, 2000, final test results were received and an assessment showed that the maximum combined train measured unfiltered in-leakage (plus the measurement uncertainty) for the control room emergency filtration system was

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4 Letter GO2-99-133, dated July 16, 1999, RL Webring (Energy Northwest) to NRC, "Withdrawal of Request for Amendment to Secondary Containment and Standby Gas Treatment System Technical Specifications"

5 Letter, dated March 6, 1995, JW Clifford (NRC) to Washington Public Power Supply System, "Summary of Meeting on Post-Accident Containment Response"

6 LER 2000-006-00, "Plant Outside Design Basis for Control Room Emergency Filtration System Unfiltered In-leakage Based Upon Tracer Gas Testing"

LER 2000-006-01, "Plant Outside Design Basis for Control Room Emergency Filtration System Unfiltered In-leakage Based Upon Tracer Gas Testing"

218 cfm. This was in excess of the current licensing and design basis limit of 10.55 cfm. The impact of the unfiltered in-leakage increase on control room dose was evaluated at the time and it was determined that the design basis thyroid dose of 30 rem to the control room operators would have been exceeded during post-accident conditions.

This is a design basis analysis issue, which is also being resolved by this amendment request (no plant equipment changes are required). Alternative source term methodology has shown that in-leakage rates in excess of 218 cfm (up to 250 cfm) would result in control room doses below the new regulatory reference values.

2.0 Technical Specification Changes

2.1 Technical Specification 1.1, "Definitions"

The definition for DOSE EQUIVALENT I-131 is being revised to remove the word "thyroid" and to replace the reference to dose conversion factors from TID-14844 with a reference to Federal Guidance Report (FGR) 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1989 and FGR 12, "External Exposure to Radionuclides in Air, Water, and Soil," 1993.

This change reflects the application of alternative source term methodology.

2.2 Technical Specification 3.3.6.2, "Secondary Containment Isolation Instrumentation"

Note (b) on Table 3.3.6.2-1 is being deleted. This note currently requires secondary containment isolation instrumentation to be operable during core alterations and during movement of irradiated fuel assemblies in the secondary containment.

With the application of alternative source term, secondary containment is not credited for the fuel handling accident.

2.3 Technical Specification 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)"

Surveillance Requirement 3.6.1.3.10 currently requires verification that the combined leakage rate for all bypass leakage paths is $\leq$ 0.74 scfh when pressurized to $\geq P_a$

The proposed change increases the allowable limit for bypass leakage from $\leq$ 0.74 scfh to 0.04%/day to allow for conservatism and additional leakage. The new value conservatively bounds bypass leakage tests performed at Columbia Generating Station to ensure isolation functionality.

2.4 Technical Specification 3.6.1.8, "Main Steam Isolation Valve Leakage Control (MSLC) System"

This entire section is being deleted because the system no longer meets the criteria of 10 CFR 50.36. With the application of alternative source term methodology, no credit is assumed for the MSLC system in the accident analyses.

2.5 Technical Specification 3.6.4.1, "Secondary Containment"

2.5.1 Technical Specification 3.6.4.1 Applicability

The applicability requirement that secondary containment be operable during movement of irradiated fuel assemblies in the secondary containment and during core alterations is being deleted.

With the application of alternative source term, secondary containment is not credited for the fuel handling accident.

2.5.2 Technical Specification 3.6.4.1 Condition

The Condition C requirement that secondary containment be operable during movement of irradiated fuel assemblies in the secondary containment and during core alterations is being deleted.

With the application of alternative source term, secondary containment is not credited for the fuel handling accident.

2.5.3 Technical Specification 3.6.4.1 Required Action

The Required Actions C.1 and C.2 requirements that secondary containment be operable during movement of irradiated fuel assemblies in the secondary containment and during core alterations respectively are being deleted. The note pertaining to applicability of LCO 3.0.3 is being deleted. Required Action C.3 is being renamed C.1.

With the application of alternative source term, no credit is assumed for secondary containment for the fuel handling accident. The LCO 3.0.3 note can be deleted because the applicability has changed.

2.5.4 Surveillance Requirement 3.6.4.1.1

Surveillance Requirement 3.6.4.1.1.a requires a verification every 24 hours that the pressure within secondary containment is $\geq$ 0.25 inch of vacuum water gauge. The surveillance requirement is being changed to require

verification every 24 hours that the pressure within secondary containment is > 0 inch of vacuum water gauge.

In NUREG-1434, the SR 3.6.4.1.1 requirement is bracketed. This requirement is bracketed in the NUREG because to maintain ≥ 0.25 inch of vacuum water gauge at all times is not required by all boiling water reactors as an initial condition of the accident analysis.

Failure to maintain 0.25 inch of vacuum water gauge with the non-safety-related secondary containment ventilation system is not necessarily indicative of an inoperable secondary containment, nor does maintaining 0.25 inch of vacuum water gauge ensure the secondary containment is operable. In the event that the normal reactor building ventilation system is secured, secondary containment could become pressurized such that the maximum accident design basis pressure of ≥ 0.25-inch vacuum water gauge on all surfaces of secondary containment is exceeded.

Therefore, when the normal reactor building ventilation system is secured, action will be taken in accordance with the current LCOs to ensure that design basis accident mitigation assumptions remain valid. Pressure surveillances are also performed in accordance with Operations procedures for shift and daily instrument checks, as applicable.

2.5.5 Surveillance Requirements (SRs) 3.6.4.1.4 and 3.6.4.1.5

SR 3.6.4.1.4

Technical Specification Surveillance Requirement (SR) 3.6.4.1.4 currently requires secondary containment to be drawn down to greater than or equal to 0.25 inch of vacuum water gauge in less than or equal to 120 seconds.

This surveillance requirement is being deleted because its function will be adequately addressed by proposed changes to current secondary containment and standby gas treatment system surveillances SR 3.6.4.1.5 and SR 3.6.4.3.3 respectively. Taken together, these revised SRs will provide a more realistic and accurate representation of secondary containment draw-down response than current SR 3.6.4.1.4.

SR 3.6.4.1.5

With the deletion of SR 3.6.4.1.4, existing SR 3.6.4.1.5 is being renumbered as SR 3.6.4.1.4. It is also being revised to require secondary containment to be drawn down to at least 0.25 inch of vacuum water gauge at a secondary containment in-leakage flow rate not to exceed 2240 cfm. This surveillance requirement verifies secondary containment integrity by

ensuring that secondary containment in-leakage does not prevent acceptable draw-down. Performance of the draw-down testing, and associated trending of leakage rates, provides assurance that secondary containment integrity is maintained and has not degraded over time.

This surveillance requirement can be satisfied by verifying that the differential pressure is at a greater vacuum (larger negative pressure) than 0.25 inch of vacuum water gauge when measured at any standby gas treatment system flow rate greater than 2240 cfm.

This surveillance requirement, in conjunction with SR 3.6.4.3.3 which is being revised to verify that the standby gas treatment system achieves ≥ 5000 cfm within two minutes after receipt of an initiation signal, demonstrates a system capability (along with containment integrity) that is within the assumptions of the applicable safety analysis.

For the secondary containment to be considered operable, it must have adequate leak tightness to ensure that the required vacuum can be established and maintained. Taken together, existing SRs 3.6.4.1.5 and 3.6.4.3.3 (as they would be modified by this amendment request) will provide an adequate demonstration of secondary containment performance and would detect unacceptable increases in secondary containment leakage or standby gas treatment system degradation.

Furthermore, a review of completed draw-down test data since 1990 shows a decline in secondary containment leakage rates from approximately 1300 cfm to the current value of approximately 700 cfm. This improved leakage rate is due primarily to the modifications associated with penetration seal and fire barrier rework projects, and enhanced control of the secondary containment barrier impairment process.

2.6 Technical Specification 3.6.4.2, "Secondary Containment Isolation Valves (SCIVs)"

2.6.1 Technical Specification 3.6.4.2 Applicability

The applicability requirement that secondary containment be operable during movement of irradiated fuel assemblies in the secondary containment and during core alterations is being deleted.

With the application of alternative source term, secondary containment is not credited for the fuel handling accident.

2.6.2 Technical Specification 3.6.4.2 Condition

The Condition D requirement that secondary containment be operable during movement of irradiated fuel assemblies in the secondary containment and during core alterations is being deleted.

With the application of alternative source term, secondary containment is not credited for the fuel handling accident.

2.6.3 Technical Specification 3.6.4.2 Required Action

The Required Actions D.1 and D.2 requirements that secondary containment be operable during movement of irradiated fuel assemblies in the secondary containment and during core alterations respectively are being deleted. The note pertaining to applicability of LCO 3.0.3 is being deleted. Required Action D.3 is being renamed D.1

With the application of alternative source term, secondary containment is not credited for the fuel handling accident. The LCO 3.0.3 note can be deleted because the applicability has changed.

2.7 Technical Specification 3.6.4.3, "Standby Gas Treatment System"

2.7.1 Technical Specification 3.6.4.3 Applicability

The applicability requirement that the standby gas treatment system be operable during movement of irradiated fuel assemblies in the secondary containment and during core alterations is being deleted.

With the application of alternative source term, secondary containment is not credited for the fuel handling accident.

2.7.2 Technical Specification 3.6.4.3 Condition

The Condition C requirement that the standby gas treatment system be operable during movement of irradiated fuel assemblies in the secondary containment and during core alterations is being deleted.

With the application of alternative source term, secondary containment is not credited for the fuel handling accident.

2.7.3 Technical Specification 3.6.4.3 Required Action

The Required Actions C.2.1 and C.2.2 requirements that the standby gas treatment system be operable during movement of irradiated fuel assemblies in the secondary containment and during core alterations respectively are being deleted. The notes pertaining to applicability of LCO 3.0.3 are being deleted. Required Action C.2.3 is being renamed C.2.

The Required Actions E.1 and E.2 requirements that the standby gas treatment system be operable during movement of irradiated fuel assemblies in the secondary containment and during core alterations respectively are being deleted. The notes pertaining to applicability of LCO 3.0.3 are being deleted. Required Action E.3 is being renamed E.1.

With the application of alternative source term, secondary containment is not credited for the fuel handling accident. The LCO 3.0.3 notes can be deleted because the applicability has changed.

2.7.4 Surveillance Requirement 3.6.4.3.3

Surveillance Requirement 3.6.4.3.3 requires a verification every 24 months that each standby gas treatment subsystem actuates on an actual or simulated initiation signal.

This surveillance is being revised to verify every 24 months that each standby gas treatment subsystem actuates on an actual or simulated initiation signal and reaches $\geq$ 5000 cfm in $\leq$ two minutes.

Reaching this flow rate within two minutes, in conjunction with SR 3.6.4.1.4, demonstrates that the standby gas treatment subsystem can draw down secondary containment within the assumptions of the applicable safety analysis.

2.8 Technical Specification 5.5.7, "Ventilation Filter Test Program (VFTP)"

Technical Specification Ventilation Filter Testing Program Requirements 5.5.7 a, b, and d specify a standby gas treatment system flow rate of 4457 cfm (nominal) for filter testing. As previously discussed, the secondary containment draw-down analysis assumes the standby gas treatment system flow rate to be 5000 cfm within two minutes of an accident start signal.

The new 5000 cfm value for standby gas treatment system flow rate has been evaluated to ensure that it is high enough to satisfy secondary containment draw-down requirements and low enough to provide adequate atmosphere residence time for 99 percent filter efficiency credit in the design basis analyses. The change to the standby gas treatment system flow rate is an analytical change only. No changes to plant equipment or equipment setpoints are required. Standby gas treatment system flow rate for filter test purposes is 4500 to 5500 cfm, which complies with American National Standards Institute (ANSI) Standard N510-1989, "Testing of Nuclear Air Treatment Systems."

The current setpoint calculation methodology considers actual plant calibration data to determine total instrument uncertainties, including drift. The method is based on American National Standards Institute/Instrument Society of America (ANSI/ISA) Standard S67.04-1988, "Setpoints for Nuclear Safety-Related Instrumentation," and guidelines in ISA draft Recommended Practice RP67.04, "Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation." Specifically, the method includes: 1) definition of loop to be analyzed; 2) determination of the analytical limit; 3) determination of the normal and accident environmental conditions for each loop component, including but not limited to pressure, humidity, seismic, temperature and radiation; 4) determination of the normal and accident environmental condition effects; 5) determination of drift effects; 6) combination of the effect terms; and 7) determination of setting range.

A standby gas treatment fan flow rate of 5000 cfm is used in the design basis analyses for both the draw-down and filtration capabilities of the standby gas treatment system, so it is appropriate that this same flow rate value be used to demonstrate both of these design basis capabilities. Therefore, the surveillance requirements demonstrating standby gas treatment system filtration capability have been revised to specify a system flow rate of 5000 cfm.

3.0 Technical Specification Bases Changes -- Highlights

3.1 General

3.1.1 Numerous references to 10 CFR 100 were replaced with reference to 10 CFR 50.67 and to reflect application of the alternative source term.

3.1.2 Several sections were revised to reflect that secondary containment and the standby gas treatment system are no longer required during movement of irradiated fuel assemblies or during core alterations. This is acceptable because, with the application of alternative source term, secondary containment is not credited for the fuel handling accident.

3.2 Main Steam Leakage Control System

This section was deleted in its entirety because the system no longer meets the criteria of 10 CFR 50.36. With the application of alternative source term methodology, no credit is assumed for the MSLC system in the accident analyses.

3.3 Primary Containment

This section was revised to reflect that the bounding radiological analysis for the LOCA is an inadequate core cooling accident that degrades to core damage, rather than a double-ended recirculation suction line break. The specific event could be a recirculation line break, a main steam line break inside containment, an intact vessel inadequate core cooling event, or a spectrum of LOCA transients. This change is justified by the alternative source term analysis.

3.4 Residual Heat Removal System

This section was revised to reflect the use of the residual heat removal drywell spray system post-accident to wash inorganic iodine and particulates from the drywell atmosphere into the suppression pool and to reduce primary containment pressure during the LOCA source term event. This change is justified by the application of alternative source term methodology.

3.5 Standby Liquid Control System

This section was revised to reflect the additional use of the standby liquid control system to buffer suppression pool pH to prevent iodine re-evolution during a postulated radiological release. This additional use of the system is a consequence of an inadequate core cooling event, using current emergency operating procedures/severe accident guidelines and application of alternative source term methodology.

3.6 Secondary Containment and Standby Gas Treatment System

3.6.1 These sections were revised to reflect the proposed changes to resolve the design basis analysis issue regarding the establishment of secondary containment vacuum under adverse environmental conditions and to also increase allowable secondary containment bypass leakage. These changes are justified by the alternative source term analysis and that the standby gas treatment subsystems can draw down secondary containment within the assumptions of the applicable safety analysis.

3.6.2 The standby gas treatment system section was also revised to better describe the existing design. The standby gas treatment system acts as part of secondary containment to minimize and control airborne radiological releases from the plant. Unfiltered release of primary containment leakage following a severe accident or the release of radioactive gases and particulates resulting from accidents outside primary containment are prevented by achieving and maintaining a secondary containment pressure of at least 0.25 inch vacuum water gauge with respect to atmospheric pressure and by filtering the effluent gases from secondary containment through a filter train. A secondary function of the standby gas treatment system is to filter the purge exhaust from primary containment whenever radiation levels within the primary containment exceed acceptable levels for direct purge to the environment by means of the reactor building exhaust system.

The standby gas treatment system consists of two filter trains (A and B), each supported by Division I and II components and controls. The active system components in each filter train, including their start logic, are redundant and configured in a lead/lag operational design.

Each filter train contains (listed in sequence from inlet to outlet) a demister (moisture separator), two banks of electric heaters in series, a prefilter, a high efficiency particulate (HEPA) filter, an electric strip heater, an activated charcoal bed for iodine adsorption, a second electric strip heater, a second activated charcoal bed for iodine adsorption, a second HEPA filter, and redundant instrumentation to measure temperature, humidity and flow. Each standby gas treatment system filter train is capable of independently processing the air flow. The demister removes water droplets from the inlet air and the two banks of heaters, one powered from the lead fan power division and one powered from the lag fan power division, maintain the relative humidity below 70

percent to ensure a standby gas treatment filtration efficiency of 99 percent.

The active system components in each filter train that are required for post-accident operation are redundant to allow for lead/lag operation. The Train A subsystems are designated A1 and A2 and the Train B subsystems are designated B1 and B2. The major components within each subsystem are identified in Figure 1. The lead subsystem of each train is powered from a separate emergency diesel divisional bus (Division 1 or 2) than the lag subsystem. As indicated in Figure 1, the Train A lead components are A1 and the lag components are A2. For Train B, the lead components are B2 and the lag components are B1.

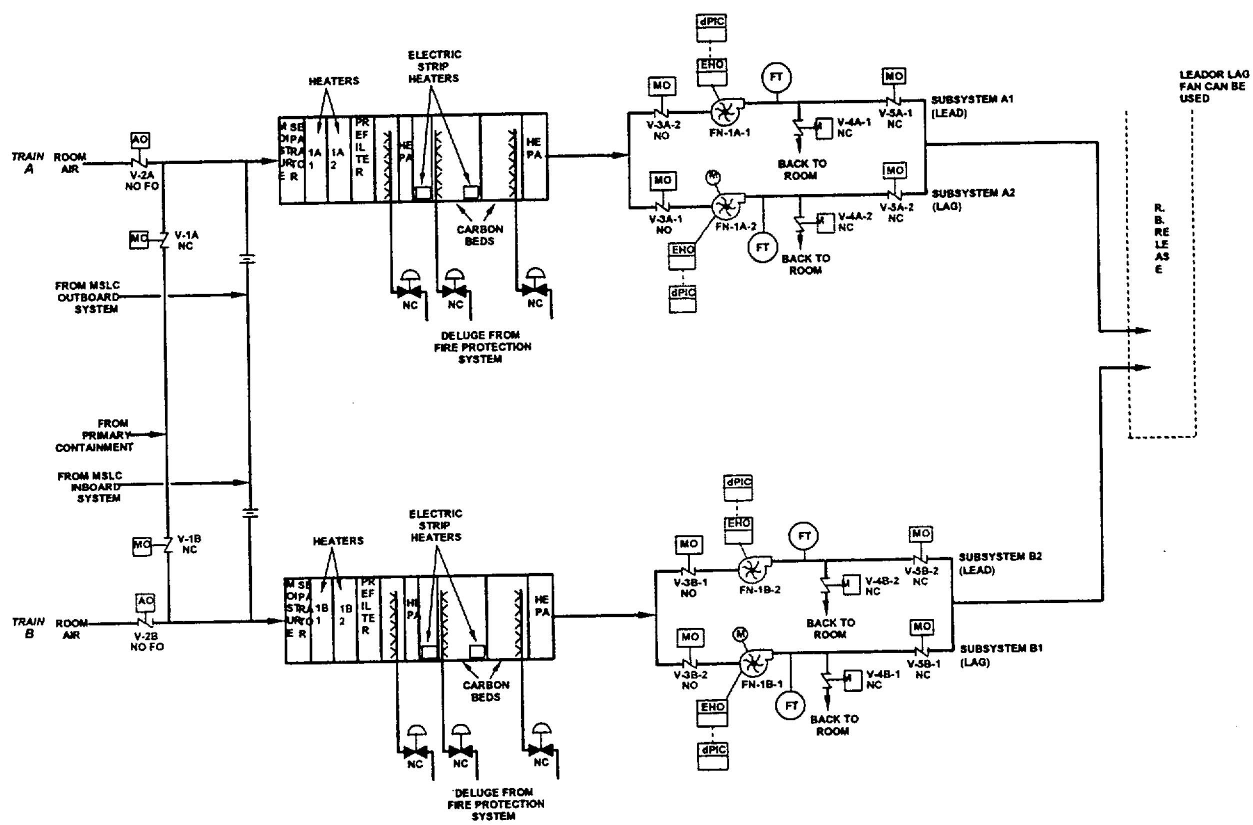

FIGURE 1. STANDBY GAS TREATMENT SYSTEM

10 CFR 50.92 Evaluation

Summary of Proposed Change

Energy Northwest is requesting a revision to the Columbia Generating Station Operating License. Specifically, we are requesting a revision to the Technical Specifications and licensing and design bases to reflect the application of alternative source term methodology.

The alternative source term analyses were performed following the guidance in accordance with Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 2000, and Standard Review Plan Section 15.0.1, "Radiological Consequences Analyses Using Alternative Source Terms."

The alternative source term analyses have been performed without crediting secondary containment during fuel handling accidents. As such, the proposed license amendment relaxes operability requirements during fuel handling and core alterations for: 1) secondary containment; 2) secondary containment isolation instrumentation; and 3) the standby gas treatment system. The alternative source term analyses have also been performed without crediting the main steam leakage control system; therefore, the licensing basis is being revised to reflect the proposed deactivation of the system.

This proposed license amendment request also resolves a Justification for Continued Operation regarding the establishment of secondary containment vacuum under adverse environmental conditions. This was reported to the staff in licensee event reports and also as an unreviewed safety question (Licensee Event Reports 88-0023-00, 88-023-01, 89-040-00 and 89-040-01). The proposed changes to the secondary containment and standby gas treatment system Technical Specifications and application of alternative source term will ensure that secondary containment draw-down and bypass leakage are within the assumptions of the applicable safety analysis.

In addition, this proposed request resolves a previously-identified Unreviewed Safety Question pertaining to increased unfiltered control room leakage into the control room envelope (Licensee Event Reports 2000-006-00 and 2000-006-01). However, application of alternative source term methodology has demonstrated that new design basis limits for in-leakage result in control room doses below the regulatory limit. The new design basis for in-leakage bounds the results identified by tracer gas testing.

No Significant Hazards Consideration Determination

The standards used to arrive at a determination that an amendment request does not involve a significant hazard are included in 10 CFR 50.92. Energy Northwest has evaluated the proposed change to the Technical Specifications and licensing and design bases using the criteria established in 10 CFR 50.92(c) and has determined that it involves no significant hazards consideration as described as follows:

1. **The operation of Columbia Generating Station in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.**

   The alternative source term does not affect the design or operation of the facility; rather, once the occurrence of an accident has been postulated, the new source term is an input to evaluate the consequence. The implementation of the alternative source term methodology has been evaluated in revisions to the analyses of the following limiting design basis accidents at Columbia Generating Station:

   - Control Rod Drop Accident
   - Fuel Handling Accident
   - Main Steam Line Break Accident
   - Loss of Coolant Accident

   Based upon the results of these analyses, it has been demonstrated that, with the requested changes, the dose consequences of these limiting events are within the regulatory guidance provided by the NRC for use with the alternative source term. This guidance is presented in 10 CFR 50.67 and associated Regulatory Guide 1.183, and Standard Review Plan Section 15.0.1.

   Requirements for secondary containment operability, secondary containment isolation valves, and the standby gas treatment system during fuel movement or core alterations are being eliminated. This is acceptable because, with the application of alternative source term methodology, secondary containment is not credited for the fuel handling accident. The licensing basis is being revised to reflect the proposed deactivation of the main steam leakage control system. This is acceptable because, with the application of alternative source term methodology, no credit is assumed for the system in the accident analyses.

   With regard to the Justification for Continued Operation regarding the establishment of secondary containment vacuum under adverse environmental conditions, the proposed changes to the secondary containment and standby gas treatment system Technical Specifications and application of alternative source term methodology ensures that secondary containment draw-down and bypass leakage are within the assumptions of the applicable safety analysis.

With regard to the previously-identified Unreviewed Safety Question pertaining to increased unfiltered control room in-leakage into the control room envelope, application of alternative source term methodology has shown that in-leakage rates in excess of tested values would result in control room doses below the regulatory limit.

Therefore, operation of Columbia Generating Station in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **The operation of Columbia Generating Station in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.**

The alternative source term does not affect the design, functional performance or operation of the facility. Similarly, it does not affect the design or operation of any structures, systems or components equipment or systems involved in the mitigation of any accidents, nor does it affect the design or operation of any component in the facility such that new equipment failure modes are created.

Requirements for the main steam leakage control system are being deleted by this proposed amendment request. This is acceptable because the system no longer meets the criteria of 10 CFR 50.36. With the application of alternative source term methodology, no credit is assumed for the system in the accident analyses. Furthermore, since the main steam leakage control system is a mitigating system, it cannot create the possibility of an accident.

Requirements for secondary containment operability, secondary containment isolation valves, and the standby gas treatment system during fuel movement or core alterations are being eliminated. This is also acceptable because, with the application of alternative source term methodology, secondary containment is not credited for the fuel handling accident.

Therefore, the operation of Columbia Generating Station in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **The operation of Columbia Generating Station in accordance with the proposed amendment will not involve a significant reduction in the margin of safety.**

The changes proposed are associated with the implementation of a new licensing basis for Columbia Generating Station. Approval of the basis change from the original source term developed in accordance with TID-14844 to a new alternative source term as described in Regulatory Guide 1.183 is requested by this submittal. The results of the accident analyses revised in support of this submittal, and the requested Technical Specification changes, are subject to revised acceptance criteria. These analyses have been performed using conservative methodologies.

Safety margins and analytical conservatisms have been evaluated and are satisfied. The analyzed events have been carefully selected and margin has been retained to ensure that the analyses adequately bound postulated event scenarios. The dose consequences of these limiting events are within the acceptance criteria also found in the latest regulatory guidance. This guidance is presented in 10 CFR 50.67 and associated Regulatory Guide 1.183.

The proposed changes can be made while still satisfying regulatory requirements and review criteria, with significant margin. The changes continue to ensure that the doses at the exclusion area and low population zone boundaries, as well as the control room, are within the corresponding regulatory limits.

Therefore, operation of Columbia Generating Station in accordance with the proposed amendment will not involve a significant reduction in the margin of safety.

In summary and based upon the above considerations, we have concluded that a significant hazard would not be introduced as a result of this proposed change.

Environmental Assessment Applicability Review

Energy Northwest has evaluated the proposed amendment against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21.

It has been determined that the proposed change meets the criteria for categorical exclusion as provided for under 10 CFR 51.22(c)(9). This conclusion has been determined because the requested change does not involve a significant hazards consideration, nor does it involve a significant change in the types or significant increase in the amounts of any effluents that may be released off-site. The following table demonstrates that Columbia Generating Station meets the radiological criteria described in 10 CFR 50.67 for the exclusion area boundary (EAB), the low population zone (LPZ) and control room. The EAB and LPZ doses represent a small fraction of the dose limits. Control room exposure to operators is less than five REM TEDE limit over 30 days for all accidents.

DOSE RESULTS (REM)			
ACCIDENT	CR	EAB	LPZ
DOSE LIMIT	5.0	25	25
LOCA	4.39	2.32	2.97
MSLB	0.28	0.71	0.20
DOSE LIMIT	5.0	6.3	6.3
CRDA	0.655	0.022	0.023
FHA	1.37	1.01	0.28

Adoption of the alternative source term and Technical Specification changes which implement certain conservative assumptions in the alternative source term analyses will not result in modifications to the plant or changes in its operation which could alter the type or amounts of effluents that may be released offsite.

The alternative source term does not affect the design or operation of the facility; rather, once the occurrence of an accident has been postulated, the alternative source term is an input to evaluate the consequence. The implementation of the alternative source term has been evaluated in revisions to the analyses of the limiting design basis accidents at Columbia Generating Station (control rod drop accident, fuel handling accident, loss of coolant accident, and main steam line break accident). Based upon the results of these analyses it has been demonstrated that, with the requested changes, the dose consequences of these limiting events are within the regulatory guidance provided by the NRC for use with alternative source term (i.e., 10 CFR 50.67 and 10 CFR 50, Appendix A, General Design Criterion 19). Thus, there will be no significant increase in either individual or cumulative occupational radiation exposure.

LICENSE AMENDMENT REQUEST
ALTERNATIVE SOURCE TERM
Attachment 6

Marked-Up Version of Technical Specifications

TABLE OF CONTENTS

(continued)

1.1 Definitions (continued)

CHANNEL FUNCTIONAL TEST

A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY, including required alarm, interlock, display, and trip functions, and channel failure trips. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is tested.

CORE ALTERATION

CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components within the reactor vessel with the vessel head removed and fuel in the vessel. The following exceptions are not considered to be CORE ALTERATIONS:

a. Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special movable detectors (including undervessel replacement); and

b. Control rod movement, provided there are no fuel assemblies in the associated core cell.

Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

CORE OPERATING LIMITS REPORT (COLR)

The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.

DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same ~~thyroid~~ dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The ~~thyroid~~ dose conversion factors used for this calculation shall be those listed in ~~Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for~~

Federal Guidance Report (FGR) 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1989;

(continued)

1.1 Definitions

FGR 12, "External Exposure to Radionuclides in Air, Water and Soil," 1993;

DOSE EQUIVALENT I-131 (continued)

~~Power and Test Reactor Sites,"~~ Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977; or ICRP 30, Supplement to Part 1, page 192-212, Table titled "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity."

EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME

The ECCS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS initiation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.

END OF CYCLE RECIRCULATION PUMP TRIP (EOC-RPT) SYSTEM RESPONSE TIME

The EOC-RPT SYSTEM RESPONSE TIME shall be that time interval from initial signal generation by the associated turbine throttle valve limit switch or from when the turbine governor valve hydraulic control oil pressure drops below the pressure switch setpoint to complete suppression of the electric arc between the fully open contacts of the recirculation pump circuit breaker. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.

ISOLATION SYSTEM RESPONSE TIME

The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation initiation setpoint at the channel sensor until the isolation valves travel to their required positions. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.

(continued)

Table 3.3.6.2-1 (page 1 of 1)
Secondary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES AND OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level — Low Low, Level 2	1,2,3,(a)	2~~(c)~~ (b)	SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≥ -58 inches
2. Drywell Pressure — High	1,2,3	2~~(c)~~ (b)	SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≤ 1.88 psig
3. Reactor Building Vent Exhaust Plenum Radiation — High	1,2,3, (a)~~,(b)~~	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≤ 16.0 mR/hr
4. Manual Initiation	1,2,3, (a)~~,(b)~~	4	SR 3.3.6.2.4	NA

(a) During operations with a potential for draining the reactor vessel.

~~(b) During CORE ALTERATIONS, and during movement of irradiated fuel assemblies in the secondary containment.~~

(b) ~~(c)~~ Also required to initiate the associated LOCA Time Delay Relay ~~Funciton~~ Function pursuant to LCO 3.3.5.1.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.6	Verify the isolation time of each MSIV is $\geq$ 3 seconds and $\leq$ 5 seconds.	In accordance with the Inservice Testing Program
SR 3.6.1.3.7	Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal.	24 months
SR 3.6.1.3.8	Verify a representative sample of reactor instrument line EFCVs actuate to the isolation position on an actual or simulated instrument line break signal.	24 months
SR 3.6.1.3.9	Remove and test the explosive squib from each shear isolation valve of the TIP System.	24 months on a STAGGERED TEST BASIS
SR 3.6.1.3.10	Verify the combined leakage rate for all secondary containment bypass leakage paths is $\leq$ ~~0.74 scfh~~ 0.04%/day when pressurized to $\geq P_a$.	In accordance with the Primary Containment Leakage Rate Testing Program

(continued)

DELETE ENTIRE SPECIFICATION 3.6.1.8

3.6 CONTAINMENT SYSTEMS

3.6.1.8 Main Steam Isolation Valve Leakage Control (MSLC) System

LCO 3.6.1.8 Two MSLC subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One MSLC subsystem inoperable.	A.1 Restore MSLC subsystem to OPERABLE status.	30 days
B. Two MSLC subsystems inoperable.	B.1 Restore one MSLC subsystem to OPERABLE status.	7 days
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.1.8.1 Operate each MSLC blower ≥ 15 minutes.	31 days

(continued)

DELETE ENTIRE SPECIFICATION 3.6.1.8

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.8.2	Verify electrical continuity of each inboard MSLC subsystem heater element circuitry.	31 days
SR 3.6.1.8.3	Perform a system functional test of each MSLC subsystem.	18 months

3.6 CONTAINMENT SYSTEMS

3.6.4.1 Secondary Containment

LCO 3.6.4.1 The secondary containment shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
~~During movement of irradiated fuel assemblies in the secondary containment,~~
~~During CORE ALTERATIONS,~~
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Secondary containment inoperable in MODE 1, 2, or 3.	A.1	Restore secondary containment to OPERABLE status.	4 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 3.	12 hours
	<u>AND</u>		
	B.2	Be in MODE 4.	36 hours

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Secondary containment inoperable ~~during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or~~ during OPDRVs.	C.1 ~~-----NOTE----- LCO 3.0.3 is not applicable. ----------~~ ~~Suspend movement of irradiated fuel assemblies in the secondary containment.~~ ~~AND~~ ~~C.2 Suspend CORE ALTERATIONS.~~ ~~AND~~ C.3 Initiate action to suspend OPDRVs.	~~Immediately~~ ~~Immediately~~ Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1.1	Verify secondary containment vacuum is ~~≥ 0.25~~ > 0 inch of vacuum water gauge.	24 hours
SR 3.6.4.1.2	Verify all secondary containment equipment hatches are closed and sealed.	31 days
SR 3.6.4.1.3	Verify each secondary containment access inner door or each secondary containment access outer door in each access opening is closed.	31 days
~~SR 3.6.4.1.4~~	~~Verify each standby gas treatment (SGT) subsystem will draw down the secondary containment to ≥ 0.25 inch of vacuum water gauge in ≤ 120 seconds.~~	~~24 months on a STAGGERED TEST BASIS~~
SR 3.6.4.1.~~5~~ 4	Verify each SGT subsystem can maintain ≥ 0.25 inch of vacuum water gauge in the secondary containment for 1 hour at ~~a~~ an inleakage flow rate ≤ 2240 cfm.	24 months on a STAGGERED TEST BASIS

3.6 CONTAINMENT SYSTEMS

3.6.4.2 Secondary Containment Isolation Valves (SCIVs)

LCO 3.6.4.2 Each SCIV shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
~~During movement of irradiated fuel assemblies in the secondary containment,~~
~~During CORE ALTERATIONS,~~
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

------------------------------------NOTES------------------------------------

1. Penetration flow paths may be unisolated intermittently under administrative controls.

2. Separate Condition entry is allowed for each penetration flow path.

3. Enter applicable Conditions and Required Actions for systems made inoperable by SCIVs.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more penetration flow paths with one SCIV inoperable.	A.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	8 hours
	<u>AND</u>		

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A or B not met ~~during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or~~ during OPDRVs.	~~D.1 ---------NOTE--------- LCO 3.0.3 is not applicable. ------------------------- Suspend movement of irradiated fuel assemblies in the secondary containment.~~	~~Immediately~~
	~~AND~~	
	~~D.2 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~	
	D.3 Initiate action to suspend OPDRVs.	Immediately

3.6 CONTAINMENT SYSTEMS

3.6.4.3 Standby Gas Treatment (SGT) System

LCO 3.6.4.3 Two SGT subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
~~During movement of irradiated fuel assemblies in the secondary containment,~~
~~During CORE ALTERATIONS,~~
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SGT subsystem inoperable.	A.1 Restore SGT subsystem to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, or 3.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours
C. Required Action and associated Completion Time of Condition A not met ~~during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or~~ during OPDRVs.	~~-----NOTE----- LCO 3.0.3 is not applicable. ----------~~ C.1 Place OPERABLE SGT subsystem in operation. OR	Immediately (continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	~~C.2.1 Suspend movement of irradiated fuel assemblies in the secondary containment.~~	~~Immediately~~
	~~AND~~	
	~~C.2.2 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~	
	~~C.2.3~~ Initiate action to suspend OPDRVs.	Immediately
D. Two SGT subsystems inoperable in MODE 1, 2, or 3.	D.1 Enter LCO 3.0.3.	Immediately
E. Two SGT subsystems inoperable ~~during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or~~ during OPDRVs.	E.1 ~~-------NOTE------- LCO 3.0.3 is not applicable. ------------------- Suspend movement of irradiated fuel assemblies in the secondary containment.~~	~~Immediately~~
	~~AND~~	
	~~E.2 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~	
	~~E.3~~ Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.3.1	Operate each SGT subsystem for $\geq$ 10 continuous hours with heaters operating.	31 days
SR 3.6.4.3.2	Perform required SGT filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.6.4.3.3	Verify each SGT subsystem actuates on an actual or simulated initiation signal.	24 months
SR 3.6.4.3.4	Verify each SGT filter cooling recirculation valve can be opened and the fan started.	24 months

and reaches $\geq$ 5000 cfm in $\leq$ 2 minutes

5.5 Programs and Manuals

5.5.6 Inservice Testing Program (continued)

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.

5.5.7 Ventilation Filter Testing Program (VFTP)

The VFTP shall establish the required testing of Engineered Safety Feature (ESF) filter ventilation systems.

Tests described in Specification 5.5.7.a and 5.5.7.b shall be performed once per 24 months; after each complete or partial replacement of the HEPA filter train or charcoal adsorber filter; after any structural maintenance on the system housing; and, following significant painting, fire, or chemical release in any ventilation zone communicating with the system while it is in operation.

Tests described in Specification 5.5.7.c shall be performed once per 24 months; after 720 hours of system operation; after any structural maintenance on the system housing; and, following significant painting, fire, or chemical release in any ventilation zone communicating with the system while it is in operation.

Tests described in Specification 5.5.7.d and 5.5.7.e shall be performed once per 24 months.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test Frequencies.

a. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < 0.05% when tested in accordance with Regulatory Guide 1.52, Revision 2, and ASME N510-1989 at the system flowrate specified below:

ESF Ventilation System	Flowrate (cfm)
SGT System	~~4012 to 4902~~ 4500 to 5500
CREF System	900 to 1100

(continued)

5.5 Programs and Manuals

5.5.7 Ventilation Filter Testing Program (VFTP) (continued)

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < 0.05% when tested in accordance with Regulatory Guide 1.52, Revision 2, and ASME N510-1989 at the system flowrate specified below:

ESF Ventilation System	Flowrate (cfm)
SGT System	~~4012 to 4902~~ 4500 to 5500
CREF System	900 to 1100

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and the relative humidity specified below. Testing of the SGT System will also be conducted at a face velocity of 75 feet per minute.

ESF Ventilation System	Penetration (%)	RH (%)
SGT System	0.5	70
CREF System	2.5	70

Allowed tolerances in the above testing parameters of temperature, relative humidity, and face velocity are as specified in ASTM D3803-1989.

d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters and the charcoal adsorbers is less than the value specified below when tested at the system flowrate specified below:

ESF Ventilation System	Delta P (inches wg)	Flowrate (cfm)
SGT System	< 8	~~4012 to 4902~~ 4500 to 5500
CREF System	< 6	900 to 1100

(continued)

Typed Version of Technical Specifications

TABLE OF CONTENTS

(continued)

(continued)

1.0 USE AND APPLICATION

1.1 Definitions

------------------------------------NOTE------------------------------------
The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

Term	Definition
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)	The APLHGR shall be applicable to a specific planar height and is equal to the sum of the LHGRs for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle at the height.
CHANNEL CALIBRATION	A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel, including the required sensor, alarm, display, and trip functions, and shall include the CHANNEL FUNCTIONAL TEST. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is calibrated.
CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.

(continued)

1.1 Definitions (continued)

CHANNEL FUNCTIONAL TEST	A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY, including required alarm, interlock, display, and trip functions, and channel failure trips. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is tested.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components within the reactor vessel with the vessel head removed and fuel in the vessel. The following exceptions are not considered to be CORE ALTERATIONS: a. Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special movable detectors (including undervessel replacement); and b. Control rod movement, provided there are no fuel assemblies in the associated core cell. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The dose conversion factors used for this calculation shall be those listed in Federal Guidance Report (FGR) 11, "Limiting Values of Radionuclide Intake and Air Concentration and

(continued)

1.1 Definitions

DOSE EQUIVALENT I-131 (continued)	Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1989; FGR 12, "External Exposure to Radionuclides in Air, Water and Soil," 1993; Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977; or ICRP 30, Supplement to Part 1, page 192-212, Table titled "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity."
EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME	The ECCS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS initiation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
END OF CYCLE RECIRCULATION PUMP TRIP (EOC-RPT) SYSTEM RESPONSE TIME	The EOC-RPT SYSTEM RESPONSE TIME shall be that time interval from initial signal generation by the associated turbine throttle valve limit switch or from when the turbine governor valve hydraulic control oil pressure drops below the pressure switch setpoint to complete suppression of the electric arc between the fully open contacts of the recirculation pump circuit breaker. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
ISOLATION SYSTEM RESPONSE TIME	The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation initiation setpoint at the channel sensor until the isolation valves travel to their required positions. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.

(continued)

1.1 Definitions (continued)

LEAKAGE	LEAKAGE shall be: a. <u>Identified LEAKAGE</u> 1. LEAKAGE into the drywell such as that from pump seals or valve packing, that is captured and conducted to a sump or collecting tank; or 2. LEAKAGE into the drywell atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; b. <u>Unidentified LEAKAGE</u> All LEAKAGE into the drywell that is not identified LEAKAGE; c. <u>Total LEAKAGE</u> Sum of the identified and unidentified LEAKAGE; and d. <u>Pressure Boundary LEAKAGE</u> LEAKAGE through a nonisolable fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall.
LINEAR HEAT GENERATION RATE (LHGR)	The LHGR shall be the heat generation rate per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.
LOGIC SYSTEM FUNCTIONAL TEST	A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all required logic components (i.e., all required relays and contacts, trip units, solid state logic elements, etc.) of a logic circuit, from as close to the sensor as practicable up to, but not including, the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total system steps so that the entire logic system is tested.

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.6.2.2 Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.6.2.3 Perform CHANNEL CALIBRATION.	18 months
SR 3.3.6.2.4 Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months

Table 3.3.6.2-1 (page 1 of 1)
Secondary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES AND OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level – Low Low, Level 2	1,2,3,(a)	2[(b)]	SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≥ -58 inches
2. Drywell Pressure – High	1,2,3	2[(b)]	SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≤ 1.88 psig
3. Reactor Building Vent Exhaust Plenum Radiation – High	1,2,3,(a)	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≤ 16.0 mR/hr
4. Manual Initiation	1,2,3,(a)	4	SR 3.3.6.2.4	NA

(a) During operations with a potential for draining the reactor vessel.

(b) Also required to initiate the associated LOCA Time Delay Relay Function pursuant to LCO 3.3.5.1.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.3	-----NOTES----- 1. Valves and blind flanges in high radiation areas may be verified by use of administrative means. 2. Not required to be met for PCIVs that are open under administrative controls. ---------- Verify each primary containment isolation manual valve and blind flange that is located inside primary containment and is required to be closed during accident conditions is closed.	Prior to entering MODE 2 or 3 from MODE 4 if primary containment was de-inerted while in MODE 4, if not performed within the previous 92 days
SR 3.6.1.3.4	Verify continuity of the traversing incore probe (TIP) shear isolation valve explosive charge.	31 days
SR 3.6.1.3.5	Verify the isolation time of each power operated and each automatic PCIV, except MSIVs, is within limits.	In accordance with the Inservice Testing Program

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.6	Verify the isolation time of each MSIV is $\geq$ 3 seconds and $\leq$ 5 seconds.	In accordance with the Inservice Testing Program
SR 3.6.1.3.7	Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal.	24 months
SR 3.6.1.3.8	Verify a representative sample of reactor instrument line EFCVs actuate to the isolation position on an actual or simulated instrument line break signal.	24 months
SR 3.6.1.3.9	Remove and test the explosive squib from each shear isolation valve of the TIP System.	24 months on a STAGGERED TEST BASIS
SR 3.6.1.3.10	Verify the combined leakage rate for all secondary containment bypass leakage paths is $\leq$ 0.04%/day when pressurized to $\geq P_a$.	In accordance with the Primary Containment Leakage Rate Testing Program

(continued)

3.6 CONTAINMENT SYSTEMS

3.6.4.1 Secondary Containment

LCO 3.6.4.1 The secondary containment shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Secondary containment inoperable in MODE 1, 2, or 3.	A.1 Restore secondary containment to OPERABLE status.	4 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours
C. Secondary containment inoperable during OPDRVs.	C.1 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1.1	Verify secondary containment vacuum is > 0 inch of vacuum water gauge.	24 hours
SR 3.6.4.1.2	Verify all secondary containment equipment hatches are closed and sealed.	31 days
SR 3.6.4.1.3	Verify each secondary containment access inner door or each secondary containment access outer door in each access opening is closed.	31 days
SR 3.6.4.1.4	Verify each SGT subsystem can maintain $\geq$ 0.25 inch of vacuum water gauge in the secondary containment for 1 hour at an inleakage flow rate $\leq$ 2240 cfm.	24 months on a STAGGERED TEST BASIS

3.6 CONTAINMENT SYSTEMS

3.6.4.2 Secondary Containment Isolation Valves (SCIVs)

LCO 3.6.4.2 Each SCIV shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

------------------------------------NOTES------------------------------------

1. Penetration flow paths may be unisolated intermittently under administrative controls.

2. Separate Condition entry is allowed for each penetration flow path.

3. Enter applicable Conditions and Required Actions for systems made inoperable by SCIVs.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more penetration flow paths with one SCIV inoperable.	A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange. AND	8 hours (continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2	--------NOTE--------- Isolation devices in high radiation areas may be verified by use of administrative means. --------------------- Verify the affected penetration flow path is isolated.	 Once per 31 days
B. ---------NOTE--------- Only applicable to penetration flow paths with two isolation valves. --------------------- One or more penetration flow paths with two SCIVs inoperable.	B.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	4 hours
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, or 3.	C.1	Be in MODE 3.	12 hours
	<u>AND</u>		
	C.2	Be in MODE 4.	36 hours
D. Required Action and associated Completion Time of Condition A or B not met during OPDRVs.	D.1	Initiate action to suspend OPDRVs.	Immediately

3.6 CONTAINMENT SYSTEMS

3.6.4.3 Standby Gas Treatment (SGT) System

LCO 3.6.4.3 Two SGT subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SGT subsystem inoperable.	A.1 Restore SGT subsystem to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, or 3.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours
C. Required Action and associated Completion Time of Condition A not met during OPDRVs.	C.1 Place OPERABLE SGT subsystem in operation. OR C.2 Initiate action to suspend OPDRVs.	Immediately Immediately

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Two SGT subsystems inoperable in MODE 1, 2, or 3.	D.1 Enter LCO 3.0.3.	Immediately
E. Two SGT subsystems inoperable during OPDRVs.	E.1 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.3.1	Operate each SGT subsystem for $\geq$ 10 continuous hours with heaters operating.	31 days
SR 3.6.4.3.2	Perform required SGT filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.6.4.3.3	Verify each SGT subsystem actuates on an actual or simulated initiation signal and reaches $\geq$ 5000 cfm in $\leq$ 2 minutes.	24 months
SR 3.6.4.3.4	Verify each SGT filter cooling recirculation valve can be opened and the fan started.	24 months

5.5 Programs and Manuals (continued)

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the FSAR, Table 3.9-1, Note 1, cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.6 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 pumps and valves.*

a. Testing Frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as follows:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;

(continued)

* The Inservice Testing Program requirement for full stroke exercise testing at each refueling outage for TIP-V-6 shall not be required for the refueling outage conducted in the Spring, 1997. This exception shall expire upon reaching MODE 4 for a plant shutdown of sufficient duration to allow TIP-V-6 testing, or May 15, 1998, whichever occurs first.

5.5 Programs and Manuals

5.5.6 Inservice Testing Program (continued)

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.

5.5.7 Ventilation Filter Testing Program (VFTP)

The VFTP shall establish the required testing of Engineered Safety Feature (ESF) filter ventilation systems.

Tests described in Specification 5.5.7.a and 5.5.7.b shall be performed once per 24 months; after each complete or partial replacement of the HEPA filter train or charcoal adsorber filter; after any structural maintenance on the system housing; and, following significant painting, fire, or chemical release in any ventilation zone communicating with the system while it is in operation.

Tests described in Specification 5.5.7.c shall be performed once per 24 months; after 720 hours of system operation; after any structural maintenance on the system housing; and, following significant painting, fire, or chemical release in any ventilation zone communicating with the system while it is in operation.

Tests described in Specification 5.5.7.d and 5.5.7.e shall be performed once per 24 months.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test Frequencies.

a. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < 0.05% when tested in accordance with Regulatory Guide 1.52, Revision 2, and ASME N510-1989 at the system flowrate specified below:

ESF Ventilation System	Flowrate (cfm)
SGT System	4500 to 5500
CREF System	900 to 1100

(continued)

5.5 Programs and Manuals

5.5.7 Ventilation Filter Testing Program (VFTP) (continued)

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < 0.05% when tested in accordance with Regulatory Guide 1.52, Revision 2, and ASME N510-1989 at the system flowrate specified below:

ESF Ventilation System	Flowrate (cfm)
SGT System	4500 to 5500
CREF System	900 to 1100

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and the relative humidity specified below. Testing of the SGT System will also be conducted at a face velocity of 75 feet per minute.

ESF Ventilation System	Penetration (%)	RH (%)
SGT System	0.5	70
CREF System	2.5	70

Allowed tolerances in the above testing parameters of temperature, relative humidity, and face velocity are as specified in ASTM D3803-1989.

d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters and the charcoal adsorbers is less than the value specified below when tested at the system flowrate specified below:

ESF Ventilation System	Delta P (inches wg)	Flowrate (cfm)
SGT System	< 8	4500 to 5500
CREF System	< 6	900 to 1100

(continued)

5.5 Programs and Manuals

5.5.7 Ventilation Filter Testing Program (VFTP) (continued)

e. Demonstrate that the heaters for each of the ESF systems dissipate the nominal value specified below when tested in accordance with ASME N510-1989:

ESF Ventilation System	Wattage (kW)
SGT System	18.6 to 22.8
CREF System	4.5 to 5.5

5.5.8 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Main Condenser Offgas Treatment System and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks.

The program shall include:

a. The limits for concentrations of hydrogen in the Main Condenser Offgas Treatment System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion); and

b. A surveillance program to ensure that the quantity of radioactivity contained in all outside temporary liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System is less than the amount that would result in concentrations greater than the limits of Appendix B, Table 2, Column 2 to 10 CFR 20.1001 - 20.2402, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program Surveillance Frequencies.

(continued)

LICENSE AMENDMENT REQUEST
ALTERNATIVE SOURCE TERM
Attachment 8

Marked-Up Version of Technical Specification Bases

(continued)

BASES

SAFETY LIMITS (continued)

water level is greater than the top of the active irradiated fuel in order to prevent elevated clad temperatures and resultant clad perforations.

APPLICABILITY

SLs 2.1.1.1, 2.1.1.2, and 2.1.1.3 are applicable in all MODES.

SAFETY LIMIT VIOLATIONS

Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR ~~100, "Reactor Site Criteria,"~~ 50.67 limits (Ref. 6). Therefore, it is required to insert all insertable control rods and restore compliance with the SL within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and the probability of an accident occurring during this period is minimal.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.

2. ANF-1125(P)(A), Revision 0, including Supplements 1 and 2, April 1990.

3. CENPD-392-P-A, "10 x 10 SVEA Critical Power Experiments and CPR Correlations: SVEA-96," September 2000

4. ANF-524(P)(A), Revision 2, including Supplements 1 and 2, November 1990.

5. CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactor Reload Fuel," July 1996.

6. 10 CFR ~~100.~~ 50.67, "Accident Source Term."

7. ANFB Critical Power Correlation Uncertainty for Limited Data Sets, ANF-1125(P)(A), Supplement 1, Appendix D, Siemens Power Corporation - Nuclear Division, July 1998.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The RCS pressure SL has been selected such that it is at a pressure below which it can be shown that the integrity of the system is not endangered. The reactor pressure vessel is designed to ASME, Boiler and Pressure Vessel Code, Section III, 1971 Edition, including Addenda through the summer of 1971 (Ref. 5), which permits a maximum pressure transient of 110%, 1375 psig, of design pressure 1250 psig. The SL of 1325 psig, as measured in the reactor steam dome, is equivalent to 1375 psig at the lowest elevation of the RCS. The RCS is designed to ASME Code, Section III, 1971 Edition, including Addenda through the summer of 1971 (Ref. 5), for the reactor recirculation piping, which permits a maximum pressure transient of 125% of design pressures of 1250 psig for suction piping and 1550 psig for discharge piping. The RCS pressure SL is selected to be the lowest transient overpressure allowed by the applicable codes.

SAFETY LIMITS

The maximum transient pressure allowable in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowable in the RCS piping, valves, and fittings is 125% of design pressures of 1250 psig for suction piping and 1550 psig for discharge piping. The most limiting of these allowances is the 110% of design pressure; therefore, the SL on maximum allowable RCS pressure is established at 1325 psig as measured at the reactor steam dome.

APPLICABILITY

SL 2.1.2 applies in all MODES.

SAFETY LIMIT VIOLATIONS

50.67

Exceeding the RCS pressure SL may cause RCS failure and create a potential for radioactive releases in excess of 10 CFR ~~100, "Reactor Site Criteria,"~~ limits (Ref. 4). Therefore, it is required to insert all insertable control rods and restore compliance with the SL within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and the probability of an accident occurring during this period is minimal.

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50, Appendix A, GDC 14, GDC 15, and GDC 28.
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000.
3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IW-5000.
4. 10 CFR ~~100~~ 50.67.
5. ASME, Boiler and Pressure Vessel Code, 1971 Edition, Addenda, summer of 1971.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Rod Pattern Control

BASES

BACKGROUND

Control rod patterns during startup conditions are controlled by the operator and the rod worth minimizer (RWM) (LCO 3.3.2.1, "Control Rod Block Instrumentation"), so that only specified control rod sequences and relative positions are allowed over the operating range of all control rods inserted to 10% RTP. The sequences effectively limit the potential amount of reactivity addition that could occur in the event of a control rod drop accident (CRDA).

This Specification assures that the control rod patterns are consistent with the assumptions of the CRDA analyses of References 1, 2, and 3.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the CRDA are summarized in References 1, 2, 3, and 4. CRDA analyses assume that the reactor operator follows prescribed withdrawal sequences. These sequences define the potential initial conditions for the CRDA analysis. The RWM (LCO 3.3.2.1) provides backup to operator control of the withdrawal sequences to ensure that the initial conditions of the CRDA analysis are not violated.

Prevention or mitigation of positive reactivity insertion events is necessary to limit the energy deposition in the fuel, thereby preventing significant fuel damage, which could result in undue release of radioactivity. Since the failure consequences for UO_2 have been shown to be insignificant below fuel energy depositions of 300 cal/gm (Ref. 5), the fuel damage limit of 280 cal/gm provides a margin of safety from significant core damage, which would result in release of radioactivity (Refs. 6 and 7). The ~~G~~eneric evaluation~~s~~ (Ref~~s~~. 4 ~~8 and 9~~) of a design basis CRDA (i.e., a CRDA resulting in a peak fuel energy deposition of 280 cal/gm) has ~~have~~ shown that if the peak fuel enthalpy remains below 280 cal/gm, then the maximum reactor pressure will be less than the required ASME Code limits (Ref. 8 ~~10~~) and the calculated offsite doses will be well within the required limits (Ref. 7).

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Control rod patterns analyzed in Reference 1 follow the banked position withdrawal sequence (BPWS) described in Reference ~~11~~ 9. The BPWS is applicable from the condition of all control rods fully inserted to 10% RTP (Ref. 2). For the BPWS, the control rods are required to be moved in groups, with all control rods assigned to a specific group required to be within specified banked positions (e.g., between notches 08 and 12). The banked positions are defined to minimize the maximum incremental control rod worths without being overly restrictive during normal plant operation. The generic BPWS analysis (Ref. ~~11~~ 9) also evaluated the effect of fully inserted, inoperable control rods not in compliance with the sequence, to allow a limited number (i.e., eight) and distribution of fully inserted, inoperable control rods.

Rod pattern control satisfies the requirements of Criterion 3 of Reference ~~12~~ 10.

LCO

Compliance with the prescribed control rod sequences minimizes the potential consequences of a CRDA by limiting the initial conditions to those consistent with the BPWS. This LCO only applies to OPERABLE control rods. For inoperable control rods required to be inserted, separate requirements are specified in LCO 3.1.3, "Control Rod OPERABILITY," consistent with the allowances for inoperable control rods in the BPWS.

APPLICABILITY

In MODES 1 and 2, when THERMAL POWER is ≤ 10% RTP, the CRDA is a Design Basis Accident (DBA) and, therefore, compliance with the assumptions of the safety analysis is required. When THERMAL POWER is > 10% RTP, there is no credible control rod configuration that results in a control rod worth that could exceed the 280 cal/gm fuel damage limit during a CRDA (Ref. 2). In MODES 3, 4, and 5, since the reactor is shut down and only a single control rod can be withdrawn from a core cell containing fuel assemblies, adequate SDM ensures that the consequences of a CRDA are acceptable, since the reactor will remain subcritical with a single control rod withdrawn.

(continued)

BASES

REFERENCES (continued)

5. NUREG-0979, "NRC Safety Evaluation Report for GESSAR II BWR/6 Nuclear Island Design, Docket No. 50-447," Section 4.2.1.3.2, April 1983.

6. NUREG-0800, "Standard Review Plan," Section 15.4.9, "Radiological Consequences of Control Rod Drop Accident (BWR)," Revision 2, July 1981.

10CFR50.67, "Accident Source Term."

7. ~~10 CFR 100.11, "Determination of Exclusion Area Low Population Zone and Population Center Distance."~~

~~8. NEDO-10527, "Rod Drop Accident Analysis for Large BWRs," (including Supplements 1 and 2), March 1972.~~

~~9. NEDO-21778-A, "Transient Pressure Rises Affected Fracture Toughness Requirements for Boiling Water Reactors," December 1978.~~

8 ~~10.~~ ASME, Boiler and Pressure Vessel Code, Section III.

9 ~~11.~~ NEDO-21231, "Banked Position Withdrawal Sequence," January 1977.

10 ~~12.~~ 10 CFR 50.36(c)(2)(ii).

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Standby Liquid Control (SLC) System

BASES

BACKGROUND

The SLC System is designed to provide the capability of bringing the reactor, at any time in a fuel cycle, from full power and minimum control rod inventory (which is at the peak of the xenon transient) to a subcritical condition with the reactor in the most reactive xenon free state without taking credit for control rod movement. The SLC System satisfies the requirements of 10 CFR 50.62 (Ref. 1) on anticipated transient without scram (ATWS).

The SLC System consists of a boron solution storage tank, two positive displacement pumps, two explosive valves, which are provided in parallel for redundancy, and associated piping and valves used to transfer borated water from the storage tank to the reactor pressure vessel (RPV). The borated solution is discharged through the high pressure core spray system sparger.

APPLICABLE SAFETY ANALYSES

It is also used to control pH in the suppression pool.

The SLC System is manually initiated from the main control room, as directed by the emergency operating procedures, if the operator believes the reactor cannot be shut down, or kept shut down, with the control rods. The SLC System is used in the event that not enough control rods can be inserted to accomplish shutdown and cooldown in the normal manner. The SLC System injects borated water into the reactor core to compensate for all of the various reactivity effects that could occur during plant operation. To meet this objective, it is necessary to inject, using both SLC pumps, a quantity of boron that produces a concentration of 660 ppm of natural boron in the reactor core, including recirculation loops, at 70°F and normal reactor water level. To allow for potential leakage and imperfect mixing in the reactor system, an additional amount of boron equal to 25% of the amount cited above is added (Ref. 2). An additional 275 ppm is provided to accommodate dilution in the RPV by the residual heat removal shutdown cooling piping. The temperature versus concentration limits in Figure 3.1.7-1

(continued)

BASES

4t The SLC system is also operated post-LOCA Source term events to maintain suppression pool pH levels to preclude re-evolution of iodine and an additional subsequent release.

APPLICABLE SAFETY ANALYSES (continued)

are calculated such that the required concentration is achieved. This quantity of borated solution is the amount that is above the pump suction shutoff level in the boron solution storage tank. No credit is taken for the portion of the tank volume that cannot be injected.

The SLC System satisfies Criterion 4 of Reference 3.

LCO

The OPERABILITY of the SLC System provides backup capability for reactivity control, independent of normal reactivity control provisions provided by the control rods. The OPERABILITY of the SLC System is based on the conditions of the borated solution in the storage tank and the availability of a flow path to the RPV, including the OPERABILITY of the pumps and valves. Two SLC subsystems are required to be OPERABLE, each containing an OPERABLE pump, an explosive valve and associated piping, valves, and instruments and controls to ensure an OPERABLE flow path.

APPLICABILITY

In MODES 1 and 2, shutdown capability is required. In MODES 3 and 4, control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate controls to ensure the reactor remains subcritical. In MODE 5, only a single control rod can be withdrawn from a core cell containing fuel assemblies. Demonstration of adequate SDM (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") ensures that the reactor will not become critical. Therefore, the SLC System is not required to be OPERABLE during these conditions, when only a single control rod can be withdrawn.

ACTIONS

A.1

If one SLC System subsystem is inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this condition, the remaining OPERABLE subsystem is adequate to perform the original licensing basis shutdown function. However, the overall capability is reduced since the remaining OPERABLE subsystem cannot meet the requirements of Reference 1. The 7 day Completion Time is based on the availability of an OPERABLE subsystem capable

(continued)

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 Scram Discharge Volume (SDV) Vent and Drain Valves

BASES

BACKGROUND

The SDV vent and drain valves are normally open and discharge any accumulated water in the SDV to ensure that sufficient volume is available at all times to allow a complete scram. During a scram, the SDV vent and drain valves close to contain reactor water. The SDV consists of header piping that connects to each hydraulic control unit (HCU) and drains into an instrument volume. There are two headers and two instrument volumes, each receiving approximately one half of the control rod drive (CRD) discharges. The two instrument volumes are connected to a common drain line with two valves in series. Each header is connected to a common vent line with two valves in series. The header piping is sized to receive and contain all the water discharged by the CRDs during a scram. The design and functions of the SDV are described in Reference 1.

APPLICABLE SAFETY ANALYSES

The Design Basis Accident and transient analyses assume all the control rods are capable of scramming. The primary function of the SDV is to limit the amount of reactor coolant discharged during a scram. The acceptance criteria for the SDV vent and drain valves are that they operate automatically to:

a. Close during scram to limit the amount of reactor coolant discharged so that adequate core cooling is maintained and offsite doses remain within the limits of 10 CFR ~~100~~ 50.67 (Ref. 2); and

b. Open on scram reset to maintain the SDV vent and drain path open so there is sufficient volume to accept the reactor coolant discharged during a scram.

Isolation of the SDV can also be accomplished by manual closure of the SDV valves. Additionally, the discharge of reactor coolant to the SDV can be terminated by scram reset or closure of the HCU manual isolation valves. For a bounding leakage case, the offsite doses are well within the limits of 10 CFR ~~100~~ 50.67 (Ref. 2) and adequate core cooling is maintained (Ref. 3). The SDV vent and drain valves also allow continuous drainage of the SDV during normal plant

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.1.8.3 (continued) unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.
REFERENCES	1. FSAR, Section 4.6.1.1.2.4.2.5. 2. 10 CFR ~~100~~. 50.67 3. NUREG-0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping," August 1981. 4. 10 CFR 50.36(c)(2)(ii).

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

BASES

BACKGROUND

The LHGR is a measure of the heat generation rate of a fuel rod in a fuel assembly at any axial location. Limits on the LHGR are specified to ensure that fuel design limits are not exceeded anywhere in the core during normal operation, including anticipated operational occurrences (AOOs). Exceeding the LHGR limit could potentially result in fuel damage and subsequent release of radioactive materials. Fuel design limits are specified to ensure that fuel system damage, fuel rod failure or inability to cool the fuel does not occur during the anticipated operating conditions identified in References 1 and 2.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel system design are presented in References 1, 2, 3, 4, 5, and 6. The fuel assembly is designed to ensure (in conjunction with the core nuclear and thermal hydraulic design, plant equipment, instrumentation, and protection system) that fuel damage will not result in the release of radioactive materials in excess of the guidelines of 10 CFR,
50.67 Parts 20, 50, and ~~100~~. The mechanisms that could cause fuel damage during operational transients and that are considered in fuel evaluations are:

a. Rupture of the fuel rod cladding caused by strain from the relative expansion of the UO_2 pellet; and

b. Severe overheating of the fuel rod cladding caused by inadequate cooling.

A value of 1% plastic strain of the fuel cladding has been defined as the limit below which fuel damage caused by overstraining of the fuel cladding is not expected to occur (Reference 7).

Fuel design evaluations have been performed and demonstrate that the 1% fuel cladding plastic strain design limit is not exceeded during continuous operation with LHGRs up to the operating limit specified in the COLR. The analysis also includes allowances for short term transient operation above the operating limit to account for AOOs.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.d. Reactor Building Vent Exhaust Plenum Radiation—High (continued)

The Reactor Building Vent Exhaust Plenum Radiation—High signals are initiated from radiation detectors that are located in the ventilation exhaust plenum. The signal from each detector is input to an individual monitor whose trip outputs are assigned to an isolation channel. Four channels of Reactor Building Vent Exhaust Plenum Radiation—High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

50.67

The Allowable Values are chosen to ensure offsite doses remain below 10 CFR ~~100~~ limits.

This Function isolates the Group 3 valves.

2.e. Manual Initiation

The Manual Initiation switch and push button channels introduce signals into the primary containment isolation logic that are redundant to the automatic protective instrumentation and provide manual isolation capability. There is no specific FSAR safety analysis that takes credit for this Function. It is retained for overall redundancy and diversity of the isolation function as required by the NRC in the plant licensing basis.

For the Group 3 valves, there are four switch and push buttons (with two channels per switch and push button) for the logic, with two switch and push buttons per trip system. For the Group 2, 4, and 5 valves, there are two switch and push buttons (with two channels per switch and push button) for the logic, one switch and push button per trip system. Eight channels of the Manual Initiation Function are available and are required to be OPERABLE in MODES 1, 2, and 3, since these are the MODES in which the Primary Containment Isolation automatic Functions are required to be OPERABLE.

There is no Allowable Value for this Function since the channels are mechanically actuated based solely on the position of the switch and push buttons.

(continued)

B 3.3 INSTRUMENTATION

B 3.3.6.2 Secondary Containment Isolation Instrumentation

BASES

BACKGROUND

The secondary containment isolation instrumentation automatically initiates closure of appropriate secondary containment isolation valves (SCIVs) and starts the Standby Gas Treatment (SGT) System. The function of these systems, in combination with other accident mitigation systems, is to limit fission product release during and following postulated Design Basis Accidents (DBAs) (Ref~~s~~. 1 ~~and 2~~),

50.67

such that offsite radiation exposures are maintained within the requirements of 10 CFR ~~100~~ that are part of the NRC staff approved licensing basis. Secondary containment isolation and establishment of vacuum with the SGT System within the assumed time limits ensures that fission products that are released during certain operations that take place inside primary containment, when primary containment is not required to be OPERABLE, or that take place outside primary containment are maintained within applicable limits.

The isolation instrumentation includes the sensors, relays, and switches that are necessary to cause initiation of secondary containment isolation. Most channels include electronic equipment (e.g., trip relays) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel outputs a secondary containment isolation signal to the isolation logic. Functional diversity is provided by monitoring a wide range of independent parameters. The input parameters to the isolation logic are (a) reactor vessel water level, (b) drywell pressure, and (c) reactor building vent exhaust plenum radiation. Redundant sensor input signals from each parameter are provided for initiation of isolation parameters. In addition, manual initiation of the logic is provided.

Most Secondary Containment Isolation instrumentation Functions receive input from four channels. The output from these channels are arranged into two two-out-of-two logic trip systems. For the Manual Initiation Function, four channels are required to actuate a trip system (a four-out-of-four logic trip system). In addition to the isolation function, the SGT subsystems are initiated. Each trip system will start one fan in each SGT subsystem, but

(continued)

BASES

BACKGROUND (continued)

will only align one SGT subsystem filter train. Automatically isolated secondary containment penetrations are isolated by two isolation valves. Each trip system initiates isolation of one of the two valves on each penetration so that operation of either trip system isolates the penetrations.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The isolation signals generated by the secondary containment isolation instrumentation are implicitly assumed in the safety analyses of Reference~~s~~ 1 ~~and 2~~ to initiate closure of valves and start the SGT System to limit offsite doses.

Refer to LCO 3.6.4.2, "Secondary Containment Isolation Valves (SCIVs)," and LCO 3.6.4.3, "Standby Gas Treatment (SGT) System," Applicable Safety Analyses Bases for more detail of the safety analyses.

The secondary containment isolation instrumentation satisfies Criterion 3 of Reference ~~3~~ 2. Certain instrumentation Functions are retained for other reasons and are described below in the individual Functions discussion.

The OPERABILITY of the secondary containment isolation instrumentation is dependent upon the OPERABILITY of the individual instrumentation channel Functions. Each Function must have the required number of OPERABLE channels with their setpoints set within the specified Allowable Values, as shown in Table 3.3.6.2-1. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions.

Allowable Values are specified for each Function specified in the Table. Nominal trip setpoints are specified in setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Values between CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2. Drywell Pressure—High (continued)

supports actions to ensure that any offsite releases are within the limits calculated in the safety analysis. However, the Drywell Pressure—High Function associated with isolation is not assumed in any FSAR accident or transient analysis. It is retained for the overall redundancy and diversity of the secondary containment isolation instrumentation as required by the NRC approved licensing basis. High drywell pressure signals are initiated from pressure switches that sense the pressure in the drywell. Four channels of Drywell—High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was chosen to be the same as the RPS Drywell Pressure—High Function Allowable Value (LCO 3.3.1.1) since this is indicative of a loss of coolant accident.

The Drywell Pressure—High Function is required to be OPERABLE in MODES 1, 2, and 3 where considerable energy exists in the RCS; thus, there is a probability of pipe breaks resulting in significant releases of radioactive steam and gas. This Function is not required in MODES 4 and 5 because the probability and consequences of these events are low due to the RCS pressure and temperature limitations of these MODES.

3. Reactor Building Vent Exhaust Plenum Radiation—High

~~NOTE~~

~~Handling a cask/canister loaded with spent fuel, after the canister is seal welded and leak tested, is not considered to be movement of irradiated fuel.~~

High secondary containment exhaust radiation is an indication of possible gross failure of the fuel cladding. The release may have originated from the primary containment due to a break in the RCPB. ~~or the refueling floor due to a fuel handling accident.~~ When Reactor Building Vent Exhaust Plenum Radiation—High is detected, secondary containment

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3. Reactor Building Vent Exhaust Plenum Radiation—High (continued)

isolation and actuation of the SGT System are initiated to limit the release of fission products. ~~as assumed in the FSAR safety analyses (Ref. 2).~~

Secondary containment is not credited for the fuel handling accident.

The Reactor Building Vent Exhaust Plenum Radiation—High signals are initiated from radiation detectors that are located in the ventilation exhaust plenum, which is the collection point of all reactor building and refueling floor air flow prior to its exhaust to atmosphere. The signal from each detector is input to an individual monitor whose trip outputs are assigned to an isolation channel. Four channels of Reactor Building Vent Exhaust Plenum Radiation—High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value is chosen to promptly detect gross failure of the fuel cladding.

The Reactor Building Vent Plenum Exhaust Radiation—High Function is required to be OPERABLE in MODES 1, 2, and 3 where considerable energy exists; thus, there is a probability of pipe breaks resulting in significant releases of radioactive steam and gas. In MODES 4 and 5, the probability and consequences of these events are low due to the RCS pressure and temperature limitations of these MODES; thus, this Function is not required. In addition, the Function is required to be OPERABLE during ~~CORE ALTERATIONS, OPDRVs, and movement of irradiated fuel assemblies in the secondary containment~~ because the capability of detecting radiation releases due to fuel failures (due to fuel uncovery ~~or dropped fuel assemblies~~) must be provided to ensure that offsite dose limits are not exceeded.

4. Manual Initiation

The Manual Initiation switch and push button channels introduce signals into the secondary containment isolation logic that are redundant to the automatic protective instrumentation channels, and provide manual isolation capability. There is no specific FSAR safety analysis that

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

4. Manual Initiation (continued)

takes credit for this Function. It is retained for the overall redundancy and diversity of the secondary containment isolation instrumentation as required by the NRC approved licensing basis.

There are four switch and push buttons (with two channels per switch and push button) for the logic, two switch and push buttons per trip system. Eight channels of the Manual Initiation Function are available and are required to be OPERABLE in MODES 1, 2, and 3 and during ~~CORE ALTERATIONS, OPDRVs, and movement of irradiated fuel assemblies in the secondary containment,~~ since these are the MODES and other specified conditions in which the Secondary Containment Isolation automatic Functions are required to be OPERABLE.

There is no Allowable Value for this Function since the channels are mechanically actuated based solely on the position of the switch and push buttons.

ACTIONS

A Note has been provided to modify the ACTIONS related to secondary containment isolation instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition discovered to be inoperable or not within limits will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable secondary containment isolation instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable secondary containment isolation instrumentation channel.

A.1

Because of the diversity of sensors available to provide isolation signals and the redundancy of the isolation design, an allowable out of service time of 12 hours or

(continued)

BASES

ACTIONS

A.1 (continued)

24 hours, depending on the Function (12 hours for those Functions that have channel components common to RPS instrumentation and 24 hours for those Functions that do not have channel components common to RPS instrumentation), has been shown to be acceptable (Refs. ~~4 and 5~~ 3and4) to permit restoration of any inoperable channel to OPERABLE status. This out of service time is only acceptable provided the associated Function is still maintaining isolation capability (refer to Required Action B.1 Bases). If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action A.1. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an isolation), Condition C must be entered and its Required Actions taken.

B.1

Required Action B.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in a complete loss of automatic isolation capability for the associated penetration flow path(s) or a complete loss of automatic initiation capability for the SGT System. A Function is considered to be maintaining isolation capability when sufficient channels are OPERABLE or in trip, such that one trip system will generate a trip signal from the given Function on a valid signal. This ensures that one of the two SCIVs in the associated penetration flow path and one SGT subsystem can be initiated on an isolation signal from the given Function. For the Functions with two two-out-of-two logic trip systems (Functions 1, 2, and 3), this would require one trip system to have two channels, each OPERABLE or in trip. The Condition does not include the Manual Initiation Function (Function 4), since it is not assumed in any accident or transient analysis. Thus, a total loss of manual initiation capability for 24 hours (as allowed by Required Action A.1) is allowed.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Surveillances are also modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours, provided the associated Function maintains isolation capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Action(s) taken. 3 and 4

This Note is based on the reliability analysis (Refs. ~~4 and 5~~) assumption of the average time required to perform channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the SCIVs will isolate the associated penetration flow paths and the SGT System will initiate when necessary.

SR 3.3.6.2.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the indicated parameter for one instrument channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.6.2.4 (continued)

The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency.

REFERENCES

1. FSAR, Sections 15.6.5 and 15.F.6.

~~2. FSAR, Section 15.7.4.~~

2 ~~3.~~ 10 CFR 50.36(c)(2)(ii).

3 ~~4.~~ NEDO-31677-P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," July 1990.

4 ~~5.~~ NEDC-30851-P-A, Supplement 2, "Technical Specifications Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," March 1989.

BASES

BACKGROUND (continued)	signal to the initiation logic. The Main Control Room Ventilation Radiation Monitors only provide alarm and indication. The radiation monitors also include electronic equipment that compares measured input signals to pre-established setpoints. When the setpoint is exceeded, the radiation monitors output relay actuates, which then outputs to an alarm in the control room.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The ability of the CREF System to maintain the habitability of the MCR is explicitly assumed for certain accidents as discussed in the FSAR safety analyses (Refs. 2 and 3). CREF System operation ensures that the radiation exposure of control room personnel, through the duration of any one of the postulated accidents, does not exceed the limits set by GDC 19 of 10 CFR 50, Appendix A. and 10 CFR 50.67.

CREF instrumentation satisfies Criterion 3 of Reference 4.

The OPERABILITY of the CREF System instrumentation is dependent upon the OPERABILITY of the individual instrumentation channel Functions specified in Table 3.3.7.1-1. Each Function must have a required number of OPERABLE channels, with their setpoints within the specified Allowable Values, where appropriate. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions.

Allowable Values are specified for each CREF System Function specified in the Table. Nominal trip setpoints are specified in the setpoint calculations. These nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint that is less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip relay) changes state. The analytic limits are derived from the limiting values of the process

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.8 RCS Specific Activity

BASES

BACKGROUND

During circulation, the reactor coolant acquires radioactive materials due to release of fission products from fuel leaks into the coolant and activation of corrosion products in the reactor coolant. These radioactive materials in the coolant can plate out in the RCS, and, at times, an accumulation will break away to spike the normal level of radioactivity. The release of coolant during a Design Basis Accident (DBA) could send radioactive materials into the environment.

Limits on the maximum allowable level of radioactivity in the reactor coolant are established to ensure, in the event of a release of any radioactive material to the environment during a DBA, radiation doses are maintained within the limits of 10 CFR ~~100~~ 50.67 (Ref. 1).

This LCO contains iodine specific activity limits. The iodine isotopic activities per gram of reactor coolant are expressed in terms of a DOSE EQUIVALENT I-131. The allowable levels are intended to limit the 2 hour radiation dose to an individual at the site boundary to a small fraction of the 10 CFR ~~100~~ 50.67 limit.

APPLICABLE SAFETY ANALYSES

Analytical methods and assumptions involving radioactive material in the primary coolant are presented in the FSAR (Ref. 2). The specific activity in the reactor coolant (the source term) is an initial condition for evaluation of the consequences of an accident due to a main steam line break (MSLB) outside containment. No fuel damage is postulated in the MSLB accident, and the release of radioactive material to the environment is assumed to end when the main steam isolation valves (MSIVs) close completely.

This MSLB release forms the basis for determining offsite doses (Ref. 2). The limits on the specific activity of the primary coolant ensure that the 2 hour thyroid and whole body doses at the site boundary, resulting from an MSLB outside containment during steady state operation, will not exceed ~~10% of~~ the dose guidelines of 10 CFR ~~100~~ 50.67

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The limit on specific activity is a value from a parametric evaluation of typical site locations. This limit is conservative because the evaluation considered more restrictive parameters than for a specific site, such as the location of the site boundary and the meteorological conditions of the site.

RCS specific activity satisfies Criterion 2 of Reference 3.

LCO

The specific iodine activity is limited to ≤ 0.2 μCi/gm DOSE EQUIVALENT I-131. This limit ensures the source term assumed in the safety analysis for the MSLB is not exceeded, so any release of radioactivity to the environment during an MSLB is less than a small fraction of the 10 CFR ~~100~~ 50.67 limits.

APPLICABILITY

In MODE 1, and MODES 2 and 3 with any main steam line not isolated, limits on the primary coolant radioactivity are applicable since there is an escape path for release of radioactive material from the primary coolant to the environment in the event of an MSLB outside of primary containment.

In MODES 2 and 3 with the main steam lines isolated, such limits do not apply since an escape path does not exist. In MODES 4 and 5, no limits are required since the reactor is not pressurized and the potential for leakage is reduced.

ACTIONS

A.1 and A.2

When the reactor coolant specific activity exceeds the LCO DOSE EQUIVALENT I-131 limit, but is ≤ 4.0 μCi/gm, samples must be analyzed for DOSE EQUIVALENT I-131 at least once every 4 hours. In addition, the specific activity must be restored to the LCO limit within 48 hours. The Completion Time of once every 4 hours is based on the time needed to take and analyze a sample. The 48 hour Completion Time to restore the activity level provides a reasonable time for temporary coolant activity increases (iodine spikes or crud bursts) to be cleaned up with the normal processing systems.

A Note to the Required Actions of Condition A excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE(S) while relying on the

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

ACTIONS even though the ACTIONS may eventually require plant shutdown. This exception is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the plant remains at, or proceeds to power operation.

B.1, B.2.1, B.2.2.1, and B.2.2.2

If the DOSE EQUIVALENT I-131 cannot be restored to ≤ 0.2 μCi/gm within 48 hours, or if at any time it is > 4.0 μCi/gm, it must be determined at least every 4 hours and all the main steam lines must be isolated within 12 hours. Isolating the main steam lines precludes the possibility of releasing radioactive material to the environment in an amount that is more than a small fraction of the requirements of 10 CFR ~~100~~ 50.67 during a postulated MSLB accident.

Alternately, the plant can be brought to MODE 3 within 12 hours and to MODE 4 within 36 hours. This option is provided for those instances when isolation of main steam lines is not desired (e.g., due to the decay heat loads). In MODE 4, the requirements of the LCO are no longer applicable.

The Completion Time of once every 4 hours is the time needed to take and analyze a sample. The 12 hour Completion Time is reasonable, based on operating experience, to isolate the main steam lines in an orderly manner and without challenging plant systems. Also, the allowed Completion Times for Required Actions B.2.2.1 and B.2.2.2 for bringing the plant to MODES 3 and 4 are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1

This Surveillance is performed to ensure iodine remains within limit during normal operation. The 7 day Frequency is adequate to trend changes in the iodine activity level. This SR is modified by a Note that requires this Surveillance to be performed only in MODE 1 because the level of fission products generated in other MODES is much less.

REFERENCES

1. 10 CFR ~~100.11~~. 50.67
2. FSAR, Section 15.6.4.
3. 10 CFR 50.36(c)(2)(ii).

BASES

BACKGROUND (continued)

This Specification ensures that the performance of the primary containment, in the event of a DBA, meets the assumptions used in the safety analyses of References 1 and 2. SR 3.6.1.1.1 leakage rate requirements are in conformance with 10 CFR 50, Appendix J, Option B (Ref. 3), as modified by approved exemptions.

APPLICABLE SAFETY ANALYSES

The safety design basis for the primary containment is that it must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate.

~~The DBA that postulates the maximum release of radioactive material within primary containment is a double-ended recirculation suction line break LOCA.~~ In the analysis, ~~of this accident,~~ it is assumed that primary containment is OPERABLE such that release of fission products to the environment is controlled by the rate of primary containment leakage.

The bounding radiological analysis for the LOCA is an inadequate core cooling accident that degrades to core damage. The specific event could be a recirculation line break, a main steam line break inside containment, an intact vessel event, or a spectrum of LOCA transients.

Analytical methods and assumptions involving the primary containment are presented in References 1 and 2. The safety analyses assume a nonmechanistic fission product release following a DBA, which forms the basis for determination of offsite doses. The fission product release is, in turn, based on an assumed leakage rate from the primary containment. OPERABILITY of the primary containment ensures that the leakage rate assumed in the safety analyses is not exceeded.

The maximum allowable leakage rate for the primary containment (L_a) is 0.5% by weight of the containment air per 24 hours at the design basis LOCA maximum peak containment pressure (P_a) of 38 psig (Ref. 4).

Primary containment satisfies Criterion 3 of Reference 5.

LCO

Primary containment OPERABILITY is maintained by limiting leakage to $\leq$ 1.0 L_a, except prior to the first startup after performing a required Primary Containment Leakage Rate Testing Program leakage test. At this time, applicable leakage limits must be met. In addition, the leakage from the drywell to the suppression chamber must be limited to ensure the pressure suppression function is accomplished and

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.5 Residual Heat Removal (RHR) Drywell Spray

BASES

BACKGROUND

The primary containment is designed with a suppression pool so that, in the event of a loss of coolant accident (LOCA), steam released from the primary system is channeled through the suppression pool water and condensed without producing significant pressurization of the primary containment. The primary containment is designed so that with the pool initially at the minimum water volume and the worst single failure of the primary containment heat removal systems, suppression pool energy absorption combined with subsequent operator controlled pool cooling will prevent the primary containment pressure from exceeding its design value. However, the primary containment must also withstand a postulated bypass leakage pathway that allows the passage of steam from the drywell directly into the suppression pool airspace, bypassing the suppression pool. The RHR Drywell Spray System is designed to mitigate the effects of bypass leakage.

It is also operated post-LOCA to wash inorganic iodines and particulates from the drywell atmosphere into the suppression pool.

There are two redundant, 100% capacity RHR drywell spray subsystems. Each subsystem consists of a suction line from the suppression pool, an RHR pump, and one spray sparger inside the drywell. Dispersion of the spray water is accomplished by spray nozzles in each subsystem.

The RHR drywell spray mode will be manually initiated, if required, following a LOCA, according to emergency procedures.

APPLICABLE SAFETY ANALYSES

Reference 1 contains the results of analyses that predict the primary containment pressure response for a LOCA with the maximum allowable bypass leakage area.

The equivalent flow path area for bypass leakage has been specified to be 0.05 ft^2. The analysis demonstrates that with drywell spray operation the primary containment pressure remains within design limits.

The drywell spray is credited for removal of inorganic iodine and particulates and for primary containment pressure reduction during the LOCA source term event.

The RHR drywell spray satisfies Criterion 3 of Reference 2.

(continued)

DELETE ENTIRE BASES SECTION B 3.6.1.8

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.8 Main Steam Isolation Valve Leakage Control (MSLC) System

BASES

BACKGROUND	The MSLC System supplements the isolation function of the MSIVs by processing the fission products that could leak through the closed MSIVs after a Design Basis Accident (DBA) loss of coolant accident (LOCA). The MSLC System consists of two independent subsystems: an inboard subsystem, which is connected between the inboard and outboard MSIVs; and an outboard subsystem, which is connected to the main steam drain line header immediately downstream of the outboard MSIVs. Each subsystem is capable of processing leakage from MSIVs following a DBA LOCA. Each subsystem consists of a blower, valves, and piping. The inboard subsystem is also provided with four electric heaters to boil off any condensate prior to the gas mixture passing through the flow limiter. Each subsystem operates in two process modes: depressurization and bleedoff. The depressurization process reduces the steam line pressure to within the operating capability of equipment used for the bleedoff mode. The effluent is discharged to the reactor building, which encloses a volume served by the Standby Gas Treatment (SGT) System. During bleedoff (long term leakage control), the blowers maintain a negative pressure in the main steam lines (Ref. 1). This ensures that leakage through the closed MSIVs is collected by the MSLC System. In this process mode, the effluent is discharged directly to the SGT System. The MSLC System is manually initiated, and is not required to be initiated until the pressure of the steam trapped between the MSIVs decreases to the reactor steam dome pressure. The pressure requirement is estimated to take at least 1 hour (Ref. 1).
APPLICABLE SAFETY ANALYSES	The MSLC System mitigates the consequences of a DBA LOCA by ensuring that fission products that may leak from the closed MSIVs are filtered by the SGT System (Ref. 2). The analyses

(continued)

DELETE ENTIRE BASES SECTION B 3.6.1.8

BASES

APPLICABLE SAFETY ANALYSES (continued)

in Reference 3 provide the evaluation of offsite dose consequences. The operation of the MSLC System prevents a release of untreated leakage for this type of event.

The MSLC System satisfies Criterion 3 of Reference 4.

LCO

One MSLC subsystem can provide the required processing of the MSIV leakage. To ensure that this capability is available, assuming worst case single failure, two MSLC subsystems must be OPERABLE.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could lead to a fission product release. Therefore, MSLC System OPERABILITY is required during these MODES. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the MSLC System OPERABLE is not required in MODE 4 or 5 to ensure MSIV leakage is processed.

ACTIONS

A.1

With one MSLC subsystem inoperable, the inoperable MSLC subsystem must be restored to OPERABLE status within 30 days. In this condition, the remaining OPERABLE MSLC subsystem is adequate to perform the required leakage control function. However, the overall reliability is reduced because a single failure in the remaining subsystem could result in a total loss of MSIV leakage control function. The 30 day Completion Time is based on the redundant capability afforded by the remaining OPERABLE MSLC subsystem and the low probability of a DBA LOCA occurring during this period.

B.1

With two MSLC subsystems inoperable, at least one subsystem must be restored to OPERABLE status within 7 days. The 7 day Completion Time is based on the low probability of the occurrence of a DBA LOCA.

(continued)

DELETE ENTIRE BASES SECTION B 3.6.1.8

ACTIONS (continued)

C.1 and C.2

If the MSLC subsystem cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.8.1

Each MSLC System blower is operated for $\geq$ 15 minutes to verify OPERABILITY. The 31 day Frequency was developed considering the known reliability of the MSLC System blower and controls, the two subsystem redundancy, and the low probability of a significant degradation of the MSLC subsystem occurring between Surveillances and has been shown to be acceptable through operating experience.

SR 3.6.1.8.2

The electrical continuity of each inboard MSLC subsystem heater is verified by a resistance check, by verifying the rate of temperature increase meets specifications, or by verifying the current or wattage draw meets specifications. The 31 day Frequency is based on operating experience that has shown that these components usually pass this Surveillance when performed at this Frequency.

SR 3.6.1.8.3

A system functional test is performed to ensure that the MSLC System will operate through its operating sequence. This includes verifying that the automatic positioning of the valves and the operation of each interlock and timer are correct, that the blowers start and develop a flow rate of $\geq$ 24 cfm and $\leq$ 36 cfm, at a vacuum of $\geq$ 17 inches water gauge, and the upstream heaters meet current or wattage draw

(continued)

DELETE ENTIRE BASES SECTION B 3.6.1.8

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.8.3 (continued)

requirements. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section 6.7.3.
2. FSAR, Section 6.7.2.1.
3. FSAR, Sections 15.6.5 and 15.F.6.
4. 10 CFR 50.36(c)(2)(ii).

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.1 Secondary Containment

BASES

BACKGROUND

The function of the secondary containment is to contain, dilute, and hold up fission products that may leak from primary containment following a Design Basis Accident (DBA). In conjunction with operation of the Standby Gas Treatment (SGT) System and closure of certain valves whose lines penetrate the secondary containment, the secondary containment is designed to reduce the activity level of the fission products prior to release to the environment and to isolate and contain fission products that are released during certain operations that take place inside primary containment, when primary containment is not required to be OPERABLE, or that take place outside primary containment.

The secondary containment is a structure that completely encloses the primary containment and those components that may be postulated to contain primary system fluid. This structure forms a control volume that serves to hold up and dilute the fission products. It is possible for the pressure in the control volume to rise relative to the environmental pressure (e.g., due to pump/motor heat load additions). To prevent ground level exfiltration while allowing the secondary containment to be designed as a conventional structure, the secondary containment requires support systems to maintain the control volume pressure at less than the external pressure. Requirements for these systems are specified separately in LCO 3.6.4.2, "Secondary Containment Isolation Valves (SCIVs)," and LCO 3.6.4.3, "Standby Gas Treatment (SGT) System."

APPLICABLE SAFETY ANALYSES

The~~re are two principal~~ accident~~s~~ for which credit is taken for secondary containment OPERABILITY is the ~~. These are a~~ loss of coolant accident (LOCA) (Ref. 1). ~~and a fuel handling accident (Ref. 2).~~ The secondary containment performs no active function in response to ~~each of these~~ this limiting event~~s~~; however, its leak tightness is required to ensure that the release of radioactive materials from the primary containment is restricted to those leakage paths and

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

associated leakage rates assumed in the accident analysis, and that fission products entrapped within the secondary containment structure will be treated by the SGT System prior to discharge to the environment.

Secondary containment satisfies Criterion 3 of Reference ~~4~~ 2

LCO

An OPERABLE secondary containment provides a control volume into which fission products that bypass or leak from primary containment, or are released from the reactor coolant pressure boundary components located in secondary containment, can be diluted and processed prior to release to the environment. For the secondary containment to be considered OPERABLE, it must have adequate leak tightness to ensure that the required vacuum can be established and maintained.

INSERT A

APPLICABILITY

~~-----NOTE-----~~
~~Handling a cask/canister loaded with spent fuel, after the canister is seal welded and leak tested, is not considered to be movement of irradiated fuel.~~
~~----------~~

In MODES 1, 2, and 3, a LOCA could lead to a fission product release to primary containment that leaks to secondary containment. Therefore, secondary containment OPERABILITY is required during the same operating conditions that require primary containment OPERABILITY.

In MODES 4 and 5, the probability and consequences of the LOCA are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining secondary containment OPERABLE is not required in MODE 4 or 5 to ensure a control volume, except for other situations for which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs)~~, during CORE ALTERATIONS, or during movement of irradiated fuel assemblies in the secondary containment~~.

(continued)

Secondary containment is not credited in the analysis of the fuel handling accident.

INSERT A [B 3.6.4.1 -- LCO]

In addition, secondary containment must be maintained at a vacuum during normal operation to ensure secondary containment effluent is monitored. In the event that the normal secondary containment ventilation system is secured, secondary containment could become pressurized such that the maximum accident design basis pressure of $\geq$ 0.25-inch vacuum water gauge on all surfaces of secondary containment is exceeded. Therefore, operating the SGT System when the normal secondary containment ventilation system is secured provides assurance that secondary containment is operable.

BASES (continued)

ACTIONS

A.1

If secondary containment is inoperable, it must be restored to OPERABLE status within 4 hours. The 4 hour Completion Time provides a period of time to correct the problem that is commensurate with the importance of maintaining secondary containment during MODES 1, 2, and 3. This time period ensures that the probability of an accident (requiring secondary containment OPERABILITY) occurring during periods where secondary containment is inoperable is minimal.

B.1 and B.2

If the secondary containment cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1~~, C.2, and C.3~~

~~Movement of irradiated fuel assemblies in the secondary containment, CORE ALTERATIONS, and~~ OPDRVs can be postulated to cause fission product release to the secondary containment. In such cases, the secondary containment is the only barrier to release of fission products to the environment. ~~CORE ALTERATIONS and movement of irradiated fuel assemblies must be immediately suspended if the secondary containment is inoperable.~~

~~Suspension of these activities shall not preclude completing an action that involves moving a component to a safe position. Also,~~ action must be immediately initiated to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

(continued)

BASES

ACTIONS

~~C.1, C.2, and C.3 (continued)~~

~~LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since irradiated fuel assembly movement can occur in MODE 1, 2, or 3, Required Action C.1 has been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.~~

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1.1

This SR ensures that the secondary containment boundary is sufficiently leak tight to preclude exfiltration under ~~expected wind~~ normal operating conditions. The 24 hour Frequency of this SR was developed based on operating experience related to secondary containment vacuum variations during the applicable MODES and the low probability of a DBA occurring between surveillances.

Furthermore, the 24 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal secondary containment vacuum condition.

SR 3.6.4.1.2 and SR 3.6.4.1.3

Verifying that secondary containment equipment hatches and each inner access door or each outer access door in each access opening are closed ensures that the infiltration of outside air of such a magnitude as to prevent maintaining the desired negative pressure does not occur. Verifying that all such openings are closed provides adequate assurance that exfiltration from the secondary containment will not occur. SR 3.6.4.1.2 also requires equipment hatches to be sealed. In this application, the term "sealed" has no connotation of leak tightness. Maintaining

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1.2 and SR 3.6.4.1.3 (continued)

secondary containment OPERABILITY requires verifying all inner doors or all outer doors in the access opening are closed. However, each secondary containment access door is normally kept closed, except when the access opening is being used for entry and exit or when maintenance is being performed on an access. The 31 day Frequency for these SRs has been shown to be adequate based on operating experience, and is considered adequate in view of the other indications of door and hatch status that are available to the operator.

SR 3.6.4.1.4 ~~and SR 3.6.4.1.5~~

The SGT System exhausts the secondary containment atmosphere to the environment through appropriate treatment equipment. To ensure that all fission products are treated, SR 3.6.4.1.4 verifies that the SGT System will ~~rapidly~~ establish and maintain a pressure in the secondary containment that is less than the lowest postulated pressure external to the secondary containment boundary. ~~This is confirmed by demonstrating that one SGT subsystem will draw down the secondary containment to ≥ 0.25 inches of vacuum water gauge in ≤ 120 seconds.~~ This cannot be accomplished if the secondary containment boundary is not intact. SR ~~3.6.4.1.5~~ 3.6.4.1.4 demonstrates that each SGT subsystem can maintain ≥ 0.25 inches of vacuum water gauge for 1 hour at ~~a~~ an inleakage flow rate ≤ 2240 cfm. The 1 hour test period allows secondary containment to be in thermal equilibrium at steady state conditions. INSERT B ~~Therefore, these two tests are used to ensure secondary containment boundary integrity.~~ Since this test is a ~~these SRs are~~ secondary containment test~~s~~, ~~they~~ it need not be performed with each SGT subsystem. The SGT subsystems are tested on a STAGGERED TEST BASIS, however, to ensure that in addition to the requirements of LCO 3.6.4.3, ~~either~~ any SGT subsystem ~~will~~ can perform this test. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

(continued)

INSERT B [B 3.6.4.1 -- SR 3.6.4.1.4]

Surveillance Requirement SR 3.6.4.1.4 requires secondary containment to be drawn down to at least 0.25 inch of vacuum water gauge at a secondary containment inleakage flow rate not to exceed 2240 cfm. This surveillance requirement verifies secondary containment integrity by ensuring that secondary containment in-leakage does not exceed a value that could prevent acceptable drawdown. The time to establish the required vacuum in secondary containment is a function of SGT System flow rate and secondary containment inleakage, in addition to outside meteorological conditions, initial secondary containment temperature and humidity, and Standby Service Water System temperature.

This surveillance requirement can be satisfied by verifying that the differential pressure is at a greater vacuum (larger negative pressure) than 0.25 inch of vacuum water gauge when measured at any standby gas treatment flow rate greater than 2240 cfm. For this surveillance, the differential pressure must be corrected for the effects of outside wind and temperature.

Based on analysis, the secondary containment differential pressure associated with standby gas treatment flow rate is a combination of linear and quadratic in-leakage relationships. The differential pressure is primarily a function of the seams in the secondary containment superstructure and leakage through doorways.

This surveillance requirement, in conjunction with SR 3.6.4.3.3 that verifies the operability of the SGT System to achieve 5000 cfm within 2 minutes after receipt of an initiation signal, demonstrates that the SGT subsystem can drawdown secondary containment within the assumptions of the applicable safety analysis.

BASES (continued)

REFERENCES

1. FSAR, Sections 15.6.5 and 15.F.6.
2. ~~FSAR, Section 15.7.4.~~
3. 2 10 CFR 50.36(c)(2)(ii).

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.2 Secondary Containment Isolation Valves (SCIVs)

BASES

BACKGROUND

The function of the SCIVs, in combination with other accident mitigation systems, is to limit fission product release during and following postulated Design Basis Accidents (DBAs) (Ref~~s~~. 1 ~~and 2~~). Secondary containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that fission products that leak from primary containment following a DBA, that are released during certain operations when primary containment is not required to be OPERABLE, or that take place outside primary containment, are maintained within the secondary containment boundary.

The OPERABILITY requirements for SCIVs help ensure that an adequate secondary containment boundary is maintained during and after an accident by minimizing potential paths to the environment. These isolation devices are either passive or active (automatic). Manual valves, de-activated automatic valves secured in their closed position (including check valves with flow through the valve secured), and blind flanges are considered passive devices. Isolation barrier(s) for the penetration are discussed in Reference ~~3~~ 2.

Automatic SCIVs close on a secondary containment isolation signal to establish a boundary for untreated radioactive material within secondary containment following a DBA or other accidents.

Other penetrations are isolated by the use of valves in the closed position or blind flanges.

APPLICABLE SAFETY ANALYSES

The SCIVs must be OPERABLE to ensure the secondary containment barrier to fission product releases is established. The ~~principal~~ accident~~s~~ for which the secondary containment boundary is required ~~are~~ is a loss of coolant accident (Ref. 1)~~, and a fuel handling accident (Ref. 2)~~. The secondary containment performs no active function in response to ~~each of these~~ this limiting event~~s~~, but

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

the boundary established by SCIVs is required to ensure that leakage from the primary containment is processed by the Standby Gas Treatment (SGT) System before being released to the environment.

Maintaining SCIVs OPERABLE with isolation times within limits ensures that fission products will remain trapped inside secondary containment so that they can be treated by the SGT System prior to discharge to the environment.

SCIVs satisfy Criterion 3 of Reference ~~4~~ 3.

LCO

SCIVs form a part of the secondary containment boundary. The SCIV safety function is related to control of offsite radiation releases resulting from DBAs.

The automatic power operated isolation valves are considered OPERABLE when their isolation times are within limits and the valves actuate on an automatic isolation signal. The valves covered by this LCO, along with their associated stroke times, are listed in Reference ~~8~~ 4.

The normally closed isolation valves or blind flanges are considered OPERABLE when manual valves are closed or open in accordance with appropriate administrative controls, automatic SCIVs are de-activated and secured in their closed position, and blind flanges are in place. These passive isolation valves or devices are listed in Reference ~~8~~ 4.

APPLICABILITY

~~NOTE~~
~~Handling a cask/canister loaded with spent fuel, after the canister is seal welded and leak tested, is not considered to be movement of irradiated fuel.~~

In MODES 1, 2, and 3, a DBA could lead to a fission product release to the primary containment that leaks to the secondary containment. Therefore, OPERABILITY of SCIVs is required.

In MODES 4 and 5, the probability and consequences of these events are reduced due to pressure and temperature limitations in these MODES. Therefore, maintaining SCIVs

(continued)

BASES

APPLICABILITY (continued)

OPERABLE is not required in MODE 4 or 5, except for other situations under which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs). ~~during CORE ALTERATIONS, or during movement of irradiated fuel assemblies in the secondary containment.~~

ACTIONS

The ACTIONS are modified by three Notes. The first Note allows penetration flow paths to be unisolated intermittently under administrative controls. These controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the isolation device. In this way, the penetration can be rapidly isolated when the need for secondary containment isolation is indicated.

The second Note provides clarification that, for the purpose of this LCO, separate Condition entry is allowed for each penetration flow path. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable SCIV. Complying with the Required Actions may allow for continued operation, and subsequent inoperable SCIVs are governed by subsequent Condition entry and application of associated Required Actions.

The third Note ensures appropriate remedial actions are taken, if necessary, if the affected system(s) are rendered inoperable by an inoperable SCIV.

<u>A.1 and A.2</u>

In the event that there are one or more penetration flow paths with one SCIV inoperable, the affected penetration flow path(s) must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criteria are a closed and de-activated automatic SCIV, a closed manual valve, and a blind flange. For penetrations isolated in accordance with Required Action A.1, the device used to isolate the penetration should be the closest available device to

(continued)

BASES

ACTIONS

B.1 (continued)

considering the time required to isolate the penetration and the low probability of a DBA, which requires the SCIVs to close, occurring during this short time.

The Condition has been modified by a Note stating that Condition B is only applicable to penetration flow paths with two isolation valves. This clarifies that only Condition A is entered if one SCIV is inoperable in each of two penetrations.

C.1 and C.2

If any Required Action and associated Completion Time cannot be met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

D.1~~, D.2, and D.3~~

If any Required Action and associated Completion Time cannot be met, the plant must be placed in a condition in which the LCO does not apply. ~~If applicable, CORE ALTERATIONS and the movement of irradiated fuel assemblies in the secondary containment must be immediately suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also,~~ if applicable, action must be immediately initiated to suspend OPDRVs in order to minimize the probability of a vessel draindown and the subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

~~LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since irradiated fuel assembly movement can occur in MODE 1, 2, or 3, Required Action D.1 has been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3~~

(continued)

BASES

ACTIONS

~~D.1, D.2, and D.3 (continued)~~

~~would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.~~

SURVEILLANCE REQUIREMENTS

SR 3.6.4.2.1

This SR verifies each secondary containment isolation manual valve and blind flange that is required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the secondary containment boundary is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification that those SCIVs in secondary containment that are capable of being mispositioned are in the correct position.

Since these SCIVs are readily accessible to personnel during normal unit operation and verification of their position is relatively easy, the 31 day Frequency was chosen to provide added assurance that the SCIVs are in the correct positions.

Two Notes have been added to this SR. The first Note applies to valves and blind flanges located in high radiation areas and allows them to be verified by use of administrative controls. Allowing verification by administrative controls is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, and 3 for ALARA reasons. Therefore, the probability of misalignment of these isolation devices, once they have been verified to be in the proper position, is low.

A second Note has been included to clarify that SCIVs that are open under administrative controls are not required to meet the SR during the time the SCIVs are open. These controls consist of stationing a dedicated operator at the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.4.2.1 (continued)

controls of the valve, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for secondary containment isolation is indicated.

SR 3.6.4.2.2

Verifying the isolation time of each power operated and each automatic SCIV listed in Licensee Controlled Specification Table 1.6.4.2-1 is within limits is required to demonstrate OPERABILITY. The isolation time test ensures that the SCIV will isolate in a time period less than or equal to that assumed in the safety analyses. The Frequency of this SR is in accordance with the Inservice Testing Program.

SR 3.6.4.2.3

Verifying that each automatic SCIV closes on a secondary containment isolation signal is required to prevent leakage of radioactive material from secondary containment following a DBA or other accidents. This SR ensures that each automatic SCIV will actuate to the isolation position on a secondary containment isolation signal. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.6.2, "Secondary Containment Isolation Instrumentation," overlaps this SR to provide complete testing of the safety function. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Sections 15.6.5 and 15.F.6.

~~2. FSAR, Section 15.7.4.~~

(continued)

BASES

REFERENCES (continued)	2 ~~3.~~	FSAR, Section 6.2.3.2.
	3 ~~4.~~	10 CFR 50.36(c)(2)(ii).
	4 ~~5.~~	Licensee Controlled Specifications Manual.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.3 Standby Gas Treatment (SGT) System

BASES

BACKGROUND

The active system components in each filter train that are required for post-accident operation are redundant and configured in two subsystems to allow for lead/lag operation. The lead subsystem of each train is powered from a separate emergency diesel divisional bus (division 1 or 2) than the lag subsystem.

The SGT System is required by 10 CFR 50, Appendix A, GDC 41, "Containment Atmosphere Cleanup" (Ref. 1). The function of the SGT System is to ensure that radioactive materials that leak from the primary containment into the secondary containment following a Design Basis Accident (DBA) are filtered and adsorbed prior to exhausting to the environment.

The SGT System consists of two fully redundant filter trains, ~~subsystems~~, each with its own filter unit, ~~set of~~ ductwork, dampers, ~~charcoal filter train,~~ and controls.

Each ~~charcoal~~ filter train consists of (components listed in order of the direction of the air flow):

a. A moisture separator;

b. Two electric heater banks (one lead ~~primary~~ and one lag ~~backup~~);

c. A prefilter bank;

d. A high efficiency particulate air (HEPA) filter bank;

e. Two charcoal adsorber banks;

f. A second HEPA filter bank; and

g. Two centrifugal fans (one lead ~~primary~~ and one lag ~~backup~~) each with inlet flow control vanes.

The sizing of the SGT System equipment and components is based on the results of an infiltration analysis, as well as an exfiltration analysis. The internal pressure of the Secondary Containment ~~SGT System~~ boundary region is maintained at a negative pressure of 0.25 inch water gauge when the SGT system is in operation, which represents the internal pressure required to ensure zero exfiltration of air from the building under adverse conditions. ~~using the 95% meteorological data.~~

(continued)

BASES

BACKGROUND (continued)

The moisture separator is provided to remove entrained water in the air, while the electric heaters reduce the relative humidity of the airstream to less than 70% (Ref. 2). The prefilter removes large particulate matter, while the HEPA filter is provided to remove fine particulate matter and protect the charcoal from fouling. The charcoal adsorber removes gaseous elemental iodine and organic iodides, and the final HEPA filter is provided to collect any carbon fines exhausted from the charcoal adsorber.

In the event that the lead subsystem in a train fails to establish proper air flow within a set time delay from an SGT system start, the lead subsystem will automatically shutdown and the lag subsystem will start.

The SGT System automatically starts and operates in response to actuation signals indicative of conditions or an accident that could require operation of the system. Following initiation, ~~one fan per~~ the lead fan in each subsystem starts. SGT System flows are controlled automatically by modulating inlet vanes installed on the SGT fans.

APPLICABLE SAFETY ANALYSES

The design basis for the SGT System is to mitigate the consequences of a loss of coolant accident ~~and fuel handling accidents~~ (Ref~~s~~. 3 ~~and 4~~). For all events analyzed, the SGT System is shown to be automatically initiated to reduce, via filtration and adsorption, the radioactive material released to the environment.

The SGT System satisfies Criterion 3 of Reference ~~5~~ 4.

LCO

Following a DBA, a minimum of one SGT subsystem is required to maintain the secondary containment at a negative pressure with respect to the environment and to process gaseous releases. Meeting the LCO requirements for two OPERABLE independent subsystems ensures operation of at least one SGT subsystem in the event of a single active failure. ~~In addition, only the primary electric heater bank and centrifugal fan are required for OPERABILITY of each SGT subsystem.~~

In addition, only the lead fan and heater (including the support equipment and controls), or the lag fan and heater (including the support equipment and controls) in each filter train are required for OPERABILITY of each SGT filter train. However, in order to meet the requirement that two independent subsystems be OPERABLE, both lead subsystems or both lag subsystems are required to be OPERABLE (i.e. the two independent subsystems required by the LCO must be powered from the separate buses).

APPLICABILITY

~~-----NOTE-----~~
~~Handling a cask/canister loaded with spent fuel, after the canister is seal welded and leak tested, is not considered to be movement of irradiated fuel.~~
~~----------~~

(continued)

BASES

APPLICABILITY (continued)

In MODES 1, 2, and 3, a DBA could lead to a fission product release to primary containment that leaks to secondary containment. Therefore, SGT System OPERABILITY is required during these MODES.

In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the SGT System OPERABLE is not required in MODE 4 or 5, except for other situations under which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs). ~~during CORE ALTERATIONS, or during movement of irradiated fuel assemblies in the secondary containment.~~

ACTIONS

A.1

With one SGT subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this condition, the remaining OPERABLE SGT subsystem is adequate to perform the required radioactivity release control function. However, the overall system reliability is reduced because a single failure in the OPERABLE subsystem could result in the radioactivity release control function not being adequately performed. The 7 day Completion Time is based on consideration of such factors as the availability of the OPERABLE redundant SGT subsystem and the low probability of a DBA occurring during this period.

B.1 and B.2

If the SGT subsystem cannot be restored to OPERABLE status within the required Completion Time in MODE 1, 2, or 3, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

and C.2

ACTIONS (continued)

C.1, C.2.1, C.2.2, and C.2.3

During ~~movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during~~ OPDRVs, when Required Action A.1 cannot be completed within the required Completion Time, the OPERABLE SGT subsystem should be immediately placed in operation. This Required Action ensures that the remaining subsystem is OPERABLE, that no failures that could prevent automatic actuation will occur, and that any other failure would be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that represent a potential for releasing radioactive material to the secondary containment, thus placing the unit in a condition that minimizes risk. ~~If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies must be immediately suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also,~~ If applicable, action must be immediately initiated to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Action must continue until OPDRVs are suspended.

~~LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since irradiated fuel assembly movement can occur in MODE 1, 2, or 3, the Required Actions of Condition C have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.~~

D.1

If both SGT subsystems are inoperable in MODE 1, 2, or 3, the SGT System may not be capable of supporting the required radioactive release control function. Therefore, actions are required to enter LCO 3.0.3 immediately.

(continued)

BASES

ACTIONS (continued)

E.1~~, E.2, and E.3~~

When two SGT subsystems are inoperable, ~~if applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies in the secondary containment must be immediately suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if~~ applicable, actions must be immediately initiated to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Action must continue until OPDRVs are suspended.

~~LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since irradiated fuel assembly movement can occur in MODE 1, 2, or 3, Required Action E.1 has been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be sufficient reason to require a reactor shutdown.~~

SURVEILLANCE REQUIREMENTS

SR 3.6.4.3.1

Operating (from the control room) each SGT subsystem for ≥ 10 continuous hours ensures that both subsystems are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. Operation with the heaters on (automatic heater cycling to maintain temperature) for ≥ 10 continuous hours every 31 days eliminates moisture on the adsorbers and HEPA filters. The 31 day Frequency was developed in consideration of the known reliability of fan motors and controls and the redundancy available in the system.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.4.3.2

This SR verifies that the required SGT filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The SGT System filter tests are in accordance with Regulatory Guide 1.52 (Ref. ~~6~~). 5 The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specified test frequencies and additional information are discussed in detail in the VFTP.

and reaches ≥ 5000 cfm within 2 minutes after the

SR 3.6.4.3.3

This SR requires verification that each SGT subsystem starts ~~upon~~ receipt of an actual or simulated initiation signal. 4 The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.6.2, "Secondary Containment Isolation Instrumentation," overlaps this SR to provide complete testing of the safety function. While this Surveillance can be performed with the reactor at power, operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.4.3.4

This SR requires verification that the ~~primary~~ SGT filter cooling recirculation valve can be opened and the ~~primary~~ fan started. This ensures that the ventilation mode of SGT System operation is available. While this Surveillance can be performed with the reactor at power, operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

(continued)

Reaching this flow rate within 2 minutes, in conjunction with SR 3.6.4.1.4, demonstrates that the SGT subsystem can drawdown secondary containment within the assumptions of the applicable safety analysis.

BASES (continued)

REFERENCES

1. 10 CFR 50, Appendix A, GDC 41.
2. FSAR, Section 6.5.1.2.
3. FSAR, Sections 15.6.5 ~~and 15.F.6~~.

~~4. FSAR, Section 15.7.4.~~

4 ~~5.~~ 10 CFR 50.36(c)(2)(ii).

5 ~~6.~~ Regulatory Guide 1.52, Rev. 2.

B 3.7 PLANT SYSTEMS

B 3.7.3 Control Room Emergency Filtration (CREF) System

BASES

BACKGROUND

The CREF System provides a radiologically controlled environment from which the unit can be safely operated following a Design Basis Accident (DBA).

The safety related function of the CREF System used to control radiation exposure consists of two independent and redundant high efficiency air filtration subsystems for treatment of outside supply air. Each subsystem consists of an electric heater, a prefilter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section, a filter unit fan, a control room recirculation fan, and the associated ductwork and dampers. The electric heater is used to limit the relative humidity of the air entering the filter train. Prefilters and HEPA filters remove particulate matter that may be radioactive. The charcoal adsorbers provide a holdup period for gaseous iodine, allowing time for decay.

The safety related CREF System is a standby system, but most of the ductwork is common to the Control Room Heating, Ventilation, and Air Conditioning (HVAC) System, which is operated to maintain the control room environment during normal operation. Upon receipt of the initiation signal(s) (indicative of conditions that could result in radiation exposure to control room personnel), the CREF System automatically switches to the pressurization mode of operation to prevent infiltration of contaminated air into the control room. A system of dampers isolates the control room (from the normal intake and exhaust), and control room outside air flow is redirected and processed through either of the two filter subsystems.

TEDE

The CREF System is designed to maintain the control room environment for a 30 day continuous occupancy after a DBA, without exceeding a 5 rem ~~whole body~~ dose. ~~or its equivalent to any part of the body.~~ CREF System operation in maintaining the control room habitability is discussed in the FSAR, Sections 6.4.1 and 9.4.1 (Refs. 1 and 2, respectively).

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.5 Main Condenser Offgas

BASES

BACKGROUND

During unit operation, steam from the low pressure turbine is exhausted directly into the main condenser. Air and noncondensible gases are collected in the main condenser, then exhausted through the steam jet air ejectors (SJAEs) to the Main Condenser Offgas System. The offgas from the main condenser normally includes radioactive gases.

The Main Condenser Offgas System has been incorporated into the unit design to reduce the gaseous radwaste emission. This system uses a catalytic recombiner to recombine radiolytically dissociated hydrogen and oxygen. The gaseous mixture is cooled by the offgas condenser; the water and condensibles are stripped out by the offgas condenser and moisture separator. The radioactivity of the remaining gaseous mixture (i.e., the offgas recombiner effluent) is monitored downstream of the moisture separator prior to entering the holdup line.

APPLICABLE SAFETY ANALYSES

The main condenser offgas gross gamma activity rate is an initial condition of the Main Condenser Offgas System failure event as discussed in the FSAR, Section 11.3 (Ref. 1). The analysis assumes a single failure of a single component in the Main Condenser Offgas System. The gross gamma activity rate is controlled to ensure that during the event, the calculated offsite doses will be well within the limits (NUREG-0800, Ref. 2) of 10 CFR ~~100~~ 50.67 (Ref. 3).

The main condenser offgas limits satisfy Criterion 2 of Reference 4.

LCO

To ensure compliance with the assumptions of the Main Condenser Offgas System failure event (Ref. 1), the fission product release rate should be consistent with a noble gas release to the reactor coolant of 100 µCi/Mwt-second after decay of 30 minutes. The LCO is established consistent with this requirement (3323 Mwt x 100 µCi/Mwt-second = 332 mCi/second) and is based on the original licensed RATED THERMAL POWER.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

This SR, on a 31 day Frequency, requires an isotopic analysis of an offgas sample (taken at the discharge of the main condenser air ejector prior to dilution) to ensure that the required limits are satisfied. The noble gases to be sampled are Xe-133, Xe-135, Xe-138, Kr-85, Kr-87, and Kr-88. If the measured rate of radioactivity increases significantly (by ≥ 50% after correcting for expected increases due to changes in THERMAL POWER), an isotopic analysis is also performed within 4 hours after the increase is noted, to ensure that the increase is not indicative of a sustained increase in the radioactivity rate. The 31 day Frequency is adequate in view of other instrumentation that continuously monitor the offgas, and is acceptable based on operating experience.

This SR is modified by a Note indicating that the SR is not required to be performed until 31 days after any main steam line is not isolated and the SJAE is in operation. Only in this condition can radioactive fission gases be in the Main Condenser Offgas System at significant rates.

REFERENCES

1. FSAR, Section 11.3.
2. NUREG-0800.
3. 10 CFR ~~100~~ 50.67.
4. 10 CFR 50.36(c)(2)(ii).

B 3.7 PLANT SYSTEMS

B 3.7.7 Spent Fuel Storage Pool Water Level

BASES

BACKGROUND

The minimum water level in the spent fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident.

A general description of the spent fuel storage pool design is found in the FSAR, Section 9.1.2 (Ref. 1). The assumptions of the fuel handling accident are found in the FSAR, Section 15.7.4 (Ref. 2).

APPLICABLE SAFETY ANALYSES

The water level above the irradiated fuel assemblies is an explicit assumption of the fuel handling accident (Ref. 2). A fuel handling accident is evaluated to ensure that the radiological consequences ~~(calculated whole body and thyroid doses at the exclusion area and low population zone boundaries) are ≤ 25% (NUREG-0800, Section 15.7.4, Ref. 3) of the 10 CFR 100 (Ref. 4) exposure guidelines.~~ are within the guidelines of the reference values established 10CFR50.67 (Ref. 3). A fuel handling accident could release a fraction of the fission product inventory by breaching the fuel rod cladding as discussed in the Regulatory Guide ~~1.25~~ 1.183 (Ref. ~~8~~ 4).

The fuel handling accident is evaluated for the dropping of an irradiated fuel assembly onto the reactor core. The consequences of a fuel handling accident over the spent fuel storage pool are no more severe than those of the fuel handling accident over the reactor core (Ref. 2). The water level in the spent fuel storage pool provides for absorption of water soluble fission product gases and transport delays of soluble and insoluble gases that must pass through the water before being released to the secondary containment atmosphere. This absorption and transport delay reduces the potential radioactivity of the release during a fuel handling accident.

The spent fuel storage pool water level satisfies Criterion 2 of Reference ~~6~~ 5.

(continued)

BASES (continued)

REFERENCES

1. FSAR, Section 9.1.2.

2. FSAR, Section 15.7.4.

~~3. NUREG-0800, Section 15.7.4, Revision 1, July 1981.~~

3 ~~4~~. 10 CFR ~~100~~ 50.67.

4 ~~5~~. Regulatory Guide ~~1.25, March 1972~~ 1.183, Appendix C, July 2000.

5 ~~6~~. 10 CFR 50.36(c)(2)(ii).

B 3.9 REFUELING OPERATIONS

B 3.9.6 Reactor Pressure Vessel (RPV) Water Level — Irradiated Fuel

BASES

BACKGROUND

The movement of irradiated fuel assemblies within the RPV requires a minimum water level of 22 ft above the top of the RPV flange. During refueling, this maintains a sufficient water level in the reactor vessel cavity and spent fuel storage pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to ~~< 25% of~~ 10 CFR ~~100~~ 50.67 limits, as provided by the guidance of Reference ~~2~~ 1.

APPLICABLE SAFETY ANALYSES

During movement of irradiated fuel assemblies the water level in the RPV is an initial condition design parameter in the analysis of a fuel handling accident in containment. ~~postulated by Regulatory Guide 1.25 (Ref. 1).~~ A minimum water level of 23 ft (Regulatory Position ~~C.1.c~~ of Ref. 1) allows a decontamination factor ~~of 100 (Regulatory Position C.1.g of Ref. 1)~~ to be used in the accident analysis for iodine. This relates to the assumption that ~~99% of the total~~ 57% of the elemental and 43% of the organic iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is ~~retained by~~ released from the refueling cavity water. ~~The fuel pellet to cladding gap is assumed to contain 10% of the total fuel rod iodine inventory (Ref. 1).~~

Analysis of the fuel handling accident inside containment is described in Reference 2. With a minimum water level of 22 ft ~~(a decontamination factor of 100 is still expected at a water level as low as 22 ft)~~ and a minimum decay time of 24 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water, and that offsite doses are maintained within allowable limits (Ref. ~~4~~ 3). While the worst case assumptions include the dropping of the irradiated fuel assembly being handled onto the reactor core, the possibility exists of the dropped assembly striking the RPV flange and releasing fission products. Therefore, the minimum depth for water coverage to ensure acceptable radiological consequences is specified from the RPV flange. Since the worst case event

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

results in failed fuel assemblies seated in the core, as well as the dropped assembly, dropping an assembly on the RPV flange will result in reduced releases of fission gases.

RPV water level satisfies Criterion 2 of Reference ~~3~~. 4

LCO

A minimum water level of 22 ft above the top of the RPV flange is required to ensure that the radiological consequences of a postulated fuel handling accident are within acceptable limits, as provided by the guidance of Reference ~~2~~. 1

APPLICABILITY

LCO 3.9.6 is applicable when moving irradiated fuel assemblies within the RPV. The LCO minimizes the possibility of a fuel handling accident in containment that is beyond the assumptions of the safety analysis. If irradiated fuel is not present within the RPV, there can be no significant radioactivity release as a result of a postulated fuel handling accident. Requirements for handling of new fuel assemblies or control rods (where water depth to the RPV flange is not of concern) are covered by LCO 3.9.7, "RPV Water Level — New Fuel or Control Rods." Requirements for fuel handling accidents in the spent fuel storage pool are covered by LCO 3.7.7, "Fuel Pool Water Level."

ACTIONS

A.1

If the water level is < 22 ft above the top of the RPV flange, all operations involving movement of irradiated fuel assemblies within the RPV shall be suspended immediately to ensure that a fuel handling accident cannot occur. The suspension of irradiated fuel movement shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

Verification of a minimum water level of 22 ft above the top of the RPV flange ensures that the design basis for the postulated fuel handling accident analysis during refueling operations is met. Water at the required level limits the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1 (continued)

consequences of damaged fuel rods, which are postulated to result from a fuel handling accident in containment (Ref. 2).

The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls on valve positions, which make significant unplanned level changes unlikely.

REFERENCES

1. Regulatory Guide ~~1.25, March 23, 1972~~. 1.183, July 2000
2. FSAR, Section 15.7.4.
~~3. NUREG-0800, Section 15.7.4.~~

3 ~~4.~~ 10 CFR ~~100.11~~. 50.67

4 ~~5.~~ 10 CFR 50.36(c)(2)(ii).

B 3.9 REFUELING OPERATIONS

B 3.9.7 Reactor Pressure Vessel (RPV) Water Level—New Fuel or Control Rods

BASES

BACKGROUND

The movement of new fuel assemblies or handling of control rods within the RPV when fuel assemblies seated within the reactor vessel are irradiated requires a minimum water level of 22 ft above the top of irradiated fuel assemblies seated within the RPV. During refueling, this maintains a sufficient water level above the irradiated fuel. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to ~~≤ 25% of~~ 10 CFR ~~100~~ 50.67 limits, as provided by the guidance of Reference ~~2~~ 1.

APPLICABLE SAFETY ANALYSES

During movement of new fuel assemblies or handling of control rods over irradiated fuel assemblies, the water level in the RPV is an initial condition design parameter in the analysis of a fuel handling accident in containment. ~~postulated by Regulatory Guide 1.25 (Ref. 1).~~ A minimum water level of 23 ft (Regulatory Position ~~C.1.c~~ of Ref. 1) allows a decontamination factor ~~of 100 (Regulatory Position C.1.g. of Ref. 1)~~ to be used in the accident analysis for iodine. This relates to the assumption that ~~99% of the total~~ 57% of the elemental and 43% of the organic iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is ~~retained by~~ released from the refueling cavity water. ~~The fuel pellet to cladding gap is assumed to contain 10% of the total fuel rod iodine inventory (Ref. 1).~~

Analysis of the fuel handling accident inside containment is described in Reference 2. With a minimum water level of 22 ft ~~(a decontamination factor of 100 is still expected at a water level as low as 22 ft)~~ and a minimum decay time of 24 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water, and that offsite doses are maintained within allowable limits (Ref. ~~4~~ 3). The related assumptions include the worst case dropping of an irradiated fuel assembly onto the reactor core loaded with irradiated fuel assemblies.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

RPV water level satisfies Criterion 2 of Reference ~~3~~. 4

LCO

A minimum water level of 22 ft above the top of irradiated fuel assemblies seated within the RPV is required to ensure that the radiological consequences of a postulated fuel handling accident are within acceptable limits, as provided by the guidance of Reference ~~3~~. 1

APPLICABILITY

LCO 3.9.7 is applicable when moving new fuel assemblies or handling control rods (i.e., movement with other than the normal control rod drive) over irradiated fuel assemblies seated within the RPV. The LCO minimizes the possibility of a fuel handling accident in containment that is beyond the assumptions of the safety analysis. If irradiated fuel is not present within the RPV, there can be no significant radioactivity release as a result of a postulated fuel handling accident. Requirements for fuel handling accidents in the spent fuel storage pool are covered by LCO 3.7.7, "Fuel Pool Water Level." Requirements for handling irradiated fuel over the RPV are covered by LCO 3.9.6, "Reactor Pressure Vessel (RPV) Water Level — Irradiated Fuel."

ACTIONS

A.1

If the water level is < 22 ft above the top of irradiated fuel assemblies seated within the RPV, all operations involving movement of new fuel assemblies and handling of control rods within the RPV shall be suspended immediately to ensure that a fuel handling accident cannot occur. The suspension of fuel movement and control rod handling shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.7.1

Verification of a minimum water level of 22 ft above the top of the irradiated fuel assemblies seated within the RPV ensures that the design basis for the postulated fuel handling accident analysis during refueling operations is

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.7.1 (continued)

met. Water at the required level limits the consequences of damaged fuel rods, which are postulated to result from a fuel handling accident in containment (Ref. 2).

The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls on valve positions, which make significant unplanned level changes unlikely.

REFERENCES

1. Regulatory Guide ~~1.25, March 23, 1972.~~ 1.183, July 2000.

2. FSAR, Section 15.7.4.

~~3. NUREG-0800, Section 15.7.4.~~

3 ~~4.~~ 10 CFR ~~100.11~~ 50.67.

4 ~~5.~~ 10 CFR 50.36(c)(2)(ii).