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PG&E Letter DCL-15-069

10 CFR 50.90

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Diablo Canyon Units 1 and 2
Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
License Amendment Request 15-03
Application of Alternative Source Term

Pursuant to 10 CFR 50.90, Pacific Gas and Electric Company (PG&E) hereby requests approval of the enclosed proposed amendment to Facility Operating License Nos. DPR-80 and DPR-82 for Units 1 and 2, respectively, of the Diablo Canyon Power Plant (DCPP). The enclosed license amendment request (LAR) proposes to revise the DCPP Units 1 and 2 licensing bases to adopt the alternative source term (AST) as allowed by 10 CFR 50.67. The following Technical Specification (TS) changes are required for AST implementation: TS 1.1 for the definition of Dose Equivalent I-131; TS 3.4.16 to revise the noble gas activity limit; TS 3.6.3 to require the 48-inch containment purge supply and exhaust valves to be sealed closed during MODES 1, 2, 3, and 4; TS 5.5.9 to revise the accident induced leakage performance criterion; TS 5.5.11 to change the allowable methyl iodide penetration testing criteria for the auxiliary building ventilation system charcoal filter: and TS 5.5.19 to replace "whole body or its equivalent to any part of the body," with "TEDE," which is the dose criteria specified in 10 CFR 50.67.

The Enclosure provides a description of the proposed changes and supporting justification including the determination of no significant hazards and environmental considerations. Attachments to the Enclosure are described within.

The changes in this LAR are not required to address an immediate safety concern. PG&E requests approval of this LAR no later than June 30, 2016. PG&E requests the license amendments be made effective upon NRC issuance, to be implemented within 365 days from the date of issuance.

This communication contains five new regulatory commitments (as defined in NEI 99-04) to be implemented following NRC approval of this LAR. The commitments are contained in Attachment 7 of the Enclosure.

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In accordance with site administrative procedures and the Quality Assurance Program, the proposed amendment has been reviewed by the Plant Staff Review Committee.

Pursuant to 10 CFR 50.91, PG&E is sending a copy of this proposed amendment to the California Department of Public Health.

If you have any questions or require additional information, please contact Hossein Hamzehee at 805-545-4720.

I state under penalty of perjury that the foregoing is true and correct.

Executed on June 17, 2015.

Sincerely,

James M. Welsch
Site Vice President

kjse/4328/50705089
Enclosure

cc: Diablo Distribution
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Evaluation of the Proposed Change

License Amendment Request 15-03
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EVALUATION

1. SUMMARY DESCRIPTION

This license amendment request (LAR) would amend Operating Licenses DPR-80 and DPR-82 for Units 1 and 2 of the Diablo Canyon Power Plant (DCPP), respectively.

Pacific Gas & Electric (PG&E) requests Nuclear Regulatory Commission (NRC) review and approval of a proposed revision to the licensing basis of DCPP Units 1 and 2 that supports a full scope application of an alternative source term (AST) methodology as allowed by 10 CFR 50.67 (Reference 1).

An application for the selective use of AST for the fuel handling accident (FHA) in the fuel handling building (FHB) was reviewed and approved by the NRC in its Safety Evaluation Report (SER) for License Amendment Nos. 163 and 165 (Reference 2). However, the FHA in the FHB has also been reanalyzed with this application and is included in this submittal to be consistent with revised inputs, as described in Attachment 4. Approval of this AST application will supersede the FHA in the FHB dose analysis and results, as discussed in the SER for License Amendment Nos. 163 and 165.

The AST methodology as established in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000 (Reference 3) is used to calculate the offsite and Control Room radiological consequence for DCPP Units 1 and 2. Attachment 4 contains a summary of the analyses and results for the following events that are expected to produce the most limiting dose consequences. Conformance to RG 1.183 is provided in Attachment 5.

- Loss of Coolant Accident (LOCA)
- FHA in the Containment
- FHA in the FHB
- Locked Rotor Accident (LRA)
- Control Rod Ejection Accident (CREA)
- Main Steam Line Break (MSLB)
- Steam Generator Tube Rupture (SGTR)
- Loss-of-Load (LOL) Event

In addition to adopting AST for design basis accidents and the associated total effective dose equivalent (TEDE) dose criteria for offsite and Control Room doses, DCPP is adopting the TEDE dose criteria of 10 CFR 50.67 for the Technical Support Center (TSC), as allowed by RG 1.183.

Full implementation of AST for DCPP Units 1 and 2 does not include revising the source terms used for environmental qualification (EQ) of safety related equipment or NUREG-0737 responses associated with shielding and vital area access. Section C.6 of RG 1.183 (Reference 3) discusses the position on performance of required EQ analyses with respect to AST and TID-14844 (Reference 4) source term assumptions. NUREG-0933, "Resolution of Generic Safety Issues," Section 3.0, Item 187 (Reference 5) resolved the issues related to the effect of increased cesium releases on EQ doses. The NRC staff concluded that there is no clear basis for a requirement to modify the design basis for EQ to adopt AST since there would be no discernible risk reduction associated with adopting AST for EQ. In addition, post-accident vital area access dose rates are not expected to be significantly impacted by the AST during the first 30 days following a LOCA based on an AST benchmarking study. The NRC SER for Fort Calhoun Station's implementation of AST (Reference 6) referenced the SECY-98-154 (Reference 7) study as the source for the conclusion that the results of analyses based on TID-14844 would be more limiting for a period up to one to four months after which time the AST results would be more limiting. Therefore, this LAR does not propose to modify the EQ design basis nor the shielding and vital area access dose rates to adopt AST.

The proposed amendment revises Technical Specification (TS) definitions, requirements, and terminology related to the use of an AST associated with offsite, Control Room, and TSC accident dose consequences. A markup of affected TS pages is included in Attachment 1 to this Enclosure.

Regulatory Issue Summary (RIS) 2006-04, "Experience with Implementation of Alternative Source Terms," dated March 7, 2006, (Reference 8) outlined twelve issues that the NRC staff has encountered during its review of AST submittals. Attachment 6 provides discussion on how DCPP has addressed the twelve issues identified in RIS 2006-04.

2. DETAILED DESCRIPTION

2.1 Proposed Changes to Current Licensing Basis

The dose consequence analyses addressed in this application have been revised to incorporate the guidance provided in RG 1.183 (Reference 3) and to resolve the findings of the Licensing Basis Verification Project (LBVP) which was voluntarily initiated by DCPP, as presented to NRC (ADAMS Accession No. ML15029A094). The LBVP findings have been addressed in prompt operability assessments, which include several temporary compensatory measures. The revised dose analyses address the LBVP findings, as well as implements the following licensing basis changes.

1. Implement RG 1.183, July 2000 (Reference 3), as the licensing basis for DCPP, as outlined in this LAR. RG 1.183 will replace DCPPs commitment

to RG 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors."

2. Remove the "expected" accident dose consequence assessments that are in DCPP Updated Final Safety Analysis Report (UFSAR) Section 15.5.

   The original DCPP licensing application included two evaluations for each accident. The first evaluation, called the expected case, used estimates of actual values expected to occur if the accident took place. The resulting doses were close to the doses expected from an accident of this type. The second evaluation, the Design Basis Accident (DBA), used the customary conservative assumptions. The calculated doses for the DBA, while not a realistic estimate of expected doses, provided the basis for determining the design adequacy of the plant safety systems.

   Current NRC guidance related to expectations for a safety analysis report for a nuclear power plant, as provided in NUREG-0800 (Reference 9), does not require the inclusion of dose consequences from "expected" accident scenarios. Since these "expected" accident scenario evaluations are not relevant for determining design adequacy of plant safety systems, PG&E is proposing to remove this information from its licensing basis. UFSAR markups showing the elimination of the "expected" cases are provided for information only in Attachment 8.

3. Eliminate the dose contribution of a containment purge via the containment hydrogen purge system following a LOCA for purposes of hydrogen control.

   The NRC revised 10 CFR 50.44 (Reference 10) to acknowledge that the amount of combustible gas generated for the design basis LOCA was not a risk significant threat to containment integrity. Thus, with the exception of demonstrating the capability of ensuring a mixed atmosphere within containment, the requirements for hydrogen control pertaining to the design basis LOCA were eliminated. In the SER for License Amendment Nos. 168 and 169 to DCPP (Reference 11), the NRC confirmed the elimination of hydrogen release concerns associated with a design-basis LOCA, and the associated requirements that necessitated the need for hydrogen recombiners and backup hydrogen vent and purge systems.

   To ensure consistency with the current licensing basis, PG&E is proposing to eliminate the dose contribution due to the containment purge pathway currently included in the LOCA dose consequence analysis in support of hydrogen control.

4. Replace dose guidelines of 10 CFR 100.11 for whole body and thyroid dose with the TEDE acceptance criterion of 10 CFR 50.67(b)(2) and Section 4.4, Table 6 of RG 1.183 (Reference 3).

   As required by 10 CFR 50.67(b)(2) (Reference 1), Attachment 4 of this LAR contains an evaluation of the consequences of applicable DBAs previously analyzed in the DCPP UFSAR (Reference 12). The TEDE dose criteria will be applied to offsite locations, as well as the Control Room and the TSC.

5. Replace General Design Criteria (GDC 19), 1971, with GDC 19, 1999 for dose only.

   10 CFR 50.67(b)(2)(iii) states that adequate radiation protection is provided to permit access to and occupancy of the Control Room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) TEDE for the duration of the accident. As part of implementation of 10 CFR 50.67, the NRC also amended GDC 19, 1971 (Reference 13) to reflect the 5 rem TEDE aspects. The DCPP license basis, from the original Final Safety Analysis Report through Amendment 85, includes GDC 19, 1971, for Control Room dose only (Reference 12). DCPP will conform to GDC 19, 1999, for dose only, for Control Room dose limits of 5 rem TEDE upon implementation of AST. This change to GDC 19, 1999, for dose only, is consistent with the adoption of AST.

6. Update the dose acceptance criterion for the TSC to 5 rem TEDE.

   The dose acceptance criterion for the TSC is based on Section 8.2.1, Item f of NUREG-0737, Supplement 1 (Reference 14), which states that any person working in the TSC would not exceed 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. The dose acceptance criterion is modified to 5 rem TEDE to be consistent with 10 CFR 50.67(b)(2) and GDC 19, 1999, for dose only. In accordance with DCPP current licensing basis, the TSC is only evaluated for the LOCA.

   Therefore, upon implementation of AST, the DCPP licensing basis for NUREG-0737, Item II.B.2 and III.A.1.2 will be the AST acceptance criteria specified in 10 CFR 50.67(b) and GDC 19, 1999, for dose only. This change to GDC 19, 1999, is consistent with the adoption of AST.

7. Update computer codes that support the AST dose consequence analyses.

   Attachment 4, Section 3 provides the computer codes utilized in support of this application. These computer codes will now be included in DCPP's

licensing basis in the manner in which they are utilized in the dose consequence analyses as outlined in Attachment 4. The codes used in support of the AST application are recommended by RG 1.183 or have been used in prior AST applications that have been approved by NRC.

8. Use inhalation dose conversion factors from Environmental Protection Agency (EPA) Federal Guidance Report (FGR) No. 11, 1988, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion." (Reference 15).

   FGR No. 11 has been part of the licensing basis for DCPP in that TS Definition 1.1 for Dose Equivalent I-131 (DEI) allowed Table 2.1 of FGR No. 11 to be used for determining DEI. FGR No. 11 is used in AST dose consequence analyses for inhalation dose conversion factors, as recommended by RG 1.183. See Section 2.2 for proposed TS changes.

9. Update offsite atmospheric dispersion factors (χ/Q) using recent 5-year meteorological data (2007 to 2011) and RG 1.145, Revision 1, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," (Reference 16) methodology.

   The methodology outlined in RG 1.145, Revision 1, is used for calculating ground level releases to determine the short-term χ/Q values for the Exclusion Area Boundary (EAB) and the Low Population Zone (LPZ) for design basis radiological analyses. All releases are conservatively treated as ground level releases, therefore Regulatory Positions C.1.3.2, C.2.1.2, and C.2.2.2 associated with elevated or stack releases are not applicable to DCPP.

   RG 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light Water Cooled Reactors," Regulatory Position C.1.c (Reference 17) is used to determine the annual average χ/Q values, which are used as input to develop the accident χ/Q values at the LPZ using RG 1.145 methodology.

   Attachment 4, Section 5 presents the development of the χ/Q values.

10. Update χ/Q factors for on-site locations such as the Control Room and the TSC using recent 5-year meteorological data (2007 to 2011) and, "Atmospheric Relative Concentrations in Building Wakes," (ARCON96) methodology (Reference 18).

   Regulatory Guide 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," dated June 2003 (Reference 19), Regulatory Position C.1 through C.3, and the adjustment factor for vertically orientated energetic releases from

steam relief valves and atmospheric dump valves (ADVs) allowed by Regulatory Position C.6 are used to determine short-term on-site χ/Q values in support of design basis radiological habitability assessments.

PG&E is requesting an exception to RG 1.194, Regulatory Position C.3.4, as part of this LAR. Two specific Control Room receptors are within 10 meters of the release (9.4 meters and 7.8 meters). The χ/Q values for these two cases were developed to establish bounding χ/Q values. However, the χ/Q values for these two locations were not the bounding values and therefore were not used in the dose consequence analyses. Attachment 4, Section 5.2 provides further discussion for the requested exception.

Credit for DCPP's dual intake design for the Control Room pressurization air intakes is taken per RG 1.194, Regulatory Position C.3.3.2.3. In addition, credit is taken for a reduction factor of 5 applied to the χ/Q values for energetic releases from the DCPP Main Steam Safety Valves (MSSVs) and the 10 percent ADVs, per RG 1.194, Regulatory Position C.6 due to the velocity and orientation of the release. This credit is used for the MSLB, SGTR, LRA, CREA, and LOL events.

Credit is also taken for the close proximity of the MSSVs/10 percent ADVs to the normal operating Control Room intake of the affected unit and the high vertical velocity of the steam discharge from the MSSVs/10 percent ADVs resulting in the post-accident plume from the MSSVs/10 percent ADVs not contaminating the normal operation Control Room intake of the affected unit. This credit is used for the MSLB, SGTR, LRA, CREA, and LOL events.

Attachment 4, Section 5 presents the development of the χ/Q values.

11. Update Control Room ventilation system (CRVS) parameters resulting from the installation of new back-draft damper in the Control Room emergency filter recirculation lines.

    Back-draft dampers were installed to prevent reverse unfiltered flow into the Control Room. The updated CRVS parameters have been included in the new Control Room transport model discussed in Attachment 4, Section 7.1.

12. Update Control Room unfiltered inleakage.

    The updated Control Room unfiltered inleakage values, including back-draft damper leakage, have been included in the new Control Room transport model discussed in Attachment 4, Section 7.1. The updated

Control Room unfiltered inleakage value bounds the unfiltered inleakage determined by the 2012 Control Room Tracer Gas Test (Reference 20).

13. Use containment spray in the recirculation mode following a LOCA for fission product cleanup.

    DCPP is designed and licensed to operate using containment spray in the recirculation mode. In accordance with the current licensing basis, and as documented in the NRC SER related to License Amendment No. 139 to Facility Operating License Nos. DPR-80 and DPR-82 (Reference 21), containment spray is not required per analyses to be actuated during recirculation, but may be actuated in accordance with the emergency operating procedures (EOPs) or at the discretion of the TSC.

    To address the delayed core damage sequence of a post-LOCA AST scenario and support fission product removal from the containment atmosphere, credit is taken in the LOCA dose analysis for using containment spray in the recirculation mode for dose mitigation. This licensing basis change to require containment spray during recirculation for fission product cleanup does not affect the conclusions of the SER related to License Amendment No. 139 with respect to the other functions of containment spray. The containment spray system will be operated in a manner consistent with the licensing basis established by the SER related to License Amendment No. 139 and no changes in operation are being proposed, other than requiring its operation within 12 minutes following terminating injection spray, instead of being optional in accordance with EOPs or at the discretion of the TSC.

    The LOCA dose analysis also credits a time critical operator action (TCOA). A TCOA is a manual action or series of actions with a specified completion time limit to meet a plant licensing basis requirement. The LOCA dose analysis assumes that containment spray is realigned from the injection mode to the recirculation mode within 12 minutes of terminating injection spray to ensure that the duration of spray operation (injection + recirculation) exceeds 6.25 hours following the event. The TCOA will be implemented as part of AST implementation, using the guidelines provided in NRC Information Notice 97-78, "Crediting of Operator Actions in Place of Automatic Actions and Modifications of Operator Actions, Including Response Times." The required actions for the new TCOA have been demonstrated on the simulator, showing that the 12 minute time requirement can be achieved with margin.

14. Update allowable engineered safety features (ESF) system leakage values and associated release points following a LOCA.

In accordance with RG 1.183, the LOCA dose analysis assumes a release of two times the allowable ESF leakage values. As such, two times the allowable leakage values outlined in Section 2.5 are used to determine doses following a LOCA, as discussed in Attachment 4, Section 7.2.3.3. The ESF leakage limits are controlled by TS 5.5.2, "Primary Coolant Sources Outside Containment."

15. Include environmental releases from the refueling water storage tank (RWST) vent due to sump water back-leakage following a LOCA.

    The LOCA dose analysis for AST includes the environmental releases from the RWST due to sump water back-leakage following a LOCA, in accordance with RG 1.183. Attachment 4, Section 7.2.3.5 presents the analysis for this dose contribution.

16. Include environmental releases from the Miscellaneous Equipment Drain Tank (MEDT).

    The LOCA dose analysis for AST includes the environmental releases from the MEDT following a LOCA, in accordance with RG 1.183. Attachment 4, Section 7.2.3.6 presents the analysis for this dose contribution.

17. Include environmental releases via the 12-inch containment vacuum/pressure relief pathway prior to containment isolation following a LOCA.

    In accordance with RG 1.183, for containments such as DCPP that can be routinely purged during normal operation, the dose consequence analysis must assume a release to the environment, through the purge pathway, occurs prior to containment isolation. As such, the LOCA dose analysis includes a dose contribution from the 12-inch containment vacuum/pressure relief pathway prior to containment isolation. Attachment 4, Section 7.2.3.1 presents the analysis for this dose contribution.

18. Preclude environmental releases via the 48-inch containment purge and exhaust system pathway prior to containment isolation following a LOCA.

    In accordance with RG 1.183, for containments such as DCPP that can be routinely purged during normal operation, the dose consequence analysis must assume a release to the environment, through the purge pathway, occurs prior to containment isolation. This release pathway is not applicable to the containment purge and exhaust system because DCPP is requesting approval of a TS change for the 48-inch containment purge valves to be sealed closed in accordance with Standard Review Plan

(SRP) Section 6.2.4, Revision 3, Items II.6 and II.14 during MODES 1, 2, 3, and 4. The 48-inch containment purge valves will be sealed closed by removing motive power to the valve operators. With this proposed TS revision, NUREG-0737, Item II.E.4.2, Position (6) (Reference 22) will be satisfied. Because the 48-inch containment purge valves will be required to be sealed closed during MODES 1, 2, 3, and 4, DCPP will no longer take credit for a Phase A isolation signal for these valves, as outlined in response to NUREG-0737, November 1980 Item II.E.4.2, Position (7). In addition, piping classification for the associated containment penetrations (61 and 62) will change from Group A to Group E.

See Section 2.2 for proposed TS changes associated with the 48-inch containment purge supply and exhaust valves.

19. Define the portion of Room 506 of the Control Room which serves as a Control Room foyer between the Control Room Assistants' office and the Shift Managers' office as a low occupancy, less frequented area.

    When determining the direct shine dose to the Control Room from external and contained sources, the analysis presented in Attachment 4, Section 7.2.5.2 takes into consideration the function of Room 506. Room 506 is used as an area where occupancy is deemed to be minimal. Thus, an "occupancy adjustment" factor is utilized for Room 506 to determine the maximum 30-day integrated dose in the Control Room (i.e., the total direct shine dose in the Control Room includes the 30-day dose in Room 506 adjusted by occupancy factor).

20. Define a minimum decay time prior to fuel movement as 72 hours.

    As part of this application, DCPP proposes to revise the definition of recently irradiated fuel as fuel that has occupied part of a critical reactor core within the previous 72 hours. This definition is used in the dose consequence analysis of the FHA to determine the release following the postulated event. Although the source term for the FHA will be slightly larger with less decay of fuel prior to fuel movement, the dose consequence analysis results show that the dose criteria are met, as shown in Table 1 and Attachment 4, Section 7.3.

21. Credit the redundant safety related gamma sensitive area radiation monitors (1-RE 25/26, 2-RE 25/26) to initiate CRVS Mode 4 following a FHA.

    These monitors are located at the Control Room normal intakes and are designed to automatically isolate the normal CRVS intakes and shift to CRVS Mode 4 (pressurized filtered emergency ventilation). These monitors are credited to perform their design function following a FHA in

the FHB or containment. See Section 2.4 for a description of a setpoint change for these radiation monitors.

22. Credit the following existing administrative controls reflected in plant procedures. These administrative controls ensure the FHB is maintained at a negative pressure relative to atmosphere during movement of irradiated fuel in the spent fuel pool, thus ensuring that the environmental releases occur via the Unit vent.

    - The movable wall is in place and secured
    - No exit door from the FHB is propped open
    - At least one FHB Ventilation System exhaust fan is running

    Attachment 4, Section 7.3 presents the FHA. Credit for the above administrative controls is taken to facilitate that the post-accident environmental release of radioactivity occurs via the plant vent.

23. Update reported doses for other UFSAR Chapter 15 events with accident source terms to TEDE doses criteria.

    The DCPP licensing basis includes dose assessments at offsite locations for several Condition III and Condition IV events. RG 1.183 does not address Condition III and Condition IV events; therefore they have not been re-analyzed with this application. SRP 15.0.1 (Reference 23) states that a complete recalculation of all design basis radiological consequence analyses may not be required for an application to be acceptable. However, SRP 15.01 also states that a full AST implementation replaces the previous accident source term used in all design basis radiological analyses and incorporates the TEDE dose criteria. Therefore, DCPP performed scoping evaluations, as allowed by RG 1.183, Section C.1.3.3 (Reference 3), to demonstrate compliance with regulatory limits at the EAB and LPZ for the DCPP UFSAR Chapter 15 Condition III and Condition IV events. The evaluation compares the accident sequence, predicted fuel damage (if applicable), and resultant dose consequences of the DBAs analyzed for AST to those parameters of the Condition III and Condition IV events. These evaluations are presented in Attachment 4, Section 2.1.

    The tank rupture events presented in UFSAR Chapter 15.5 represent the accidental release of radioactivity accumulated in tanks resulting from normal plant operations and are not affected by accident source terms associated with AST. Therefore, the tank rupture events are not reanalyzed in support of this LAR and the dose acceptance criteria for the tank rupture events will remain unchanged.

As part of its original licensing basis, DCPP provides an estimated radiation exposure to the Control Room operator during egress and ingress between the Control Room and the site boundary following a LOCA, as presented in UFSAR 15.5.17.10. Although RG 1.183 does not address or provide guidance for determining this dose contribution, DCPP is retaining this access dose in its licensing basis. The whole body gamma dose and thyroid dose reported in the UFSAR are converted to reflect the estimated TEDE dose by using organ weighting factors provided in 10 CFR 20.1003. Attachment 4, Section 7.2.6 demonstrates that the access dose, converted to TEDE dose, is minimal in that it is 1 percent of the estimated operator dose due to Control Room occupancy following a LOCA.

It is noted that the dose received by the operator during transit outside the Control Room is not a measure of the "habitability" of the Control Room, which is defined by the radiation protection provided to the operator by the Control Room shielding and ventilation system design. Thus, the estimated dose to the operator during routine post-LOCA access to the Control Room is addressed separately from the Control Room occupancy dose and is not included with the Control Room occupancy dose for the demonstration of Control Room habitability.

2.2 Proposed Technical Specification Changes

The following TS changes are proposed to reflect the licensing basis changes outlined in Section 2.1. Brief descriptions of the associated proposed TS changes are provided below along with justification for each change. The specific wording changes to the TS are provided in Attachments 1 and 2 to this enclosure.

- TS 1.1, "Definitions," is revised to change the definition of Dose Equivalent I-131 (DEI).

  This TS provides a definition for DEI, which currently references Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites;" Table E-7 of Regulatory Guide 1.109, Revision 1, NRC, 1977; International Commission on Radiological Protection Publication 30, 1979, Supplement to Part 1, pages 192-212, Table titled "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity;" and Table 2.1 of EPA Federal Guidance Report No. 11, 1988, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."

  This TS change will be revised to only reference the committed thyroid dose equivalent conversion factors from Table 2.1 of FGR No. 11

(Reference 15). The change is consistent with the recommendations of RG 1.183. Under AST, the doses are reported as TEDE dose.

NRC RIS 2006-04 (Reference 8) states:

> *"Although different references are available for dose conversion factors, the TS definition should be based on the same dose conversion factors that are used in the determination of the reactor coolant dose equivalent iodine curie content for the main steamline break and steam generator tube rupture accident analyses."*

Dose conversion factors from Table 2.1 of FGR No. 11 (Reference 15) are used by DCPP to determine the reactor coolant dose equivalent iodine curie content for the MSLB and SGTR accident analyses. Thus, the TS change is consistent with item 10, Definition of Dose Equivalent I-131, of NRC RIS 2006-04.

The TS change will remain consistent with the approved Industry Improved Standard Technical Specification Traveler, TSTF-490 (Reference 24).

- TS 3.4.16, "RCS Specific Activity," is revised to change the Noble gas activity limit from less than or equal to 600 μCi/gm Dose Equivalent XE-133 (DEX) to less than or equal to 270 μCi/gm DEX.

  DEX limit is equivalent to approximately 0.5 percent fuel defects. The current limit of 600 μCi/gm DEX corresponds to approximately 1 percent fuel defects, which is the DCPP design basis value for system and shielding design. The limit is being reduced by DCPP to control the noble gas activity in the coolant to levels below the design basis values.

- TS 3.6.3, "Containment Isolation Valves," is revised to modify Note 1 of Limiting Condition of Operation (LCO) 3.6.3 concerning the 48-inch containment purge supply and exhaust valves. The TS currently allows the operation of these valves for less than 200 hours per year during operating MODES 1, 2, 3, and 4. The proposed revision eliminates the administratively controlled operation of the 48-inch containment purge valves during MODES 1, 2, 3, and 4. The proposed revision will now require the 48-inch containment purge supply and exhaust valves to remain sealed closed during MODES 1, 2, 3, and 4. This change will eliminate a potential dose contribution due to an open containment purge pathway at the initiation of a LOCA.

  The TS revision includes a new surveillance requirement for verifying the 48-inch purge valves are sealed closed, removes the 48-inch purge valves

from Surveillance Requirements (SRs) 3.6.3.2 and modifies the frequency for SR 3.6.3.7.

The proposed revision is consistent with NUREG-1431, Volume 1, Standard Technical Specifications, Westinghouse Plants (Reference 25).

The 48-inch containment purge valves are to be sealed closed in accordance with SRP Section 6.2.4, Revision 3, Item II.6 and II.14 during MODES 1, 2, 3, and 4. The 48-inch containment purge valves will be sealed closed by removing motive power to the valve operators. With this proposed TS revision, NUREG-0737, Item II.E.4.2, Position (6) (Reference 22) will be satisfied. Because the 48-inch containment purge valves will be required to be sealed closed during MODES 1, 2, 3, and 4, DCPP will no longer take credit for a Phase A isolation signal for these valves, as outlined in response to NUREG-0737, November 1980 Item II.E.4.2, Position (7).

- TS 5.5.9, "Steam Generator (SG) Tube Inspection Program," is revised to lower the accident induced leakage performance criterion from 1 gallon per minute (gpm) per steam generator to 0.75 gpm total for all four steam generators. The accident induced leakage performance criterion shall not exceed the leakage rate assumed in the dose consequence analysis.

  A primary-to-secondary SG tube leakage of 0.75 gpm at standard temperature and pressure is used in the dose consequence analysis for the LRA, CREA, MSLB, SGTR, and LOL events. DCPP TS 3.4.13d limits primary-to-secondary SG tube leakage to 150 gallons per day (gpd) per SG for a total of 600 gpd for all 4 SGs. The revised testing criterion for the primary-to-secondary leakage is more restrictive than the current testing criteria and represents the leakage rate assumed in the dose consequence analyses presented in Attachment 4. The 0.75 gpm from all 4 SGs (or a total of 1080 gpd) conservatively bounds the TS 3.4.13.d limit in that the analyzed leakage rate accounts for higher leakage than the TS 3.4.13.d limit, and thus a higher analyzed release of radioactivity.

- TS 5.5.11, "Ventilation Filter Testing Program (VFTP)," is revised to change the allowable methyl iodide penetration testing criteria for the auxiliary building ventilation system (ABVS) charcoal filter from 15 percent to 5 percent.

  The allowable methyl iodide penetration is used to determine charcoal filter efficiency for removing iodine from atmospheric releases. Credit for filtration of the release of a residual heat removal (RHR) system pump seal passive failure by the ABVS is taken in determining the dose

consequences to the public at the EAB and LPZ, and to personnel in the Control Room and TSC.

- TS 5.5.19, "Control Room Envelope Habitability Program," is revised to replace "whole body or its equivalent to any part of the body" with "TEDE," which is the dose criteria specified in 10 CFR 50.67 (Reference 1).

2.3 Technical Specification Bases Changes

The TS Bases will be revised to reflect the licensing basis changes outlined in Section 2.1 and the TS changes identified in Section 2.2. A markup of the TS Bases changes is provided for information only in Attachment 3 to this Enclosure. These TS Bases changes will be implemented in accordance with TS 5.5.14, "Technical Specification (TS) Bases Control Program," upon NRC approval of this LAR.

2.4 Plant Changes

The following plant design modifications will be performed as part of AST implementation. These plant modifications support the AST analyses provided in Attachment 4.

- Install shielding material, equivalent to that provided by the Control Room outer walls, at the external concrete west wall of the Control Room briefing room.
- Install a high efficiency particulate air filter (HEPA) in the TSC normal ventilation system intake.
- Update setpoints for the redundant safety related gamma sensitive area radiation monitors (1-RE 25/26, 2-RE 25/26). These monitors are located at the Control Room normal intakes and are designed to automatically isolate the normal CRVS intakes and shift to CRVS Mode 4 (pressurized filtered emergency ventilation). These monitors are relied upon to perform their design function following a FHA in the FHB or Containment. Setpoints for these monitors are contained in the Offsite Dose Calculation Manual, which is controlled by TS 5.5.1, requirements.
- Reclassify a portion of the 40-inch Containment Penetration Area (GE/GW) Ventilation line from PG&E Design Class II to PG&E Design Class I and upgrade the damper actuators, pressure switches, and the damper solenoid valves to PG&E Design Class I. See Attachment 4, Section 5.2 for further details.
- Reclassify a portion of the 2-inch gaseous radwaste system line which connects to the Plant Vent as PG&E Design Class I. This line is currently

classified as PG&E Design Class II. See Attachment 4, Section 5.2 for further details.

2.5 Procedure Changes

As part of AST implementation, the following procedural updates will include:

- Update Equipment Control Guideline (ECG) 42.1, "Refueling Operations – Decay Time," to lower the restriction on fuel movement from 100 hours to 72 hours post-shutdown.
- Update ECG 42.5, "Refueling Operations – Water Level – Reactor Vessel (Control Rods)" to reflect the FHA AST analysis assumptions.
- Update ECG 23.3, "Containment Ventilation System," to reflect changes to TS 3.6.3.
- Review and update, as necessary, the EOPs and operator training procedures to ensure that the requisite steps to select the least contaminated CRVS pressurization intake are in use throughout the event. This review is to be performed as verification, since the EOPs currently include steps to select the least contaminated CRVS intake.
- Update Surveillance Test Procedure M-57, "Control Room Ventilation System (CRVS) Tracer Gas Test," to include the new Control Room inleakage test acceptance criteria and the range of CRVS ventilation flows deemed acceptable by the AST dose consequence analyses.
- Update the TSC administrative procedures to ensure that:
  a. The nominal normal operation TSC ventilation air intake flowrate is 500 cubic feet per minute (cfm).
  b. Following a LOCA, the TSC will be manually placed in Mode 4 operation such that filtered pressurization and recirculation can be credited within 2 hours of accident initiation.
  c. The nominal post-LOCA TSC ventilation filtered pressurization and recirculation flowrates are 500 cfm, respectively.
- Review EOPs to verify valve alignment information to manually initiate containment spray in the recirculation mode. Update EOPs to include direction to perform the realignment action within 12 minutes of termination of injection spray to ensure that the duration of spray operation (injection plus recirculation) exceeds 6.25 hours following the event. An associated TCOA will be implemented.
- Update ESF system leak testing procedures that are controlled by TS 5.5.2, "Primary Coolant Sources Outside Containment," to establish administrative acceptance criteria to ensure:

a. The total as-tested leakage from ESF systems that recirculate sump fluid outside containment is less than 126 cubic centimeters per minute (cc/min), with the following breakdown:

   i. In areas covered by the ABVS, the as-tested leakage is less than 120 cc/min,

   ii. In the containment penetration area, the as-tested leakage is less than 6 cc/min.

b. The total as-tested back leakage into the RWST from the containment recirculation sump is less than 1 gpm.

c. The total as-tested flow hard piped to the MEDT is less than the following values:

   i. Leakage from systems carrying non-radioactive fluids is less than 484 cc/min.

   ii. Leakage from ESF systems that recirculated sump fluids is less than 950 cc/min.

- Review and update the Emergency Plan to reflect AST, as necessary. There will be no change to the Emergency Planning Zone.

2.6 Updated Final Safety Analysis Report Changes

The UFSAR will be revised to reflect AST dose consequence analyses and the proposed licensing basis changes outlined in Section 2.1. UFSAR changes are provided in Attachment 8, for information only.

2.7 Presentation of Current Licensing Basis and Alternative Source Term Analysis Inputs

This section provides a summary of changes from current design and licensing basis analysis input values to revised AST inputs for each analysis.

The NRC's traditional methods for calculating the radiological consequences of DBAs are described in a series of regulatory guides and SRP chapters. That guidance was developed to be consistent with TID-14844 source term and the whole body and thyroid dose guidelines stated in 10 CFR 100.11. Many of those analysis assumptions and methods are inconsistent with the AST and with the TEDE criteria provided in 10 CFR 50.67. In addition, many of DCPP's analyses pre-date SRP guidance.

RG 1.183 provides assumptions and methods that are acceptable to the NRC staff for performing design basis radiological analyses using an AST. As stated in RG 1.183, the RG 1.183 guidance supersedes corresponding radiological analysis assumptions provided in other regulatory guides and SRP chapters

when used in conjunction with an approved AST and TEDE criteria provided in 10 CFR 50.67 (Reference 1).

DCPP used the guidance provided by RG 1.183 for analysis assumptions and methods for design basis radiological analyses. Conformance to RG 1.183 guidance is presented in Attachment 5. A summary of design inputs, assumptions, and methodology used in the AST analyses is provided in Chapter 7 of Attachment 4. Appendix B to Attachment 4 provides a comparison between the design input values used in the current licensing basis dose consequence analyses and the values used to support the AST analyses supporting this LAR.

As noted above and in Attachment 4, many of DCPP's analyses pre-date SRP guidance. Specifically, the DCPP's current licensing basis for the LRA, CREA, and LOL events are DCPP-specific with pre-SRP assumptions and only address offsite dose consequences. Updated analyses performed for AST now include Control Room doses.

3. TECHNICAL EVALUATION

The DCPP Units 1 and 2 current licensing basis for the radiological consequences analyses of accidents is based on source term methodology and assumptions derived from TID-14844 (Reference 4). An application for the selective use of AST for the FHA in the FHB was reviewed and approved by the NRC in its SER for License Amendment Nos. 163 and 165 (Reference 2). The design basis accidents are discussed in Chapter 15 of the DCPP UFSAR (Reference 12). The current DCPP dose consequences of design basis events, other than the FHA in the FHB, are based on acceptance criteria stated in 10 CFR Part 100 and 10 CFR 50, Appendix A, GDC 19, 1971. The current licensing basis for the radiological consequences of the FHA in the FHB is 10 CFR 50.67, "Accident Source Term."

The AST and methodology described in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," (Reference 25) and in RG 1.183 (Reference 3), provide regulatory guidance for the implementation of the AST. Revision of a plant licensing basis from the TID-14844 (Reference 4) source term to an AST involves the preparation of dose consequence analyses. Demonstration that the results satisfy the regulatory acceptance criteria and NRC approval of the requested change establishes the acceptability of the use of the AST for DCPP.

DCPP has performed radiological consequence analyses of the DBAs documented in Chapter 15 of the DCPP UFSAR that potentially result in the most significant Control Room and offsite exposures. These analyses were performed to support full scope implementation of AST. The AST analyses have been performed in accordance with the guidance in RG 1.183 and SRP Section 15.0.1

(Reference 23). Acceptance criteria consistent with those required by 10 CFR 50.67 and RG 1.183, Table 6, were used to replace the current design basis source term acceptance criteria. This represents a full implementation of AST in which the RG 1.183 source term will become the licensing basis for DCPP DBAs.

The technical justification for full implementation of the AST methodology, as defined in RG 1.183, of the DCPP DBA analyses is provided in Attachment 4. The following DBAs are addressed:

- LOCA
- FHA in the Containment
- FHA in the FHB (reanalysis)
- LRA
- CREA
- MSLB
- SGTR
- LOL Event

Conformance to RG 1.183 is documented in Attachment 5. The AST as defined in RG 1.183 has been incorporated into the DCPP site boundary and Control Room dose re-analyses discussed in Attachment 4. The estimated DCPP dose consequences for all design basis events addressed in RG 1.183, meet the acceptance criteria specified in 10 CFR 50.67 and RG 1.183, as shown in Table 1 for offsite locations and Control Room personnel.

In addition, the TSC dose is re-analyzed in Attachment 4. In accordance with current licensing basis, the 30-day integrated dose to an operator in the TSC due to immersion, inhalation, and direct shine is evaluated for the LOCA. The resultant post-LOCA dose is estimated to be 4.1 rem TEDE, which is within the acceptance criteria of 5 rem TEDE.

As stated in Section 2.4, some changes to the facility are required to implement AST for Control Room and TSC dose consequences. These changes include additional shielding for the Control Room, setpoint change for radiation monitors 1-RE 25/26 and 2-RE 25/26, and component upgrades (damper actuators, pressure switches, and damper solenoid valves) for a portion of the 40-inch Containment Penetration Area (GE/GW) Ventilation system. A HEPA filter will be installed at the normal intake of the TSC ventilation system.

The TSC is designed to meet NUREG-0696, Functional Criteria for Emergency Response Facilities, (Reference 27), which states that the TSC ventilation system need not be seismic Category 1 qualified, redundant, instrumented in the Control Room, or automatically activated to fulfill its role. Thus, the addition of the HEPA filter is not considered a safety-related component.

The Engineering Change Package supporting AST Implementation will ensure that any Design Class I equipment located adjacent to the ABVS and CRVS filters and the new TSC filter are qualified to any potential increase in the estimated total integrated radiation dose resulting from the additional post-LOCA radiological release pathways and higher χ/Q values addressed in this application.

Based on the preceding paragraphs, the methodology and dose consequence analyses presented in Attachment 4 do not rely on any newly installed safety-related systems, structure, or components. The containment spray system will now be credited during sump water recirculation following a LOCA for dose mitigation, but DCPP is already licensed for recirculation containment spray operation. Therefore, there are no additions to the EQ list or RG 1.97 instrumentation list.

No changes have been made in the system responses to accidents. Therefore, there are no additional or new emergency diesel generator (EDG) loads and the timing of the EDG loads did not change as a result of AST.

4. REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

<u>Title 10 Code of Federal Regulations Section 50.36, "Technical specifications"</u>

10 CFR 50.36:

(c) Technical specifications will include items in the following categories:

2) *Limiting conditions for operation.*

(i) Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met.

LCO 3.4.16, "RCS Specific Activity," provides the limiting condition for operation of the Reactor Coolant System (RCS) DEI and DEX. The limit established for DEI is not being revised by this LAR; however, the definition of DEI in TS 1.1, "Definition," is being revised to reference Table 2.1 of FGR No. 11 (Reference 15) as the only acceptable dose conversion factors for determining DEI. Thus, the definition of DEI will reference the same dose conversion factors used to

determine the reactor coolant dose equivalent iodine curie content for the MSLB and SGTR analysis, as requested by NRC RIS 2006-04 (Reference 8).

(ii) A technical specification limiting condition for operation of a nuclear reactor must be established for each item meeting one or more of the following criteria:

(A) Criterion 1. Installed instrumentation that is used to detect, and indicate in the Control Room, a significant abnormal degradation of the reactor coolant pressure boundary.

(B) Criterion 2. A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

LCO 3.6.3, "Containment Isolation Valves," Note 1 is revised so that the containment purge supply and exhaust flow paths are sealed closed during operating MODES 1, 2, 3, and 4. This change is in support of the LOCA dose analysis assumptions that the containment purge supply and exhaust paths are closed and therefore, not a release path for radionuclides present in the containment following an accident prior to containment isolation. Therefore, this proposed TS change will eliminate a potential radiological release pathway.

(C) Criterion 3. A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or integrity of a fission product barrier.

LCO 3.6.3 is part of the success path which functions to mitigate a LOCA. The revision to LCO 3.6.3, Note 1 support the LOCA dose analysis assumption that the 48-inch valves in the containment purge supply and exhaust paths are sealed closed and therefore, are not a release path for radionuclides present in the containment following a LOCA prior to containment isolation.

3) *Surveillance requirements.* Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

SR 3.4.16.1 is revised to reduce the specific activity of DEX. The revised AST analyses that base the released radioactive source terms on RCS specific activity uses a limit of less than or equal to 270μCi/gm DEX. The limit has been

reduced by DCPP to control the noble gas activity in the coolant to levels below the design basis values.

SR 3.6.3.1, 3.6.3.2, and 3.6.3.7 are being revised to support the revision to LCO 3.6.3 to ensure that the LCO will be met.

5) *Administrative Controls.* Administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner

TS 5.5.9, "Steam Generator (SG) Tube Inspection Program," is revised to change the accident induced leakage performance criteria. The primary-to-secondary accident induced leakage rate for any design basis accident, other than a SGTR, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs. Except during a SGTR, leakage is not to exceed 0.75 gpm total for all four SGs. The revised testing criterion for the primary-to-secondary leakage is more restrictive than the current testing criteria and represents the leakage rate assumed in the AST dose consequence analyses.

TS 5.5.11, "Ventilation Filter Testing Program (VFTP)," is revised to change the allowable methyl iodine penetration testing criteria for the ABVS charcoal filter from 15 percent to 5 percent. The allowable methyl iodide penetration is used to determine charcoal filter efficiency for removing iodine from atmospheric releases. This proposed revision will support the LOCA dose analysis assumptions with respect to the releases from an RHR pump seal passive failure, and has been demonstrated to be acceptable.

TS 5.5.19, "Control Room Envelope Habitability Program," is revised to replace "whole body or its equivalent to any part of the body" to "TEDE," to be consistent with the dose criteria specified in 10 CFR 50.67 for AST.

In summary, the changes proposed in this LAR in Section 2 support the AST analysis assumptions and have been demonstrated to be acceptable. The AST analysis results meet the dose criteria specified in 10 CFR 50.67 and Table 6 of RG 1.183, therefore the requirements of 10 CFR 50.36 continue to be met.

Title 10 Code of Federal Regulations Section 50.67, "Alternate Source Term"

On December 23, 1999, the NRC published 10 CFR 50.67, "Accident Source Term," in the Federal Register. This regulation provides a mechanism for licensed power reactors to replace the current accident source term used in the DBA analyses with an AST. The direction provided in 10 CFR 50.67 is that licensees who seek to revise their current accident source term in design basis radiological consequence analyses shall apply for a LAR under 10 CFR 50.90. Thus, this LAR meets 10 CFR 50.67.

General Design Criteria

The construction of DCPP Units 1 and 2 was significantly complete prior to issuance of 10 CFR 50, Appendix A GDC. DCPP was designed and constructed to comply with Atomic Energy Commission GDC as proposed on July 10, 1967 (AEC GDC), except as noted and described in the DCPP UFSAR Chapter 3.

Criterion 19, 1971 – Control Room, describes provisions for a Control Room that provides adequate radiation protection to permit access and occupancy under accident conditions. The dose criterion of GDC 19 was modified to 5 rem TEDE in 1999 to be consistent with 10 CFR 50.67. The results from the dose analyses using AST source terms and methodologies show that the predicted dose consequence results are within the allowable regulatory limits of 10 CFR 50.67 and GDC 19, 1999. DCPP will conform to GDC 19, 1999, for dose only, for Control Room dose limits of 5 rem TEDE upon implementation of AST. Thus, with the changes proposed in this LAR, DCPP will continue to meet the requirements of 10 CFR 50, Appendix A, GDC Criterion 19.

Criterion 52, 1967 – Containment Heat Removal Systems (Category A), describes two functions of the containment spray system. One function is to facilitate heat removal from the containment following an accident. The second function is to remove radioactive iodine isotopes from the containment atmosphere should these fission products be released in the event of an accident. The AST LOCA dose analysis assumes that the containment spray system now operates during the mode of operation that recirculates containment sump water for dose mitigation. This change in containment spray system operation assumptions in the LOCA dose consequence analysis does not change the function of the containment spray system or the actual operation of containment spray system. DCPP is not crediting the containment spray system in the recirculation mode for heat removal and thus not changing the licensing commitment to Criterion 52, 1967. Therefore Criterion 52, 1967, continues to be met and the plant will continue to provide the basis for safe plant operation.

Criterion 54, 1971 - Piping Systems Penetrating Containment, describes requirements for isolation, including leak detection, and periodic testing. No new containment penetrations or lines penetrating the containment are being proposed with this LAR. Changes to TS 3.6.3 concerning the containment purge supply and exhaust paths isolation, including leak detection surveillances, includes an enhanced requirement to ensure that these lines remain sealed closed during MODES 1, 2, 3, and 4 operation. Therefore Criterion 54, 1971, continues to be met and the piping systems penetrating containment will continue to provide the basis for safe plant operation.

Criterion 56, 1971– Primary Containment Isolation, describes the provisions for providing isolating lines that penetrate the containment. No new containment

penetrations or lines penetrating the containment are being proposed with this LAR. Changes to TS 3.6.3 concerning the containment purge supply and exhaust paths isolation, includes an enhanced requirement to ensure that these lines remain sealed closed during MODES 1, 2, 3, and 4 operation. Therefore Criterion 56, 1971, continues to be met.

Criterion 58, 1967 – Inspection of Containment Pressure-Reducing Systems (Category A), describes the design provision requirements to facilitate periodic physical inspection of components of the containment pressure-reducing systems. UFSAR 3.1.8.22 provides a discussion on how DCPP meets Criterion 58, 1967, with a brief discussion of the containment pressure-reducing systems. Containment spray is a containment pressure reducing system and the description states that during the recirculation phase, containment spray operation is not required. While this statement remains correct for the containment pressure-reducing function of containment spray, due to the timing of fission product releases associated with AST, containment spray will be required during the recirculation phase for fission product cleanup. This LAR does not change how DCPP meets Criterion 58, 1967; therefore, the criterion continues to be met.

Criterion 59, 1967 – Testing of Containment Pressure-Reducing Systems Components (Category A), discusses how the active components of the containment pressure-reducing systems are to be tested periodically for operability and required functional performance. UFSAR 3.1.8.23 provides a discussion on how DCPP meets Criterion 59, 1967. Containment spray is a containment pressure reducing system; however, no changes are being made to the system. The AST LOCA dose analysis now credits the operation of the system during recirculation for dose mitigation. Therefore, Criterion 59, 1967, continues to be met.

Criterion 60, 1967 – Testing of Containment Spray Systems (Category A), discusses the capability to test the delivery capability of the containment spray system at a position as close to the spray nozzles as practical. UFSAR 3.1.8.24 provides a discussion on how DCPP meets Criterion 60, 1967. No changes are being made to the containment spray system. The only change is that the AST LOCA dose analysis now credits the operation of the system during recirculation for dose mitigation. Therefore, Criterion 60, 1967, continues to be met.

Criterion 62, 1967 – Inspection of Air Cleanup Systems (Category A), discusses the physical inspection of containment air cleanup systems, such as ducts, filters, fans, and dampers. UFSAR 3.1.8.26 provides a discussion on how DCPP meets Criterion 62, 1967. The containment spray system, using sodium hydroxide, serves as the air cleanup system. No changes are being made to the containment spray system. The only change is that the AST LOCA dose analysis

now credits the operation of the system during recirculation for dose mitigation. Therefore, Criterion 62, 1967, continues to be met.

Criterion 63, 1967 – Testing of Air Cleanup Systems Components (Category A), discusses the provisions for testing containment air cleanup systems, such as ducts, filters, fans, and dampers. UFSAR 3.1.8.27 provides a discussion on how DCPP meets Criterion 63, 1967. The containment spray system, using sodium hydroxide, serves as the air cleanup system. No changes are being made to the containment spray system. The only change is that the AST LOCA dose analysis now credits the operation of the system during recirculation for dose mitigation. Therefore, Criterion 63, 1967, continues to be met.

Criterion 64, 1967 – Testing of Air Cleanup Systems (Category A), discusses the provisions for testing containment air cleanup systems. UFSAR 3.1.8.28 provides a discussion on how DCPP meets Criterion 64, 1967. The containment spray system serves as the air cleanup system. No changes are being made to the containment spray system. The only change is that the AST LOCA dose analysis now credits the operation of the system during recirculation for dose mitigation. Therefore, Criterion 64, 1967, continues to be met.

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

The AST methodology used to perform the dose consequence analyses for DCPP is consistent with the guidance of RG 1.183 (Reference 3). Documentation of conformance to RG 1.183 is presented in Attachment 5, with cross-references to specific sections of Attachment 4, where more detail is provided.

RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," dated June 2003.

RG 1.194, dated June 2003 (Reference 19), Regulatory Position C.1 through C.3, and the adjustment factor for vertically orientated energetic releases from steam relief valves and ADVs allowed by Regulatory Position C.6 are used to determine short-term onsite χ/Q values in support of design basis radiological habitability assessments. Credit is taken for DCPP's dual intake design for the Control Room pressurization air intakes per Regulatory Position C.3.3.2.3.

PG&E takes an exception to Regulatory Position C.3.4, for two specific Control Room receptors (9.4 meters for Unit 1 containment building to Unit 1 Control Room normal intake and 7.8 meters for Unit 2 containment building to Unit 2 Control Room normal intake). Use of ARCON96 methodology for these two release point-to-receptor distances is considered acceptable since the dominating factors in the calculation are building cross-sectional area and plume meander and not the normal atmospheric dispersion coefficients. Note that the

χ/Q values for these two release-receptor cases were developed to establish the bounding χ/Q values. However, the χ/Q values for these two cases were not bounding, and therefore not used in the dose consequence analyses. See section 2.1 and Attachment 4, Section 5.2 for further detail.

RG 1.145, Revision 1, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," dated February 1983.

The methodology outlined in RG 1.145, Revision 1, is used for calculating ground level releases to determine the short-term χ/Q values for the EAB and LPZ for design basis radiological analyses. All releases are conservatively treated as ground level releases, therefore Regulatory Positions C.1.3.2, C.2.1.2, and C.2.2.2 associated with elevated or stack releases are not applicable to DCPP.

10 CFR 50.34

10 CFR 50.34(b) specifies content requirements for the UFSAR including evaluations required to show that accident dose criteria are met. Attachment 8 contains UFSAR changes (for information only) to support AST implementation. Upon approval of this LAR, UFSAR changes will be made to fulfill these requirements.

4.2 Precedent

The NRC has previously approved implementation of the AST methodology at a number of nuclear power plants. In a LAR dated April 26, 2004, PSEG Nuclear LLC, proposed to adopt the AST methodology for Salem Units 1 and 2 (Reference 28, ADAMS Accession No. ML041280067). The DCPP LAR is similar to the PSEG submittal in that PSEG also proposed to credit recirculation sprays following the LOCA for long term containment iodine removal. PSEG also adjusted the Control Room assumed in-leakage by replacing it with values based upon their Tracer Gas Test. The NRC reviewed and approved the AST LAR for PSEG in a SER dated February 17, 2006 (Reference 29, ADAMS Accession No. ML060040322).

In a LAR dated June 5, 2002, FirstEnergy Nuclear Operating Company proposed to adopt the AST methodology for Beaver Valley Power Station Units 1 and 2 (Reference 30, ADAMS Accession No. ML021620298). The Beaver Valley amendment used CBI S&W Proprietary computer codes, listed in Section 3 of Attachment 4, in similar applications. The NRC reviewed and approved the AST license amendment, including the use of the SBI S&W Proprietary computer codes, in a SER dated September 10, 2003 (Reference 31, ADAMS Accession No. ML032530204).

In a LAR dated February 7, 2001, Omaha Public Power District proposed to adopt the AST methodology for Fort Calhoun Station Unit 1 (Reference 32,

ADAMS Accession No. ML010400079). The Fort Calhoun amendment used CBI S&W Proprietary computer codes, listed in Section 3 of Attachment 4, in similar applications. The NRC reviewed and approved the AST license amendment, including the use of the SBI S&W Proprietary computer codes, in a SER dated December 5, 2001 (Reference 6, ADAMS Accession No. ML013030027).

In a LAR dated September 26, 2002, Millstone Unit 2 proposed alternate non-LOCA gap fractions similar to the non-LOCA gap fractions DCPP proposes in Section 4.3 of Attachment 4, as enhancement to RG 1.183 for higher burnup fuel (Reference 33, ADAMS Accession No. ML023040334). The NRC reviewed and approved the AST license amendment, including the use of the non-LOCA gap fractions, in a SER dated September 20, 2004 (Reference 34, ADAMS Accession No. ML042360671).

Similarly, in a LAR dated June 5, 2002, Indian Point Unit 3 proposed alternate non-LOCA gap fractions similar to the non-LOCA gap fractions DCPP proposes in Section 4.3 of Attachment 4 (Reference 35, ADAMS Accession No. ML021840136). The NRC reviewed and approved the AST license amendment, including the use of the non-LOCA gap fractions, in a SER dated March 17, 2003 (Reference 36, ADAMS Accession No. ML030760135).

4.3 No Significant Hazards Consideration

As provided by 10 CFR 50.67, Pacific Gas & Electric (PG&E) is implementing the use of an Alternative Source Term (AST) and the dose calculation methodology described in Regulatory Guide (RG) 1.183 to calculate the accident doses to the Control Room, Technical Support Center (TSC), and offsite receptors following postulated design basis events that result in the release of radioactive material from reactor fuel at Diablo Canyon Power Plant (DCPP) Units 1 and 2. The AST and associated methodology for full implementation of AST define the amount, isotopic composition, physical and chemical characteristics, and timing of radioactive material releases following postulated events. Transport of the material to the Control Room, TSC, and offsite areas is modeled, and the resulting Total Effective Dose Equivalent (TEDE) is determined. Regulatory acceptance criteria account for the sum of the deep-dose equivalent (for external exposures) and the committed effective dose equivalent (for internal exposures). In accordance with 10 CFR 50.67(b), licensees wishing to adopt an AST must apply for a license amendment in accordance with 10 CFR 50.90.

In support of the revised analyses applying AST, the following Technical Specification (TS) changes are being made: the definition for Dose Equivalent Iodine-131 (DEI) is revised to be consistent with AST dose conversion factor usage, the limit for reactor coolant system Dose Equivalent Xenon-133 (DEX) activity is decreased to control the noble gas activity in the coolant to levels below the design basis values, the requirement for containment penetrations is

revised to require the 48-inch containment purge supply and exhaust valves to be sealed closed during operation MODES 1, 2, 3, and 4 eliminating a potential dose contribution release path, the accident induced leakage performance criterion for the steam generator tube inspection program is revised to be more restrictive, and the testing requirement for the auxiliary building ventilation system charcoal filter is also revised to be more restrictive. Other changes to the TSs involve the adoption of terminology on which AST is based.

AST methods have been utilized in the analysis of the limiting design basis accidents, as follows: loss of coolant accident (LOCA), fuel handing accident (FHA) in the containment and in the fuel handling building, locked rotor accident (LRA), control rod ejection accident (CREA), main steam line break (MSLB), and steam generator tube rupture (SGTR). AST methods have also been utilized in the analysis of the limiting Condition II event, the loss of load (LOL) accident. Other changes incorporated in the revised analyses include revising atmospheric dispersion factors (χ/Q), reducing the minimum decay time before fuel movement, adding shielding to the Control Room for additional protection of Control Room personnel and adding a high efficiency particulate air (HEPA) filter for additional protection of TSC personnel. In addition, a portion of the 40-inch Containment Penetration Area Ventilation line and a portion of the 2-inch gaseous radwaste system line which connect to the Plant Vent are being reclassified from PG&E Design Class II to PG&E Design Class I. Because AST methodologies better represent the physical characteristics and timing of the radionuclide release following a postulated LOCA, containment spray is now relied upon during the recirculation of sump water for continued removal of iodine and particulate from the containment atmosphere for spray duration (injection plus recirculation) greater than 6.25 hours. In addition, setpoint changes are being made to the Control Room intake radiation monitors to incorporate the effect of all possible release points from a FHA.

PG&E has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   This license amendment does not physically impact any system, structure, or component (SSC) that is a potential initiator of an accident. Therefore, implementation of AST, the AST assumptions and inputs, the proposed TS changes, and new χ/Q values have no impact on the probability for initiation of any design basis accident. Once the occurrence of an accident has been postulated, the new accident source term and χ/Q

values are inputs to analyses that evaluate the radiological consequences of the postulated events.

Reactor coolant specific activity, testing criteria of charcoal filters, and the accident induced primary-to-secondary system leakage performance criterion are not initiators for any accident previously evaluated. The proposed change to require the 48-inch containment purge valves to be sealed closed during operating MODES 1, 2, 3, and 4 is not an accident initiator for any accident previously evaluated. The change in the classifications of a portion of the 40-inch Containment Penetration Area Ventilation line and a portion of the 2-inch gaseous radwaste system line is also not an accident initiator for any accident previously evaluated. Thus, the proposed TS changes and AST implementation will not increase the probability of an accident.

The change to the decay time prior to fuel movement is not an accident initiator. Decay time is used to determine the source term for the dose consequence calculation following a potential FHA and has no effect on the probability of the accident. Likewise, the change to the Control Room radiation monitors setpoint cannot cause an accident and the operation of containment spray during the recirculation phase is used for mitigation of a LOCA, and thus not an accident initiator.

As a result, there are no proposed changes to the parameters or conditions that could contribute to the initiation of an accident previously evaluated in Chapter 15 of the Updated Final Safety Analysis Report (UFSAR). As such, the AST cannot affect the probability of an accident previously evaluated.

Regarding accident consequences, equipment and components affected by the proposed changes are mitigative in nature and relied upon once the accident has been postulated. The license amendment implements a new calculation methodology for determining accident consequences and does not adversely affect any plant component or system that is credited to mitigate fuel damage. Subsequently, no conditions have been created that could significantly increase the consequences of any accidents previously evaluated.

Requiring that the 48-inch containment purge supply and exhaust valves be sealed closed during operating MODES 1, 2, 3, and 4 eliminates a potential path for radiological release following events that result in radioactive material releases to the containment, thus reducing potential consequences of the event. The steam generator tube inspection testing criterion for accident induced leakage is being changed, resulting in lower leakage rates, and thus less potential releases due to primary-to-

secondary leakage. The auxiliary building ventilation system allowable methyl iodide penetration limit is being changed, which results in more stringent testing requirements, and thus higher filter efficiencies for reducing potential releases.

Changes to the operation of the containment spray system to require operation during the recirculation mode are also mitigative in nature. While the plant design basis has always included the ability to implement containment spray during recirculation, this license amendment now requires operation of containment spray in the recirculation mode for dose mitigation. DCPP is designed and licensed to operate using containment spray in the recirculation mode. As such, operation of containment spray in the recirculation mode has already been analyzed, evaluated, and is currently controlled by Emergency Operating Procedures. Usage of recirculation spray reduces the consequence of the postulated event. Likewise, the additional shielding to the Control Room and the addition of a HEPA filter to the TSC ventilation system reduces the consequences of the postulated event to the Control Room and TSC personnel. Lowering the limit for DEX lowers potential releases. By reclassifying a portion of the 40-inch Containment Penetration Area Ventilation line and a portion of the 2-inch gaseous radwaste system line to PG&E Design Class I, these lines will be seismically qualified, thus assuring that post-LOCA release points are the same as those used for determining χ/Q values.

The change to the decay time from 100 hours to 72 hours prior to fuel movement is an input to the FHA. Although less decay will result in higher released activity, the results of the FHA dose consequence analysis remain within the dose acceptance criteria of the event. Also, the radiation levels to an operator from a raised fuel assembly may increase due to a lower decay time, however, any exposure will continue to be maintained under 10 CFR 20 limits by the plant Radiation Protection Program.

Plant-specific radiological analyses have been performed using the AST methodology, assumption and inputs, as well as new χ/Q values. The results of the dose consequences analyses demonstrate that the regulatory acceptance criteria are met for each analyzed event. Implementing the AST involves no facility equipment, procedure, or process changes that could significantly affect the radioactive material actually released during an event. Subsequently, no conditions have been created that could significantly increase the consequences of any of the events being evaluated.

Based on the above discussion, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different accident from any accident previously evaluated?

   Response: No.

   This license amendment does not alter or place any SSC in a configuration outside its design or analysis limits and does not create any new accident scenarios.

   The AST methodology is not an accident initiator, as it is a method used to estimate resulting postulated design basis accident doses. The proposed TS changes reflect the plant configuration that supports implementation of the new methodology and supports reduction in dose consequences. DCPP is designed and licensed to operate using containment spray in the recirculation mode. This change will not affect any operational aspect of the system or any other system, thus no new modes of operation are introduced by the proposed change.

   The function of the radiation monitors has not changed; only the setpoint has changed as a result of an assessment of all potential release pathways. The continued operation of containment spray and the radiation monitor setpoint change do not create any new failure modes, alter the nature of events postulated in the UFSAR, nor introduce any unique precursor mechanism.

   Requiring the 48-inch containment purge valves to be sealed closed during operating MODES 1, 2, 3, and 4 does not introduce any new accident precursor. This change only eliminates a potential release path for radionuclides following a LOCA.

   The proposed TS testing criteria for the auxiliary building ventilation system charcoal filters and the proposed performance criteria for steam generator tube integrity also cannot create an accident, but results in requiring more efficient filtration of potentially released iodine and less allowable primary-to-secondary leakage. The proposed changes to the DEX activity limit, the TS terminology, and the decay time of the fuel before movement are also unrelated to accident initiators.

   The only physical changes to the plant being made in support of AST is the addition of Control Room shielding in an area previously modified, the addition of a HEPA filter at the intake of the TSC normal ventilation system, and the upgrade to the damper actuators, pressure switches, and

damper solenoid valves to support reclassifying a portion of the Containment Penetration Area Ventilation line to PG&E Design Class I. Both Control Room shielding and HEPA filtration are mitigative in nature and do not have any impact on plant operation or system response following an accident. The Control Room modification for adding the shielding will meet applicable loading limits, so the addition of the shielding cannot initiate a failure. Upgrading damper actuators, pressure switches, and damper solenoid valves involve replacing existing components with components that are PG&E Design Class I. Therefore, the addition of shielding, a HEPA filter, and upgrading components cannot create a new or different kind of accident.

Since the function of the SSCs has not changed for AST implementation, no new failure modes are created by this proposed change. The AST change itself does not have the capability to initiate accidents.

Therefore, the proposed change does not create the possibility of a new or different type of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

Implementing the AST is relevant only to calculated dose consequences of potential design basis accidents evaluated in Chapter 15 of the UFSAR. The changes proposed in this license amendment involve the use of a new analysis methodology and related regulatory acceptance criteria. New atmospheric dispersion factors, which are based on site specific meteorological data, were calculated in accordance with regulatory guidelines. The proposed TS, TS Bases, and UFSAR changes reflect the plant configuration that will support implementation of the new methodology and result in operation in accordance with regulatory guidelines that support the revisions to the radiological analyses of the limiting design basis accidents. Conservative methodologies, per the guidance of RG 1.183, have been used in performing the accident analyses. The radiological consequences of these accidents are all within the regulatory acceptance criteria associated with the use of AST methodology.

The change to the minimum decay time prior to fuel movement results in higher fission product releases after a FHA. However, the results of the FHA dose consequence analysis remain within the dose acceptance criteria of the event.

The proposed changes continue to ensure that the dose consequences of design basis accidents at the exclusion area, low population zone boundaries, in the TSC, and in the Control Room are within the corresponding acceptance criteria presented in RG 1.183 and 10 CFR 50.67. The margin of safety for the radiological consequences of these accidents is provided by meeting the applicable regulatory limits, which are set at or below the 10 CFR 50.67 limits. An acceptable margin of safety is inherent in these limits.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above evaluation, PG&E concludes that the proposed change does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of "no significant hazards consideration" is justified.

4.4 Conclusions

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

5. ENVIRONMENTAL CONSIDERATION

PG&E has evaluated the proposed amendment and has determined that the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure.

Based on the evaluation performed under the standards set forth in 10 CFR 50.92(c), PG&E concludes that the proposed amendment does not involve a significant hazards consideration. AST only involves a change in accident dose calculation inputs and methodology. Calculated doses meet TEDE criteria.

No aspect of implementing the AST involves facility equipment, procedure, or process changes that would increase actual onsite doses if an event were to occur.

The AST does not result in actual or calculated changes in the normal radiation levels in the facility or in the type or quantity of radioactive materials processed

during normal operation. Implementation of the AST also has no effect on the actual or calculated effluents arising from normal operation.

Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6. REFERENCES

1. Code of Federal Regulations (CFR), 10 CFR 50.67, "Accident Source Terms."
2. NRC Letter "Diablo Canyon Power Plant, Units 1 and 2 - Issuance of Amendments RE: Control Room, Auxiliary Building, and Fuel Handling Building Ventilation Systems (TAC Nos. MB8485 and MB8486)," dated February 27, 2004.
3. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.
4. Technical Information Document 14844, "Calculation of Distance Factors for Power and Test Reactor Sites," 1962.
5. NUREG-0933, "Resolution of Generic Safety Issues," dated December 2011.
6. NRC Letter "Fort Calhoun Station, Unit No. 1 – Issuance of Amendment (TAC No. MB1221)," dated December 5, 2001 (ADAMS Accession No. ML013030027).
7. SECY-98-154, "Results of the Revised (NUREG-1465) Source Term Rebaselining For Operating Reactors," dated June 30, 1998.
8. NRC Regulatory Issue Summary (RIS) 2006-04, "Experience with Implementation of Alternative Source Terms," dated March 7, 2006.
9. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," dated June 1987.
10. Code of Federal Regulations 10 CFR 50.44, "Combustible Gas Control for Nuclear Power Reactors."
11. NRC Letter "Diablo Canyon Power Plant, Unit 1 (TAC No. MC1678) and Unit No. 2 (TAC No. MC1679) – Issuance of Amendments RE: Elimination of Requirements for Hydrogen Recombiners and Hydrogen Monitors," dated May 4, 2004.
12. Diablo Canyon Power Plant Updated Final Safety Analysis Report, Revision 21.
13. Code of Federal Regulations, 10 CFR 50, Appendix A, General Design Criteria 19, "Control Room," dated 1971.
14. NUREG-0737, Supplement 1, "Clarification of TMI Action Plan Requirements," dated January 1983.
15. Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," dated 1988.
16. Regulatory Guide 1.145, Revision 1, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," dated February 1983.

17. Regulatory Guide 1.111, Revision 1, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light Water Cooled Reactors," dated July 1977.

18. Ramsdell, J. V. Jr. and C. A. Simonen, "Atmospheric Relative Concentrations in Building Wakes." Prepared by Pacific Northwest Laboratory for the U.S. Nuclear Regulatory Commission, PNL-10521, NUREG/CR-6331, Revision 1, May 1997.

19. Regulatory Guide 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," dated June 2003.

20. NUCON International Inc., "Control Room Habitability Tracer Gas Leak Testing at Diablo Canyon Power Plant," dated December 2012.

21. NRC Letter "Diablo Canyon Power Plant, Unit Nos. 1 and 2 - Issuance of Amendments RE: Containment Spray During the Recirculation Phase of a LOCA (TAC Nos. MA1408 and MA1409)," dated February 9, 2000.

22. NUREG-0737, "Clarification of TMI Action Plan Requirements," dated November 1980.

23. NUREG-0800, Standard Review Plan 15.0.1, Revision 0, "Radiological Consequence Analyses using Alternative Source Terms," dated July 2000.

24. Improved Standard Technical Specification Traveler, TSTF-490, Revision 0, "Deletion of E Bar Definition and Revision to RCS Specific Activity Tech Spec," dated September 13, 2005.

25. NUREG-1431, Revision 4, Volume 1, "Standard Technical Specifications, Westinghouse Plants," dated April 2012.

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

27. NUREG-0696, "Functional Criteria for Emergency Response Facilities," dated February 1981.

28. PSEG Nuclear LLC Letter "Implementation of Alternative Source Term (AST) Request for Changes to Technical Specifications and Updated Final Safety Analysis Report Salem Nuclear Generating Station, Units 1 and 2, Facility Operating Licenses DBR-70 and DPR-75, Docket Nos. 50.272 and 50-311," dated April 26, 2004 (ADAMS Accession No. ML041280067).

29. NRC Letter "Salem Nuclear Generating Station, Unit Nos. 1 and 2, Issuance of Amendments RE: Alternate Source Term (TAC Nos. MC3094 and MC3095)," dated February 17, 2006, (ADAMS Accession No. ML060040322).

30. FirstEnergy Nuclear Operating Company Letter L-02-069 "Beaver Valley Power Station, Unit 1 No. 1 and No. 2 BV-1 Docket No. 50-334, License No. DPR-66, BV-2 Docket No. 50-412, License No. NPF-73, License Amendment Request Nos. 300 and 172," dated June 5, 2002 (ADAMS Accession No. ML021620298).

31. NRC Letter "Beaver Valley Power Station, Unit Nos. 1 and 2 - Issuance of Amendment RE: Selective Implementation of Alternate Source Term and Control Room Habitability Technical Specification Changes (TAC Nos. MB5303 and MB5304)," dated September 10, 2003 (ADAMS Accession No. ML032530204).

32. Omaha Public Power District Letter LIC-01-0010, "Application for Amendment of Operating License," dated February 7, 2001 (ADAMS Accession No. ML010400079).

33. Dominion Nuclear Connecticut, Inc. Letter, "Millstone Power Station, Unit No. 2, License Basis Document Change Request (LBDCR) 2-18-02, Selective Implementation of the Alternative Source Term – Fuel Handling Accident Analyses," dated September 26, 2002 (ADAMS Accession No. ML023040334).

34. NRC Letter "Millstone Power Station, Unit Nos. 2 – Issuance of Amendment RE: Selective Implementation of Alternate Source Term (TAC No. MB6479)," dated September 20, 2004 (ADAMS Accession No. ML042360671).

35. Entergy Nuclear Northeast Letter IPN-02-044, "Indian Point Nuclear Generating Unit No. 3, Docket No. 50-286, Proposed Changes to Technical Specifications: Selective Adoption of Alternate Source Term and Incorporation of Generic Changes; TSTF-51, TSTF-68, and TSTF-312," dated June 5, 2002 (ADAMS Accession No. ML021840136).

36. NRC Letter "Indian Point Nuclear Generating Unit No. 3 – Issuance of Amendment RE: Selective Adoption of Alternate Source Term (TAC No. MB5382)," dated March 17, 2003 (ADAMS Accession No. ML030760135).

Table 1 - AST Site Boundary and Control Room TEDE (rem)

Accident	EAB[1]	LPZ[2]	Regulatory Limit	Control Room	Regulatory Limit
Loss of Coolant Accident	5.6[4]	1	25	3.7 (0.7) [3]	5
Fuel Handling Accident in Fuel Handling Building	1.5	0.2	6.3	1.1	5
Fuel Handling Accident in Containment	1.5	0.2	6.3	4.7	5
Locked Rotor Accident	0.8	0.2	2.5	2.4	5
Control Rod Ejection Accident Containment Release Secondary Release	 0.7 0.7	 0.3 0.2	 6.3	 3.4 0.5	 5
Main Steam Line Break Pre-incident iodine Spike Accident-Initiated Iodine Spike	 0.1 0.7	 <0.1 0.2	 25 2.5	 2.0 4.1	 5
Steam Generator Tube Rupture Pre-incident iodine Spike Accident-Initiated Iodine Spike	 1.3 0.7	 0.1 <0.1	 25 2.5	 0.6 0.3	 5
Loss of Load Pre-incident iodine Spike Accident-Initiated Iodine Spike	 <0.1 <0.1	 <0.1 <0.1	 2.5 2.5	 <0.1 <0.1	 5

Notes

(1) EAB doses are based on worst 2-hour period following onset of accident. Except as noted, the maximum 2-hour dose period for the EAB dose for each of the accidents is the 0 to 2 hours' time period.
- LOCA: 24-26 hours (based on RHR Pump Seal Failure; see note 4 below for additional information)
- LRA: 8.73 to 10.73 hours
- MSLB (Accident-Initiated Spike case): 7.6 to 9.6 hours
- LOL (Accident-Initiated Spike case): 8.73 to 10.73 hours.

(2) LPZ Doses are based on the duration of the release.

(3) The dose presented represents the operator dose due to occupancy. Value shown in parenthesis represents that portion of the total dose reported that is the contribution of direct shine from contained sources/external cloud. The dose to the Control Room operator during routine access for the 30 day duration of the accident is discussed in Attachment 4, Section 7.2.6 and summarized in the text of Attachment 4, Section 8.0.

(4) The maximum 2 hour EAB dose is based on the assumed RHR pump seal passive failure resulting in a 50 gpm leak of sump water occurring at t=24 hours for 30 minutes. The RHR pump seal passive failure is considered a part of DCPP licensing basis with respect to passive system failure. If this assumed passive failure were not included, the maximum 2 hour dose at the EAB would occur between t=0.5 hours to t=2.5 hours (i.e., during the post-LOCA ex-vessel release phase) and would be 3.4 rem.

ATTACHMENT 1

Proposed Technical Specification Changes (MARKUP)

Changes are proposed to the following Technical Specifications:

1. Specification 1.1, Definitions, Dose Equivalent I-131
2. Specification 3.4.16, RCS Specific Activity
3. Specification 3.6.3, Containment Isolation Valves
4. Specification 5.5.9, Steam Generator (SG) Tube Inspection Program
5. Specification 5.5.11, Ventilation Filter Testing Program (VFTP)
6. Specification 5.5.19, Control Room Envelope Habitability Program

1.1 Definitions (continued)

DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) that alone would produce the same dose when inhaled as the combined activities of iodine isotopes I-131, I-132, I-133, I-134, and I-135 actually present. The determination of DOSE EQUIVALENT I-131 shall be performed using ~~thyroid dose conversion factors from Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites," or Table E-7 of Regulatory Guide 1.109, Revision 1, NRC, 1977, or International Commission on Radiological Protection (ICRP) Publication 30, 1979, Supplement to Part 1, pages 192-212, Table titled "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity," or~~ the committed thyroid dose conversion factors from Table 2.1 of EPA Federal Guidance Report No. 11, 1988, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."
DOSE EQUIVALENT XE-133	DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 (microcuries per gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-87, Kr-88, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT XE-133 shall be performed using effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil."

(continued)

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.16.1	------------------------------NOTE------------------------ Only required to be performed in MODE 1. -- Verify reactor coolant DOSE EQUIVALENT XE-133 specific activity ≤ ~~600.0~~270.0 μCi/gm.	In accordance with the Surveillance Frequency Control Program
SR 3.4.16.2	---------------------------NOTE------------------------------- Only required to be performed in MODE 1. -- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 μCi/gm.	In accordance with the Surveillance Frequency Control Program AND Between 2 and 6 hours after a THERMAL POWER change of ≥ 15% RTP within a 1 hour period.

3.6 CONTAINMENT SYSTEMS

3.6.3 Containment Isolation Valves

LCO 3.6.3 Each containment isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

-----NOTES-----

1. Penetration flow path(s) except ~~no more than two of three flow paths for containment purge supply and exhaust and containment vacuum/pressure relief paths at one time~~ for 48-inch purge valve 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 containment isolation valves.
4. Enter applicable Conditions and Required Actions of LCO 3.6.1, "Containment," when isolation valve leakage results in exceeding the overall containment leakage rate acceptance criteria.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. -----NOTE----- Only applicable to penetration flow paths with two containment isolation valves. ---------- One or more penetration flow paths with one containment isolation valve inoperable except for a containment purge supply and exhaust valve or pressure/vacuum relief valve leakage not within limit.	A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured. AND	4 hours (continued)

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.6.3.1	~~Not used~~Verify each 48 inch purge valve is sealed closed, except for one purge valve in a penetration flow path while in Condition D of this LCO.	In accordance with the Surveillance Frequency Control Program
SR 3.6.3.2	Verify each ~~48 inch containment purge supply and exhaust and~~ 12 inch vacuum/pressure relief valve is closed, except when these valves are open for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open.	In accordance with the Surveillance Frequency Control Program
SR 3.6.3.3	-----------------------------NOTE--------------------------- Valves and blind flanges in high radiation areas may be verified by use of administrative controls. --- Verify each containment isolation manual valve and blind flange that is located outside containment and not locked, sealed or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.	In accordance with the Surveillance Frequency Control Program
SR 3.6.3.4	-----------------------------NOTE--------------------------- Valves and blind flanges in high radiation areas may be verified by use of administrative means. --- Verify each containment isolation manual valve and blind flange that is located inside containment and not locked, sealed or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.	Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days
SR 3.6.3.5	Verify the isolation time of each automatic power operated containment isolation valve is within limits.	In accordance with the Inservice Testing Program
SR 3.6.3.6	Not used	

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.6.3.7	---NOTE--- This surveillance is not required when the penetration flow path is isolated by a leak tested blank flange. --- Perform leakage rate testing for containment purge supply and exhaust and vacuum/pressure relief valves with resilient seals.	In accordance with the Surveillance Frequency Control Program ~~AND~~ ~~For containment purge supply and exhaust valves only, within 92 days after opening the valve~~
SR 3.6.3.8	Verify each automatic containment isolation valve that is not locked, sealed or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.6.3.9	Not used	
SR 3.6.3.10	Verify each 12 inch containment vacuum/pressure relief valve is blocked to restrict the valve from opening > 50°.	In accordance with the Surveillance Frequency Control Program
SR 3.6.3.11	Not used	

5.5 Programs and Manuals (continued)

5.5.9 Steam Generator (SG) Tube Inspection Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

a. Provisions for condition monitoring assessments.

Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity.

SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs ~~and leakage rate for an individual SG~~. Except during a SG tube rupture, leakage is ~~also~~ not to exceed ~~1~~0.75 gallon per minute ~~per~~ total for all four SGs.

(continued)

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal absorber, 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 and at the relative humidity specified below. Laboratory testing shall be completed at least once per 24 months and after every 720 hours of charcoal operation.

ESF Ventilation System	Penetration	RH
Control Room	2.5%	95%
Auxiliary Building	~~1~~5.0%	95%
Fuel Handling Building	15.0%	95%

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 in accordance with ANSI N510-1980 at the system flowrate specified below ± 10% at least once per 24 months.

ESF Ventilation System	Delta P	Flowrate
Control Room	3.5 in. WG	2100 cfm
Auxiliary Building	3.7 in. WG	73,500 cfm
Fuel Handling Building	4.1 in. WG	35,750 cfm

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

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Waste Gas Holdup System, the quantity of radioactivity contained in gas storage tanks, and the quantity of radioactivity contained in temporary unprotected outdoor liquid storage tanks.

The gaseous radioactivity quantities shall be determined following the methodology in Regulatory Guide 1.24 "Assumptions Used For Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure." The liquid radwaste quantities shall be maintained such that 10 CFR Part 20 limits are met.

(continued)

5.5 Programs and Manuals (continued)

5.5.19 Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Ventilation System (CRVS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem ~~whole body~~ TEDE ~~or its equivalent to any part of the body~~ for the duration of the accident. The program shall include the following elements:

a. The definition of the CRE and the CRE boundary.

b. Requirements for maintaining the CRE boundary in its design condition, including configuration control and preventive maintenance.

c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CRVS, operating at the flow rate required by the VFTP, at a Frequency of 24 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 24 month assessment of the CRE boundary.

e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

f. The provisions of SR 3.0.2 are applicable to the Frequencies required by paragraphs c and d for determining CRE unfiltered inleakage and assessing CRE habitability, and measuring CRE pressure and assessing the CRE boundary.

ATTACHMENT 2

Proposed Technical Specification Changes
(RETYPED)

Remove Pages	Insert Pages
1.1-3	1.1-3
3.4-36	3.4-36
3.6-5	3.6-5
3.6-9	3.6-9
3.6-10	3.6-10
5.0-10	5.0-10
5.0-13	5.0-13
5.0-17a	5.0-17a

1.1 Definitions (continued)

DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) that alone would produce the same dose when inhaled as the combined activities of iodine isotopes I-131, I-132, I-133, I-134, and I-135 actually present. The determination of DOSE EQUIVALENT I-131 shall be performed using the committed thyroid dose conversion factors from Table 2.1 of EPA Federal Guidance Report No. 11, 1988, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."
DOSE EQUIVALENT XE-133	DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 (microcuries per gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-87, Kr-88, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT XE-133 shall be performed using effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil."

(continued)

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.16.1	-----NOTE----- Only required to be performed in MODE 1. ----- Verify reactor coolant DOSE EQUIVALENT XE-133 specific activity ≤ 270.0 μCi/gm.	In accordance with the Surveillance Frequency Control Program
SR 3.4.16.2	-----NOTE----- Only required to be performed in MODE 1. ----- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 μCi/gm.	In accordance with the Surveillance Frequency Control Program <u>AND</u> Between 2 and 6 hours after a THERMAL POWER change of ≥ 15% RTP within a 1 hour period.

3.6 CONTAINMENT SYSTEMS

3.6.3 Containment Isolation Valves

LCO 3.6.3 Each containment isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

---NOTES---

1. Penetration flow path(s) except for 48-inch purge valve 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 containment isolation valves.
4. Enter applicable Conditions and Required Actions of LCO 3.6.1, "Containment," when isolation valve leakage results in exceeding the overall containment leakage rate acceptance criteria.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. --------------NOTE------------- Only applicable to penetration flow paths with two containment isolation valves. ------------------------------------ One or more penetration flow paths with one containment isolation valve inoperable except for a containment purge supply and exhaust valve or pressure/vacuum relief valve leakage not within limit.	A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured. <u>AND</u>	4 hours (continued)

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.6.3.1	Verify each 48 inch purge valve is sealed closed, except for one purge valve in a penetration flow path while in Condition D of this LCO.	In accordance with the Surveillance Frequency Control Program
SR 3.6.3.2	Verify each 12 inch vacuum/pressure relief valve is closed, except when these valves are open for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open.	In accordance with the Surveillance Frequency Control Program
SR 3.6.3.3	-----------------------------NOTE-------------------------- Valves and blind flanges in high radiation areas may be verified by use of administrative controls. ---	
	Verify each containment isolation manual valve and blind flange that is located outside containment and not locked, sealed or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.	In accordance with the Surveillance Frequency Control Program
SR 3.6.3.4	-----------------------------NOTE-------------------------- Valves and blind flanges in high radiation areas may be verified by use of administrative means. ---	
	Verify each containment isolation manual valve and blind flange that is located inside containment and not locked, sealed or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.	Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days
SR 3.6.3.5	Verify the isolation time of each automatic power operated containment isolation valve is within limits.	In accordance with the Inservice Testing Program
SR 3.6.3.6	Not used	

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.3.7	----------------------------NOTE--------------------------- This surveillance is not required when the penetration flow path is isolated by a leak tested blank flange. --- Perform leakage rate testing for containment purge supply and exhaust and vacuum/pressure relief valves with resilient seals.	In accordance with the Surveillance Frequency Control Program
SR 3.6.3.8	Verify each automatic containment isolation valve that is not locked, sealed or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.6.3.9	Not used	
SR 3.6.3.10	Verify each 12 inch containment vacuum/pressure relief valve is blocked to restrict the valve from opening > 50°.	In accordance with the Surveillance Frequency Control Program
SR 3.6.3.11	Not used	

5.5 Programs and Manuals (continued)

5.5.9 Steam Generator (SG) Tube Inspection Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

a. Provisions for condition monitoring assessments.

Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity.

SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs. Except during a SG tube rupture, leakage is not to exceed 0.75 gallon per minute total for all four SGs.

(continued)

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal absorber, 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 and at the relative humidity specified below. Laboratory testing shall be completed at least once per 24 months and after every 720 hours of charcoal operation.

ESF Ventilation System	Penetration	RH
Control Room	2.5%	95%
Auxiliary Building	5.0%	95%
Fuel Handling Building	15.0%	95%

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 in accordance with ANSI N510-1980 at the system flowrate specified below ± 10% at least once per 24 months.

ESF Ventilation System	Delta P	Flowrate
Control Room	3.5 in. WG	2100 cfm
Auxiliary Building	3.7 in. WG	73,500 cfm
Fuel Handling Building	4.1 in. WG	35,750 cfm

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

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Waste Gas Holdup System, the quantity of radioactivity contained in gas storage tanks, and the quantity of radioactivity contained in temporary unprotected outdoor liquid storage tanks.

The gaseous radioactivity quantities shall be determined following the methodology in Regulatory Guide 1.24 "Assumptions Used For Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure." The liquid radwaste quantities shall be maintained such that 10 CFR Part 20 limits are met.

(continued)

5.5 Programs and Manuals (continued)

5.5.19 Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Ventilation System (CRVS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem TEDE for the duration of the accident. The program shall include the following elements:

a. The definition of the CRE and the CRE boundary.

b. Requirements for maintaining the CRE boundary in its design condition, including configuration control and preventive maintenance.

c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CRVS, operating at the flow rate required by the VFTP, at a Frequency of 24 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 24 month assessment of the CRE boundary.

e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

f. The provisions of SR 3.0.2 are applicable to the Frequencies required by paragraphs c and d for determining CRE unfiltered inleakage and assessing CRE habitability, and measuring CRE pressure and assessing the CRE boundary.

ATTACHMENT 3

Technical Specification Bases Markup
(For Information Only)

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on RCS pressure protects the integrity of the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. By establishing an upper limit on RCS pressure, the continued integrity of the RCS is ensured. According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the reactor pressure coolant boundary (RCPB) design conditions are not to be exceeded during normal operation and anticipated operational occurrences (AOOs). Also, in accordance with GDC 28, "Reactivity Limits" (Ref. 1), reactivity accidents, including rod ejection, do not result in damage to the RCPB greater than limited local yielding.

The design pressure of the RCS is 2500 psia. During normal operation and AOOs, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2). To ensure system integrity, all RCS components were hydrostatically tested at 125% of design pressure, according to the ASME Code requirements prior to initial operation when there is no fuel in the core. Following inception of unit operation, RCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3).

Overpressurization of the RCS could result in a breach of the RCPB. If such a breach occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere, raising concerns relative to limits on radioactive releases specified in 10 CFR ~~100~~50.67, "~~Reactor Site Criteria~~Accident Source Term" (Ref. 4).

(continued)

BASES (continued)

APPLICABILITY	SL 2.1.2 applies in MODES 1, 2, 3, 4, and 5 because this SL could be approached or exceeded in these MODES due to overpressurization events. The SL is not applicable in MODE 6 because the reactor vessel head closure bolts are not fully tightened, or the reactor vessel is sufficiently vented, making it unlikely that the RCS can be pressurized.
SAFETY LIMIT VIOLATIONS	If the RCS pressure SL is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance and be in MODE 3 within 1 hour. Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR ~~100~~50.67, "~~Reactor Site Criteria~~Accident Source Term" limits (Ref. 4). The allowable Completion Time of 1 hour recognizes the importance of reducing power level to a MODE of operation where the potential for challenges to safety systems is minimized. If the RCS pressure SL is exceeded in MODE 3, 4, or 5, RCS pressure must be restored to within the SL value within 5 minutes. Exceeding the RCS pressure SL in MODE 3, 4, or 5 is more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SL within 5 minutes. The action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.
REFERENCES	1. 10 CFR 50, Appendix A, GDC 14 (associated with 1967 GDC 9 per FSAR Appendix 3.1A), GDC 15 (no direct correlation to 1967 GDC; however, intent of 1971 GDC is per met per FSAR Appendix 3.1A), and GDC 28 (associated with 1967 GDC 30 per FSAR Appendix 3.1A). 2. ASME, Boiler and Pressure Vessel Code, Section III, Summer 1969. 3. ASME, Boiler and Pressure Vessel Code, Section XI. 4. 10 CFR ~~100~~50.67. 5. FSAR, Section 7.2. 6. DCM S-7, 3.4.1.

BASES

APPLICABLE SAFETY ANALYSIS (continued)

In the boron dilution analysis, the required SDM defines the reactivity difference between an initial subcritical boron concentration and the corresponding critical boron concentration. These values, in conjunction with the configuration of the RCS and the assumed dilution flow rate, directly affect the results of the analysis. This event is most limiting at the beginning of core life, when critical boron concentrations are highest.

Depending on the system initial conditions and reactivity insertion rate, the uncontrolled rod withdrawal transient is terminated by either a high power level trip or a high pressurizer pressure trip. In all cases, power level, RCS pressure, linear heat rate, and the DNBR do not exceed allowable limits.

The ejection of a control rod rapidly adds reactivity to the reactor core, causing both the core power level and heat flux to increase with corresponding increases in reactor coolant temperatures and pressure. The ejection of a rod also produces a time dependent redistribution of core power.

The startup of an inactive RCP in MODES 1 or 2 is precluded. In MODE 3, the startup of an inactive RCP cannot result in a "cold water" criticality, even if the maximum difference in temperature exists between the SG and the core. The maximum positive reactivity addition that can occur due to an inadvertent start is less than half the minimum required SDM. Startup of an idle RCP cannot, therefore, produce a return to power from the hot standby condition.

SDM satisfies Criterion 2 of 10CFR50.36(c)(2)(ii). Even though it is not directly observed from the control room, SDM is considered an initial condition process variable because it is periodically monitored to ensure that the unit is operating within the bounds of accident analysis assumptions.

LCO

SDM is a core design condition that can be ensured during operation through control rod positioning (control and shutdown banks) and through the soluble boron concentration.

The MSLB (Ref. 2) and the boron dilution (Ref. 3) accidents are the most limiting analyses that establish the SDM value of the LCO. For MSLB accidents, if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed 10 CFR ~~100~~50.67, "~~Reactor Site Criteria~~Accident Source Term," limits (Ref. 4). For the boron dilution accident, if the LCO is violated, the minimum required time assumed for operator action to terminate dilution may no longer be sufficient. The required SDM is specified in the COLR.

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.1.1.1 (continued) Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical, and the fuel temperature will be changing at the same rate as the RCS. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the *Surveillance Frequency Control Program*.
REFERENCES	1 10 CFR 50, Appendix A, GDC 26. 2 FSAR, Chapter 15, Section 15.4.2.1. 3 FSAR, Chapter 15, Section 15.2.4. 4 10 CFR ~~100~~50.67. 5 FSAR, Chapter 15, Section 15.4.6.1.6.

B 3.3 INSTRUMENTATION

B 3.3.1 Reactor Trip System (RTS) Instrumentation

BASES

BACKGROUND

The RTS initiates a unit shutdown, based on the values of selected unit parameters, to protect against violating the core fuel design limits and Reactor Coolant System (RCS) pressure boundary during anticipated operational occurrences (AOOs) and to assist the Engineered Safety Features (ESF) Systems in mitigating accidents.

The protection and monitoring systems have been designed to assure safe operation of the reactor. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the RTS, as well as specifying LCOs on other reactor system parameters and equipment performance.

The LSSS, as defined in 10 CFR 50.36, are defined in this specification as the Allowable Values, and in conjunction with the LCOs, establish the threshold for protective system action to prevent exceeding acceptable limits during Design Basis Accidents (DBAs).

During AOOs, which are those events expected to occur more than once during the unit life, the acceptable limits are:

1. The Departure from Nucleate Boiling Ratio (DNBR) shall be maintained above the Safety Limit (SL) value to prevent departure from nucleate boiling (DNB);
2. Fuel centerline melt shall not occur; and
3. The RCS pressure SL of 2735 psig shall not be exceeded.

Operation within the SLs of Specification 2.0, "Safety Limits (SLs)," also maintains the above values and assures that offsite dose will be within the 10 CFR 50 and 10 CFR ~~100~~50.67 criteria during AOOs.

Accidents are events that are analyzed even though they are not expected to occur during the unit life. The acceptable limit during accidents is that offsite dose shall be maintained within an acceptable fraction of 10 CFR ~~100~~50.67 limits. Different accident categories are allowed a different fraction of these limits, based on probability of occurrence. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event.

(continued)

BASES

BACKGROUND (continued)

During AOOs, which are those events expected to occur one or more times during the unit life, the acceptable limits are:

1. The Departure from Nucleate Boiling Ratio (DNBR) shall be maintained above the Safety Limit (SL) value to prevent departure from nucleate boiling (DNB).
2. Fuel centerline melt shall not occur, and
3. The RCS pressure SL of 2750 psia shall not be exceeded.

Operation within the SLs of Specification 2.0, "Safety Limits (SLs)," also maintains the above values and assures that offsite dose will be within the 10 CFR 50 and 10 CFR ~~100~~50.67 criteria during AOOs.

Accidents are events that are analyzed even though they are not expected to occur during the unit life. The acceptable limit during accidents is that offsite dose shall be maintained within an acceptable fraction of 10 CFR ~~100~~50.67 limits. Different accident categories are allowed a different fraction of these limits, based on probability of occurrence. Meeting the acceptable consequences for that event is considered having acceptable consequences for that event. However, these values and their associated NTSPs are not considered to be LSSS as defined in 10 CFR 50.36.

The ESFAS instrumentation is segmented into three distinct but interconnected modules as identified below:

- Field transmitters or process sensors and instrumentation: provide a measurable electronic signal based on the physical characteristics of the parameter being measured;
- Signal processing equipment including digital protection system, field contacts, and protection channel sets: provide signal conditioning, bistable setpoint comparison, process algorithm actuation, compatible electrical signal output to protection system devices, and control board/control room/miscellaneous indications; and
- Solid State Protection System (SSPS) including input, logic, and output bays: initiates the proper unit shutdown or engineered safety feature (ESF) actuation in accordance with the defined logic and based on the bistable outputs from the signal process control and protection system. The residual heat removal pump trip or refueling water storage tank level-low signal is not processed by the SSPS. The associated relays are located in the residual heat removal pumps control system.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

They are also the primary means for automatically isolating containment in the event of a fuel handling accident or any other source within containment during shutdown. Containment isolation in turn ensures meeting the containment leakage rate assumptions of the safety analyses, and ensures that the calculated accidental offsite radiological doses are below 10 CFR ~~100~~50.67 (Ref. 1) limits. Due to radioactive decay, containment is only required to isolate during fuel handling accidents involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous ~~100~~72 hours.)

The containment ventilation isolation instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requirements ensure that the instrumentation necessary to initiate Containment Ventilation Isolation, listed in Table 3.3.6-1, is OPERABLE.

1. Manual Initiation - Not used

2. Automatic Actuation Logic and Actuation Relays

The LCO requires two trains of Automatic Actuation Logic and Actuation Relays OPERABLE to ensure that no single random failure can prevent automatic actuation.

Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b, SI, and ESFAS Function 3.a, Containment Phase A Isolation. The applicable MODES and specified conditions for the Containment Ventilation Isolation portion of these Functions are different and less restrictive than those for their Phase A isolation and SI roles. If one or more of the SI or Phase A isolation Functions becomes inoperable in such a manner that only the Containment Ventilation Isolation Function is affected, the Conditions applicable to their SI and Phase A isolation Functions need not be entered. The less restrictive Actions specified for inoperability of the Containment Ventilation Isolation Functions specify sufficient compensatory measures for this case.

3. Containment Radiation

The LCO specifies two required channels of radiation monitors to ensure that the radiation monitoring instrumentation necessary to initiate Containment ventilation Isolation remains OPERABLE in MODES 1-4.

The LCO only requires one monitor to be OPERABLE during movement of recently irradiated fuel assemblies in containment. In order to provide the CVI function under these conditions without placing the entire SSPS in service, an alternate circuit is provided to power the output relays and provide logic actuation signals independent of the SSPS.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.6.7 (continued)

The test verifies that the channel responds to a measured parameter within the necessary range and accuracy. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.3.6.8

This SR assures that the individual channel RESPONSE TIMES for the CVI from Containment Purge Radiation Gaseous and Particulate function are less than or equal to the maximum values assumed in the accident analysis. Response Time testing acceptance criteria are included in ECG 38.2. Individual component response times are not modeled in the analyses. The analyses model the overall or elapsed time, from the point at which the parameter exceeds the Trip Setpoint value at the sensor, to the point at which the equipment in both trains reaches the required functional state (e.g., valves in full closed position). The response time may be measured by a series of overlapping tests such that the entire response time is measured.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. 10 CFR ~~100.11~~50.67.
2. NUREG-1366, December 1992.
3. DCM No. T-16, Containment Function.
4. WCAP-15376-P-A, Revision 1, "Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times," March 2003.
5. License Amendment 184/186, January 3, 2006.

BASES

BACKGROUND (continued)

The CRVS has two additional manually selected emergency operating modes; smoke removal and recirculation. Neither of these modes are required for the CRVS to be OPERABLE, but they are useful for certain non-DBA circumstances.

APPLICABLE SAFETY ANALYSES

The control room must be kept habitable for the operators stationed there during accident recovery and post accident operations.

The CRVS acts to terminate the supply of unfiltered outside air to the control room, initiate filtration, and pressurize the control room. These actions are necessary to ensure the control room is kept habitable for the operators stationed there during accident recovery and post accident operations by minimizing the radiation exposure of control room personnel.

In MODES 1, 2, 3, and 4, the radiation monitor (located at the control room intakes) actuation of the CRVS is a backup for the Phase A signal actuation. This ensures initiation of the CRVS during a loss of coolant accident, ~~or~~ steam generator tube rupture, control rod ejection accident and Main Steam Line Break ~~involving a release of radioactive materials~~.

The radiation monitor actuation of the CRVS in MODES 5 and 6, during movement of recently irradiated fuel assemblies (i.e., fuel that has occupied part of a critical reactor core within the previous ~~100~~72 hours), is the primary means to ensure control room habitability in the event of a fuel handling or waste gas decay tank rupture accident. This actuation is credited in the FHA. The CRVS pressurization system actuation instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

In MODES 1, 2, 3, 4, 5 and 6, credit is taken for the dual ventilation intake design of the CR pressurization air intakes. Based on the availability of redundant PG&E Design Class I radiation monitors at each pressurization intake location, the DCPP design has the capability of initial selection of the cleaner intake, but does not have the capability of automatic selection of the clean intake throughout the event. Based on the CRVS pressurization intake design, and the expectation that the operator will manually make the proper intake selection throughout the event, and per RG 1.194, June 2003, Regulatory Position C.3.3.2.3, when the CRVS is in Mode 4, the X/Q values for the more favorable CR intake is reduced by a factor of 4 and utilized to estimate the dose consequences.

LCO

The LCO requirements ensure that instrumentation necessary to initiate the CRVS pressurization system is OPERABLE.

1. Manual Initiation

   The LCO requires two trains OPERABLE. The operator can initiate the CRVS pressurization mode at any time by using either of two switches in the control room. This action will cause actuation of all components in the same manner as any of the

automatic actuation signals.

The LCO for Manual Initiation ensures the proper amount of redundancy is maintained in the manual actuation circuitry to ensure the operator has manual initiation capability.

2. Automatic Actuation Relays

The LCO requires two trains of Actuation Relays OPERABLE to ensure that no single random failure can prevent automatic actuation of the pressurization system. Since each unit has one train of Actuation Relays consisting of two sets of actuation logic, each unit must have at least one logic set for both trains to be considered OPERABLE.

(continued)

(Spillover from previous page.)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.7.5

SR 3.3.7.5 is the performance of a SLAVE RELAY TEST. This test energizes the Slave Relays and verifies actuation of the equipment to the pressurization mode. Although there are no "Slave Relays" as in the SSPS, this surveillance was maintained to preserve the format of the standard specification. The surveillance is intended to ensure that the actuation relays, downstream of the logic, function to actuate the pressurization mode equipment. Since the radiation monitors directly actuate the actuation relays, this test is performed as part of the performance of SR 3.3.7.2.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.3.7.6

SR 3.3.7.6 is the performance of a TADOT. This test is a check of the Manual Actuation Functions. Each Manual Actuation Function is tested up to, and including, the master relay coils. In some instances, the test includes actuation of the end device (i.e., pump starts, valve cycles, etc.).

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program. The SR is modified by a Note that excludes verification of setpoints during the TADOT. The Functions tested have no setpoints associated with them.

SR 3.3.7.7

CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. WCAP-13878, "Reliability of Potter & Brumfield MDR Relays", June 1994.
2. WCAP-13900, "Extension of Slave Relay Surveillance Test Intervals", April 1994.
3. License Amendment 184/186, January 3, 2006.
4. RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," June 2003.

B 3.3 INSTRUMENTATION

B 3.3.8 Fuel Building Ventilation System (FBVS) Actuation Instrumentation

BASES

BACKGROUND

The FBVS ensures that radioactive materials in the fuel building atmosphere following a fuel handling accident involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous ~~100~~72 hours) are filtered and adsorbed prior to exhausting to the environment. The system is described in the Bases for LCO 3.7.13, "Fuel Handling Building Ventilation System." The system initiates filtered ventilation of the fuel building automatically following receipt of a high radiation signal from the Spent Fuel Pool Monitor or from the New Fuel Storage Vault Monitor. Initiation may also be performed manually as needed from the main control room or fuel handling building.

High radiation, from either of the two monitors, provides FBVS initiation. These actions function to prevent exfiltration of contaminated air by initiating filtered ventilation, which imposes a negative pressure on the fuel building.

APPLICABLE SAFETY ANALYSES

The FBVS ensures that radioactive materials in the fuel building atmosphere following a fuel handling accident involving handling recently irradiated fuel are filtered and adsorbed prior to being exhausted to the environment. This action reduces the radioactive content in the fuel building exhaust following a fuel handling accident ~~so that offsite doses remain within the limits specified in 10 CFR 100 (Ref. 1)~~.

The FBVS actuation instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requirements ensure that instrumentation necessary to initiate the FBVS is OPERABLE.

1. Manual Initiation

   The LCO requires two channels OPERABLE. The operator can initiate the FBVS at any time by using either of two switches, one in the control room and another in the fuel handling building. This action will cause actuation of all components in the same manner as any of the automatic actuation signals.

   The LCO for Manual Initiation ensures the proper amount of redundancy is maintained in the manual actuation circuitry to ensure the operator has manual initiation capability.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.8.1 (continued)

The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.3.8.2

A CFT is performed on each required channel to ensure the entire channel will perform the intended function. This test verifies the capability of the instrumentation to provide the FBACS actuation. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.3.8.3 - Not used

SR 3.3.8.4

SR 3.3.8.4 is the performance of a TADOT. This test is a check of the manual actuation functions. Each manual actuation function is tested up to, and including, the master relay coils. In some instances, the test includes actuation of the end device (e.g., pump starts, valve cycles, etc.). The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program. The SR is modified by a Note that excludes verification of setpoints during the TADOT. The Functions tested have no setpoints associated with them.

SR 3.3.8.5

CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. ~~10 CFR 100.11~~Not used.
2. License Amendment 184/186, January 3, 2006.
3. PG&E Letter DCL-05-124

BASES (continued)

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. Safety analyses for design basis events that model primary to secondary LEAKAGE result in steam discharge to the atmosphere. ~~The safety analysis for the SLB event assumes that primary to secondary LEAKAGE is 10.5 gpm (room temperature conditions) from the faulted SG or increases to 10.5 gpm as a result of accident induced conditions, and 0.1 gpm (room temperature conditions) from each intact SG.~~ The safety analyses for events resulting in steam discharge to the atmosphere~~, other than SGTR and SLB,~~ assume that primary to secondary LEAKAGE from all SGs is 0.75 gpm (~~hot conditions~~Standard Temperature and Pressure) under accident conditions. For conservatism, the SLB assumes that the total 0.75 gpm tube leakage is assigned to the faulted steam generator and the SGTR assumes that the total 0.75 gpm tube leakage is assigned to the 3 intact steam generators. The LCO requirement to limit primary to secondary LEAKAGE through any one SG to less than or equal to 150 gallons per day is significantly less than the conditions assumed in the SLB safety analysis for the faulted SG.

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The SGTR (Ref. 3) is more limiting for radiological releases at the site boundary. The radiological dose analysis assumes loss of off-site power at the time of reactor trip with no subsequent condenser cooling available. The SGTR assumes that the total 0.75 gpm tube leakage is assigned to the 3 intact steam generators. The steam generator (SG) PORV for the SG that has sustained the tube rupture is assumed to fail open for 30 minutes, at which time the operator closes the block valve to the PORV. The dose consequences resulting from the SGTR accident are within the limits defined in 10 CFR ~~100~~50.67 (Ref. 6).

The SLB is more limiting for site radiation releases for events other than SGTR. ~~The safety analysis for the SLB accident assumes 10.5 gpm primary to secondary LEAKAGE is through the faulted SG.~~ The dose consequences resulting from the SLB accident are ~~well~~ within the limits defined in 10 CFR ~~100~~50.67~~ or the staff approved licensing basis (i.e., small fraction of these limits)~~.

The safety analysis for RCS main loop piping for GDC-4 (Ref. 1) assumes 1 gpm unidentified leakage and monitoring per RG 1.45 (Ref. 2) are maintained (Ref. 4 and 5).

The RCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50, Appendix A, GDC 4 and 30.
2. Regulatory Guide 1.45, May 1973.
3. FSAR, Section 15.
4. FSAR, Section 3.
5. NUREG-1061, Volume 3, November, 1984.
6. 10 CFR ~~100~~50.67.
7. NEI 97-06, "Steam Generator Program Guidelines."
8. EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 RCS Specific Activity

BASES

BACKGROUND	The maximum dose ~~to the whole body and the thyroid~~ that an individual at the exclusion area boundary can receive for 2 hours following an accident or at the low population zone outer boundary for the radiological release duration, is specified in 10 CFR ~~100.11~~50.67 (Ref. 1). Doses to the control room operators must be limited per GDC 19. The limits on specific activity ensure that the doses are appropriately limited during analyzed transients and accidents. The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the dose consequences in the event of a steam line break (SLB) or steam generator tube rupture (SGTR) accident. The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133. The allowable levels are intended to ensure that offsite and control room doses meet the appropriate acceptance criteria in ~~the Standard Review Plan~~ RG 1.183 (Ref. 2).
APPLICABLE SAFETY ANALYSES	The LCO limits on the specific activity of the reactor coolant ensures that the resulting offsite and control room doses meet the appropriate SRP acceptance criteria following a SLB or a SGTR accident. The safety analyses (Refs. 3 and 4) assume the specific activity of the reactor coolant is at ~~or more conservative than~~ the LCO limits, and an existing reactor coolant steam generator (SG) tube leakage rate of ~~1~~0.75 gpm ~~exists~~. The safety analyses assume the specific activity of the secondary coolant is at its limit of 0.1 μCi/gm DOSE EQUIVALENT I-131 from LCO 3.7.18, "Secondary Specific Activity." The analysis for the SLB and SGTR accidents establish the acceptance limits for RCS specific activity. Reference to these analyses is used to assess changes to the unit that could affect RCS specific activity, as they relate to the acceptance limits. The analyses consider two cases of reactor coolant specific activity. One case assumes specific activity at 1.0 μCi/gm DOSE EQUIVALENT I-131 with a concurrent large iodine spike that increases the rate of release of iodine from the fuel rods containing cladding defects to the primary coolant immediately after a SLB (by a factor of 500) or SGTR (by a factor of 335), respectively.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The second case assumes the initial reactor coolant iodine activity at 60.0 μCi/gm DOSE EQUIVALENT I-131 due to a pre-accident iodine spike caused by an RCS transient. In both cases, the noble gas specific activity is assumed to be ~~651~~270 μCi/gm DOSE EQUIVALENT XE-133.

The SGTR analysis also assumes a loss of offsite power at the same time as the reactor trip. The SGTR causes a reduction in reactor coolant inventory. The reduction initiates a reactor trip from a low pressurizer pressure signal or an RCS overtemperature ΔT signal.

The loss of offsite power causes the steam dump valves to close to protect the condenser. The rise in pressure in the ruptured SG discharges radioactively contaminated steam to the atmosphere through the SG power operated relief valves and the main steam safety valves. The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends and the RHR system is placed in service.

Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed 60.0 μCi/gm DOSE EQUIVALENT I-131, for more than 48 hours.

The limits on RCS specific activity are also used for establishing standardization in radiation shielding and plant personnel radiation protection practices.

RCS specific activity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The iodine specific activity in the reactor coolant is limited to 1.0 μCi/gm DOSE EQUIVALENT I-131, and the noble gas specific activity in the reactor coolant is limited to ~~600.0~~270.0 μCi/gm DOSE EQUIVALENT XE-133, as contained in SR 3.4.16.2 and SR 3.4.16.1 respectively. The limits on specific activity ensure that offsite and control room doses will meet the appropriate ~~SRP~~ acceptance criteria (Refs. 1 and 2).

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

The definition of DOSE EQUIVALENT XE-133 in Specification 1.1, "Definitions," requires that the determination of DOSE EQUIVALENT XE-133 shall be performed using the effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil." These dose conversion factors are consistent with the dose conversion factors used in the applicable dose consequence analyses.

The Note modifies this SR to allow entry into and operation in MODE 4, MODE 3, and MODE 2 prior to performing the SR. This allows the Surveillance to be performed in those MODES, prior to entering MODE 1.

SR 3.4.16.2

This Surveillance is performed to ensure iodine specific activity remains within the LCO limit during normal operation and following fast power changes when iodine spiking is more apt to occur. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program. The Frequency, between 2 and 6 hours after a power change ≥ 15% RTP within a 1 hour period, is established because the iodine levels peak during this time following iodine spike initiation; samples at other times would provide inaccurate results.

The definition of DOSE EQUIVALENT I-131 in Specification 1.1, "Definitions," specifies the ~~thyroid~~ dose conversion factors which may be used to determine DOSE EQUIVALENT I-131. The ~~thyroid~~ dose conversion factors used to determine DOSE EQUIVALENT I-131 are the committed thyroid dose conversion factors from Table 2.1 of EPA Federal Guidance Report No. 11, 1988, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion," and are~~to be~~ consistent with the dose conversion factors used in the applicable dose consequence analyses~~, or be conservative with respect to the dose conversion factors used in the applicable dose consequence analyses such that a higher DOSE EQUIVALENT I-131 is determined~~.

The Note modifies this SR to allow entry into and operation in MODE 4, MODE 3, and MODE 2 prior to performing the SR. This allows the Surveillance to be performed in those MODES, prior to entering MODE 1.

(continued)

BASES (continued)

REFERENCES

1. 10 CFR ~~100.11, 1973~~50.67.
2. ~~Standard Review Plan (SRP), Section 6.4 (SLB and SGTR control room dose limits), Section 15.1.5 Appendix A (SLB offsite dose limits) and Section 15.6.3 (SGTR offsite dose limits).~~Regulatory Guide 1.183, July 2000.
3. FSAR, Sections 15.4.3 and 15.5.20.
4. FSAR Section ~~15.1.5~~15.5.18.

BASES (continued)

APPLICABLE SAFETY ANALYSES

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this Specification. The analysis of a SGTR event assumes a ~~total~~ primary to secondary LEAKAGE rate of ~~1~~0.75 gpm from the intact SGs plus the leakage rate associated with a double-ended rupture of a single tube. The SGTR radiological dose analysis assumes loss of off-site power at the time of reactor trip with no subsequent condenser cooling available. The SG PORV for the SG that has sustained the tube rupture is assumed to fail open for 30 minutes, at which time the operator closes the block valve to the PORV. The SGTR radiological dose analysis assumes the contaminated secondary fluid is released briefly to the atmosphere from all the PORVs following reactor trip, is released from the ruptured SG PORV for 30 minutes, is released from the intact SG PORVs during the cooldown, and is released from all PORVs following cooldown until termination of the event.

The analysis for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture.) ~~For the SLB event, the primary to secondary LEAKAGE is 10.5 gpm from the faulted SG or is assumed to increase to 10.5 gpm as a result of accident induced conditions, and 0.1 gpm from each intact SG. For other events, the~~The steam discharge to the atmosphere is based on the total primary to secondary LEAKAGE from all SGs of 0.75 gpm under accident conditions. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the LCO 3.4.16, "RCS Specific Activity," limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2), 10 CFR ~~100~~50.67 (Ref. 3) ~~or the NRC approved licensing basis (e.g., a small fraction of these limits)~~ and RG 1.183 (Ref. 7).

Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the repair criteria be plugged in accordance with the Steam Generator Program.

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. If a tube was determined to satisfy the repair criteria but was not plugged, the tube may still have tube integrity.

(continued)

BASES

LCO (continued)

Structural integrity requires that the primary membrane stress intensity in a tube not exceed the yield strength for all ASME Code, Section III, Service Level A (normal operating conditions) and Service Level B (upset or abnormal conditions) transients included in the design specification. This includes safety factors and applicable design basis loads based on ASME Code, Section III, Subsection NB (Ref. 4) and Draft Regulatory Guide 1.121 (Ref. 5).

The accident induced leakage performance criterion ensures (a) that the primary to secondary LEAKAGE caused by a design basis accident, other than a SGTR, is within the accident analysis assumptions, and (b) that the primary to secondary LEAKAGE will not exceed ~~1~~0.75 gpm total for all four ~~per~~ SGs (except for specific types of degradation at specific locations where the NRC has approved greater accident induced leakage) to ensure that the potential for induced leakage during severe accidents will be maintained at a level that will not increase risk. The accident analysis for the SLB event, the SGTR event and other events resulting in steam release to the atmosphere assumes that accident induced leakage does not ~~exceed 10 gpm in the faulted SG and 0.1 gpm in each intact SG. For the faulted SG in the SLB event, 10.5 gpm is the accident induced leakage limit, of which no more than 1 gpm can come from sources not specifically exempted by the NRC from this 1 gpm limit. The accident analyses for events other than SGTR and SLB assume that leakage does not~~ exceed 0.75 gpm total under accident conditions. The accident induced leakage rate includes any primary to secondary LEAKAGE existing prior to the accident in addition to primary to secondary LEAKAGE induced during the accident.

The operational LEAKAGE performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational LEAKAGE is contained in LCO 3.4.13, "RCS Operational LEAKAGE," and limits primary to secondary LEAKAGE through any one SG to 150 gallons per day. This limit is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line break. If this amount of LEAKAGE is due to more than one crack, the cracks are very small, and the above assumption is conservative.

APPLICABILITY

Steam generator tube integrity is challenged when the pressure differential across the tubes is large. Large differential pressures across SG tubes can only be experienced in MODE 1, 2, 3, or 4.

RCS conditions are far less challenging in MODES 5 and 6 than during MODES 1, 2, 3, and 4. In MODES 5 and 6, primary to secondary differential pressure is low, resulting in lower stresses and reduced potential for LEAKAGE.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

In addition, Specification 5.5.9 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

SR 3.4.17.2

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. The tube repair criteria delineated in Specification 5.5.9 are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size measurement and for future flaw growth. In addition, the tube repair criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). Reference 1 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria.

The Frequency of prior to entering MODE 4 following a SG inspection ensures that the Surveillance has been completed and all tubes meeting the repair criteria are plugged prior to subjecting the SG tubes to significant primary to secondary pressure differential.

REFERENCES

1. NEI 97-06, "Steam Generator Program Guidelines."
2. 10 CFR 50 Appendix A, GDC 19, 1999.
3. 10 CFR ~~100~~50.67.
4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.
5. Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976.
6. EPRI, "Pressurized Water Reactor Steam Generator Examination Guidelines."
7. Regulatory Guide 1.183, July 2000.

BASES

BACKGROUND (continued)

Containment Purge System (48 inch purge valves)

The Containment Purge System operates to supply outside air into the containment for ventilation and cooling or heating needed for prolonged containment access following a shutdown and during refueling. ~~The system may also be used to reduce the concentration of noble gases within containment prior to and during personnel access.~~ The supply and exhaust lines each contain two isolation valves. ~~The 48 inch Containment Purge valves are qualified for automatic closure from their open position under DBA conditions.~~ The safety analyses assume that the 48-inch supply and exhaust line valves are closed at the start of the DBA. Therefore, the 48 inch Containment Purge supply and exhaust isolation valves are ~~normally maintained~~ sealed closed in MODES 1, 2, 3, and 4 to ensure the containment boundary is maintained. ~~The Containment Purge Supply and Exhaust Isolation valves are supplied with an internal block which prevents opening the valve beyond 80 degrees. This block was provided by the manufacture to allow limiting the valve's opening. Calculations performed during qualification to Branch Technical Position CSB 6-4 showed the block to be unnecessary to assure closure time within 2 seconds under DBA conditions (SSER 9, June 1980 and Calculation M-661). Adjustments of this block to values greater than or less than 80 degrees will not affect the valve's ability to close. This design assures that containment boundary is maintained. These valves may be opened as necessary to:~~

~~a. Reduce noble gases within containment prior to and during personnel access, and~~

~~b. Mitigate the effects of controller leakage and other sources which may effect the habitability of the containment for personnel entry.~~

~~Operation in Modes 1, 2, 3, or 4 with the 48-inch purge valves or the 12-inch vacuum/pressure relief valves open providing a flow path is limited to no more than 200 hours per calendar year.~~

Containment Pressure/Vacuum Relief (12 inch isolation valves)

The Containment Pressure/Vacuum Relief valves are operated as necessary to:

a. Reduce the concentration of noble gases within containment prior to and during personnel access, and

b. Equalize containment internal and external pressures.

Since the 12 inch Containment Pressure/Vacuum Relief valves are designed to meet the requirements for automatic containment isolation within 5 seconds if mechanical blocks are installed to prevent opening more then 50°, these valves may be opened as needed in MODES 1, 2, 3, and 4.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The containment isolation valve LCO was derived from the assumptions related to minimizing the loss of reactor coolant inventory and establishing the containment boundary during major accidents. As part of the containment boundary, containment isolation valve OPERABILITY supports leak tightness of the containment. Therefore, the safety analyses of any event requiring isolation of containment is applicable to this LCO.

The DBA that results in a release of radioactive material within containment in MODES 1, 2, 3, or 4 is a loss of coolant accident (LOCA) (Ref. 1). In the analyses for this accident, it is assumed that containment isolation valves are either closed or function to close within the required isolation time following event initiation. This ensures that potential paths to the environment through containment isolation valves (including the ~~Containment Purge, and~~ Containment Vacuum/Pressure Relief valves) are minimized. The safety analyses assume that the 48 inch purge valves are closed at event initiation. If the ~~48 inch Containment Purge supply and exhaust valves close within 2 seconds and the~~ 12 inch pressure/vacuum relief valves close within 5 seconds after the DBA initiation, the safety analysis shows that offsite dose release will be less than 10 CFR~~100~~ 50.67 guidelines.

The DBA analysis assumes that containment isolation occurs and leakage is prevented except for the design leakage rate, L_a.

The LOCA offsite dose analysis assumes leakage from the containment at a maximum leak rate of 0.10 percent of the containment volume per day for the first 24 hours, and at 0.05 percent of the containment volume per day for the duration of the accident.

The single failure criterion required to be imposed in the conduct of plant safety analyses was considered in the original design of ~~the 48 inch Containment Purge supply and exhaust and~~ the 12 inch Containment Pressure/Vacuum Relief valves. Two valves in series provide assurance that the flow paths can be isolated even if a single failure occurred. The inboard and outboard isolation valves are provided with diverse power sources and are pneumatically operated spring closed valves that will fail closed on the loss of power or air.

The ~~48 inch Containment Purge supply and exhaust and~~ 12 inch Containment Pressure/Vacuum Relief valves are able to close in the environment following a LOCA. Therefore, each of the ~~Containment Purge supply and exhaust and~~ Containment Vacuum/pressure Relief valves may be opened to provide a flow path. The ~~48 inch Containment Purge supply and exhaust valves and/or~~ 12-inch vacuum/pressure relief valves may be open no more then 200 hours per calendar year while in MODES 1, 2, 3, and 4. ~~Additionally, only two of the three flow paths (containment purge supply and exhaust, and containment vacuum/pressure relief) may be open at one time.~~

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The system is designed to preclude a single failure from compromising the containment boundary as long as the system is operated in accordance with the subject LCO.

The 48 inch Containment Purge supply and exhaust valves may be unable to close in the environment following a LOCA in sufficient time to support DBA acceptance criteria. Therefore, each of the purge valves is required to remain sealed closed during MODES 1, 2, 3, and 4. In this case, the single failure criterion remains applicable to the containment purge valves due to failure in the control circuit associated with each valve. Again, the purge system valve design precludes a single failure from compromising the containment boundary as long as the system is operated in accordance with the subject LCO.

The containment isolation valves satisfy Criterion 3 of 10CFR50.36(c)(2)(ii).

LCO

Containment isolation valves form a part of the containment boundary. The containment isolation valves' safety function is related to minimizing the loss of reactor coolant inventory and establishing the containment boundary during a DBA. The automatic power operated isolation valves are required to have isolation times within limits and to actuate on an automatic isolation signal. The 48 inch Containment Purge supply and exhaust valves ~~and the~~ must be sealed closed during MODES 1, 2, 3, and 4. The Pressure/Vacuum Relief valves must have blocks installed to prevent full opening. These blocked valves also actuate on an automatic isolation signal. The valves covered by this LCO are listed along with their associated stroke times in Plant Procedure AD13.DC1 (Ref. 5).

Normally closed passive containment isolation valves/devices are considered OPERABLE when manual valves are closed, automatic valves are de-activated and secured in their closed position, blind flanges are in place, and closed systems are intact. These passive isolation valves/devices are those listed in Reference 5.

Containment Purge supply and exhaust valves, and Containment Pressure/Vacuum Relief valves with resilient seals must meet additional leakage rate surveillance frequency requirements. The other containment isolation valve leakage rates are addressed by LCO 3.6.1, "Containment."

This LCO provides assurance that the containment isolation valves and the Containment Purge supply and exhaust, and Containment Pressure/Vacuum Relief valves will perform their designed safety function to minimize the loss of reactor coolant inventory and establish the containment boundary during accidents.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment

isolation valves are not required to be OPERABLE in MODE 5. The requirements for containment isolation valves during MODE 6 are addressed in LCO 3.9.4, "Containment Penetrations."

(continued)

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BASES (*continued*)

ACTIONS

The ACTIONS are modified by a Note allowing penetration flow paths, except 48-inch purge valve flow paths, to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a person at the valve controls, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for containment isolation is indicated. Due to the size of the containment purge line penetration and the fact that those penetrations exhaust directly from the containment atmosphere to the environment, the penetration flow path containing these valves may not be opened under administrative controls. A single purge valve in a penetration flow path may be opened to effect repairs to an inoperable valve, as allowed by SR 3.6.3.1.~~This Note also limits operation of the normally isolated Containment Supply and Exhaust valves (2 penetration flow paths) and the Vacuum/Pressure Relief valves (1 penetration flow path) to no more than 2 of 3 penetration flow paths open at one time.~~

A second Note has been added to provide clarification that, for 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 containment isolation valve. Complying with the Required Actions may allow for continued operation, and subsequent inoperable containment isolation valves are governed by subsequent Condition entry and application of associated Required Actions.

The ACTIONS are further modified by a third Note, which ensures appropriate remedial actions are taken, if necessary, if the affected systems are rendered inoperable by an inoperable containment isolation valve.

In the event the containment isolation valve leakage results in exceeding the overall containment leakage rate acceptance criteria, Note 4 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1.

Plant Procedure AD13.DC1 Attachment 7.7 (Ref. 5) provides the applicable CONDITION to enter for each containment isolation valve if the valve is inoperable.

A.1 and A.2

In the event one containment isolation valve in one or more penetration flow paths requiring isolation following a DBA is inoperable except for Containment Purge supply and exhaust, and Containment Pressure/Vacuum Relief isolation valve leakage not within limit, the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that

(continued)

BASES

ACTIONS

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

condition, it is prudent to perform the SR more often. Therefore, a Frequency of once per 92 days was chosen and has been shown to be acceptable based on operating experience.

Required Action D.2 is modified by two Notes. Note 1 applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned. Therefore, the probability of misalignment of these valves, once they have been verified to be in the proper position, is small.

E.1 and E.2

If the Required Actions and associated Completion Times are not 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 6 hours and to MODE 5 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 3.6.3.1

~~Not Used~~ Each 48 inch Containment Purge supply and exhaust valve is required to be verified sealed closed. This Surveillance is designed to ensure that a gross breach of containment is not caused by an inadvertent or spurious opening of a Containment Purge valve. These valves are assumed to be closed at the start of a DBA. Therefore, these valves are required to be in the sealed closed position during MODES 1, 2, 3, and 4. A Containment Purge valve that is sealed closed must have motive power to the valve operator removed. This can be accomplished by de-energizing the source of electric power or by removing the air supply to the valve operator. In the event the purge valve leakage requires entry into Condition D, the surveillance permits opening one purge valve in a penetration flow path to perform repairs. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.3.2

This SR ensures that ~~the 48 inch Containment Purge supply and exhaust and~~ the 12 inch Containment Pressure/Vacuum Relief valves are closed as required or, if open, open for an allowable reason. If a

purge or pressure relief valve is open in violation of this SR, the valve is considered inoperable. If the inoperable valve is not otherwise known to have excessive leakage when closed, it is not considered to have leakage outside of limits. The SR is not required to be met when the ~~Containment Purge supply and exhaust or~~ Containment Pressure Relief valves are open for the reasons stated. The valves may be opened for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open. The ~~Containment Purge supply and exhaust or~~ Containment Pressure/Vacuum Relief valves are capable of closing in the

(continued)

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BASES

BACKGROUND

Containment Spray System (continued)

In the recirculation mode of operation, containment spray is supplied by manual realignment of the residual heat removal (RHR) pumps after the RWST is empty.

The Containment Spray System provides a spray of cold borated water mixed with sodium hydroxide (NaOH) from the spray additive tank into the upper regions of containment to reduce the containment pressure and temperature, and to reduce fission products from the containment atmosphere during a DBA. The RWST solution temperature is an important factor in determining the heat removal capability of the Containment Spray System during the injection phase. In the recirculation mode of operation, heat is removed from the containment sump water by the RHR heat exchangers. Each train of the Containment Spray System provides adequate spray coverage to meet the system design requirements for containment atmospheric heat removal.

The Spray Additive System injects an NaOH solution into the spray. The resulting alkaline pH of the spray enhances the ability of the spray to scavenge fission products from the containment atmosphere. The NaOH added in the spray also ensures an alkaline pH for the solution recirculated in the containment sump. The alkaline pH of the containment sump water maximizes the retention of iodine and minimizes the occurrence of chloride and caustic stress corrosion on mechanical systems and components exposed to the fluid.

The Containment Spray System is actuated either automatically by a containment High-High pressure signal or manually. If an "S" signal is present, the High-High pressure signal automatically starts the two containment spray pumps, opens the containment spray pump discharge valves, opens the spray additive tank outlet valves, initiates a phase "B" isolation signal, and begins the injection phase. A manual actuation of the Containment Spray System will begin the same sequence and can be initiated by operator action from the main control board. The injection phase of containment spray continues until an RWST Low-Low level alarm is received. The Low-Low level alarm for the RWST signals the operator to manually secure the system. After re-alignment of the RHR system to the containment recirculation sump, the associated RHR spray header isolation valve ~~may be~~is opened to allow continued spray operation of one train of spray utilizing the RHR pump to supply flow. The LOCA dose analysis takes credit for this manual initiation of Containment Spray during recirculation to take place within 12 minutes following the termination of Containment Spray during the injection phase.

Containment Spray is ~~not~~ required to be actuated during the recirculation phase of a LOCA~~, but may be actuated at the discretion of the Technical Support Center~~. Containment Spray operation (injection plus recirculation) is credited until 6.25 hours following initiation of a

LOCA. During the recirculation phase of a LOCA, the Containment Spray System must be capable of

(continued)

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BASES

APPLICABLE SAFETY ANALYSES (continued)

Analyses and evaluation show that containment spray is not required during the recirculation phase of a LOCA for containment pressure and temperature control (Ref. 7). However, for dose consequences, containment spray is required during the recirculation phase of a LOCA for removing radioactive iodine and particulates from the containment atmosphere.

If only one RHR pump is available during the recirculation phase of a LOCA, it may not be possible to obtain significant containment spray without closing valves 8809A or B. If recirculation spray is used with only one train of RHR in operation, ECCS flow to the reactor will be reduced, but analysis has shown that the flow to the reactor in this situation is still in excess of that needed to supply the required core cooling.

The effect of an inadvertent containment spray actuation has been analyzed. An inadvertent spray actuation results in a -1.80 psid containment pressure decrease and is based on a sudden cooling effect of 70°F in the interior of the leak tight containment. Additional discussion is provided in the Bases for LCO 3.6.4.

The modeled Containment Spray System actuation from the containment analysis is based on a response time associated with exceeding the containment High-High pressure setpoint to achieving full flow through the containment spray nozzles. The Containment Spray System total response time includes diesel generator (DG) startup (for loss of offsite power),

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

sequenced loading of equipment, containment spray pump startup, and spray line filling (Ref. 4). The CFCUs performance for post accident conditions is given in Reference 4. The result of the analysis is that each train (two CFCUs) combined with one train of containment spray can provide 100% of the required peak cooling capacity during the post accident condition.

The modeled Containment Cooling System actuation from the containment analysis is based upon a response time associated with exceeding the containment High-High pressure setpoint to achieving full Containment Cooling System air and safety grade cooling water flow. The Containment Cooling System total response time includes signal delay, DG startup (for loss of offsite power), and component cooling water pump startup times.

The Containment Spray System and the Containment Cooling System satisfies Criterion 3 of 10CFR50.36(c)(2)(ii).

LCO

During a DBA LOCA, a minimum of two CFCUs and one containment spray train are required to maintain the containment peak pressure and temperature below the design limits (Refs. 4). Additionally, one containment spray train is also required to remove radioactive iodine and particulates from the containment atmosphere and maintain concentrations below those assumed in the safety analysis. To ensure that these requirements are met, two containment spray trains and the CFCU system consisting of four CFCUs or three CFCUs each supplied by a different vital bus must be OPERABLE. Therefore, in the event of an accident, at least one train of containment spray and two CFCUs operate, assuming the worst case single active failure occurs. Each Containment Spray train typically includes a spray pump, spray headers, nozzles, valves, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST upon an ESF actuation signal. Upon actuation of the RWST Low-Low alarm, the containment spray pumps are secured. Containment spray ~~could~~is then ~~be~~ supplied ~~as required~~ by an RHR pump taking suction from the containment sump for a total spray operation (injection and recirculation) of 6.25 hours.

Each CFCU includes cooling coils, dampers, fans, instruments, and controls to ensure an OPERABLE flow path.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment and an increase in containment pressure and temperature requiring the operation of the containment spray trains and CFCUs.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Thus, the Containment Spray System and the Containment Cooling System are not required to be OPERABLE in MODES 5 and 6.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

c. A break downstream of the MSIVs will be isolated by the closure of the MSIVs.

d. Following a steam generator tube rupture, closure of the MSIVs isolates the ruptured steam generator from the intact steam generators to minimize radiological releases.

e. The MSIVs are also utilized during other events such as a feedwater line break. This event is less limiting so far as MSIV OPERABILITY is concerned.

The MSIVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO requires that four MSIVs in the steam lines be OPERABLE. The MSIVs are considered OPERABLE when the isolation times are within limits, and they close on an isolation actuation signal.

This LCO provides assurance that the MSIVs will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR ~~100~~50.67 (Ref. 4) limits or the NRC staff approved licensing basis.

APPLICABILITY

The MSIVs must be OPERABLE in MODE 1, and in MODES 2 and 3 except when closed and de-activated (vented or prevented from opening), when there is significant mass and energy in the RCS and steam generators. When the MSIVs are closed, they are already performing the safety function.

In MODE 4, the steam generator energy is low, thus OPERABILITY in MODE 4 is not required.

In MODE 5 or 6, the steam generators do not contain much energy because their temperature is below the boiling point of water; therefore, the MSIVs are not required for isolation of potential high energy secondary system pipe breaks in these MODES.

ACTIONS

A.1

With one MSIV inoperable in MODE 1, action must be taken to restore OPERABLE status within 8 hours. Some repairs to the MSIV can be made with the unit hot. The 8 hour Completion Time is reasonable, considering the low probability of an accident occurring during this time period that would require a closure of the MSIVs.

The 8 hour Completion Time is greater than that normally allowed for containment isolation valves because the MSIVs are valves that isolate a closed system penetrating containment. These valves differ from other containment isolation valves in that the closed system provides an additional means for containment isolation.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1 (continued)

analyses. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. The MSIVs should not be tested at power, since even a part stroke exercise increases the risk of a valve closure when the unit is generating power.

As the MSIVs are not tested at power, they are exempt from the ASME Code, Section XI (Ref. 5), requirements during operation in MODE 1 or 2.

The Frequency is in accordance with the Inservice Testing Program.

This test may be conducted in MODE 3 with the unit at operating temperature and pressure. This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. However, the test is normally conducted in MODE 5 as permitted by the cold shutdown frequency justification provided in the Inservice Testing Program (IST) and as permitted by Reference 6, Subsection ISTC-3521(c).

SR 3.7.2.2

This SR verifies that each MSIV can close on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the plant to operation following a refueling outage. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Section 10.3.
2. FSAR, Section 6, Appendix 6.2 D.
3. FSAR, Section 15.4.2.
4. 10 CFR ~~100.11~~50.67.
5. ASME, Boiler and Pressure Vessel Code, Section XI.
6. ASME Code for Operation and Maintenance of Nuclear Power Plants, 2001 Edition including 2002 and 2003 Addenda.

BASES

BACKGROUND (continued)	Redundant supply and recirculation trains provide the required filtration should an excessive pressure drop develop across the other filter train. Normally open isolation dampers are arranged in series pairs so that the failure of one damper to shut will not result in a breach of isolation. The CRVS is designed in accordance with Seismic Category I requirements. The CRVS is designed to maintain a habitable environment in the CRE for the duration of the most severe Design Basis Accident (DBA) without exceeding a 5 rem ~~whole body~~TEDE dose ~~or its equivalent to any part of the body~~.
APPLICABLE SAFETY ANALYSES	The CRVS components are arranged in redundant, safety related ventilation trains. The location of components and ducting within the CRE ensures an adequate supply of filtered air to all areas requiring access. The CRVS provides airborne radiological protection for the CRE occupants, as demonstrated by the CRE occupant dose analyses for the most limiting design basis accident, fission product release presented in the FSAR, Chapter 15 (Ref. 2). There are no offsite or onsite hazardous chemicals that would pose a credible threat to control room habitability. Consequently, engineered controls for the control room are not required to ensure habitability against a hazardous chemical threat. The amount of CRE unfiltered inleakage is not incorporated into PG&E's hazardous chemical assessment. The evaluation of a smoke challenge demonstrated that smoke will not result in the inability of the CRE occupants to control the reactor either from the control room or from the remote shutdown panels (Ref. 1). The assessment verified that a fire or smoke event anywhere within the plant would not simultaneously render the Hot Shutdown Panel (HSP) and the CRE uninhabitable, nor would it prevent access from the CRE to the HSP in the event remote shutdown is required. No CRVS automatic actuation is required for hazardous chemical releases or smoke and no Surveillance Requirements are required to verify operability in cases of hazardous chemicals or smoke.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	The worst case single active failure of a component of the CRVS, assuming a loss of offsite power, does not impair the ability of the system to perform its design function. The CRVS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	Two independent and redundant CRVS trains are required to be OPERABLE to ensure that at least one is available if a single active failure disables the other train. The redundant train means a second train from the other unit (Ref. 5). Total system failure, such as from a loss of both ventilation trains or from an inoperable CRE boundary, could result in exceeding a dose of 5 rem ~~whole body~~TEDE ~~or its equivalent to any part of the body~~ to the CRE occupants in the event of a large radioactive release. Each CRVS train is considered OPERABLE when the individual components necessary to limit CRE occupant exposure are OPERABLE. A CRVS train is OPERABLE when the associated: a. main supply fan (one), filter booster fan (one) and pressurization fan (one) are OPERABLE; b. HEPA filters and charcoal adsorbers are not excessively restricting flow, and are capable of performing their filtration functions; and c. Ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained. In order for the CRVS trains to be considered OPERABLE, the CRE boundary must be maintained such that the CRE occupant dose from a large radioactive release does not exceed the calculated dose in the licensing basis consequence analyses for DBAs. In the event of an inoperable CRE boundary in MODES 1, 2, 3, or 4, mitigating actions are required to ensure CRE occupants are protected from hazardous chemicals and smoke. DCPP does not have CRVS automatic actuation for hazardous chemicals or smoke. Current practices at DCPP do not utilize chemicals in sufficient quantity to present a chemical hazard to the control room. Smoke is not considered in the DCPP safety analyses. Therefore, there are no specific limits at DCPP for hazardous chemicals or smoke.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, 3, 4, 5, and 6, and during movement of recently irradiated fuel assemblies (i.e., fuel that has occupied part of a critical reactor core within the previous ~~100~~72 hours) the CRVS must be OPERABLE to ensure that the CRE will remain habitable during and following a DBA or the release from the rupture of an outside waste gas tank.

During movement of recently irradiated fuel assemblies, the CRVS must be OPERABLE to cope with the release from a fuel handling accident involving handling recently irradiated fuel.

CRVS OPERABILITY requires that for MODE 5 and 6 and during movement of recently irradiated fuel assemblies in either unit, when there is only one OPERABLE train of CRVS, the OPERABLE CRVS train must be capable of being powered from an OPERABLE diesel generator that is directly associated with the bus which is energizing the OPERABLE CRVS train. This is an exception to LCO 3.0.6.

ACTIONS

The ACTIONS are modified by a NOTE that states that ACTIONS apply simultaneously to both units. The CRVS is common to both units.

A.1

When one CRVS train is inoperable for reasons other than an inoperable CRE boundary, action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining OPERABLE CRVS train is adequate to perform the CRE occupant protection function. However, the overall reliability is reduced because a single failure in the OPERABLE CRVS train could result in loss of CRVS function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and ability of the remaining train to provide the required capability.

B.1, B.2, and B.3

The CRE boundary is inoperable if unfiltered inleakage past the CRE boundary can result in CRE occupant radiological dose greater than the calculated dose of the licensing basis analyses of DBA consequences (allowed to be up to 5 rem ~~whole body or its equivalent to any part of the body~~TEDE).

In the event of an inoperable CRE boundary in MODES 1, 2, 3, or 4, action must be initiated to implement mitigating actions to lessen the effect on CRE occupants from the potential hazards of a radiological or chemical event or a challenge from smoke. Actions must be taken within 24 hours to verify that in the event of a DBA, the mitigating actions will ensure that CRE occupant radiological exposures will not exceed the calculated dose of the licensing basis analyses of DBA consequences, and that CRE occupants are protected from potential smoke and chemical hazards.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.10.3

This SR verifies that the required CRVS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The CRVS filter tests are in accordance with ANSI N510-1980 (Ref. 3). The VFTP includes testing the performance of the HEPA filter, charcoal adsorber efficiency, minimum flow rate, and the physical properties of the activated charcoal. Specific test Frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.10.4

This SR verifies that each CRVS train automatically starts and operates in the pressurization mode on an actual or simulated actuation signal generated from a Phase "A" Isolation. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.7.10.5

This SR verifies the OPERABILITY of the CRE boundary by testing for unfiltered air inleakage past the CRE boundary and into the CRE. The details of the testing are specified in the Control Room Envelope Habitability Program. Any changes to the most limiting configuration of the CRVS testing alignment for determining unfiltered air inleakage past the CRE boundary into the CRE must be made using a conservative decision making process (References 11-13).

The CRE is considered habitable when the radiological dose to CRE occupants calculated in the licensing basis analyses of DBA consequences is no more than 5 rem ~~whole body or its equivalent to any part of the body~~TEDE and the CRE occupants are protected from hazardous chemicals and smoke. For DCPP, there is no CRVS automatic actuation for hazardous chemical releases or smoke and there are no CRVS Surveillance Requirements that verify operability in cases of hazardous chemicals or smoke. This SR verifies that the unfiltered air inleakage into the CRE is no greater than the flow rate assumed in the licensing basis analyses of DBA consequences. When unfiltered air inleakage is greater than the assumed flow rate, Condition B must be entered. Required Action B.3 allows time to restore the CRE boundary to OPERABLE status provided mitigating actions can ensure that the CRE remains within the licensing basis habitability limits for the occupants following an accident.

(continued)

BASES

BACKGROUND (continued)

The ABVS is discussed in the FSAR, Sections 9.4 2, and 15.5 (Refs. 1, and 2, respectively) since it may be used for normal, as well as post accident, ventilation and atmospheric cleanup functions. The primary purpose of the single manually initiated heater is to maintain the relative humidity at an acceptable level, consistent with iodine removal efficiencies per ASTM D 3803-1989 (Ref. 3). There is no redundant heater since the failure of the charcoal adsorber and heater train would constitute a second failure in addition to the RHR pump seal failure assumed in conjunction with a LOCA (Ref.7). The heaters are not required for ABVS operability.

APPLICABLE SAFETY ANALYSES

The design basis of the ABVS is established by the large break LOCA. The system evaluation assumes a passive failure of the ECCS outside containment, such as an RHR pump seal failure, during the recirculation mode. In such a case, the system limits radioactive release to within the 10 CFR ~~100~~50.67 (Ref. 5) limits. The analysis of the effects and consequences of a large break LOCA is presented in Reference 2. The ABVS also functions, following a LOCA, in those cases where the ECCS goes into the recirculation mode of long term cooling, to clean up releases of smaller leaks, such as from valve stem packing.

The ventilation flow is also required to maintain the temperatures of the operating ECCS motors within allowable limits. The ventilation function has been designed for single failure and the system will continue to function to provide its ECCS motor cooling function.

Two types of system failures are considered in the accident analysis for radiation release: complete loss of function of one train, and excessive RHR pump seal LEAKAGE. Either type of failure may result in a lower efficiency of removal for any gaseous and particulate activity released to the ECCS pump rooms following a LOCA.

The ABVS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Two trains of the ABVS are required to be OPERABLE to ensure that at least one is available, assuming that a single failure disables the other train coincident with loss of offsite power. Total system failure could result in the atmospheric release from the ECCS pump room exceeding 10 CFR ~~100~~50.67 limits in the event of a Design Basis Accident (DBA).

ABVS is considered OPERABLE when the individual components necessary to maintain the ECCS pump room filtration and temperature are OPERABLE in both trains.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.12.6

This SR verifies the leak tightness of dampers that isolate flow to the normally operating filter train. This SR assures that the flow from the auxiliary building passes through the HEPA filter and charcoal adsorber unit when the ABVS Buildings and Safeguards or Safeguards Only modes have been actuated coincident with an SI. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Section 9.4.2.
2. FSAR, Section 15.5.
3. ASTM D 3803-1989
4. ANSI N510-1980
5. 10 CFR ~~100.11~~50.67.
6. NUREG-0800, Section 6.5.1, Rev. 2, July 1981.
7. DCM S-23B, "Main Auxiliary Building Heating and Ventilation System".

BASES

APPLICABLE SAFETY ANALYSES (continued)

FHBVS is only required to isolate during fuel handling accidents involving the handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous ~~100~~72 hours). In accordance with assumptions made in the fuel handling accident analysis, loss of offsite power is not considered concurrent with a fuel handling accident. ~~However, loss of power is enveloped by the fuel handling accident analysis.~~ To maximize FHBVS capability to mitigate the consequences of a fuel handling accident, at least one of the FHBVS trains must be capable of being supplied from an operable emergency diesel generator at all times whenever movement of recently irradiated fuel is taking place in the spent fuel pool. These assumptions and the analysis follow the guidance provided in Regulatory Guide ~~1.25~~1.183 (Ref. 3).

The FHBVS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Two independent and redundant trains of the FHBVS are required to be OPERABLE to ensure that at least one train is available, assuming a single failure that disables the other train. In accordance with assumptions made in the fuel handling accident analysis, loss of offsite power is not considered concurrent with a fuel handling accident. ~~However, loss of power is enveloped by the fuel handling accident analysis.~~ This requires that when two trains of the FHBVS are OPERABLE, at least one train of the FHBVS must be capable of being powered from an OPERABLE diesel generator that is directly associated with the bus which energizes the FHBVS train. When only one train is OPERABLE, an OPERABLE diesel generator must be directly associated with the bus which energizes that one OPERABLE FHBVS train. Total system failure could result in the atmospheric release from the fuel handling building exceeding the 10 CFR ~~100~~50.67 (Ref. 4) limits in the event of a fuel handling accident.

The FHBVS is considered OPERABLE when the individual components necessary to control releases from fuel handling building are OPERABLE in both trains. An FHBVS train is considered OPERABLE when its associated:

a. Exhaust fan is OPERABLE;

b. HEPA filter and charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration function; and

c. Ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.13.4

This SR verifies the integrity of the fuel handling building enclosure. The ability of the fuel handling building to maintain negative pressure with respect to potentially uncontaminated adjacent areas is periodically tested to verify proper function of the FHBVS. During the post accident mode of operation, the FHBVS is designed to maintain a slight negative pressure in the fuel handling building, to prevent unfiltered LEAKAGE. The FHBVS is designed to maintain the building pressure ≤ -0.125 inches water gauge with respect to atmospheric pressure. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.7.13.5

Operation of damper M-29 is necessary to ensure that the system functions properly. The operability of damper M-29 is verified if it can be closed. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Section 9.4.4.
2. FSAR, Section 15.5.
3. Regulatory Guide ~~1.25~~1.183, July 2000.
4. 10 CFR ~~100~~50.67.
5. ASTM D 3802-1989
6. ANSI N510-1980.
7. NUREG-0800, Section 6.5.1, Rev. 2, July 1981.
8. DCM S-23D, "Fuel handling Building HVAC System."
9. Not used
10. License Amendment 184/186, January 3, 2006.
11. PG&E Letter DCL-05-124

B 3.7 PLANT SYSTEMS

B 3.7.15 Spent Fuel Storage Pool Water Level

BASES

BACKGROUND

The minimum water level in the spent fuel pool meets the assumptions of iodine decontamination factors following a fuel handling accident. The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water also provides shielding during the movement of spent fuel.

A general description of the spent fuel pool design is given in the FSAR, Section 9.1.2 (Ref. 1). A description of the Spent Fuel Pool Cooling and Cleanup System is given in the FSAR, Section 9.1.3 (Ref. 2). The assumptions of the fuel handling accident are given in the FSAR, Section 9.1.4.3.4, 15.4.5 and 15.5.22 (Ref. 3).

APPLICABLE SAFETY ANALYSES

The minimum water level in the spent fuel pool meets the assumptions of the fuel handling accident described in Regulatory Guide ~~1.25~~1.183 (Ref. 4). The resultant 2 hour ~~thyroid~~ dose per person at the exclusion area boundary is a small fraction of the 10 CFR ~~100~~50.67 (Ref. 5) limits.

According to Reference 4, there is 23 ft of water between the top of the damaged fuel rods and the fuel pool surface during a fuel handling accident. With 23 ft of water, the assumptions of Reference 4 can be used directly. Although there are other spent fuel pool elevations where fuel handling accidents can occur, the design basis fuel handling accident, which uses the conservative assumptions of RG ~~1.25~~1.183, is expected to be bounding. To add conservatism, the analysis assumes that all fuel rods of the damaged fuel assembly fail.

In practice, the water level maintained for fuel handling provides more than 23 feet of water over the top of irradiated fuel assemblies seated in the storage racks. FSAR Section 9.1.4.3.4 requires the water level provide a minimum of 8 feet of water shielding during fuel handling. This assures more than 24 feet 6 inches of water shielding over the top of the fuel assemblies in the racks and more than 23 feet of water shielding over a fuel assembly lying horizontally on top of the racks.

The spent fuel pool water level satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The spent fuel pool water level is required to be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks. The specified water level preserves the assumptions of the fuel handling accident analysis (Ref. 3). As such, it is the minimum required for fuel storage and movement within the fuel storage pool.

(continued)

BASES (continued)

APPLICABILITY

This LCO applies during movement of irradiated fuel assemblies in the spent fuel pool, since the potential for a release of fission products exists.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

When the initial conditions for prevention of an accident cannot be met, steps should be taken to preclude the accident from occurring. When the spent fuel pool water level is lower than the required level, the movement of irradiated fuel assembly in the spent fuel pool is immediately suspended. This does not preclude movement of a fuel assembly to a safe position.

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODES 1, 2, 3, and 4, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.15.1

This SR is done during the movement of irradiated fuel assemblies as stated in the Applicability. This SR verifies sufficient fuel storage pool water is available in the event of a fuel handling accident. The water level in the spent fuel pool must be checked periodically. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

During refueling operations, the level in the spent fuel pool is in equilibrium with the refueling canal, and the level in the refueling canal is checked daily in accordance with SR 3.9.7.1.

REFERENCES

1. FSAR, Section 9.1.2.
2. FSAR, Section 9.1.3.
3. FSAR, Section 9.1.4.3.4, 15.4.5 and 15.5.22.
4. Regulatory Guide ~~1.25~~1.183, July 2000.
5. 10 CFR ~~100.11~~50.67.

B 3.7 PLANT SYSTEMS

B 3.7.18 Secondary Specific Activity

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator tube outleakage from the Reactor Coolant System (RCS). Under steady state conditions, the activity is primarily iodines with relatively short half lives and, thus, indicates current conditions. During transients, I-131 spikes have been observed as well as increased releases of some noble gases. Other fission product isotopes, as well as activated corrosion products in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the environment because of normal operation, anticipated operational occurrences, and accidents.

This limit is lower than the activity value that might be expected from a ~~1~~0.75 gpm tube leak (LCO 3.4.13, "RCS Operational LEAKAGE") of primary coolant at the limit of 1.0 μCi/gm (LCO 3.4.16, "RCS Specific Activity"). The steam line failure is assumed to result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and the reactor coolant LEAKAGE. Most of the iodine isotopes have short half lives, (i.e., < 20 hours). Operating at or below 0.1 μCi/gm ensures that in the event of a DBA, offsite doses will be less than 10 CFR ~~100~~50.67 requirements.

APPLICABLE SAFETY ANALYSES

The accident analysis of the main steam line break (MSLB), as discussed in the FSAR, Chapter 15 (Ref. 2) assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of 0.10 μCi/gm DOSE EQUIVALENT I-131. This assumption is used in the analysis for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of an MSLB do not exceed 10 CFR ~~100~~50.67 limits (Ref. 1)~~ for whole body and thyroid dose rates~~.

With the loss of offsite power, the remaining steam generators are available for core decay heat dissipation by venting steam to the atmosphere through the MSSVs and steam generator atmospheric dump valves (ADVs). The Auxiliary Feedwater System supplies the necessary makeup to the steam generators. Venting continues until the reactor coolant temperature and pressure have decreased sufficiently for the Residual Heat Removal System to complete the cooldown.

(continued)

BASES

ACTIONS	A.1 and A.2 (continued) least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.
SURVEILLANCE REQUIREMENTS	SR 3.7.18.1 This SR verifies that the secondary specific activity is within the limits of the accident analysis. A gamma isotopic analysis of the secondary coolant, which determines DOSE EQUIVALENT I-131, confirms the validity of the safety analysis assumptions as to the source terms in post accident releases. It also serves to identify and trend any unusual isotopic concentrations that might indicate changes in reactor coolant activity or LEAKAGE. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.
REFERENCES	1. 10 CFR ~~100.11~~50.67. 2. FSAR, Chapter 15.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.2 AC Sources—Shutdown

BASES

BACKGROUND

A description of the AC sources is provided in the Bases for LCO 3.8.1, "AC Sources — Operating."

APPLICABLE SAFETY ANALYSES

The OPERABILITY of the minimum AC sources during MODES 5 and 6 and during movement of recently irradiated fuel assemblies ensures that:

a. The unit can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate AC electrical power is provided to mitigate events postulated during shutdown, such as a fuel handling accident involving handling recently irradiated fuel. Due to radioactive decay, AC electrical power is only required to mitigate fuel handling accidents involving recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous ~~100~~72 hours).

In general, when the unit is shut down, the Technical Specifications requirements ensure that the unit has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required. The rationale for this is based on the fact that many Design Basis Accidents (DBAs) that are analyzed in MODES 1, 2, 3, and 4 have no specific analyses in MODES 5 and 6. Worst case bounding events are deemed not credible in MODES 5 and 6 because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and in minimal consequences. These deviations from DBA analysis assumptions and design requirements during shutdown conditions are allowed by the LCO for required systems.

During MODES 1, 2, 3, and 4, various deviations from the analysis assumptions and design requirements are allowed within the Required Actions. This allowance is in recognition that certain testing and maintenance activities must be conducted, provided an acceptable level of risk is not exceeded. During MODES 5 and 6, performance of a significant number of required testing and maintenance activities is also required. In MODES 5 and 6, the activities are generally planned and administratively controlled. Relaxations from MODE 1, 2, 3, and 4 LCO requirements are acceptable during shutdown modes based on:

a. The fact that time in an outage is limited. This is a risk prudent goal as well as a utility economic consideration.

(continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.5 DC Sources-Shutdown

BASES

BACKGROUND

A description of the DC sources is provided in the Bases for LCO 3.8.4, "DC Sources-Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident and transient analyses in the FSAR, Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume that Engineered Safety Feature systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the diesel generators, emergency auxiliaries, and control and switching during all MODES of operation.

The OPERABILITY of the DC subsystems is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum DC electrical power sources during MODES 5 and 6 and during movement of recently irradiated fuel assemblies ensures that:

a. The unit can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate DC electrical power is provided to mitigate events postulated during shutdown, such as a fuel handling accident involving handling recently irradiated fuel. Due to radioactive decay, DC electrical power is only required to mitigate fuel handling accidents involving recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous ~~100~~72 hours).

The DC sources satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The DC electrical power subsystems, each subsystem consisting of one battery, one battery charger per battery, and the corresponding control equipment and interconnecting class 1E cabling within the subsystem, are required to be OPERABLE to support required trains of the distribution systems required OPERABLE by LCO 3.8.10, "Distribution Systems-Shutdown." An OPERABLE subsystem consists of a DC bus connected to a battery with an OPERABLE battery charger which is fed from an OPERABLE AC vital bus (Ref B.3.8.10).

(continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.8 Inverters-Shutdown

BASES

BACKGROUND

A description of the inverters is provided in the Bases for LCO 3.8.7, "Inverters - Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume Engineered Safety Feature systems are OPERABLE. The Class 1E UPS inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the Reactor Protective System and Engineered Safety Features Actuation System instrumentation and controls so that the fuel, Reactor Coolant System, and containment design limits are not exceeded.

The OPERABILITY of the inverters is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum inverters to each 120 VAC vital bus during MODES 5 and 6 and during movement of recently irradiated fuel assemblies ensures that:

a. The unit can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate power is available to mitigate events postulated during shutdown, such as a fuel handling accident involving handling recently irradiated fuel. Due to radioactive decay, AC and DC inverters are only required to mitigate fuel handling accidents involving recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous ~~100~~72 hours).

The inverters were previously identified as part of the distribution system and, as such, satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This ensures the availability of sufficient inverter power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents involving handling recently irradiated fuel).

(continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.10 Distribution Systems - Shutdown

BASES

BACKGROUND

A description of the Class 1E AC, DC, and 120 VAC vital bus electrical power distribution systems is provided in the Bases for LCO 3.8.9, "Distribution Systems - Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident and transient analyses in the FSAR, Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume Engineered Safety Feature (ESF) systems are OPERABLE. The Class 1E AC, DC, and 120 VAC vital bus electrical power distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System, and containment design limits are not exceeded.

The OPERABILITY of the Class 1E AC, DC, and 120 VAC vital bus electrical power distribution system is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum Class 1E AC, DC, and 120 VAC vital bus electrical power distribution subsystems during MODES 5 and 6, and during movement of recently irradiated fuel assemblies ensures that:

a. The unit can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate power is provided to mitigate events postulated during shutdown, such as a fuel handling accident involving handling recently irradiated fuel. Due to radioactive decay, AC and DC electrical power is only required to mitigate fuel handling accidents involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous ~~100~~72 hours).

The Class 1E AC, DC, and 120 VAC electrical power distribution systems satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Various combinations of subsystems, equipment, and components are required OPERABLE by other LCOs, depending on the specific plant condition. An OPERABLE AC subsystem shall consist of a 4kV vital bus powered from at least one energized offsite power source with the capability of being powered from an OPERABLE DG. The DG may be the DG associated with that bus or, with administrative controls in place, a DG that can be cross-tied (via the startup cross-tie feeder breakers) to another bus. However, credit for this cross-tie capability

(continued)

B 3.9 REFUELING OPERATIONS

B 3.9.4 Containment Penetrations

BASES

BACKGROUND

In MODES 1, 2, 3, and 4, the containment serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained ~~well~~ within the requirements of 10 CFR ~~100~~50.67. Additionally, in all operating modes the containment provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions. However during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to maintain the pressure boundary can be less stringent. An analysis has been performed that shows by meeting the LCO, during CORE ALTERATION and movement of irradiated fuel assemblies in containment, the potential release as a result of a fuel handling accident (FHA) will remain ~~well~~ within the requirements of 10 CFR ~~100~~50.67 limits.

The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. The LCO requires that during CORE ALTERATIONS or the movement of irradiated fuel assemblies the equipment hatch must be capable of being closed and held in place by at least four bolts. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced.

The containment Personnel Air Lock (PAL) and Emergency Air Lock (EAL), which are also part of the containment pressure boundary, provide a means for personnel and emergency access during MODES 1, 2, 3, and 4 unit operation in accordance with LCO 3.6.2, "Containment Air Locks." Each of these air locks has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of unit shutdown when the PAL and EAL are not required to be closed, the door interlock mechanisms may be disabled, allowing both doors of each of the air locks to remain open for extended periods when frequent containment entry is necessary.

(continued)

BASES

BACKGROUND (continued)

Per the FHA inside containment analysis, there are no closure restrictions required to limit any release to ~~well~~ within the requirements of 10 CFR ~~100~~50.67 limits for offsite dose as the result of a fuel handling accident during refueling. The LCO requirements for containment penetration closure are not provided to meet regulatory requirements, but rather to reduce the potential volume of the release of fission product radioactivity within containment to the environment.

The Containment Purge and Exhaust System includes two subsystems. The normal subsystem includes a 48 inch purge penetration and a 48 inch exhaust penetration ~~in which the flow path is limited to being open 200 hour or less per calendar year~~. The second subsystem, a pressure equalization system provides a single 12 inch supply and exhaust penetration. ~~The three valves in the 12 inch pressure equalization penetration can be opened intermittently.~~ Each of these systems are qualified to closed automatically by the Engineered Safety Features Actuation System (ESFAS). ~~Neither of the subsystems is subject to a Specification in MODE 5.~~

In MODE 6, large air exchanges are necessary to conduct refueling operations. The normal 48 inch purge system is used for this purpose, and all four valves are closed by the ESFAS in accordance with LCO 3.3.6, "Containment Purge and Exhaust Isolation Instrumentation."

The pressure equalization system is disassembled and used in MODE 6 for other outage functions.

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side if they are not opened under administrative controls. Isolation may be achieved by an OPERABLE automatic isolation valve, or by a manual isolation valve, blind flange, or equivalent. The fuel transfer tube is open but closure is provided by an equivalent isolation of a water loop seal. Equivalent isolation methods must be approved and may include use of a material that can provide a temporary, ventilation barrier for the other containment penetrations during fuel movements (Ref. 1).

Although the historic severe weather patterns for DCPP do not require consideration of tornados as part of the design basis, severe weather conditions might occur at the site that could necessitate closure of open penetrations with direct access to the outside atmosphere during refueling operations with core alterations or irradiated fuel movement inside containment. As a result, administrative procedures shall require that closure of these penetrations be initiated immediately if severe weather warnings are in effect. All fuel handling activities inside containment shall be suspended until closure of the equipment hatch is completed.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSIS

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 2). Fuel handling accident inside the containment is based on dropping a single irradiated fuel assembly of which all 264 fuel rods rupture. In addition the analysis assumes free and rapid communication of air from the containment to the outside environment; the accident occurs ~~100~~72 hours after reactor shutdown; almost instantaneous release of the entire containment volume to the outside atmosphere; ~~thyroid dose conversion factors based on ICRP 30 (Ref. 4);~~ a radial peaking factor of 1.65 based on 105% full power operation; and the other guidance from RG ~~1.25~~1.183. (Ref 5).

The requirements of LCO 3.9.7, "Refueling Cavity Water Level," and the minimum decay time of ~~100~~72 hours prior to CORE ALTERATIONS *ensure that the release of fission product radioactivity, subsequent to a* fuel handling accident, results in doses ~~that are well within the guideline values specified in 10 CFR 100. Standard Review Plan, Section 15.7.4,~~

~~Rev. 1 (Ref. 3), defines "well within" 10 CFR 100 to be 25% or less of the 10 CFR 100 values. The acceptance limits for offsite radiation exposure will be 25% of 10 CFR 100 values.~~less than the accident dose criteria specified in Table 6 of RG 1.183 (Ref. 5).

Containment penetrations satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

(continued)

BASES (continued)

REFERENCES

1. Design Criteria Memorandum T-16, Containment Functions.
2. FSAR, Section 15.4.5 and 15.5.22.
3. ~~NUREG-0800, Section 15.7.4, Rev. 1, July 1981~~Not Used.
4. ~~International Commission on Radiological Protection Publication 30, "Limits for Intakes of Radionuclides by Workers," 1979~~ Not Used.
5. RG ~~1.25~~1.183, July 2000.

B 3.9 REFUELING OPERATIONS

B 3.9.7 Refueling Cavity Water Level

BASES

BACKGROUND

The movement of irradiated fuel assemblies within containment requires a minimum water level of 23 ft above the top of the reactor vessel flange. During refueling, this maintains sufficient water level in the containment, refueling canal, fuel transfer canal, refueling cavity, and spent fuel pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. ~~2 and 6~~1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to ~~≤ 25% of 10 CFR 100 limits, as provided by the guidance of Reference 3~~the acceptance criteria of 10 CFR 50.67 (Ref. 4) and RG 1.183 (Ref. 1).

APPLICABLE SAFETY ANALYSIS

During CORE ALTERATIONS and movement of irradiated fuel assemblies, the water level in the refueling canal and the refueling cavity is an initial condition design parameter in the analysis of a fuel handling accident in containment, as postulated by Regulatory Guide ~~1.25~~ 1.183 (Ref. 1). A minimum water level of 23 ft allows a decontamination factor of 200 (Appendix B (2) of Ref. ~~6~~1 approved in Ref. 7) to be used in the accident analysis for iodine. This relates to the assumption that 99.5% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the refueling cavity water. The fuel pellet to cladding gap is assumed to contain ~~10~~12% of I-131 and 10% of core~~the total fuel rod iodine~~ inventory of all other iodine isotopes (Ref. ~~1~~2).

The fuel handling accident analysis inside containment is described in Reference 2. With a minimum water level of 23 ft and a minimum decay time of ~~100~~72 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 offsite doses are maintained ~~well~~ within allowable limits (Refs. 1and 4~~, and 5~~).

Refueling cavity water level satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

A minimum refueling cavity water level of 23 ft above the reactor vessel flange is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment are within acceptable limits, as provided by the guidance of Reference ~~3~~1.

(continued)

BASES (continued)

APPLICABILITY	LCO 3.9.7 is applicable during CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts, and when moving irradiated fuel assemblies within containment. The LCO minimizes the possibility of a fuel handling accident in containment that is beyond the assumptions of the safety analysis. If irradiated fuel assemblies are not present in containment, 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 pool are covered by LCO 3.7.15, "Fuel Storage Pool Water Level."
ACTIONS	A.1 With a water level of < 23 ft above the top of the reactor vessel flange, all operations involving movement of irradiated fuel assemblies within the containment shall be suspended immediately to ensure that a fuel handling accident cannot occur. The suspension of fuel movement 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 23 ft above the top of the reactor vessel flange ensures that the design basis for the analysis of the postulated fuel handling accident during refueling operations is met. Water at the required level above the top of the reactor vessel flange limits the consequences of damaged fuel rods that are postulated to result from a fuel handling accident inside containment (Ref. 2). The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.
REFERENCES	1. Regulatory Guide ~~1.25, March 23, 1972~~1.183, July 2000. 2. FSAR, Section 15.4.5 and 15.5.22. 3. ~~NUREG-0800, Section 15.7.4.~~ Not Used 4. 10 CFR ~~100.10~~50.67. 5. ~~Malinowski, D. D., Bell, M. J., Duhn, E., and Locante, J., WCAP-828, Radiological Consequences of a Fuel Handling Accident, December 1971.~~ Not Used 6. ~~Appendix B (2) of Regulatory Guide 1.183, July 2000~~Not Used. 7. License Amendment 155/155, October 21, 2002