Exelon Nuclear
200 Exelon Way
Kennett Square, PA 19348

www.exeloncorp.com

Exelon Nuclear

10 CFR 50.90

July 14, 2003

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

Peach Bottom Atomic Power Station, Units 2 & 3
Facility Operating License Nos. DPR-44 and DPR-56
NRC Docket Nos. 50-277 and 50-278

Subject: Request for License Amendments Related to Application of Alternative Source Term

References:
(1) U. S. Nuclear Regulatory Commission, Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000

(2) U. S. Nuclear Regulatory Commission Standard Review Plan 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Revision 0, July 2000

(3) Technical Specification Task Force (TSTF) Traveler, TSTF-51, "Revise Containment Requirements During Handling of Irradiated Fuel and Core Alterations," Revision 2

Pursuant to 10 CFR 50.67, "Accident source term," and 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company, LLC (EGC) hereby requests an amendment to the Facility Operating Licenses listed above. The proposed change is requested to support application of an alternative source term (AST) methodology, with the exception that Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactor Sites," will continue to be used as the radiation dose basis for equipment qualification.

On December 23, 1999, the NRC published regulation 10 CFR 50.67 in the Federal Register. This regulation provides a mechanism for operating license holders to revise the current accident source term used in design-basis radiological analyses with an AST. Regulatory guidance for the implementation of AST is provided in Reference 1. This regulatory guide provides guidance on acceptable applications of ASTs. The use of AST changes only the regulatory assumptions regarding the analytical treatment of the design basis accidents (DBAs)

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EGC has performed radiological consequence analyses of the four DBAs that result in offsite exposure to support a full-scope implementation of AST as described in Reference 1. The AST analyses for Peach Bottom Atomic Power Station (PBAPS), Units 2 & 3, were performed following the guidance in References 1 and 2.

The proposed changes to the current licensing basis for PBAPS that are justified by the AST analyses include:

- Technical Specifications (TS) and associated Bases revisions to reflect implementation of AST assumptions;
- TS and associated Bases revisions to increase primary containment allowable leakage;
- TS and associated Bases revisions to increase main steam isolation valve allowable leakage;
- TS and associated Bases revisions to change the applicability requirements for the following systems during movement of irradiated fuel assemblies in secondary containment and to reflect that these systems are no longer required to be operable during core alterations:
    - Standby Gas Treatment (SGT),
    - Secondary Containment, and
    - Main Control Room Emergency Ventilation (MCREV)
- TS and associated Bases revisions to reflect use of the Standby Liquid Control (SLC) System to buffer suppression pool pH to prevent iodine re-evolution during a postulated radiological release;
- TS revisions to the Ventilation Filter Testing Program to reflect that the AST analyses do not take credit for SGT filters and to revise the acceptance criteria for MCREV methyl iodide penetration.

The proposed changes related to the applicability requirements during movement of irradiated fuel assemblies are consistent with Technical Specification Task Force Traveler (TSTF)-51, Revision 2 (Reference 3). TSTF-51, Revision 2, was approved by the NRC on November 1, 1999. TSTF-51 changes the TS operability requirements for certain engineered safety features such that they are not required after sufficient radioactive decay has occurred to ensure that offsite doses remain within limits. Since a portion of this license amendment request is based on TSTF-51, EGC is committing to the applicable provisions of Nuclear Utilities Management and Resources Council (NUMARC) 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 3, as described in TSTF-51. NUMARC 93-01 provides recommendations on the need to initiate actions to verify and/or re-establish secondary containment, and if needed, primary containment, in the event of a dropped fuel assembly.

There is an additional proposed change to the UFSAR. UFSAR section 5.2.4.3.2, "Minimum Containment Pressure Available," has been revised to reflect the impact of AST assumptions. Portions of the text and Figure 5.2.16, "Minimum Containment Pressure Available and Containment Overpressure License," were previously approved by the NRC in a Safety Evaluation dated August 14, 2000.

EGC has been an active participant on the NEI Control Room Habitability (CRH) Task Force and understands the NRC position regarding CRH and acknowledges the fact that a Generic Letter has been issued. This submittal does not directly address the CRH issue other than to provide an increase in the assumed unfiltered inleakage value. However, EGC will provide a formal response to the Generic Letter. Although an ASTM E741 tracer gas test has not been performed to date, the assumed unfiltered inleakage value in the AST dose analyses is greater than 50% of the full Control Room pressurization air flow. With the assumed inleakage value this high, it is EGC's judgment that the measured value is not reasonably expected to exceed this assumed value. Other CRH actions will be addressed via the Generic Letter response.

This request is subdivided as follows.

1. Attachment 1 provides a Description of Proposed Changes, Technical Analysis, and Regulatory Analysis.

2. Attachment 2 provides the Markup of Technical Specification pages.

3. Attachment 3 provides the Markup of Technical Specification Bases pages (for Information only).

4. Attachment 4 provides the Retyped Technical Specification pages.

5. Attachment 5 provides the Retyped Technical Specification Bases pages (for Information only).

6. Attachment 6 provides the List of Commitments resulting from the proposed changes.

7. Attachment 7 provides a compact disk (CD) containing PBAPS meteorological data for the calculation of the atmospheric dispersion factors (X/Qs). The CD also provides the PAVAN and ARCON96 input parameters.

8. Attachment 8 provides the Mark-up of UFSAR section 5.2.4.3.2 including Figure 5.2.16.

The proposed changes have been reviewed by the Plant Operations Review Committee and approved by the Nuclear Safety Review Board in accordance with the requirements of the EGC Quality Assurance Program.

EGC requests approval of the proposed amendments by July 14, 2004. Once approved, the amendments shall be implemented within 60 days. This implementation period will provide adequate time for the affected station documents to be revised using the appropriate change control mechanisms.

In accordance with 10 CFR 50.91(b), EGC is notifying the State of Pennsylvania of this application for changes to the TS by transmitting a copy of this letter and its attachments to the designated State Official.

If you have any questions or require additional information, please contact me at (610) 765- 5664.

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

Respectfully,

Executed on 07-14-03

Michael P. Gallagher

Michael P. Gallagher
Director, Licensing and Regulatory Affairs
Mid-Atlantic Regional Operating Group

Attachments:
1. Description of Proposed Changes, Technical Analysis, and Regulatory Analysis
2. Markup of Technical Specification pages
3. Markup of Technical Specification Bases pages *(Information only)*
4. Retyped Technical Specification pages
5. Retyped Technical Specification Bases pages *(Information only)*
6. List of Commitments
7. PBAPS Meteorological data *(Information only)*
8. Mark-up of UFSAR section 5.2.4.3.2

cc:
H. J. Miller, Administrator, Region I, USNRC
A. C. McMurtray, USNRC Senior Resident Inspector, PBAPS
J. Boska, Senior Project Manager, USNRC (by FedEx)
R. R. Janati - Commonwealth of Pennsylvania

ATTACHMENT 1

Peach Bottom Atomic Power Station
Units 2 & 3

License Amendment Request
"PBAPS Alternative Source Term Implementation"

1.0 DESCRIPTION

In accordance with 10 CFR 50.67, "Accident source term," and 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company, LLC (EGC) requests a change to Appendix A, Technical Specifications (TS), of Facility Operating License Nos. DPR-44 and DPR-56 for the Peach Bottom Atomic Power Station (PBAPS), Units 2 & 3. The proposed changes are requested to support application of an alternative source term (AST) methodology, with the exception that Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactor Sites," (Reference 7.1) will continue to be used as the radiation dose basis for equipment qualification.

EGC has performed radiological consequence analyses of the four Design Basis Accidents (DBAs) that result in offsite exposure (i.e., Loss of Coolant Accident (LOCA), Main Steam Line Break (MSLB), Fuel Handling Accident (FHA), and Control Rod Drop Accident (CRDA)) to support a full-scope implementation of AST as described in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (Reference 7.2). The AST analyses for PBAPS were performed following the guidance in Reference 7.2 and Standard Review Plan 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms" (Reference 7.3).

The proposed changes to the TS will allow PBAPS to apply the results of the plant-specific AST analyses using the guidance in Reference 7.2 and meet the requirements of 10 CFR 50.67. Approval of this change will provide a realistic source term for PBAPS that will result in a more accurate assessment of DBA radiological doses. This allows relaxation of some current licensing basis requirements as described in Section 2.0, Proposed Changes. Adopting the AST methodology may also support future evaluations and license amendments.

This proposed change would increase allowable Main Steam Isolation Valve (MSIV) and Primary Containment Isolation Valve (PCIV) leakage. Unplanned MSIV and PCIV repairs are potential contributors to increased outage duration and unplanned personnel exposure. These leakage limits are associated with TS for the operability of primary containment isolation valves (i.e., TS 3.6.1.3). Under the AST assumptions proposed, the MSIV and PCIV work can be strategically planned via work planning process to maintain work ALARA and maximize the incremental benefit of work being performed in high dose areas.

In addition, implementation of AST will not require secondary containment to be operable throughout the refueling outage (except during Operations with the Potential for Draining the Reactor Vessel (OPDRVs) and movement of recently irradiated fuel). This proposed change provides the flexibility of performing fuel floor activities (such as control rod blade exchanges and fuel movements) as well as movement of large equipment through the secondary containment boundary in support of outage activities while remaining within all safety limits.

Other benefits of AST are the cost savings that will be achieved by extending the frequency of charcoal and HEPA filter testing for Main Control Room Emergency Ventilation (MCREV).

2.0 PROPOSED CHANGES

The proposed changes related to the applicability requirements during movement of irradiated fuel assemblies are consistent with Technical Specification Task Force Traveler (TSTF)-51, "Revise Containment Requirements During Handling of Irradiated Fuel and Core Alterations," Revision 2. TSTF-51, Revision 2, was approved by the NRC on October 15, 1999. TSTF-51 changes the TS operability requirements for engineered safety features such that they are not required after sufficient radioactive decay has occurred to ensure that offsite doses remain within limits. Since a portion of this license amendment request is based on TSTF-51, EGC is committing to the applicable provisions of Nuclear Utilities Management and Resources Council (NUMARC) 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 3, as described in TSTF-51. NUMARC 93-01 provides recommendations on the need to initiate actions to verify and/or re-establish secondary containment, and if needed, primary containment, in the event of a dropped fuel assembly.

2.1 TS Section 1.1, "Definitions"

The proposed change revises the definition of DOSE EQUIVALENT I-131 in TS Section 1.1 to remove the word "thyroid" and to add a reference to Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1989. This change reflects the application of AST assumptions.

2.2 TS Section 3.1.7, "Standby Liquid Control (SLC) System"

The proposed change revises the Applicability of TS Section 3.1.7 to add the requirement for the LCO to be met in Mode 3. This change implements AST assumptions regarding the use of the SLC System to buffer the suppression pool following a LOCA involving significant fission product release. The required actions for Condition D are being revised to add an additional requirement to be in Mode 4 with a completion time of 36 hours.

2.3 TS Section 3.3.6.1, "Primary Containment Isolation Instrumentation"

TS Section 3.3.6.1, Table 3.3.6.1-1 lists the applicability requirements for Primary Containment Isolation Instrumentation. The proposed change revises the applicability of the SLC System Initiation Function of the Reactor Water Cleanup System Isolation instrumentation to add the requirement for this function to be operable in Mode 3. The revised applicability for this function is consistent with the revised applicability for the SLC System.

2.4 TS Section 3.3.6.2, "Secondary Containment Isolation Instrumentation"

The proposed change revises footnote (b) of TS Table 3.3.6.2-1 by deleting, "CORE ALTERATIONS, and during," which eliminates the requirement for Function 3 (i.e., Reactor Building Ventilation Exhaust Radiation – High) and Function 4 (i.e., Refueling Floor Ventilation Exhaust Radiation – High) of the Secondary Containment Isolation Instrumentation to be operable during core alterations. The proposed change also relaxes TS requirements to require these functions to be operable only when handling recently irradiated fuel. Changes to the TS Bases define the time period that must elapse to consider fuel to be beyond recently irradiated. With the application of AST, secondary containment is not credited for the FHA after a 24-hour decay period.

2.5 TS Section 3.3.7.1, "Main Control Room Emergency Ventilation (MCREV) System Instrumentation"

The proposed change deletes "During CORE ALTERATIONS" from the applicability statement for TS LCO 3.3.7.1 and relaxes TS requirements for LCO 3.3.7.1 to be applicable only when handling recently irradiated fuel. Changes to the TS Bases define the time period that must elapse to consider the fuel to be beyond recently irradiated. With the application of AST, the MCREV System is not credited for the FHA after a 24-hour decay period.

2.6 TS Section 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)"

The proposed change revises Surveillance Requirement (SR) 3.6.1.3.14 to increase the allowable limit for the combined leakage rate for all MSIV leakage paths. Currently, the allowable limit is less than or equal to 11.5 scfh for each MSIV when tested at greater than or equal to 25 psig. This limit will be increased to less than or equal to 174 scfh for all four main steam lines and less than or equal to 100 scfh for any one main steam line, when tested at greater than or equal to 25 psig. The revised SR 3.6.1.3.14 reads:

Verify combined MSIV leakage rate for all four main steam lines is $\leq$ 174 scfh, and $\leq$100 scfh for any one steam line, when tested at $\geq$ 25 psig.

The Frequency for SR 3.6.1.3.14 is "In accordance with the Primary Containment Leakage Rate Testing Program," and this frequency is not being changed.

2.7 TS Section 3.6.4.1, "Secondary Containment"

The proposed change deletes "During CORE ALTERATIONS" from the applicability statement for TS LCO 3.6.4.1 and relaxes TS requirements to require LCO 3.6.4.1 to be applicable only when handling recently irradiated fuel. Changes to the TS Bases define the time period that must elapse to consider the fuel to be beyond recently irradiated. In addition, the proposed change revises Condition C, and associated required actions and completion times, to reflect the revision of the applicability requirements for LCO 3.6.4.1. With the application of AST, secondary containment is not credited for the FHA after a 24-hour decay period. The final proposed change to TS 3.6.4.1 is to SR 3.6.4.1.3, increasing the secondary containment drawdown time from less than or equal to 120 seconds to less than or equal to 15 minutes.

2.8 TS Section 3.6.4.2, "Secondary Containment Isolation Valves (SCIVs)"

The proposed change deletes "During CORE ALTERATIONS" from the applicability statement for TS LCO 3.6.4.2 and relaxes TS requirements to require LCO 3.6.4.2 to be applicable only when handling recently irradiated fuel. Changes to the TS Bases define the time period that must elapse to consider the fuel to be beyond recently irradiated. In addition, the proposed change revises Condition D, and associated required actions and completion times, to reflect the revision of the applicability requirements for LCO 3.6.4.2. With the application of AST, closure of secondary containment isolation valves is not credited for the FHA after a 24-hour decay period.

2.9 TS Section 3.6.4.3, "Standby Gas Treatment (SGT) System"

The proposed change deletes "During CORE ALTERATIONS" from the applicability statement for TS LCO 3.6.4.3 and relaxes TS requirements to require LCO 3.6.4.3 to be applicable only when handling recently irradiated fuel. Changes to the TS Bases define the time period that must elapse to consider the fuel to be beyond recently irradiated. In addition, the proposed change revises Condition C and Condition E, and associated required actions and completion times, to reflect the revision of the applicability requirements for LCO 3.6.4.3. These changes are being made to reflect that with application of AST, the SGT System is no longer required to be operable during movement of irradiated fuel assemblies, that have decayed at least 24-hours, in the secondary containment, or during core alterations, since this system is not credited for the FHA after a 24-hour decay period. SR 3.6.4.3.1 is also revised to reflect that the SGT system heater is no longer required.

2.10 TS Section 3.7.4, "Main Control Room Emergency Ventilation (MCREV) System"

The proposed change deletes "During CORE ALTERATIONS" from the applicability statement for TS LCO 3.7.4 and relaxes TS requirements to require LCO 3.7.4 to be applicable only when handling recently irradiated fuel. Changes to the TS Bases define the time period that must elapse to consider the fuel to be beyond recently irradiated. In addition, the proposed change revises Condition C and Condition E, and associated required actions and completion times, to reflect the revision of the applicability requirements for LCO 3.7.4. The AST analyses do not take credit for MCREV System operation during movement of irradiated fuel that has decayed at least 24-hours, in secondary containment, or during core alterations.

2.11 TS Section 3.8.1, "AC Sources – Operating"

The proposed change revises the note for SR 3.8.1.21 to add "recently" to the phrase referencing movement of irradiated fuel assemblies in secondary containment. This change is needed to be consistent with the proposed change to TS Section 3.8.2, as described below.

2.12 TS Section 3.8.2, "AC Sources – Shutdown"

The proposed change relaxes TS requirements to require LCO 3.8.2 to be applicable only when handling recently irradiated fuel. Changes to the TS Bases define the time period that must elapse to consider the fuel to be beyond recently irradiated. In

addition, the proposed change revises Required Actions A.2.2, B.2.2, and C.2 to reflect the revision of the applicability requirements for LCO 3.8.2. AST analyses do not take credit for operation of the MCREV and SGT Systems, or secondary containment, following a FHA involving movement of irradiated fuel assemblies that have decayed at least 24-hours.

2.13 TS Section 3.8.4, "DC Sources – Operating"

The proposed change relaxes TS requirements to require the SR 3.8.4.9 Note to be applicable only when handling recently irradiated fuel.

2.14 TS Section 3.8.5, "DC Sources – Shutdown"

The proposed change relaxes TS requirements to require LCO 3.8.5 and the SR 3.8.5.2 Note to be applicable only when handling recently irradiated fuel. Changes to the TS Bases define the time period that must elapse to consider the fuel to be beyond recently irradiated. In addition, the proposed change revises Required Action A.2.2 to reflect the revision of the applicability requirements for LCO 3.8.5. AST analyses do not take credit for operation of the MCREV and SGT Systems, or secondary containment, following a FHA involving movement of irradiated fuel assemblies that have decayed at least 24-hours.

2.15 TS Section 3.8.8, "Distribution Systems – Shutdown"

The proposed change relaxes TS requirements to require LCO 3.8.8 to be applicable only when handling recently irradiated fuel. Changes to the TS Bases define the time period that must elapse to consider the fuel to be beyond recently irradiated. In addition, the proposed change revises Required Action A.2.2 to reflect the revision of the applicability requirements for LCO 3.8.8. AST analyses do not take credit for operation of the MCREV and SGT Systems, or secondary containment, following a FHA involving movement of irradiated fuel assemblies that have decayed at least 24-hours.

2.16 TS Section 5.5.7, "Ventilation Filter Testing Program (VFTP)"

The proposed change deletes the TS requirements for SGT System charcoal adsorbers and HEPA filters from applicable portions of TS 5.5.7 since this equipment is no longer credited in the LOCA accident analyses. As a result, TS 5.5.7.a, TS 5.5.7.b, and TS 5.5.7.c are revised to delete all references to the SGT System. Although the SGT charcoal adsorbers and HEPA filters requirements are being eliminated, EGC does not plan to remove this equipment from PBAPS.

The proposed change revises Section 5.5.7.c to increase methyl iodide penetration acceptance criteria for the MCREV System from 5% to 15%. Application of AST supports increasing the methyl iodide penetration percentage. The SGT system function of assuring an elevated release through the Main Stack remains an important safety function; therefore, there is no change to the TS 5.5.7.d requirement for testing SGT System pressure drop.

The proposed change also revises Section 5.5.7.e to delete the requirement for periodic heater testing. Since the proposed change deletes the requirement for SGT

System charcoal adsorbers from the TS, the need to periodically verify a SGT System heater dissipates the required wattage is eliminated. Although the requirement to verify the heater dissipation is being eliminated, EGC does not plan to remove this heater from the SGT system.

2.17 TS Section 5.5.12, "Primary Containment Leakage Rate Testing Program"

The proposed change increases the maximum allowable primary containment leakage rate, L_a, at P_a, from 0.5% to 0.7% of primary containment air weight per day. Application of AST supports the increase in maximum allowable primary containment leakage rate.

2.18 UFSAR Section 5.2.4.3.2, "Minimum Containment Pressure Available"

The proposed change to the UFSAR revises the Containment Overpressure License (COPL) of UFSAR Figure 5.2.16 to reduce the COPL such that additional margin is maintained between the revised Minimum Containment Pressure Available (MCPA) and the proposed COPL.

3.0 BACKGROUND

On December 23, 1999, the NRC published regulation 10 CFR 50.67 in the Federal Register. This regulation provides a mechanism for operating license holders to revise the current accident source term used in design-basis radiological analyses with an AST. Regulatory guidance for the implementation of AST is provided in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors", July 2000 (Reference 7.2). This regulatory guide provides guidance on acceptable applications of ASTs. The use of AST changes only the regulatory assumptions regarding the analytical treatment of the design basis accidents (DBAs).

The fission product release from the reactor core into containment is referred to as the "source term," and it is characterized by the composition and magnitude of the radioactive material, the chemical and physical properties of the material, and the timing of the release from the reactor core. Since the publication of Reference 7.1, significant advances have been made in understanding the composition and magnitude, chemical form, and timing of fission product releases from severe nuclear power plant accidents. Many of these insights developed out of the major research efforts started by the NRC and the nuclear industry after the accident at Three Mile Island. NUREG-1465 (Reference 7.4) was published in 1995 with revised ASTs for use in the licensing of future Light Water Reactors (LWRs). The NRC, in 10 CFR 50.67, later allowed the use of the ASTs described in NUREG-1465 at operating plants. This NUREG represents the result of decades of research on fission product release and transport in LWRs under accident conditions. One of the major insights summarized in NUREG-1465 involves the timing and duration of fission product releases.

The five release phases representing the progress of a severe accident in a LWR are described in NUREG-1465 as:

1. Coolant Activity Release

2. Gap Activity Release
3. Early In-Vessel Release
4. Ex-Vessel Release
5. Late In-Vessel Release

Phases 1, 2, and 3 are considered in current DBA evaluations; however, they are all assumed to occur instantaneously. Phases 4 and 5 are related to severe accident evaluations. Under the AST, the coolant activity release is assumed to occur instantaneously and end with the onset of the gap activity release.

The requested license amendment involves a full-scope application of the AST, addressing the composition and magnitude of the radioactive material, its chemical and physical form, and the timing of its release as described in Reference 7.2.

EGC has performed radiological consequence analyses of the four DBAs that result in offsite exposure (i.e., LOCA, MSLB, FHA, and CRDA). These analyses were performed to support full scope implementation of AST. The AST analyses have been performed in accordance with the guidance in References 7.2 and 7.3. The implementation consisted of the following steps:

- Identification of the AST based on plant-specific analysis of core fission product inventory,
- Calculation of the release fractions for the four DBAs that could potentially result in control room and offsite doses (i.e., LOCA, MSLB, FHA, and CRDA),
- Analysis of the atmospheric dispersion for the radiological propagation pathways,
- Calculation of fission product deposition rates and transport and removal mechanisms,
- Calculation of offsite and control room personnel Total Effective Dose Equivalent (TEDE) doses, and
- Evaluation of suppression pool pH to ensure that the iodine deposited into the suppression pool during a DBA LOCA does not re-evolve and become airborne as elemental iodine.

The analysis assumptions for the transport, reduction, and release of the radioactive material from the fuel and the reactor coolant are consistent with the guidance provided in applicable appendices of Reference 7.2 for the four analyzed DBAs.

Accordingly, Exelon Generation Company (EGC), as a holder of an operating license issued prior to January 10 1997, is requesting the use of AST for several areas of operational relief for systems used in the event of a Design Basis Accident (DBA), and without crediting the use of certain previously assumed safety systems/functions.

4.0 TECHNICAL ANALYSIS

4.1 Evaluation

4.1.1 Scope

4.1.1.1 Accident Radiological Consequence Analyses

The DBA accident analyses documented in the PBAPS UFSAR that could potentially result in control room and offsite doses were addressed using methods and input assumptions consistent with AST. The following DBAs were addressed:

- CRDA, UFSAR Section 14.6.2;
- MSLB, UFSAR Section 14.6.5;
- LOCA, UFSAR Section 14.9.2; and
- FHA, UFSAR Section 14.6.4.

The analyses were performed in accordance with Reference 7.2 to confirm compliance with the acceptance criteria presented in 10 CFR 50.67.

4.1.2 NUREG-0737, Item II.B.2

EGC has determined that continued compliance will be maintained with NUREG-0737, Item II.B.2, "Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems Which May be Used in Post-Accident Operations." The source term associated with environmental qualification of equipment will remain consistent with previous commitments under 10 CFR 50.49.

4.2 Method of Evaluation

4.2.1 Fission Product Inventory

Pre-AST core source terms were determined based on TID-14844 methodology. That is, inventory was based on the fission product equilibrium based on U-235 fission product yields and isotopic decay constants. This simplified approach is replaced, per Reference 7.2 guidance, with ORIGEN 2.1 (Reference 7.5) methodology used to determine core inventory. This program provides a more complete and accurate simulation of isotopic buildup and depletion, including consideration of fission product yields from all isotopes, and activation as well as decay.

The power level used is approximately equal to the current licensed reactor thermal power of 3514 MWt. These source terms were evaluated at end-of-cycle and at beginning of cycle (100 effective full power days (EFPD) to achieve equilibrium) conditions and worst case inventory used for the selected isotopes. These values were then divided by the power level to obtain activity in units of Ci/MWt. Accident

analyses are based on a 3528 MWt power level that includes the current accident analysis design basis allowance for instrument uncertainty.

Source terms were based on a 2-year fuel cycle with a nominal 711 EFPD per cycle.

These source terms were developed using ORIGEN 2.1. The values extracted from the ORIGEN 2.1 runs are for the standard 60 isotope RADTRAD (Reference 7.6) library except that the activation products Co-58 and Co-60 used RADTRAD default library values.

The reactor coolant fission product inventory for MSLB analysis was based on the Technical Specification limits in terms of Dose Equivalent I-131 (the concentration of I-131 that alone would produce the same dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually assumed), using inhalation Committed Effective Dose Equivalent (CEDE) dose conversion factors from Federal Guidance Report 11 (Reference 7.15).

4.2.2 Radiological Consequence

New calculations were prepared for the simulation of the radionuclide release, transport, removal, and dose estimates associated with the postulated accidents listed in Section 4.1.1.1.

The RADTRAD computer code was used for these calculations. The RADTRAD program is a radiological consequence analysis code used to estimate post-accident doses at plant offsite locations and in the control room. The RADTRAD code is publicly available and is used by the NRC in safety reviews.

Offsite exclusion area boundary (EAB) and low population zone (LPZ) atmospheric dispersion factors (χ/Qs) were calculated using the guidance of Regulatory Guide 1.145 (Reference 7.7) and the PAVAN computer code (Reference 7.8). This code has been used by the NRC in safety reviews.

The χ/Q values resulting at the Control Room Intake were calculated using the NRC-sponsored computer codes ARCON96 (Reference 7.9) and PAVAN, consistent with the procedures in Draft Regulatory Guide DG-1111 (Reference 7.19).

Figure 1 shows the "Layout of Intake and Release Points for PBAPS."

Airborne radioactivity drawn into the control room envelope results in both internal and external dose components that are used in the TEDE dose calculation. The noble gas inventory within the control room is the main contributor to the gamma ray whole body (i.e., external) dose component of the TEDE; the non-noble gas radionuclides, principally iodines, contribute to the internal organ dose component via the inhalation pathway.

The post-accident dose rate in the control room and adjacent areas is due to the shine from the refueling floor airborne source. An additional low-level post-accident external-source gamma ray dose rate component in the control room and adjacent

areas is due to the shine from the refueling floor noble gas airborne source. The shielding provided by the refueling floor slab and the reactor building walls, as well as other concrete structures, was modeled. The post-accident dose rate in the control room due to airborne activity is also directly calculated using the RADTRAD program (Reference 7.6).

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

4.3 Inputs and Assumptions

4.3.1 Accident Radiological Consequence Analyses

Release Mode

Releases were evaluated for full power conditions. The power level used is as described in Section 4.2.1 for each event evaluated.

Onsite Meteorological Measurements Program

The PBAPS meteorological measurement program meets the guidelines of Regulatory Guide 1.23 (Safety Guide 23), "Onsite Meteorological Programs." The tower base areas are on natural surfaces (e.g., short natural vegetation) with towers free from obstructions and micro-scale influences. This ensures that data is representative of the overall site area.

All sensors and related equipment are calibrated according to written procedures designed to ensure adherence to Regulatory Guide 1.23 guidelines for accuracy. Calibrations occur at least every six (6) months, with component checks and adjustments performed when required.

Inspections and maintenance of all equipment is accomplished in accordance with procedures in the instrument manufacturer's manuals. This inspection occurs at least once per week by qualified technicians capable of performing the maintenance if required. In the event that the required maintenance could affect the instrument's calibration, another calibration is performed prior to returning the instrument to service.

Data from the towers are digitized and transmitted to the control room and to an on-site computer for archive storage. Periodically, all digital and analog data are sent to the approved meteorological consultant for data processing and analysis. The digital data acquisition systems are remotely interrogated by the consultant to perform a daily quality check on system performance with the objective of identifying potential problems and to notify plant personnel as soon as possible in order to minimize down-time. This is performed each working day. All analog chart data are subject to a quality check by the consultant. This quality check consists of time continuity, instrument malfunction, inking problems, directional switching problems, negative speeds, missing data, and digital/analog correlation.

Data are compared with other site or regional data for consistency. If deviations occur, they are evaluated and dispositioned as appropriate. Site instrument technicians perform additional checks weekly on the instruments and collect charts for storage.

Meteorological data utilized for this calculation were selected from the historical record of the PBAPS meteorological monitoring tower network. Monitoring records dating back to 1967 and extending through 2001 were reviewed. Examination of the release locations and configurations in conjunction with the sharply varying topography (both in the vicinity of the release and at the desired receptor locations to be addressed) resulted in the selection of three (3) different towers from which representative data for the χ/Q modeling analyses were used. It was desired that this calculation be based upon a continuous five-year period of data common to all 3 towers, and meet NRC Regulatory Guide 1.23 (Safety Guide 23) (Reference 7.11) specifications. Data from the period of 1984 through 1988 was selected because it meets these criteria.

The meteorological measurements program at PBAPS consists of monitoring wind direction, wind speed, temperature, and precipitation. The method used for determining atmospheric stability is delta temperature (T), which measures the vertical temperature difference. When the delta-T method is not available, the sigma theta method can be used. However, for the data used in the calculation, no sigma theta data was used. These data, referenced in ANSI/ANS-2.5-1984 (Reference 7.10), are used to determine the meteorological conditions prevailing at the plant site.

The meteorological towers are equipped with instrumentation that conforms to the system accuracy recommendations of References 7.10 and 7.11. The equipment is placed on booms oriented into the generally prevailing wind at the site.

Recorded meteorological data are used to generate joint frequency distributions of wind direction, wind speed, and atmospheric stability class used to provide estimates of airborne concentrations of gaseous effluents and projected offsite radiation dose. Better than 90% data recovery is attained from each measuring and recording system.

MSExcel© spreadsheet software was used to convert hour-by-hour delta-T data values recorded in "°F", as measured over a height range specified in "feet", into "delta-T/height" values in units of "°C/100 meters", which were then assigned the appropriate hourly stability class values as prescribed by Safety Guide 23. Also, in order to provide wind speed data compatible with the ARCON96 input requirement for "wind speed times 10", raw wind speed values were reformatted within MSExcel© by appropriately adjusting the decimal in the wind speed data, as applicable.

Wind roses and joint frequency distributions were reviewed for meteorological and climatological reasonableness and found to be acceptable prior to use. A review was also conducted on specific hourly data prior to the execution of the atmospheric dispersion calculations in PAVAN and ARCON96. This consisted of manual spot checks of the MSExcel© spreadsheet reformatted data in comparison with the raw data provided by the vendor.

Transport Mode

Atmospheric dispersion coefficients were calculated, for the identified release paths, based on site-specific meteorological data collected between 1984 through 1988. The dispersion coefficients developed represent a change to those used in the current UFSAR analyses. The values currently in the UFSAR are based on Regulatory Guide 1.3 (Reference 7.12). The Regulatory Guide 1.145 results were used for the offsite atmospheric dispersion coefficients for the AST analyses.

The inleakage of unfiltered air into the control room occurs through the control room boundary, system components, and backflow at the control room doors as a result of ingress or egress to or from the control room.

During emergency pressurized modes of operation, the control room ventilation system supplies 3,000 cfm of filtered, outdoor air to maintain the control room at 0.1-inch water column positive pressure with respect to the adjacent areas. Intentionally admitting outdoor air into the control room facilitates reduction of infiltration through the control room boundary by assuring that air is exfiltrating from the zone at an adequate velocity (i.e., a velocity through the control room boundary to develop and maintain a pressure of 0.1-inch water column).

During the isolation mode, infiltration through the control room boundary is initially negligible because the control room will be at a positive pressure at the time of system isolation. Infiltration following isolation is assumed to be 1,600 cfm of unfiltered inleakage, which includes impacts of ingress and egress.

The infiltration through the system components located outside the control room occurs through joints and seams in the ductwork, around damper shafts, through joints and penetrations in the air-handling units, and through the dampers that isolate the control room from non-habitable areas. The inleakage has not been measured via tracer gas testing. This AST analysis assumes a value of 1,600 cfm, which is greater than 50% of the filtered intake. This is a conservative estimate that should easily pass a tracer gas test since no driving force greater than the supplied intake would be expected. Such a test will be scheduled in conjunction with the resolution of the on-going Control Room Habitability (CRH) issue.

The opening and closing of boundary doors can induce infiltration to the control room. The backflow infiltration is conservatively assumed at 10 cfm as recommended by Reference 7.13. This 10 cfm is included in the 1600 cfm total unfiltered inleakage value assumed in the analysis.

Potential adverse interactions between the control room and adjacent zones that may allow the transfer of toxic or radioactive gases into the control room are minimized by maintaining the control room at a positive pressure of 0.1-inch water column with respect to adjacent areas, during emergency pressurized modes.

The standard breathing rates used for control room personnel dose assessments and for the offsite personnel are shown in Table 1, Personnel Dose Inputs. Control room occupancy factors used are also included in Table 1.

Removal Mode

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

4.3.1.1 LOCA Inputs and Assumptions

The key inputs and assumptions used in this analysis are included in Tables 2a through 5. These inputs and assumptions are grouped into three main categories: release, transport, and removal.

LOCA Release Inputs

The LOCA analysis assumes the total containment leak rate of 0.7 percent of primary containment air weight per day. This is an increase from the previously evaluated value of 0.5 percent in TS (0.635 percent in analysis). Primary containment leakage is assumed to be at 0.7% of containment mass per day for 24 hours, 0.392% from 24 to 38 hours, and 0.35% per day from 38 to 720 hours. This is based on conservative post-LOCA containment pressure history.

All of this leakage is considered to be released to the environment unfiltered for an assumed 15-minute secondary containment drawdown time (revised from the current 120 seconds in the SR 3.6.4.1.3). During the drawdown period, a zero velocity vent release is assumed from the RB/TB vent location. This treatment is effectively a ground release assumption (i.e., no upward velocity or buoyancy effect). No credit is taken for mixing in the secondary containment. The drawdown period is completed when the SGTS fans have successfully restored the secondary containment vacuum such that all secondary containment releases are elevated through the Main Stack. This release location provides ample dispersion. Therefore, the SGTS HEPA filters and Charcoal Adsorbers are no longer credited in the LOCA analyses. These changes in assumptions allow elimination of the TS 5.5.7 in-place HEPA and charcoal adsorber testing and the laboratory charcoal adsorber penetration testing for SGTS. The SGTS function of assuring an elevated release through the Main Stack after secondary containment drawdown remains an important safety function. Therefore, the TS 5.5.7 testing of SGTS pressure drop is retained. This, combined with TS testing that assures that a 0.25-inch W.G. vacuum can be developed within the allowed drawdown time (SR 3.6.4.1.3) and maintained (SR 3.6.4.1.4), provides the assurance of release through the Main Stack.

Except for MSIV leakage, no secondary containment bypass leakage pathways have been identified for PBAPS. Analyzed MSIV leakage is increased from the historically used 11.5 scfh per steam line (a total of 46 scfh for all 4 steam lines) to a total of 174 scfh of leakage for all 4 steam lines, and a maximum of 100 scfh for any single line (at a test pressure of $\geq$ 25 psig). The total value limit is based on a plant specific calculation with consideration of ECCS NPSH requirements. MSIV leakage would normally be expected to be transported to the condenser. Main steam piping has been seismically analyzed and supported out to the main stop valves. Therefore, the balance of main steam piping and the condenser are not credited. Based on the shortest steam line and the maximum 100 scfh flow rate, a 12 hour transport delay is determined. This conservative delay is applied to all steam lines including those that are longer or have a lower flow rate. The delay is also

conservative in that the fluid is assumed fully expanded to atmospheric conditions for the entire length.

MSIV leakage is assumed to be to the Turbine Building, and is released instantly and unfiltered without mixing credit through the RB/TB vent stacks on a zero velocity (i.e., no upward velocity or buoyancy effect) vent release (ground level) basis. After 24 hours, the MSIV leak rate is reduced to 77.2%, to 65.4% at 48 hours, to 59.0% at 72 hours, to 55.5% at 96 hours, and to 50% at 157 hours. This reduction timing is different than other primary containment leakage paths since MSIV leakage is measured at $\geq$ 25 psig rather than the $\geq$ 49.1 psig used for other penetrations.

In addition, the LOCA analysis assumes an ECCS leakage rate of 5 gpm outside of primary containment into the secondary containment, from where it is considered to be released to the environment unfiltered as an airborne release based on a flashing fraction of 1.41% for the assumed 15 minute secondary containment drawdown time. This leak rate is conservatively assumed to begin at the onset of the accident and to continue throughout the 30-day duration of the postulated accident. This is a new release path for PBAPS, analyzed to comply with Regulatory Guide 1.183 requirements. This provides the basis for an acceptance criterion for the TS 5.5.2 program.

The Standby Liquid Control (SLC) System has an added design function to maintain suppression pool pH greater than 7 throughout the 30-day accident duration.

Figure 2 illustrates the "LOCA Release Pathways", with an associated Table of "Leakage Rates and Secondary Containment Mixing Parameters".

LOCA Transport Inputs

PBAPS has a once-through main control room emergency ventilation (MCREV) system. This system is initiated on high radiation and/or low flow, and for this accident is assumed to be in service by the start of gap release. The control room free air volume is 176,000 ft^3 and the filtered intake rate is nominally 3,000 cfm. Filter efficiency credit has been modified from the historical 90% assumption to a 70% filter efficiency for elemental and organic iodines, and to 99% for aerosols, based on the presence of HEPA filters before and after the charcoal adsorbers. In addition, a 1,600 cfm allowance for unfiltered inleakage has been used to provide ample margin for this phenomena. This value is more than 1/2 of the pressurization flow, and would typically be a value applied to an unpressurized control room.

These changes in assumptions support relaxation of TS 5.5.7.c laboratory charcoal adsorber penetration testing requirement to 15% for the MCREV, based on the reduced credit taken for filter efficiency, and safety factors allowed per NRC Generic Letter 99-02 (Reference 7.17).

LOCA Removal Inputs

The activity of elemental iodine and aerosols released from the core into the primary containment is reduced by deposition (i.e., plate-out) and settling utilizing the Powers' natural deposition values identified in the RADTRAD code at the 10% probability level. The deposition removal mechanisms are characteristics of the AST methodology and represent a change in the plant design and licensing basis.

Main steam line pipe deposition was modeled using the Brockmann-Bixler model contained in the RADTRAD computer code, as applied to horizontal segments only and based on using the highest flows in the shortest steam lines (i.e., most rapid transport, least deposition). No credit is taken for holdup or plate-out in the main steam lines beyond the turbine stop valves, which is the end of the seismically analyzed and supported main steam piping. No credit is taken for holdup and plate-out in down stream piping, the main condenser, or the turbine building.

4.3.1.2 MSLB Accident and Assumptions

The key inputs and assumptions used in the AST MSLB analysis are included in Table 6. The postulated MSLB accident assumes a double-ended break of one main steam line outside the primary containment with displacement of the pipe ends that permits maximum blowdown rates. Two activity release cases corresponding to the pre-accident spike and maximum equilibrium concentration allowed by Technical Specifications of 4.0 µCi/gm and 0.2 µCi/gm dose equivalent I-131, respectively were assumed, with inhalation CEDE dose conversion factors from Federal Guidance Report 11 (Reference 7.15) and external EDE dose conversion factors from Federal Guidance Report 12 (Reference 7.16). The released activity assumptions are consistent with the guidance provided in Appendix D of Regulatory Guide 1.183, as indicated in Table 6 below.

The analysis assumes an instantaneous ground level release. The released reactor coolant and steam is assumed to expand to a hemispheric volume at atmospheric pressure and temperature (consistent with an assumption of no Turbine Building holdup credit). This hemisphere is then assumed to move at a speed of 1 meter per second downwind past the control room intake. No credit is taken for buoyant rise of the steam cloud or for decay, and dispersion of the activity of the plume was conservatively ignored. For offsite locations, the buoyant rise of the steam cloud is similarly ignored, and the ground level dispersion is based on the conservative and simplified Regulatory Guide 1.5 (Reference 7.14) methodology.

The radiological consequences following an MSLB accident were determined utilizing Regulatory Guide 1.183 guidance. The following assumptions were made:

- There is instantaneous release from the break to the environment. No holdup in the Turbine Building or dilution by mixing with Turbine Building air volume is credited.
- The activity in the steam cloud is based on the total mass of water released from the break, not just the portion that flashes to steam. This assumption is conservative because it considers the maximum release of fission products.

- The fraction of liquid water contained in steam, which carries activity into the cloud is conservatively assumed to be 2.0%.
- The flashing fraction of liquid water released is 40%. However, all activity in the water is conservatively assumed to be released.
- No credit for control room operator action or filtration of the control room intake for the duration of the event is taken.

4.3.1.3 FHA Analysis Inputs and Assumptions

The key inputs and assumptions used in the AST FHA analysis are included in Table 7. The design basis FHA involves the drop of an assembly over the reactor core to maximize the fuel damage potential because of fall height.

The postulated FHA involves the drop of a fuel assembly on top of the reactor core during refueling operations. The analysis assumes that 172 GE-14 fuel rods in the full core are damaged. A radial peaking factor of 1.7 was assumed in the analysis in addition to the source term corrections discussed in Section 4.2.1. A post-shutdown 24-hour decay period was used to determine the release activity inventory. This assumption is conservative when compared to actual plant refueling outage history. The analysis assumes that gap activity in the affected rods was released instantaneously into the water in the reactor well. The analysis assumes the fuel bundle is dropped into the vessel, but assumes a water depth of greater than 23 feet above the assemblies seated in the reactor pressure vessel. The decontamination provided by the 23 feet is determined from guidance in Regulatory Guide 1.183 and is consistent with the limits in the TS.

In accordance with Regulatory Guide 1.183, the analysis assumes that the activity in the reactor building environment is released within two hours, from the reactor building through the reactor building vent stack, as a zero velocity vent release with no further credit for reactor building holdup or dilution, or SGT System operation.

TS Bases 3.9.6 assumes that a fuel assembly is dropped onto the reactor vessel flange. However, this is not an evaluated design basis accident. The bounding fuel handling accident is one in which a fuel assembly is dropped from the highest position onto the core. This produces the maximum kinetic energy, which results in the maximum damage. The drop of a fuel assembly onto the flange will not generate the kinetic energy to damage as many fuel rods as a drop into the vessel.

Similarly, a drop of a fuel assembly onto fuel in the spent fuel pool will also be bounded by a drop into the vessel.

The analysis assumes that the MCREV System and control room isolation are not initiated.

4.3.1.4 CRDA Analysis Inputs and Assumptions

The key inputs and assumptions used in the AST CRDA analysis are included in Table 8. The design basis CRDA involves the rapid removal of the highest worth control rod resulting in a reactivity excursion that encompasses the consequences of any other postulated CRDA. For the dose consequence analysis, it was assumed

that 1,200 of the fuel rods in the core were damaged, with melting occurring in 0.77 percent of the damaged rods (Reference 7.18). A conservative core average radial peaking factor of 1.7 was assumed in the analysis. For releases from the breached fuel, 10% of the core inventory of noble gases and iodines are assumed to be in the fuel gap. For releases attributed to fuel melting, 100% of the noble gases and 50% of the iodines are assumed to be released to the reactor coolant.

Instantaneous mixing of the activity released from the fuel in the reactor coolant is assumed, with 100% of the noble gases, 10% of the iodines and 1% of the remaining radionuclides that are released into the reactor coolant assumed to reach the turbine and condenser. 100% of the noble gases, 10% of the iodines and 1% of the particulate radionuclides reaching the turbine and condenser are available for release to the environment.

Unfiltered release from the turbine building is via the RB/TB Vent Stack at the rate of 1.0% of the condenser activity per day for 24 hours.

PBAPS retains the Main Steam Line Radiation Monitor (MSLRM) System's reactor trip and MSIV isolation functions. Additionally, the MSLRM also provides a signal to trip the mechanical vacuum pump and isolate its suction, if running. Therefore, there are no forced flow release paths.

The analysis takes no credit for control room operator action or filtration of the control room intake air for the duration of the event. In addition, both the 20,600 cfm normal intake flow and the assumed 1,600 cfm allowance for unfiltered inleakage are treated as intake sources.

The analysis assumptions for the transport, reduction, and release of the radioactive material from the fuel and the reactor coolant are consistent with the guidance provided in Appendix C of Regulatory Guide 1.183, as indicated in Table 8 below.

4.4 RESULTS

4.4.1 Evaluation Results

4.4.1.1.Accident Radiological Consequence Analyses

The postulated accident radiological consequence analyses were reviewed and updated for AST implementation impact and determined not to exceed regulatory limits.

4.4.1.2 LOCA Radiological Consequence Analyses

The radiological consequences of the DBA LOCA were analyzed with the RADTRAD code, using the inputs and assumptions discussed in Section 4.3.1.1.

The postulated sources of activity in the control room include contributions from filtered intake, and unfiltered inleakage. Dose contributors include internal cloud immersion and inhalation, and gamma shine from sources outside the Control Room.

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

The post-accident doses are the result of four distinct activity releases as discussed below.

Primary Containment to Secondary Containment Leakage

The leakage, captured by the secondary containment (reactor building), is exhausted as a zero velocity vent release as analyzed during the 15-minute drawdown period. After this period, this activity is collected by the SGTS, and then released to the environment through the main stack as an elevated release without filter credit. No other unfiltered exhaust from the reactor building is considered after the initial secondary containment drawdown period.

The primary leakage, secondary containment bypass pathway considers piping systems from primary containment to points outside of secondary containment and then to the environment. Except for MSIV leakage, no secondary bypass leakage pathways have been identified for PBAPS.

MSIV Leakage from the Primary Containment into the Environment

The MSIV leakage is released as an unfiltered, zero-velocity vent release at the RB/TB vent stack. No credit for upward velocity or buoyancy is considered.

ECCS Leakage to Secondary Containment

This leakage is assumed to start immediately after the onset of a LOCA and continue for 30 days. A flashing fraction of 1.41% is used. The flashed activity is collected by the SGTS prior to release to the environment except during the 15-minute drawdown period.

Dose Assessment from Sources External to Control Room

The doses from the following external sources were evaluated.

- External cloud contribution from the primary containment, secondary containment bypass, engineered safety feature systems, and MSIV leakage releases. This term takes credit for control room structural shielding.

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

- A direct shine from the MCREV System and SGT System filters.

The LOCA control room dose corresponds to an assumed unfiltered inleakage rate of 1600 cfm. Table 9 presents the results of the LOCA radiological consequence

analysis. As indicated, the control room, EAB, and LPZ calculated doses are within the regulatory limits for implementation of AST.

4.4.1.3 MSLB Accident Radiological Consequence Analysis

The radiological consequences of the postulated MSLB are given in Table 10. As indicated, the control room, EAB, and LPZ calculated doses are within regulatory limits after AST implementation.

4.4.1.4 FHA Radiological Consequence Analysis

The radiological consequences of the postulated FHA are given in Table 11. As indicated, the control room, EAB, and LPZ calculated doses are within regulatory limits after AST implementation.

4.4.1.5 CRDA Radiological Consequence Analysis

The radiological consequences of the postulated CRDA are given in Table 12. As indicated, the control room, EAB, and LPZ calculated doses are within regulatory limits after AST implementation.

4.4.2 Atmospheric Dispersion Factors

Figure 1 illustrates the release and intake points for PBAPS. The χ/Q values for these release-intake combinations are summarized in Tables 13, 14a, and 14b.

Table 13 lists χ/Q values used for the control room dose assessments. The ground level release χ/Q values (i.e., LOCA-MSIV and FHA release) were calculated by the ARCON96 computer code. The elevated release χ/Q values (i.e., LOCA main stack release) were calculated using ARCON96, and also the PAVAN code in accordance with Regulatory Guide 1.145 methodology. The separate χ/Q results of each of these two models were then analyzed according to the methodology in DG-1111, and the appropriate controlling χ/Q values for the elevated release were determined and are given in Table 13. These results are based on site-specific hourly meteorological data in a five-year period of record.

Tables 14a and 14b list χ/Q values for the EAB and LPZ boundaries. These values, including an initial 30-minute fumigation period, were calculated using the PAVAN code and Regulatory Guide 1.145 guidance using the same five-year period of record of site hourly meteorological data.

4.4.3 Post-Accident Suppression Pool Water Chemistry Management

The re-evolution of elemental iodine from the suppression pool is strongly dependent on pool pH. The analysis assumed that the borated solution was injected within 24 hours of the onset of a DBA LOCA and mixed within the suppression pool. The modeling of the PBAPS containment cabling maximized the production of hydrochloric acid. The analysis demonstrated that the suppression pool pH remains above 7 for the 30-day LOCA duration. The final pH and other related parameters are presented in Table 15.

4.4.4 Evaluation Conclusions

As shown in Tables 9 through 12, the plant accident radiological consequence analyses demonstrate that the post-accident offsite and control room doses will be maintained within regulatory limits following AST implementation. Furthermore, it has determined that continued compliance with NUREG-0737, Item II.B.2, will be maintained.

4.5 PROPOSED CHANGE TO UFSAR

The Peach Bottom Units 2 and 3 licensing basis has always included reliance on containment overpressure for the residual heat removal (RHR) and the core spray (CS) pumps following a postulated design basis loss of coolant accident (LOCA). A station analysis is maintained that conservatively estimates the Minimum Containment Pressure Available (MCPA) following a postulated design basis LOCA. A brief discussion of this analysis is included in PBAPS UFSAR Section 5.2.4.3.2. Although the MCPA analysis is conservative, a change to the PBAPS Operating License was approved by the NRC (Reference 7.22) to define a Containment Overpressure License (COPL) less than the evaluated MCPA for the design basis LOCA. Both the MCPA for the design basis LOCA and the COPL are shown in PBAPS UFSAR Figure 5.2.16. The PBAPS OL permits credit for containment overpressure for any design basis event up to the MCPA for that event, but not greater than the COPL.

The MCPA analysis for the design basis LOCA includes consideration of containment leakage. The proposed license amendment request includes an increase in the PBAPS Technical Specification limits for general containment leakage from 0.5% per day to 0.7% per day (both at a test pressure of 49.1 psig), and for MSIV leakage from 11.5 scfh to 174 scfh (both at a test pressure of 25 psig). These proposed leak rates have been incorporated into a revision to the MCPA analysis. The result is a reduction in the MCPA and the margin between the MCPA and the current COPL. Exelon is therefore also requesting a change to the Operating License by a change to the COPL of UFSAR Figure 5.2.16 to reduce the COPL such that additional margin is maintained between the revised MCPA and the proposed COPL.

Exelon has included a revision to PBAPS UFSAR Section 5.2.4.3.2 and Figure 5.2.16 in this License Amendment Request. The revised Figure 5.2.16 shows the revised MCPA and the proposed new COPL. Text changes to PBAPS UFSAR Section 5.2.4.3.2 include revision to the discussion of containment leakage. The text now includes containment leakage assumptions from all leakage sources and identifies that these leakage values are consistent with the proposed changes to the PBAPS Technical Specifications.

The revised UFSAR text also identifies a change in methodology regarding how this containment leakage is addressed in the MCPA analysis. Previously, containment leakage was assumed to be constant at 0.5% per day throughout the event, even though the containment pressure was on the order of 7 psig and the Technical Specification limit for containment leakage was 0.5% per day at a containment pressure of 49.1 psig. This previous approach for containment leakage was ultra-

conservative and clearly non-physical. Although such containment leakage is expected to consist of very small cracks such that the flow through the cracks would be characterized as laminar flow, the revised MCPA analysis conservatively treats the containment leakage as turbulent leakage.

Although the dose assessment evaluation discussed in this LAR accommodates a MSIV leakage rate of 250 scfh at the test pressure of 25 psig, the proposed Technical Specification change includes an increase in MSIV leakage rate to 174 scfh at the test pressure of 25 psig to maintain adequate margin between MCPA, COPL, and the containment overpressure required (COPR) for the most limiting ECCS system and design basis LOCA.

4.6 SUMMARY

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

Implementation of the AST provides the basis for several changes to the licensing and design bases for PBAPS. The principal changes affect primary containment and MSIV allowable leakage; elimination of requirements for several systems during movement of irradiated fuel assemblies that have decayed at least 24 hours and during other core alterations; and elimination of requirements for testing SGT charcoal and HEPA filters.

In the dose consequence analyses for the control room occupants, the assumed unfiltered inleakage was increased to a value that would be expected to bound credible inleakage values. Further evaluation of the analyses performed in support of the AST implementation support the conclusion that exposures to onsite and offsite receptors would not result in doses exceeding the values specified in 10 CFR 50.67.

Figure 1: Layout of Intake and Release Points for PBAPS

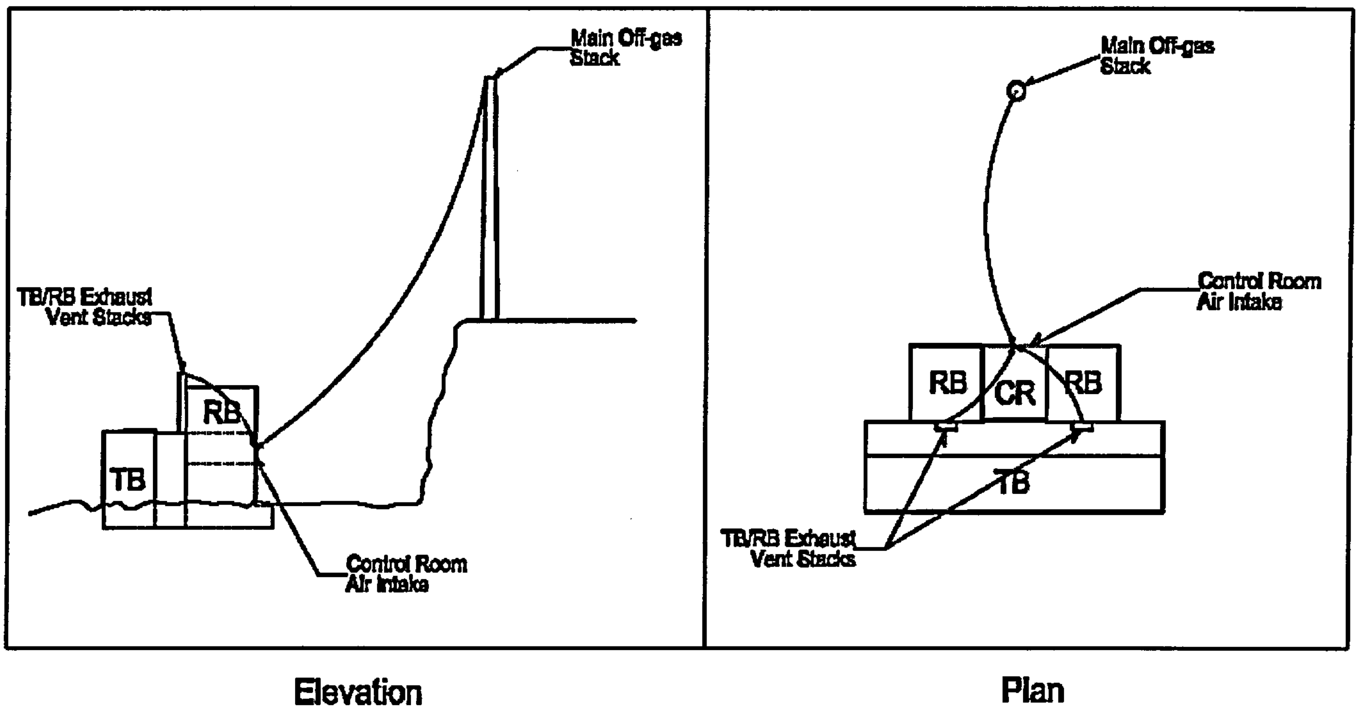

Figure 1 Parameters: Dimensional Data For Dispersion Analyses

Parameter	Value
Distance from Main Stack to Control Room Intake	685 feet
Direction, CR Intake to Main Stack	244 degrees
Distance from RB/TB Stacks to Control Room Intake	192 feet
Direction, CR Intake to Unit 2 RB/TB Stack	113 degrees
Direction, CR Intake to Unit 3 RB/TB Stack	15 degrees
Elevation at Plant Grade	116 feet above mean sea level (MSL)
Elevation at Center of Control Room Intake	185 feet above MSL
Elevation at Top of RB/TB Stacks	305 feet above MSL
Elevation at Base of Main Stack	280 feet above MSL
Elevation at Top of Main Stack	780 feet above MSL

Figure 2: LOCA Release Pathways

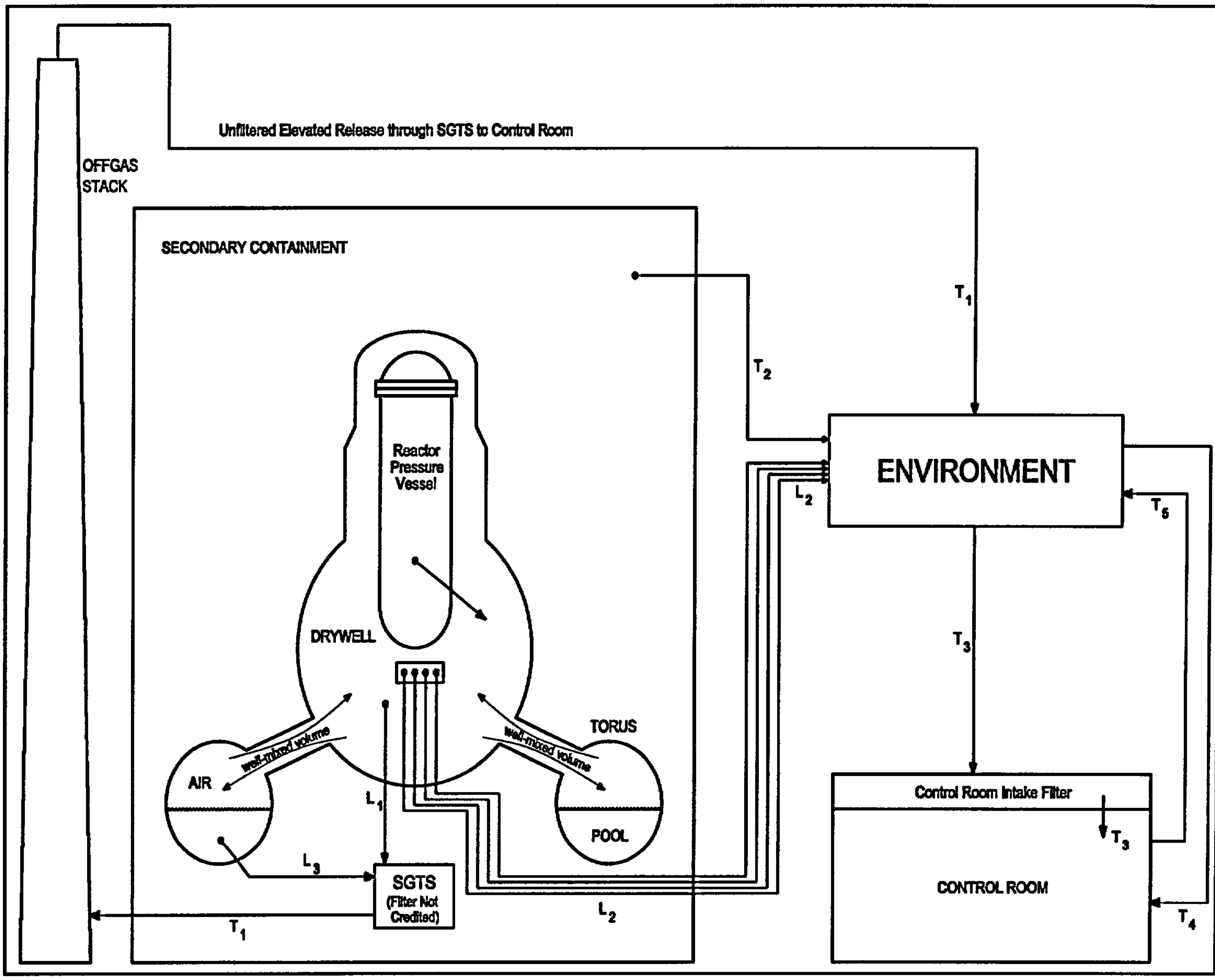

Figure 2 Parameters: LOCA Leakage Rates and Secondary Containment Mixing Parameters		
Path	**Description**	**Parameters & Values**
L_1	Primary Containment Leakage to Secondary Containment	Leak Rate: L_a = 0.7%/day, 0 – 15 min after start of gap release. Release is unfiltered through RB/TB Exhaust Vent Stacks during drawdown period. L_a = 0.7%/day, 15 min – 24 hr Release is unfiltered through Main Off-gas Stack. 0.56 x L_a = 0.392%/day, 24 - 38 hrs. Release is unfiltered through Main Off-gas Stack. 0.50 x L_a = 0.350%/day, 38 - 720 hrs. Release is unfiltered through Main Off-gas Stack.
L_2	MSIV Leakage to Environment	Leak Rate: 250 scfh for all main steam lines (for dose limit). 100 scfh for maximum for any one Main Steam line. Containment condition flows are derived, for example, by: 100 scfh *(14.7 psia/(25 psig+14.7psia))/60 min/hr = 0.62 cfm 12-hour delay transport credit. Leak Rate reduced to 77.2% at 24 hrs., 65.4% at 48 hrs., 59.0% at 72 hrs., 55.5% at 96 hrs., and 50% at 157 hrs. Release is unfiltered through RB/TB Exhaust Vent Stacks.
L_3	ECCS Leakage (Torus Water Source) to Secondary Containment Note: A flashing fraction of 1.41% is applied to the liquid release.	Leak Rate: 5 gpm, 0 – 15 min Release is unfiltered through RB/TB Exhaust Vent Stacks during drawdown period. 5 gpm, 15 min – 30 days Release is unfiltered through Main Off-gas Stack.
T_1 T_2	Release of Secondary Containment Atmosphere to the Environment through Main Stack Release of Secondary Containment Atmosphere to the Environment through the RB/TB Stack during drawdown.	No Secondary Containment mixing credit Volume (for analysis) = 1 cu. ft. Outflow (for analysis) = 100,000 cfm For first 15 minutes, leakage is directed through the RB/TB Exhaust Vent Stack directly to the Environment. Modeled as a zero velocity RB/TB vent release. After first 15 minutes, leakage is directed to the Main Off-gas Stack. Modeled as zero velocity RB/TB vent release. No credit is assumed for upward velocity or buoyancy effects.
T_3 T_4	Control Room Filtered Intake from the Environment Control Room Unfiltered Inleakage from the Environment	3,000 cfm filtered intake plus 1,600 cfm unfiltered inleakage, for a total of 4,600 cfm. MCREV filters credited at 99% for aerosols (based on HEPA) and 70% for charcoal adsorbers for elemental and organic iodine.
T_5	Control Room Exhaust to Environment	4,600 cfm to balance with intake

Table 1: Personnel Dose Inputs	
Input/Assumption	**Value**
Onsite Breathing Rate	3.47E-04 m^3/sec
Offsite Breathing Rate	0-8 hours: 3.47E-04 m^3/sec 8-24 hours: 1.75E-04 m^3/sec 1-30 days: 2.32E-04 m^3/sec
Control Room Occupancy Factors	0-1 day: 1.0 1-4 days: 0.6 4-30 days: 0.4

Table 2a: Key Analysis Inputs and Assumptions	
Release Inputs - LOCA Radionuclide Source Term	
Input/Assumption	**Value**
Core Fission Product Inventory	ORIGEN-2 Only the 60 nuclides considered by RADTRAD are utilized in the analysis
Core Power Level	3,528 MWt
Core Burnup	711 EFPD per 2-year cycle
Fission Product Release Fractions for LOCA	**RG 1.183, Table 1** **BWR Core Inventory Fraction Released Into Containment**
Fission Product Release Timing (Per RG 1.183, the release phases are modeled sequentially)	**RG 1.183, Table 4** **LOCA Release Phases** BWRs

BWR Core Inventory Fraction Released Into Containment

Group	Gap Release Phase	Early In-vessel Phase	Total
Noble Gases	0.05	0.95	1.0
Halogens	0.05	0.25	0.3
Alkali Metals	0.05	0.20	0.25
Tellurium Metals	0.00	0.05	0.05
Ba, Sr	0.00	0.02	0.02
Noble Metals	0.00	0.0025	0.0025
Cerium Group	0.00	0.0005	0.0005
Lanthanides	0.00	0.0002	0.0002

LOCA Release Phases (BWRs)

Phase	Onset	Duration
Gap Release	2 min	0.5 hr
Early In-Vessel	0.5 hr	1.5 hr

Table 2b: Key Analysis Inputs and Assumptions	
Release Inputs - Non-LOCA Radionuclide Source Term	
Input/Assumption	**Value**
Core Fission Product Inventory	ORIGEN-2 Only the 60 nuclides considered by RADTRAD are utilized in the analysis
Core Power Level	3,528 MWt
Core Burnup	711 EFPD (per 2-year cycle)
Fission Product Gap Release Fractions for Non-LOCA Accidents	**RG 1.183, Table 3** **Non-LOCA Fraction of Fission Product Inventory In Gap** **Group** — **Fraction** I-131 — 0.08 Kr-85 — 0.10 Other Noble Gases — 0.05 Other Halogens — 0.05 Alkali Metals — 0.12

Table 3: Key LOCA Analysis Inputs and Assumptions	
Release Inputs - Primary and Secondary Containment Parameters	
Input/Assumption	**Value**
Containment Free Volume	293,900 cubic feet
Minimum Suppression Pool Volume	122,900 cubic feet
Primary Containment Total Leak Rate	0.7% per day for first 24 hours (L_a) 0.392% per day 24 – 38 hours 0.35% per day after 38 hours
Total MSIV leak rate	250 scfh for all main steam lines (for dose limit). 100 scfh for maximum for any one Main Steam line. Containment condition flows are derived, for example, by: 100 scfh *(14.7 psia/(25 psig+14.7psia))/60 min/hr = 0.62 cfm 12-hour delay transport credit. Leak Rate reduced to 77.2% at 24 hrs., 65.4% at 48 hrs., 59.0% at 72 hrs., 55.5% at 96 hrs., and 50% at 157 hrs. Release is unfiltered through RB/TB Vent Stacks.
Fraction of Secondary Containment Available for Mixing	0
Maximum SGTS Flow Rate	10,500 cfm

Table 3: Key LOCA Analysis Inputs and Assumptions	
Release Inputs - Primary and Secondary Containment Parameters	
Input/Assumption	**Value**
Secondary Containment Drawdown Time	15 minutes
Secondary Containment Bypass	None
ECCS Systems Leak Rate Outside of Primary Containment (includes factor of 2)	5 gpm
ECCS Leakage Duration	0-30 days
<u>Release Pathways</u> ECCS/Containment Leakage MSIV Leakage	<u>Location</u> Main Stack (elevated release) RB/TB Vent Stacks (ground release)
<u>Release Pathways</u> ECCS/Containment Leakage MSIV Leakage	<u>Duration</u> 0-30 days 12 hours to 30 days

Table 4: Key LOCA Analysis Inputs and Assumptions	
Transport Inputs - Control Room Parameters	
Input/Assumption	**Value**
Nuclide Release Locations	See Figure 1
MCREV System Initiation	Control Room ventilation hi-radiation signal – effectively instantaneous
Control Room Free Volume	176,000 cubic feet
Control Room Filtered Air Intake Flow Rate Elemental and Organic Iodine Removal Efficiencies Aerosols Removal Efficiency	3,000 cfm 70% 99%
Control Room Unfiltered Inleakage Rate	1,600 cfm

Table 5: Key LOCA Analysis Inputs and Assumptions	
Removal Inputs	
Input/Assumption	Value
Aerosol DW Spray Removal Rates	Not Credited
Aerosol Natural Deposition Coefficients Used in the Drywell	Credit is taken for natural deposition of aerosols based on equations for the Power's model in NUREG/CR 6189 and input directly by RADTRAD as natural deposition time dependent lambdas. No credit is assumed for natural deposition of elemental or organic iodine, or for suppression pool scrubbing.
SGT System Filter Efficiencies – Elemental and Organic Iodine Aerosols	SGT System HEPA filters and charcoal adsorbers are not credited. The credited safety function is by the SGTS fans, which generate the secondary containment vacuum required to assure elevated release through the main stack.
Deposition/Plate-out (where credited)	Calculated for horizontal segments only using the RADTRAD Brockmann-Bixler model.
Main Steam Line and Condenser Holdup Holdup Credit for MSIV Leakage	Based on the maximum single steam line leakage rate and the volume in the shortest steam line from the reactor vessel nozzle to the turbine stop valves, a 12-hour transport delay is calculated. No credit is taken for holdup or plate-out downstream of the turbine stop valves or in the condenser since these components have not been evaluated for seismic ruggedness.

Table 6: Key MSLB Accident Analysis Inputs and Assumptions	
Input/Assumption	Value
Break Discharge Mass Release	190,920 pounds (25,800 as steam and 165,120 as liquid)
Pre-Accident Spike Iodine Concentration	4.0 µCi/gm I-131 equivalent
Maximum Equilibrium Iodine Concentration	0.2 µCi/gm I-131 equivalent
Transport Model for Control Room	Steam cloud moves past the Control Room intake at 1 m/sec
Turbine Building Holdup/Control Room Filtration	Not Credited

Table 7: Key FHA Analysis Inputs and Assumptions

Input/Assumption	Value
Core Damage	172 fuel rods failed based on GE14 fuel and "Heavy Mast"
Radial Peaking Factor	1.7
Fuel Decay Period	24 hours
Iodine Decontamination Factor	DF = 200
Release Period	2 hours
Refuel Floor Air Removal Rate	6 air changes per hour to assure activity exhaust within 2 hours
Control Room Filtration	Not Credited
Control Room Intake Flow	20,600 cfm Normal Intake plus 1,600 cfm unfiltered inleakage
Release Location	RB/TB vent stacks unfiltered, zero velocity vent release (Ground Level equivalent)
MCREV/SGT System Initiation	Not Credited

Table 8: Key CRDA Analysis Inputs and Assumptions	
Input/Assumption	Value
Core Damage	1,200 fuel rods failed (66,720 fuel rods in core)
Percent of Damaged Fuel with Melt	0.77%
Radial Peaking Factor	1.7
Mechanical Vacuum Pump Operation Before Isolation	MVP isolation by MSLRM Hi Rad signal
Condenser Leak Rate	1% of condenser activity per day throughout entire release period
Release Period	24 hours
Forced Flow Paths	None. Main Steam Line Radiation Monitor high radiation causes reactor trip, Main Steam Isolation Valve closure, and Mechanical Vacuum Pump trip and isolation, if required.
Release Location	See Figure 1 Zero-velocity vent release through the RB/TB vent stacks
MCREV System Initiation	Not Credited
Charcoal Delay Bed Noble Gas Delay for SJAE pathway	Not applicable. MSIV closure prevents release from this pathway

Table 9: LOCA Radiological Consequence Analysis			
Location	**Duration**	**TEDE (rem)**	**Regulatory Limit TEDE (rem)**
Control Room	30 days	4.1*	5
EAB	Maximum, 2 hours	4.8	25
LPZ	30 days	4.3	25

* The doses here include the direct shine and inhalation doses from radioactivity drawn into the control room as well as the dose from external sources. Dose is based on an assumed MSIV total leakage of 250 scfh. Actual leakage is limited by TS to 174 scfh.

Table 10: MSLB Accident Radiological Consequence Analysis				
Location	**Duration**	**4.0 µCi/gm Dose Equivalent I-131 TEDE (rem)**	**0.2 µCi/gm Dose Equivalent I-131 TEDE (rem)**	**Regulatory Limit TEDE (rem)**
Control Room	30-day integrated dose	3.3	0.17	5
EAB	Worst 2-hour integrated dose	1.6	0.080	25 (4.0 µCi/gm) 2.5 (0.2 µCi/gm)
LPZ	30-day integrated dose	0.28	0.014	25 (4.0 µCi/gm) 2.5 (0.2 µCi/gm)

Table 11: FHA Radiological Consequence Analysis			
Location	**Duration**	**TEDE (rem)**	**Regulatory Limit TEDE (rem)**
Control Room	30 days	2.4	5
EAB	Maximum, 2 hours	1.2	6.3
LPZ	30 days	0.14	6.3

Table 12: CRDA Radiological Consequence Analysis			
Location	**Duration**	**TEDE (rem)**	**Regulatory Limit TEDE (rem)**
Control Room	30 days	0.30	5
EAB	Maximum, 2 hours	0.065	6.3
LPZ	30 days	0.012	6.3

Table 13: Control Room χ/Q Values for the Different Release and Intake Combinations (Notes 1, 2)				
Time Period	χ/Q (sec/m^3)			
	LOCA Main Stack	LOCA RB/TB Vent Stacks (Note 4)	FHA (Note 4)	CRDA (Note 4)
0 - 2 hrs	2.72E-06	1.18E-03	1.18E-03	1.18E-03
2 - 8 hrs	1.00E-09 (Note 3)	9.08E-04	-	9.08E-04
8 - 24 hrs	1.00E-09 (Note 3)	4.14E-04	-	4.14E-04
1 - 4 days	1.46E-08	2.90E-04	-	-
4-30 days	4.21E-09	2.26E-04	-	-

Notes:

1. Main stack elevated release χ/Q values are based on Regulatory Guide 1.145 methodology; zero-velocity RB/TB vent stacks χ/Q values are based on ARCON 96.
2. Control room intake χ/Q values are applicable for control room inleakage.
3. The 2-8 hour and 8-24 hour period χ/Q's were conservatively increased from the value of 1.00E-15, to an artificial floor of 1.00E-09.
4. Bounding Unit 2 values were utilized.

Table 14a: Main Stack Elevated Release χ/Q (sec/m^3) Values Using RG 1.145 Methodology for the EAB and LPZ		
Time Period	EAB χ/Q (sec/m^3)	LPZ χ/Q (sec/m^3)
0 - 0.5 hrs	5.30E-05	1.75E-05
0 - 2 hrs	8.89E-06	8.87E-06
0 - 8 hrs	-	3.94E-06
8 - 24 hrs	-	2.62E-06
1 - 4 days	-	1.09E-06
4 - 30 days	-	3.06E-07

Table 14b: Zero-velocity RB/TB Vent Stack Release χ/Q (sec/m^3) Values Using RG 1.145 Methodology for the EAB and LPZ		
Time Period	EAB χ/Q (sec/m^3)	LPZ χ/Q (sec/m^3)
0 - 2 hrs	4.25E-04	4.81E-05
0 - 8 hrs	-	2.08E-05
8 - 24 hrs	-	1.37E-05
1 - 4 days	-	5.49E-06
4 - 30 days	-	1.49E-06

Table 15: Suppression Pool pH Results (Based on TS Minimum of value of 162.7 lb_m B-10 with a bounding enrichment assumption of 65% B-10)	
Condition	**Value**
Initial Suppression Pool pH	5.3
SLC injection time	Required Sodium Pentaborate to be injected within 24 hours
Suppression Pool pH throughout the 30-day accident duration	Greater than 7

REGULATORY GUIDE 1.183 COMPARISON

Table A: Conformance with Regulatory Guide (RG) 1.183 Main Sections			
RG Section	RG Position	PBAPS Analysis	Comments
3.1	The inventory of fission products in the reactor core and available for release to the containment should be based on the maximum full power operation of the core with, as a minimum, current licensed values for fuel enrichment, fuel burnup, and an assumed core power equal to the current licensed rated thermal power times the ECCS evaluation uncertainty. The period of irradiation should be of sufficient duration to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values. The core inventory should be determined using an appropriate isotope generation and depletion computer code such as ORIGEN 2 or ORIGEN-ARP. Core inventory factors (Ci/MWt) provided in TID 14844 and used in some analysis computer codes were derived for low burnup, low enrichment fuel and should not be used with higher burnup and higher enrichment fuels.	Conforms	ORIGEN 2.1 based methodology was used to determine core inventory. Power level used was 3514.9 MWt which is approximately the current licensed reactor thermal power or 3514 MWt. These source terms were evaluated at end-of-cycle and at beginning of cycle (100 effective full power days (EFPD) to achieve equilibrium) conditions and worst case inventory used for the selected isotopes. These values were then divided by 3514.9 MWt to obtain activity in units of Ci/MWt. Accident analyses are based on a 3528 MWt power level that is the current accident analysis design basis allowance for instrument uncertainty. Source terms are based on a 2 year fuel cycle with a nominal 711 EFPD per cycle.
3.1	For the DBA LOCA, all fuel assemblies in the core are assumed to be affected and the core average inventory should be used. For DBA events that do not involve the entire core, the fission product inventory of each of the damaged fuel rods is determined by dividing the total core inventory by the number of fuel rods in the core. To account for differences in power level across the core, radial peaking factors from the facility's core operating limits report (COLR) or technical specifications should be applied in determining the inventory of the damaged rods.	Conforms	Peaking factors of 1.7 are used for DBA events that do not involve the entire core, with fission product inventories for damaged fuel rods determined by dividing the total core inventory by the number of fuel rods in the core.
3.1	No adjustment to the fission product inventory should be made for events postulated to occur during power operations at less than full rated power or those postulated to occur at the beginning of core life. For events postulated to occur while the facility is shutdown, e.g., a fuel handling accident, radioactive decay from the time of shutdown may be modeled.	Conforms	No adjustments for less than full power are made in any analyses.

<table>
<tr><th colspan="4">Table A: Conformance with Regulatory Guide (RG) 1.183 Main Sections</th></tr>
<tr><th>RG Section</th><th>RG Position</th><th>PBAPS Analysis</th><th>Comments</th></tr>
<tr>
<td>3.2</td>
<td>The core inventory release fractions, by radionuclide groups, for the gap release and early in-vessel damage phases for DBA LOCAs are listed in Table 1 for BWRs and Table 2 for PWRs. These fractions are applied to the equilibrium core inventory described in Regulatory Position 3.1.

Table 1
BWR Core Inventory Fraction Released Into Containment
<table>
<tr><th>Group</th><th>Gap Release Phase</th><th>Early In-Vessel Phase</th><th>Total</th></tr>
<tr><td>Noble Gases</td><td>0.05</td><td>0.95</td><td>1.0</td></tr>
<tr><td>Halogens</td><td>0.05</td><td>0.25</td><td>0.3</td></tr>
<tr><td>Alkali Metals</td><td>0.05</td><td>0.20</td><td>0.25</td></tr>
<tr><td>Tellurium Metals</td><td>0.00</td><td>0.05</td><td>0.05</td></tr>
<tr><td>Ba, Sr</td><td>0.00</td><td>0.02</td><td>0.02</td></tr>
<tr><td>Noble Metals</td><td>0.00</td><td>0.0025</td><td>0.0025</td></tr>
<tr><td>Cerium Group</td><td>0.00</td><td>0.0005</td><td>0.0005</td></tr>
<tr><td>Lanthanides</td><td>0.00</td><td>0.0002</td><td>0.0002</td></tr>
</table>
Footnote 10:
The release fractions listed here have been determined to be acceptable for use with currently approved LWR fuel with a peak rod burnup up to 62,000 MWD/MTU. The data in this section may not be applicable to cores containing mixed oxide (MOX) fuel.</td>
<td>Conforms</td>
<td>The fractions from Regulatory Position 3.1, Table 1 are used.

Footnote 10 criteria are met.</td>
</tr>
</table>

Table A: Conformance with Regulatory Guide (RG) 1.183 Main Sections

RG Section	RG Position	PBAPS Analysis	Comments
3.2	For non-LOCA events, the fractions of the core inventory assumed to be in the gap for the various radionuclides are given in Table 3. The release fractions from Table 3 are used in conjunction with the fission product inventory calculated with the maximum core radial peaking factor. **Table 3** **Non-LOCA Fraction of Fission Product Inventory in Gap** **Group** — **Fraction** I-131 — 0.08 Kr-85 — 0.10 Other Noble Gases — 0.05 Other Halogens — 0.05 Alkali Metals — 0.12 Footnote 11: *The release fractions listed here have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup up to 62,000 MWD/MTU provided that the maximum linear heat generation rate does not exceed 6.3 kw/ft peak rod average power for rods with burnups that exceed 54 GWD/MTU. As an alternative, fission gas release calculations performed using NRC-approved methodologies may be considered on a case-by-case basis. To be acceptable, these calculations must use a projected power history that will bound the limiting projected plant-specific power history for the specific fuel load. For the BWR rod drop accident and the PWR rod ejection accident, the gap fractions are assumed to be 10% for iodines and noble gases.*	Conforms	Complies with Note 11 of Table 3. Peaking factor of 1.7 used for DBA events that do not involve the entire core.
3.3	Table 4 tabulates the onset and duration of each sequential release phase for DBA LOCAs at PWRs and BWRs. The specified onset is the time following the initiation of the accident (i.e., time = 0). The early in-vessel phase immediately follows the gap release phase. The activity released from the core during each release phase should be modeled as increasing in a linear fashion over the duration of the phase. For non-LOCA DBAs, in which fuel damage	Conforms	The BWR durations from Table 4 are used. LOCA is modeled in a linear fashion. Non-LOCA is modeled as an instantaneous release.

Table A: Conformance with Regulatory Guide (RG) 1.183 Main Sections

RG Section	RG Position	PBAPS Analysis	Comments
	is projected, the release from the fuel gap and the fuel pellet should be assumed to occur instantaneously with the onset of the projected damage. **Table 4** **LOCA Release Phases** **PWRs / BWRs** **Phase — Onset — Duration — Onset — Duration** Gap Release — 30 sec — 0.5 hr — 2 min — 0.5 hr Early In-Vessel — 0.5 hr — 1.3 hr — 0.5 hr — 1.5 hr		
3.3	For facilities licensed with leak-before-break methodology, the onset of the gap release phase may be assumed to be 10 minutes. A licensee may propose an alternative time for the onset of the gap release phase, based on facility-specific calculations using suitable analysis codes or on an accepted topical report shown to be applicable for the specific facility. In the absence of approved alternatives, the gap release phase onsets in Table 4 should be used.	Not Applicable	PBAPS does not use leak-before-break methodology for DBA analyses.
3.4	Table 5 lists the elements in each radionuclide group that should be considered in design basis analyses. **Table 5** **Radionuclide Groups** **Group — Elements** Noble Gases — Xe, Kr Halogens — I, Br Alkali Metals — Cs, Rb Tellurium Group — Te, Sb, Se, Ba, Sr Noble Metals — Ru, Rh, Pd, Mo, Tc, Co Lanthanides — La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am Cerium — Ce, Pu, Np	Conforms	The nuclides used are the 60 identified as being potentially important dose contributors to total effective dose equivalent (TEDE) in the RADTRAD code, which encompasses those listed in RG 1.183, Table 5.
3.5	Of the radioiodine released from the reactor coolant system (RCS) to the containment in a postulated accident, 95 percent of the iodine released should be assumed to be cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. This	Conforms	This guidance is applied in the analyses.

Table A: Conformance with Regulatory Guide (RG) 1.183 Main Sections			
RG Section	RG Position	PBAPS Analysis	Comments
	includes releases from the gap and the fuel pellets. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form. The same chemical form is assumed in releases from fuel pins in FHAs and from releases from the fuel pins through the RCS in DBAs other than FHAs or LOCAs. However, the transport of these iodine species following release from the fuel may affect these assumed fractions. The accident-specific appendices to this regulatory guide provide additional details.		
3.6	The amount of fuel damage caused by non-LOCA design basis events should be analyzed to determine, for the case resulting in the highest radioactivity release, the fraction of the fuel that reaches or exceeds the initiation temperature of fuel melt and the fraction of fuel elements for which the fuel clad is breached. Although the NRC staff has traditionally relied upon the departure from nucleate boiling ratio (DNBR) as a fuel damage criterion, licensees may propose other methods to the NRC staff, such as those based upon enthalpy deposition, for estimating fuel damage for the purpose of establishing radioactivity releases.	Conforms	Fuel damage assessment for CRDA and FHA are based on GESTAR standard analyses for GE14 fuel.
4.1.1	The dose calculations should determine the TEDE. TEDE is the sum of the committed effective dose equivalent (CEDE) from inhalation and the deep dose equivalent (DDE) from external exposure. The calculation of these two components of the TEDE should consider all radionuclides, including progeny from the decay of parent radionuclides that are significant with regard to dose consequences and the released radioactivity.	Conforms	TEDE is calculated, with significant progeny included.
4.1.2	The exposure-to-CEDE factors for inhalation of radioactive material should be derived from the data provided in ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers" (Ref. 19). Table 2.1 of Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion" (Ref. 20), provides tables of conversion factors acceptable to the NRC staff.	Conforms	Federal Guidance Report 11 dose conversion factors (DCFs) are used.

Table A: Conformance with Regulatory Guide (RG) 1.183 Main Sections

RG Section	RG Position	PBAPS Analysis	Comments
	The factors in the column headed "effective" yield doses corresponding to the CEDE.		
4.1.3	For the first 8 hours, the breathing rate of persons offsite should be assumed to be 3.5×10^{-4} cubic meters per second. From 8 to 24 hours following the accident, the breathing rate should be assumed to be 1.8×10^{-4} cubic meters per second. After that and until the end of the accident, the rate should be assumed to be 2.3×10^{-4} cubic meters per second.	Conforms	The analysis uses values to three significant figures that correspond to the rounded values in Section 4.1.3 of RG 1.183.
4.1.4	The DDE should be calculated assuming submergence in semi-infinite cloud assumptions with appropriate credit for attenuation by body tissue. The DDE is nominally equivalent to the effective dose equivalent (EDE) from external exposure if the whole body is irradiated uniformly. Since this is a reasonable assumption for submergence exposure situations, EDE may be used in lieu of DDE in determining the contribution of external dose to the TEDE. Table III.1 of Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water, and Soil" (Ref. 21), provides external EDE conversion factors acceptable to the NRC staff. The factors in the column headed "effective" yield doses corresponding to the EDE.	Conforms	Federal Guidance Report 12 conversion factors are used.
4.1.5	The TEDE should be determined for the most limiting person at the EAB. The maximum EAB TEDE for any two-hour period following the start of the radioactivity release should be determined and used in determining compliance with the dose criteria in 10 CFR 50.67. The maximum two-hour TEDE should be determined by calculating the postulated dose for a series of small time increments and performing a "sliding" sum over the increments for successive two-hour periods. The maximum TEDE obtained is submitted. The time increments should appropriately reflect the progression of the accident to capture the peak dose interval between the start of the event and the end of radioactivity release (see also Table 6). Footnote 14: *With regard to the EAB TEDE, the maximum two-hour value is the*	Conforms	The maximum two-hour LOCA EAB dose is as follows: PC Leakage: 0.0 to 2.0 hours (3.50 Rem TEDE) due to the 15-minute unfiltered, ground-level SC drawdown time. MSIV Leakage: 12.0 to 14.0 hours (1.50 Rem TEDE) due to the 12-hour delay in transport. ECCS Leakage: 2.0 to 4.0 hours (0.043 Rem TEDE) Conservatively, the maximum 2-hour period dose was determined by adding the maximum 2-hour dose for each of the components listed

Table A: Conformance with Regulatory Guide (RG) 1.183 Main Sections

RG Section	RG Position	PBAPS Analysis	Comments
	basis for screening and evaluation under 10 CFR 50.59. Changes to doses outside of the two-hour window are only considered in the context of their impact on the maximum two-hour EAB TEDE.		above even though they do not occur simultaneously. This yields: 3.50 + 1.50 + 0.043 = 5.043 Rem TEDE (Rounded up to 5.1 Rem TEDE).
4.1.6	TEDE should be determined for the most limiting receptor at the outer boundary of the low population zone (LPZ) and should be used in determining compliance with the dose criteria in 10 CFR 50.67.	Conforms	This guidance is applied in the analyses.
4.1.7	No correction should be made for depletion of the effluent plume by deposition on the ground.	Conforms	No such corrections made in the analyses.
4.2.1	The TEDE analysis should consider all sources of radiation that will cause exposure to control room personnel. The applicable sources will vary from facility to facility, but typically will include: Contamination of the control room atmosphere by the intake or infiltration of the radioactive material contained in the radioactive plume released from the facility, Contamination of the control room atmosphere by the intake or infiltration of airborne radioactive material from areas and structures adjacent to the control room envelope, Radiation shine from the external radioactive plume released from the facility, Radiation shine from radioactive material in the reactor containment, Radiation shine from radioactive material in systems and components inside or external to the control room envelope, e.g., radioactive material buildup in recirculation filters.	Conforms	The principal source of dose within the control room is due to airborne activity. The dose estimates from post LOCA primary containment and external cloud sources external to the control room were based on TID-14844 source terms and the small contributor is considered to continue to be conservatively applicable. Calculations of doses from reactor building airborne activity have been recalculated with AST source term assumptions, the activity well mixed in the reactor building free air volume, no credit for contained structures except floors, and with a relatively detailed geometry treatment. In the worst case, external dose component represents only 1% of the control room dose limit. SGTS and MCREV filters are well away and/or shielded from the Control Room and have not historically been considered a source for operator doses. AST assessments would reduce filter loadings because of the credited natural deposition in containment. Therefore, the historical conclusion continues to apply.

Table A: Conformance with Regulatory Guide (RG) 1.183 Main Sections

RG Section	RG Position	PBAPS Analysis	Comments
4.2.2	The radioactive material releases and radiation levels used in the control room dose analysis should be determined using the same source term, transport, and release assumptions used for determining the EAB and the LPZ TEDE values, unless these assumptions would result in non-conservative results for the control room.	Conforms	The source term, transport, and release methodology is the same for both the control room and offsite locations.
4.2.3	The models used to transport radioactive material into and through the control room, and the shielding models used to determine radiation dose rates from external sources, should be structured to provide suitably conservative estimates of the exposure to control room personnel.	Conforms	This guidance is applied in the analyses.
4.2.4	Credit for engineered safety features that mitigate airborne radioactive material within the control room may be assumed. Such features may include control room isolation or pressurization, or intake or recirculation filtration. Refer to Section 6.5.1, "ESF Atmospheric Cleanup System," of the SRP (Ref. 3) and Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Post-accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants" (Ref. 25), for guidance.	Conforms	Control Room pressurization and intake filtration are credited in the LOCA accident analysis. No credit is taken in the FHA, MSLB and CRDA accident analyses. No credit is taken for SGTS HEPA or charcoal adsorber filtration in any accident.
4.2.5	Credit should generally not be taken for the use of personal protective equipment or prophylactic drugs. Deviations may be considered on a case-by-case basis.	Conforms	Such credits are not taken.
4.2.6	The dose receptor for these analyses is the hypothetical maximum exposed individual who is present in the control room for 100% of the time during the first 24 hours after the event, 60% of the time between 1 and 4 days, and 40% of the time from 4 days to 30 days. For the duration of the event, the breathing rate of this individual should be assumed to be 3.5×10^{-4} cubic meters per second.	Conforms	The cited occupancy factors and breathing rate are used. An unrounded breathing rate of 3.47E-04 m^3/sec is used.
4.2.7	Control room doses should be calculated using dose conversion factors identified in Regulatory Position 4.1 above for use in offsite dose analyses. The DDE from photons may be corrected for the difference between finite cloud geometry in the control room and	Conforms	The equation given is utilized for finite cloud correction when calculating external doses due to the airborne activity inside the control room.

Table A: Conformance with Regulatory Guide (RG) 1.183 Main Sections

RG Section	RG Position	PBAPS Analysis	Comments
	the semi-infinite cloud assumption used in calculating the dose conversion factors. The following expression may be used to correct the semi-infinite cloud dose, DDE_∞, to a finite cloud dose, DDE_{finite}, where the control room is modeled as a hemisphere that has a volume, V, in cubic feet, equivalent to that of the control room (Ref. 22). $DDE_{finite} = \frac{DDE_\infty V^{0.338}}{1173}$		
4.3	The guidance provided in Regulatory Positions 4.1 and 4.2 should be used, as applicable, in re-assessing the radiological analyses identified in Regulatory Position 1.3.1, such as those in NUREG-0737 (Ref. 2). Design envelope source terms provided in NUREG-0737 should be updated for consistency with the AST. In general, radiation exposures to plant personnel identified in Regulatory Position 1.3.1 should be expressed in terms of TEDE. Integrated radiation exposure of plant equipment should be determined using the guidance of Appendix I of this guide.	Conforms	The Technical Support Center at PBAPS is in the Unit 1 Control Room. A review of the current TID-14844 based analysis indicates that it is unnecessary to reanalyze doses therein to assure accessibility. For other areas requiring plant personnel access, a qualitative assessment of the regulatory positions on source terms indicates that, with no new operator actions required, radiation exposures are bounded by those previously analyzed.
5.1.1	The evaluations required by 10 CFR 50.67 are re-analyses of the design basis safety analyses and evaluations required by 10 CFR 50.34; they are considered to be a significant input to the evaluations required by 10 CFR 50.92 or 10 CFR 50.59. These analyses should be prepared, reviewed, and maintained in accordance with quality assurance programs that comply with Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50.	Conforms	These analyses were prepared as specified in the guidance.
5.1.2	Credit may be taken for accident mitigation features that are classified as safety-related, are required to be operable by technical specifications, are powered by emergency power sources, and are either automatically actuated or, in limited cases, have actuation requirements explicitly addressed in emergency operating procedures. The single active component failure that results in the	Conforms	The analyses take credit for SLC System operation. The SLC System is safety-related, required to be operable by Technical Specifications, and supplied with emergency power . The SLC System is manually initiated from the main control room, as directed by the

Table A: Conformance with Regulatory Guide (RG) 1.183 Main Sections			
RG Section	RG Position	PBAPS Analysis	Comments
	most limiting radiological consequences should be assumed. Assumptions regarding the occurrence and timing of a loss of offsite power should be selected with the objective of maximizing the postulated radiological consequences.		emergency operating procedures. There are four proceduralized injection methods for SLC with at least one alternate method for SLC injection that does not require personnel access into the secondary containment.
5.1.3	The numeric values that are chosen as inputs to the analyses required by 10 CFR 50.67 should be selected with the objective of determining a conservative postulated dose. In some instances, a particular parameter may be conservative in one portion of an analysis but be non-conservative in another portion of the same analysis.	Conforms	Conservative assumptions are used.
5.1.4	Licensees should ensure that analysis assumptions and methods are compatible with the AST and the TEDE criteria.	Conforms	Analysis assumptions and methods were made per this guidance.
5.3	Atmospheric dispersion values (X/Q) for the EAB, the LPZ, and the control room that were approved by the staff during initial facility licensing or in subsequent licensing proceedings may be used in performing the radiological analyses identified by this guide. Methodologies that have been used for determining X/Q values are documented in Regulatory Guides 1.3 and 1.4, Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," and the paper, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19". The NRC computer code PAVAN implements Regulatory Guide 1.145 and its use is acceptable to the NRC staff. The methodology of the NRC computer code ARCON96 is generally acceptable to the NRC staff for use in determining control room X/Q values.	Conforms	New atmospheric dispersion values (X/Q) for the EAB, the LPZ, and the control room were developed, using meteorological data for the years 1984-1988. ARCON96 and PAVAN were used with these data to determine control room and EAB/LPZ atmospheric dispersion values. Control room X/Qs from releases from the Main Stack were developed in conformance with DG-1111.

Table B: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	PBAPS Analysis	Comments
1	Acceptable assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide.	Conforms	*Fission Product Inventory:* Core source terms are developed using ORIGEN-2.1 based methodology. *Release Fractions:* Release fractions are per Table 1 of RG 1.183, and are implemented by RADTRAD. *Timing of Release Phases:* Release Phases are per Table 4 of RG 1.183, and are implemented by RADTRAD. *Radionuclide Composition:* Radionuclide grouping is per Table 5 of RG 1.183, as implemented in RADTRAD. *Chemical Form:* Treatment of release chemical form is per RG 1.183, Section 3.5.
2	If the sump or suppression pool pH is controlled at values of 7 or greater, the chemical form of radioiodine released to the containment should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. Iodine species, including those from iodine re-evolution, for sump or suppression pool pH values less than 7 will be evaluated on a case-by-case basis. Evaluations of pH should consider the effect of acids and bases created during the LOCA event, e.g., radiolysis products. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form.	Conforms	The stated distributions of iodine chemical forms are used. The post-LOCA suppression pool pH has been evaluated, including consideration of the effects of acids and bases created during the LOCA event, the effects of key fission product releases, and the impact of SLC injection. Suppression pool pH remains above 7 for at least 30 days.
3.1	The radioactivity released from the fuel should be assumed to mix instantaneously and homogeneously throughout the free air volume of the primary containment in PWRs or the drywell in BWRs as it is released. This distribution should be adjusted if there are internal compartments that have limited ventilation exchange. The suppression pool free air volume may be included provided there is a mechanism to ensure mixing between the drywell to the wetwell. The release into the	Conforms	The radioactivity release from the fuel is assumed to instantaneously and homogeneously mix throughout the drywell and suppression chamber air space. The suppression chamber free air volume is included based on expected steam flow from the drywell

Table B: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	PBAPS Analysis	Comments
	containment or drywell should be assumed to terminate at the end of the early in-vessel phase.		to the suppression chamber, even after the initial blowdown.
3.2	Reduction in airborne radioactivity in the containment by natural deposition within the containment may be credited. Acceptable models for removal of iodine and aerosols are described in Chapter 6.5.2, "Containment Spray as a Fission Product Cleanup System," of the Standard Review Plan (SRP), NUREG-0800 (Ref. A-1) and in NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments" (Ref. A-2). The latter model is incorporated into the analysis code RADTRAD (Ref. A-3).	Conforms	Credit is taken for natural deposition per the methodology of NUREG/CR-6189, as implemented in RADTRAD. No deterministically assumed initial plateout is credited.
3.3	Reduction in airborne radioactivity in the containment by containment spray systems that have been designed and are maintained in accordance with Chapter 6.5.2 of the SRP (Ref. A-1) may be credited. Acceptable models for the removal of iodine and aerosols are described in Chapter 6.5.2 of the SRP and NUREG/CR-5966, "A Simplified Model of Aerosol Removal by Containment Sprays"1 (Ref. A-4). This simplified model is incorporated into the analysis code RADTRAD (Refs. A-1 to A-3). The evaluation of the containment sprays should address areas within the primary containment that are not covered by the spray drops. The mixing rate attributed to natural convection between sprayed and unsprayed regions of the containment building, provided that adequate flow exists between these regions, is assumed to be two turnovers of the unsprayed regions per hour, unless other rates are justified. The containment building atmosphere may be considered a single, well-mixed volume if the spray covers at least 90% of the volume and if adequate mixing of unsprayed compartments can be shown. The SRP sets forth a maximum decontamination factor (DF) for elemental iodine based on the maximum iodine activity in the primary containment atmosphere when the sprays actuate, divided by the activity of iodine remaining at some time after decontamination. The SRP also states that the particulate iodine removal rate should be reduced by a factor of 10 when a DF of 50 is reached. The reduction in the removal rate is not required if the removal rate is based on the	Not Applicable	While containment sprays are a design feature that is available at PBAPS, no credit is taken for aerosol removal by them in the LOCA AST reanalysis.

Table B: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	PBAPS Analysis	Comments
	calculated time-dependent airborne aerosol mass. There is no specified maximum DF for aerosol removal by sprays. The maximum activity to be used in determining the DF is defined as the iodine activity in the columns labeled "Total" in Tables 1 and 2 of this guide multiplied by 0.05 for elemental iodine and by 0.95 for particulate iodine (i.e., aerosol treated as particulate in SRP methodology).		
3.4	Reduction in airborne radioactivity in the containment by in-containment recirculation filter systems may be credited if these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. A-5 and A-6). The filter media loading caused by the increased aerosol release associated with the revised source term should be addressed.	Not Applicable	No in-containment recirculation filter systems exist at PBAPS.
3.5	Reduction in airborne radioactivity in the containment by suppression pool scrubbing in BWRs should generally not be credited. However, the staff may consider such reduction on an individual case basis. The evaluation should consider the relative timing of the blowdown and the fission product release from the fuel, the force driving the release through the pool, and the potential for any bypass of the suppression pool (Ref. 7). Analyses should consider iodine re-evolution if the suppression pool liquid pH is not maintained greater than 7.	Conforms	No credit is taken for suppression pool scrubbing in the LOCA AST reanalysis. Analyses have been performed that determined that the suppression pool liquid pH is maintained greater than 7, and that, therefore, iodine re-evolution is not expected.
3.6	Reduction in airborne radioactivity in the containment by retention in ice condensers, or other engineering safety features not addressed above, should be evaluated on an individual case basis. See Section 6.5.4 of the SRP (Ref. A-1).	Not Applicable	PBAPS does not have ice condensers. No other removal mechanisms are credited other than natural deposition.
3.7	The primary containment (i.e., drywell for Mark I and II containment designs) should be assumed to leak at the peak pressure technical specification leak rate for the first 24 hours. For PWRs, the leak rate may be reduced after the first 24 hours to 50% of the technical specification leak rate. For BWRs, leakage may be reduced after the first 24 hours, if supported by plant configuration and analyses, to a value not less than 50% of the technical specification leak rate. Leakage from subatmospheric containments is assumed to terminate when the containment is brought to and maintained at a subatmospheric condition as defined by technical specifications.	Conforms	Primary containment leakage is assumed to be at the 0.7% of containment mass per day for 24 hours, 0.392% from 24 to 38 hours, and 0.35% per day from 38 to 720 hours. This is based on the results of the leak characteristic methodology evaluation performed (turbulent flow). The Darcy's Formula evaluation methodology is considered the most

Table B: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	PBAPS Analysis	Comments
	For BWRs with Mark III containments, the leakage from the drywell into the primary containment should be based on the steaming rate of the heated reactor core, with no credit for core debris relocation. This leakage should be assumed during the two-hour period between the initial blowdown and termination of the fuel radioactivity release (gap and early in-vessel release phases). After two hours, the radioactivity is assumed to be uniformly distributed throughout the drywell and the primary containment.		conservative approach for the evaluation of the primary containment leak rate. The large break LOCA was found to be the bounding containment pressurization event. Even if a LOCA were to occur during purging, isolation valve closure would occur within a small fraction of the time before start of the gap release. Dose due to this purge would be negligible as compared to other dose contributors. PBAPS uses a Mark I containment.
3.8	If the primary containment is routinely purged during power operations, releases via the purge system prior to containment isolation should be analyzed and the resulting doses summed with the postulated doses from other release paths. The purge release evaluation should assume that 100% of the radionuclide inventory in the reactor coolant system liquid is released to the containment at the initiation of the LOCA. This inventory should be based on the technical specification reactor coolant system equilibrium activity. Iodine spikes need not be considered. If the purge system is not isolated before the onset of the gap release phase, the release fractions associated with the gap release and early in-vessel phases should be considered as applicable.	Conforms	The PBAPS primary containment is not routinely purged during power operation. Purging is limited to inerting, de-inerting and occasional short pressure control activities.
4.1	Leakage from the primary containment should be considered to be collected, processed by engineered safety feature (ESF) filters, if any, and released to the environment via the secondary containment exhaust system during periods in which the secondary containment has a negative pressure as defined in technical specifications. Credit for an elevated release should be assumed only if the point of physical release is more than two and one-half times the height of any adjacent structure.	Conforms	Secondary Containment elevated release via the Main Stack credit is taken at 15 minutes after the start of gap release. Gap release begins at ~ 2 minutes after LOCA initiation. For EAB and LPZ doses, ground level releases are assumed. For Control Room doses, releases are based on zero-velocity RB/TB vent stack release assumptions, yielding

Table B: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	PBAPS Analysis	Comments
			ground level release equivalent dispersion factors.
4.2	Leakage from the primary containment is assumed to be released directly to the environment as a ground-level release during any period in which the secondary containment does not have a negative pressure as defined in technical specifications.	Conforms	For EAB and LPZ doses, ground level releases are assumed. For Control Room doses, releases are based on zero-velocity RB/TB vent stack release assumptions.
4.3	The effect of high wind speeds on the ability of the secondary containment to maintain a negative pressure should be evaluated on an individual case basis. The wind speed to be assumed is the 1-hour average value that is exceeded only 5% of the total number of hours in the data set. Ambient temperatures used in these assessments should be the 1-hour average value that is exceeded only 5% or 95% of the total numbers of hours in the data set, whichever is conservative for the intended use (e.g., if high temperatures are limiting, use those exceeded only 5%).	Conforms	The wind speed exceeded only 5% of the time at PBAPS in the secondary containment vicinity is approximately 11 mph. It has been determined that a 23 mph wind speed would be required before the secondary containment pressures would be positive relative to outside air pressures at the downwind side of the reactor enclosure.
4.4	Credit for dilution in the secondary containment may be allowed when adequate means to cause mixing can be demonstrated. Otherwise, the leakage from the primary containment should be assumed to be transported directly to exhaust systems without mixing. Credit for mixing, if found to be appropriate, should generally be limited to 50%. This evaluation should consider the magnitude of the containment leakage in relation to contiguous building volume or exhaust rate, the location of exhaust plenums relative to projected release locations, the recirculation ventilation systems, and internal walls and floors that impede stream flow between the release and the exhaust.	Conforms	No credit is taken for dilution/mixing in secondary containment. An artificially small secondary containment volume is assumed in the RADTRAD analysis in conjunction with a large SGTS flow rate to ensure mixing is not an issue.

Table B: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)

RG Section	RG Position	PBAPS Analysis	Comments
4.5	Primary containment leakage that bypasses the secondary containment should be evaluated at the bypass leak rate incorporated in the technical specifications. If the bypass leakage is through water, e.g., via a filled piping run that is maintained full, credit for retention of iodine and aerosols may be considered on a case-by-case basis. Similarly, deposition of aerosol radioactivity in gas-filled lines may be considered on a case-by-case basis.	Conforms	No primary containment leakage, with the exception of MSIV leakage, has been identified which bypasses the secondary containment. Only the MSIV pathway leak rates are incorporated into the Technical Specifications. The AST analysis is based on an MSIV leakage limit of 250 scfh total leakage with not more that 100 scfh per line when tested at ≥ 25 psig. However, based on revised containment pressure analysis, the revised TS MSIV leakage is limited to 174 scfh. Refer to Section 4.5 for additional information. The dose consequences for releases through this pathway (with piping deposition credit) are separately calculated. Therefore, MSIV leakage can continue to be excluded from Type B and C leakage total evaluated against the revised L_a of 0.7% per day. Piping deposition credit is determined using the Brockmann-Bixler routines available in RADTRAD. Delay in transit through these piping system is also credited, based on volume divided by standard pressure flow rates. The resulting delay values are 12 hours based on the smallest volume steam line and highest allowable leakage flow rate.

Table B: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	PBAPS Analysis	Comments
			The credited piping is that which has previously been seismically qualified and is from the reactor vessel to the Turbine Stop Valves.
4.6	Reduction in the amount of radioactive material released from the secondary containment because of ESF filter systems may be taken into account provided that these systems meet the guidance of Regulatory Guide 1.52 (Ref. A-5) and Generic Letter 99-02 (Ref. A-6).	Conforms	SGTS HEPA and charcoal adsorber filters are not credited in the evaluation of analyzed accidents onsite and offsite dose consequences.
5.1	With the exception of noble gases, all the fission products released from the fuel to the containment (as defined in Tables 1 and 2 of this guide) should be assumed to instantaneously and homogeneously mix in the primary containment sump water (in PWRs) or suppression pool (in BWRs) at the time of release from the core. In lieu of this deterministic approach, suitably conservative mechanistic models for the transport of airborne activity in containment to the sump water may be used. Note that many of the parameters that make spray and deposition models conservative with regard to containment airborne leakage are non-conservative with regard to the buildup of sump activity.	Conforms	With the exception of noble gases, all the fission products released from the fuel to the containment are assumed to instantaneously and homogeneously mix in the suppression pool at the time of release from the core.
5.2	The leakage should be taken as two times the sum of the simultaneous leakage from all components in the ESF recirculation systems above which the technical specifications, or licensee commitments to item III.D.1.1 of NUREG-0737 (Ref. A-8), would require declaring such systems inoperable. The leakage should be assumed to start at the earliest time the recirculation flow occurs in these systems and end at the latest time the releases from these systems are terminated. Consideration should also be given to design leakage through valves isolating ESF recirculation systems from tanks vented to atmosphere, e.g., emergency core cooling system (ECCS) pump miniflow return to the refueling water storage tank.	Conforms	The 5 gpm leak rate is assumed to be two times the sum of the simultaneous leakage from all ECCS components as discussed in the dose calculations. ECCS leakage is minimized at PBAPS through implementation of the Program committed to in T.S. 5.5.2 "Primary Coolant Sources Outside Containment". Since certain ECCS systems take suction immediately from the suppression pool, this leak path is assumed to start at time 0. Leakage to atmospheric tanks is

Table B: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)

RG Section	RG Position	PBAPS Analysis	Comments
			credible only for lines connecting from ECCS pump discharges to such a tank, because of relative elevations. The sole leakage paths to a tank vented to atmosphere meeting this condition are the High Pressure Coolant Injection / Reactor Core Isolation Cooling test lines that discharge to the Condensate Storage Tank (CST). These lines are isolated by two normally closed valves. Since the CST contents are demineralized water, ECCS leakage would quickly turn the water basic. Therefore, minimal elemental iodine is expected, and as a result, negligible iodine volatilization.
5.3	With the exception of iodine, all radioactive materials in the recirculating liquid should be assumed to be retained in the liquid phase.	Conforms	With the exception of iodine, all radioactive materials in ECCS liquids are assumed to be retained in the liquid phase.
5.4	If the temperature of the leakage exceeds 212°F, the fraction of total iodine in the liquid that becomes airborne should be assumed equal to the fraction of the leakage that flashes to vapor. This flash fraction, FF, should be determined using a constant enthalpy, h, process, based on the maximum time-dependent temperature of the sump water circulating outside the containment: $FF = \frac{h_{f1} - h_{f2}}{h_{fg}}$ *Where:* h_{f1} is the enthalpy of liquid at system design temperature and pressure; h_{f2} is the enthalpy of liquid at saturation conditions (14.7 psia, 212°F); and h_{fg} is the heat of vaporization at 212°F.	Not Applicable	The temperature of the leakage does not exceed 212°F.
5.5	If the temperature of the leakage is less than 212°F or the calculated	Conforms	An airborne release fraction of 1.41%

Table B: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	PBAPS Analysis	Comments
	flash fraction is less than 10%, the amount of iodine that becomes airborne should be assumed to be 10% of the total iodine activity in the leaked fluid, unless a smaller amount can be justified based on the actual sump pH history and area ventilation rates.		is used. Suppression Pool water pH is maintained above 7 for the entire 30 days of the accident dose assessment period. Under these conditions virtually none of the iodine will be in elemental form, and organic iodine formation will be inhibited. Because of the subcooled condition no flashing is expected. Neverthe-less, this value, derived based on ORNL-TM-2412 methodology for iodine partition factor determination, is used.
5.6	The radioiodine that is postulated to be available for release to the environment is assumed to be 97% elemental and 3% organic. Reduction in release activity by dilution or holdup within buildings, or by ESF ventilation filtration systems, may be credited where applicable. Filter systems used in these applications should be evaluated against the guidance of Regulatory Guide 1.52 (Ref. A-5) and Generic Letter 99-02 (Ref. A-6).	Conforms	The credited Control Room intake charcoal and HEPA filters meet the requirements of RG 1.52 and Generic Letter 99-02. These are credited at 70% efficiency for elemental and organic iodines. Aerosol removal efficiencies are assumed to be 99% based on the HEPA/charcoal combination.
6.1	For the purpose of this analysis, the activity available for release via MSIV leakage should be assumed to be that activity determined to be in the drywell for evaluating containment leakage (see Regulatory Position 3). No credit should be assumed for activity reduction by the steam separators or by iodine partitioning in the reactor vessel.	Conforms	The radioactivity release from the fuel is assumed to instantaneously and homogeneously mix throughout the drywell and suppression chamber air space. The suppression chamber free air volume is included based on expected steam flow from the drywell to the suppression chamber, even after initial blowdown.
6.2	All the MSIVs should be assumed to leak at the maximum leak rate above which the technical specifications would require declaring the MSIVs inoperable. The leakage should be assumed to continue for the	Conforms	MSIV leakage assumed in this accident analysis is 250 scfh (TS limit is 174 scfh) for all steam lines

Table B: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)

RG Section	RG Position	PBAPS Analysis	Comments
	duration of the accident. Postulated leakage may be reduced after the first 24 hours, if supported by site-specific analyses, to a value not less than 50% of the maximum leak rate.		and 100 scfh for any one line when tested at ≥ 25 psig. Reduction in leakage rates after 24 hours are, as previously discussed, based on calculated post-accident containment pressures. No credit is taken for leakage rate reductions below 50% of the MSIV leakage limit.
6.3	Reduction of the amount of released radioactivity by deposition and plateout on steam system piping upstream of the outboard MSIVs may be credited, but the amount of reduction in concentration allowed will be evaluated on an individual case basis. Generally, the model should be based on the assumption of well-mixed volumes, but other models such as slug flow may be used if justified.	Conforms	Modeling is per RADTRAD Brockmann-Bixler approach, with a conservatively derived transport delay credit of 12 hours.
6.4	In the absence of collection and treatment of releases by ESFs such as the MSIV leakage control system, or as described in paragraph 6.5 below, the MSIV leakage should be assumed to be released to the environment as an unprocessed, ground- level release. Holdup and dilution in the turbine building should not be assumed.	Conforms	Releases are assumed to be from the RB/TB vent stacks, without credit for holdup or dilution in the condenser or turbine building. The zero velocity RB/TB vent stacks release assumption is effectively a ground level release assumption.
6.5	A reduction in MSIV releases that is due to holdup and deposition in main steam piping downstream of the MSIVs and in the main condenser, including the treatment of air ejector effluent by offgas systems, may be credited if the components and piping systems used in the release path are capable of performing their safety function during and following a safe shutdown earthquake (SSE). The amount of reduction allowed will be evaluated on an individual case basis. References A-9 and A-10 provide guidance on acceptable models.	Conforms	Main steam piping that is capable of performing its safety function during and following an SSE is credited. No credit is taken for holdup and deposition in piping downstream of the qualified main steam piping, or in the condenser. The modeling is per the RADTRAD Brockmann-Bixler approach.
7.0	The radiological consequences from post-LOCA primary containment purging as a combustible gas or pressure control measure should be analyzed. If the installed containment purging capabilities are	Conforms	Containment purging as a combustible gas or pressure control measure is not required nor credited

Table B: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)

RG Section	RG Position	PBAPS Analysis	Comments
	maintained for purposes of severe accident management and are not credited in any design basis analysis, radiological consequences need not be evaluated. If the primary containment purging is required within 30 days of the LOCA, the results of this analysis should be combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA. Reduction in the amount of radioactive material released via ESF filter systems may be taken into account provided that these systems meet the guidance in Regulatory Guide 1.52 (Ref. A-5) and Generic Letter 99-02 (Ref. A-6).		in any design basis analysis for 30 days following a design basis LOCA at PBAPS.

Table C: Conformance with Regulatory Guide 1.183 Appendix B (Fuel Handling Accident)			
RG Section	RG Position	PBAPS Analysis	Comments
1	Acceptable assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide.	Conforms	
1.1	The number of fuel rods damaged during the accident should be based on a conservative analysis that considers the most limiting case. This analysis should consider parameters such as the weight of the dropped heavy load or the weight of a dropped fuel assembly (plus any attached handling grapples), the height of the drop, and the compression, torsion, and shear stresses on the irradiated fuel rods. Damage to adjacent fuel assemblies, if applicable (e.g., events over the reactor vessel), should be considered.	Conforms	This is based on generic evaluation of GE11 and GE14 fuel, with heavy mast, yielding 172 failed rods, based on 87.33 rods per assembly and 764 assemblies in the core. Damage due to a fuel assembly drop into the reactor vessel bounds a drop in the spent fuel pool. This is due to the 33 foot drop in the vessel as opposed to the less than 4-foot fall height in the pool.
1.2	The fission product release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached. All the gap activity in the damaged rods is assumed to be instantaneously released. Radionuclides that should be considered include xenons, kryptons, halogens, cesiums, and rubidiums.	Conforms	Gap activity assumed is per this guidance.
1.3	The chemical form of radioiodine released from the fuel to the spent fuel pool should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. The CsI released from the fuel is assumed to completely dissociate in the pool water. Because of the low pH of the pool water, the iodine re-evolves as elemental iodine. This is assumed to occur instantaneously. The NRC staff will consider, on a case-by-case basis, justifiable mechanistic treatment of the iodine release from the pool.	Conforms	All iodine added to pool assumed to dissociate.
2	If the depth of water above the damaged fuel is 23 feet or greater, the decontamination factors for the elemental and organic species are 500 and 1, respectively, giving an overall effective decontamination factor of 200 (i.e., 99.5% of the total iodine released from the damaged rods is retained by the water). This difference in decontamination factors for elemental (99.85%) and organic iodine (0.15%) species results in the iodine above the water being composed of 57% elemental and 43%	Conforms	The analyzed water depth above damaged fuel is 23 feet. Although the actual water coverage over damaged fuel in the reactor vessel is 52 feet, no further credit is applied for the additional (i.e., >23 feet) water depth in accordance with regulatory guidance.

Table C: Conformance with Regulatory Guide 1.183 Appendix B (Fuel Handling Accident)			
RG Section	RG Position	PBAPS Analysis	Comments
	organic species. If the depth of water is not 23 feet, the decontamination factor will have to be determined on a case-by-case method (Ref. B-1).		An overall DF of 200 is used. For a drop over the spent fuel pool, coverage over the "striking" fuel assembly as it lies across the top of the "struck" fuel is 21.5 feet. It has been calculated that this coverage is sufficient to maintain a DF of 200 based on RG 1.183 Appendix C recommended DF of 500 for inorganic iodines and the recommended inorganic/organic iodine ratio. If a lower inorganic iodine DF is selected in order to force an overall DF of 200 with 23 feet of pool coverage, the reduction in DF (15%) is offset by the reduction in the amount of fuel damage (30%) due to a much shorter fall height over the spent fuel pool (< 4 feet) as compared to that in the vessel (> 33 feet). Therefore, 23 feet of water coverage is used since a drop into the vessel is the most limiting case.
3	The retention of noble gases in the water in the fuel pool or reactor cavity is negligible (i.e., decontamination factor of 1). Particulate radionuclides are assumed to be retained by the water in the fuel pool or reactor cavity (i.e., infinite decontamination factor).	Conforms	DF = 1 for noble gas isotopes.
4.1	The radioactive material that escapes from the fuel pool to the fuel building is assumed to be released to the environment over a 2-hour time period.	Conforms	The release is assumed to occur over a two hour period.
4.2	A reduction in the amount of radioactive material released from the fuel pool by engineered safety feature (ESF) filter systems may be taken into	Conforms	No credit is taken for the Standby Gas Treatment System or its elevated

Table C: Conformance with Regulatory Guide 1.183 Appendix B (Fuel Handling Accident)			
RG Section	RG Position	PBAPS Analysis	Comments
	account provided these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. B-2, B-3). Delays in radiation detection, actuation of the ESF filtration system, or diversion of ventilation flow to the ESF filtration system(21) should be determined and accounted for in the radioactivity release analyses.		release.
4.3	The radioactivity release from the fuel pool should be assumed to be drawn into the ESF filtration system without mixing or dilution in the fuel building. If mixing can be demonstrated, credit for mixing and dilution may be considered on a case-by-case basis. This evaluation should consider the magnitude of the building volume and exhaust rate, the potential for bypass to the environment, the location of exhaust plenums relative to the surface of the pool, recirculation ventilation systems, and internal walls and floors that impede stream flow between the surface of the pool and the exhaust plenums.	Conforms	Two-hour release to the environment is assumed, without SGTS or MCREV filtration, and without elevated release through the Main Stack.
5.1	If the containment is isolated during fuel handling operations, no radiological consequences need to be analyzed.	Conforms	Secondary Containment isolation is not credited.
5.2	If the containment is open during fuel handling operations, but designed to automatically isolate in the event of a fuel handling accident, the release duration should be based on delays in radiation detection and completion of containment isolation. If it can be shown that containment isolation occurs before radioactivity is released to the environment, no radiological consequences need to be analyzed.	Conforms	Automatic Secondary Containment isolation is not credited.
5.3	If the containment is open during fuel handling operations (e.g., personnel air lock or equipment hatch is open), the radioactive material that escapes from the reactor cavity pool to the containment is released to the environment over a 2-hour time period. Note 3: *The staff will generally require that technical specifications allowing such operations include administrative controls to close the airlock, hatch, or penetrations within 30 minutes. Such administrative controls will generally require that a dedicated individual be present, with the necessary equipment available, to restore containment closure should a*	Site-specific controls to be implement-ed	Secondary containment closure will be accomplished within a 1-hour time period. Administrative controls will be in place associated with closure of doors and penetrations.

Table C: Conformance with Regulatory Guide 1.183 Appendix B (Fuel Handling Accident)			
RG Section	RG Position	PBAPS Analysis	Comments
	fuel handling accident occur. Radiological analyses should generally not credit this manual isolation.		
5.4	A reduction in the amount of radioactive material released from the containment by ESF filter systems may be taken into account provided that these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. B-2 and B-3). Delays in radiation detection, actuation of the ESF filtration system, or diversion of ventilation flow to the ESF filtration system should be determined and accounted for in the radioactivity release analyses.	Not Applicable	No credit is being taken for filtration of release from the reactor building.
5.5	Credit for dilution or mixing of the activity released from the reactor cavity by natural or forced convection inside the containment may be considered on a case-by-case basis. Such credit is generally limited to 50% of the containment free volume. This evaluation should consider the magnitude of the containment volume and exhaust rate, the potential for bypass to the environment, the location of exhaust plenums relative to the surface of the reactor cavity, recirculation ventilation systems, and internal walls and floors that impede stream flow between the surface of the reactor cavity and the exhaust plenums.	Not Applicable	Two-hour release to the environment is assumed.

Table D: Conformance with Regulatory Guide 1.183 Appendix C (Control Rod Drop Accident)			
RG Section	RG Position	PBAPS Analysis	Comments
1	Assumptions acceptable to the NRC staff regarding core inventory are provided in Regulatory Position 3 of this guide. For the rod drop accident, the release from the breached fuel is based on the estimate of the number of fuel rods breached and the assumption that 10% of the core inventory of the noble gases and iodines is in the fuel gap. The release attributed to fuel melting is based on the fraction of the fuel that reaches or exceeds the initiation temperature for fuel melting and on the assumption that 100% of the noble gases and 50% of the iodines contained in that fraction are released to the reactor coolant.	Conforms	Breached/melted fuel rods and release fractions have been updated to reflect GE14 fuel, and release fractions per RG 1.183. Releases are based on 1,200 fuel rods breached and melting in 0.77% of the fuel contained in the breached rods. A conservative radial peaking factor of 1.7 is used in agreement with the AST Calculation for the Fuel Handling Accident. In addition to noble gas and iodine releases, releases of 12% of the core inventory of Cesium (an alkali metal, per Table 5 in Regulatory Position 3 of the guide) is assumed, based on Table 3 in Regulatory Position 3 of the guide. Radionuclide grouping is per Table 5 in Regulatory Position 3 of the guide, as implemented in RADTRAD.
2	If no or minimal fuel damage is postulated for the limiting event, the released activity should be the maximum coolant activity (typically 4 μCi/gm DE I-131) allowed by the technical specifications.	Conforms	Substantial fuel damage is postulated.
3.1	The activity released from the fuel from either the gap or from fuel pellets is assumed to be instantaneously mixed in the reactor coolant within the pressure vessel.	Conforms	Instantaneous mixing is assumed per this guidance.
3.2	Credit should not be assumed for partitioning in the pressure vessel or for removal by the steam separators.	Conforms	No partitioning is assumed.
3.3	Of the activity released from the reactor coolant within the pressure vessel, 100% of the noble gases, 10% of the iodine, and 1% of the remaining radionuclides are assumed to reach the turbine and condensers.	Conforms	Released activity is per this guidance.

Table D: Conformance with Regulatory Guide 1.183 Appendix C (Control Rod Drop Accident)			
RG Section	RG Position	PBAPS Analysis	Comments
3.4	Of the activity that reaches the turbine and condenser, 100% of the noble gases, 10% of the iodine, and 1% of the particulate radionuclides are available for release to the environment. The turbine and condensers leak to the atmosphere as a ground- level release at a rate of 1% per day for a period of 24 hours, at which time the leakage is assumed to terminate. No credit should be assumed for dilution or holdup within the turbine building. Radioactive decay during holdup in the turbine and condenser may be assumed.	Conforms	The condenser leak rate of 1% per day for a period of 24 hours is assumed. All releases are assumed to be at ground level and based on zero-velocity RB/TB vent stacks release assumptions. Radioactive decay during holdup in the condenser is assumed.
3.5	In lieu of the transport assumptions provided in paragraphs 3.2 through 3.4 above, a more mechanistic analysis may be used on a case-by-case basis. Such analyses account for the quantity of contaminated steam carried from the pressure vessel to the turbine and condensers based on a review of the minimum transport time from the pressure vessel to the first main steam isolation (MSIV) and considers MSIV closure time.	Not Applicable	Paragraphs 3.2 through 3.4 are used in the analysis.
3.6	The iodine species released from the reactor coolant within the pressure vessel should be assumed to be 95% CsI as an aerosol, 4.85% elemental, and 0.15% organic. The release from the turbine and condenser should be assumed to be 97% elemental and 3% organic.	Conforms	No credit for SGTS or MCREV filters is taken, and therefore variation in iodine species has no effect.
Foot-note 1	The activity assumed in the analysis should be based on the activity associated with the projected fuel damage or the maximum technical specification values, whichever maximizes the radiological consequences. In determining the dose equivalent I-131 (DE I-131), only the radioiodine associated with normal operations or iodine spikes should be included. Activity from projected fuel damage should not be included.	Conforms	Projected fuel damage is the limiting case.
Foot-note 2	If there are forced flow paths from the turbine or condenser, such as unisolated motor vacuum pumps or unprocessed air ejectors, the leakage rate should be assumed to be the flow rate associated with the most limiting of these paths. Credit for collection and processing of releases, such as by off gas or standby gas treatment, will be considered on a case-by-case basis.	Conforms	Upon detection of high radiation levels by the Main Steam Line Radiation Monitor system, the MSIVs close and the mechanical vacuum pump trips. Therefore, no forced flow path is applicable to PBAPS.

Table E: Conformance with Regulatory Guide 1.183 Appendix D (Main Steam Line Break)			
RG Section	RG Position	PBAPS Analysis	Comments
1	Assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide. The release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached.	Not Applicable	No fuel damage, release estimate based on coolant activity.
2	If no or minimal fuel damage is postulated for the limiting event, the released activity should be the maximum coolant activity allowed by technical specification. The iodine concentration in the primary coolant is assumed to correspond to the following two cases in the nuclear steam supply system vendor's standard technical specifications.	Conforms	Technical Specification SR 3.6.1.3.9 verifies the isolation time of each MSIV is between 3 and 5 seconds, well within the 10.5 seconds up to full MSIV closure assumed for the release period. Technical Specification LCO 3.4.6 limits the reactor coolant Dose Equivalent (DE) I-131 specific activity to 0.2 µCi/gm, with action to isolate all main steam lines if the reactor coolant DE I-131 specific activity exceeds 4.0 µCi/gm during Power Operation or Startup.
2.1	The concentration that is the maximum value (typically 4.0 µCi/gm DE I-131) permitted and corresponds to the conditions of an assumed pre-accident spike, and	Conforms	See Item 2 above.
2.2	The concentration that is the maximum equilibrium value (typically 0.2 µCi/gm DE I-131) permitted for continued full power operation.	Conforms	See Item 2 above.
3	The activity released from the fuel should be assumed to mix instantaneously and homogeneously in the reactor coolant. Noble gases should be assumed to enter the steam phase instantaneously.	Conforms	Mixing is per this guidance.
4.1	The main steam line isolation valves (MSIV) should be assumed to close in the maximum time allowed by technical specifications.	Conforms	See Item 2 above.
4.2	The total mass of coolant released should be assumed to be that amount in the steam line and connecting lines at the time of the break plus the amount that passes through the valves prior to closure.	Conforms	Mass of coolant released is per this guidance.
4.3	All the radioactivity in the released coolant should be assumed to be	Conforms	This guidance was used in the

Table E: Conformance with Regulatory Guide 1.183 Appendix D (Main Steam Line Break)

RG Section	RG Position	PBAPS Analysis	Comments
	released to the atmosphere instantaneously as a ground-level release. No credit should be assumed for plateout, holdup, or dilution within facility buildings.		analysis.
4.4	The iodine species released from the main steam line should be assumed to be 95% CsI as an aerosol, 4.85% elemental, and 0.15% organic.	Conforms	No filtration is credited, so the iodine species are irrelevant.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

Exelon Generation Company, LLC (EGC) is requesting a revision to the Facility Operating Licenses for Peach Bottom Atomic Power Station, Units 2 and 3. Specifically, we are requesting a revision to the Technical Specifications and licensing and design bases to reflect the application of alternative source term (AST) assumptions.

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

According to 10 CFR 50.92, "Issuance of amendment," paragraph (c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

In support of this determination, an evaluation of each of the three criteria set forth in 10 CFR 50.92 is provided below regarding the proposed license amendment.

5.1.1 The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

The implementation of alternative source term (AST) assumptions has been evaluated in revisions to the analyses of the following limiting design basis accidents (DBAs) at Peach Bottom Atomic Power Station (PBAPS):

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

Based upon the results of these analyses, it has been demonstrated that, with the requested changes, the dose consequences of these limiting events are within the regulatory guidance provided by the NRC for use with the AST. This guidance is presented in 10 CFR 50.67 and associated Regulatory Guide 1.183, and Standard Review Plan Section 15.0.1. The Alternative Source Term is an input to calculations used to evaluate the consequences of an accident, and does not by itself affect the

plant response, or the actual pathway of the radiation released from the fuel. It does however, better represent the physical characteristics of the release, so that appropriate mitigation techniques may be applied. Therefore, the consequences of an accident previously evaluated are not significantly increased.

The equipment affected by the proposed changes is mitigative in nature, and relied upon after an accident has been initiated. Application of the Alternative Source Term (AST) does not involve any physical changes to the plant design. While the operation of various systems do change as a result of these proposed changes, these systems are not accident initiators. Application of the AST is not an initiator of a design basis accident. The proposed changes to the Technical Specifications (TS), while they revise certain performance requirements, do not involve any physical modifications to the plant. As a result, the proposed changes do not affect any of the parameters or conditions that could contribute to the initiation of any accidents. As such, removal of operability requirements during the specified conditions will not significantly increase the probability of occurrence for an accident previously analyzed. Since design basis accident initiators are not being altered by adoption of the Alternative Source Term analyses, the probability of an accident previously evaluated is not affected.

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

5.1.2 The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed amendment does not involve a physical alteration of the plant (no new or different type of equipment will be installed and there are no physical modifications to existing equipment associated with the proposed changes). Similarly, it does not physically change any structures, systems or components involved in the mitigation of any accidents, thus, no new initiators or precursors of a new or different kind of accident are created. New equipment or personnel failure modes that might initiate a new type of accident are not created as a result of the proposed amendment.

As such the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

5.1.3 The proposed change does not involve a significant reduction in a margin of safety.

Safety margins and analytical conservatisms have been evaluated and have been found acceptable. The analyzed events have been carefully selected and margin has been retained to ensure that the analyses adequately bound postulated event scenarios. The dose consequences due to design basis accidents comply with the requirements of 10 CFR 50.67 and the guidance of Regulatory Guide 1.183.

The proposed amendment is associated with the implementation of a new licensing basis for PBAPS Design Basis Accidents (DBAs). Approval of the change from the original source term to a new source term taken from Regulatory Guide 1.183 is

being requested. The results of the accident analyses, revised in support of the proposed license amendment, are subject to revised acceptance criteria. The analyses have been performed using conservative methodologies, as specified in Regulatory Guide 1.183. Safety margins have been evaluated and analytical conservatism has been utilized to ensure that the analyses adequately bound the postulated limiting event scenario. The dose consequences of these DBAs remain within the acceptance criteria presented in 10 CFR 50.67, "Accident Source Term", and Regulatory Guide 1.183.

The proposed changes continue to ensure that the doses at the exclusion area boundary (EAB) and low population zone boundary (LPZ), as well as the Control Room, are within corresponding regulatory limits.

Therefore, operation of PBAPS in accordance with the proposed changes will not involve a significant reduction in a margin of safety.

Conclusion

Exelon Generation Company, LLC (EGC) concludes that the proposed amendments present no 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.

5.2 Applicable Regulatory Requirements/Criteria

The NRC's traditional methods (prior to the AST) for calculating the radiological consequences of design basis accidents are described in a series of regulatory guides and Standard Review Plan (SRP) chapters. That guidance was developed to be consistent with the 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 ASTs and with the Total Effective Dose Equivalent (TEDE) criteria provided in 10 CFR 50.67. Regulatory Guide 1.183 provides assumptions and methods that are acceptable to the NRC staff for performing design basis radiological analyses using an AST. This guidance supersedes corresponding radiological analysis assumptions provided in the older regulatory guides and SRP chapters when used in conjunction with an approved AST and the TEDE criteria provided in 10 CFR 50.67.

Due to the comprehensive nature of Regulatory Guide 1.183, the Tables in Section 4 above were incorporated into this submittal to show how each section of the new guidance is being addressed.

Also, the NRC published a new SRP section to address AST. It is Standard Review Plan Section 15.0.1, Rev. 0, entitled "Radiological Consequence Analyses Using Alternative Source Terms". It provides guidance on which NRC branches will review various aspects of an AST license amendment request, but otherwise is consistent with the guidance found in Regulatory Guide 1.183. The plant-specific information provided in this license amendment request is believed to adequately address the guidance found in SRP 15.0.1.

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 the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

Exelon Generation Company, LLC (EGC) has evaluated the proposed changes against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21, "Criteria for and identification of licensing and regulatory actions requiring environmental assessments." EGC has determined that the proposed changes meet the criteria for a categorical exclusion as set forth in 10 CFR 51.22, "Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review," paragraph (c)(9), and as such, has determined that no irreversible consequences exist in accordance with 10 CFR 50.92, "Issuance of amendment," paragraph (b). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities," which changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20," Standards for Protection Against Radiation," or that changes an inspection or a surveillance requirement, and the amendment meets the following specific criteria.

(i) **The amendment involves no significant hazards consideration.**

As demonstrated in Section 5.1 above, the proposed changes do not involve a significant hazards consideration.

(ii) **There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.**

The following table demonstrates that EGC meets the radiological criteria described in 10 CFR 50.67 for the exclusion area boundary (EAB) and the low population zone (LPZ). The EAB and LPZ doses represent a small fraction of the dose limits.

Dose Results (rem TEDE)				
Accident	EAB Doses and Limit		LPZ Doses and Limit	
	Dose	Limit	Dose	Limit
Loss of Coolant Accident	4.8	25	4.3	25

Main Steam Line Break	1.6[1] 0.08[2]	25[1] 2.5[2]	0.28[1] 0.014[2]	25[1] 2.5[2]
Control Rod Drop Accident	0.065	6.3	0.012	6.3
Fuel Handling Accident	1.2	6.3	0.14	6.3

Notes: 1. Based on a pre-accident spike concentration of 4.0 µCi/gm dose equivalent I-131.
2. Based on a maximum equilibrium concentration of 0.2 µCi/gm dose equivalent I-131.

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

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

The following table demonstrates that EGC meets the radiological criteria described in 10 CFR 50.67 for the control room (CR). CR exposure to operators is less than the five rem total effective dose equivalent (TEDE) over 30 days for all accidents.

CR Dose Results (rem TEDE)		
Accident	**Dose**	**Limit**
Loss of Coolant Accident	4.1	5.0
Main Steam Line Break	3.3[1] 0.17[2]	5.0
Control Rod Drop Accident	0.3	5.0
Fuel Handling Accident	2.4	5.0

Notes: 1. Based on a pre-accident spike concentration of 4.0 µCi/gm dose equivalent I-131.
2. Based on a maximum equilibrium concentration of 0.2 µCi/gm dose equivalent I-131.

The implementation of the alternative source term has been evaluated in revisions to the analyses of the limiting design basis accidents at Peach Bottom Atomic Power

Station, Units 2 and 3. These accidents include the control rod drop accident, fuel handling accident, loss of coolant accident, and main steam line break accident. Based upon the results of these analyses, it has been demonstrated that with the requested changes, the dose consequences of these limiting events are within the regulatory guidance provided by the NRC for use with alternative source term (i.e., 10 CFR 50.67 and 10 CFR 50, Appendix A, General Design Criterion 19). Thus, there will be no significant increase in either individual or cumulative occupational radiation exposure.

7.0 REFERENCES

7.1 U. S. Atomic Energy Commission, Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactor Sites," March 23, 1962

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

7.3 U. S. Nuclear Regulatory Commission Standard Review Plan 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Revision 0, July 2000

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

7.5 A. G. Croff, "A User's Manual for the ORIGEN 2 Computer Code," ORNL/TM-7175, Oak Ridge National Laboratory, July 1980

7.6 S. L. Humphreys et al., "RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation," NUREG/CR-6604, U. S. Nuclear Regulatory Commission, April 1998

7.7 U. S. Nuclear Regulatory Commission Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Revision 1, November 1982

7.8 T. J. Bander, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations," NUREG-2858, U. S. Nuclear Regulatory Commission, November 1982

7.9 J. V. Ramsdell and C. A. Simonen, "Atmospheric Relative Concentrations in Building Wakes," NUREG-6331, Revision 1, U. S. Nuclear Regulatory Commission, May 1997. (ARCON96)

7.10 ANSI/ANS-2.5-1984, "Standard for Determining Meteorological Information at Nuclear Power Sites"

7.11 U. S. Nuclear Regulatory Commission Regulatory Guide 1.23 (Safety Guide 23), "Onsite Meteorological Programs," February 17, 1972

7.12 U. S. Nuclear Regulatory Commission Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors," Revision 2, June 1974

7.13 U. S. Nuclear Regulatory Commission Standard Review Plan 6.4, "Control Room Habitability Systems," Revision 2, July 1981

7.14 U. S. Nuclear Regulatory Commission Regulatory Guide 1.5, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors," March 1971

7.15 Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion", 1988

7.16 Federal Guidance Report No. 12, "External Exposure to Radionuclides in Air, Water, and Soil", 1993.

7.17 Generic Letter 99-02, Laboratory Testing of Nuclear-Grade Activated Charcoal.

7.18 GE Report NEDE-31152P, "General Electric Fuel Bundle Designs", Revision 7, June 2000.

7.19 Draft Regulatory Guide DG-1111; Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants; U.S. Nuclear Regulatory Commission; December 2001.

7.20 Technical Specification Task Force (TSTF) Traveler, TSTF-51, "Revise Containment Requirements During Handling Irradiated Fuel and Core Alterations," Revision 2

7.21 NUREG-0737, "Clarification of TMI Action Plan Requirements", October 1980

7.22 Letter from US NRC to Mr. J. A. Hutton, PECO Energy Company, "Peach Bottom Atomic Power Station, Unit Nos. 2 and 3 – Issuance of Amendment Regarding Crediting of Containment Overpressure for Net Positive Suction Head Calculations for Emergency Core Cooling Pumps", dated August 14, 2000

ATTACHMENT 2

PEACH BOTTOM ATOMIC POWER STATION
UNITS 2 AND 3

Docket Nos. 50-277
50-278

License Nos. DPR-44
DPR-56

License Amendment Request
"PBAPS Alternative Source Term Implementation"

Markup of Technical Specification Pages

UNIT 2
1.1-2
3.1-20 to 21
3.3-54
3.3-58 to 59
3.6-16
3.6-34 to 36
3.6-38
3.6-40 to 42
3.7-7 to 8
3.8-19 to 23
3.8-33 to 36
3.8-44 to 45
5.0-12 to 14
5.0-17

UNIT 3
1.1-2
3.1-20 to 21
3.3-54
3.3-58 to 59
3.6-16
3.6-34 to 36
3.6-38
3.6-40 to 42
3.7-7 to 8
3.8-19 to 23
3.8-33 to 36
3.8-44 to 45
5.0-12 to 14
5.0-17

TS Inserts For PBAPS AST LAR

Insert 1

or Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1989.

Insert 2

Verify combined MSIV leakage rate for all four main steam lines is ≤ 174 scfh, and ≤ 100 scfh for any one steam line, when tested at ≥ 25 psig.

1.1 Definitions (continued)

CHANNEL FUNCTIONAL TEST

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

CORE ALTERATION

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

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

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

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

CORE OPERATING LIMITS REPORT (COLR)

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

DOSE EQUIVALENT I-131

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

(continued)

INSERT 1

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Standby Liquid Control (SLC) System

LCO 3.1.7 Two SLC subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, ~~and~~ 2, and 3

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Concentration of boron in solution > 9.82% weight.	A.1	Verify the concentration and temperature of boron in solution and pump suction piping temperature are within the limits of Figure 3.1.7-1.	8 hours AND Once per 12 hours thereafter
	AND		
	A.2	Restore concentration of boron in solution to ≤ 9.82% weight.	72 hours AND 10 days from discovery of failure to meet the LCO
B. One SLC subsystem inoperable for reasons other than Condition A.	B.1	Restore SLC subsystem to OPERABLE status.	7 days AND 10 days from discovery of failure to meet the LCO

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Two SLC subsystems inoperable for reasons other than Condition A.	C.1	Restore one SLC subsystem to OPERABLE status.	8 hours
D. Required Action and associated Completion Time not met.	D.1	Be in MODE 3.	12 hours
	AND		
	D.2	Be in MODE 4.	36 hours

INSERT

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.1	Verify level of sodium pentaborate solution in the SLC tank is ≥ 46%.	24 hours
SR 3.1.7.2	Verify temperature of sodium pentaborate solution is ≥ 53°F.	24 hours
SR 3.1.7.3	Verify temperature of pump suction piping is ≥ 53°F.	24 hours
SR 3.1.7.4	Verify continuity of explosive charge.	31 days

(continued)

Table 3.3.6.1-1 (page 3 of 3)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
5. Reactor Water Cleanup (RWCU) System Isolation					
a. RWCU Flow—High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.7	≤ 125% rated flow (23.0 in-wc)
b. SLC System Initiation	1,2,3	1	H	SR 3.3.6.1.7	NA
c. Reactor Vessel Water Level —Low (Level 3)	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.7	≥ 1.0 inches
6. RHR Shutdown Cooling System Isolation					
a. Reactor Pressure—High	1,2,3	1	F	SR 3.3.6.1.3 SR 3.3.6.1.7	≤ 70.0 psig
b. Reactor Vessel Water Level —Low (Level 3)	3,4,5	2[(a)]	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.7	≥ 1.0 inches
7. Feedwater Recirculation Isolation					
a. Reactor Pressure—High	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.7	≤ 600 psig

(a) In MODES 4 and 5, provided RHR Shutdown Cooling System integrity is maintained, only one channel per trip system with an isolation signal available to one shutdown cooling pump suction isolation valve is required.

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

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level —Low (Level 3)	1,2,3, (a)	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.4 SR 3.3.6.2.5	≥ 1.0 inches
2. Drywell Pressure —High	1,2,3	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.4 SR 3.3.6.2.5	≤ 2.0 psig
3. Reactor Building Ventilation Exhaust Radiation —High	1,2,3, (a),(b)	2	SR 3.3.6.2.1 SR 3.3.6.2.3 SR 3.3.6.2.5	≤ 16.0 mR/hr
4. Refueling Floor Ventilation Exhaust Radiation —High	1,2,3, (a),(b)	2	SR 3.3.6.2.1 SR 3.3.6.2.3 SR 3.3.6.2.5	≤ 16.0 mR/hr

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

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

recently

3.3 INSTRUMENTATION

3.3.7.1 Main Control Room Emergency Ventilation (MCREV) System Instrumentation

LCO 3.3.7.1 Two channels per trip system of the Control Room Air Intake Radiation—High Function shall be OPERABLE.

recently

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

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Declare associated MCREV subsystems inoperable.	1 hour from discovery of loss of MCREV System initiation capability
	AND	
	A.2 Place channel in trip.	6 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.14	~~Verify leakage rate through each MSIV is ≤ 11.5 scfh when tested at ≥ 25 psig.~~ INSERT 2	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.3.15	Verify each 6 inch and 18 inch primary containment purge valve and each 18 inch primary containment exhaust valve is blocked to restrict opening greater than the required maximum opening angle.	24 months
SR 3.6.1.3.16	Replace the inflatable seal of each 6 inch and 18 inch primary containment purge valve and each 18 inch primary containment exhaust valve.	96 months

3.6 CONTAINMENT SYSTEMS

3.6.4.1 Secondary Containment

LCO 3.6.4.1 The secondary containment shall be OPERABLE.

recently

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

ACTIONS

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	~~C.2 Suspend CORE ALTERATIONS.~~ ~~AND~~ C.3 Initiate action to suspend OPDRVs. 1	~~Immediately~~ Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1.1	Verify all secondary containment equipment hatches are closed and sealed.	31 days
SR 3.6.4.1.2	Verify one secondary containment access door in each access opening is closed.	31 days
SR 3.6.4.1.3	Verify each standby gas treatment (SGT) subsystem will draw down the secondary containment to ≥ 0.25 inch of vacuum water gauge in ≤ ~~120 seconds~~. 15 minutes	24 months on a STAGGERED TEST BASIS
SR 3.6.4.1.4	Verify each SGT subsystem can maintain ≥ 0.25 inch of vacuum water gauge in the secondary containment for 1 hour at a flow rate ≤ 10,500 cfm.	24 months on a STAGGERED TEST BASIS

3.6 CONTAINMENT SYSTEMS

3.6.4.2 Secondary Containment Isolation Valves (SCIVs)

LCO 3.6.4.2 Each SCIV shall be OPERABLE.

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

ACTIONS

-----NOTES-----

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

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

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

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

ACTIONS (continued)

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

3.6 CONTAINMENT SYSTEMS

3.6.4.3 Standby Gas Treatment (SGT) System

LCO 3.6.4.3 Two SGT subsystems shall be OPERABLE.

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

ACTIONS

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

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.3.1	Operate each SGT subsystem for ≥ 15 minutes ~~with heaters operating~~.	31 days
SR 3.6.4.3.2	Perform required SGT filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.6.4.3.3	Verify each SGT subsystem actuates on an actual or simulated initiation signal.	24 months

3.7 PLANT SYSTEMS

3.7.4 Main Control Room Emergency Ventilation (MCREV) System

LCO 3.7.4 Two MCREV subsystems shall be OPERABLE.

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

ACTIONS

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

ACTIONS

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

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.20	-------------------NOTES------------------ 1. All DG starts may be preceded by an engine prelube period. 2. A single test at the specified Frequency will satisfy this Surveillance for both units. -- Verify, when started simultaneously from standby condition, each DG achieves, in ≤ 10 seconds, voltage ≥ 4160 V and frequency ≥ 58.8 Hz.	10 years
SR 3.8.1.21 recently	-------------------NOTE------------------- When Unit 3 is in MODE 4 or 5, or moving irradiated fuel assemblies in the secondary containment, the Note to Unit 3 SR 3.8.2.1 is applicable. -- For required Unit 3 AC sources, the SRs of Unit 3 Specification 3.8.1, except SR 3.8.1.8 (when only one Unit 3 offsite circuit is required), SR 3.8.1.12, SR 3.8.1.13, SR 3.8.1.17, SR 3.8.1.18 (ECCS load block requirement only), and SR 3.8.1.19, are applicable.	In accordance with applicable SRs

3.8 ELECTRICAL POWER SYSTEMS

3.8.2 AC Sources—Shutdown

LCO 3.8.2 The following AC electrical power sources shall be OPERABLE:

a. One qualified circuit between the offsite transmission network and the Unit 2 onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.8, "Distribution Systems—Shutdown";

b. Two DGs each capable of supplying one Unit 2 onsite Class 1E AC electrical power distribution subsystem required by LCO 3.8.8;

c. One qualified circuit between the offsite transmission network and the Unit 3 onsite Class 1E AC electrical power distribution subsystem(s) needed to support the Unit 3 powered equipment required to be OPERABLE by LCO 3.6.4.3, "Standby Gas Treatment (SGT) System," and LCO 3.8.5, "DC Sources—Shutdown"; and

d. One DG capable of supplying one Unit 3 onsite Class 1E AC electrical power distribution subsystem needed to support the Unit 3 powered equipment required to be OPERABLE by:

1. LCO 3.6.4.3.

<u>OR</u>

2. LCO 3.8.5.

recently

APPLICABILITY: MODES 4 and 5,
During movement of irradiated fuel assemblies in the secondary containment.

ACTIONS

-----------------------------------NOTE-----------------------------------
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required offsite circuits inoperable.	------------NOTE------------ Enter applicable Condition and Required Actions of LCO 3.8.8, with one or more required 4 kV emergency buses de-energized as a result of Condition A. ----------------------------	
	A.1 Declare affected required feature(s), with no offsite power available inoperable.	Immediately
	OR	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2.2 Suspend movement of recently irradiated fuel assemblies in the secondary containment.	Immediately
	AND	
	A.2.3 Initiate action to suspend operations with a potential for draining the reactor vessel (OPDRVs).	Immediately
	AND	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.4 Initiate action to restore required offsite power circuit(s) to OPERABLE status.	Immediately
B. One required DG inoperable.	B.1 Declare affected required feature(s) with no DG available inoperable.	Immediately
	OR	
	B.2.1 Suspend CORE ALTERATIONS	Immediately
	AND	
	B.2.2 Suspend movement of recently irradiated fuel assemblies in the secondary containment.	Immediately
	AND	
	B.2.3 Initiate action to suspend OPDRVs.	Immediately
	AND	
	B.2.4 Initiate action to restore required DGs to OPERABLE status.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Two or more required DGs inoperable.	C.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	C.2 Suspend movement of recently irradiated fuel assemblies in secondary containment.	Immediately
	AND	
	C.3 Initiate action to suspend OPDRVs.	Immediately
	AND	
	C.4 Initiate action to restore required DG(s) to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.8.4.9 recently	-------------------NOTE-------------------- When Unit 3 is in MODE 4 or 5, or moving irradiated fuel assemblies in the secondary containment, the Note to Unit 3 SR 3.8.5.1 is applicable. --- For required Unit 3 DC electrical power subsystems, the SRs of Unit 3 Specification 3.8.4 are applicable.	In accordance with applicable SRs

3.8 ELECTRICAL POWER SYSTEMS

3.8.5 DC Sources—Shutdown

LCO 3.8.5 The following DC electrical power subsystems shall be OPERABLE:

a. Unit 2 DC electrical power subsystems needed to support the DC electrical power distribution subsystem(s) required by LCO 3.8.8, "Distribution Systems—Shutdown"; and

b. Unit 3 DC electrical power subsystems needed to support the DC electrical power distribution subsystem(s) required by LCO 3.8.8, "Distribution Systems—Shutdown."

APPLICABILITY: MODES 4 and 5,
During movement of recently irradiated fuel assemblies in the secondary containment.

ACTIONS

------------------------------------NOTE------------------------------------
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more required DC electrical power subsystems inoperable.	A.1	Declare affected required feature(s) inoperable.	Immediately
	<u>OR</u>		
	A.2.1	Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.2 Suspend movement of irradiated fuel assemblies in the secondary containment. recently	Immediately
	AND	
	A.2.3 Initiate action to suspend operations with a potential for draining the reactor vessel.	Immediately
	AND	
	A.2.4 Initiate action to restore required DC electrical power subsystems to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.5.1	-------------------NOTE------------------- The following SRs are not required to be performed: SR 3.8.4.7 and SR 3.8.4.8. -- For required Unit 2 DC electrical power subsystems, the following SRs are applicable: SR 3.8.4.1 SR 3.8.4.4 SR 3.8.4.7 SR 3.8.4.2 SR 3.8.4.5 SR 3.8.4.8. SR 3.8.4.3 SR 3.8.4.6	In accordance with applicable SRs
SR 3.8.5.2	-------------------NOTE------------------- When Unit 3 is in MODE 4 or 5, or moving recently irradiated fuel assemblies in the secondary containment, the Note to Unit 3 SR 3.8.5.1 is applicable. -- For required Unit 3 DC electrical power subsystems, the SRs of Unit 3 Specification 3.8.4 are applicable.	In accordance with applicable SRs

3.8 ELECTRICAL POWER SYSTEMS

3.8.8 Distribution Systems—Shutdown

LCO 3.8.8 The necessary portions of the following AC and DC electrical power distribution subsystems shall be OPERABLE:

a. Unit 2 AC and DC electrical power distribution subsystems needed to support equipment required to be OPERABLE; and

b. Unit 3 AC and DC electrical power distribution subsystems needed to support equipment required to be OPERABLE by LCO 3.4.8, "Residual Heat Removal (RHR) Shutdown Cooling System—Cold Shutdown," LCO 3.5.2, "ECCS—Shutdown," LCO 3.6.4.3, "Standby Gas Treatment (SGT) System," LCO 3.8.2, "AC Sources—Shutdown," LCO 3.9.7, "RHR—High Water Level," and LCO 3.9.8, "RHR—Low Water Level."

APPLICABILITY: MODES 4 and 5,
During movement of recently irradiated fuel assemblies in the secondary containment.

ACTIONS

------------------------------------NOTE------------------------------------
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required AC or DC electrical power distribution subsystems inoperable.	A.1 Declare associated supported required feature(s) inoperable.	Immediately
	<u>OR</u>	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2.2	Suspend handling of recently irradiated fuel assemblies in the secondary containment.	Immediately
	AND		
	A.2.3	Initiate action to suspend operations with a potential for draining the reactor vessel.	Immediately
	AND		
	A.2.4	Initiate actions to restore required AC and DC electrical power distribution subsystems to OPERABLE status.	Immediately
	AND		
	A.2.5	Declare associated required shutdown cooling subsystem(s) inoperable and not in operation.	Immediately

5.5 Programs and Manuals

5.5.7 Ventilation Filter Testing Program (VFTP) (continued)

1) Once per 12 months for standby service or after 720 hours of system operation; and,

2) After each complete or partial replacement of the HEPA filter train or charcoal adsorber filter; after any structural maintenance on the system housing; and, following significant painting, fire, or chemical release in any ventilation zone communicating with the system while it is in operation.

The Test~~s~~ described in Specification~~s~~ 5.5.7.d ~~and 5.5.7.e~~ shall be performed once per 24 months.

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

a. Demonstrate for ~~each of~~ the ~~ESF~~ MCREV system~~s~~ that an inplace test of the HEPA filters shows a penetration and system bypass < 1.0% when tested in accordance with Regulatory Guide 1.52, Revision 2, Section 5c, and ASME N510-1989, Section~~s 6 (Standby Gas Treatment (SGT) System only) and~~ 10, at the system flowrate specified below.

ESF Ventilation System	Flowrate (cfm)
~~SGT System~~	~~7200 to 8800~~
Main Control Room Emergency Ventilation (MCREV) System	2700 to 3300

(continued)

5.5 Programs and Manuals

5.5.7 Ventilation Filter Testing Program (VFTP) (continued)

b. Demonstrate for ~~each of~~ the ESF MCREV systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < 1.0% when tested in accordance with Regulatory Guide 1.52, Revision 2, Section 5d, and ASME N510-1989, Section~~s 6 (SGT System only) and~~ 11, at the system flowrate specified below.

ESF Ventilation System	Flowrate (cfm)
~~SGT System~~	~~7200 to 8800~~
MCREV System	2700 to 3300

c. Demonstrate for ~~each of~~ the ESF MCREV systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, Section 6b, shows the methyl iodide penetration less than the value specified below when tested in accordance with the laboratory testing criteria of ASTM D3803-1989 at a temperature of 30 degrees C [86 degrees F], face velocity, and the relative humidity specified below.

	~~ESF Ventilation System~~	
	~~SGT System~~	MCREV System
Penetration (%)	~~5~~	~~5~~ 15
Face Velocity (FPM)	~~60~~	57
Relative Humidity: (%)	~~70~~	95

(continued)

5.5 Programs and Manuals

5.5.7 Ventilation Filter Testing Program (VFTP) (continued)

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

ESF Ventilation System	Delta P (inches wg)	Flowrate (cfm)
SGT System	< 3.9	7200 to 8800
MCREV System	< 8	2700 to 3300

~~e. Demonstrate that the heaters for the SGT System dissipate ≥ 40 kw.~~

5.5.8 Explosive Gas Monitoring Program

This program provides controls for potentially explosive gas mixtures contained downstream of the off-gas recombiners.

The program shall include:

a. The limit for the concentration of hydrogen downstream of the off-gas recombiners and a surveillance program to ensure the limit is maintained. This limit shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion);

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas Monitoring Program surveillance frequencies.

5.5.9 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

(continued)

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

1. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or

2. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

3. A required system redundant to support system(s) for the supported systems (b.1) and (b.2) above is also inoperable.

c. The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.12 Primary Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exception to NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Section 10.2:

a. MSIV leakage is excluded from the combined total of 0.6 L_a for the Type B and C tests.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 49.1 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.5% of primary containment air weight per day.

0.7%

Leakage Rate acceptance criteria are:

a. Primary Containment leakage rate acceptance criterion is $\leq$ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq$ 0.60 L_a for the Type B and Type C tests and $\leq$ 0.75 L_a for Type A tests;

(continued)

1.1 Definitions (continued)

CHANNEL FUNCTIONAL TEST

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

CORE ALTERATION

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

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

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

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

CORE OPERATING LIMITS REPORT (COLR)

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

DOSE EQUIVALENT I-131

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

(continued)

INSERT 1

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Standby Liquid Control (SLC) System

LCO 3.1.7 Two SLC subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, ~~and~~ 2, and 3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Concentration of boron in solution > 9.82% weight.	A.1 Verify the concentration and temperature of boron in solution and pump suction piping temperature are within the limits of Figure 3.1.7-1.	8 hours AND Once per 12 hours thereafter
	AND	
	A.2 Restore concentration of boron in solution to ≤ 9.82% weight.	72 hours AND 10 days from discovery of failure to meet the LCO
B. One SLC subsystem inoperable for reasons other than Condition A.	B.1 Restore SLC subsystem to OPERABLE status.	7 days AND 10 days from discovery of failure to meet the LCO

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Two SLC subsystems inoperable for reasons other than Condition A.	C.1	Restore one SLC subsystem to OPERABLE status.	8 hours
D. Required Action and associated Completion Time not met.	D.1	Be in MODE 3.	12 hours
	AND		
	D.2	Be in MODE 4.	36 hours

INSERT

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.1	Verify level of sodium pentaborate solution in the SLC tank is ≥ 46%.	24 hours
SR 3.1.7.2	Verify temperature of sodium pentaborate solution is ≥ 53°F.	24 hours
SR 3.1.7.3	Verify temperature of pump suction piping is ≥ 53°F.	24 hours
SR 3.1.7.4	Verify continuity of explosive charge.	31 days

(continued)

Table 3.3.6.1-1 (page 3 of 3)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
5. Reactor Water Cleanup (RWCU) System Isolation					
a. RWCU Flow — High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.7	≤ 125% rated flow (23.0 in-wc)
b. SLC System Initiation	1,2,3	1	H	SR 3.3.6.1.7	NA
c. Reactor Vessel Water Level — Low (Level 3)	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.7	≥ 1.0 inches
6. RHR Shutdown Cooling System Isolation					
a. Reactor Pressure — High	1,2,3	1	F	SR 3.3.6.1.3 SR 3.3.6.1.7	≤ 70.0 psig
b. Reactor Vessel Water Level — Low (Level 3)	3,4,5	2[a]	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.7	≥ 1.0 inches
7. Feedwater Recirculation Isolation					
a. Reactor Pressure — High	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.7	≤ 600 psig

(a) In MODES 4 and 5, provided RHR Shutdown Cooling System integrity is maintained, only one channel per trip system with an isolation signal available to one shutdown cooling pump suction isolation valve is required.

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

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level —Low (Level 3)	1,2,3, (a)	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.4 SR 3.3.6.2.5	≥ 1.0 inches
2. Drywell Pressure —High	1,2,3	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.4 SR 3.3.6.2.5	≤ 2.0 psig
3. Reactor Building Ventilation Exhaust Radiation —High	1,2,3, (a),(b)	2	SR 3.3.6.2.1 SR 3.3.6.2.3 SR 3.3.6.2.5	≤ 16.0 mR/hr
4. Refueling Floor Ventilation Exhaust Radiation —High	1,2,3, (a),(b)	2	SR 3.3.6.2.1 SR 3.3.6.2.3 SR 3.3.6.2.5	≤ 16.0 mR/hr

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

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

3.3 INSTRUMENTATION

3.3.7.1 Main Control Room Emergency Ventilation (MCREV) System Instrumentation

LCO 3.3.7.1 Two channels per trip system of the Control Room Air Intake Radiation—High Function shall be OPERABLE.

recently

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

ACTIONS

---NOTE---
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Declare associated MCREV subsystems inoperable.	1 hour from discovery of loss of MCREV System initiation capability
	AND	
	A.2 Place channel in trip.	6 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.14	~~Verify leakage rate through each MSIV is ≤ 11.5 scfh when tested at ≥ 25 psig.~~ INSERT 2	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.3.15	Verify each 6 inch and 18 inch primary containment purge valve and each 18 inch primary containment exhaust valve is blocked to restrict opening greater than the required maximum opening angle.	24 months
SR 3.6.1.3.16	Replace the inflatable seal of each 6 inch and 18 inch primary containment purge valve and each 18 inch primary containment exhaust valve.	96 months

3.6 CONTAINMENT SYSTEMS

3.6.4.1 Secondary Containment

LCO 3.6.4.1 The secondary containment shall be OPERABLE.

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

ACTIONS

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	~~C.2 Suspend CORE ALTERATIONS.~~ ~~AND~~	~~Immediately~~
	C.~~3~~ 2 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1.1	Verify all secondary containment equipment hatches are closed and sealed.	31 days
SR 3.6.4.1.2	Verify one secondary containment access door in each access opening is closed.	31 days
SR 3.6.4.1.3	Verify each standby gas treatment (SGT) subsystem will draw down the secondary containment to ≥ 0.25 inch of vacuum water gauge in ≤ ~~120 seconds~~ 15 minutes	24 months on a STAGGERED TEST BASIS
SR 3.6.4.1.4	Verify each SGT subsystem can maintain ≥ 0.25 inch of vacuum water gauge in the secondary containment for 1 hour at a flow rate ≤ 10,500 cfm.	24 months on a STAGGERED TEST BASIS

3.6 CONTAINMENT SYSTEMS

3.6.4.2 Secondary Containment Isolation Valves (SCIVs)

LCO 3.6.4.2 Each SCIV shall be OPERABLE.

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

ACTIONS

-----NOTES-----

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

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

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

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

ACTIONS (continued)

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

3.6 CONTAINMENT SYSTEMS

3.6.4.3 Standby Gas Treatment (SGT) System

LCO 3.6.4.3 Two SGT subsystems shall be OPERABLE.

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

ACTIONS

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

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.3.1	Operate each SGT subsystem for ≥ 15 minutes ~~with heaters operating~~.	31 days
SR 3.6.4.3.2	Perform required SGT filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.6.4.3.3	Verify each SGT subsystem actuates on an actual or simulated initiation signal.	24 months

3.7 PLANT SYSTEMS

3.7.4 Main Control Room Emergency Ventilation (MCREV) System

LCO 3.7.4 Two MCREV subsystems shall be OPERABLE.

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

ACTIONS

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

ACTIONS

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

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.8.1.20	-------------------NOTES------------------ 1. All DG starts may be preceded by an engine prelube period. 2. A single test at the specified Frequency will satisfy this Surveillance for both units. -- Verify, when started simultaneously from standby condition, each DG achieves, in ≤ 10 seconds, voltage ≥ 4160 V and frequency ≥ 58.8 Hz.	10 years
SR 3.8.1.21	-------------------NOTE------------------- When Unit 2 is in MODE 4 or 5, or moving recently irradiated fuel assemblies in the secondary containment, the Note to Unit 2 SR 3.8.2.1 is applicable. -- For required Unit 2 AC sources, the SRs of Unit 2 Specification 3.8.1, except SR 3.8.1.8 (when only one Unit 2 offsite circuit is required), SR 3.8.1.12, SR 3.8.1.13, SR 3.8.1.17, SR 3.8.1.18 (ECCS load block requirement only), and SR 3.8.1.19, are applicable.	In accordance with applicable SRs

3.8 ELECTRICAL POWER SYSTEMS

3.8.2 AC Sources—Shutdown

LCO 3.8.2 The following AC electrical power sources shall be OPERABLE:

a. One qualified circuit between the offsite transmission network and the Unit 3 onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.8, "Distribution Systems—Shutdown";

b. Two DGs each capable of supplying one Unit 3 onsite Class 1E AC electrical power distribution subsystem required by LCO 3.8.8;

c. One qualified circuit between the offsite transmission network and the Unit 2 onsite Class 1E AC electrical power distribution subsystem(s) needed to support the Unit 2 powered equipment required to be OPERABLE by LCO 3.6.4.3, "Standby Gas Treatment (SGT) System", LCO 3.7.4, "Main Control Room Emergency Ventilation (MCREV) System," and LCO 3.8.5, "DC Sources—Shutdown"; and

d. The DG(s) capable of supplying one subsystem of each of the Unit 2 powered equipment required to be OPERABLE by LCO 3.6.4.3, LCO 3.7.4, and LCO 3.8.5.

APPLICABILITY: MODES 4 and 5,
recently
During movement of irradiated fuel assemblies in the secondary containment.

ACTIONS

-----------------------------------NOTE-----------------------------------
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required offsite circuits inoperable.	------------NOTE------------ Enter applicable Condition and Required Actions of LCO 3.8.8, with one or more required 4 kV emergency buses de-energized as a result of Condition A. ----------------------------	
	A.1 Declare affected required feature(s), with no offsite power available inoperable.	Immediately
	<u>OR</u>	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	A.2.2 Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	<u>AND</u> recently	
	A.2.3 Initiate action to suspend operations with a potential for draining the reactor vessel (OPDRVs).	Immediately
	<u>AND</u>	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.4 Initiate action to restore required offsite power circuit(s) to OPERABLE status.	Immediately
B. One required DG inoperable.	B.1 Declare affected required feature(s) with no DG available inoperable.	Immediately
	OR	
	B.2.1 Suspend CORE ALTERATIONS	Immediately
	AND	
	B.2.2 Suspend movement of recently irradiated fuel assemblies in the secondary containment.	Immediately
	AND	
	B.2.3 Initiate action to suspend OPDRVs.	Immediately
	AND	
	B.2.4 Initiate action to restore required DGs to OPERABLE status.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Two or more required DGs inoperable.	C.1	Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>		
	C.2	Suspend movement of irradiated fuel assemblies in secondary containment. recently	Immediately
	<u>AND</u>		
	C.3	Initiate action to suspend OPDRVs.	Immediately
	<u>AND</u>		
	C.4	Initiate action to restore required DG(s) to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.4.9	-------------------NOTE------------------- When Unit 2 is in MODE 4 or 5, or moving recently irradiated fuel assemblies in the secondary containment, the Note to Unit 2 SR 3.8.5.1 is applicable. ---	
	For required Unit 2 DC electrical power subsystems, the SRs of Unit 2 Specification 3.8.4 are applicable.	In accordance with applicable SRs

3.8 ELECTRICAL POWER SYSTEMS

3.8.5 DC Sources—Shutdown

LCO 3.8.5 The following DC electrical power subsystems shall be OPERABLE:

a. Unit 3 DC electrical power subsystems needed to support the DC electrical power distribution subsystem(s) required by LCO 3.8.8, "Distribution Systems—Shutdown"; and

b. Unit 2 DC electrical power subsystems needed to support the DC electrical power distribution subsystem(s) required by LCO 3.8.8, "Distribution Systems—Shutdown."

APPLICABILITY: MODES 4 and 5,
During movement of recently irradiated fuel assemblies in the secondary containment.

ACTIONS

------------------------------------NOTE------------------------------------
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more required DC electrical power subsystems inoperable.	A.1	Declare affected required feature(s) inoperable.	Immediately
	<u>OR</u>		
	A.2.1	Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.2 Suspend movement of irradiated fuel assemblies in the secondary containment. recently	Immediately
	<u>AND</u>	
	A.2.3 Initiate action to suspend operations with a potential for draining the reactor vessel.	Immediately
	<u>AND</u>	
	A.2.4 Initiate action to restore required DC electrical power subsystems to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.5.1	-------------------NOTE------------------- The following SRs are not required to be performed: SR 3.8.4.7 and SR 3.8.4.8. -- For required Unit 3 DC electrical power subsystems, the following SRs are applicable: SR 3.8.4.1 SR 3.8.4.4 SR 3.8.4.7 SR 3.8.4.2 SR 3.8.4.5 SR 3.8.4.8. SR 3.8.4.3 SR 3.8.4.6	In accordance with applicable SRs
SR 3.8.5.2 recently	-------------------NOTE------------------- When Unit 2 is in MODE 4 or 5, or moving irradiated fuel assemblies in the secondary containment, the Note to Unit 2 SR 3.8.5.1 is applicable. -- For required Unit 2 DC electrical power subsystems, the SRs for Unit 2 Specification 3.8.4 are applicable.	In accordance with applicable SRs

3.8 ELECTRICAL POWER SYSTEMS

3.8.8 Distribution Systems—Shutdown

LCO 3.8.8 The necessary portions of the following AC and DC electrical power distribution subsystems shall be OPERABLE:

a. Unit 3 AC and DC electrical power distribution subsystems needed to support equipment required to be OPERABLE; and

b. Unit 2 AC and DC electrical power distribution subsystems needed to support equipment required to be OPERABLE by LCO 3.4.8, "Residual Heat Removal (RHR) Shutdown Cooling System—Cold Shutdown," LCO 3.5.2, "ECCS—Shutdown," LCO 3.6.4.3, "Standby Gas Treatment (SGT) System," LCO 3.7.4, "Main Control Room Emergency Ventilation (MCREV) System," LCO 3.8.2, "AC Sources—Shutdown," LCO 3.9.7, "RHR—High Water Level," and LCO 3.9.8, "RHR—Low Water Level."

APPLICABILITY: MODES 4 and 5,
During movement of recently irradiated fuel assemblies in the secondary containment.

ACTIONS

-----------------------------------NOTE-----------------------------------
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required AC or DC electrical power distribution subsystems inoperable.	A.1 Declare associated supported required feature(s) inoperable.	Immediately
	<u>OR</u>	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.2 Suspend handling of recently irradiated fuel assemblies in the secondary containment. AND	Immediately
	A.2.3 Initiate action to suspend operations with a potential for draining the reactor vessel. AND	Immediately
	A.2.4 Initiate actions to restore required AC and DC electrical power distribution subsystems to OPERABLE status. AND	Immediately
	A.2.5 Declare associated required shutdown cooling subsystem(s) inoperable and not in operation.	Immediately

5.5 Programs and Manuals

5.5.7 Ventilation Filter Testing Program (VFTP) (continued)

1) Once per 12 months for standby service or after 720 hours of system operation; and,

2) After each complete or partial replacement of the HEPA filter train or charcoal adsorber filter; after any structural maintenance on the system housing; and, following significant painting, fire, or chemical release in any ventilation zone communicating with the system while it is in operation.

The Test~~s~~ described in Specification~~s~~ 5.5.7.d ~~and 5.5.7.e~~ shall be performed once per 24 months.

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

a. Demonstrate for ~~each of~~ the ~~ESF~~ MCREV system~~s~~ that an inplace test of the HEPA filters shows a penetration and system bypass < 1.0% when tested in accordance with Regulatory Guide 1.52, Revision 2, Section 5c, and ASME N510-1989, Section~~s 6 (Standby Gas Treatment (SGT) System only) and~~ 10, at the system flowrate specified below.

ESF Ventilation System	Flowrate (cfm)
~~SGT System~~	~~7200 to 8800~~
Main Control Room Emergency Ventilation (MCREV) System	2700 to 3300

(continued)

5.5 Programs and Manuals

5.5.7 Ventilation Filter Testing Program (VFTP) (continued)

b. Demonstrate for ~~each of~~ the ESF MCREV systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < 1.0% when tested in accordance with Regulatory Guide 1.52, Revision 2, Section 5d, and ASME N510-1989, Section~~s 6 (SGT System only) and~~ 11, at the system flowrate specified below.

ESF Ventilation System	Flowrate (cfm)
~~SGT System~~	~~7200 to 8800~~
MCREV System	2700 to 3300

c. Demonstrate for ~~each of~~ the ESF MCREV systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, Section 6b, shows the methyl iodide penetration less than the value specified below when tested in accordance with the laboratory testing criteria of ASTM D3803-1989 at a temperature of 30 degrees C [86 degrees F], face velocity, and the relative humidity specified below.

	~~ESF Ventilation System~~	
	~~SGT System~~	MCREV System
Penetration (%)	~~5~~	~~5~~ 15
Face Velocity (FPM)	~~60~~	57
Relative Humidity: (%)	~~70~~	95

(continued)

5.5 Programs and Manuals

5.5.7 Ventilation Filter Testing Program (VFTP) (continued)

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

ESF Ventilation System	Delta P (inches wg)	Flowrate (cfm)
SGT System	< 3.9	7200 to 8800
MCREV System	< 8	2700 to 3300

~~e. Demonstrate that the heaters for the SGT System dissipate ≥ 40 kw.~~

5.5.8 Explosive Gas Monitoring Program

This program provides controls for potentially explosive gas mixtures contained downstream of the off-gas recombiners.

The program shall include:

a. The limit for the concentration of hydrogen downstream of the off-gas recombiners and a surveillance program to ensure the limit is maintained. This limit shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion);

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas Monitoring Program surveillance frequencies.

5.5.9 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

(continued)

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

1. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or

2. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

3. A required system redundant to support system(s) for the supported systems (b.1) and (b.2) above is also inoperable.

c. The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.12 Primary Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exceptions to NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J":

a. Section 10.2: MSIV leakage is excluded from the combined total of 0.6 L_a for the Type B and C tests.

b. Section 9.2.3: The first Type A test performed after the December, 1991 Type A test shall be performed no later than December, 2006.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 49.1 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, shall be ~~0.5%~~ of primary containment air weight per day.

Leakage Rate acceptance criteria are:

0.7%.

a. Primary Containment leakage rate acceptance criterion is ≤ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are ≤ 0.60 L_a for the Type B and Type C tests and ≤ 0.75 L_a for Type A tests;

(continued)

ATTACHMENT 3

PEACH BOTTOM ATOMIC POWER STATION
UNITS 2 AND 3

Docket Nos. 50-277
50-278

License Nos. DPR-44
DPR-56

License Amendment Request
"PBAPS Alternative Source Term Implementation"

Markup of Technical Specification Bases Pages
(For information only)

<u>UNITS 2 & 3</u>

B 2.0-5 to 7
B 2.0-9 to 10
B 3.1-37
B 3.1-39 to 41
B 3.1-44
B 3.1-47
B 3.3-147 to 150 (Unit 2 only)
B 3.3-148 to 151 (Unit 3 only)
B 3.3-156 (Unit 2 only)
B 3.3-157 (Unit 3 only)
B 3.3-174
B 3.3-182
B 3.4-29 to 32
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B 3.6-73 to 76
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B 3.7-16 to 20
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B 3.8-38
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PBAPS Units 2 and 3 Technical Specification Bases Markup Inserts

INSERT A {pg. B 3.6-29}

Total leakage through all four main steam lines must be $\leq$ 250 scfh, and $\leq$ 100 scfh for any one steam line, when tested at $\geq$ 25 psig. The analysis in Reference 1 is based on treatment of MSIV leakage as secondary containment bypass leakage, independent of the primary to secondary containment leakage analyzed at L_a. The Frequency is in accordance with the Primary Containment Leakage Rate Testing Program.

INSERT B {pg. B 3.1-39}

The SLC System is also used to maintain suppression pool pH at or above 7 following a loss of coolant accident (LOCA) involving significant fission product releases. Maintaining suppression pool pH levels at or above 7 following an accident ensures that iodine will be retained in the suppression pool water.

INSERT C {pg. B 3.1-41}

In MODES 1, 2, and 3, the SLC System must be OPERABLE to ensure that offsite doses remain within 10 CFR 50.67 (Ref. 3) limits following a LOCA involving significant fission product releases. The SLC System is designed to maintain suppression pool pH at or above 7 following a LOCA to ensure that iodine will be retained in the suppression pool water.

INSERT D {pg. B 3.3-156}

. Both channels are also required to be OPERABLE in MODES 1, 2, and 3, since the SLC System is also designed to maintain suppression pool pH above 7 following a LOCA to ensure that iodine will be retained in the suppression pool water. These

INSERT E {pg. B 3.6-73}

The function of the secondary containment is to receive fission products that may leak from primary containment or from systems in secondary containment following a Design Basis Accident (DBA) and, in conjunction with the Standby Gas Treatment System (SGT) and closure of certain valves whose lines penetrate the secondary containment, to provide for elevated release through the Main Stack.

INSERT F {pg. B 3.6-76}

The SGT System exhausts the secondary containment atmosphere to the environment through the elevated release point provided by the Main Stack.

To ensure that this exhaust pathway is used, SR 3.6.4.1.3

INSERT G {pg. B 3.6-85}

The primary function of the SGT System is to ensure that radioactive materials that leak from primary containment into the secondary containment following a Design Basis Accident (DBA) are discharged through the elevated release provided by the Main Stack.

INSERT H {pg. B 3.6-85}

These filters are not credited in any DBA analysis.

INSERT I {pg. B 3.6-86}

The design basis for the SGT System is to mitigate the consequences of a loss of coolant accident by providing a controlled, elevated release path. The SGT system also provides this function for OPDRVs. For all events where required, the SGT System automatically initiates to reduce, via an elevated release, the consequences of radioactive material released to the environment.

The HEPA filter and charcoal adsorber provided in the SGT System are not credited for any DBA analysis.

INSERT J {pg. B 3.6-90}

The only credited safety function of the SGT System is to provide a secondary containment vacuum sufficient to assure that discharges from the secondary containment will be through the Main Stack. The VFTP test 5.5.7.d. provides verification that the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is acceptable. SR 3.6.4.1.3 and SR 3.6.4.1.4 provide assurance that sufficient vacuum in the secondary containment is established with the time period as used in the DBA LOCA analysis.

INSERT K {pg. B 3.7-16}

Additionally, the MCREV System is designed to maintain the control room environment for a 30-day occupancy after a DBA without exceeding 5 rem TEDE.

INSERT L {pg. B 3.7-16}

The MCREV System is credited as operating following a loss of coolant accident. The MCREV System is not credited in the analysis of the fuel handling accident, the main steam line break, or the control rod drop accident,

INSERT M {pg B 3.6-74}

Secondary containment is only required to be OPERABLE during handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

INSERT N {pg B 3.6-87}

The SGT System is only required to be OPERABLE during handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

INSERT P {pg B 3.6-79}

SCIVs are only required to be OPERABLE during handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

INSERT Q {pg B 3.8-40}

involving recently irradiated fuel. 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 24 hours)

INSERT R {PG B 3.8-42, 43, 72, 73, 74, 94, and 95}

involving recently irradiated fuel

INSERT S {pg B 3.8-94}

AC and DC electrical power 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 24 hours).

INSERT T {pg B 3.8-74}

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 24 hours).

INSERT U {pg B 3.6-75}

, since the movement of recently irradiated fuel can only be performed in MODES 4 and 5.

INSERT V {pg B 3.8-44, 74}

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply since the movement of recently irradiated fuel can only be performed in MODES 4 and 5.

INSERT W {pg B 3.3-174}

The Functions are only required to be OPERABLE during handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

INSERT X {pg B 3.3-182}

The MCREV System is only required to be OPERABLE during handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

INSERT Y {pg B 3.1-40}

The sodium pentaborate solution in the SLC System is also used, post-LOCA, to maintain ECCS fluid pH above 7. The system parameters used in the calculation are the Boron-10 minimum mass of 162.7 lbm, and an upper bound Boron-10 enrichment of 65%.

BASES (continued)

SAFETY LIMITS

The reactor core SLs are established to protect the integrity of the fuel clad barrier to the release of radioactive materials to the environs. SL 2.1.1.1 and SL 2.1.1.2 ensure that the core operates within the fuel design criteria. SL 2.1.1.3 ensures that the reactor vessel water level is greater than the top of the active irradiated fuel in order to prevent elevated clad temperatures and resultant clad perforations.

APPLICABILITY

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

SAFETY LIMIT VIOLATIONS

2.2.1

If any SL is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. 5).

10 CFR 50.67, "Accident Source Term,"

2.2.2

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

2.2.3

If any SL is violated, the senior management of the nuclear plant and the utility shall be notified within 24 hours. The 24 hour period provides time for plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to the senior management.

(continued)

BASES

SAFETY LIMIT VIOLATIONS (continued)

2.2.4

If any SL is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Ref. 7). A copy of the report shall also be provided to the senior management of the nuclear plant and the utility.

2.2.5

If any SL is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.

REFERENCES

1. GE Nuclear Energy 23A7188, Revision 1, September 1992.
2. ABB Atom Report BR 90-004, October 1990.
3. ANF-90-133 (P), Revision 2, August 1992.
4. NEDE-24011-P-A-10, February 1991.
5. 10 CFR 50.72.
6. ~~10 CFR 100.~~ 10 CFR 50.67
7. 10 CFR 50.73.

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on reactor steam dome pressure protects 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. Establishing an upper limit on reactor steam dome pressure ensures continued RCS integrity with regard to pressure excursions. Per the UFSAR (Ref. 1), the reactor coolant pressure boundary (RCPB) shall be designed with sufficient margin to ensure that the design conditions are not exceeded during normal operation and abnormal operational transients.

During normal operation and abnormal operational transients, 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 are hydrostatically tested at 125% of design pressure, in accordance with ASME Code requirements, prior to initial operation when there is no fuel in the core. Any further hydrostatic testing with fuel in the core may be done under LCO 3.10.1, "Inservice Leak and Hydrostatic Testing Operation." 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 reducing the number of protective barriers designed to prevent radioactive releases from exceeding the limits specified in ~~10 CFR 100, "Reactor Site Criteria"~~ (Ref. 4). If this occurred in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere.

APPLICABLE SAFETY ANALYSES

The RCS safety/relief valves and the Reactor Protection System Reactor Pressure—High Function have settings established to ensure that the RCS pressure SL will not be exceeded.

(continued)

10 CFR 50.67, "Accident Source Term,"

BASES

SAFETY LIMIT VIOLATIONS (continued)

2.2.2

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of ~~10 CFR 100, "Reactor Site Criteria,"~~ 10 CFR 50.67, "Accident Source Term," limits (Ref. 4). Therefore, it is required to insert all insertable control rods and restore compliance with the SL within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also assures that the probability of an accident occurring during the period is minimal.

2.2.3

If any SL is violated, the senior management of the nuclear plant and the utility, shall be notified within 24 hours. The 24 hour period provides time for plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to the senior management.

2.2.4

If any SL is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Ref. 8). A copy of the report shall also be provided to the senior management of the nuclear plant and the utility.

2.2.5

If any SL is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.

REFERENCES

1. UFSAR, Section 1.5.2.2.
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000.

(continued)

BASES

REFERENCES (continued)

3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IW-5000.

4. 10 CFR ~~100~~ 50.67.

5. ASME, Boiler and Pressure Vessel Code, Section III, 1965 Edition, including Addenda to winter of 1965.

6. ASME, Boiler and Pressure Vessel Code, Section III, 1980 Edition, Addenda to winter of 1981.

7. 10 CFR 50.72.

8. 10 CFR 50.73.

BASES

ACTIONS

B.1 and B.2 (continued)

control rods has less impact on control rod worth than withdrawals have. Required Action B.1 is modified by a Note which allows the RWM to be bypassed to allow the affected control rods to be returned to their correct position.

LCO 3.3.2.1 requires verification of control rod movement by a second licensed operator or a qualified member of the technical staff.

When nine or more OPERABLE control rods are not in compliance with BPWS, the reactor mode switch must be placed in the shutdown position within 1 hour. With the mode switch in shutdown, the reactor is shut down, and as such, does not meet the applicability requirements of this LCO. The allowed Completion Time of 1 hour is reasonable to allow insertion of control rods to restore compliance, and is appropriate relative to the low probability of a CRDA occurring with the control rods out of sequence.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

The control rod pattern is verified to be in compliance with the BPWS at a 24 hour Frequency to ensure the assumptions of the CRDA analyses are met. The 24 hour Frequency was developed considering that the primary check on compliance with the BPWS is performed by the RWM (LCO 3.3.2.1), which provides control rod blocks to enforce the required sequence and is required to be OPERABLE when operating at $\leq$ 10% RTP.

REFERENCES

1. NEDE-24011-P-A-10-US, "General Electric Standard Application for Reactor Fuel, Supplement for United States," Section 2.2.3.1, February 1991.
2. Letter (BWROG-8644) from T. Pickens (BWROG) to G. C. Lainas (NRC), "Amendment 17 to General Electric Licensing Topical Report NEDE-24011-P-A."
3. UFSAR, Section 14.6.2.3.
4. NUREG-0800, Section 15.4.9, Revision 2, July 1981.
5. 10 CFR ~~100.11.~~ 50.67

(continued)

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Standby Liquid Control (SLC) System

BASES

BACKGROUND

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

INSERT B

Reference 1 requires a SLC System with a minimum flow capacity and boron content equivalent in control capacity to 86 gpm of 13 weight percent sodium pentaborate solution. Natural sodium pentaborate solution is 19.8% atom Boron-10. Therefore, the system parameters of concern, boron concentration (C), SLC pump flow rate (Q), and Boron-10 enrichment (E), may be expressed as a multiple of ratios. The expression is as follows:

$$\frac{C}{13\%\ \text{weight}} \times \frac{Q}{86\ \text{gpm}} \times \frac{E}{19.8\%\ \text{atom}}$$

If the product of this expression is ≥ 1, then the SLC System satisfies the criteria of Reference 1. As such, the equation forms the basis for acceptance criteria for the surveillances of concentration, flow rate, and boron enrichment and is presented in Table 3.1.7-1.

The SLC System consists of a boron solution storage tank, two positive displacement pumps, two explosive valves that are provided in parallel for redundancy, and associated piping and valves used to transfer borated water from the storage tank to the reactor pressure vessel (RPV). The borated solution is discharged near the bottom of the core shroud, where it then mixes with the cooling water rising through the core. A smaller tank containing demineralized water is provided for testing purposes.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The SLC System is manually initiated from the main control room, as directed by the emergency operating procedures, if the operator believes the reactor cannot be shut down, or kept shut down, with the control rods. The SLC System is used in the event that enough control rods cannot be inserted to accomplish shutdown and cooldown in the normal manner. The SLC System injects borated water into the reactor core to add negative reactivity to compensate for all of the various reactivity effects that could occur during plant operations. To meet this objective, it is necessary to inject a quantity of boron, which produces a concentration of 660 ppm of natural boron, in the reactor coolant at 68°F. To allow for potential leakage and imperfect mixing in the reactor system, an additional amount of boron equal to 25% of the amount cited above is added (Ref. 2). The minimum mass of Boron-10 (162.7 lbm) needed for injection is calculated such that the required quantity is achieved accounting for dilution in the RPV with normal water level and including the water volume in the residual heat removal shutdown cooling piping and in the recirculation loop piping. This quantity of borated solution is the amount that is above the pump suction shutoff level in the boron solution storage tank. No credit is taken for the portion of the tank volume that cannot be injected. The maximum concentration of sodium pentaborate listed in Table 3.1.7-1 has been established to ensure that the solution saturation temperature does not exceed 43°F.

INSERT Y

The SLC System satisfies Criterion 4 of the NRC Policy Statement.

LCO

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

(continued)

BASES (continued)

APPLICABILITY

INSERT C

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

ACTIONS

A.1 and A.2

If the boron solution concentration is > 9.82% weight but the concentration and temperature of boron in solution and pump suction piping temperature are within the limits of Figure 3.1.7-1, operation is permitted for a limited period since the SLC subsystems are capable of performing the intended function. It is not necessary under these conditions to declare both SLC subsystems inoperable since the SLC subsystems are capable of performing their intended function.

The concentration and temperature of boron in solution and pump suction piping temperature must be verified to be within the limits of Figure 3.1.7-1 within 8 hours and once per 12 hours thereafter (Required Action A.1). The temperature versus concentration curve of Figure 3.1.7-1 ensures a 10°F margin will be maintained above the saturation temperature. This verification ensures that boron does not precipitate out of solution in the storage tank or in the pump suction piping due to low boron solution temperature (below the saturation temperature for the given concentration). The Completion Time for performing Required Action A.1 is considered acceptable given the low probability of a Design Basis Accident (DBA) or transient occurring concurrent with the failure of the control rods to shut down the reactor and operating experience which has shown there are relatively slow variations in the measured parameters of concentration and temperature over these time periods.

(continued)

BASES

ACTIONS

D.1 (continued) and MODE 4 within 36 hours

Times are

brought to MODE 3 within 12 hours. The allowed Completion ~~Time of 12 hours is~~ reasonable, based on operating experience, to reach ~~MODE 3~~ from full power conditions in an orderly manner and without challenging plant systems.

the required MODES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1, SR 3.1.7.2, and SR 3.1.7.3

SR 3.1.7.1 through SR 3.1.7.3 are 24 hour Surveillances verifying certain characteristics of the SLC System (e.g., the level and temperature of the borated solution in the storage tank), thereby ensuring SLC System OPERABILITY without disturbing normal plant operation. These Surveillances ensure that the proper borated solution level and temperature, including the temperature of the pump suction piping, are maintained. Maintaining a minimum specified borated solution temperature is important in ensuring that the boron remains in solution and does not precipitate out in the storage tank or in the pump suction piping. The temperature limit specified in SR 3.1.7.2 and SR 3.1.7.3 and the maximum sodium pentaborate concentration specified in Table 3.1.7-1 ensures that a 10°F margin will be maintained above the saturation temperature. Control room alarms for low SLC storage tank temperature and low SLC System piping temperature are available and are set at 55°F. As such, SR 3.1.7.2 and SR 3.1.7.3 may be satisfied by verifying the absence of low temperature alarms for the SLC storage tank and SLC System piping. The 24 hour Frequency is based on operating experience and has shown there are relatively slow variations in the measured parameters of level and temperature.

SR 3.1.7.4 and SR 3.1.7.6

SR 3.1.7.4 verifies the continuity of the explosive charges in the injection valves to ensure that proper operation will occur if required. Other administrative controls, such as those that limit the shelf life of the explosive charges, must be followed. The 31 day Frequency is based on operating experience and has demonstrated the reliability of the explosive charge continuity.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.9 (continued)

Surveillance when performed at the 24 month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.1.7.10

Enriched sodium pentaborate solution is made by mixing granular, enriched sodium pentaborate with water. In order to ensure the proper B-10 atom percentage (in accordance with Table 3.1.7-1) is being used, calculations must be performed to verify the actual B-10 enrichment within 8 hours after addition of the solution to the SLC tank. The calculations may be performed using the results of isotopic tests on the granular sodium pentaborate or vendor certification documents. The Frequency is acceptable considering that boron enrichment is verified during the procurement process and any time boron is added to the SLC tank.

REFERENCES

1. 10 CFR 50.62.
2. UFSAR, Section 3.8.4.
3. 10 CFR 50.67.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.a. Reactor Vessel Water Level—Low Low Low (Level 1) (continued)

The Reactor Vessel Water Level—Low Low Low (Level 1) Allowable Value is chosen to be the same as the ECCS Level 1 Allowable Value (LCO 3.3.5.1) to ensure that the MSLs isolate on a potential loss of coolant accident (LOCA) to prevent offsite doses from exceeding 10 CFR ~~100~~ limits. 50.67

This Function isolates MSIVs, MSL drains, MSL sample lines and recirculation loop sample line valves.

1.b. Main Steam Line Pressure—Low

Low MSL pressure indicates that there may be a problem with the turbine pressure regulation, which could result in a low reactor vessel water level condition and the RPV cooling down more than 100°F/hr if the pressure loss is allowed to continue. The Main Steam Line Pressure—Low Function is directly assumed in the analysis of the pressure regulator failure (Ref. 3). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hr) is not reached. In addition, this Function supports actions to ensure that Safety Limit 2.1.1.1 is not exceeded. (This Function closes the MSIVs prior to pressure decreasing below 785 psig, which results in a scram due to MSIV closure, thus reducing reactor power to < 25% RTP.)

The MSL low pressure signals are initiated from four transmitters that are connected to the MSL header. The transmitters are arranged such that, even though physically separated from each other, each transmitter is able to detect low MSL pressure. Four channels of Main Steam Line Pressure—Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be high enough to prevent excessive RPV depressurization.

The Main Steam Line Pressure—Low Function is only required to be OPERABLE in MODE 1 since this is when the assumed transient can occur (Ref. 1).

This Function isolates MSIVs, MSL drains, MSL sample lines and recirculation loop sample line valves.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

1.c. Main Steam Line Flow—High

Main Steam Line Flow—High is provided to detect a break of the MSL and to initiate closure of the MSIVs. If the steam were allowed to continue flowing out of the break, the reactor would depressurize and the core could uncover. If the RPV water level decreases too far, fuel damage could occur. Therefore, the isolation is initiated on high flow to prevent or minimize core damage. The Main Steam Line Flow—High Function is directly assumed in the analysis of the main steam line break (MSLB) (Ref. 3). The isolation action, along with the scram function of the Reactor Protection System (RPS), ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46 and offsite doses do not exceed the ~~10 CFR 100~~ 10 CFR 50.67 limits.

The MSL flow signals are initiated from 16 transmitters that are connected to the four MSLs. The transmitters are arranged such that, even though physically separated from each other, all four connected to one MSL would be able to detect the high flow. Four channels of Main Steam Line Flow—High Function for each MSL (two channels per trip system) are available and are required to be OPERABLE so that no single instrument failure will preclude detecting a break in any individual MSL.

The Allowable Value is chosen to ensure that offsite dose limits are not exceeded due to the break.

This Function isolates MSIVs, MSL drains, MSL sample lines and recirculation loop sample line valves.

1.d. Main Steam Line—High Radiation

The Main Steam Line—High Radiation Function is provided to detect gross release of fission products from the fuel and to initiate closure of the MSIVs. The trip setting is set low enough so that a high radiation trip results from a design basis rod drop accident and high enough above background radiation levels in the vicinity of the main steam lines so that spurious trips at rated power are avoided. The Main Steam Line—High Radiation Function is directly assumed in the analysis of the control rod drop accident (Ref. 3).

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.d. Main Steam Line—High Radiation (continued)

The Main Steam Line—High Radiation signals are initiated from four gamma sensitive instruments. Four channels are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value is chosen to ensure that offsite dose limits are not exceeded.

This Function isolates MSIVs, MSL drains, MSL sample lines and recirculation loop sample line valves.

1.e. Main Steam Tunnel Temperature—High

The Main Steam Tunnel Temperature Function is provided to detect a break in a main steam line and provides diversity to the high flow instrumentation.

Main Steam Tunnel Temperature signals are initiated from resistance temperature detectors (RTDs) located along the main steam line between the drywell wall and the turbine. Sixteen channels of Main Steam Tunnel Temperature—High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value is chosen to detect a leak equivalent to between 1% and 10% rated steam flow.

This Function isolates MSIVs, MSL drains, MSL sample lines and recirculation loop sample line valves.

Primary Containment Isolation

2.a. Reactor Vessel Water Level—Low (Level 3)

Low RPV water level indicates that the capability to cool the fuel may be threatened. The valves whose penetrations communicate with the primary containment are isolated to limit the release of fission products. The isolation of the primary containment on Level 3 supports actions to ensure that offsite dose limits of 10 CFR ~~100~~ 50.67 are not exceeded.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.a. Reactor Vessel Water Level—Low (Level 3) (continued)

The Reactor Vessel Water Level—Low (Level 3) Function associated with isolation is implicitly assumed in the UFSAR analysis as these leakage paths are assumed to be isolated post LOCA.

Reactor Vessel Water Level—Low (Level 3) signals are initiated from level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level—Low (Level 3) Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level—Low (Level 3) Allowable Value was chosen to be the same as the RPS Level 3 scram Allowable Value (LCO 3.3.1.1), since isolation of these valves is not critical to orderly plant shutdown.

This Function isolates the Group II(A) valves listed in Reference 1 with the exception of RWCU isolation valves and RHR shutdown cooling pump suction valves which are addressed in Functions 5.c and 6.b, respectively.

2.b. Drywell Pressure—High

50.67

High drywell pressure can indicate a break in the RCPB inside the primary containment. The isolation of some of the primary containment isolation valves on high drywell pressure supports actions to ensure that offsite dose limits of 10 CFR 100 are not exceeded. The Drywell Pressure—High Function, associated with isolation of the primary containment, is implicitly assumed in the UFSAR accident analysis as these leakage paths are assumed to be isolated post LOCA.

High drywell pressure signals are initiated from pressure transmitters that sense the pressure in the drywell. Four channels of Drywell Pressure—High are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

5.a. RWCU Flow—High (continued)

The high RWCU flow signals are initiated from transmitters that are connected to the pump suction line of the RWCU System. Two channels of RWCU Flow—High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The RWCU Flow—High Allowable Value ensures that a break of the RWCU piping is detected.

This Function isolates the inboard and outboard RWCU pump suction penetration and the outboard valve at the RWCU connection to reactor feedwater.

5.b. Standby Liquid Control (SLC) System Initiation

The isolation of the RWCU System is required when the SLC System has been initiated to prevent dilution and removal of the boron solution by the RWCU System (Ref. 5). SLC System initiation signals are initiated from the remote SLC System start switch.

There is no Allowable Value associated with this Function since the channels are mechanically actuated based solely on the position of the SLC System initiation switch.

INSERT O

Two channels of the SLC System Initiation Function are available and are required to be OPERABLE only in MODES 1 and 2, since these are the only MODES where the reactor can be critical~~, and these~~ MODES are consistent with the Applicability for the SLC System (LCO 3.1.7).

This Function isolates the inboard and outboard RWCU pump suction penetration and the outboard valve at the RWCU connection to reactor feedwater.

5.c. Reactor Vessel Water Level—Low (Level 3)

Low RPV water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, isolation of some interfaces with the reactor vessel occurs to isolate the potential sources of a break. The isolation of the RWCU System on Level 3 supports actions to ensure that the fuel

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3, 4. Reactor Building Ventilation and Refueling Floor Ventilation Exhaust Radiation—High (continued)

channels of Reactor Building Ventilation Exhaust Radiation—High Function and four channels of Refueling Floor Ventilation Exhaust Radiation—High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Values are chosen to promptly detect gross failure of the fuel cladding.

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

ACTIONS

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The Control Room Air Intake Radiation—High Function consists of four independent monitors. Two channels of Control Room Air Intake Radiation—High per trip system are available and are required to be OPERABLE to ensure that no single instrument failure can preclude MCREV System initiation. The Allowable Value was selected to ensure protection of the control room personnel.

The Control Room Air Intake Radiation—High Function is required to be OPERABLE in MODES 1, 2, and 3 and during ~~CORE ALTERATIONS,~~ OPDRVs, and movement of recently irradiated fuel assemblies in the secondary containment, to ensure that control room personnel are protected during a LOCA, fuel handling event, or vessel draindown event. INSERT X During MODES 4 and 5, when these specified conditions are not in progress (e.g., ~~CORE ALTERATIONS~~ OPDRVs), the probability of a LOCA or fuel damage is low; thus, the Function is not required.

ACTIONS

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

A.1 and A.2

Because of the redundancy of sensors available to provide initiation signals and the redundancy of the MCREV System design, an allowable out of service time of 6 hours has been shown to be acceptable (Ref. 4), to permit restoration of any inoperable channel to OPERABLE status. However, this out of service time is only acceptable provided the Control Room Air Intake Radiation—High Function is still maintaining MCREV System initiation capability. The Function is considered to be maintaining MCREV System

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 RCS Specific Activity

BASES

BACKGROUND

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

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

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

APPLICABLE SAFETY ANALYSES

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

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The limits on specific activity are values from a parametric evaluation of typical site locations. These limits are conservative because the evaluation considered more restrictive parameters than for a specific site, such as the location of the site boundary and the meteorological conditions of the site.

RCS specific activity satisfies Criterion 2 of the NRC Policy Statement.

LCO

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

APPLICABILITY

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

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

ACTIONS

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

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

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

A Note to the Required Actions of Condition A excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE(S) while relying on the ACTIONS even though the ACTIONS may eventually require plant shutdown. This exception is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the plant remains at, or proceeds to, power operation.

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

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

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

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

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

This Surveillance is performed to ensure iodine remains within limit during normal operation. The 7 day Frequency is adequate to trend changes in the iodine activity level.

This SR is modified by a Note that requires this Surveillance to be performed only in MODE 1 because the level of fission products generated in other MODES is much less.

REFERENCES

1. 10 CFR ~~100.11, 1973.~~ 50.67

2. UFSAR, Section 14.6.5.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.7 (continued)

position, since these valves were verified to be in the correct position prior to locking or securing. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves. The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.6.1.3.8

Verifying the isolation time of each power operated and each automatic PCIV is within limits is required to demonstrate OPERABILITY. MSIVs may be excluded from this SR since MSIV full closure isolation time is demonstrated by SR 3.6.1.3.9. The isolation time test ensures that the valve will isolate in a time period less than or equal to that assumed in the safety analyses. The isolation time is in accordance with Reference 2 or the requirements of the Inservice Testing Program which ever is more conservative. The Frequency of this SR is in accordance with the requirements of the Inservice Testing Program.

SR 3.6.1.3.9

50.67

Verifying that the isolation time of each MSIV is within the specified limits is required to demonstrate OPERABILITY. The isolation time test ensures that the MSIV will isolate in a time period that does not exceed the times assumed in the DBA analyses. This ensures that the calculated radiological consequences of these events remain within 10 CFR ~~100~~ limits. The Frequency of this SR is in accordance with the requirements of the Inservice Testing Program.

SR 3.6.1.3.10

Automatic PCIVs close on a primary containment isolation signal to prevent leakage of radioactive material from primary containment following a DBA. This SR ensures that each automatic PCIV will actuate to its isolation position on a primary containment isolation signal. The LOGIC SYSTEM

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.13

This SR ensures that in case the non-safety grade instrument air system is unavailable, the SGIG System will perform its design function to supply nitrogen gas at the required pressure for valve operators and valve seals supported by the SGIG System. The 24 month Frequency was developed considering it is prudent that this Surveillance be performed only during a plant outage. Operating experience has shown that these components will usually pass this Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.1.3.14

~~Leakage through each MSIV must be $\leq$ 11.5 scfh when tested at $\geq P_t$ (25 psig). The analyses in Reference 1 are based on treatment of MSIV leakage as a secondary containment bypass leakage, independent of a primary to secondary containment leakage analyzed at 1.27 L_a. In the Reference 1 analysis all 4 steam lines are assumed to leak at the TS Limit. This ensures that MSIV leakage is properly accounted for in determining the overall impacts of primary containment leakage. The Frequency is required by the Primary Containment Leakage Rate Testing Program.~~

INSERT A

SR 3.6.1.3.15

Verifying the opening of each 6 inch and 18 inch primary containment purge valve and each 18 inch primary containment exhaust valve is restricted by a blocking device to less than or equal to the required maximum opening angle specified in the UFSAR (Ref. 4) is required to ensure that the valves can close under DBA conditions within the times in the analysis of Reference 1. If a LOCA occurs, the purge and exhaust valves must close to maintain primary containment leakage within the values assumed in the accident analysis. At other times pressurization concerns are not present, thus the purge and exhaust valves can be fully open. The 24 month Frequency is appropriate because the blocking devices may be removed during a refueling outage.

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.1 Secondary Containment

BASES

BACKGROUND

INSERT E

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

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

APPLICABLE SAFETY ANALYSES

There ~~are two~~ is one principal accident~~s~~ for which credit is taken for secondary containment OPERABILITY. That is ~~These are~~ a loss of coolant accident (LOCA) (Ref. 1) ~~and a fuel handling accident inside secondary containment (Ref. 2)~~. The secondary containment performs no active function in response to ~~each of these~~ this limiting event~~s~~; however, its leak

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

tightness is required to ensure that fission products entrapped within the secondary containment structure will be treated by the SGT System prior to discharge to the environment.

collected

for

via the main stack.

Secondary containment satisfies Criterion 3 of the NRC Policy Statement.

LCO

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

discharged

APPLICABILITY

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

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

INSERT M

recently

ACTIONS

A.1

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

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

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

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

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

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

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

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1.1 and SR 3.6.4.1.2

Verifying that secondary containment equipment hatches and one access door in each access opening are closed ensures that the infiltration of outside air of such a magnitude as to prevent maintaining the desired negative pressure does not occur. Verifying that all such openings are closed provides adequate assurance that exfiltration from the secondary containment will not occur. In this application, the term "sealed" has no connotation of leak tightness. Maintaining secondary containment OPERABILITY requires verifying one door in the access opening is closed. An access opening contains one inner and one outer door. In some cases, secondary containment access openings are shared such that a secondary containment barrier may have multiple inner or multiple outer doors. The intent is to not breach secondary containment at any time when secondary containment is required. This is achieved by maintaining the inner or outer portion of the barrier closed at all times. However, all secondary containment access doors are normally kept closed, except when the access opening is being used for entry and exit or when maintenance is being performed on an access opening. The 31 day Frequency for these SRs has been shown to be adequate, based on operating experience, and is considered adequate in view of the other indications of door and hatch status that are available to the operator.

SR 3.6.4.1.3 and SR 3.6.4.1.4

INSERT F

~~The SGT System exhausts the secondary containment atmosphere to the environment through appropriate treatment equipment.~~

~~To ensure that fission products are treated, SR 3.6.4.1.3~~ verifies that the SGT System will rapidly establish and maintain a pressure in the secondary containment that is less than the pressure external to the secondary containment boundary. This is confirmed by demonstrating that one SGT subsystem will draw down the secondary containment to ≥ 0.25 inches of vacuum water gauge in ≤ ~~120 seconds~~ 15 minutes. This cannot be accomplished if the secondary containment boundary is not intact.

SR 3.6.4.1.4 demonstrates that one SGT subsystem can maintain ≥ 0.25 inches of vacuum water gauge for 1 hour at a flow rate ≤ 10,500 cfm. The 1 hour test period allows secondary containment to be in thermal equilibrium at steady

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.2 Secondary Containment Isolation Valves (SCIVs)

BASES

BACKGROUND

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

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

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

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

APPLICABLE SAFETY ANALYSES

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

(continued)

exhausted

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

via the main stack.

for elevated release

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

until discharged

via the main stack

SCIVs satisfy Criterion 3 of the NRC Policy Statement.

LCO

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

The power operated isolation valves are considered OPERABLE when their isolation times are within limits and the valves actuate on an automatic isolation signal. The valves covered by this LCO, along with their associated stroke times, are listed in Reference ~~3~~. 2

The normally closed isolation valves or blind flanges are considered OPERABLE when manual valves are closed or open in accordance with appropriate administrative controls, automatic SCIVs are de-activated and secured in their closed position, and blind flanges are in place. These passive isolation valves or devices are listed in Reference ~~3~~.

APPLICABILITY

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

In MODES 4 and 5, the probability and consequences of these events are reduced due to pressure and temperature limitations in these MODES. Therefore, maintaining SCIVs OPERABLE is not required in MODE 4 or 5, except for other situations under which significant radioactive releases can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs)~~, during CORE ALTERATIONS,~~ or during movement of recently irradiated fuel assemblies in the secondary containment. Moving irradiated fuel assemblies in the secondary containment may also occur in MODES 1, 2, and 3.

INSERT P

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

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

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

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

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

SURVEILLANCE REQUIREMENTS

SR 3.6.4.2.1

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.4.2.3 (continued)

under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components will usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section ~~14.6.3.~~ 14.9.2.1.

~~2. UFSAR, Section 14.6.4.~~

2. ~~3.~~ Technical Requirements Manual.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.3 Standby Gas Treatment (SGT) System

BASES

BACKGROUND

INSERT G

~~The SGT System is required by UFSAR design criteria (Ref. 1). The function of the SGT System is to ensure that radioactive materials that leak from the primary containment into the secondary containment following a Design Basis Accident (DBA) are filtered and adsorbed prior to exhausting to the environment.~~

A single SGT System is common to both Unit 2 and Unit 3 and consists of two fully redundant subsystems, each with its own set of ductwork, dampers, valves, charcoal filter train, and controls. Both SGT subsystems share a common inlet plenum. This inlet plenum is connected to the refueling floor ventilation exhaust duct for each Unit and to the suppression chamber and drywell of each Unit. Both SGT subsystems exhaust to the plant offgas stack through a common exhaust duct served by three 100% capacity system fans. SGT System fans 0AV020 and 0BV020 automatically start on Unit 2 secondary containment isolation signals. SGT System fans 0CV020 and 0BV020 automatically start on Unit 3 secondary containment isolation signals.

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

a. A demister or moisture separator;

b. An electric heater;

c. A prefilter;

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

e. A charcoal adsorber; and

f. A second HEPA filter.

INSERT H

The SGT System is sized such that each 100% capacity fan will provide a flow rate of 10,500 cfm at 20 inches water gauge static pressure to support the control of fission product releases. The SGT System is designed to restore and maintain secondary containment at a negative pressure of 0.25 inches water gauge relative to the atmosphere following

(continued)

Although not credited in any DBA analysis, the

BASES

BACKGROUND (continued)

the receipt of a secondary containment isolation signal. Maintaining this negative pressure is based upon the existence of calm wind conditions (up to 5 mph), a maximum SGT System flow rate of 10,500 cfm, outside air temperature of 95°F and a temperature of 150°F for air entering the SGT System from inside secondary containment.

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

The SGT System automatically starts and operates in response to actuation signals indicative of conditions or an accident that could require operation of the system. Following initiation, two charcoal filter train fans (0AV020 and 0BV020) start. Upon verification that both subsystems are operating, the redundant subsystem is normally shut down.

APPLICABLE SAFETY ANALYSES

INSERT I

~~The design basis for the SGT System is to mitigate the consequences of a loss of coolant accident and fuel handling accidents (Ref. 2). For all events analyzed, the SGT System is shown to be automatically initiated to reduce, via filtration and adsorption, the radioactive material released to the environment.~~

The SGT System satisfies Criterion 3 of the NRC Policy Statement.

LCO

Following a DBA, a minimum of one SGT subsystem is required to maintain the secondary containment at a negative pressure with respect to the environment and to process gaseous releases. Meeting the LCO requirements for two OPERABLE subsystems ensures operation of at least one SGT subsystem in the event of a single active failure.

(continued)

BASES

LCO (continued)

For Unit 2, one SGT subsystem is OPERABLE when ~~one charcoal filter train,~~ one fan (0AV020) and associated ductwork, dampers, valves, and controls are OPERABLE. The second SGT subsystem is OPERABLE when the other ~~charcoal filter train,~~ ~~one~~ fan (0BV020) and associated ductwork, damper, valves, and controls are OPERABLE.

APPLICABILITY

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

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

ACTIONS

A.1

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

B.1 and B.2

If the SGT subsystem cannot be restored to OPERABLE status within the required Completion Time in MODE 1, 2, or 3, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

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.

and C.2.2

recently

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

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

a significant amount of

recently

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

this activity

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

INSERT a

(continued)

BASES

ACTIONS (continued)

D.1

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

and

recently

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

When two SGT subsystems are inoperable, if applicable, ~~CORE ALTERATIONS and~~ movement of irradiated fuel assemblies in secondary containment must immediately be suspended.

this activity

Suspension of ~~these activities~~ shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must immediately be initiated to suspend OPDRVs in order to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

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

INSERT u

SURVEILLANCE REQUIREMENTS

SR 3.6.4.3.1

Operating each SGT subsystem (including each filter train fan) for ≥ 15 minutes ensures that both subsystems are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. ~~Operation with the heaters on (automatic heater cycling to maintain temperature) for ≥ 15 minutes every 31 days is sufficient to eliminate moisture on the adsorbers and HEPA filters since during idle periods instrument air is injected into the filter plenum to keep the filters dry.~~ The 31 day Frequency was developed in consideration of the known reliability of fan motors and controls and the redundancy available in the system.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.4.3.2

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

INSERT J

SR 3.6.4.3.3

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

REFERENCES

1. UFSAR, Section 1.5.1.6.
2. UFSAR, Section 14.9.

BASES

BACKGROUND (continued)

initiate an emergency shutdown of non-essential equipment and lighting to reduce the heat generation to a minimum. Heat removal would be accomplished by conduction through the floors, ceilings, and walls to adjacent rooms and to the environment. ~~Additionally, the MCREV System is designed to maintain the control room environment for a 30 day continuous occupancy after a DBA without exceeding 5 rem whole body dose.~~ A single MCREV subsystem will pressurize the control room to prevent infiltration of air from surrounding buildings. MCREV System operation in maintaining control room habitability is discussed in the UFSAR, Chapters 7, 10, and 12, (Refs. 1, 2, and 3, respectively).

INSERT K

APPLICABLE SAFETY ANALYSES

The ability of the MCREV System to maintain the habitability of the control room is an explicit assumption for the safety analyses presented in the UFSAR, Chapters 10 and 12 (Refs. 2 and 3, respectively). ~~The MCREV System is assumed to operate following a loss of coolant accident, fuel handling accident, main steam line break, and control rod drop accident,~~ as discussed in the UFSAR, Section 14.9.1.5 (Ref. 4). The radiological doses to control room personnel as a result of the various DBAs are summarized in Reference 4. No single active or passive failure will cause the loss of outside or recirculated air from the control room.

INSERT L

The MCREV System satisfies Criterion 3 of the NRC Policy Statement.

LCO

Two redundant subsystems of the MCREV System are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other subsystem. Total system failure could result in exceeding a dose of 5 rem TEDE to the control room operators in the event of a ~~DBA.~~ LOCA.

The MCREV System is considered OPERABLE when the individual components necessary to control operator exposure are OPERABLE in both subsystems. A subsystem is considered OPERABLE when its associated:

a. Fan is OPERABLE;

(continued)

BASES

LCO (continued)

b. HEPA filter 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 flow can be maintained.

In addition, the control room boundary must be maintained, including the integrity of the walls, floors, ceilings, and ductwork. Temporary seals may be used to maintain the boundary. In addition, an access door may be opened provided the ability to pressurize the control room is maintained and the capability exists to close the affected door in an expeditious manner.

APPLICABILITY

In MODES 1, 2, and 3, the MCREV System must be OPERABLE to control operator exposure during and following a DBA LOCA since the ~~DBA~~ LOCA could lead to a fission product release.

In MODES 4 and 5, the probability and consequences of a ~~DBA~~ are reduced because of the pressure and temperature limitations in these MODES. Therefore, maintaining the MCREV System OPERABLE is not required in MODE 4 or 5, except for the following situations under which significant radioactive releases can be postulated:

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

~~b. During CORE ALTERATIONS; and~~

~~c.~~ b. During movement of recently irradiated fuel assemblies in the secondary containment.

ACTIONS

A.1

With one MCREV subsystem inoperable, the inoperable MCREV subsystem must be restored to OPERABLE status within 7 days. With the unit in this condition, the remaining OPERABLE MCREV subsystem is adequate to maintain control room temperature and to perform control room radiation protection. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could

(continued)

BASES

ACTIONS

A.1 (continued)

result in reduced MCREV System capability. The 7 day Completion Time is based on the low probability of a ~~DBA~~ LOCA occurring during this time period, and that the remaining subsystem can provide the required capabilities.

B.1 and B.2

In MODE 1, 2, or 3, if the inoperable MCREV subsystem cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE that minimizes risk. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours and in MODE 4 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.

INSERT U

and

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

The Required Actions of Condition C are modified by a Note indicating that LCO 3.0.3 does not apply. ~~If moving irradiated fuel assemblies while in MODE 1, 2, or 3, 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.~~

recently

During movement of irradiated fuel assemblies in the secondary containment~~, during CORE ALTERATIONS,~~ or during OPDRVs, if the inoperable MCREV subsystem cannot be restored to OPERABLE status within the required Completion Time, the OPERABLE MCREV subsystem may be placed in operation. This action ensures that the remaining subsystem is OPERABLE, that no failures that would prevent automatic actuation will occur, and that any active failure will be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

(continued)

and

BASES

ACTIONS

C.1, C.2.1, C.2.2, ~~and C.2.3~~ (continued)

recently

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

this activity

D.1

If both MCREV subsystems are inoperable in MODE 1, 2, or 3, the MCREV System may not be capable of performing the intended function and the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

and

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

The Required Actions of Condition E are modified by a Note indicating that LCO 3.0.3 does not apply. ~~If moving irradiated fuel assemblies while in MODE 1, 2, or 3, 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.~~

INSERT U

During movement of irradiated fuel assemblies in the secondary containment, ~~during CORE ALTERATIONS,~~ or during OPDRVs, with two MCREV subsystems inoperable, action must be taken immediately to suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

recently

this activity

If applicable, ~~CORE ALTERATIONS and~~ movement of irradiated fuel assemblies in the secondary containment must be suspended immediately. Suspension of ~~these activities~~ shall not preclude completion of movement of a component to a safe position. If applicable, actions must be initiated

(continued)

and

BASES

ACTIONS

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

immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

This SR verifies that a subsystem in a standby mode starts on demand and continues to operate for ≥ 15 minutes. Standby systems should be checked periodically to ensure that they start and function properly. As the environmental and normal operating conditions of this system are not severe, testing each subsystem once every month provides an adequate check on this system. Furthermore, the 31 day Frequency is based on the known reliability of the equipment and the two subsystem redundancy available.

SR 3.7.4.2

This SR verifies that the required MCREV testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.4.3

This SR verifies that on an actual or simulated initiation signal, each MCREV subsystem starts and operates. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.7.1.4 overlaps this SR to provide complete testing of the safety function. Operating experience has shown that these components will usually pass the SR when performed at the 24 month Frequency. Therefore, this Frequency is concluded to be acceptable from a reliability standpoint.

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.5 Main Condenser Offgas

BASES

BACKGROUND

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

The Main Condenser Offgas System has been incorporated into the unit design to reduce the gaseous radwaste emission. This system uses a catalytic recombiner to recombine radiolytically dissociated hydrogen and oxygen. The gaseous mixture is cooled and water vapor removed by the offgas recombiner condenser; the remaining water and condensibles are stripped out by the cooler condenser and moisture separator. The remaining gaseous mixture (i.e., the offgas recombiner effluent) is then processed by a charcoal adsorber bed prior to release.

APPLICABLE SAFETY ANALYSES

The main condenser offgas gross gamma activity rate is an initial condition of the Main Condenser Offgas System failure event, discussed in the UFSAR, Section 9.4.5 (Ref. 1). The analysis assumes a gross failure in the Main Condenser Offgas System that results in the rupture of the Main Condenser Offgas System pressure boundary. The gross gamma activity rate is controlled to ensure that, during the event, the calculated offsite doses will be well within the limits of 10 CFR ~~100~~ (Ref. 2) or the NRC staff approved licensing basis.

50.67

The main condenser offgas limits satisfy Criterion 2 of the NRC Policy Statement.

LCO

To ensure compliance with the assumptions of the Main Condenser Offgas System failure event (Ref. 1), the fission product release rate should be consistent with a noble gas release to the reactor coolant of 100 μCi/MWt-second after decay of 30 minutes. The LCO is established consistent

(continued)

BASES

ACTIONS

B.1, B.2, B.3.1, and B.3.2 (continued)

experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

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

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

REFERENCES

1. UFSAR, Section 9.4.5.
2. 10 CFR ~~100.~~ 50.67.

B 3.7 PLANT SYSTEMS

B 3.7.7 Spent Fuel Storage Pool Water Level

BASES

BACKGROUND

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

A general description of the spent fuel storage pool design is found in the UFSAR, Section 10.3 (Ref. 1). The assumptions of the fuel handling accident are found in the UFSAR, Section 14.6.4 (Ref. 2).

APPLICABLE SAFETY ANALYSES

The water level above the irradiated fuel assemblies is an implicit assumption of the fuel handling accident. A fuel handling accident is evaluated to ensure that the radiological consequences ~~(calculated whole body and thyroid doses at the site boundary)~~ are well below the guidelines set forth in 10 CFR ~~100~~ 50.67 (Ref. 3). A fuel handling accident could release a fraction of the fission product inventory by breaching the fuel rod cladding as discussed in Reference 2.

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

The spent fuel storage pool water level satisfies Criteria 2 and 3 of the NRC Policy Statement.

LCO

The specified water level (232 ft 3 inches plant elevation, which is equivalent to 22 ft over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks) preserves the assumptions of the fuel handling accident analysis (Ref. 2). As such, it is the minimum required for fuel movement within the spent fuel storage pool.

(continued)

BASES (continued)

APPLICABILITY

This LCO applies during movement of fuel assemblies in the spent fuel storage 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. If moving fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of fuel assemblies is not a sufficient reason to require a reactor shutdown.

When the initial conditions for an accident cannot be met, action must be taken to preclude the accident from occurring. If the spent fuel storage pool level is less than required, the movement of fuel assemblies in the spent fuel storage pool is suspended immediately. Suspension of this activity shall not preclude completion of movement of a fuel assembly to a safe position. This effectively precludes a spent fuel handling accident from occurring.

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

This SR verifies that sufficient water is available in the event of a fuel handling accident. The water level in the spent fuel storage pool must be checked periodically. The 7 day Frequency is acceptable, based on operating experience, considering that the water volume in the pool is normally stable, and all water level changes are controlled by unit procedures.

REFERENCES

1. UFSAR, Section 10.3.
2. UFSAR, Section 14.6.4.
3. 10 CFR ~~100.~~ 50.67.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.3 (continued)

Note 1 modifies this Surveillance to indicate that diesel engine runs for this Surveillance may include gradual loading, as recommended by the manufacturer, so that mechanical stress and wear on the diesel engine are minimized.

Note 2 modifies this Surveillance by stating that momentary transients because of changing bus loads do not invalidate this test. Similarly, momentary power factor transients above the limit do not invalidate the test.

Note 3 indicates that this Surveillance should be conducted on only one DG at a time in order to avoid common cause failures that might result from offsite circuit or grid perturbations.

Note 4 stipulates a prerequisite requirement for performance of this SR. A successful DG start must precede this test to credit satisfactory performance.

To minimize testing of the DGs, Note 5 allows a single test (instead of two tests, one for each unit) to satisfy the requirements for both units, with the DG synchronized to the 4 kV emergency bus of Unit 2 for one periodic test and synchronized to the 4 kV emergency bus of Unit 3 during the next periodic test. This is allowed since the main purpose of the Surveillance, to ensure DG OPERABILITY, is still being verified on the proper frequency, and each unit's breaker control circuitry, which is only being tested every second test (due to the staggering of the tests), historically have a very low failure rate. Note 5 modifies the specified frequency for each unit's breaker control circuitry to be 62 days. If the DG fails one of these Surveillances, the DG should be considered inoperable on both units, unless the cause of the failure can be directly related to only one unit. In addition, if the test is scheduled to be performed on Unit 3, and the Unit 3 TS allowance that provides an exception to performing the test is used (i.e., when Unit 3 is in MODE 4 or 5, or moving

recently

irradiated fuel assemblies in the secondary containment, the Note to Unit 3 SR 3.8.2.1 provides an exception to performing this test) or if it is not preferable to perform the test on a unit due to operational concerns (however time is not to exceed 62 days plus grace), then the test shall be performed synchronized to the Unit 2 4 kV emergency bus.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.20 (continued)

The 10 year Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 8). This SR is modified by two Notes. The reason for Note 1 is to minimize wear on the DG during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations. To minimize testing of the DGs, Note 2 allows a single test (instead of two tests, one for each unit) to satisfy the requirements for both units. This is allowed since the main purpose of the Surveillance can be met by performing the test on either unit. If a DG fails one of these Surveillances, a DG should be considered inoperable on both units, unless the cause of the failure can be directly related to only one unit.

SR 3.8.1.21

With the exception of this Surveillance, all other Surveillances of this Specification (SR 3.8.1.1 through SR 3.8.1.20) are applied only to the Unit 2 AC sources. This Surveillance is provided to direct that the appropriate Surveillances for the required Unit 3 AC sources are governed by the applicable Unit 3 Technical Specifications. Performance of the applicable Unit 3 Surveillances will satisfy Unit 3 requirements, as well as satisfying this Unit 2 Surveillance Requirement. Six exceptions are noted to the Unit 3 SRs of LCO 3.8.1. SR 3.8.1.8 is excepted when only one Unit 3 offsite circuit is required by the Unit 2 Specification, since there is not a second circuit to transfer to. SR 3.8.1.12, SR 3.8.1.13, SR 3.8.1.17, SR 3.8.1.18 (ECCS load block requirements only), and SR 3.8.1.19 are excepted since these SRs test the Unit 3 ECCS initiation signal, which is not needed for the AC sources to be OPERABLE on Unit 2.

The Frequency required by the applicable Unit 3 SR also governs performance of that SR for Unit 2.

As Noted, if Unit 3 is in MODE 4 or 5, or moving recently irradiated fuel assemblies in the secondary containment, the Note to Unit 3 SR 3.8.2.1 is applicable. This ensures that a Unit 2 SR will not require a Unit 3 SR to be performed, when the

(continued)

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 4 and 5 and during movement of irradiated fuel assemblies in secondary containment ensures that: recently

a. The facility 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 an inadvertent draindown of the vessel or a fuel handling accident. INSERT Q

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 loss of 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, and 3 have no specific analyses in MODES 4 and 5. Worst case bounding events are deemed not credible in MODES 4 and 5 because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and corresponding stresses result in the probabilities of occurrences significantly reduced or eliminated, and 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, and 3, various deviations from the analysis assumptions and design requirements are allowed within the ACTIONS. This allowance is in recognition that

(continued)

BASES

LCO (continued)

offsite circuit. In addition, some equipment that may be required by Unit 2 is powered from Unit 3 sources (e.g., Standby Gas Treatment (SGT) System). Therefore, one qualified circuit between the offsite transmission network and the Unit 3 onsite Class 1E AC electrical power distribution subsystem(s), and one DG (not necessarily a different DG than those being used to meet LCO 3.8.2.b requirements) capable of supplying power to one of the required Unit 3 subsystems of each of the required components must also be OPERABLE. Together, OPERABILITY of the required offsite circuit(s) and required DG(s) ensures the availability of sufficient AC sources to operate the plant in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents, INSERT R and reactor vessel draindown).

The qualified Unit 2 offsite circuit must be capable of maintaining rated frequency and voltage while connected to the respective Unit 2 4 kV emergency bus(es), and of accepting required loads during an accident. Qualified offsite circuits are those that are described in the UFSAR, Technical Specification Bases Section 3.8.1 and are part of the licensing basis for the unit. A Unit 2 offsite circuit consists of the incoming breaker and disconnect to the startup and emergency auxiliary transformer, the respective circuit path to the emergency auxiliary transformer, and the circuit path to the Unit 2 4 kV emergency buses required by LCO 3.8.8, including feeder breakers to the required Unit 2 4 kV emergency buses. A qualified Unit 3 offsite circuit's requirements are the same as the Unit 2 circuit's requirements, except that the circuit path, including the feeder breakers, is to the Unit 3 4 kV emergency buses required to be OPERABLE by LCO 3.8.8.

The required DGs must be capable of starting, accelerating to rated speed and voltage, and connecting to their respective Unit 2 emergency bus on detection of bus undervoltage. This sequence must be accomplished within 10 seconds. Each DG must also be capable of accepting required loads within the assumed loading sequence intervals, and must continue to operate until offsite power can be restored to the 4 kV emergency buses. These capabilities are required to be met from a variety of initial conditions such as DG in standby with engine hot and DG in standby with engine at ambient conditions. Additional

(continued)

BASES

LCO (continued)

DG capabilities must be demonstrated to meet required Surveillances, e.g., capability of the DG to revert to standby status on an ECCS signal while operating in parallel test mode. Proper sequencing of loads is a required function for DG OPERABILITY. The necessary portions of the Emergency Service Water System are also required to provide appropriate cooling to each required DG.

The OPERABILITY requirements for the DG capable of supplying power to the Unit 3 powered equipment are the same as described above, except that the required DG must be capable of connecting to its respective Unit 3 4 kV emergency bus. (In addition, the Unit 3 ECCS initiation logic SRs are not applicable, as described in SR 3.8.2.2 Bases.)

It is acceptable for 4 kV emergency buses to be cross tied during shutdown conditions, permitting a single offsite power circuit to supply all required buses. No automatic transfer capability is required for offsite circuits to be considered OPERABLE.

recently

APPLICABILITY

The AC sources are required to be OPERABLE in MODES 4 and 5 and during movement of irradiated fuel assemblies in the secondary containment to provide assurance that:

recently

a. Systems providing adequate coolant inventory makeup are available for the irradiated fuel assemblies in the core in case of an inadvertent draindown of the reactor vessel;

INSERT R

b. Systems needed to mitigate a fuel handling accident are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

AC power requirements for MODES 1, 2, and 3 are covered in LCO 3.8.1.

(continued)

BASES (continued)

ACTIONS

INSERT V

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

A.1 and B.1

recently irradiated

With one or more required offsite circuits inoperable, or with one DG inoperable, the remaining required sources may be capable of supporting sufficient required features (e.g., system, subsystem, division, component, or device) to allow continuation of CORE ALTERATIONS, fuel movement, and operations with a potential for draining the reactor vessel. For example, if two or more 4 kV emergency buses are required per LCO 3.8.8, one 4 kV emergency bus with offsite power available may be capable of supplying sufficient required features. By the allowance of the option to declare required features inoperable that are not powered from offsite power (Required Action A.1) or capable of being powered by the required DG (Required Action B.1), appropriate restrictions can be implemented in accordance with the affected feature(s) LCOs' ACTIONS. Required features remaining powered from a qualified offsite power circuit, even if that circuit is considered inoperable because it is not powering other required features, are not declared inoperable by this Required Action. If a single DG is credited with meeting both LCO 3.8.2.d and one of the DG requirements of LCO 3.8.2.b, then the required features remaining capable of being powered by the DG are not declared inoperable by this Required Action, even if the DG is considered inoperable because it is not capable of powering other required features.

A.2.1, A.2.2, A.2.3, A.2.4, B.2.1, B.2.2, B.2.3, B.2.4, C.1, C.2, C.3, and C.4

With an offsite circuit not available to all required 4 kV emergency buses or one required DG inoperable, the option still exists to declare all required features inoperable

(continued)

BASES

ACTIONS

A.2.1, A.2.2, A.2.3, A.2.4, B.2.1, B.2.2, B.2.3, B.2.4, C.1, C.2, C.3, and C.4 (continued)

(per Required Actions A.1 and B.1). Since this option may involve undesired administrative efforts, the allowance for sufficiently conservative actions is made. With two or more required DGs inoperable, the minimum required diversity of AC power sources may not be available. It is, therefore, required to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies in the secondary containment, and activities that could result in inadvertent draining of the reactor vessel. recently

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required AC sources and to continue this action until restoration is accomplished in order to provide the necessary AC power to the plant safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required AC electrical power sources should be completed as quickly as possible in order to minimize the time during which the plant safety systems may be without sufficient power.

Pursuant to LCO 3.0.6, the Distribution System ACTIONS would not be entered even if all AC sources to it are inoperable, resulting in de-energization. Therefore, the Required Actions of Condition A have been modified by a Note to indicate that when Condition A is entered with no AC power to any required 4 kV emergency bus, ACTIONS for LCO 3.8.8 must be immediately entered. This Note allows Condition A to provide requirements for the loss of the offsite circuit whether or not a required bus is de-energized. LCO 3.8.8 provides the appropriate restrictions for the situation involving a de-energized bus.

SURVEILLANCE REQUIREMENTS

SR 3.8.2.1

SR 3.8.2.1 requires the SRs from LCO 3.8.1 that are necessary for ensuring the OPERABILITY of the Unit 2 AC sources in other than MODES 1, 2, and 3. SR 3.8.1.8 is not

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.4.8 (continued)

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required DC electrical power subsystem from service, perturb the electrical distribution system, and challenge safety systems. Credit may be taken for unplanned events that satisfy the Surveillance. The DC batteries of the other unit are exempted from this restriction since they are required to be OPERABLE by both units and the Surveillance cannot be performed in the manner required by the Note without resulting in a dual unit shutdown.

SR 3.8.4.9

With the exception of this Surveillance, all other Surveillances of this Specification (SR 3.8.4.1 through SR 3.8.4.8) are applied only to the Unit 2 DC electrical power subsystems. This Surveillance is provided to direct that the appropriate Surveillances for the required Unit 3 DC electrical power subsystems are governed by the Unit 3 Technical Specifications. Performance of the applicable Unit 3 Surveillances will satisfy Unit 3 requirements, as well as satisfying this Unit 2 Surveillance Requirement.

The Frequency required by the applicable Unit 3 SR also governs performance of that SR for Unit 2. As Noted, if Unit 3 is in MODE 4 or 5, or moving recently irradiated fuel assemblies in the secondary containment, the Note to Unit 3 SR 3.8.5.1 is applicable. This ensures that a Unit 2 SR will not require a Unit 3 SR to be performed, when the Unit 3 Technical Specifications exempts performance of a Unit 3 SR. (However, as stated in the Unit 3 SR 3.8.5.1 Note, while performance of the SR is exempted, the SR still must be met.)

REFERENCES

1. UFSAR, Chapter 14.
2. "Proposed IEEE Criteria for Class 1E Electrical Systems for Nuclear Power Generating Stations," June 1969.
3. IEEE Standard 485, 1983.

(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 UFSAR, Chapter 14 (Ref. 1), assume that Engineered Safety Feature systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the diesel generators (DGs), 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.

recently

The OPERABILITY of the minimum DC electrical power sources during MODES 4 and 5 and during movement of irradiated fuel assemblies in secondary containment ensures that:

a. The facility 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 an inadvertent draindown of the vessel or a fuel handling accident.

INSERT R

The DC sources satisfy Criterion 3 of the NRC Policy Statement.

LCO

The Unit 2 DC electrical power subsystems, with each DC subsystem consisting of two 125 V station batteries in series, two battery chargers (one per battery), and the corresponding control equipment and interconnecting cabling supplying power to the associated bus, are required to be

(continued)

BASES

LCO (continued)

OPERABLE to support Unit 2 DC distribution subsystems required OPERABLE by LCO 3.8.8, "Distribution Systems—Shutdown." When the equipment required OPERABLE: 1) does not require 250 VDC from the DC electrical power subsystem; and 2) does not require 125 VDC from one of the two 125 V batteries of the DC electrical power subsystem, the Unit 2 DC electrical power subsystem requirements can be modified to only include one 125 V battery (the battery needed to provide power to required equipment), an associated battery charger, and the corresponding control equipment and interconnecting cabling supplying 125 V power to the associated bus. This exception is allowed only if all 250 VDC loads are removed from the associated bus. In addition, DC control power (which provides control power for the 4 kV load circuit breakers and the feeder breakers to the 4 kV emergency bus) for two of the four 4 kV emergency buses, as well as control power for two of the diesel generators, is provided by the Unit 3 DC electrical power subsystems. Therefore, the Unit 3 DC electrical power subsystems needed to support required components are also required to be OPERABLE. The Unit 3 DC electrical power subsystem OPERABILITY requirements are the same as those required for a Unit 2 DC electrical power subsystem. In addition, battery chargers (Unit 2 and Unit 3) can be powered from the opposite unit's AC source (as described in the Background section of the Bases for LCO 3.8.4, "DC Sources—Operating"), and be considered OPERABLE for the purpose of meeting this LCO.

This requirement ensures the availability of sufficient DC electrical 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, and inadvertent reactor vessel draindown). INSERT R

APPLICABILITY

The DC electrical power sources required to be OPERABLE in MODES 4 and 5 and during movement of irradiated fuel assemblies in the secondary containment provide assurance that:

recently

a. Required features to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies in the core in case of an inadvertent draindown of the reactor vessel;

(continued)

BASES

APPLICABILITY (continued)

b. Required features needed to mitigate a fuel handling accident are available;

INSERT R

c. Required features necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

INSERT T

The DC electrical power requirements for MODES 1, 2, and 3 are covered in LCO 3.8.4.

ACTIONS

recently

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

INSERT V

A.1, A.2.1, A.2.2, A.2.3, and A.2.4

If more than one DC distribution subsystem is required according to LCO 3.8.8, the DC electrical power subsystems remaining OPERABLE with one or more DC electrical power subsystems inoperable may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS, fuel movement, and operations with a potential for draining the reactor vessel.

recently irradiated

By allowance of the option to declare required features inoperable with associated DC electrical power subsystems inoperable, appropriate restrictions are implemented in accordance with the affected system LCOs' ACTIONS. However, in many instances, this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies in secondary containment, and any activities that could result in inadvertent draining of the reactor vessel).

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.5.2 (continued)

recently

As Noted, if Unit 3 is in MODE 4 or 5, or moving irradiated fuel assemblies in the secondary containment, the Note to Unit 3 SR 3.8.5.1 is applicable. This ensures that a Unit 2 SR will not require a Unit 3 SR to be performed, when the Unit 3 Technical Specifications exempts performance of a Unit 3 SR. (However, as stated in the Unit 3 SR 3.8.5.1 Note, while performance of an SR is exempted, the SR still must be met.)

REFERENCES

1. UFSAR, Chapter 14.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.8 Distribution Systems—Shutdown

BASES

BACKGROUND	A description of the AC and DC electrical power distribution system is provided in the Bases for LCO 3.8.7, "Distribution Systems—Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident and transient analyses in the UFSAR, Chapter 14 (Ref. 1), assume Engineered Safety Feature (ESF) systems are OPERABLE. The AC and DC 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 AC and DC 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 AC and DC electrical power sources and associated power distribution subsystems during MODES 4 and 5 and during movement of recently irradiated fuel assemblies in the secondary containment ensures that:

a. The facility 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 an inadvertent draindown of the vessel or a fuel handling accident. INSERT R

INSERT S

The AC and DC electrical power distribution systems satisfy Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

Various combinations of subsystems, equipment, and components are required OPERABLE by other LCOs, depending on the specific plant condition. Implicit in those requirements is the required OPERABILITY of necessary support required features. This LCO explicitly requires energization of the portions of the Unit 2 electrical distribution system necessary to support OPERABILITY of Technical Specifications required systems, equipment, and components—both specifically addressed by their own LCO, and implicitly required by the definition of OPERABILITY. In addition, some components that may be required by Unit 2 receive power through Unit 3 electrical power distribution subsystems (e.g., Standby Gas Treatment (SGT) System and DC control power for two of the four 4 kV emergency buses, as well as control power for two of the diesel generators). Therefore, Unit 3 AC and DC electrical power distribution subsystems needed to support the required equipment must also be OPERABLE.

In addition, it is acceptable for required buses to be cross-tied during shutdown conditions, permitting a single source to supply multiple redundant buses, provided the source is capable of maintaining proper frequency (if required) and voltage.

Maintaining these portions of the distribution system energized ensures the availability of sufficient power to operate the plant in a safe manner to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents and inadvertent reactor vessel draindown). INSERT R

APPLICABILITY

The AC and DC electrical power distribution subsystems required to be OPERABLE in MODES 4 and 5 and during movement of recently irradiated fuel assemblies in the secondary containment provide assurance that:

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core in case of an inadvertent draindown of the reactor vessel;

b. Systems needed to mitigate a fuel handling accident INSERT R are available;

(continued)

BASES

APPLICABILITY (continued)

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The AC and DC electrical power distribution subsystem requirements for MODES 1, 2, and 3 are covered in LCO 3.8.7.

ACTIONS

INSERT V

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

A.1, A.2.1, A.2.2, A.2.3, A.2.4, and A.2.5

Although redundant required features may require redundant electrical power distribution subsystems to be OPERABLE, one OPERABLE distribution subsystem may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS, recently irradiated fuel movement, and operations with a potential for draining the reactor vessel. By allowing the option to declare required features inoperable with associated electrical power distribution subsystems inoperable, appropriate restrictions are implemented in accordance with the affected distribution subsystem LCO's Required Actions. However, in many instances this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made, (i.e., to suspend CORE ALTERATIONS, movement of recently irradiated fuel assemblies in the secondary containment, and any activities that could result in inadvertent draining of the reactor vessel).

(continued)

BASES (continued)

SAFETY LIMITS

The reactor core SLs are established to protect the integrity of the fuel clad barrier to the release of radioactive materials to the environs. SL 2.1.1.1 and SL 2.1.1.2 ensure that the core operates within the fuel design criteria. SL 2.1.1.3 ensures that the reactor vessel water level is greater than the top of the active irradiated fuel in order to prevent elevated clad temperatures and resultant clad perforations.

APPLICABILITY

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

SAFETY LIMIT VIOLATIONS

2.2.1

If any SL is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. 3).

10 CFR 50.67, "Accident Source Term,"

2.2.2

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

2.2.3

If any SL is violated, the senior management of the nuclear plant and the utility shall be notified within 24 hours. The 24 hour period provides time for plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to the senior management.

(continued)

BASES

SAFETY LIMIT VIOLATIONS (continued)	<u>2.2.4</u> If any SL is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Ref. 5). A copy of the report shall also be provided to the senior management of the nuclear plant and the utility. <u>2.2.5</u> If any SL is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.
REFERENCES	1. EMF-93-115 (P), July 1993. 2. NEDE-24011-P-A-10, February 1991. 3. 10 CFR 50.72. 4. ~~10 CFR 100.~~ 10 CFR 50.67 5. 10 CFR 50.73.

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on reactor steam dome pressure protects 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. Establishing an upper limit on reactor steam dome pressure ensures continued RCS integrity with regard to pressure excursions. Per the UFSAR (Ref. 1), the reactor coolant pressure boundary (RCPB) shall be designed with sufficient margin to ensure that the design conditions are not exceeded during normal operation and abnormal operational transients.

During normal operation and abnormal operational transients, 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 are hydrostatically tested at 125% of design pressure, in accordance with ASME Code requirements, prior to initial operation when there is no fuel in the core. Any further hydrostatic testing with fuel in the core may be done under LCO 3.10.1, "Inservice Leak and Hydrostatic Testing Operation." 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 reducing the number of protective barriers designed to prevent radioactive releases from exceeding the limits specified in ~~10 CFR 100, "Reactor Site Criteria"~~ (Ref. 4). If this occurred in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere.

APPLICABLE SAFETY ANALYSES

The RCS safety/relief valves and the Reactor Protection System Reactor Pressure—High Function have settings established to ensure that the RCS pressure SL will not be exceeded.

(continued)

10 CFR 50.67, "Accident Source Term,"

BASES

10 CFR 50.67, "Accident Source Term,"

SAFETY LIMIT VIOLATIONS (continued)

2.2.2

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

2.2.3

If any SL is violated, the senior management of the nuclear plant and the utility shall be notified within 24 hours. The 24 hour period provides time for plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to the senior management.

2.2.4

If any SL is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Ref. 8). A copy of the report shall also be provided to the senior management of the nuclear plant and the utility.

2.2.5

If any SL is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.

REFERENCES

1. UFSAR, Section 1.5.2.2.

2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000.

(continued)

BASES

REFERENCES (continued)

3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IW-5000.
4. 10 CFR ~~100.~~ 50.67.
5. ASME, Boiler and Pressure Vessel Code, Section III, 1965 Edition, including Addenda to summer of 1966.
6. ASME, Boiler and Pressure Vessel Code, Section III, 1980 Edition, Addenda to winter of 1981.
7. 10 CFR 50.72.
8. 10 CFR 50.73.

BASES

ACTIONS

B.1 and B.2 (continued)

control rods has less impact on control rod worth than withdrawals have. Required Action B.1 is modified by a Note which allows the RWM to be bypassed to allow the affected control rods to be returned to their correct position.

LCO 3.3.2.1 requires verification of control rod movement by a second licensed operator or a qualified member of the technical staff.

When nine or more OPERABLE control rods are not in compliance with BPWS, the reactor mode switch must be placed in the shutdown position within 1 hour. With the mode switch in shutdown, the reactor is shut down, and as such, does not meet the applicability requirements of this LCO. The allowed Completion Time of 1 hour is reasonable to allow insertion of control rods to restore compliance, and is appropriate relative to the low probability of a CRDA occurring with the control rods out of sequence.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

The control rod pattern is verified to be in compliance with the BPWS at a 24 hour Frequency to ensure the assumptions of the CRDA analyses are met. The 24 hour Frequency was developed considering that the primary check on compliance with the BPWS is performed by the RWM (LCO 3.3.2.1), which provides control rod blocks to enforce the required sequence and is required to be OPERABLE when operating at ≤ 10% RTP.

REFERENCES

1. NEDE-24011-P-A-10-US, "General Electric Standard Application for Reactor Fuel, Supplement for United States," Section 2.2.3.1, February 1991.

2. Letter (BWROG-8644) from T. Pickens (BWROG) to G. C. Lainas (NRC), "Amendment 17 to General Electric Licensing Topical Report NEDE-24011-P-A."

3. UFSAR, Section 14.6.2.3.

4. NUREG-0800, Section 15.4.9, Revision 2, July 1981.

5. 10 CFR ~~100.11.~~ 50.67.

(continued)

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Standby Liquid Control (SLC) System

BASES

BACKGROUND

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

INSERT B

Reference 1 requires a SLC System with a minimum flow capacity and boron content equivalent in control capacity to 86 gpm of 13 weight percent sodium pentaborate solution. Natural sodium pentaborate solution is 19.8% atom Boron-10. Therefore, the system parameters of concern, boron concentration (C), SLC pump flow rate (Q), and Boron-10 enrichment (E), may be expressed as a multiple of ratios. The expression is as follows:

$$\frac{C}{13\%\ \text{weight}} \times \frac{Q}{86\ \text{gpm}} \times \frac{E}{19.8\%\ \text{atom}}$$

If the product of this expression is ≥ 1, then the SLC System satisfies the criteria of Reference 1. As such, the equation forms the basis for acceptance criteria for the surveillances of concentration, flow rate, and boron enrichment and is presented in Table 3.1.7-1.

The SLC System consists of a boron solution storage tank, two positive displacement pumps, two explosive valves that are provided in parallel for redundancy, and associated piping and valves used to transfer borated water from the storage tank to the reactor pressure vessel (RPV). The borated solution is discharged near the bottom of the core shroud, where it then mixes with the cooling water rising through the core. A smaller tank containing demineralized water is provided for testing purposes.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The SLC System is manually initiated from the main control room, as directed by the emergency operating procedures, if the operator believes the reactor cannot be shut down, or kept shut down, with the control rods. The SLC System is used in the event that enough control rods cannot be inserted to accomplish shutdown and cooldown in the normal manner. The SLC System injects borated water into the reactor core to add negative reactivity to compensate for all of the various reactivity effects that could occur during plant operations. To meet this objective, it is necessary to inject a quantity of boron, which produces a concentration of 660 ppm of natural boron, in the reactor coolant at 68°F. To allow for potential leakage and imperfect mixing in the reactor system, an additional amount of boron equal to 25% of the amount cited above is added (Ref. 2). The minimum mass of Boron-10 (162.7 lbm) needed for injection is calculated such that the required quantity is achieved accounting for dilution in the RPV with normal water level and including the water volume in the residual heat removal shutdown cooling piping and in the recirculation loop piping. This quantity of borated solution is the amount that is above the pump suction shutoff level in the boron solution storage tank. No credit is taken for the portion of the tank volume that cannot be injected. The maximum concentration of sodium pentaborate listed in Table 3.1.7-1 has been established to ensure that the solution saturation temperature does not exceed 43°F.

INSERT Y

The SLC System satisfies Criterion 4 of the NRC Policy Statement.

LCO

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

(continued)

BASES (continued)

APPLICABILITY

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

INSERT C

ACTIONS

A.1 and A.2

If the boron solution concentration is > 9.82% weight but the concentration and temperature of boron in solution and pump suction piping temperature are within the limits of Figure 3.1.7-1, operation is permitted for a limited period since the SLC subsystems are capable of performing the intended function. It is not necessary under these conditions to declare both SLC subsystems inoperable since the SLC subsystems are capable of performing their intended function.

The concentration and temperature of boron in solution and pump suction piping temperature must be verified to be within the limits of Figure 3.1.7-1 within 8 hours and once per 12 hours thereafter (Required Action A.1). The temperature versus concentration curve of Figure 3.1.7-1 ensures a 10°F margin will be maintained above the saturation temperature. This verification ensures that boron does not precipitate out of solution in the storage tank or in the pump suction piping due to low boron solution temperature (below the saturation temperature for the given concentration). The Completion Time for performing Required Action A.1 is considered acceptable given the low probability of a Design Basis Accident (DBA) or transient occurring concurrent with the failure of the control rods to shut down the reactor and operating experience which has shown there are relatively slow variations in the measured parameters of concentration and temperature over these time periods.

(continued)

BASES

ACTIONS

D.1 (continued)

AND MODE 4 WITHIN 36 HOURS

TIMES ARE

brought to MODE 3 within 12 hours. The allowed Completion ~~Time of 12 hours is~~ reasonable, based on operating experience, to reach ~~MODE 3~~ from full power conditions in an orderly manner and without challenging plant systems.

THE REQUIRED MODES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1, SR 3.1.7.2, and SR 3.1.7.3

SR 3.1.7.1 through SR 3.1.7.3 are 24 hour Surveillances verifying certain characteristics of the SLC System (e.g., the level and temperature of the borated solution in the storage tank), thereby ensuring SLC System OPERABILITY without disturbing normal plant operation. These Surveillances ensure that the proper borated solution level and temperature, including the temperature of the pump suction piping, are maintained. Maintaining a minimum specified borated solution temperature is important in ensuring that the boron remains in solution and does not precipitate out in the storage tank or in the pump suction piping. The temperature limit specified in SR 3.1.7.2 and SR 3.1.7.3 and the maximum sodium pentaborate concentration specified in Table 3.1.7-1 ensures that a 10°F margin will be maintained above the saturation temperature. Control room alarms for low SLC storage tank temperature and low SLC System piping temperature are available and are set at 55°F. As such, SR 3.1.7.2 and SR 3.1.7.3 may be satisfied by verifying the absence of low temperature alarms for the SLC storage tank and SLC System piping. The 24 hour Frequency is based on operating experience and has shown there are relatively slow variations in the measured parameters of level and temperature.

SR 3.1.7.4 and SR 3.1.7.6

SR 3.1.7.4 verifies the continuity of the explosive charges in the injection valves to ensure that proper operation will occur if required. Other administrative controls, such as those that limit the shelf life of the explosive charges, must be followed. The 31 day Frequency is based on operating experience and has demonstrated the reliability of the explosive charge continuity.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.9 (continued)

Surveillance when performed at the 24 month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.1.7.10

Enriched sodium pentaborate solution is made by mixing granular, enriched sodium pentaborate with water. In order to ensure the proper B-10 atom percentage (in accordance with Table 3.1.7-1) is being used, calculations must be performed to verify the actual B-10 enrichment within 8 hours after addition of the solution to the SLC tank. The calculations may be performed using the results of isotopic tests on the granular sodium pentaborate or vendor certification documents. The Frequency is acceptable considering that boron enrichment is verified during the procurement process and any time boron is added to the SLC tank.

REFERENCES

1. 10 CFR 50.62.

2. UFSAR, Section 3.8.4.

3. 10 CFR 50.67.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.a. Reactor Vessel Water Level—Low Low Low (Level 1) (continued)

The Reactor Vessel Water Level—Low Low Low (Level 1) Allowable Value is chosen to be the same as the ECCS Level 1 Allowable Value (LCO 3.3.5.1) to ensure that the MSLs isolate on a potential loss of coolant accident (LOCA) to prevent offsite doses from exceeding 10 CFR ~~100~~ 50.67 limits.

This Function isolates MSIVs, MSL drains, MSL sample lines and recirculation loop sample line valves.

1.b. Main Steam Line Pressure—Low

Low MSL pressure indicates that there may be a problem with the turbine pressure regulation, which could result in a low reactor vessel water level condition and the RPV cooling down more than 100°F/hr if the pressure loss is allowed to continue. The Main Steam Line Pressure—Low Function is directly assumed in the analysis of the pressure regulator failure (Ref. 3). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hr) is not reached. In addition, this Function supports actions to ensure that Safety Limit 2.1.1.1 is not exceeded. (This Function closes the MSIVs prior to pressure decreasing below 785 psig, which results in a scram due to MSIV closure, thus reducing reactor power to < 25% RTP.)

The MSL low pressure signals are initiated from four transmitters that are connected to the MSL header. The transmitters are arranged such that, even though physically separated from each other, each transmitter is able to detect low MSL pressure. Four channels of Main Steam Line Pressure—Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be high enough to prevent excessive RPV depressurization.

The Main Steam Line Pressure—Low Function is only required to be OPERABLE in MODE 1 since this is when the assumed transient can occur (Ref. 1).

This Function isolates MSIVs, MSL drains, MSL sample lines and recirculation loop sample line valves.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

1.c. Main Steam Line Flow—High

Main Steam Line Flow—High is provided to detect a break of the MSL and to initiate closure of the MSIVs. If the steam were allowed to continue flowing out of the break, the reactor would depressurize and the core could uncover. If the RPV water level decreases too far, fuel damage could occur. Therefore, the isolation is initiated on high flow to prevent or minimize core damage. The Main Steam Line Flow—High Function is directly assumed in the analysis of the main steam line break (MSLB) (Ref. 3). The isolation action, along with the scram function of the Reactor Protection System (RPS), ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46 and offsite doses do not exceed the ~~10 CFR 100~~ 10 CFR 50.67 limits.

The MSL flow signals are initiated from 16 transmitters that are connected to the four MSLs. The transmitters are arranged such that, even though physically separated from each other, all four connected to one MSL would be able to detect the high flow. Four channels of Main Steam Line Flow—High Function for each MSL (two channels per trip system) are available and are required to be OPERABLE so that no single instrument failure will preclude detecting a break in any individual MSL.

The Allowable Value is chosen to ensure that offsite dose limits are not exceeded due to the break.

This Function isolates MSIVs, MSL drains, MSL sample lines and recirculation loop sample line valves.

1.d. Main Steam Line—High Radiation

The Main Steam Line—High Radiation Function is provided to detect gross release of fission products from the fuel and to initiate closure of the MSIVs. The trip setting is set low enough so that a high radiation trip results from a design basis rod drop accident and high enough above background radiation levels in the vicinity of the main steam lines so that spurious trips at rated power are avoided. The Main Steam Line—High Radiation Function is directly assumed in the analysis of the control rod drop accident (Ref. 3).

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.d. Main Steam Line—High Radiation (continued)

The Main Steam Line—High Radiation signals are initiated from four gamma sensitive instruments. Four channels are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value is chosen to ensure that offsite dose limits are not exceeded.

This Function isolates MSIVs, MSL drains, MSL sample lines and recirculation loop sample line valves.

1.e. Main Steam Tunnel Temperature—High

The Main Steam Tunnel Temperature Function is provided to detect a break in a main steam line and provides diversity to the high flow instrumentation.

Main Steam Tunnel Temperature signals are initiated from resistance temperature detectors (RTDs) located along the main steam line between the drywell wall and the turbine. Sixteen channels of Main Steam Tunnel Temperature—High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value is chosen to detect a leak equivalent to between 1% and 10% rated steam flow.

This Function isolates MSIVs, MSL drains, MSL sample lines and recirculation loop sample line valves.

Primary Containment Isolation

2.a. Reactor Vessel Water Level—Low (Level 3)

Low RPV water level indicates that the capability to cool the fuel may be threatened. The valves whose penetrations communicate with the primary containment are isolated to limit the release of fission products. The isolation of the primary containment on Level 3 supports actions to ensure that offsite dose limits of 10 CFR ~~100~~ are not exceeded.
50.67

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.a. Reactor Vessel Water Level—Low (Level 3) (continued)

The Reactor Vessel Water Level—Low (Level 3) Function associated with isolation is implicitly assumed in the UFSAR analysis as these leakage paths are assumed to be isolated post LOCA.

Reactor Vessel Water Level—Low (Level 3) signals are initiated from level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level—Low (Level 3) Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level—Low (Level 3) Allowable Value was chosen to be the same as the RPS Level 3 scram Allowable Value (LCO 3.3.1.1), since isolation of these valves is not critical to orderly plant shutdown.

This Function isolates the Group II(A) valves listed in Reference 1 with the exception of RWCU isolation valves and RHR shutdown cooling pump suction valves which are addressed in Functions 5.c and 6.b, respectively.

2.b. Drywell Pressure—High

50.67

High drywell pressure can indicate a break in the RCPB inside the primary containment. The isolation of some of the primary containment isolation valves on high drywell pressure supports actions to ensure that offsite dose limits of 10 CFR ~~100~~ are not exceeded. The Drywell Pressure—High Function, associated with isolation of the primary containment, is implicitly assumed in the UFSAR accident analysis as these leakage paths are assumed to be isolated post LOCA.

High drywell pressure signals are initiated from pressure transmitters that sense the pressure in the drywell. Four channels of Drywell Pressure—High are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

5.a. RWCU Flow—High (continued)

The high RWCU flow signals are initiated from transmitters that are connected to the pump suction line of the RWCU System. Two channels of RWCU Flow—High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The RWCU Flow—High Allowable Value ensures that a break of the RWCU piping is detected.

This Function isolates the inboard and outboard RWCU pump suction penetration and the outboard valve at the RWCU connection to reactor feedwater.

5.b. Standby Liquid Control (SLC) System Initiation

The isolation of the RWCU System is required when the SLC System has been initiated to prevent dilution and removal of the boron solution by the RWCU System (Ref. 5). SLC System initiation signals are initiated from the remote SLC System start switch.

There is no Allowable Value associated with this Function since the channels are mechanically actuated based solely on the position of the SLC System initiation switch.

INSERT D

Two channels of the SLC System Initiation Function are available and are required to be OPERABLE ~~only~~ in MODES 1 ~~and 2, since these are~~ the only MODES where the reactor can be critical~~, and these~~ MODES are consistent with the Applicability for the SLC System (LCO 3.1.7).

This Function isolates the inboard and outboard RWCU pump suction penetration and the outboard valve at the RWCU connection to reactor feedwater.

5.c. Reactor Vessel Water Level—Low (Level 3)

Low RPV water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, isolation of some interfaces with the reactor vessel occurs to isolate the potential sources of a break. The isolation of the RWCU System on Level 3 supports actions to ensure that the fuel

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3, 4. Reactor Building Ventilation and Refueling Floor Ventilation Exhaust Radiation—High (continued)

channels of Reactor Building Ventilation Exhaust Radiation—High Function and four channels of Refueling Floor Ventilation Exhaust Radiation—High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Values are chosen to promptly detect gross failure of the fuel cladding.

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

ACTIONS

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The Control Room Air Intake Radiation—High Function consists of four independent monitors. Two channels of Control Room Air Intake Radiation—High per trip system are available and are required to be OPERABLE to ensure that no single instrument failure can preclude MCREV System initiation. The Allowable Value was selected to ensure protection of the control room personnel.

RECENTLY

The Control Room Air Intake Radiation—High Function is required to be OPERABLE in MODES 1, 2, and 3 and during ~~CORE ALTERATIONS,~~ OPDRVs, and movement of irradiated fuel assemblies in the secondary containment, to ensure that control room personnel are protected during a LOCA, fuel handling event, or vessel draindown event. During MODES 4 and 5, when these specified conditions are not in progress (e.g., ~~CORE ALTERATIONS~~), the probability of a LOCA or fuel damage is low; thus, the Function is not required.

INSERT X

OPDRVs

ACTIONS

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

A.1 and A.2

Because of the redundancy of sensors available to provide initiation signals and the redundancy of the MCREV System design, an allowable out of service time of 6 hours has been shown to be acceptable (Ref. 4), to permit restoration of any inoperable channel to OPERABLE status. However, this out of service time is only acceptable provided the Control Room Air Intake Radiation—High Function is still maintaining MCREV System initiation capability. The Function is considered to be maintaining MCREV System

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 RCS Specific Activity

BASES

BACKGROUND

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

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

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

APPLICABLE SAFETY ANALYSES

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

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The limits on specific activity are values from a parametric evaluation of typical site locations. These limits are conservative because the evaluation considered more restrictive parameters than for a specific site, such as the location of the site boundary and the meteorological conditions of the site.

RCS specific activity satisfies Criterion 2 of the NRC Policy Statement.

LCO

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

APPLICABILITY

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

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

ACTIONS

A.1 and A.2

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

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

A Note to the Required Actions of Condition A excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE(S) while relying on the ACTIONS even though the ACTIONS may eventually require plant shutdown. This exception is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the plant remains at, or proceeds to, power operation.

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

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

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

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

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

This Surveillance is performed to ensure iodine remains within limit during normal operation. The 7 day Frequency is adequate to trend changes in the iodine activity level.

This SR is modified by a Note that requires this Surveillance to be performed only in MODE 1 because the level of fission products generated in other MODES is much less.

REFERENCES

1. 10 CFR ~~100.11, 1973.~~ 50.67.
2. UFSAR, Section 14.6.5.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.7 (continued)

position, since these valves were verified to be in the correct position prior to locking or securing. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves. The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.6.1.3.8

Verifying the isolation time of each power operated and each automatic PCIV is within limits is required to demonstrate OPERABILITY. MSIVs may be excluded from this SR since MSIV full closure isolation time is demonstrated by SR 3.6.1.3.9. The isolation time test ensures that the valve will isolate in a time period less than or equal to that assumed in the safety analyses. The isolation time is in accordance with Reference 2 or the requirements of the Inservice Testing Program which ever is more conservative. The Frequency of this SR is in accordance with the requirements of the Inservice Testing Program.

SR 3.6.1.3.9

Verifying that the isolation time of each MSIV is within the specified limits is required to demonstrate OPERABILITY. The isolation time test ensures that the MSIV will isolate in a time period that does not exceed the times assumed in the DBA analyses. This ensures that the calculated radiological consequences of these events remain within

50.67

10 CFR ~~100~~ limits. The Frequency of this SR is in accordance with the requirements of the Inservice Testing Program.

SR 3.6.1.3.10

Automatic PCIVs close on a primary containment isolation signal to prevent leakage of radioactive material from primary containment following a DBA. This SR ensures that each automatic PCIV will actuate to its isolation position on a primary containment isolation signal. The LOGIC SYSTEM

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.13

This SR ensures that in case the non-safety grade instrument air system is unavailable, the SGIG System will perform its design function to supply nitrogen gas at the required pressure for valve operators and valve seals supported by the SGIG System. The 24 month Frequency was developed considering it is prudent that this Surveillance be performed only during a plant outage. Operating experience has shown that these components will usually pass this Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.1.3.14

~~Leakage through each MSIV must be ≤ 11.5 scfh when tested at ≥ P_t (25 psig). The analyses in Reference 1 are based on treatment of MSIV leakage as a secondary containment bypass leakage, independent of a primary to secondary containment leakage analyzed at 1.27 L_a. In the Reference 1 analysis all 4 steam lines are assumed to leak at the TS Limit. This ensures that MSIV leakage is properly accounted for in determining the overall impacts of primary containment leakage. The Frequency is required by the Primary Containment Leakage Rate Testing Program.~~

INSERT A

SR 3.6.1.3.15

Verifying the opening of each 6 inch and 18 inch primary containment purge valve and each 18 inch primary containment exhaust valve is restricted by a blocking device to less than or equal to the required maximum opening angle specified in the UFSAR (Ref. 4) is required to ensure that the valves can close under DBA conditions within the times in the analysis of Reference 1. If a LOCA occurs, the purge and exhaust valves must close to maintain primary containment leakage within the values assumed in the accident analysis. At other times pressurization concerns are not present, thus the purge and exhaust valves can be fully open. The 24 month Frequency is appropriate because the blocking devices may be removed during a refueling outage.

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.1 Secondary Containment

BASES

BACKGROUND

INSERT E

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

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

APPLICABLE SAFETY ANALYSES

There ~~are two~~ is one principal acciden~~ts~~t for which credit is taken for secondary containment OPERABILITY. ~~These are~~ that is a loss of coolant accident (LOCA) (Ref. 1) ~~and a fuel handling accident inside secondary containment (Ref. 2)~~. The secondary containment performs no active function in response to ~~each of these~~ this limiting even~~ts~~t; however, its leak

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

tightness is required to ensure that fission products entrapped within the secondary containment structure will be ~~treated~~ COLLECTED by the SGT System ~~prior to~~ FOR discharge to the environment VIA THE MAIN STACK.

Secondary containment satisfies Criterion 3 of the NRC Policy Statement.

LCO

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

APPLICABILITY

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

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

ACTIONS

A.1

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

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

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

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

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

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

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

INSERT U

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1.1 and SR 3.6.4.1.2

Verifying that secondary containment equipment hatches and one access door in each access opening are closed ensures that the infiltration of outside air of such a magnitude as to prevent maintaining the desired negative pressure does not occur. Verifying that all such openings are closed provides adequate assurance that exfiltration from the secondary containment will not occur. In this application, the term "sealed" has no connotation of leak tightness. Maintaining secondary containment OPERABILITY requires verifying one door in the access opening is closed. An access opening contains one inner and one outer door. In some cases, secondary containment access openings are shared such that a secondary containment barrier may have multiple inner or multiple outer doors. The intent is to not breach secondary containment at any time when secondary containment is required. This is achieved by maintaining the inner or outer portion of the barrier closed at all times. However, all secondary containment access doors are normally kept closed, except when the access opening is being used for entry and exit or when maintenance is being performed on an access opening. The 31 day Frequency for these SRs has been shown to be adequate, based on operating experience, and is considered adequate in view of the other indications of door and hatch status that are available to the operator.

SR 3.6.4.1.3 and SR 3.6.4.1.4

INSERT F

~~The SGT System exhausts the secondary containment atmosphere to the environment through appropriate treatment equipment.~~

~~To ensure that fission products are treated, SR 3.6.4.1.3 verifies that the SGT System will rapidly establish and~~ maintain a pressure in the secondary containment that is less than the pressure external to the secondary containment boundary. This is confirmed by demonstrating that one SGT subsystem will draw down the secondary containment to ≥ 0.25 inches of vacuum water gauge in ≤ ~~120 seconds~~ 15 MINUTES. This cannot be accomplished if the secondary containment boundary is not intact.

SR 3.6.4.1.4 demonstrates that one SGT subsystem can maintain ≥ 0.25 inches of vacuum water gauge for 1 hour at a flow rate ≤ 10,500 cfm. The 1 hour test period allows secondary containment to be in thermal equilibrium at steady

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.2 Secondary Containment Isolation Valves (SCIVs)

BASES

BACKGROUND

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

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

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

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

APPLICABLE SAFETY ANALYSES

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

established by SCIVs is required to ensure that leakage from the primary containment is ~~processed~~ EXHAUSTED by the Standby Gas Treatment (SGT) System ~~before being released~~ FOR ELEVATED RELEASE to the environment VIA THE MAIN STACK.

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

SCIVs satisfy Criterion 3 of the NRC Policy Statement.

LCO

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

The power operated isolation valves are considered OPERABLE when their isolation times are within limits and the valves actuate on an automatic isolation signal. The valves covered by this LCO, along with their associated stroke times, are listed in Reference ~~3~~ 2.

The normally closed isolation valves or blind flanges are considered OPERABLE when manual valves are closed or open in accordance with appropriate administrative controls, automatic SCIVs are de-activated and secured in their closed position, and blind flanges are in place. These passive isolation valves or devices are listed in Reference ~~3~~ 2.

APPLICABILITY

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

In MODES 4 and 5, the probability and consequences of these events are reduced due to pressure and temperature limitations in these MODES. Therefore, maintaining SCIVs OPERABLE is not required in MODE 4 or 5, except for other situations under which significant radioactive releases can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs) ~~, during CORE ALTERATIONS,~~ or during movement of RECENTLY irradiated fuel assemblies in the secondary containment. INSERT P Moving irradiated fuel assemblies in the secondary containment may also occur in MODES 1, 2, and 3.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

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

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

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

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

SURVEILLANCE REQUIREMENTS

SR 3.6.4.2.1

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.4.2.3 (continued)

under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components will usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section ~~14.6.3.~~ 14.9.2.1.
~~2. UFSAR, Section 14.6.4.~~
2. ~~3.~~ Technical Requirements Manual.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.3 Standby Gas Treatment (SGT) System

BASES

BACKGROUND

~~The SGT System is required by UFSAR design criteria (Ref. 1). The function of the SGT System is to ensure that radioactive materials that leak from the primary containment into the secondary containment following a Design Basis Accident (DBA) are filtered and adsorbed prior to exhausting to the environment.~~

INSERT G

A single SGT System is common to both Unit 2 and Unit 3 and consists of two fully redundant subsystems, each with its own set of ductwork, dampers, valves, charcoal filter train, and controls. Both SGT subsystems share a common inlet plenum. This inlet plenum is connected to the refueling floor ventilation exhaust duct for each Unit and to the suppression chamber and drywell of each Unit. Both SGT subsystems exhaust to the plant offgas stack through a common exhaust duct served by three 100% capacity system fans. SGT System fans 0AV020 and 0BV020 automatically start on Unit 2 secondary containment isolation signals. SGT System fans 0CV020 and 0BV020 automatically start on Unit 3 secondary containment isolation signals.

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

a. A demister or moisture separator;

b. An electric heater;

c. A prefilter;

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

e. A charcoal adsorber; and

f. A second HEPA filter.

INSERT H

The SGT System is sized such that each 100% capacity fan will provide a flow rate of 10,500 cfm at 20 inches water gauge static pressure to support the control of fission product releases. The SGT System is designed to restore and maintain secondary containment at a negative pressure of 0.25 inches water gauge relative to the atmosphere following

(continued)

BASES

ALTHOUGH NOT CREDITED IN ANY DBA ANALYSIS, THE

BACKGROUND (continued)

the receipt of a secondary containment isolation signal. Maintaining this negative pressure is based upon the existence of calm wind conditions (up to 5 mph), a maximum SGT System flow rate of 10,500 cfm, outside air temperature of 95°F and a temperature of 150°F for air entering the SGT System from inside secondary containment.

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

The SGT System automatically starts and operates in response to actuation signals indicative of conditions or an accident that could require operation of the system. Following initiation, two charcoal filter train fans (OCV020 and OBV020) start. Upon verification that both subsystems are operating, the redundant subsystem is normally shut down.

APPLICABLE SAFETY ANALYSES

~~The design basis for the SGT System is to mitigate the consequences of a loss of coolant accident and fuel handling accidents (Ref. 2). For all events analyzed, the SGT System is shown to be automatically initiated to reduce, via filtration and adsorption, the radioactive material released to the environment.~~

INSERT I

The SGT System satisfies Criterion 3 of the NRC Policy Statement.

LCO

Following a DBA, a minimum of one SGT subsystem is required to maintain the secondary containment at a negative pressure with respect to the environment and to process gaseous releases. Meeting the LCO requirements for two OPERABLE subsystems ensures operation of at least one SGT subsystem in the event of a single active failure.

(continued)

BASES

LCO (continued)

For Unit 3, one SGT subsystem is OPERABLE when ~~one charcoal filter train,~~ one fan (OCV020) and associated ductwork, dampers, valves, and controls are OPERABLE. The second SGT subsystem is OPERABLE when the other ~~charcoal filter train,~~ ~~one~~ fan (OBV020) and associated ductwork, damper, valves, and controls are OPERABLE.

APPLICABILITY

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

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

ACTIONS

A.1

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

B.1 and B.2

If the SGT subsystem cannot be restored to OPERABLE status within the required Completion Time in MODE 1, 2, or 3, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

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

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

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

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

(continued)

BASES

ACTIONS (continued)

D.1

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

AND

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

RECENTLY

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

INSERT U

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

SURVEILLANCE REQUIREMENTS

SR 3.6.4.3.1

Operating each SGT subsystem (including each filter train fan) for ≥ 15 minutes ensures that both subsystems are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. ~~Operation with the heaters on (automatic heater cycling to maintain temperature) for ≥ 15 minutes every 31 days is sufficient to eliminate moisture on the adsorbers and HEPA filters since during idle periods instrument air is injected into the filter plenum to keep the filters dry.~~ The 31 day Frequency was developed in consideration of the known reliability of fan motors and controls and the redundancy available in the system.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.4.3.2

INSERT J

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

SR 3.6.4.3.3

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

REFERENCES

1. UFSAR, Section 1.5.1.6.
2. UFSAR, Section 14.9.

BASES

BACKGROUND (continued)

initiate an emergency shutdown of non-essential equipment and lighting to reduce the heat generation to a minimum. Heat removal would be accomplished by conduction through the floors, ceilings, and walls to adjacent rooms and to the environment. INSERT K ~~Additionally, the MCREV System is designed to maintain the control room environment for a 30 day continuous occupancy after a DBA without exceeding 5 rem whole body dose.~~ A single MCREV subsystem will pressurize the control room to prevent infiltration of air from surrounding buildings. MCREV System operation in maintaining control room habitability is discussed in the UFSAR, Chapters 7, 10, and 12, (Refs. 1, 2, and 3, respectively).

APPLICABLE SAFETY ANALYSES

The ability of the MCREV System to maintain the habitability of the control room is an explicit assumption for the safety analyses presented in the UFSAR, Chapters 10 and 12 (Refs. 2 and 3, respectively). INSERT L ~~The MCREV System is assumed to operate following a loss of coolant accident, fuel handling accident, main steam line break, and control rod drop accident,~~ as discussed in the UFSAR, Section 14.9.1.5 (Ref. 4). The radiological doses to control room personnel as a result of the various DBAs are summarized in Reference 4. No single active or passive failure will cause the loss of outside or recirculated air from the control room.

The MCREV System satisfies Criterion 3 of the NRC Policy Statement.

LCO

Two redundant subsystems of the MCREV System are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other subsystem. Total system failure could result in exceeding a dose of 5 rem TEDE to the control room operators in the event of a ~~DBA.~~ LOCA.

The MCREV System is considered OPERABLE when the individual components necessary to control operator exposure are OPERABLE in both subsystems. A subsystem is considered OPERABLE when its associated:

a. Fan is OPERABLE;

(continued)

BASES

LCO (continued)

b. HEPA filter 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 flow can be maintained.

In addition, the control room boundary must be maintained, including the integrity of the walls, floors, ceilings, and ductwork. Temporary seals may be used to maintain the boundary. In addition, an access door may be opened provided the ability to pressurize the control room is maintained and the capability exists to close the affected door in an expeditious manner.

APPLICABILITY

In MODES 1, 2, and 3, the MCREV System must be OPERABLE to control operator exposure during and following a ~~DBA~~ LOCA, since the ~~DBA~~ LOCA could lead to a fission product release.

In MODES 4 and 5, the probability and consequences of a ~~DBA~~ LOCA are reduced because of the pressure and temperature limitations in these MODES. Therefore, maintaining the MCREV System OPERABLE is not required in MODE 4 or 5, except for the following situations under which significant radioactive releases can be postulated:

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

~~b. During CORE ALTERATIONS; and~~

b. ~~c.~~ During movement of RECENTLY irradiated fuel assemblies in the secondary containment.

ACTIONS

A.1

With one MCREV subsystem inoperable, the inoperable MCREV subsystem must be restored to OPERABLE status within 7 days. With the unit in this condition, the remaining OPERABLE MCREV subsystem is adequate to maintain control room temperature and to perform control room radiation protection. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could

(continued)

BASES

ACTIONS

A.1 (continued)

result in reduced MCREV System capability. The 7 day Completion Time is based on the low probability of a ~~DBA~~ LOCA occurring during this time period, and that the remaining subsystem can provide the required capabilities.

B.1 and B.2

In MODE 1, 2, or 3, if the inoperable MCREV subsystem cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE that minimizes risk. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours and in MODE 4 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.

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

Insert U

The Required Actions of Condition C are modified by a Note indicating that LCO 3.0.3 does not apply. ~~If moving irradiated fuel assemblies while in MODE 1, 2, or 3, 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.~~

During movement of RECENTLY irradiated fuel assemblies in the secondary containment~~, during CORE ALTERATIONS,~~ or during OPDRVs, if the inoperable MCREV subsystem cannot be restored to OPERABLE status within the required Completion Time, the OPERABLE MCREV subsystem may be placed in operation. This action ensures that the remaining subsystem is OPERABLE, that no failures that would prevent automatic actuation will occur, and that any active failure will be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

(continued)

BASES

ACTIONS

AND

C.1, C.2.1, C.2.2~~, and C.2.3~~ (continued) RECENTLY

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

THIS ACTIVITY

D.1

If both MCREV subsystems are inoperable in MODE 1, 2, or 3, the MCREV System may not be capable of performing the intended function and the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

AND

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

The Required Actions of Condition E are modified by a Note indicating that LCO 3.0.3 does not apply. ~~If moving irradiated fuel assemblies while in MODE 1, 2, or 3, 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.~~

INSERT U

During movement of irradiated fuel assemblies in the secondary containment~~, during CORE ALTERATIONS,~~ or during OPDRVs, with two MCREV subsystems inoperable, action must be taken immediately to suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

RECENTLY THIS ACTIVITY

If applicable, ~~CORE ALTERATIONS and~~ movement of irradiated fuel assemblies in the secondary containment must be suspended immediately. Suspension of ~~these activities~~ shall not preclude completion of movement of a component to a safe position. If applicable, actions must be initiated

(continued)

BASES

ACTIONS

E.1, AND E.2, ~~and E.3~~ (continued)

immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

This SR verifies that a subsystem in a standby mode starts on demand and continues to operate for ≥ 15 minutes. Standby systems should be checked periodically to ensure that they start and function properly. As the environmental and normal operating conditions of this system are not severe, testing each subsystem once every month provides an adequate check on this system. Furthermore, the 31 day Frequency is based on the known reliability of the equipment and the two subsystem redundancy available.

SR 3.7.4.2

This SR verifies that the required MCREV testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.4.3

This SR verifies that on an actual or simulated initiation signal, each MCREV subsystem starts and operates. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.7.1.4 overlaps this SR to provide complete testing of the safety function. Operating experience has shown that these components will usually pass the SR when performed at the 24 month Frequency. Therefore, this Frequency is concluded to be acceptable from a reliability standpoint.

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.5 Main Condenser Offgas

BASES

BACKGROUND

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

The Main Condenser Offgas System has been incorporated into the unit design to reduce the gaseous radwaste emission. This system uses a catalytic recombiner to recombine radiolytically dissociated hydrogen and oxygen. The gaseous mixture is cooled and water vapor removed by the offgas recombiner condenser; the remaining water and condensibles are stripped out by the cooler condenser and moisture separator. The remaining gaseous mixture (i.e., the offgas recombiner effluent) is then processed by a charcoal adsorber bed prior to release.

APPLICABLE SAFETY ANALYSES

The main condenser offgas gross gamma activity rate is an initial condition of the Main Condenser Offgas System failure event, discussed in the UFSAR, Section 9.4.5 (Ref. 1). The analysis assumes a gross failure in the Main Condenser Offgas System that results in the rupture of the Main Condenser Offgas System pressure boundary. The gross gamma activity rate is controlled to ensure that, during the event, the calculated offsite doses will be well within the limits of 10 CFR 100 (Ref. 2) or the NRC staff approved licensing basis.

50.67

The main condenser offgas limits satisfy Criterion 2 of the NRC Policy Statement.

LCO

To ensure compliance with the assumptions of the Main Condenser Offgas System failure event (Ref. 1), the fission product release rate should be consistent with a noble gas release to the reactor coolant of 100 μCi/MWt-second after decay of 30 minutes. The LCO is established consistent

(continued)

BASES

ACTIONS

B.1, B.2, B.3.1, and B.3.2 (continued)

experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

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

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

REFERENCES

1. UFSAR, Section 9.4.5.
2. 10 CFR ~~100.~~ 50.67.

B 3.7 PLANT SYSTEMS

B 3.7.7 Spent Fuel Storage Pool Water Level

BASES

BACKGROUND

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

A general description of the spent fuel storage pool design is found in the UFSAR, Section 10.3 (Ref. 1). The assumptions of the fuel handling accident are found in the UFSAR, Section 14.6.4 (Ref. 2).

APPLICABLE SAFETY ANALYSES

50.67

The water level above the irradiated fuel assemblies is an implicit assumption of the fuel handling accident. A fuel handling accident is evaluated to ensure that the radiological consequences ~~(calculated whole body and thyroid doses at the site boundary)~~ are well below the guidelines set forth in 10 CFR ~~100~~ (Ref. 3). A fuel handling accident could release a fraction of the fission product inventory by breaching the fuel rod cladding as discussed in Reference 2.

Less

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

The spent fuel storage pool water level satisfies Criteria 2 and 3 of the NRC Policy Statement.

LCO

The specified water level (232 ft 3 inches plant elevation, which is equivalent to 22 ft over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks) preserves the assumptions of the fuel handling accident analysis (Ref. 2). As such, it is the minimum required for fuel movement within the spent fuel storage pool.

(continued)

BASES (continued)

APPLICABILITY

This LCO applies during movement of fuel assemblies in the spent fuel storage 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. If moving fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of fuel assemblies is not a sufficient reason to require a reactor shutdown.

When the initial conditions for an accident cannot be met, action must be taken to preclude the accident from occurring. If the spent fuel storage pool level is less than required, the movement of fuel assemblies in the spent fuel storage pool is suspended immediately. Suspension of this activity shall not preclude completion of movement of a fuel assembly to a safe position. This effectively precludes a spent fuel handling accident from occurring.

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

This SR verifies that sufficient water is available in the event of a fuel handling accident. The water level in the spent fuel storage pool must be checked periodically. The 7 day Frequency is acceptable, based on operating experience, considering that the water volume in the pool is normally stable, and all water level changes are controlled by unit procedures.

REFERENCES

1. UFSAR, Section 10.3.
2. UFSAR, Section 14.6.4.
3. 10 CFR ~~100.~~ 50.67.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.3 (continued)

Note 1 modifies this Surveillance to indicate that diesel engine runs for this Surveillance may include gradual loading, as recommended by the manufacturer, so that mechanical stress and wear on the diesel engine are minimized.

Note 2 modifies this Surveillance by stating that momentary transients because of changing bus loads do not invalidate this test. Similarly, momentary power factor transients above the limit do not invalidate the test.

Note 3 indicates that this Surveillance should be conducted on only one DG at a time in order to avoid common cause failures that might result from offsite circuit or grid perturbations.

Note 4 stipulates a prerequisite requirement for performance of this SR. A successful DG start must precede this test to credit satisfactory performance.

To minimize testing of the DGs, Note 5 allows a single test (instead of two tests, one for each unit) to satisfy the requirements for both units, with the DG synchronized to the 4 kV emergency bus of Unit 3 for one periodic test and synchronized to the 4 kV emergency bus of Unit 2 during the next periodic test. This is allowed since the main purpose of the Surveillance, to ensure DG OPERABILITY, is still being verified on the proper frequency, and each unit's breaker control circuitry, which is only being tested every second test (due to the staggering of the tests), historically have a very low failure rate. Note 5 modifies the specified frequency for each unit's breaker control circuitry to be 62 days. If the DG fails one of these Surveillances, the DG should be considered inoperable on both units, unless the cause of the failure can be directly related to only one unit. In addition, if the test is scheduled to be performed on Unit 2, and the Unit 2 TS allowance that provides an exception to performing the test is used (i.e., when Unit 2 is in MODE 4 or 5, or moving RECENTLY irradiated fuel assemblies in the secondary containment, the Note to Unit 2 SR 3.8.2.1 provides an exception to performing this test) or if it is not preferable to perform the test on a unit due to operational concerns (however time is not to exceed 62 days plus grace), then the test shall be performed synchronized to the Unit 3 4 kV emergency bus.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.20 (continued)

The 10 year Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 8). This SR is modified by two Notes. The reason for Note 1 is to minimize wear on the DG during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations. To minimize testing of the DGs, Note 2 allows a single test (instead of two tests, one for each unit) to satisfy the requirements for both units. This is allowed since the main purpose of the Surveillance can be met by performing the test on either unit. If a DG fails one of these Surveillances, a DG should be considered inoperable on both units, unless the cause of the failure can be directly related to only one unit.

SR 3.8.1.21

With the exception of this Surveillance, all other Surveillances of this Specification (SR 3.8.1.1 through SR 3.8.1.20) are applied only to the Unit 3 AC sources. This Surveillance is provided to direct that the appropriate Surveillances for the required Unit 2 AC sources are governed by the applicable Unit 2 Technical Specifications. Performance of the applicable Unit 2 Surveillances will satisfy Unit 2 requirements, as well as satisfying this Unit 3 Surveillance Requirement. Six exceptions are noted to the Unit 2 SRs of LCO 3.8.1. SR 3.8.1.8 is excepted when only one Unit 2 offsite circuit is required by the Unit 3 Specification, since there is not a second circuit to transfer to. SR 3.8.1.12, SR 3.8.1.13, SR 3.8.1.17, SR 3.8.1.18 (ECCS load block requirements only), and SR 3.8.1.19 are excepted since these SRs test the Unit 2 ECCS initiation signal, which is not needed for the AC sources to be OPERABLE on Unit 3.

The Frequency required by the applicable Unit 2 SR also governs performance of that SR for Unit 3.

recently

As Noted, if Unit 2 is in MODE 4 or 5, or moving irradiated fuel assemblies in the secondary containment, the Note to Unit 2 SR 3.8.2.1 is applicable. This ensures that a Unit 3 SR will not require a Unit 2 SR to be performed, when the

(continued)

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 4 and 5 and during movement of irradiated fuel assemblies in secondary containment ensures that:

RECENTLY

a. The facility 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 an inadvertent draindown of the vessel or a fuel handling accident.

INSERT Q

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 loss of 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, and 3 have no specific analyses in MODES 4 and 5. Worst case bounding events are deemed not credible in MODES 4 and 5 because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and corresponding stresses result in the probabilities of occurrences significantly reduced or eliminated, and 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, and 3, various deviations from the analysis assumptions and design requirements are allowed within the ACTIONS. This allowance is in recognition that

(continued)

BASES

LCO (continued)

offsite circuit. In addition some equipment that may be required by Unit 3 is powered from Unit 2 sources (e.g., Containment Atmospheric Dilution System, Standby Gas Treatment System, Emergency Service Water System, and Main Control Room Emergency Ventilation System). Therefore, qualified circuits between the offsite transmission network and the Unit 2 onsite Class 1E AC electrical power distribution subsystem(s), and the DG(s) (not necessarily different DG(s) from those being used to meet LCO 3.8.2.b requirements) capable of supplying power to the required Unit 2 subsystems of each of the required components must also be OPERABLE. Together, OPERABILITY of the required offsite circuit(s) and required DG(s) ensures the availability of sufficient AC sources to operate the plant in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents and reactor vessel draindown).

INSERT R

The qualified Unit 3 offsite circuit must be capable of maintaining rated frequency and voltage while connected to the respective Unit 3 4 kV emergency bus(es), and of accepting required loads during an accident. Qualified offsite circuits are those that are described in the UFSAR, Technical Specification Bases Section 3.8.1 and are part of the licensing basis for the unit. A Unit 3 offsite circuit consists of the incoming breaker and disconnect to the startup and emergency auxiliary transformer, the respective circuit path to the emergency auxiliary transformer and the circuit path to the Unit 3 4 kV emergency buses required by LCO 3.8.8, including feeder breakers to the required Unit 3 4 kV emergency buses. A qualified Unit 2 offsite circuit's requirements are the same as the Unit 3 circuit's requirements, except that the circuit path, including the feeder breakers, is to the Unit 2 4 kV emergency buses required to be OPERABLE by LCO 3.8.8.

The required DGs must be capable of starting, accelerating to rated speed and voltage, and connecting to their respective Unit 3 emergency bus on detection of bus undervoltage. This sequence must be accomplished within 10 seconds. Each DG must also be capable of accepting required loads within the assumed loading sequence intervals, and must continue to operate until offsite power can be restored to the 4 kV emergency buses. These capabilities are required to be met from a variety of initial conditions such as DG in standby with engine hot and DG in standby with engine at ambient conditions. Additional

(continued)

BASES

LCO (continued)

DG capabilities must be demonstrated to meet required Surveillances, e.g., capability of the DG to revert to standby status on an ECCS signal while operating in parallel test mode. Proper sequencing of loads is a required function for DG OPERABILITY. The necessary portions of the Emergency Service Water System are also required to provide appropriate cooling to each required DG.

The OPERABILITY requirements for the DG capable of supplying power to the Unit 2 powered equipment are the same as described above, except that the required DG must be capable of connecting to its respective Unit 2 4 kV emergency bus. (In addition, the Unit 2 ECCS initiation logic SRs are not applicable, as described in SR 3.8.2.2 Bases.)

It is acceptable for 4 kV emergency buses to be cross tied during shutdown conditions, permitting a single offsite power circuit to supply all required buses. No automatic transfer capability is required for offsite circuits to be considered OPERABLE.

APPLICABILITY

The AC sources are required to be OPERABLE in MODES 4 and 5 and during movement of irradiated fuel assemblies in the secondary containment to provide assurance that:

RECENTLY

a. Systems providing adequate coolant inventory makeup are available for the irradiated fuel assemblies in the core in case of an inadvertent draindown of the reactor vessel;

b. Systems needed to mitigate a fuel handling accident are available;

INSERT R

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

AC power requirements for MODES 1, 2, and 3 are covered in LCO 3.8.1.

(continued)

BASES (continued)

ACTIONS

INSERT V

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

A.1 and B.1

With one or more required offsite circuits inoperable, or with one DG inoperable, the remaining required sources may be capable of supporting sufficient required features (e.g., system, subsystem, division, component, or device) to allow continuation of CORE ALTERATIONS, RECENTLY IRRADIATED fuel movement, and operations with a potential for draining the reactor vessel. For example, if two or more 4 kV emergency buses are required per LCO 3.8.8, one 4 kV emergency bus with offsite power available may be capable of supplying sufficient required features. By the allowance of the option to declare required features inoperable that are not powered from offsite power (Required Action A.1) or capable of being powered by the required DG (Required Action B.1), appropriate restrictions can be implemented in accordance with the affected feature(s) LCOs' ACTIONS. Required features remaining powered from a qualified offsite power circuit, even if that circuit is considered inoperable because it is not powering other required features, are not declared inoperable by this Required Action. If a single DG is credited with meeting both LCO 3.8.2.d and one of the DG requirements of LCO 3.8.2.b, then the required features remaining capable of being powered by the DG are not declared inoperable by this Required Action, even if the DG is considered inoperable because it is not capable of powering other required features.

A.2.1, A.2.2, A.2.3, A.2.4, B.2.1, B.2.2, B.2.3, B.2.4, C.1, C.2, C.3, and C.4

With an offsite circuit not available to all required 4 kV emergency buses or one required DG inoperable, the option still exists to declare all required features inoperable

(continued)

BASES

ACTIONS

A.2.1, A.2.2, A.2.3, A.2.4, B.2.1, B.2.2, B.2.3, B.2.4, C.1, C.2, C.3, and C.4 (continued)

(per Required Actions A.1 and B.1). Since this option may involve undesired administrative efforts, the allowance for sufficiently conservative actions is made. With two or more required DGs inoperable, the minimum required diversity of AC power sources may not be available. It is, therefore, required to suspend CORE ALTERATIONS, movement of RECENTLY irradiated fuel assemblies in the secondary containment, and activities that could result in inadvertent draining of the reactor vessel.

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required AC sources and to continue this action until restoration is accomplished in order to provide the necessary AC power to the plant safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required AC electrical power sources should be completed as quickly as possible in order to minimize the time during which the plant safety systems may be without sufficient power.

Pursuant to LCO 3.0.6, the Distribution System ACTIONS would not be entered even if all AC sources to it are inoperable, resulting in de-energization. Therefore, the Required Actions of Condition A have been modified by a Note to indicate that when Condition A is entered with no AC power to any required 4 kV emergency bus, ACTIONS for LCO 3.8.8 must be immediately entered. This Note allows Condition A to provide requirements for the loss of the offsite circuit whether or not a required bus is de-energized. LCO 3.8.8 provides the appropriate restrictions for the situation involving a de-energized bus.

SURVEILLANCE REQUIREMENTS

SR 3.8.2.1

SR 3.8.2.1 requires the SRs from LCO 3.8.1 that are necessary for ensuring the OPERABILITY of the Unit 3 AC sources in other than MODES 1, 2, and 3. SR 3.8.1.8 is not

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.4.8 (continued)

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required DC electrical power subsystem from service, perturb the electrical distribution system, and challenge safety systems. Credit may be taken for unplanned events that satisfy the Surveillance. The DC batteries of the other unit are exempted from this restriction since they are required to be OPERABLE by both units and the Surveillance cannot be performed in the manner required by the Note without resulting in a dual unit shutdown.

SR 3.8.4.9

With the exception of this Surveillance, all other Surveillances of this Specification (SR 3.8.4.1 through SR 3.8.4.8) are applied only to the Unit 3 DC electrical power subsystems. This Surveillance is provided to direct that the appropriate Surveillances for the required Unit 2 DC electrical power subsystems are governed by the Unit 2 Technical Specifications. Performance of the applicable Unit 2 Surveillances will satisfy Unit 2 requirements, as well as satisfying this Unit 3 Surveillance Requirement.

The Frequency required by the applicable Unit 2 SR also governs performance of that SR for Unit 3. As Noted, if Unit 2 is in MODE 4 or 5, or moving RECENTLY irradiated fuel assemblies in the secondary containment, the Note to Unit 2 SR 3.8.5.1 is applicable. This ensures that a Unit 3 SR will not require a Unit 2 SR to be performed, when the Unit 2 Technical Specifications exempts performance of a Unit 2 SR. (However, as stated in the Unit 2 SR 3.8.5.1 Note, while performance of the SR is exempted, the SR still must be met.)

REFERENCES

1. UFSAR, Chapter 14.

2. "Proposed IEEE Criteria for Class 1E Electrical Systems for Nuclear Power Generating Stations," June 1969.

3. IEEE Standard 485, 1983.

(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 UFSAR, Chapter 14 (Ref. 1), assume that Engineered Safety Feature systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the diesel generators (DGs), 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.

RECENTLY

The OPERABILITY of the minimum DC electrical power sources during MODES 4 and 5 and during movement of irradiated fuel assemblies in secondary containment ensures that:

a. The facility 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 an inadvertent draindown of the vessel or a fuel handling accident.

INSERT R

The DC sources satisfy Criterion 3 of the NRC Policy Statement.

LCO

The Unit 3 DC electrical power subsystems, with each DC subsystem consisting of two 125 V station batteries in series, two battery chargers (one per battery), and the corresponding control equipment and interconnecting cabling supplying power to the associated bus, are required to be

(continued)

BASES

LCO (continued)

OPERABLE to support Unit 3 DC distribution subsystems required OPERABLE by LCO 3.8.8, "Distribution Systems—Shutdown." When the equipment required OPERABLE: 1) does not require 250 VDC from the DC electrical power subsystem; and 2) does not require 125 VDC from one of the two 125 V batteries of the DC electrical power subsystem, the Unit 3 DC electrical power subsystem requirements can be modified to only include one 125 V battery (the battery needed to provide power to required equipment), an associated battery charger, and the corresponding control equipment and interconnecting cabling supplying 125 V power to the associated bus. This exception is allowed only if all 250 VDC loads are removed from the associated bus. In addition, DC control power (which provides control power for the 4 kV load circuit breakers and the feeder breakers to the 4 kV emergency bus) for two of the four 4 kV emergency buses, as well as control power for two of the diesel generators, is provided by the Unit 2 DC electrical power subsystems. Therefore, the Unit 2 DC electrical power subsystems needed to support required components are also required to be OPERABLE. The Unit 2 DC electrical power subsystem OPERABILITY requirements are the same as those required for a Unit 3 DC electrical power subsystem. In addition, battery chargers (Unit 2 and Unit 3) can be powered from the opposite unit's AC source (as described in the Background section of the Bases for LCO 3.8.4, "DC Sources—Operating"), and be considered OPERABLE for the purpose of meeting this LCO.

This requirement ensures the availability of sufficient DC electrical 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, and inadvertent reactor vessel draindown). INSERT R

APPLICABILITY

The DC electrical power sources required to be OPERABLE in MODES 4 and 5 and during movement of irradiated fuel assemblies in the secondary containment provide assurance that:

a. Required features to provide adequate coolant inventory makeup are available for the RECENTLY irradiated fuel assemblies in the core in case of an inadvertent draindown of the reactor vessel;

(continued)

BASES

APPLICABILITY (continued)

b. Required features needed to mitigate a fuel handling accident are available;

INSERT R

c. Required features necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

INSERT T

The DC electrical power requirements for MODES 1, 2, and 3 are covered in LCO 3.8.4.

ACTIONS

INSERT V

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

A.1, A.2.1, A.2.2, A.2.3, and A.2.4

If more than one DC distribution subsystem is required according to LCO 3.8.8, the DC electrical power subsystems remaining OPERABLE with one or more DC electrical power subsystems inoperable may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS, fuel movement, and operations with a potential for draining the reactor vessel.

RECENTLY IRRADIATED

By allowance of the option to declare required features inoperable with associated DC electrical power subsystems inoperable, appropriate restrictions are implemented in accordance with the affected system LCOs' ACTIONS. However, in many instances, this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies in secondary containment, and any activities that could result in inadvertent draining of the reactor vessel).

RECENTLY

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.5.2 (continued)

RECENTLY

As Noted, if Unit 2 is in MODE 4 or 5, or moving irradiated fuel assemblies in the secondary containment, the Note to Unit 2 SR 3.8.5.1 is applicable. This ensures that a Unit 3 SR will not require a Unit 2 SR to be performed, when the Unit 2 Technical Specifications exempts performance of a Unit 2 SR. (However, as stated in the Unit 2 SR 3.8.5.1 Note, while performance of an SR is exempted, the SR still must be met.)

REFERENCES

1. UFSAR, Chapter 14.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.8 Distribution Systems—Shutdown

BASES

BACKGROUND

A description of the AC and DC electrical power distribution system is provided in the Bases for LCO 3.8.7, "Distribution Systems—Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident and transient analyses in the UFSAR, Chapter 14 (Ref. 1), assume Engineered Safety Feature (ESF) systems are OPERABLE. The AC and DC 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 AC and DC electrical power distribution system is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

RECENTLY

The OPERABILITY of the minimum AC and DC electrical power sources and associated power distribution subsystems during MODES 4 and 5 and during movement of irradiated fuel assemblies in the secondary containment ensures that:

a. The facility 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

INSERT R

c. Adequate power is provided to mitigate events postulated during shutdown, such as an inadvertent draindown of the vessel or a fuel handling accident.

INSERT S

The AC and DC electrical power distribution systems satisfy Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

Various combinations of subsystems, equipment, and components are required OPERABLE by other LCOs, depending on the specific plant condition. Implicit in those requirements is the required OPERABILITY of necessary support required features. This LCO explicitly requires energization of the portions of the Unit 3 electrical distribution system necessary to support OPERABILITY of Technical Specifications required systems, equipment, and components—both specifically addressed by their own LCO, and implicitly required by the definition of OPERABILITY. In addition some components that may be required by Unit 3 receive power through Unit 2 electrical power distribution subsystems (e.g., Standby Gas Treatment System, Main Control Room Emergency Ventilation System, and DC control power for two of the four 4 kV emergency buses, as well as control power for two of the diesel generators). Therefore, Unit 2 AC and DC electrical power distribution subsystems needed to support the required equipment must also be OPERABLE.

In addition, it is acceptable for required buses to be cross-tied during shutdown conditions, permitting a single source to supply multiple redundant buses, provided the source is capable of maintaining proper frequency (if required) and voltage.

Maintaining these portions of the distribution system energized ensures the availability of sufficient power to operate the plant in a safe manner to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents INSERT R and inadvertent reactor vessel draindown).

APPLICABILITY

The AC and DC electrical power distribution subsystems required to be OPERABLE in MODES 4 and 5 and during movement of RECENTLY irradiated fuel assemblies in the secondary containment provide assurance that:

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core in case of an inadvertent draindown of the reactor vessel;

b. Systems needed to mitigate a fuel handling accident INSERT R are available;

(continued)

BASES

APPLICABILITY (continued)

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The AC and DC electrical power distribution subsystem requirements for MODES 1, 2, and 3 are covered in LCO 3.8.7.

ACTIONS

INSERT V

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

A.1, A.2.1, A.2.2, A.2.3, A.2.4, and A.2.5

Although redundant required features may require redundant electrical power distribution subsystems to be OPERABLE, one OPERABLE distribution subsystem may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS, RECENTLY IRRADIATED fuel movement, and operations with a potential for draining the reactor vessel. By allowing the option to declare required features inoperable with associated electrical power distribution subsystems inoperable, appropriate restrictions are implemented in accordance with the affected distribution subsystem LCO's Required Actions. However, in many instances this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made, (i.e., to suspend CORE ALTERATIONS, movement of RECENTLY irradiated fuel assemblies in the secondary containment, and any activities that could result in inadvertent draining of the reactor vessel).

(continued)

ATTACHMENT 4

PEACH BOTTOM ATOMIC POWER STATION
UNITS 2 AND 3

Docket Nos. 50-277
50-278

License Nos. DPR-44
DPR-56

License Amendment Request
"PBAPS Alternative Source Term Implementation"

Retyped Technical Specification Pages

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3.3-54
3.3-58 to 59
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3.6-34 to 36
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<u>UNIT 3</u>
1.1-2
3.1-20 to 21
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3.8-19 to 23
3.8-33 to 36
3.8-44 to 45
5.0-12 to 14
5.0-17

1.1 Definitions (continued)

CHANNEL FUNCTIONAL TEST	A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY, including required alarm, interlock, display, and trip functions, and channel failure trips. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is tested.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components within the reactor vessel with the vessel head removed and fuel in the vessel. The following exceptions are not considered to be CORE ALTERATIONS: a. Movement of wide range neutron monitors, local power range monitors, traversing incore probes, or special movable detectors (including undervessel replacement); and b. Control rod movement, provided there are no fuel assemblies in the associated core cell. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites," or Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1989.

(continued)

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Standby Liquid Control (SLC) System

LCO 3.1.7 Two SLC subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Concentration of boron in solution > 9.82% weight.	A.1 Verify the concentration and temperature of boron in solution and pump suction piping temperature are within the limits of Figure 3.1.7-1. AND A.2 Restore concentration of boron in solution to ≤ 9.82% weight.	8 hours AND Once per 12 hours thereafter 72 hours AND 10 days from discovery of failure to meet the LCO
B. One SLC subsystem inoperable for reasons other than Condition A.	B.1 Restore SLC subsystem to OPERABLE status.	7 days AND 10 days from discovery of failure to meet the LCO

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Two SLC subsystems inoperable for reasons other than Condition A.	C.1	Restore one SLC subsystem to OPERABLE status.	8 hours
D. Required Action and associated Completion Time not met.	D.1	Be in MODE 3.	12 hours
	AND		
	D.2	Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.1	Verify level of sodium pentaborate solution in the SLC tank is ≥ 46%.	24 hours
SR 3.1.7.2	Verify temperature of sodium pentaborate solution is ≥ 53°F.	24 hours
SR 3.1.7.3	Verify temperature of pump suction piping is ≥ 53°F.	24 hours
SR 3.1.7.4	Verify continuity of explosive charge.	31 days

(continued)

Table 3.3.6.1-1 (page 3 of 3)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
5. Reactor Water Cleanup RWCU) System Isolation					
a. RWCU Flow - High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.7	≤ 125% rated flow (23.0 in-wc)
b. SLC System Initiation	1,2,3	1	H	SR 3.3.6.1.7	NA
c. Reactor Vessel Water Level - Low (Level 3)	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.7	≥ 1.0 inches
6. RHR Shutdown Cooling System Isolation					
a. Reactor Pressure - High	1,2,3	1	F	SR 3.3.6.1.3 SR 3.3.6.1.7	≤ 70.0 psig
b. Reactor Vessel Water Level - Low (Level 3)	3,4,5	2[(a)]	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.7	≥ 1.0 inches
7. Feedwater Recirculation Isolation					
a. Reactor Pressure - High	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.7	≤ 600 psig

(a) In MODES 4 and 5, provided RHR Shutdown Cooling System integrity is maintained, only one channel per trip system with an isolation signal available to one shutdown cooling pump suction isolation valve is required.

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

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level—Low (Level 3)	1,2,3, (a)	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.4 SR 3.3.6.2.5	≥ 1.0 inches
2. Drywell Pressure—High	1,2,3	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.4 SR 3.3.6.2.5	≤ 2.0 psig
3. Reactor Building Ventilation Exhaust Radiation—High	1,2,3, (a),(b)	2	SR 3.3.6.2.1 SR 3.3.6.2.3 SR 3.3.6.2.5	≤ 16.0 mR/hr
4. Refueling Floor Ventilation Exhaust Radiation—High	1,2,3, (a),(b)	2	SR 3.3.6.2.1 SR 3.3.6.2.3 SR 3.3.6.2.5	≤ 16.0 mR/hr

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

(b) During movement of recently irradiated fuel assemblies in secondary containment.

3.3 INSTRUMENTATION

3.3.7.1 Main Control Room Emergency Ventilation (MCREV) System Instrumentation

LCO 3.3.7.1 Two channels per trip system of the Control Room Air Intake Radiation—High Function shall be OPERABLE.

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

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Declare associated MCREV subsystems inoperable.	1 hour from discovery of loss of MCREV System initiation capability
	<u>AND</u>	
	A.2 Place channel in trip.	6 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.14	Verify combined MSIV leakage rate for all four main steam lines is ≤ 174 scfh, and ≤ 100 scfh for any one steam line, when tested at ≥ 25 psig.	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.3.15	Verify each 6 inch and 18 inch primary containment purge valve and each 18 inch primary containment exhaust valve is blocked to restrict opening greater than the required maximum opening angle.	24 months
SR 3.6.1.3.16	Replace the inflatable seal of each 6 inch and 18 inch primary containment purge valve and each 18 inch primary containment exhaust valve.	96 months

3.6 CONTAINMENT SYSTEMS

3.6.4.1 Secondary Containment

LCO 3.6.4.1 The secondary containment shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Secondary containment inoperable in MODE 1, 2, or 3.	A.1	Restore secondary containment to OPERABLE status.	4 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 3.	12 hours
	<u>AND</u>		
	B.2	Be in MODE 4.	36 hours
C. Secondary containment inoperable during movement of recently irradiated fuel assemblies in the secondary containment or during OPDRVs.	C.1	--------NOTE--------- LCO 3.0.3 is not applicable. --------------------- Suspend movement of recently irradiated fuel assemblies in the secondary containment.	Immediately
	<u>AND</u>		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	C.2 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1.1	Verify all secondary containment equipment hatches are closed and sealed.	31 days
SR 3.6.4.1.2	Verify one secondary containment access door in each access opening is closed.	31 days
SR 3.6.4.1.3	Verify each standby gas treatment (SGT) subsystem will draw down the secondary containment to ≥ 0.25 inch of vacuum water gauge in ≤ 15 minutes.	24 months on a STAGGERED TEST BASIS
SR 3.6.4.1.4	Verify each SGT subsystem can maintain ≥ 0.25 inch of vacuum water gauge in the secondary containment for 1 hour at a flow rate ≤ 10,500 cfm.	24 months on a STAGGERED TEST BASIS

3.6 CONTAINMENT SYSTEMS

3.6.4.2 Secondary Containment Isolation Valves (SCIVs)

LCO 3.6.4.2 Each SCIV shall be OPERABLE.

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

ACTIONS

-----NOTES-----

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

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

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

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

ACTIONS (continued)

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

3.6 CONTAINMENT SYSTEMS

3.6.4.3 Standby Gas Treatment (SGT) System

LCO 3.6.4.3 Two SGT subsystems shall be OPERABLE.

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

ACTIONS

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

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.3.1	Operate each SGT subsystem for ≥ 15 minutes.	31 days
SR 3.6.4.3.2	Perform required SGT filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.6.4.3.3	Verify each SGT subsystem actuates on an actual or simulated initiation signal.	24 months

3.7 PLANT SYSTEMS

3.7.4 Main Control Room Emergency Ventilation (MCREV) System

LCO 3.7.4 Two MCREV subsystems shall be OPERABLE.

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

ACTIONS

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

ACTIONS

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

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.20	-------------------NOTES------------------- 1. All DG starts may be preceded by an engine prelube period. 2. A single test at the specified Frequency will satisfy this Surveillance for both units. --- Verify, when started simultaneously from standby condition, each DG achieves, in ≤ 10 seconds, voltage ≥ 4160 V and frequency ≥ 58.8 Hz.	10 years
SR 3.8.1.21	-------------------NOTE------------------- When Unit 3 is in MODE 4 or 5, or moving recently irradiated fuel assemblies in the secondary containment, the Note to Unit 3 SR 3.8.2.1 is applicable. --- For required Unit 3 AC sources, the SRs of Unit 3 Specification 3.8.1, except SR 3.8.1.8 (when only one Unit 3 offsite circuit is required), SR 3.8.1.12, SR 3.8.1.13, SR 3.8.1.17, SR 3.8.1.18 (ECCS load block requirement only), and SR 3.8.1.19, are applicable.	In accordance with applicable SRs

3.8 ELECTRICAL POWER SYSTEMS

3.8.2 AC Sources—Shutdown

LCO 3.8.2 The following AC electrical power sources shall be OPERABLE:

a. One qualified circuit between the offsite transmission network and the Unit 2 onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.8, "Distribution Systems—Shutdown";

b. Two DGs each capable of supplying one Unit 2 onsite Class 1E AC electrical power distribution subsystem required by LCO 3.8.8;

c. One qualified circuit between the offsite transmission network and the Unit 3 onsite Class 1E AC electrical power distribution subsystem(s) needed to support the Unit 3 powered equipment required to be OPERABLE by LCO 3.6.4.3, "Standby Gas Treatment (SGT) System," and LCO 3.8.5, "DC Sources—Shutdown"; and

d. One DG capable of supplying one Unit 3 onsite Class 1E AC electrical power distribution subsystem needed to support the Unit 3 powered equipment required to be OPERABLE by:

1. LCO 3.6.4.3.

<u>OR</u>

2. LCO 3.8.5.

APPLICABILITY: MODES 4 and 5,
During movement of recently irradiated fuel assemblies in the secondary containment.

ACTIONS

-----------------------------------NOTE-----------------------------------
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required offsite circuits inoperable.	------------NOTE------------ Enter applicable Condition and Required Actions of LCO 3.8.8, with one or more required 4 kV emergency buses de-energized as a result of Condition A. ----------------------------	
	A.1 Declare affected required feature(s), with no offsite power available inoperable.	Immediately
	<u>OR</u>	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	A.2.2 Suspend movement of recently irradiated fuel assemblies in the secondary containment.	Immediately
	<u>AND</u>	
	A.2.3 Initiate action to suspend operations with a potential for draining the reactor vessel (OPDRVs).	Immediately
	<u>AND</u>	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2.4	Initiate action to restore required offsite power circuit(s) to OPERABLE status.	Immediately
B. One required DG inoperable.	B.1	Declare affected required feature(s) with no DG available inoperable.	Immediately
	OR		
	B.2.1	Suspend CORE ALTERATIONS	Immediately
	AND		
	B.2.2	Suspend movement of recently irradiated fuel assemblies in the secondary containment.	Immediately
	AND		
	B.2.3	Initiate action to suspend OPDRVs.	Immediately
	AND		
	B.2.4	Initiate action to restore required DGs to OPERABLE status.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Two or more required DGs inoperable.	C.1	Suspend CORE ALTERATIONS.	Immediately
	AND		
	C.2	Suspend movement of recently irradiated fuel assemblies in secondary containment.	Immediately
	AND		
	C.3	Initiate action to suspend OPDRVs.	Immediately
	AND		
	C.4	Initiate action to restore required DG(s) to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.4.9	------------------NOTE------------------- When Unit 3 is in MODE 4 or 5, or moving recently irradiated fuel assemblies in the secondary containment, the Note to Unit 3 SR 3.8.5.1 is applicable. ---	
	For required Unit 3 DC electrical power subsystems, the SRs of Unit 3 Specification 3.8.4 are applicable.	In accordance with applicable SRs

3.8 ELECTRICAL POWER SYSTEMS

3.8.5 DC Sources—Shutdown

LCO 3.8.5 The following DC electrical power subsystems shall be OPERABLE:

a. Unit 2 DC electrical power subsystems needed to support the DC electrical power distribution subsystem(s) required by LCO 3.8.8, "Distribution Systems—Shutdown"; and

b. Unit 3 DC electrical power subsystems needed to support the DC electrical power distribution subsystem(s) required by LCO 3.8.8, "Distribution Systems—Shutdown."

APPLICABILITY: MODES 4 and 5,
During movement of recently irradiated fuel assemblies in the secondary containment.

ACTIONS

------------------------------------NOTE------------------------------------
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required DC electrical power subsystems inoperable.	A.1 Declare affected required feature(s) inoperable.	Immediately
	OR	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.2 Suspend movement of recently irradiated fuel assemblies in the secondary containment. AND A.2.3 Initiate action to suspend operations with a potential for draining the reactor vessel. AND A.2.4 Initiate action to restore required DC electrical power subsystems to OPERABLE status.	Immediately Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.5.1	------------------NOTE------------------ The following SRs are not required to be performed: SR 3.8.4.7 and SR 3.8.4.8. -- For required Unit 2 DC electrical power subsystems, the following SRs are applicable: SR 3.8.4.1 SR 3.8.4.4 SR 3.8.4.7 SR 3.8.4.2 SR 3.8.4.5 SR 3.8.4.8. SR 3.8.4.3 SR 3.8.4.6	In accordance with applicable SRs
SR 3.8.5.2	------------------NOTE------------------ When Unit 3 is in MODE 4 or 5, or moving recently irradiated fuel assemblies in the secondary containment, the Note to Unit 3 SR 3.8.5.1 is applicable. -- For required Unit 3 DC electrical power subsystems, the SRs of Unit 3 Specification 3.8.4 are applicable.	In accordance with applicable SRs

3.8 ELECTRICAL POWER SYSTEMS

3.8.8 Distribution Systems—Shutdown

LCO 3.8.8 The necessary portions of the following AC and DC electrical power distribution subsystems shall be OPERABLE:

a. Unit 2 AC and DC electrical power distribution subsystems needed to support equipment required to be OPERABLE; and

b. Unit 3 AC and DC electrical power distribution subsystems needed to support equipment required to be OPERABLE by LCO 3.4.8, "Residual Heat Removal (RHR) Shutdown Cooling System—Cold Shutdown," LCO 3.5.2, "ECCS—Shutdown," LCO 3.6.4.3, "Standby Gas Treatment (SGT) System," LCO 3.8.2, "AC Sources—Shutdown," LCO 3.9.7, "RHR—High Water Level," and LCO 3.9.8, "RHR—Low Water Level."

APPLICABILITY: MODES 4 and 5,
During movement of recently irradiated fuel assemblies in the secondary containment.

ACTIONS

------------------------------------NOTE------------------------------------
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more required AC or DC electrical power distribution subsystems inoperable.	A.1	Declare associated supported required feature(s) inoperable.	Immediately
	<u>OR</u>		
	A.2.1	Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.2 Suspend handling of recently irradiated fuel assemblies in the secondary containment.	Immediately
	AND	
	A.2.3 Initiate action to suspend operations with a potential for draining the reactor vessel.	Immediately
	AND	
	A.2.4 Initiate actions to restore required AC and DC electrical power distribution subsystems to OPERABLE status.	Immediately
	AND	
	A.2.5 Declare associated required shutdown cooling subsystem(s) inoperable and not in operation.	Immediately

5.5 Programs and Manuals

5.5.7 <u>Ventilation Filter Testing Program (VFTP)</u> (continued)

1) Once per 12 months for standby service or after 720 hours of system operation; and,

2) After each complete or partial replacement of the HEPA filter train or charcoal adsorber filter; after any structural maintenance on the system housing; and, following significant painting, fire, or chemical release in any ventilation zone communicating with the system while it is in operation.

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

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

a. Demonstrate for the MCREV system that an inplace test of the HEPA filters shows a penetration and system bypass < 1.0% when tested in accordance with Regulatory Guide 1.52, Revision 2, Section 5c, and ASME N510-1989, Section 10, at the system flowrate specified below.

ESF Ventilation System	Flowrate (cfm)
Main Control Room Emergency Ventilation (MCREV) System	2700 to 3300

(continued)

* see mark-up for additional changes

5.5 Programs and Manuals

5.5.7 Ventilation Filter Testing Program (VFTP) (continued)

b. Demonstrate for the MCREV system that an inplace test of the charcoal adsorber shows a penetration and system bypass < 1.0% when tested in accordance with Regulatory Guide 1.52, Revision 2, Section 5d, and ASME N510-1989, Section 11, at the system flowrate specified below.

ESF Ventilation System	Flowrate (cfm)
MCREV System	2700 to 3300

for

c. Demonstrate of the MCREV system that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, Section 6b, shows the methyl iodide penetration less than the value specified below when tested in accordance with the laboratory testing criteria of ASTM D3803-1989 at a temperature of 30 degrees C [86 degrees F], face velocity, and the relative humidity specified below.

	MCREV System
Penetration (%)	15
Face Velocity (FPM)	57
Relative Humidity: (%)	95

(continued)

5.5 Programs and Manuals

5.5.7 Ventilation Filter Testing Program (VFTP) (continued)

1) Once per 12 months for standby service or after 720 hours of system operation; and,

2) After each complete or partial replacement of the HEPA filter train or charcoal adsorber filter; after any structural maintenance on the system housing; and, following significant painting, fire, or chemical release in any ventilation zone communicating with the system while it is in operation.

The Test described in Specification 5.5.7.d shall be performed once per 24 months.

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

a. Demonstrate for the MCREV system that an inplace test of the HEPA filters shows a penetration and system bypass < 1.0% when tested in accordance with Regulatory Guide 1.52, Revision 2, Section 5c, and ASME N510-1989, Section 10, at the system flowrate specified below.

ESF Ventilation System	Flowrate (cfm)
Main Control Room Emergency Ventilation (MCREV) System	2700 to 3300

(continued)

5.5 Programs and Manuals

5.5.7 Ventilation Filter Testing Program (VFTP) (continued)

1) Once per 12 months for standby service or after 720 hours of system operation; and,

2) After each complete or partial replacement of the HEPA filter train or charcoal adsorber filter; after any structural maintenance on the system housing; and, following significant painting, fire, or chemical release in any ventilation zone communicating with the system while it is in operation.

The test described in Specification 5.5.7.d shall be performed once per 24 months.

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

a. Demonstrate for the MCREV system that an inplace test of the HEPA filters shows a penetration and system bypass < 1.0% when tested in accordance with Regulatory Guide 1.52, Revision 2, Section 5c, and ASME N510-1989, Section 10, at the system flowrate specified below.

ESF Ventilation System	Flowrate (cfm)
Main Control Room Emergency Ventilation (MCREV) System	2700 to 3300

(continued)

5.5 Programs and Manuals

5.5.7 Ventilation Filter Testing Program (VFTP) (continued)

b. Demonstrate for the MCREV system that an inplace test of the charcoal adsorber shows a penetration and system bypass < 1.0% when tested in accordance with Regulatory Guide 1.52, Revision 2, Section 5d, and ASME N510-1989, Section 11, at the system flowrate specified below.

ESF Ventilation System	Flowrate (cfm)
MCREV System	2700 to 3300

c. Demonstrate for the MCREV system that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, Section 6b, shows the methyl iodide penetration less than the value specified below when tested in accordance with the laboratory testing criteria of ASTM D3803-1989 at a temperature of 30 degrees C [86 degrees F], face velocity, and the relative humidity specified below.

	MCREV System
Penetration (%)	15
Face Velocity (FPM)	57
Relative Humidity: (%)	95

(continued)

5.5 Programs and Manuals

5.5.7 Ventilation Filter Testing Program (VFTP) (continued)

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

ESF Ventilation System	Delta P (inches wg)	Flowrate (cfm)
SGT System	< 3.9	7200 to 8800
MCREV System	< 8	2700 to 3300

5.5.8 Explosive Gas Monitoring Program

This program provides controls for potentially explosive gas mixtures contained downstream of the off-gas recombiners.

The program shall include:

a. The limit for the concentration of hydrogen downstream of the off-gas recombiners and a surveillance program to ensure the limit is maintained. This limit shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion);

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas Monitoring Program surveillance frequencies.

5.5.9 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

(continued)

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

1. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or

2. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

3. A required system redundant to support system(s) for the supported systems (b.1) and (b.2) above is also inoperable.

c. The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.12 Primary Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exception to NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Section 10.2:

a. MSIV leakage is excluded from the combined total of 0.6 L_a for the Type B and C tests.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 49.1 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.7% of primary containment air weight per day.

Leakage Rate acceptance criteria are:

a. Primary Containment leakage rate acceptance criterion is $\leq$ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq$ 0.60 L_a for the Type B and Type C tests and $\leq$ 0.75 L_a for Type A tests;

(continued)

1.1 Definitions (continued)

CHANNEL FUNCTIONAL TEST	A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY, including required alarm, interlock, display, and trip functions, and channel failure trips. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is tested.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components within the reactor vessel with the vessel head removed and fuel in the vessel. The following exceptions are not considered to be CORE ALTERATIONS: a. Movement of wide range neutron monitors, local power range monitors, traversing incore probes, or special movable detectors (including undervessel replacement); and b. Control rod movement, provided there are no fuel assemblies in the associated core cell. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites," or Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1989.

(continued)

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Standby Liquid Control (SLC) System

LCO 3.1.7 Two SLC subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Concentration of boron in solution > 9.82% weight.	A.1 Verify the concentration and temperature of boron in solution and pump suction piping temperature are within the limits of Figure 3.1.7-1.	8 hours AND Once per 12 hours thereafter
	AND	
	A.2 Restore concentration of boron in solution to ≤ 9.82% weight.	72 hours AND 10 days from discovery of failure to meet the LCO
B. One SLC subsystem inoperable for reasons other than Condition A.	B.1 Restore SLC subsystem to OPERABLE status.	7 days AND 10 days from discovery of failure to meet the LCO

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Two SLC subsystems inoperable for reasons other than Condition A.	C.1 Restore one SLC subsystem to OPERABLE status.	8 hours
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.1	Verify level of sodium pentaborate solution in the SLC tank is ≥ 46%.	24 hours
SR 3.1.7.2	Verify temperature of sodium pentaborate solution is ≥ 53°F.	24 hours
SR 3.1.7.3	Verify temperature of pump suction piping is ≥ 53°F.	24 hours
SR 3.1.7.4	Verify continuity of explosive charge.	31 days

(continued)

Table 3.3.6.1-1 (page 3 of 3)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
5. Reactor Water Cleanup (RWCU) System Isolation					
a. RWCU Flow—High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.7	≤ 125% rated flow (23.0 in-wc)
b. SLC System Initiation	1,2,3	1	H	SR 3.3.6.1.7	NA
c. Reactor Vessel Water Level—Low (Level 3)	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.7	≥ 1.0 inches
6. RHR Shutdown Cooling System Isolation					
a. Reactor Pressure—High	1,2,3	1	F	SR 3.3.6.1.3 SR 3.3.6.1.7	≤ 70.0 psig
b. Reactor Vessel Water Level—Low (Level 3)	3,4,5	2[(a)]	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.7	≥ 1.0 inches
7. Feedwater Recirculation Isolation					
a. Reactor Pressure—High	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.7	≤ 600 psig

(a) In MODES 4 and 5, provided RHR Shutdown Cooling System integrity is maintained, only one channel per trip system with an isolation signal available to one shutdown cooling pump suction isolation valve is required.

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

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level—Low (Level 3)	1,2,3, (a)	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.4 SR 3.3.6.2.5	≥ 1.0 inches
2. Drywell Pressure—High	1,2,3	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.4 SR 3.3.6.2.5	≤ 2.0 psig
3. Reactor Building Ventilation Exhaust Radiation—High	1,2,3, (a),(b)	2	SR 3.3.6.2.1 SR 3.3.6.2.3 SR 3.3.6.2.5	≤ 16.0 mR/hr
4. Refueling Floor Ventilation Exhaust Radiation—High	1,2,3, (a),(b)	2	SR 3.3.6.2.1 SR 3.3.6.2.3 SR 3.3.6.2.5	≤ 16.0 mR/hr

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

(b) During movement of recently irradiated fuel assemblies in secondary containment.

3.3 INSTRUMENTATION

3.3.7.1 Main Control Room Emergency Ventilation (MCREV) System Instrumentation

LCO 3.3.7.1 Two channels per trip system of the Control Room Air Intake Radiation—High Function shall be OPERABLE.

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

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Declare associated MCREV subsystems inoperable.	1 hour from discovery of loss of MCREV System initiation capability
	<u>AND</u>	
	A.2 Place channel in trip.	6 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.14	Verify combined MSIV leakage rate for all four main steam lines is ≤ 174 scfh, and ≤ 100 scfh for any one steam line, when tested at ≥ 25 psig.	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.3.15	Verify each 6 inch and 18 inch primary containment purge valve and each 18 inch primary containment exhaust valve is blocked to restrict opening greater than the required maximum opening angle.	24 months
SR 3.6.1.3.16	Replace the inflatable seal of each 6 inch and 18 inch primary containment purge valve and each 18 inch primary containment exhaust valve.	96 months

3.6 CONTAINMENT SYSTEMS

3.6.4.1 Secondary Containment

LCO 3.6.4.1 The secondary containment shall be OPERABLE.

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

ACTIONS

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	C.2 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1.1	Verify all secondary containment equipment hatches are closed and sealed.	31 days
SR 3.6.4.1.2	Verify one secondary containment access door in each access opening is closed.	31 days
SR 3.6.4.1.3	Verify each standby gas treatment (SGT) subsystem will draw down the secondary containment to ≥ 0.25 inch of vacuum water gauge in ≤ 15 minutes.	24 months on a STAGGERED TEST BASIS
SR 3.6.4.1.4	Verify each SGT subsystem can maintain ≥ 0.25 inch of vacuum water gauge in the secondary containment for 1 hour at a flow rate ≤ 10,500 cfm.	24 months on a STAGGERED TEST BASIS

3.6 CONTAINMENT SYSTEMS

3.6.4.2 Secondary Containment Isolation Valves (SCIVs)

LCO 3.6.4.2 Each SCIV shall be OPERABLE.

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

ACTIONS

-----NOTES-----

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

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

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

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

ACTIONS (continued)

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

3.6 CONTAINMENT SYSTEMS

3.6.4.3 Standby Gas Treatment (SGT) System

LCO 3.6.4.3 Two SGT subsystems shall be OPERABLE.

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

ACTIONS

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

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.3.1	Operate each SGT subsystem for ≥ 15 minutes.	31 days
SR 3.6.4.3.2	Perform required SGT filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.6.4.3.3	Verify each SGT subsystem actuates on an actual or simulated initiation signal.	24 months

3.7 PLANT SYSTEMS

3.7.4 Main Control Room Emergency Ventilation (MCREV) System

LCO 3.7.4 Two MCREV subsystems shall be OPERABLE.

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

ACTIONS

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

ACTIONS

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

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.20	-------------------NOTES------------------- 1. All DG starts may be preceded by an engine prelube period. 2. A single test at the specified Frequency will satisfy this Surveillance for both units. -- Verify, when started simultaneously from standby condition, each DG achieves, in ≤ 10 seconds, voltage ≥ 4160 V and frequency ≥ 58.8 Hz.	10 years
SR 3.8.1.21	-------------------NOTE------------------- When Unit 2 is in MODE 4 or 5, or moving recently irradiated fuel assemblies in the secondary containment, the Note to Unit 2 SR 3.8.2.1 is applicable. -- For required Unit 2 AC sources, the SRs of Unit 2 Specification 3.8.1, except SR 3.8.1.8 (when only one Unit 2 offsite circuit is required), SR 3.8.1.12, SR 3.8.1.13, SR 3.8.1.17, SR 3.8.1.18 (ECCS load block requirement only), and SR 3.8.1.19, are applicable.	In accordance with applicable SRs

3.8 ELECTRICAL POWER SYSTEMS

3.8.2 AC Sources — Shutdown

LCO 3.8.2 The following AC electrical power sources shall be OPERABLE:

a. One qualified circuit between the offsite transmission network and the Unit 3 onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.8, "Distribution Systems — Shutdown";

b. Two DGs each capable of supplying one Unit 3 onsite Class 1E AC electrical power distribution subsystem required by LCO 3.8.8;

c. One qualified circuit between the offsite transmission network and the Unit 2 onsite Class 1E AC electrical power distribution subsystem(s) needed to support the Unit 2 powered equipment required to be OPERABLE by LCO 3.6.4.3, "Standby Gas Treatment (SGT) System", LCO 3.7.4, "Main Control Room Emergency Ventilation (MCREV) System," and LCO 3.8.5, "DC Sources — Shutdown"; and

d. The DG(s) capable of supplying one subsystem of each of the Unit 2 powered equipment required to be OPERABLE by LCO 3.6.4.3, LCO 3.7.4, and LCO 3.8.5.

APPLICABILITY: MODES 4 and 5,
During movement of recently irradiated fuel assemblies in the secondary containment.

ACTIONS

-----------------------------------NOTE-----------------------------------
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required offsite circuits inoperable.	------------NOTE------------ Enter applicable Condition and Required Actions of LCO 3.8.8, with one or more required 4 kV emergency buses de-energized as a result of Condition A. ----------------------------	
	A.1 Declare affected required feature(s), with no offsite power available inoperable.	Immediately
	OR	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2.2 Suspend movement of recently irradiated fuel assemblies in the secondary containment.	Immediately
	AND	
	A.2.3 Initiate action to suspend operations with a potential for draining the reactor vessel (OPDRVs).	Immediately
	AND	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2.4	Initiate action to restore required offsite power circuit(s) to OPERABLE status.	Immediately
B. One required DG inoperable.	B.1	Declare affected required feature(s) with no DG available inoperable.	Immediately
	<u>OR</u>		
	B.2.1	Suspend CORE ALTERATIONS	Immediately
	<u>AND</u>		
	B.2.2	Suspend movement of recently irradiated fuel assemblies in the secondary containment.	Immediately
	<u>AND</u>		
	B.2.3	Initiate action to suspend OPDRVs.	Immediately
	<u>AND</u>		
	B.2.4	Initiate action to restore required DGs to OPERABLE status.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Two or more required DGs inoperable.	C.1	Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>		
	C.2	Suspend movement of recently irradiated fuel assemblies in secondary containment.	Immediately
	<u>AND</u>		
	C.3	Initiate action to suspend OPDRVs.	Immediately
	<u>AND</u>		
	C.4	Initiate action to restore required DG(s) to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.4.9	-------------------NOTE------------------- When Unit 2 is in MODE 4 or 5, or moving recently irradiated fuel assemblies in the secondary containment, the Note to Unit 2 SR 3.8.5.1 is applicable. ---	
	For required Unit 2 DC electrical power subsystems, the SRs of Unit 2 Specification 3.8.4 are applicable.	In accordance with applicable SRs

3.8 ELECTRICAL POWER SYSTEMS

3.8.5 DC Sources — Shutdown

LCO 3.8.5 The following DC electrical power subsystems shall be OPERABLE:

a. Unit 3 DC electrical power subsystems needed to support the DC electrical power distribution subsystem(s) required by LCO 3.8.8, "Distribution Systems — Shutdown"; and

b. Unit 2 DC electrical power subsystems needed to support the DC electrical power distribution subsystem(s) required by LCO 3.8.8, "Distribution Systems — Shutdown."

APPLICABILITY: MODES 4 and 5,
During movement of recently irradiated fuel assemblies in the secondary containment.

ACTIONS

------------------------------------NOTE------------------------------------
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more required DC electrical power subsystems inoperable.	A.1	Declare affected required feature(s) inoperable.	Immediately
	<u>OR</u>		
	A.2.1	Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.2 Suspend movement of recently irradiated fuel assemblies in the secondary containment. **AND**	Immediately
	A.2.3 Initiate action to suspend operations with a potential for draining the reactor vessel. **AND**	Immediately
	A.2.4 Initiate action to restore required DC electrical power subsystems to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.5.1	-------------------NOTE------------------- The following SRs are not required to be performed: SR 3.8.4.7 and SR 3.8.4.8. -- For required Unit 3 DC electrical power subsystems, the following SRs are applicable: SR 3.8.4.1 SR 3.8.4.4 SR 3.8.4.7 SR 3.8.4.2 SR 3.8.4.5 SR 3.8.4.8. SR 3.8.4.3 SR 3.8.4.6	In accordance with applicable SRs
SR 3.8.5.2	-------------------NOTE------------------- When Unit 2 is in MODE 4 or 5, or moving recently irradiated fuel assemblies in the secondary containment, the Note to Unit 2 SR 3.8.5.1 is applicable. -- For required Unit 2 DC electrical power subsystems, the SRs for Unit 2 Specification 3.8.4 are applicable.	In accordance with applicable SRs

3.8 ELECTRICAL POWER SYSTEMS

3.8.8 Distribution Systems—Shutdown

LCO 3.8.8 The necessary portions of the following AC and DC electrical power distribution subsystems shall be OPERABLE:

a. Unit 3 AC and DC electrical power distribution subsystems needed to support equipment required to be OPERABLE; and

b. Unit 2 AC and DC electrical power distribution subsystems needed to support equipment required to be OPERABLE by LCO 3.4.8, "Residual Heat Removal (RHR) Shutdown Cooling System—Cold Shutdown," LCO 3.5.2, "ECCS—Shutdown," LCO 3.6.4.3, "Standby Gas Treatment (SGT) System," LCO 3.7.4, "Main Control Room Emergency Ventilation (MCREV) System," LCO 3.8.2, "AC Sources—Shutdown," LCO 3.9.7, "RHR—High Water Level," and LCO 3.9.8, "RHR—Low Water Level."

APPLICABILITY: MODES 4 and 5,
During movement of recently irradiated fuel assemblies in the secondary containment.

ACTIONS

------------------------------------NOTE------------------------------------
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required AC or DC electrical power distribution subsystems inoperable.	A.1 Declare associated supported required feature(s) inoperable. OR A.2.1 Suspend CORE ALTERATIONS. AND	Immediately Immediately (continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.2 Suspend handling of recently irradiated fuel assemblies in the secondary containment.	Immediately
	<u>AND</u>	
	A.2.3 Initiate action to suspend operations with a potential for draining the reactor vessel.	Immediately
	<u>AND</u>	
	A.2.4 Initiate actions to restore required AC and DC electrical power distribution subsystems to OPERABLE status.	Immediately
	<u>AND</u>	
	A.2.5 Declare associated required shutdown cooling subsystem(s) inoperable and not in operation.	Immediately

5.5 Programs and Manuals

5.5.7 Ventilation Filter Testing Program (VFTP) (continued)

1) Once per 12 months for standby service or after 720 hours of system operation; and,

2) After each complete or partial replacement of the HEPA filter train or charcoal adsorber filter; after any structural maintenance on the system housing; and, following significant painting, fire, or chemical release in any ventilation zone communicating with the system while it is in operation.

The test described in Specification 5.5.7.d shall be performed once per 24 months.

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

a. Demonstrate for the MCREV system that an inplace test of the HEPA filters shows a penetration and system bypass < 1.0% when tested in accordance with Regulatory Guide 1.52, Revision 2, Section 5c, and ASME N510-1989, Section 10, at the system flowrate specified below.

ESF Ventilation System	Flowrate (cfm)
Main Control Room Emergency Ventilation (MCREV) System	2700 to 3300

(continued)

5.5 Programs and Manuals

5.5.7 Ventilation Filter Testing Program (VFTP) (continued)

1) Once per 12 months for standby service or after 720 hours of system operation; and,

2) After each complete or partial replacement of the HEPA filter train or charcoal adsorber filter; after any structural maintenance on the system housing; and, following significant painting, fire, or chemical release in any ventilation zone communicating with the system while it is in operation.

The Test described in Specification 5.5.7.d shall be performed once per 24 months.

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

a. Demonstrate for the MCREV system that an inplace test of the HEPA filters shows a penetration and system bypass < 1.0% when tested in accordance with Regulatory Guide 1.52, Revision 2, Section 5c, and ASME N510-1989, Section 10, at the system flowrate specified below.

ESF Ventilation System	Flowrate (cfm)
Main Control Room Emergency Ventilation (MCREV) System	2700 to 3300

(continued)

5.5 Programs and Manuals

5.5.7 Ventilation Filter Testing Program (VFTP) (continued)

b. Demonstrate for the MCREV system that an inplace test of the charcoal adsorber shows a penetration and system bypass < 1.0% when tested in accordance with Regulatory Guide 1.52, Revision 2, Section 5d, and ASME N510-1989, Section 11, at the system flowrate specified below.

ESF Ventilation System	Flowrate (cfm)
MCREV System	2700 to 3300

c. Demonstrate for the MCREV system that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, Section 6b, shows the methyl iodide penetration less than the value specified below when tested in accordance with the laboratory testing criteria of ASTM D3803-1989 at a temperature of 30 degrees C [86 degrees F], face velocity, and the relative humidity specified below.

	MCREV System
Penetration (%)	15
Face Velocity (FPM)	57
Relative Humidity: (%)	95

(continued)

5.5 Programs and Manuals

5.5.7 Ventilation Filter Testing Program (VFTP) (continued)

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

ESF Ventilation System	Delta P (inches wg)	Flowrate (cfm)
SGT System	< 3.9	7200 to 8800
MCREV System	< 8	2700 to 3300

5.5.8 Explosive Gas Monitoring Program

This program provides controls for potentially explosive gas mixtures contained downstream of the off-gas recombiners.

The program shall include:

a. The limit for the concentration of hydrogen downstream of the off-gas recombiners and a surveillance program to ensure the limit is maintained. This limit shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion);

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas Monitoring Program surveillance frequencies.

5.5.9 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

(continued)

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

1. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or

2. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

3. A required system redundant to support system(s) for the supported systems (b.1) and (b.2) above is also inoperable.

c. The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.12 Primary Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exceptions to NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J":

a. Section 10.2: MSIV leakage is excluded from the combined total of 0.6 L_a for the Type B and C tests.

b. Section 9.2.3: The first Type A test performed after the December, 1991 Type A test shall be performed no later than December, 2006.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 49.1 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.7% of primary containment air weight per day.

Leakage Rate acceptance criteria are:

a. Primary Containment leakage rate acceptance criterion is $\leq$ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq$ 0.60 L_a for the Type B and Type C tests and $\leq$ 0.75 L_a for Type A tests;

(continued)

ATTACHMENT 5

PEACH BOTTOM ATOMIC POWER STATION
UNITS 2 AND 3

Docket Nos. 50-277
50-278

License Nos. DPR-44
DPR-56

License Amendment Request
"PBAPS Alternative Source Term Implementation"

Retyped Technical Specification Bases Pages
(For information only)

UNITS 2 & 3
B 2.0-5 to 2.07
B 2.0-9 to 2.0-10
B 3.1-37
B 3.1-39 to 41
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B 3.1-47
B 3.3-147 to 150
B 3.3-156
B 3.3-174
B 3.3-182
B 3.4-29 to 32
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B 3.6-84 to 90
B 3.7-16 to 20
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B 3.7-24
B 3.7-29 to 30
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B 3.8-40
B 3.8-42 to 45
B 3.8-70
B 3.8-72 to 74
B 3.8-76
B 3.8-94 to 96
B 3.9-17 to 19

BASES (continued)

SAFETY LIMITS	The reactor core SLs are established to protect the integrity of the fuel clad barrier to the release of radioactive materials to the environs. SL 2.1.1.1 and SL 2.1.1.2 ensure that the core operates within the fuel design criteria. SL 2.1.1.3 ensures that the reactor vessel water level is greater than the top of the active irradiated fuel in order to prevent elevated clad temperatures and resultant clad perforations.
APPLICABILITY	SLs 2.1.1.1, 2.1.1.2, and 2.1.1.3 are applicable in all MODES.

SAFETY LIMIT VIOLATIONS

2.2.1

If any SL is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. 5).

2.2.2

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

2.2.3

If any SL is violated, the senior management of the nuclear plant and the utility shall be notified within 24 hours. The 24 hour period provides time for plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to the senior management.

(continued)

BASES

SAFETY LIMIT VIOLATIONS (continued)

2.2.4

If any SL is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Ref. 7). A copy of the report shall also be provided to the senior management of the nuclear plant and the utility.

2.2.5

If any SL is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.

REFERENCES

1. GE Nuclear Energy 23A7188, Revision 1, September 1992.
2. ABB Atom Report BR 90-004, October 1990.
3. ANF-90-133 (P), Revision 2, August 1992.
4. NEDE-24011-P-A-10, February 1991.
5. 10 CFR 50.72.
6. 10 CFR 50.67.
7. 10 CFR 50.73.

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on reactor steam dome pressure protects 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. Establishing an upper limit on reactor steam dome pressure ensures continued RCS integrity with regard to pressure excursions. Per the UFSAR (Ref. 1), the reactor coolant pressure boundary (RCPB) shall be designed with sufficient margin to ensure that the design conditions are not exceeded during normal operation and abnormal operational transients.

During normal operation and abnormal operational transients, 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 are hydrostatically tested at 125% of design pressure, in accordance with ASME Code requirements, prior to initial operation when there is no fuel in the core. Any further hydrostatic testing with fuel in the core may be done under LCO 3.10.1, "Inservice Leak and Hydrostatic Testing Operation." 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 reducing the number of protective barriers designed to prevent radioactive releases from exceeding the limits specified in 10 CFR 50.67, "Accident Source Term," (Ref. 4). If this occurred in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere.

APPLICABLE SAFETY ANALYSES

The RCS safety/relief valves and the Reactor Protection System Reactor Pressure—High Function have settings established to ensure that the RCS pressure SL will not be exceeded.

(continued)

BASES

SAFETY LIMIT VIOLATIONS (continued)

2.2.2

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR 50.67, "Accident Source Term," limits (Ref. 4). Therefore, it is required to insert all insertable control rods and restore compliance with the SL within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also assures that the probability of an accident occurring during the period is minimal.

2.2.3

If any SL is violated, the senior management of the nuclear plant and the utility, shall be notified within 24 hours. The 24 hour period provides time for plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to the senior management.

2.2.4

If any SL is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Ref. 8). A copy of the report shall also be provided to the senior management of the nuclear plant and the utility.

2.2.5

If any SL is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.

REFERENCES

1. UFSAR, Section 1.5.2.2.

2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000.

(continued)

BASES

REFERENCES (continued)

3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IW-5000.
4. 10 CFR 50.67.
5. ASME, Boiler and Pressure Vessel Code, Section III, 1965 Edition, including Addenda to winter of 1965.
6. ASME, Boiler and Pressure Vessel Code, Section III, 1980 Edition, Addenda to winter of 1981.
7. 10 CFR 50.72.
8. 10 CFR 50.73.

BASES

ACTIONS

B.1 and B.2 (continued)

control rods has less impact on control rod worth than withdrawals have. Required Action B.1 is modified by a Note which allows the RWM to be bypassed to allow the affected control rods to be returned to their correct position.

LCO 3.3.2.1 requires verification of control rod movement by a second licensed operator or a qualified member of the technical staff.

When nine or more OPERABLE control rods are not in compliance with BPWS, the reactor mode switch must be placed in the shutdown position within 1 hour. With the mode switch in shutdown, the reactor is shut down, and as such, does not meet the applicability requirements of this LCO. The allowed Completion Time of 1 hour is reasonable to allow insertion of control rods to restore compliance, and is appropriate relative to the low probability of a CRDA occurring with the control rods out of sequence.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

The control rod pattern is verified to be in compliance with the BPWS at a 24 hour Frequency to ensure the assumptions of the CRDA analyses are met. The 24 hour Frequency was developed considering that the primary check on compliance with the BPWS is performed by the RWM (LCO 3.3.2.1), which provides control rod blocks to enforce the required sequence and is required to be OPERABLE when operating at ≤ 10% RTP.

REFERENCES

1. NEDE-24011-P-A-10-US, "General Electric Standard Application for Reactor Fuel, Supplement for United States," Section 2.2.3.1, February 1991.
2. Letter (BWROG-8644) from T. Pickens (BWROG) to G. C. Lainas (NRC), "Amendment 17 to General Electric Licensing Topical Report NEDE-24011-P-A."
3. UFSAR, Section 14.6.2.3.
4. NUREG-0800, Section 15.4.9, Revision 2, July 1981.
5. 10 CFR 50.67.

(continued)

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Standby Liquid Control (SLC) System

BASES

BACKGROUND

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

The SLC System is also used to maintain suppression pool pH at or above 7 following a loss of coolant accident (LOCA) involving significant fission product releases. Maintaining suppression pool pH levels at or above 7 following an accident ensures that iodine will be retained in the suppression pool water.

Reference 1 requires a SLC System with a minimum flow capacity and boron content equivalent in control capacity to 86 gpm of 13 weight percent sodium pentaborate solution. Natural sodium pentaborate solution is 19.8% atom Boron-10. Therefore, the system parameters of concern, boron concentration (C), SLC pump flow rate (Q), and Boron-10 enrichment (E), may be expressed as a multiple of ratios. The expression is as follows:

$$\frac{C}{13\%\ \text{weight}} \times \frac{Q}{86\ \text{gpm}} \times \frac{E}{19.8\%\ \text{atom}}$$

If the product of this expression is ≥ 1, then the SLC System satisfies the criteria of Reference 1. As such, the equation forms the basis for acceptance criteria for the surveillances of concentration, flow rate, and boron enrichment and is presented in Table 3.1.7-1.

The SLC System consists of a boron solution storage tank, two positive displacement pumps, two explosive valves that are provided in parallel for redundancy, and associated piping and valves used to transfer borated water from the storage tank to the reactor pressure vessel (RPV). The borated solution is discharged near the bottom of the core shroud, where it then mixes with the cooling water rising through the core. A smaller tank containing demineralized water is provided for testing purposes.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The SLC System is manually initiated from the main control room, as directed by the emergency operating procedures, if the operator believes the reactor cannot be shut down, or kept shut down, with the control rods. The SLC System is used in the event that enough control rods cannot be inserted to accomplish shutdown and cooldown in the normal manner. The SLC System injects borated water into the reactor core to add negative reactivity to compensate for all of the various reactivity effects that could occur during plant operations. To meet this objective, it is necessary to inject a quantity of boron, which produces a concentration of 660 ppm of natural boron, in the reactor coolant at 68°F. To allow for potential leakage and imperfect mixing in the reactor system, an additional amount of boron equal to 25% of the amount cited above is added (Ref. 2). The minimum mass of Boron-10 (162.7 lbm) needed for injection is calculated such that the required quantity is achieved accounting for dilution in the RPV with normal water level and including the water volume in the residual heat removal shutdown cooling piping and in the recirculation loop piping. This quantity of borated solution is the amount that is above the pump suction shutoff level in the boron solution storage tank. No credit is taken for the portion of the tank volume that cannot be injected. The maximum concentration of sodium pentaborate listed in Table 3.1.7-1 has been established to ensure that the solution saturation temperature does not exceed 43°F. The sodium pentaborate solution in the SLC System is also used, post-LOCA, to maintain ECCS fluid pH above 7. The system parameters used in the calculation are the Boron-10 minimum mass of 162.7 lbm, and an upper bound Boron-10 enrichment of 65%.

The SLC System satisfies Criterion 4 of the NRC Policy Statement.

LCO

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

(continued)

BASES (continued)

APPLICABILITY

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

In MODES 1, 2, and 3, the SLC System must be OPERABLE to ensure that offsite doses remain within 10 CRF 50.67 (Ref. 3) limits following a LOCA involving significant fission product releases. The SLC System is designed to maintain suppression pool pH at or above 7 following a LOCA to ensure that iodine will be retained in the suppression pool water.

ACTIONS

A.1 and A.2

If the boron solution concentration is > 9.82% weight but the concentration and temperature of boron in solution and pump suction piping temperature are within the limits of Figure 3.1.7-1, operation is permitted for a limited period since the SLC subsystems are capable of performing the intended function. It is not necessary under these conditions to declare both SLC subsystems inoperable since the SLC subsystems are capable of performing their intended function.

The concentration and temperature of boron in solution and pump suction piping temperature must be verified to be within the limits of Figure 3.1.7-1 within 8 hours and once per 12 hours thereafter (Required Action A.1). The temperature versus concentration curve of Figure 3.1.7-1 ensures a 10°F margin will be maintained above the saturation temperature. This verification ensures that boron does not precipitate out of solution in the storage tank or in the pump suction piping due to low boron solution temperature (below the saturation temperature for the given concentration). The Completion Time for performing Required Action A.1 is considered acceptable given the low probability of a Design Basis Accident (DBA) or transient occurring concurrent with the failure of the control rods to shut down the reactor and operating experience which has shown there are relatively slow variations in the measured parameters of concentration and temperature over these time periods.

(continued)

BASES

ACTIONS

D.1 (continued)

brought to MODE 3 within 12 hours and MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1, SR 3.1.7.2, and SR 3.1.7.3

SR 3.1.7.1 through SR 3.1.7.3 are 24 hour Surveillances verifying certain characteristics of the SLC System (e.g., the level and temperature of the borated solution in the storage tank), thereby ensuring SLC System OPERABILITY without disturbing normal plant operation. These Surveillances ensure that the proper borated solution level and temperature, including the temperature of the pump suction piping, are maintained. Maintaining a minimum specified borated solution temperature is important in ensuring that the boron remains in solution and does not precipitate out in the storage tank or in the pump suction piping. The temperature limit specified in SR 3.1.7.2 and SR 3.1.7.3 and the maximum sodium pentaborate concentration specified in Table 3.1.7-1 ensures that a 10°F margin will be maintained above the saturation temperature. Control room alarms for low SLC storage tank temperature and low SLC System piping temperature are available and are set at 55°F. As such, SR 3.1.7.2 and SR 3.1.7.3 may be satisfied by verifying the absence of low temperature alarms for the SLC storage tank and SLC System piping. The 24 hour Frequency is based on operating experience and has shown there are relatively slow variations in the measured parameters of level and temperature.

SR 3.1.7.4 and SR 3.1.7.6

SR 3.1.7.4 verifies the continuity of the explosive charges in the injection valves to ensure that proper operation will occur if required. Other administrative controls, such as those that limit the shelf life of the explosive charges, must be followed. The 31 day Frequency is based on operating experience and has demonstrated the reliability of the explosive charge continuity.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.9 (continued)

Surveillance when performed at the 24 month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.1.7.10

Enriched sodium pentaborate solution is made by mixing granular, enriched sodium pentaborate with water. In order to ensure the proper B-10 atom percentage (in accordance with Table 3.1.7-1) is being used, calculations must be performed to verify the actual B-10 enrichment within 8 hours after addition of the solution to the SLC tank. The calculations may be performed using the results of isotopic tests on the granular sodium pentaborate or vendor certification documents. The Frequency is acceptable considering that boron enrichment is verified during the procurement process and any time boron is added to the SLC tank.

REFERENCES

1. 10 CFR 50.62.
2. UFSAR, Section 3.8.4.
3. 10 CFR 50.67.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.a. Reactor Vessel Water Level—Low Low Low (Level 1) (continued)

The Reactor Vessel Water Level—Low Low Low (Level 1) Allowable Value is chosen to be the same as the ECCS Level 1 Allowable Value (LCO 3.3.5.1) to ensure that the MSLs isolate on a potential loss of coolant accident (LOCA) to prevent offsite doses from exceeding 10 CFR 50.67 limits.

This Function isolates MSIVs, MSL drains, MSL sample lines and recirculation loop sample line valves.

1.b. Main Steam Line Pressure—Low

Low MSL pressure indicates that there may be a problem with the turbine pressure regulation, which could result in a low reactor vessel water level condition and the RPV cooling down more than 100°F/hr if the pressure loss is allowed to continue. The Main Steam Line Pressure—Low Function is directly assumed in the analysis of the pressure regulator failure (Ref. 3). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hr) is not reached. In addition, this Function supports actions to ensure that Safety Limit 2.1.1.1 is not exceeded. (This Function closes the MSIVs prior to pressure decreasing below 785 psig, which results in a scram due to MSIV closure, thus reducing reactor power to < 25% RTP.)

The MSL low pressure signals are initiated from four transmitters that are connected to the MSL header. The transmitters are arranged such that, even though physically separated from each other, each transmitter is able to detect low MSL pressure. Four channels of Main Steam Line Pressure—Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be high enough to prevent excessive RPV depressurization.

The Main Steam Line Pressure—Low Function is only required to be OPERABLE in MODE 1 since this is when the assumed transient can occur (Ref. 1).

This Function isolates MSIVs, MSL drains, MSL sample lines and recirculation loop sample line valves.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

1.c. Main Steam Line Flow—High

Main Steam Line Flow—High is provided to detect a break of the MSL and to initiate closure of the MSIVs. If the steam were allowed to continue flowing out of the break, the reactor would depressurize and the core could uncover. If the RPV water level decreases too far, fuel damage could occur. Therefore, the isolation is initiated on high flow to prevent or minimize core damage. The Main Steam Line Flow—High Function is directly assumed in the analysis of the main steam line break (MSLB) (Ref. 3). The isolation action, along with the scram function of the Reactor Protection System (RPS), ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46 and offsite doses do not exceed the 10 CFR 50.67 limits.

The MSL flow signals are initiated from 16 transmitters that are connected to the four MSLs. The transmitters are arranged such that, even though physically separated from each other, all four connected to one MSL would be able to detect the high flow. Four channels of Main Steam Line Flow—High Function for each MSL (two channels per trip system) are available and are required to be OPERABLE so that no single instrument failure will preclude detecting a break in any individual MSL.

The Allowable Value is chosen to ensure that offsite dose limits are not exceeded due to the break.

This Function isolates MSIVs, MSL drains, MSL sample lines and recirculation loop sample line valves.

1.d. Main Steam Line—High Radiation

The Main Steam Line—High Radiation Function is provided to detect gross release of fission products from the fuel and to initiate closure of the MSIVs. The trip setting is set low enough so that a high radiation trip results from a design basis rod drop accident and high enough above background radiation levels in the vicinity of the main steam lines so that spurious trips at rated power are avoided. The Main Steam Line—High Radiation Function is directly assumed in the analysis of the control rod drop accident (Ref. 3).

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.d. Main Steam Line-High Radiation (continued)

The Main Steam Line-High Radiation signals are initiated from four gamma sensitive instruments. Four channels are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value is chosen to ensure that offsite dose limits are not exceeded.

This Function isolates MSIVs, MSL drains, MSL sample lines and recirculation loop sample line valves.

1.e. Main Steam Tunnel Temperature-High

The Main Steam Tunnel Temperature Function is provided to detect a break in a main steam line and provides diversity to the high flow instrumentation.

Main Steam Tunnel Temperature signals are initiated from resistance temperature detectors (RTDs) located along the main steam line between the drywell wall and the turbine. Sixteen channels of Main Steam Tunnel Temperature-High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value is chosen to detect a leak equivalent to between 1% and 10% rated steam flow.

This Function isolates MSIVs, MSL drains, MSL sample lines and recirculation loop sample line valves.

Primary Containment Isolation

2.a. Reactor Vessel Water Level-Low (Level 3)

Low RPV water level indicates that the capability to cool the fuel may be threatened. The valves whose penetrations communicate with the primary containment are isolated to limit the release of fission products. The isolation of the primary containment on Level 3 supports actions to ensure that offsite dose limits of 10 CFR 50.67 are not exceeded.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.a. Reactor Vessel Water Level—Low (Level 3) (continued)

The Reactor Vessel Water Level—Low (Level 3) Function associated with isolation is implicitly assumed in the UFSAR analysis as these leakage paths are assumed to be isolated post LOCA.

Reactor Vessel Water Level—Low (Level 3) signals are initiated from level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level—Low (Level 3) Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level—Low (Level 3) Allowable Value was chosen to be the same as the RPS Level 3 scram Allowable Value (LCO 3.3.1.1), since isolation of these valves is not critical to orderly plant shutdown.

This Function isolates the Group II(A) valves listed in Reference 1 with the exception of RWCU isolation valves and RHR shutdown cooling pump suction valves which are addressed in Functions 5.c and 6.b, respectively.

2.b. Drywell Pressure—High

High drywell pressure can indicate a break in the RCPB inside the primary containment. The isolation of some of the primary containment isolation valves on high drywell pressure supports actions to ensure that offsite dose limits of 10 CFR 50.67 are not exceeded. The Drywell Pressure—High Function, associated with isolation of the primary containment, is implicitly assumed in the UFSAR accident analysis as these leakage paths are assumed to be isolated post LOCA.

High drywell pressure signals are initiated from pressure transmitters that sense the pressure in the drywell. Four channels of Drywell Pressure—High are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

5.a. RWCU Flow-High (continued)

The high RWCU flow signals are initiated from transmitters that are connected to the pump suction line of the RWCU System. Two channels of RWCU Flow-High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The RWCU Flow-High Allowable Value ensures that a break of the RWCU piping is detected.

This Function isolates the inboard and outboard RWCU pump suction penetration and the outboard valve at the RWCU connection to reactor feedwater.

5.b. Standby Liquid Control (SLC) System Initiation

The isolation of the RWCU System is required when the SLC System has been initiated to prevent dilution and removal of the boron solution by the RWCU System (Ref. 5). SLC System initiation signals are initiated from the remote SLC System start switch.

There is no Allowable Value associated with this Function since the channels are mechanically actuated based solely on the position of the SLC System initiation switch.

Two channels of the SLC System Initiation Function are available and are required to be OPERABLE in MODES 1 and 2, since these are the only MODES where the reactor can be critical. Both channels are also required to be OPERABLE in MODES 1, 2, and 3, since the SLC System is also designed to maintain suppression pool pH above 7 following a LOCA to ensure that iodine will be retained in the suppression pool water. These MODES are consistent with the Applicability for the SLC System (LCO 3.1.7).

This Function isolates the inboard and outboard RWCU pump suction penetration and the outboard valve at the RWCU connection to reactor feedwater.

5.c. Reactor Vessel Water Level-Low (Level 3)

Low RPV water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, isolation of some interfaces with the reactor vessel occurs to isolate the potential sources of a break. The isolation of the RWCU System on Level 3 supports actions to ensure that the fuel

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3, 4. Reactor Building Ventilation and Refueling Floor Ventilation Exhaust Radiation—High (continued)

channels of Reactor Building Ventilation Exhaust Radiation—High Function and four channels of Refueling Floor Ventilation Exhaust Radiation—High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Values are chosen to promptly detect gross failure of the fuel cladding.

The Reactor Building Ventilation and Refueling Floor Ventilation Exhaust Radiation—High Functions are required to be OPERABLE in MODES 1, 2, and 3 where considerable energy exists; thus, there is a probability of pipe breaks resulting in significant releases of radioactive steam and gas. In MODES 4 and 5, the probability and consequences of these events are low due to the RCS pressure and temperature limitations of these MODES; thus, these Functions are not required. In addition, the Functions are also required to be OPERABLE during OPDRVs and movement of recently irradiated fuel assemblies in the secondary containment, because the capability of detecting radiation releases due to fuel failures (due to fuel uncovery or dropped fuel assemblies) must be provided to ensure that offsite dose limits are not exceeded. The Functions are only required to be OPERABLE during handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

ACTIONS

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The Control Room Air Intake Radiation—High Function consists of four independent monitors. Two channels of Control Room Air Intake Radiation—High per trip system are available and are required to be OPERABLE to ensure that no single instrument failure can preclude MCREV System initiation. The Allowable Value was selected to ensure protection of the control room personnel.

The Control Room Air Intake Radiation—High Function is required to be OPERABLE in MODES 1, 2, and 3 and during OPDRVs and movement of recently irradiated fuel assemblies in the secondary containment, to ensure that control room personnel are protected during a LOCA, fuel handling event, or vessel draindown event. The MCREV System is only required to be OPERABLE during handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours). During MODES 4 and 5, when these specified conditions are not in progress (e.g., OPDRVs), the probability of a LOCA or fuel damage is low; thus, the Function is not required.

ACTIONS

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

A.1 and A.2

Because of the redundancy of sensors available to provide initiation signals and the redundancy of the MCREV System design, an allowable out of service time of 6 hours has been shown to be acceptable (Ref. 4), to permit restoration of any inoperable channel to OPERABLE status. However, this out of service time is only acceptable provided the Control Room Air Intake Radiation—High Function is still maintaining MCREV System initiation capability. The Function is considered to be maintaining MCREV System

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 RCS Specific Activity

BASES

BACKGROUND

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

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

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

APPLICABLE SAFETY ANALYSES

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

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	The limits on specific activity are values from a parametric evaluation of typical site locations. These limits are conservative because the evaluation considered more restrictive parameters than for a specific site, such as the location of the site boundary and the meteorological conditions of the site. RCS specific activity satisfies Criterion 2 of the NRC Policy Statement.
LCO	The specific iodine activity is limited to ≤ 0.2 μCi/gm DOSE EQUIVALENT I-131. This limit ensures the source term assumed in the safety analysis for the MSLB is not exceeded, so any release of radioactivity to the environment during an MSLB is well within the 10 CFR 50.67 limits.
APPLICABILITY	In MODE 1, and MODES 2 and 3 with any main steam line not isolated, limits on the primary coolant radioactivity are applicable since there is an escape path for release of radioactive material from the primary coolant to the environment in the event of an MSLB outside of primary containment. In MODES 2 and 3 with the main steam lines isolated, such limits do not apply since an escape path does not exist. In MODES 4 and 5, no limits are required since the reactor is not pressurized and the potential for leakage is reduced.
ACTIONS	A.1 and A.2 When the reactor coolant specific activity exceeds the LCO DOSE EQUIVALENT I-131 limit, but is ≤ 4.0 μCi/gm, samples must be analyzed for DOSE EQUIVALENT I-131 at least once every 4 hours. In addition, the specific activity must be restored to the LCO limit within 48 hours. The Completion Time of once every 4 hours is based on the time needed to take and analyze a sample. The 48 hour Completion Time to restore the activity level provides a reasonable time for temporary coolant activity increases (iodine spikes) to be cleaned up with the normal processing systems.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

A Note to the Required Actions of Condition A excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE(S) while relying on the ACTIONS even though the ACTIONS may eventually require plant shutdown. This exception is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the plant remains at, or proceeds to, power operation.

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

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

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

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

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

This Surveillance is performed to ensure iodine remains within limit during normal operation. The 7 day Frequency is adequate to trend changes in the iodine activity level.

This SR is modified by a Note that requires this Surveillance to be performed only in MODE 1 because the level of fission products generated in other MODES is much less.

REFERENCES

1. 10 CFR 50.67.
2. UFSAR, Section 14.6.5.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.7 (continued)

position, since these valves were verified to be in the correct position prior to locking or securing. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves. The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.6.1.3.8

Verifying the isolation time of each power operated and each automatic PCIV is within limits is required to demonstrate OPERABILITY. MSIVs may be excluded from this SR since MSIV full closure isolation time is demonstrated by SR 3.6.1.3.9. The isolation time test ensures that the valve will isolate in a time period less than or equal to that assumed in the safety analyses. The isolation time is in accordance with Reference 2 or the requirements of the Inservice Testing Program which ever is more conservative. The Frequency of this SR is in accordance with the requirements of the Inservice Testing Program.

SR 3.6.1.3.9

Verifying that the isolation time of each MSIV is within the specified limits is required to demonstrate OPERABILITY. The isolation time test ensures that the MSIV will isolate in a time period that does not exceed the times assumed in the DBA analyses. This ensures that the calculated radiological consequences of these events remain within 10 CFR 50.67 limits. The Frequency of this SR is in accordance with the requirements of the Inservice Testing Program.

SR 3.6.1.3.10

Automatic PCIVs close on a primary containment isolation signal to prevent leakage of radioactive material from primary containment following a DBA. This SR ensures that each automatic PCIV will actuate to its isolation position on a primary containment isolation signal. The LOGIC SYSTEM

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.13

This SR ensures that in case the non-safety grade instrument air system is unavailable, the SGIG System will perform its design function to supply nitrogen gas at the required pressure for valve operators and valve seals supported by the SGIG System. The 24 month Frequency was developed considering it is prudent that this Surveillance be performed only during a plant outage. Operating experience has shown that these components will usually pass this Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.1.3.14

Total leakage through all four main steam lines must be ≤ 174 scfh, and ≤ 100 scfh for any one steam line, when tested at ≥ 25 psig. The analysis in Reference 1 is based on treatment of MSIV leakage as secondary containment bypass leakage, independent of the primary to secondary containment leakage analyzed at L_a. The Frequency is in accordance with the Primary Containment Leakage Rate Testing Program.

SR 3.6.1.3.15

Verifying the opening of each 6 inch and 18 inch primary containment purge valve and each 18 inch primary containment exhaust valve is restricted by a blocking device to less than or equal to the required maximum opening angle specified in the UFSAR (Ref. 4) is required to ensure that the valves can close under DBA conditions within the times in the analysis of Reference 1. If a LOCA occurs, the purge and exhaust valves must close to maintain primary containment leakage within the values assumed in the accident analysis. At other times pressurization concerns are not present, thus the purge and exhaust valves can be fully open. The 24 month Frequency is appropriate because the blocking devices may be removed during a refueling outage.

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.1 Secondary Containment

BASES

BACKGROUND

The function of the secondary containment is to receive fission products that may leak from primary containment or from systems in secondary containment following a Design Basis Accident (DBA) and, in conjunction with the Standby Gas Treatment System (SGT) and closure of certain valves whose lines penetrate the secondary containment, to provide for elevated release through the Main Stack.

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

APPLICABLE SAFETY ANALYSES

There is one principal accident for which credit is taken for secondary containment OPERABILITY. That is a loss of coolant accident (LOCA) (Ref. 1). The secondary containment performs no active function in response to this limiting event; however, its leak

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

tightness is required to ensure that fission products entrapped within the secondary containment structure will be collected by the SGT System for discharge to the environment via the main stack.

Secondary containment satisfies Criterion 3 of the NRC Policy Statement.

LCO

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

APPLICABILITY

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

In MODES 4 and 5, the probability and consequences of the LOCA are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining secondary containment OPERABLE is not required in MODE 4 or 5 to ensure a control volume, except for other situations for which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs), or during movement of recently irradiated fuel assemblies in the secondary containment. Secondary containment is only required to be OPERABLE during handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

ACTIONS

A.1

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

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

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

C.1 and C.2

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

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

Required Action C.1 has been modified by a Note stating that LCO 3.0.3 is not applicable, since the movement of recently irradiated fuel can only be performed in MODES 4 and 5.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1.1 and SR 3.6.4.1.2

Verifying that secondary containment equipment hatches and one access door in each access opening are closed ensures that the infiltration of outside air of such a magnitude as to prevent maintaining the desired negative pressure does not occur. Verifying that all such openings are closed provides adequate assurance that exfiltration from the secondary containment will not occur. In this application, the term "sealed" has no connotation of leak tightness. Maintaining secondary containment OPERABILITY requires verifying one door in the access opening is closed. An access opening contains one inner and one outer door. In some cases, secondary containment access openings are shared such that a secondary containment barrier may have multiple inner or multiple outer doors. The intent is to not breach secondary containment at any time when secondary containment is required. This is achieved by maintaining the inner or outer portion of the barrier closed at all times. However, all secondary containment access doors are normally kept closed, except when the access opening is being used for entry and exit or when maintenance is being performed on an access opening. The 31 day Frequency for these SRs has been shown to be adequate, based on operating experience, and is considered adequate in view of the other indications of door and hatch status that are available to the operator.

SR 3.6.4.1.3 and SR 3.6.4.1.4

The SGT System exhausts the secondary containment atmosphere to the environment through the elevated release point provided by the Main Stack.

To ensure that this exhaust pathway is used, SR 3.6.4.1.3 verifies that the SGT System will rapidly establish and maintain a pressure in the secondary containment that is less than the pressure external to the secondary containment boundary. This is confirmed by demonstrating that one SGT subsystem will draw down the secondary containment to $\geq$ 0.25 inches of vacuum water gauge in $\leq$ 15 minutes. This cannot be accomplished if the secondary containment boundary is not intact.

SR 3.6.4.1.4 demonstrates that one SGT subsystem can maintain $\geq$ 0.25 inches of vacuum water gauge for 1 hour at a flow rate $\leq$ 10,500 cfm. The 1 hour test period allows secondary containment to be in thermal equilibrium at steady

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.2 Secondary Containment Isolation Valves (SCIVs)

BASES

BACKGROUND

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

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

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

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

APPLICABLE SAFETY ANALYSES

The SCIVs must be OPERABLE to ensure the secondary containment barrier to fission product releases is established. The principal accident for which the secondary containment boundary is required is a loss of coolant accident (Ref. 2). The secondary containment performs no active function in response to either of these limiting events, but the boundary

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	established by SCIVs is required to ensure that leakage from the primary containment is exhausted by the Standby Gas Treatment (SGT) System for elevated release to the environment via the main stack. Maintaining SCIVs OPERABLE with isolation times within limits ensures that fission products will remain trapped inside secondary containment until discharged to the environment via the main stack. SCIVs satisfy Criterion 3 of the NRC Policy Statement.
LCO	SCIVs form a part of the secondary containment boundary. The SCIV safety function is related to control of offsite radiation releases resulting from DBAs. The power operated isolation valves are considered OPERABLE when their isolation times are within limits and the valves actuate on an automatic isolation signal. The valves covered by this LCO, along with their associated stroke times, are listed in Reference 2. The normally closed isolation valves or blind flanges are considered OPERABLE when manual valves are closed or open in accordance with appropriate administrative controls, automatic SCIVs are de-activated and secured in their closed position, and blind flanges are in place. These passive isolation valves or devices are listed in Reference 2.
APPLICABILITY	In MODES 1, 2, and 3, a DBA could lead to a fission product release to the primary containment that leaks to the secondary containment. Therefore, the OPERABILITY of SCIVs is required. In MODES 4 and 5, the probability and consequences of these events are reduced due to pressure and temperature limitations in these MODES. Therefore, maintaining SCIVs OPERABLE is not required in MODE 4 or 5, except for other situations under which significant radioactive releases can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs) or during movement of recently irradiated fuel assemblies in the secondary containment. SCIVs are only required to be OPERABLE during handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours). Moving irradiated fuel assemblies in the secondary containment may also occur in MODES 1, 2, and 3.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

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

D.1 and D.2

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

Required Action D.1 has been modified by a Note stating that LCO 3.0.3 is not applicable, since the movement of recently irradiated fuel can only be performed in MODES 4 and 5.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.2.1

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.6.4.2.3 (continued) under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components will usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.
REFERENCES	1. UFSAR, Section 14.9.2.1. 2. Technical Requirements Manual.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.3 Standby Gas Treatment (SGT) System

BASES

BACKGROUND

The primary function of the SGT System is to ensure that radioactive materials that leak from primary containment into the secondary containment following a Design Basis Accident (DBA) are discharged through the elevated release provided by the Main Stack.

A single SGT System is common to both Unit 2 and Unit 3 and consists of two fully redundant subsystems, each with its own set of ductwork, dampers, valves, charcoal filter train, and controls. Both SGT subsystems share a common inlet plenum. This inlet plenum is connected to the refueling floor ventilation exhaust duct for each Unit and to the suppression chamber and drywell of each Unit. Both SGT subsystems exhaust to the plant offgas stack through a common exhaust duct served by three 100% capacity system fans. SGT System fans OAV020 and OBV020 automatically start on Unit 2 secondary containment isolation signals. SGT System fans OCV020 and OBV020 automatically start on Unit 3 secondary containment isolation signals.

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

a. A demister or moisture separator;

b. An electric heater;

c. A prefilter;

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

e. A charcoal adsorber; and

f. A second HEPA filter.

These filters are not credited in any DBA analysis.

The SGT System is sized such that each 100% capacity fan will provide a flow rate of 10,500 cfm at 20 inches water gauge static pressure to support the control of fission product releases. The SGT System is designed to restore and maintain secondary containment at a negative pressure of 0.25 inches water gauge relative to the atmosphere following

(continued)

BASES

BACKGROUND (continued)

the receipt of a secondary containment isolation signal. Maintaining this negative pressure is based upon the existence of calm wind conditions (up to 5 mph), a maximum SGT System flow rate of 10,500 cfm, outside air temperature of 95°F and a temperature of 150°F for air entering the SGT System from inside secondary containment.

The demister is provided to remove entrained water in the air, while the electric heater reduces the relative humidity of the airstream to less than 70% (Ref. 2). Although not credited in any DBA analysis, the prefilter removes large particulate matter, while the HEPA filter removes fine particulate matter and protects the charcoal from fouling. The charcoal adsorber removes gaseous elemental iodine and organic iodides, and the final HEPA filter collects any carbon fines exhausted from the charcoal adsorber.

The SGT System automatically starts and operates in response to actuation signals indicative of conditions or an accident that could require operation of the system. Following initiation, two charcoal filter train fans (0AV020 and 0BV020) start. Upon verification that both subsystems are operating, the redundant subsystem is normally shut down.

APPLICABLE SAFETY ANALYSES

The design basis for the SGT System is to mitigate the consequences of a loss of coolant accident by providing a controlled, elevated release path. The SGT system also provides this function for OPDRVs. For all events where required, the SGT System automatically initiates to reduce, via an elevated release, the consequences of radioactive material released to the environment.

The HEPA filter and charcoal adsorber provided in the SGT System are not credited for any DBA analysis.

The SGT System satisfies Criterion 3 of the NRC Policy Statement.

LCO

Following a DBA, a minimum of one SGT subsystem is required to maintain the secondary containment at a negative pressure with respect to the environment and to process gaseous releases. Meeting the LCO requirements for two OPERABLE subsystems ensures operation of at least one SGT subsystem in the event of a single active failure.

(continued)

BASES

LCO (continued)

For Unit 2, one SGT subsystem is OPERABLE when one fan (0AV020) and associated ductwork, dampers, valves, and controls are OPERABLE. The second SGT subsystem is OPERABLE when the other fan (0BV020) and associated ductwork, damper, valves, and controls are OPERABLE.

APPLICABILITY

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

In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the SGT System in OPERABLE status is not required in MODE 4 or 5, except for other situations under which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs) or during movement of recently irradiated fuel assemblies in the secondary containment. The SGT System is only required to be OPERABLE during handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

ACTIONS

A.1

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

B.1 and B.2

If the SGT subsystem cannot be restored to OPERABLE status within the required Completion Time in MODE 1, 2, or 3, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

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

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

An alternative to Required Action C.1 is to immediately suspend activities that represent a potential for releasing a significant amount of radioactive material to the secondary containment, thus placing the plant in a condition that minimizes risk. If applicable, movement of recently irradiated fuel assemblies must immediately be suspended. Suspension of this activity must not preclude completion of movement of a component to a safe position. Also, if applicable, actions must immediately be initiated to suspend OPDRVs in order to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

The Required Actions of Condition C have been modified by a Note stating that LCO 3.0.3 is not applicable, since the movement of recently irradiated fuel can only be performed in MODES 4 and 5.

(continued)

BASES

ACTIONS (continued)

D.1

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

E.1 and E.2

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

Required Action E.1 has been modified by a Note stating that LCO 3.0.3 is not applicable, since the movement of recently irradiated fuel can only be performed in MODES 4 and 5.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.3.1

Operating each SGT subsystem (including each filter train fan) for ≥ 15 minutes ensures that both subsystems are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. The 31 day Frequency was developed in consideration of the known reliability of fan motors and controls and the redundancy available in the system.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.4.3.2

This SR verifies that the required SGT filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The only credited safety function of the SGT System is to provide a secondary containment vacuum sufficient to assure that discharges from the secondary containment will be through the Main Stack. The VFTP test 5.5.7.d. provides verification that the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is acceptable. SR 3.6.4.1.3 and SR 3.6.4.1.4 provide assurance that sufficient vacuum in the secondary containment is established with the time period as used in the DBA LOCA analysis. Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.6.4.3.3

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

REFERENCES

1. UFSAR, Section 1.5.1.6.
2. UFSAR, Section 14.9.

BASES

BACKGROUND (continued)

initiate an emergency shutdown of non-essential equipment and lighting to reduce the heat generation to a minimum. Heat removal would be accomplished by conduction through the floors, ceilings, and walls to adjacent rooms and to the environment. Additionally, the MCREV System is designed to maintain the control room environment for a 30-day occupancy after a DBA without exceeding 5 rem TEDE. A single MCREV subsystem will pressurize the control room to prevent infiltration of air from surrounding buildings. MCREV System operation in maintaining control room habitability is discussed in the UFSAR, Chapters 7, 10, and 12, (Refs. 1, 2, and 3, respectively).

APPLICABLE SAFETY ANALYSES

The ability of the MCREV System to maintain the habitability of the control room is an explicit assumption for the safety analyses presented in the UFSAR, Chapters 10 and 12 (Refs. 2 and 3, respectively). The MCREV System is credited as operating following a loss of coolant accident. The MCREV System is not credited in the analysis of the fuel handling accident, the main steam line break, or the control rod drop accident, as discussed in the UFSAR, Section 14.9.1.5 (Ref. 4). The radiological doses to control room personnel as a result of the various DBAs are summarized in Reference 4. No single active or passive failure will cause the loss of outside or recirculated air from the control room.

The MCREV System satisfies Criterion 3 of the NRC Policy Statement.

LCO

Two redundant subsystems of the MCREV System are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other subsystem. Total system failure could result in exceeding a dose of 5 rem TEDE to the control room operators in the event of a LOCA.

The MCREV System is considered OPERABLE when the individual components necessary to control operator exposure are OPERABLE in both subsystems. A subsystem is considered OPERABLE when its associated:

a. Fan is OPERABLE;

(continued)

BASES

LCO (continued)	b. HEPA filter 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 flow can be maintained. In addition, the control room boundary must be maintained, including the integrity of the walls, floors, ceilings, and ductwork. Temporary seals may be used to maintain the boundary. In addition, an access door may be opened provided the ability to pressurize the control room is maintained and the capability exists to close the affected door in an expeditious manner.
APPLICABILITY	In MODES 1, 2, and 3, the MCREV System must be OPERABLE to control operator exposure during and following a LOCA, since the LOCA could lead to a fission product release. In MODES 4 and 5, the probability and consequences of a LOCA are reduced because of the pressure and temperature limitations in these MODES. Therefore, maintaining the MCREV System OPERABLE is not required in MODE 4 or 5, except for the following situations under which significant radioactive releases can be postulated: a. During operations with potential for draining the reactor vessel (OPDRVs); and b. During movement of recently irradiated fuel assemblies in the secondary containment.
ACTIONS	<u>A.1</u> With one MCREV subsystem inoperable, the inoperable MCREV subsystem must be restored to OPERABLE status within 7 days. With the unit in this condition, the remaining OPERABLE MCREV subsystem is adequate to maintain control room temperature and to perform control room radiation protection. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could

(continued)

BASES

ACTIONS

A.1 (continued)

result in reduced MCREV System capability. The 7 day Completion Time is based on the low probability of a LOCA occurring during this time period, and that the remaining subsystem can provide the required capabilities.

B.1 and B.2

In MODE 1, 2, or 3, if the inoperable MCREV subsystem cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE that minimizes risk. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours and in MODE 4 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.

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

The Required Actions of Condition C are modified by a Note indicating that LCO 3.0.3 does not apply, since the movement of recently irradiated fuel can only be performed in MODES 4 and 5.

During movement of recently irradiated fuel assemblies in the secondary containment or during OPDRVs, if the inoperable MCREV subsystem cannot be restored to OPERABLE status within the required Completion Time, the OPERABLE MCREV subsystem may be placed in operation. This action ensures that the remaining subsystem is OPERABLE, that no failures that would prevent automatic actuation will occur, and that any active failure will be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

(continued)

BASES

ACTIONS

C.1, C.2.1, and C.2.2 (continued)

If applicable, movement of recently irradiated fuel assemblies in the secondary containment must be suspended immediately. Suspension of this activity shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and the subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended.

D.1

If both MCREV subsystems are inoperable in MODE 1, 2, or 3, the MCREV System may not be capable of performing the intended function and the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

E.1 and E.2

The Required Actions of Condition E are modified by a Note indicating that LCO 3.0.3 does not apply, since the movement of recently irradiated fuel can only be performed in MODES 4 and 5.

During movement of recently irradiated fuel assemblies in the secondary containment or during OPDRVs, with two MCREV subsystems inoperable, action must be taken immediately to suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

If applicable, movement of recently irradiated fuel assemblies in the secondary containment must be suspended immediately. Suspension of this activity shall not preclude completion of movement of a component to a safe position. If applicable, actions must be initiated

(continued)

BASES

ACTIONS

E.1 and E.2 (continued)

immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

This SR verifies that a subsystem in a standby mode starts on demand and continues to operate for ≥ 15 minutes. Standby systems should be checked periodically to ensure that they start and function properly. As the environmental and normal operating conditions of this system are not severe, testing each subsystem once every month provides an adequate check on this system. Furthermore, the 31 day Frequency is based on the known reliability of the equipment and the two subsystem redundancy available.

SR 3.7.4.2

This SR verifies that the required MCREV testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.4.3

This SR verifies that on an actual or simulated initiation signal, each MCREV subsystem starts and operates. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.7.1.4 overlaps this SR to provide complete testing of the safety function. Operating experience has shown that these components will usually pass the SR when performed at the 24 month Frequency. Therefore, this Frequency is concluded to be acceptable from a reliability standpoint.

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.5 Main Condenser Offgas

BASES

BACKGROUND

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

The Main Condenser Offgas System has been incorporated into the unit design to reduce the gaseous radwaste emission. This system uses a catalytic recombiner to recombine radiolytically dissociated hydrogen and oxygen. The gaseous mixture is cooled and water vapor removed by the offgas recombiner condenser; the remaining water and condensibles are stripped out by the cooler condenser and moisture separator. The remaining gaseous mixture (i.e., the offgas recombiner effluent) is then processed by a charcoal adsorber bed prior to release.

APPLICABLE SAFETY ANALYSES

The main condenser offgas gross gamma activity rate is an initial condition of the Main Condenser Offgas System failure event, discussed in the UFSAR, Section 9.4.5 (Ref. 1). The analysis assumes a gross failure in the Main Condenser Offgas System that results in the rupture of the Main Condenser Offgas System pressure boundary. The gross gamma activity rate is controlled to ensure that, during the event, the calculated offsite doses will be well within the limits of 10 CFR 50.67 (Ref. 2) or the NRC staff approved licensing basis.

The main condenser offgas limits satisfy Criterion 2 of the NRC Policy Statement.

LCO

To ensure compliance with the assumptions of the Main Condenser Offgas System failure event (Ref. 1), the fission product release rate should be consistent with a noble gas release to the reactor coolant of 100 μCi/MWt-second after decay of 30 minutes. The LCO is established consistent

(continued)

BASES

ACTIONS	B.1, B.2, B.3.1, and B.3.2 (continued) experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.
SURVEILLANCE REQUIREMENTS	SR 3.7.5.1 This SR, on a 31 day Frequency, requires an isotopic analysis of an offgas sample to ensure that the required limits are satisfied. The noble gases to be sampled are Xe-133, Xe-135, Xe-138, Kr-85m, Kr-87, and Kr-88. If the measured rate of radioactivity increases significantly (by ≥ 50% after correcting for expected increases due to changes in THERMAL POWER), an isotopic analysis is also performed within 4 hours after the increase is noted, to ensure that the increase is not indicative of a sustained increase in the radioactivity rate. The 31 day Frequency is adequate in view of other instrumentation that continuously monitor the offgas, and is acceptable, based on operating experience. This SR is modified by a Note indicating that the SR is not required to be performed until 31 days after any main steam line is not isolated and the SJAE is in operation. Only in this condition can radioactive fission gases be in the Main Condenser Offgas System at significant rates.
REFERENCES	1. UFSAR, Section 9.4.5. 2. 10 CFR 50.67.

B 3.7 PLANT SYSTEMS

B 3.7.7 Spent Fuel Storage Pool Water Level

BASES

BACKGROUND	The minimum water level in the spent fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident. A general description of the spent fuel storage pool design is found in the UFSAR, Section 10.3 (Ref. 1). The assumptions of the fuel handling accident are found in the UFSAR, Section 14.6.4 (Ref. 2).
APPLICABLE SAFETY ANALYSES	The water level above the irradiated fuel assemblies is an implicit assumption of the fuel handling accident. A fuel handling accident is evaluated to ensure that the radiological consequences are well below the guidelines set forth in 10 CFR 50.67 (Ref. 3). A fuel handling accident could release a fraction of the fission product inventory by breaching the fuel rod cladding as discussed in Reference 2. The fuel handling accident is evaluated for the dropping of an irradiated fuel assembly onto the reactor core. The consequences of a fuel handling accident over the spent fuel storage pool are less severe than those of the fuel handling accident over the reactor core. The water level in the spent fuel storage pool provides for absorption of water soluble fission product gases before being released to the secondary containment atmosphere. The spent fuel storage pool water level satisfies Criteria 2 and 3 of the NRC Policy Statement.
LCO	The specified water level (232 ft 3 inches plant elevation, which is equivalent to 22 ft over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks) preserves the assumptions of the fuel handling accident analysis (Ref. 2). As such, it is the minimum required for fuel movement within the spent fuel storage pool.

(continued)

BASES (continued)

APPLICABILITY	This LCO applies during movement of fuel assemblies in the spent fuel storage pool since the potential for a release of fission products exists.
ACTIONS	<u>A.1</u> Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply. If moving fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of fuel assemblies is not a sufficient reason to require a reactor shutdown. When the initial conditions for an accident cannot be met, action must be taken to preclude the accident from occurring. If the spent fuel storage pool level is less than required, the movement of fuel assemblies in the spent fuel storage pool is suspended immediately. Suspension of this activity shall not preclude completion of movement of a fuel assembly to a safe position. This effectively precludes a spent fuel handling accident from occurring.
SURVEILLANCE REQUIREMENTS	<u>SR 3.7.7.1</u> This SR verifies that sufficient water is available in the event of a fuel handling accident. The water level in the spent fuel storage pool must be checked periodically. The 7 day Frequency is acceptable, based on operating experience, considering that the water volume in the pool is normally stable, and all water level changes are controlled by unit procedures.
REFERENCES	1. UFSAR, Section 10.3. 2. UFSAR, Section 14.6.4. 3. 10 CFR 50.67.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.3 (continued)

Note 1 modifies this Surveillance to indicate that diesel engine runs for this Surveillance may include gradual loading, as recommended by the manufacturer, so that mechanical stress and wear on the diesel engine are minimized.

Note 2 modifies this Surveillance by stating that momentary transients because of changing bus loads do not invalidate this test. Similarly, momentary power factor transients above the limit do not invalidate the test.

Note 3 indicates that this Surveillance should be conducted on only one DG at a time in order to avoid common cause failures that might result from offsite circuit or grid perturbations.

Note 4 stipulates a prerequisite requirement for performance of this SR. A successful DG start must precede this test to credit satisfactory performance.

To minimize testing of the DGs, Note 5 allows a single test (instead of two tests, one for each unit) to satisfy the requirements for both units, with the DG synchronized to the 4 kV emergency bus of Unit 2 for one periodic test and synchronized to the 4 kV emergency bus of Unit 3 during the next periodic test. This is allowed since the main purpose of the Surveillance, to ensure DG OPERABILITY, is still being verified on the proper frequency, and each unit's breaker control circuitry, which is only being tested every second test (due to the staggering of the tests), historically have a very low failure rate. Note 5 modifies the specified frequency for each unit's breaker control circuitry to be 62 days. If the DG fails one of these Surveillances, the DG should be considered inoperable on both units, unless the cause of the failure can be directly related to only one unit. In addition, if the test is scheduled to be performed on Unit 3, and the Unit 3 TS allowance that provides an exception to performing the test is used (i.e., when Unit 3 is in MODE 4 or 5, or moving recently irradiated fuel assemblies in the secondary containment, the Note to Unit 3 SR 3.8.2.1 provides an exception to performing this test) or if it is not preferable to perform the test on a unit due to operational concerns (however time is not to exceed 62 days plus grace), then the test shall be performed synchronized to the Unit 2 4 kV emergency bus.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.20 (continued)

The 10 year Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 8). This SR is modified by two Notes. The reason for Note 1 is to minimize wear on the DG during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations. To minimize testing of the DGs, Note 2 allows a single test (instead of two tests, one for each unit) to satisfy the requirements for both units. This is allowed since the main purpose of the Surveillance can be met by performing the test on either unit. If a DG fails one of these Surveillances, a DG should be considered inoperable on both units, unless the cause of the failure can be directly related to only one unit.

SR 3.8.1.21

With the exception of this Surveillance, all other Surveillances of this Specification (SR 3.8.1.1 through SR 3.8.1.20) are applied only to the Unit 2 AC sources. This Surveillance is provided to direct that the appropriate Surveillances for the required Unit 3 AC sources are governed by the applicable Unit 3 Technical Specifications. Performance of the applicable Unit 3 Surveillances will satisfy Unit 3 requirements, as well as satisfying this Unit 2 Surveillance Requirement. Six exceptions are noted to the Unit 3 SRs of LCO 3.8.1. SR 3.8.1.8 is excepted when only one Unit 3 offsite circuit is required by the Unit 2 Specification, since there is not a second circuit to transfer to. SR 3.8.1.12, SR 3.8.1.13, SR 3.8.1.17, SR 3.8.1.18 (ECCS load block requirements only), and SR 3.8.1.19 are excepted since these SRs test the Unit 3 ECCS initiation signal, which is not needed for the AC sources to be OPERABLE on Unit 2.

The Frequency required by the applicable Unit 3 SR also governs performance of that SR for Unit 2.

As Noted, if Unit 3 is in MODE 4 or 5, or moving recently irradiated fuel assemblies in the secondary containment, the Note to Unit 3 SR 3.8.2.1 is applicable. This ensures that a Unit 2 SR will not require a Unit 3 SR to be performed, when the

(continued)

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 4 and 5 and during movement of recently irradiated fuel assemblies in secondary containment ensures that:

a. The facility 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 an inadvertent draindown of the vessel or a fuel handling accident involving recently irradiated fuel. 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 24 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 loss of 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, and 3 have no specific analyses in MODES 4 and 5. Worst case bounding events are deemed not credible in MODES 4 and 5 because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and corresponding stresses result in the probabilities of occurrences significantly reduced or eliminated, and 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, and 3, various deviations from the analysis assumptions and design requirements are allowed within the ACTIONS. This allowance is in recognition that

(continued)

BASES

LCO (continued)

offsite circuit. In addition, some equipment that may be required by Unit 2 is powered from Unit 3 sources (e.g., Standby Gas Treatment (SGT) System). Therefore, one qualified circuit between the offsite transmission network and the Unit 3 onsite Class 1E AC electrical power distribution subsystem(s), and one DG (not necessarily a different DG than those being used to meet LCO 3.8.2.b requirements) capable of supplying power to one of the required Unit 3 subsystems of each of the required components must also be OPERABLE. Together, OPERABILITY of the required offsite circuit(s) and required DG(s) ensures the availability of sufficient AC sources to operate the plant in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents involving recently irradiated fuel and reactor vessel draindown).

The qualified Unit 2 offsite circuit must be capable of maintaining rated frequency and voltage while connected to the respective Unit 2 4 kV emergency bus(es), and of accepting required loads during an accident. Qualified offsite circuits are those that are described in the UFSAR, Technical Specification Bases Section 3.8.1 and are part of the licensing basis for the unit. A Unit 2 offsite circuit consists of the incoming breaker and disconnect to the startup and emergency auxiliary transformer, the respective circuit path to the emergency auxiliary transformer, and the circuit path to the Unit 2 4 kV emergency buses required by LCO 3.8.8, including feeder breakers to the required Unit 2 4 kV emergency buses. A qualified Unit 3 offsite circuit's requirements are the same as the Unit 2 circuit's requirements, except that the circuit path, including the feeder breakers, is to the Unit 3 4 kV emergency buses required to be OPERABLE by LCO 3.8.8.

The required DGs must be capable of starting, accelerating to rated speed and voltage, and connecting to their respective Unit 2 emergency bus on detection of bus undervoltage. This sequence must be accomplished within 10 seconds. Each DG must also be capable of accepting required loads within the assumed loading sequence intervals, and must continue to operate until offsite power can be restored to the 4 kV emergency buses. These capabilities are required to be met from a variety of initial conditions such as DG in standby with engine hot and DG in standby with engine at ambient conditions. Additional

(continued)

BASES

LCO (continued)

DG capabilities must be demonstrated to meet required Surveillances, e.g., capability of the DG to revert to standby status on an ECCS signal while operating in parallel test mode. Proper sequencing of loads is a required function for DG OPERABILITY. The necessary portions of the Emergency Service Water System are also required to provide appropriate cooling to each required DG.

The OPERABILITY requirements for the DG capable of supplying power to the Unit 3 powered equipment are the same as described above, except that the required DG must be capable of connecting to its respective Unit 3 4 kV emergency bus. (In addition, the Unit 3 ECCS initiation logic SRs are not applicable, as described in SR 3.8.2.2 Bases.)

It is acceptable for 4 kV emergency buses to be cross tied during shutdown conditions, permitting a single offsite power circuit to supply all required buses. No automatic transfer capability is required for offsite circuits to be considered OPERABLE.

APPLICABILITY

The AC sources are required to be OPERABLE in MODES 4 and 5 and during movement of irradiated fuel assemblies in the secondary containment to provide assurance that:

a. Systems providing adequate coolant inventory makeup are available for the recently irradiated fuel assemblies in the core in case of an inadvertent draindown of the reactor vessel;

b. Systems needed to mitigate a fuel handling accident involving recently irradiated fuel are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

AC power requirements for MODES 1, 2, and 3 are covered in LCO 3.8.1.

(continued)

BASES (continued)

ACTIONS

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply since the movement of recently irradiated fuel can only be performed in MODES 4 and 5.

A.1 and B.1

With one or more required offsite circuits inoperable, or with one DG inoperable, the remaining required sources may be capable of supporting sufficient required features (e.g., system, subsystem, division, component, or device) to allow continuation of CORE ALTERATIONS, recently irradiated fuel movement, and operations with a potential for draining the reactor vessel. For example, if two or more 4 kV emergency buses are required per LCO 3.8.8, one 4 kV emergency bus with offsite power available may be capable of supplying sufficient required features. By the allowance of the option to declare required features inoperable that are not powered from offsite power (Required Action A.1) or capable of being powered by the required DG (Required Action B.1), appropriate restrictions can be implemented in accordance with the affected feature(s) LCOs' ACTIONS. Required features remaining powered from a qualified offsite power circuit, even if that circuit is considered inoperable because it is not powering other required features, are not declared inoperable by this Required Action. If a single DG is credited with meeting both LCO 3.8.2.d and one of the DG requirements of LCO 3.8.2.b, then the required features remaining capable of being powered by the DG are not declared inoperable by this Required Action, even if the DG is considered inoperable because it is not capable of powering other required features.

A.2.1, A.2.2, A.2.3, A.2.4, B.2.1, B.2.2, B.2.3, B.2.4, C.1, C.2, C.3, and C.4

With an offsite circuit not available to all required 4 kV emergency buses or one required DG inoperable, the option still exists to declare all required features inoperable

(continued)

BASES

ACTIONS

A.2.1, A.2.2, A.2.3, A.2.4, B.2.1, B.2.2, B.2.3, B.2.4, C.1, C.2, C.3, and C.4 (continued)

(per Required Actions A.1 and B.1). Since this option may involve undesired administrative efforts, the allowance for sufficiently conservative actions is made. With two or more required DGs inoperable, the minimum required diversity of AC power sources may not be available. It is, therefore, required to suspend CORE ALTERATIONS, movement of recently irradiated fuel assemblies in the secondary containment, and activities that could result in inadvertent draining of the reactor vessel.

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required AC sources and to continue this action until restoration is accomplished in order to provide the necessary AC power to the plant safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required AC electrical power sources should be completed as quickly as possible in order to minimize the time during which the plant safety systems may be without sufficient power.

Pursuant to LCO 3.0.6, the Distribution System ACTIONS would not be entered even if all AC sources to it are inoperable, resulting in de-energization. Therefore, the Required Actions of Condition A have been modified by a Note to indicate that when Condition A is entered with no AC power to any required 4 kV emergency bus, ACTIONS for LCO 3.8.8 must be immediately entered. This Note allows Condition A to provide requirements for the loss of the offsite circuit whether or not a required bus is de-energized. LCO 3.8.8 provides the appropriate restrictions for the situation involving a de-energized bus.

SURVEILLANCE REQUIREMENTS

SR 3.8.2.1

SR 3.8.2.1 requires the SRs from LCO 3.8.1 that are necessary for ensuring the OPERABILITY of the Unit 2 AC sources in other than MODES 1, 2, and 3. SR 3.8.1.8 is not

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.4.8 (continued)

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required DC electrical power subsystem from service, perturb the electrical distribution system, and challenge safety systems. Credit may be taken for unplanned events that satisfy the Surveillance. The DC batteries of the other unit are exempted from this restriction since they are required to be OPERABLE by both units and the Surveillance cannot be performed in the manner required by the Note without resulting in a dual unit shutdown.

SR 3.8.4.9

With the exception of this Surveillance, all other Surveillances of this Specification (SR 3.8.4.1 through SR 3.8.4.8) are applied only to the Unit 2 DC electrical power subsystems. This Surveillance is provided to direct that the appropriate Surveillances for the required Unit 3 DC electrical power subsystems are governed by the Unit 3 Technical Specifications. Performance of the applicable Unit 3 Surveillances will satisfy Unit 3 requirements, as well as satisfying this Unit 2 Surveillance Requirement.

The Frequency required by the applicable Unit 3 SR also governs performance of that SR for Unit 2. As Noted, if Unit 3 is in MODE 4 or 5, or moving recently irradiated fuel assemblies in the secondary containment, the Note to Unit 3 SR 3.8.5.1 is applicable. This ensures that a Unit 2 SR will not require a Unit 3 SR to be performed, when the Unit 3 Technical Specifications exempts performance of a Unit 3 SR. (However, as stated in the Unit 3 SR 3.8.5.1 Note, while performance of the SR is exempted, the SR still must be met.)

REFERENCES

1. UFSAR, Chapter 14.
2. "Proposed IEEE Criteria for Class 1E Electrical Systems for Nuclear Power Generating Stations," June 1969.
3. IEEE Standard 485, 1983.

(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 UFSAR, Chapter 14 (Ref. 1), assume that Engineered Safety Feature systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the diesel generators (DGs), 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 4 and 5 and during movement of recently irradiated fuel assemblies in secondary containment ensures that:

a. The facility 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 an inadvertent draindown of the vessel or a fuel handling accident involving recently irradiated fuel.

The DC sources satisfy Criterion 3 of the NRC Policy Statement.

LCO

The Unit 2 DC electrical power subsystems, with each DC subsystem consisting of two 125 V station batteries in series, two battery chargers (one per battery), and the corresponding control equipment and interconnecting cabling supplying power to the associated bus, are required to be

(continued)

BASES

LCO (continued)

OPERABLE to support Unit 2 DC distribution subsystems required OPERABLE by LCO 3.8.8, "Distribution Systems—Shutdown." When the equipment required OPERABLE: 1) does not require 250 VDC from the DC electrical power subsystem; and 2) does not require 125 VDC from one of the two 125 V batteries of the DC electrical power subsystem, the Unit 2 DC electrical power subsystem requirements can be modified to only include one 125 V battery (the battery needed to provide power to required equipment), an associated battery charger, and the corresponding control equipment and interconnecting cabling supplying 125 V power to the associated bus. This exception is allowed only if all 250 VDC loads are removed from the associated bus. In addition, DC control power (which provides control power for the 4 kV load circuit breakers and the feeder breakers to the 4 kV emergency bus) for two of the four 4 kV emergency buses, as well as control power for two of the diesel generators, is provided by the Unit 3 DC electrical power subsystems. Therefore, the Unit 3 DC electrical power subsystems needed to support required components are also required to be OPERABLE. The Unit 3 DC electrical power subsystem OPERABILITY requirements are the same as those required for a Unit 2 DC electrical power subsystem. In addition, battery chargers (Unit 2 and Unit 3) can be powered from the opposite unit's AC source (as described in the Background section of the Bases for LCO 3.8.4, "DC Sources—Operating"), and be considered OPERABLE for the purpose of meeting this LCO.

This requirement ensures the availability of sufficient DC electrical 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 recently irradiated fuel and inadvertent reactor vessel draindown).

APPLICABILITY

The DC electrical power sources required to be OPERABLE in MODES 4 and 5 and during movement of irradiated fuel assemblies in the secondary containment provide assurance that:

a. Required features to provide adequate coolant inventory makeup are available for the recently irradiated fuel assemblies in the core in case of an inadvertent draindown of the reactor vessel;

(continued)

BASES

APPLICABILITY (continued)

b. Required features needed to mitigate a fuel handling accident involving recently irradiated fuel are available;

c. Required features necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

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 24 hours).

The DC electrical power requirements for MODES 1, 2, and 3 are covered in LCO 3.8.4.

ACTIONS

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply since the movement of recently irradiated fuel can only be performed in MODES 4 and 5.

<u>A.1, A.2.1, A.2.2, A.2.3, and A.2.4</u>

If more than one DC distribution subsystem is required according to LCO 3.8.8, the DC electrical power subsystems remaining OPERABLE with one or more DC electrical power subsystems inoperable may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS, recently irradiated fuel movement, and operations with a potential for draining the reactor vessel.

By allowance of the option to declare required features inoperable with associated DC electrical power subsystems inoperable, appropriate restrictions are implemented in accordance with the affected system LCOs' ACTIONS. However, in many instances, this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of recently irradiated fuel assemblies in secondary containment, and any activities that could result in inadvertent draining of the reactor vessel).

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.5.2 (continued)

As Noted, if Unit 3 is in MODE 4 or 5, or moving recently irradiated fuel assemblies in the secondary containment, the Note to Unit 3 SR 3.8.5.1 is applicable. This ensures that a Unit 2 SR will not require a Unit 3 SR to be performed, when the Unit 3 Technical Specifications exempts performance of a Unit 3 SR. (However, as stated in the Unit 3 SR 3.8.5.1 Note, while performance of an SR is exempted, the SR still must be met.)

REFERENCES

1. UFSAR, Chapter 14.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.8 Distribution Systems—Shutdown

BASES

BACKGROUND	A description of the AC and DC electrical power distribution system is provided in the Bases for LCO 3.8.7, "Distribution Systems—Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident and transient analyses in the UFSAR, Chapter 14 (Ref. 1), assume Engineered Safety Feature (ESF) systems are OPERABLE. The AC and DC 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 AC and DC 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 AC and DC electrical power sources and associated power distribution subsystems during MODES 4 and 5 and during movement of recently irradiated fuel assemblies in the secondary containment ensures that:

a. The facility 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 an inadvertent draindown of the vessel or a fuel handling accident involving recently irradiated fuel.

AC and DC electrical power 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 24 hours).

The AC and DC electrical power distribution systems satisfy Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

Various combinations of subsystems, equipment, and components are required OPERABLE by other LCOs, depending on the specific plant condition. Implicit in those requirements is the required OPERABILITY of necessary support required features. This LCO explicitly requires energization of the portions of the Unit 2 electrical distribution system necessary to support OPERABILITY of Technical Specifications required systems, equipment, and components—both specifically addressed by their own LCO, and implicitly required by the definition of OPERABILITY. In addition, some components that may be required by Unit 2 receive power through Unit 3 electrical power distribution subsystems (e.g., Standby Gas Treatment (SGT) System and DC control power for two of the four 4 kV emergency buses, as well as control power for two of the diesel generators). Therefore, Unit 3 AC and DC electrical power distribution subsystems needed to support the required equipment must also be OPERABLE.

In addition, it is acceptable for required buses to be cross-tied during shutdown conditions, permitting a single source to supply multiple redundant buses, provided the source is capable of maintaining proper frequency (if required) and voltage.

Maintaining these portions of the distribution system energized ensures the availability of sufficient power to operate the plant in a safe manner to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents involving recently irradiated fuel and inadvertent reactor vessel draindown).

APPLICABILITY

The AC and DC electrical power distribution subsystems required to be OPERABLE in MODES 4 and 5 and during movement of recently irradiated fuel assemblies in the secondary containment provide assurance that:

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core in case of an inadvertent draindown of the reactor vessel;

b. Systems needed to mitigate a fuel handling accident involving recently irradiated fuel are available;

(continued)

BASES

APPLICABILITY (continued)

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The AC and DC electrical power distribution subsystem requirements for MODES 1, 2, and 3 are covered in LCO 3.8.7.

ACTIONS

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply since the movement of recently irradiated fuel can only be performed in MODES 4 and 5..

A.1, A.2.1, A.2.2, A.2.3, A.2.4, and A.2.5

Although redundant required features may require redundant electrical power distribution subsystems to be OPERABLE, one OPERABLE distribution subsystem may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS, recently irradiated fuel movement, and operations with a potential for draining the reactor vessel. By allowing the option to declare required features inoperable with associated electrical power distribution subsystems inoperable, appropriate restrictions are implemented in accordance with the affected distribution subsystem LCO's Required Actions. However, in many instances this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made, (i.e., to suspend CORE ALTERATIONS, movement of recently irradiated fuel assemblies in the secondary containment, and any activities that could result in inadvertent draining of the reactor vessel).

(continued)

BASES (continued)

SAFETY LIMITS	The reactor core SLs are established to protect the integrity of the fuel clad barrier to the release of radioactive materials to the environs. SL 2.1.1.1 and SL 2.1.1.2 ensure that the core operates within the fuel design criteria. SL 2.1.1.3 ensures that the reactor vessel water level is greater than the top of the active irradiated fuel in order to prevent elevated clad temperatures and resultant clad perforations.
APPLICABILITY	SLs 2.1.1.1, 2.1.1.2, and 2.1.1.3 are applicable in all MODES.

SAFETY LIMIT VIOLATIONS

2.2.1

If any SL is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. 3).

2.2.2

Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 50.67, "Accident Source Term," limits (Ref. 4). Therefore, it is required to insert all insertable control rods and restore compliance with the SLs within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also ensures that the probability of an accident occurring during this period is minimal.

2.2.3

If any SL is violated, the senior management of the nuclear plant and the utility shall be notified within 24 hours. The 24 hour period provides time for plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to the senior management.

(continued)

BASES

SAFETY LIMIT VIOLATIONS (continued)

2.2.4

If any SL is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Ref. 5). A copy of the report shall also be provided to the senior management of the nuclear plant and the utility.

2.2.5

If any SL is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.

REFERENCES

1. EMF-93-115 (P), July 1993.
2. NEDE-24011-P-A-10, February 1991.
3. 10 CFR 50.72.
4. 10 CFR 50.67.
5. 10 CFR 50.73.

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on reactor steam dome pressure protects 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. Establishing an upper limit on reactor steam dome pressure ensures continued RCS integrity with regard to pressure excursions. Per the UFSAR (Ref. 1), the reactor coolant pressure boundary (RCPB) shall be designed with sufficient margin to ensure that the design conditions are not exceeded during normal operation and abnormal operational transients.

During normal operation and abnormal operational transients, 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 are hydrostatically tested at 125% of design pressure, in accordance with ASME Code requirements, prior to initial operation when there is no fuel in the core. Any further hydrostatic testing with fuel in the core may be done under LCO 3.10.1, "Inservice Leak and Hydrostatic Testing Operation." 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 reducing the number of protective barriers designed to prevent radioactive releases from exceeding the limits specified in 10 CFR 50.67, "Accident Source Term," (Ref. 4). If this occurred in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere.

APPLICABLE SAFETY ANALYSES

The RCS safety/relief valves and the Reactor Protection System Reactor Pressure—High Function have settings established to ensure that the RCS pressure SL will not be exceeded.

(continued)

BASES

SAFETY LIMIT VIOLATIONS (continued)

2.2.2

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR 50.67, "Accident Source Term," limits (Ref. 4). Therefore, it is required to insert all insertable control rods and restore compliance with the SL within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also assures that the probability of an accident occurring during the period is minimal.

2.2.3

If any SL is violated, the senior management of the nuclear plant and the utility shall be notified within 24 hours. The 24 hour period provides time for plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to the senior management.

2.2.4

If any SL is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Ref. 8). A copy of the report shall also be provided to the senior management of the nuclear plant and the utility.

2.2.5

If any SL is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.

REFERENCES

1. UFSAR, Section 1.5.2.2.

2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000.

(continued)

BASES

REFERENCES (continued)

3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IW-5000.
4. 10 CFR 50.67.
5. ASME, Boiler and Pressure Vessel Code, Section III, 1965 Edition, including Addenda to summer of 1966.
6. ASME, Boiler and Pressure Vessel Code, Section III, 1980 Edition, Addenda to winter of 1981.
7. 10 CFR 50.72.
8. 10 CFR 50.73.

BASES

ACTIONS

B.1 and B.2 (continued)

control rods has less impact on control rod worth than withdrawals have. Required Action B.1 is modified by a Note which allows the RWM to be bypassed to allow the affected control rods to be returned to their correct position.

LCO 3.3.2.1 requires verification of control rod movement by a second licensed operator or a qualified member of the technical staff.

When nine or more OPERABLE control rods are not in compliance with BPWS, the reactor mode switch must be placed in the shutdown position within 1 hour. With the mode switch in shutdown, the reactor is shut down, and as such, does not meet the applicability requirements of this LCO. The allowed Completion Time of 1 hour is reasonable to allow insertion of control rods to restore compliance, and is appropriate relative to the low probability of a CRDA occurring with the control rods out of sequence.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

The control rod pattern is verified to be in compliance with the BPWS at a 24 hour Frequency to ensure the assumptions of the CRDA analyses are met. The 24 hour Frequency was developed considering that the primary check on compliance with the BPWS is performed by the RWM (LCO 3.3.2.1), which provides control rod blocks to enforce the required sequence and is required to be OPERABLE when operating at ≤ 10% RTP.

REFERENCES

1. NEDE-24011-P-A-10-US, "General Electric Standard Application for Reactor Fuel, Supplement for United States," Section 2.2.3.1, February 1991.

2. Letter (BWROG-8644) from T. Pickens (BWROG) to G. C. Lainas (NRC), "Amendment 17 to General Electric Licensing Topical Report NEDE-24011-P-A."

3. UFSAR, Section 14.6.2.3.

4. NUREG-0800, Section 15.4.9, Revision 2, July 1981.

5. 10 CFR 50.67.

(continued)

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Standby Liquid Control (SLC) System

BASES

BACKGROUND

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

The SLC System is also used to maintain suppression pool pH at or above 7 following a loss of coolant accident (LOCA) involving significant fission product releases. Maintaining suppression pool pH levels at or above 7 following an accident ensures that iodine will be retained in the suppression pool water.

Reference 1 requires a SLC System with a minimum flow capacity and boron content equivalent in control capacity to 86 gpm of 13 weight percent sodium pentaborate solution. Natural sodium pentaborate solution is 19.8% atom Boron-10. Therefore, the system parameters of concern, boron concentration (C), SLC pump flow rate (Q), and Boron-10 enrichment (E), may be expressed as a multiple of ratios. The expression is as follows:

$$\frac{C}{13\%\ \text{weight}} \times \frac{Q}{86\ \text{gpm}} \times \frac{E}{19.8\%\ \text{atom}}$$

If the product of this expression is ≥ 1, then the SLC System satisfies the criteria of Reference 1. As such, the equation forms the basis for acceptance criteria for the surveillances of concentration, flow rate, and boron enrichment and is presented in Table 3.1.7-1.

The SLC System consists of a boron solution storage tank, two positive displacement pumps, two explosive valves that are provided in parallel for redundancy, and associated piping and valves used to transfer borated water from the storage tank to the reactor pressure vessel (RPV). The borated solution is discharged near the bottom of the core shroud, where it then mixes with the cooling water rising through the core. A smaller tank containing demineralized water is provided for testing purposes.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES	The SLC System is manually initiated from the main control room, as directed by the emergency operating procedures, if the operator believes the reactor cannot be shut down, or kept shut down, with the control rods. The SLC System is used in the event that enough control rods cannot be inserted to accomplish shutdown and cooldown in the normal manner. The SLC System injects borated water into the reactor core to add negative reactivity to compensate for all of the various reactivity effects that could occur during plant operations. To meet this objective, it is necessary to inject a quantity of boron, which produces a concentration of 660 ppm of natural boron, in the reactor coolant at 68°F. To allow for potential leakage and imperfect mixing in the reactor system, an additional amount of boron equal to 25% of the amount cited above is added (Ref. 2). The minimum mass of Boron-10 (162.7 lbm) needed for injection is calculated such that the required quantity is achieved accounting for dilution in the RPV with normal water level and including the water volume in the residual heat removal shutdown cooling piping and in the recirculation loop piping. This quantity of borated solution is the amount that is above the pump suction shutoff level in the boron solution storage tank. No credit is taken for the portion of the tank volume that cannot be injected. The maximum concentration of sodium pentaborate listed in Table 3.1.7-1 has been established to ensure that the solution saturation temperature does not exceed 43°F. The sodium pentaborate solution in the SLC System is also used, post-LOCA, to maintain ECCS fluid pH above 7. The system parameters used in the calculation are the Boron-10 minimum mass of 162.7 lbm, and an upper bound Boron-10 enrichment of 65%. The SLC System satisfies Criterion 4 of the NRC Policy Statement.
LCO	The OPERABILITY of the SLC System provides backup capability for reactivity control independent of normal reactivity control provisions provided by the control rods. The OPERABILITY of the SLC System is based on the conditions of the borated solution in the storage tank and the availability of a flow path to the RPV, including the OPERABILITY of the pumps and valves. Two SLC subsystems are required to be OPERABLE; each contains an OPERABLE pump, an explosive valve, and associated piping, valves, and instruments and controls to ensure an OPERABLE flow path.

(continued)

BASES (continued)

APPLICABILITY

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

In MODES 1, 2, and 3, the SLC System must be OPERABLE to ensure that offsite doses remain within 10 CRF 50.67 (Ref. 3) limits following a LOCA involving significant fission product releases. The SLC System is designed to maintain suppression pool pH at or above 7 following a LOCA to ensure that iodine will be retained in the suppression pool water.

ACTIONS

A.1 and A.2

If the boron solution concentration is > 9.82% weight but the concentration and temperature of boron in solution and pump suction piping temperature are within the limits of Figure 3.1.7-1, operation is permitted for a limited period since the SLC subsystems are capable of performing the intended function. It is not necessary under these conditions to declare both SLC subsystems inoperable since the SLC subsystems are capable of performing their intended function.

The concentration and temperature of boron in solution and pump suction piping temperature must be verified to be within the limits of Figure 3.1.7-1 within 8 hours and once per 12 hours thereafter (Required Action A.1). The temperature versus concentration curve of Figure 3.1.7-1 ensures a 10°F margin will be maintained above the saturation temperature. This verification ensures that boron does not precipitate out of solution in the storage tank or in the pump suction piping due to low boron solution temperature (below the saturation temperature for the given concentration). The Completion Time for performing Required Action A.1 is considered acceptable given the low probability of a Design Basis Accident (DBA) or transient occurring concurrent with the failure of the control rods to shut down the reactor and operating experience which has shown there are relatively slow variations in the measured parameters of concentration and temperature over these time periods.

(continued)

BASES

ACTIONS

D.1 (continued)

brought to MODE 3 within 12 hours and MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1, SR 3.1.7.2, and SR 3.1.7.3

SR 3.1.7.1 through SR 3.1.7.3 are 24 hour Surveillances verifying certain characteristics of the SLC System (e.g., the level and temperature of the borated solution in the storage tank), thereby ensuring SLC System OPERABILITY without disturbing normal plant operation. These Surveillances ensure that the proper borated solution level and temperature, including the temperature of the pump suction piping, are maintained. Maintaining a minimum specified borated solution temperature is important in ensuring that the boron remains in solution and does not precipitate out in the storage tank or in the pump suction piping. The temperature limit specified in SR 3.1.7.2 and SR 3.1.7.3 and the maximum sodium pentaborate concentration specified in Table 3.1.7-1 ensures that a 10°F margin will be maintained above the saturation temperature. Control room alarms for low SLC storage tank temperature and low SLC System piping temperature are available and are set at 55°F. As such, SR 3.1.7.2 and SR 3.1.7.3 may be satisfied by verifying the absence of low temperature alarms for the SLC storage tank and SLC System piping. The 24 hour Frequency is based on operating experience and has shown there are relatively slow variations in the measured parameters of level and temperature.

SR 3.1.7.4 and SR 3.1.7.6

SR 3.1.7.4 verifies the continuity of the explosive charges in the injection valves to ensure that proper operation will occur if required. Other administrative controls, such as those that limit the shelf life of the explosive charges, must be followed. The 31 day Frequency is based on operating experience and has demonstrated the reliability of the explosive charge continuity.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.9 (continued)

Surveillance when performed at the 24 month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.1.7.10

Enriched sodium pentaborate solution is made by mixing granular, enriched sodium pentaborate with water. In order to ensure the proper B-10 atom percentage (in accordance with Table 3.1.7-1) is being used, calculations must be performed to verify the actual B-10 enrichment within 8 hours after addition of the solution to the SLC tank. The calculations may be performed using the results of isotopic tests on the granular sodium pentaborate or vendor certification documents. The Frequency is acceptable considering that boron enrichment is verified during the procurement process and any time boron is added to the SLC tank.

REFERENCES

1. 10 CFR 50.62.
2. UFSAR, Section 3.8.4.
3. 10 CFR 50.67.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.a. Reactor Vessel Water Level—Low Low Low (Level 1) (continued)

The Reactor Vessel Water Level—Low Low Low (Level 1) Allowable Value is chosen to be the same as the ECCS Level 1 Allowable Value (LCO 3.3.5.1) to ensure that the MSLs isolate on a potential loss of coolant accident (LOCA) to prevent offsite doses from exceeding 10 CFR 50.67 limits.

This Function isolates MSIVs, MSL drains, MSL sample lines and recirculation loop sample line valves.

1.b. Main Steam Line Pressure—Low

Low MSL pressure indicates that there may be a problem with the turbine pressure regulation, which could result in a low reactor vessel water level condition and the RPV cooling down more than 100°F/hr if the pressure loss is allowed to continue. The Main Steam Line Pressure—Low Function is directly assumed in the analysis of the pressure regulator failure (Ref. 3). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hr) is not reached. In addition, this Function supports actions to ensure that Safety Limit 2.1.1.1 is not exceeded. (This Function closes the MSIVs prior to pressure decreasing below 785 psig, which results in a scram due to MSIV closure, thus reducing reactor power to < 25% RTP.)

The MSL low pressure signals are initiated from four transmitters that are connected to the MSL header. The transmitters are arranged such that, even though physically separated from each other, each transmitter is able to detect low MSL pressure. Four channels of Main Steam Line Pressure—Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be high enough to prevent excessive RPV depressurization.

The Main Steam Line Pressure—Low Function is only required to be OPERABLE in MODE 1 since this is when the assumed transient can occur (Ref. 1).

This Function isolates MSIVs, MSL drains, MSL sample lines and recirculation loop sample line valves.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

1.c. Main Steam Line Flow—High

Main Steam Line Flow—High is provided to detect a break of the MSL and to initiate closure of the MSIVs. If the steam were allowed to continue flowing out of the break, the reactor would depressurize and the core could uncover. If the RPV water level decreases too far, fuel damage could occur. Therefore, the isolation is initiated on high flow to prevent or minimize core damage. The Main Steam Line Flow—High Function is directly assumed in the analysis of the main steam line break (MSLB) (Ref. 3). The isolation action, along with the scram function of the Reactor Protection System (RPS), ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46 and offsite doses do not exceed the 10 CFR 50.67 limits.

The MSL flow signals are initiated from 16 transmitters that are connected to the four MSLs. The transmitters are arranged such that, even though physically separated from each other, all four connected to one MSL would be able to detect the high flow. Four channels of Main Steam Line Flow—High Function for each MSL (two channels per trip system) are available and are required to be OPERABLE so that no single instrument failure will preclude detecting a break in any individual MSL.

The Allowable Value is chosen to ensure that offsite dose limits are not exceeded due to the break.

This Function isolates MSIVs, MSL drains, MSL sample lines and recirculation loop sample line valves.

1.d. Main Steam Line—High Radiation

The Main Steam Line—High Radiation Function is provided to detect gross release of fission products from the fuel and to initiate closure of the MSIVs. The trip setting is set low enough so that a high radiation trip results from a design basis rod drop accident and high enough above background radiation levels in the vicinity of the main steam lines so that spurious trips at rated power are avoided. The Main Steam Line—High Radiation Function is directly assumed in the analysis of the control rod drop accident (Ref. 3).

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.d. Main Steam Line-High Radiation (continued)

The Main Steam Line-High Radiation signals are initiated from four gamma sensitive instruments. Four channels are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value is chosen to ensure that offsite dose limits are not exceeded.

This Function isolates MSIVs, MSL drains, MSL sample lines and recirculation loop sample line valves.

1.e. Main Steam Tunnel Temperature-High

The Main Steam Tunnel Temperature Function is provided to detect a break in a main steam line and provides diversity to the high flow instrumentation.

Main Steam Tunnel Temperature signals are initiated from resistance temperature detectors (RTDs) located along the main steam line between the drywell wall and the turbine. Sixteen channels of Main Steam Tunnel Temperature-High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value is chosen to detect a leak equivalent to between 1% and 10% rated steam flow.

This Function isolates MSIVs, MSL drains, MSL sample lines and recirculation loop sample line valves.

Primary Containment Isolation

2.a. Reactor Vessel Water Level-Low (Level 3)

Low RPV water level indicates that the capability to cool the fuel may be threatened. The valves whose penetrations communicate with the primary containment are isolated to limit the release of fission products. The isolation of the primary containment on Level 3 supports actions to ensure that offsite dose limits of 10 CFR 50.67 are not exceeded.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.a. Reactor Vessel Water Level—Low (Level 3) (continued)

The Reactor Vessel Water Level—Low (Level 3) Function associated with isolation is implicitly assumed in the UFSAR analysis as these leakage paths are assumed to be isolated post LOCA.

Reactor Vessel Water Level—Low (Level 3) signals are initiated from level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level—Low (Level 3) Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level—Low (Level 3) Allowable Value was chosen to be the same as the RPS Level 3 scram Allowable Value (LCO 3.3.1.1), since isolation of these valves is not critical to orderly plant shutdown.

This Function isolates the Group II(A) valves listed in Reference 1 with the exception of RWCU isolation valves and RHR shutdown cooling pump suction valves which are addressed in Functions 5.c and 6.b, respectively.

2.b. Drywell Pressure—High

High drywell pressure can indicate a break in the RCPB inside the primary containment. The isolation of some of the primary containment isolation valves on high drywell pressure supports actions to ensure that offsite dose limits of 10 CFR 50.67 are not exceeded. The Drywell Pressure—High Function, associated with isolation of the primary containment, is implicitly assumed in the UFSAR accident analysis as these leakage paths are assumed to be isolated post LOCA.

High drywell pressure signals are initiated from pressure transmitters that sense the pressure in the drywell. Four channels of Drywell Pressure—High are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

5.a. RWCU Flow-High (continued)

The high RWCU flow signals are initiated from transmitters that are connected to the pump suction line of the RWCU System. Two channels of RWCU Flow-High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The RWCU Flow-High Allowable Value ensures that a break of the RWCU piping is detected.

This Function isolates the inboard and outboard RWCU pump suction penetration and the outboard valve at the RWCU connection to reactor feedwater.

5.b. Standby Liquid Control (SLC) System Initiation

The isolation of the RWCU System is required when the SLC System has been initiated to prevent dilution and removal of the boron solution by the RWCU System (Ref. 5). SLC System initiation signals are initiated from the remote SLC System start switch.

There is no Allowable Value associated with this Function since the channels are mechanically actuated based solely on the position of the SLC System initiation switch.

Two channels of the SLC System Initiation Function are available and are required to be OPERABLE in MODES 1 and 2, since these are the only MODES where the reactor can be critical. Both channels are also required to be OPERABLE in MODES 1, 2, and 3, since the SLC System is also designed to maintain suppression pool pH above 7 following a LOCA to ensure that iodine will be retained in the suppression pool water. These MODES are consistent with the Applicability for the SLC System (LCO 3.1.7).

This Function isolates the inboard and outboard RWCU pump suction penetration and the outboard valve at the RWCU connection to reactor feedwater.

5.c. Reactor Vessel Water Level-Low (Level 3)

Low RPV water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, isolation of some interfaces with the reactor vessel occurs to isolate the potential sources of a break. The isolation of the RWCU System on Level 3 supports actions to ensure that the fuel

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

<u>3, 4. Reactor Building Ventilation and Refueling Floor Ventilation Exhaust Radiation—High</u> (continued)

channels of Reactor Building Ventilation Exhaust Radiation—High Function and four channels of Refueling Floor Ventilation Exhaust Radiation—High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Values are chosen to promptly detect gross failure of the fuel cladding.

The Reactor Building Ventilation and Refueling Floor Ventilation Exhaust Radiation—High Functions are required to be OPERABLE in MODES 1, 2, and 3 where considerable energy exists; thus, there is a probability of pipe breaks resulting in significant releases of radioactive steam and gas. In MODES 4 and 5, the probability and consequences of these events are low due to the RCS pressure and temperature limitations of these MODES; thus, these Functions are not required. In addition, the Functions are also required to be OPERABLE during OPDRVs and movement of recently irradiated fuel assemblies in the secondary containment, because the capability of detecting radiation releases due to fuel failures (due to fuel uncovery or dropped fuel assemblies) must be provided to ensure that offsite dose limits are not exceeded. The Functions are only required to be OPERABLE during handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

ACTIONS

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)	The Control Room Air Intake Radiation—High Function consists of four independent monitors. Two channels of Control Room Air Intake Radiation—High per trip system are available and are required to be OPERABLE to ensure that no single instrument failure can preclude MCREV System initiation. The Allowable Value was selected to ensure protection of the control room personnel. The Control Room Air Intake Radiation—High Function is required to be OPERABLE in MODES 1, 2, and 3 and during OPDRVs and movement of recently irradiated fuel assemblies in the secondary containment, to ensure that control room personnel are protected during a LOCA, fuel handling event, or vessel draindown event. The MCREV System is only required to be OPERABLE during handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours). During MODES 4 and 5, when these specified conditions are not in progress (e.g., OPDRVs), the probability of a LOCA or fuel damage is low; thus, the Function is not required.
ACTIONS	A Note has been provided to modify the ACTIONS related to MCREV System instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable MCREV System instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable MCREV System instrumentation channel. <u>A.1 and A.2</u> Because of the redundancy of sensors available to provide initiation signals and the redundancy of the MCREV System design, an allowable out of service time of 6 hours has been shown to be acceptable (Ref. 4), to permit restoration of any inoperable channel to OPERABLE status. However, this out of service time is only acceptable provided the Control Room Air Intake Radiation—High Function is still maintaining MCREV System initiation capability. The Function is considered to be maintaining MCREV System

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 RCS Specific Activity

BASES

BACKGROUND

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

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

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

APPLICABLE SAFETY ANALYSES

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

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	The limits on specific activity are values from a parametric evaluation of typical site locations. These limits are conservative because the evaluation considered more restrictive parameters than for a specific site, such as the location of the site boundary and the meteorological conditions of the site. RCS specific activity satisfies Criterion 2 of the NRC Policy Statement.
LCO	The specific iodine activity is limited to ≤ 0.2 μCi/gm DOSE EQUIVALENT I-131. This limit ensures the source term assumed in the safety analysis for the MSLB is not exceeded, so any release of radioactivity to the environment during an MSLB is well within the 10 CFR 50.67 limits.
APPLICABILITY	In MODE 1, and MODES 2 and 3 with any main steam line not isolated, limits on the primary coolant radioactivity are applicable since there is an escape path for release of radioactive material from the primary coolant to the environment in the event of an MSLB outside of primary containment. In MODES 2 and 3 with the main steam lines isolated, such limits do not apply since an escape path does not exist. In MODES 4 and 5, no limits are required since the reactor is not pressurized and the potential for leakage is reduced.
ACTIONS	A.1 and A.2 When the reactor coolant specific activity exceeds the LCO DOSE EQUIVALENT I-131 limit, but is ≤ 4.0 μCi/gm, samples must be analyzed for DOSE EQUIVALENT I-131 at least once every 4 hours. In addition, the specific activity must be restored to the LCO limit within 48 hours. The Completion Time of once every 4 hours is based on the time needed to take and analyze a sample. The 48 hour Completion Time to restore the activity level provides a reasonable time for temporary coolant activity increases (iodine spikes) to be cleaned up with the normal processing systems.

(continued)

ACTIONS

A.1 and A.2 (continued)

A Note to the Required Actions of Condition A excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE(S) while relying on the ACTIONS even though the ACTIONS may eventually require plant shutdown. This exception is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the plant remains at, or proceeds to, power operation.

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

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

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

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

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS	SR 3.4.6.1 This Surveillance is performed to ensure iodine remains within limit during normal operation. The 7 day Frequency is adequate to trend changes in the iodine activity level. This SR is modified by a Note that requires this Surveillance to be performed only in MODE 1 because the level of fission products generated in other MODES is much less.
REFERENCES	1. 10 CFR 50.67. 2. UFSAR, Section 14.6.5.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.7 (continued)

position, since these valves were verified to be in the correct position prior to locking or securing. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves. The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.6.1.3.8

Verifying the isolation time of each power operated and each automatic PCIV is within limits is required to demonstrate OPERABILITY. MSIVs may be excluded from this SR since MSIV full closure isolation time is demonstrated by SR 3.6.1.3.9. The isolation time test ensures that the valve will isolate in a time period less than or equal to that assumed in the safety analyses. The isolation time is in accordance with Reference 2 or the requirements of the Inservice Testing Program which ever is more conservative. The Frequency of this SR is in accordance with the requirements of the Inservice Testing Program.

SR 3.6.1.3.9

Verifying that the isolation time of each MSIV is within the specified limits is required to demonstrate OPERABILITY. The isolation time test ensures that the MSIV will isolate in a time period that does not exceed the times assumed in the DBA analyses. This ensures that the calculated radiological consequences of these events remain within 10 CFR 50.67 limits. The Frequency of this SR is in accordance with the requirements of the Inservice Testing Program.

SR 3.6.1.3.10

Automatic PCIVs close on a primary containment isolation signal to prevent leakage of radioactive material from primary containment following a DBA. This SR ensures that each automatic PCIV will actuate to its isolation position on a primary containment isolation signal. The LOGIC SYSTEM

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.13

This SR ensures that in case the non-safety grade instrument air system is unavailable, the SGIG System will perform its design function to supply nitrogen gas at the required pressure for valve operators and valve seals supported by the SGIG System. The 24 month Frequency was developed considering it is prudent that this Surveillance be performed only during a plant outage. Operating experience has shown that these components will usually pass this Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.1.3.14

Total leakage through all four main steam lines must be ≤ 174 scfh, and ≤ 100 scfh for any one steam line, when tested at ≥ 25 psig. The analysis in Reference 1 is based on treatment of MSIV leakage as secondary containment bypass leakage, independent of the primary to secondary containment leakage analyzed at L_a. The Frequency is in accordance with the Primary Containment Leakage Rate Testing Program.

SR 3.6.1.3.15

Verifying the opening of each 6 inch and 18 inch primary containment purge valve and each 18 inch primary containment exhaust valve is restricted by a blocking device to less than or equal to the required maximum opening angle specified in the UFSAR (Ref. 4) is required to ensure that the valves can close under DBA conditions within the times in the analysis of Reference 1. If a LOCA occurs, the purge and exhaust valves must close to maintain primary containment leakage within the values assumed in the accident analysis. At other times pressurization concerns are not present, thus the purge and exhaust valves can be fully open. The 24 month Frequency is appropriate because the blocking devices may be removed during a refueling outage.

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.1 Secondary Containment

BASES

BACKGROUND

The function of the secondary containment is to receive fission products that may leak from primary containment or from systems in secondary containment following a Design Basis Accident (DBA) and, in conjunction with the Standby Gas Treatment System (SGT) and closure of certain valves whose lines penetrate the secondary containment, to provide for elevated release through the Main Stack.

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

APPLICABLE SAFETY ANALYSES

There is one principal accident for which credit is taken for secondary containment OPERABILITY. That is a loss of coolant accident (LOCA) (Ref. 1). The secondary containment performs no active function in response to this limiting event; however, its leak

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

tightness is required to ensure that fission products entrapped within the secondary containment structure will be collected by the SGT System for discharge to the environment via the main stack.

Secondary containment satisfies Criterion 3 of the NRC Policy Statement.

LCO

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

APPLICABILITY

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

In MODES 4 and 5, the probability and consequences of the LOCA are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining secondary containment OPERABLE is not required in MODE 4 or 5 to ensure a control volume, except for other situations for which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs) or during movement of recently irradiated fuel assemblies in the secondary containment. Secondary containment is only required to be OPERABLE during handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

ACTIONS

A.1

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

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

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

C.1 and C.2

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

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

Required Action C.1 has been modified by a Note stating that LCO 3.0.3 is not applicable, since the movement of recently irradiated fuel can only be performed in MODES 4 and 5.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1.1 and SR 3.6.4.1.2

Verifying that secondary containment equipment hatches and one access door in each access opening are closed ensures that the infiltration of outside air of such a magnitude as to prevent maintaining the desired negative pressure does not occur. Verifying that all such openings are closed provides adequate assurance that exfiltration from the secondary containment will not occur. In this application, the term "sealed" has no connotation of leak tightness. Maintaining secondary containment OPERABILITY requires verifying one door in the access opening is closed. An access opening contains one inner and one outer door. In some cases, secondary containment access openings are shared such that a secondary containment barrier may have multiple inner or multiple outer doors. The intent is to not breach secondary containment at any time when secondary containment is required. This is achieved by maintaining the inner or outer portion of the barrier closed at all times. However, all secondary containment access doors are normally kept closed, except when the access opening is being used for entry and exit or when maintenance is being performed on an access opening. The 31 day Frequency for these SRs has been shown to be adequate, based on operating experience, and is considered adequate in view of the other indications of door and hatch status that are available to the operator.

SR 3.6.4.1.3 and SR 3.6.4.1.4

The SGT System exhausts the secondary containment atmosphere to the environment through the elevated release point provided by the Main Stack.

To ensure that this exhaust pathway is used, SR 3.6.4.1.3 verifies that the SGT System will rapidly establish and maintain a pressure in the secondary containment that is less than the pressure external to the secondary containment boundary. This is confirmed by demonstrating that one SGT subsystem will draw down the secondary containment to $\geq$ 0.25 inches of vacuum water gauge in $\leq$ 15 minutes. This cannot be accomplished if the secondary containment boundary is not intact.

SR 3.6.4.1.4 demonstrates that one SGT subsystem can maintain $\geq$ 0.25 inches of vacuum water gauge for 1 hour at a flow rate $\leq$ 10,500 cfm. The 1 hour test period allows secondary containment to be in thermal equilibrium at steady

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.2 Secondary Containment Isolation Valves (SCIVs)

BASES

BACKGROUND

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

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

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

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

APPLICABLE SAFETY ANALYSES

The SCIVs must be OPERABLE to ensure the secondary containment barrier to fission product releases is established. The principal accident for which the secondary containment boundary is required is a loss of coolant accident (Ref. 2). The secondary containment performs no active function in response to either of these limiting events, but the boundary

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	established by SCIVs is required to ensure that leakage from the primary containment is exhausted by the Standby Gas Treatment (SGT) System for elevated release to the environment via the main stack. Maintaining SCIVs OPERABLE with isolation times within limits ensures that fission products will remain trapped inside secondary containment until discharged to the environment via the main stack. SCIVs satisfy Criterion 3 of the NRC Policy Statement.
LCO	SCIVs form a part of the secondary containment boundary. The SCIV safety function is related to control of offsite radiation releases resulting from DBAs. The power operated isolation valves are considered OPERABLE when their isolation times are within limits and the valves actuate on an automatic isolation signal. The valves covered by this LCO, along with their associated stroke times, are listed in Reference 2. The normally closed isolation valves or blind flanges are considered OPERABLE when manual valves are closed or open in accordance with appropriate administrative controls, automatic SCIVs are de-activated and secured in their closed position, and blind flanges are in place. These passive isolation valves or devices are listed in Reference 2.
APPLICABILITY	In MODES 1, 2, and 3, a DBA could lead to a fission product release to the primary containment that leaks to the secondary containment. Therefore, the OPERABILITY of SCIVs is required. In MODES 4 and 5, the probability and consequences of these events are reduced due to pressure and temperature limitations in these MODES. Therefore, maintaining SCIVs OPERABLE is not required in MODE 4 or 5, except for other situations under which significant radioactive releases can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs) or during movement of recently irradiated fuel assemblies in the secondary containment. SCIVs are only required to be OPERABLE during handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours). Moving irradiated fuel assemblies in the secondary containment may also occur in MODES 1, 2, and 3.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

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

D.1 and D.2

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

Required Action D.1 has been modified by a Note stating that LCO 3.0.3 is not applicable, since the movement of recently irradiated fuel can only be performed in MODES 4 and 5.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.2.1

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.4.2.3 (continued)

under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components will usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 14.9.2.1.
2. Technical Requirements Manual.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.3 Standby Gas Treatment (SGT) System

BASES

BACKGROUND

The primary function of the SGT System is to ensure that radioactive materials that leak from primary containment into the secondary containment following a Design Basis Accident (DBA) are discharged through the elevated release provided by the Main Stack.

A single SGT System is common to both Unit 2 and Unit 3 and consists of two fully redundant subsystems, each with its own set of ductwork, dampers, valves, charcoal filter train, and controls. Both SGT subsystems share a common inlet plenum. This inlet plenum is connected to the refueling floor ventilation exhaust duct for each Unit and to the suppression chamber and drywell of each Unit. Both SGT subsystems exhaust to the plant offgas stack through a common exhaust duct served by three 100% capacity system fans. SGT System fans 0AV020 and 0BV020 automatically start on Unit 2 secondary containment isolation signals. SGT System fans 0CV020 and 0BV020 automatically start on Unit 3 secondary containment isolation signals.

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

a. A demister or moisture separator;

b. An electric heater;

c. A prefilter;

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

e. A charcoal adsorber; and

f. A second HEPA filter.

These filters are not credited in any DBA analysis.

The SGT System is sized such that each 100% capacity fan will provide a flow rate of 10,500 cfm at 20 inches water gauge static pressure to support the control of fission product releases. The SGT System is designed to restore and maintain secondary containment at a negative pressure of 0.25 inches water gauge relative to the atmosphere following

(continued)

BASES

BACKGROUND (continued)

the receipt of a secondary containment isolation signal. Maintaining this negative pressure is based upon the existence of calm wind conditions (up to 5 mph), a maximum SGT System flow rate of 10,500 cfm, outside air temperature of 95°F and a temperature of 150°F for air entering the SGT System from inside secondary containment.

The demister is provided to remove entrained water in the air, while the electric heater reduces the relative humidity of the airstream to less than 70% (Ref. 2). Although not credited in any DBA analysis, the prefilter removes large particulate matter, while the HEPA filter removes fine particulate matter and protects the charcoal from fouling. The charcoal adsorber removes gaseous elemental iodine and organic iodides, and the final HEPA filter collects any carbon fines exhausted from the charcoal adsorber.

The SGT System automatically starts and operates in response to actuation signals indicative of conditions or an accident that could require operation of the system. Following initiation, two charcoal filter train fans (OCV020 and OBV020) start. Upon verification that both subsystems are operating, the redundant subsystem is normally shut down.

APPLICABLE SAFETY ANALYSES

The design basis for the SGT System is to mitigate the consequences of a loss of coolant accident by providing a controlled, elevated release path. The SGT system also provides this function for OPDRVs. For all events where required, the SGT System automatically initiates to reduce, via an elevated release, the consequences of radioactive material released to the environment.

The HEPA filter and charcoal adsorber provided in the SGT System are not credited for any DBA analysis.

The SGT System satisfies Criterion 3 of the NRC Policy Statement.

LCO

Following a DBA, a minimum of one SGT subsystem is required to maintain the secondary containment at a negative pressure with respect to the environment and to process gaseous releases. Meeting the LCO requirements for two OPERABLE subsystems ensures operation of at least one SGT subsystem in the event of a single active failure.

(continued)

BASES

LCO (continued)

For Unit 3, one SGT subsystem is OPERABLE when one fan (OCV020) and associated ductwork, dampers, valves, and controls are OPERABLE. The second SGT subsystem is OPERABLE when the other fan (OBV020) and associated ductwork, damper, valves, and controls are OPERABLE.

APPLICABILITY

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

In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the SGT System in OPERABLE status is not required in MODE 4 or 5, except for other situations under which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs) or during movement of recently irradiated fuel assemblies in the secondary containment. The SGT System is only required to be OPERABLE during handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

ACTIONS

A.1

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

B.1 and B.2

If the SGT subsystem cannot be restored to OPERABLE status within the required Completion Time in MODE 1, 2, or 3, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

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

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

An alternative to Required Action C.1 is to immediately suspend activities that represent a potential for releasing a significant amount of radioactive material to the secondary containment, thus placing the plant in a condition that minimizes risk. If applicable, movement of recently irradiated fuel assemblies must immediately be suspended. Suspension of this activity must not preclude completion of movement of a component to a safe position. Also, if applicable, actions must immediately be initiated to suspend OPDRVs in order to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

The Required Actions of Condition C have been modified by a Note stating that LCO 3.0.3 is not applicable, since the movement of recently irradiated fuel can only be performed in MODES 4 and 5.

(continued)

BASES

ACTIONS (continued)

D.1

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

E.1 and E.2

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

Required Action E.1 has been modified by a Note stating that LCO 3.0.3 is not applicable. , since the movement of recently irradiated fuel can only be performed in MODES 4 and 5.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.3.1

Operating each SGT subsystem (including each filter train fan) for ≥ 15 minutes ensures that both subsystems are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. The 31 day Frequency was developed in consideration of the known reliability of fan motors and controls and the redundancy available in the system.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.4.3.2

This SR verifies that the required SGT filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The only credited safety function of the SGT System is to provide a secondary containment vacuum sufficient to assure that discharges from the secondary containment will be through the Main Stack. The VFTP test 5.5.7.d. provides verification that the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is acceptable. SR 3.6.4.1.3 and SR 3.6.4.1.4 provide assurance that sufficient vacuum in the secondary containment is established with the time period as used in the DBA LOCA analysis. Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.6.4.3.3

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

REFERENCES

1. UFSAR, Section 1.5.1.6.
2. UFSAR, Section 14.9.

BASES

BACKGROUND (continued)

initiate an emergency shutdown of non-essential equipment and lighting to reduce the heat generation to a minimum. Heat removal would be accomplished by conduction through the floors, ceilings, and walls to adjacent rooms and to the environment. Additionally, the MCREV System is designed to maintain the control room environment for a 30-day occupancy after a DBA without exceeding 5 rem TEDE. A single MCREV subsystem will pressurize the control room to prevent infiltration of air from surrounding buildings. MCREV System operation in maintaining control room habitability is discussed in the UFSAR, Chapters 7, 10, and 12, (Refs. 1, 2, and 3, respectively).

APPLICABLE SAFETY ANALYSES

The ability of the MCREV System to maintain the habitability of the control room is an explicit assumption for the safety analyses presented in the UFSAR, Chapters 10 and 12 (Refs. 2 and 3, respectively). The MCREV System is credited as operating following a loss of coolant accident. The MCREV System is not credited in the analysis of the fuel handling accident, the main steam line break, or the control rod drop accident, as discussed in the UFSAR, Section 14.9.1.5 (Ref. 4). The radiological doses to control room personnel as a result of the various DBAs are summarized in Reference 4. No single active or passive failure will cause the loss of outside or recirculated air from the control room.

The MCREV System satisfies Criterion 3 of the NRC Policy Statement.

LCO

Two redundant subsystems of the MCREV System are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other subsystem. Total system failure could result in exceeding a dose of 5 rem TEDE to the control room operators in the event of a LOCA.

The MCREV System is considered OPERABLE when the individual components necessary to control operator exposure are OPERABLE in both subsystems. A subsystem is considered OPERABLE when its associated:

a. Fan is OPERABLE;

(continued)

BASES

LCO (continued)

b. HEPA filter 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 flow can be maintained.

In addition, the control room boundary must be maintained, including the integrity of the walls, floors, ceilings, and ductwork. Temporary seals may be used to maintain the boundary. In addition, an access door may be opened provided the ability to pressurize the control room is maintained and the capability exists to close the affected door in an expeditious manner.

APPLICABILITY

In MODES 1, 2, and 3, the MCREV System must be OPERABLE to control operator exposure during and following a LOCA, since the LOCA could lead to a fission product release.

In MODES 4 and 5, the probability and consequences of a LOCA are reduced because of the pressure and temperature limitations in these MODES. Therefore, maintaining the MCREV System OPERABLE is not required in MODE 4 or 5, except for the following situations under which significant radioactive releases can be postulated:

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

b. During movement of recently irradiated fuel assemblies in the secondary containment.

ACTIONS

A.1

With one MCREV subsystem inoperable, the inoperable MCREV subsystem must be restored to OPERABLE status within 7 days. With the unit in this condition, the remaining OPERABLE MCREV subsystem is adequate to maintain control room temperature and to perform control room radiation protection. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could

(continued)

BASES

ACTIONS

A.1 (continued)

result in reduced MCREV System capability. The 7 day Completion Time is based on the low probability of a LOCA occurring during this time period, and that the remaining subsystem can provide the required capabilities.

B.1 and B.2

In MODE 1, 2, or 3, if the inoperable MCREV subsystem cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE that minimizes risk. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours and in MODE 4 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.

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

The Required Actions of Condition C are modified by a Note indicating that LCO 3.0.3 does not apply, since the movement of recently irradiated fuel can only be performed in MODES 4 and 5.

During movement of recently irradiated fuel assemblies in the secondary containment or during OPDRVs, if the inoperable MCREV subsystem cannot be restored to OPERABLE status within the required Completion Time, the OPERABLE MCREV subsystem may be placed in operation. This action ensures that the remaining subsystem is OPERABLE, that no failures that would prevent automatic actuation will occur, and that any active failure will be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

(continued)

BASES

ACTIONS

C.1, C.2.1, and C.2.2 (continued)

If applicable, movement of recently irradiated fuel assemblies in the secondary containment must be suspended immediately. Suspension of this activity shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and the subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended.

D.1

If both MCREV subsystems are inoperable in MODE 1, 2, or 3, the MCREV System may not be capable of performing the intended function and the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

E.1 and E.2

The Required Actions of Condition E are modified by a Note indicating that LCO 3.0.3 does not apply, since the movement of recently irradiated fuel can only be performed in MODES 4 and 5.

During movement of recently irradiated fuel assemblies in the secondary containment or during OPDRVs, with two MCREV subsystems inoperable, action must be taken immediately to suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

If applicable, movement of recently irradiated fuel assemblies in the secondary containment must be suspended immediately. Suspension of this activity shall not preclude completion of movement of a component to a safe position. If applicable, actions must be initiated

(continued)

BASES

ACTIONS

E.1 and E.2 (continued)

immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

This SR verifies that a subsystem in a standby mode starts on demand and continues to operate for ≥ 15 minutes. Standby systems should be checked periodically to ensure that they start and function properly. As the environmental and normal operating conditions of this system are not severe, testing each subsystem once every month provides an adequate check on this system. Furthermore, the 31 day Frequency is based on the known reliability of the equipment and the two subsystem redundancy available.

SR 3.7.4.2

This SR verifies that the required MCREV testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.4.3

This SR verifies that on an actual or simulated initiation signal, each MCREV subsystem starts and operates. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.7.1.4 overlaps this SR to provide complete testing of the safety function. Operating experience has shown that these components will usually pass the SR when performed at the 24 month Frequency. Therefore, this Frequency is concluded to be acceptable from a reliability standpoint.

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.5 Main Condenser Offgas

BASES

BACKGROUND	During unit operation, steam from the low pressure turbine is exhausted directly into the condenser. Air and noncondensible gases are collected in the condenser, then exhausted through the steam jet air ejectors (SJAEs) to the Main Condenser Offgas System. The offgas from the main condenser normally includes radioactive gases. The Main Condenser Offgas System has been incorporated into the unit design to reduce the gaseous radwaste emission. This system uses a catalytic recombiner to recombine radiolytically dissociated hydrogen and oxygen. The gaseous mixture is cooled and water vapor removed by the offgas recombiner condenser; the remaining water and condensibles are stripped out by the cooler condenser and moisture separator. The remaining gaseous mixture (i.e., the offgas recombiner effluent) is then processed by a charcoal adsorber bed prior to release.
APPLICABLE SAFETY ANALYSES	The main condenser offgas gross gamma activity rate is an initial condition of the Main Condenser Offgas System failure event, discussed in the UFSAR, Section 9.4.5 (Ref. 1). The analysis assumes a gross failure in the Main Condenser Offgas System that results in the rupture of the Main Condenser Offgas System pressure boundary. The gross gamma activity rate is controlled to ensure that, during the event, the calculated offsite doses will be well within the limits of 10 CFR 50.67 (Ref. 2) or the NRC staff approved licensing basis. The main condenser offgas limits satisfy Criterion 2 of the NRC Policy Statement.
LCO	To ensure compliance with the assumptions of the Main Condenser Offgas System failure event (Ref. 1), the fission product release rate should be consistent with a noble gas release to the reactor coolant of 100 μCi/MWt-second after decay of 30 minutes. The LCO is established consistent

(continued)

BASES

ACTIONS

B.1, B.2, B.3.1, and B.3.2 (continued)

experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

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

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

REFERENCES

1. UFSAR, Section 9.4.5.

2. 10 CFR 50.67.

B 3.7 PLANT SYSTEMS

B 3.7.7 Spent Fuel Storage Pool Water Level

BASES

BACKGROUND

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

A general description of the spent fuel storage pool design is found in the UFSAR, Section 10.3 (Ref. 1). The assumptions of the fuel handling accident are found in the UFSAR, Section 14.6.4 (Ref. 2).

APPLICABLE SAFETY ANALYSES

The water level above the irradiated fuel assemblies is an implicit assumption of the fuel handling accident. A fuel handling accident is evaluated to ensure that the radiological consequences are well below the guidelines set forth in 10 CFR 50.67 (Ref. 3). A fuel handling accident could release a fraction of the fission product inventory by breaching the fuel rod cladding as discussed in Reference 2.

The fuel handling accident is evaluated for the dropping of an irradiated fuel assembly onto the reactor core. The consequences of a fuel handling accident over the spent fuel storage pool are less severe than those of the fuel handling accident over the reactor core. The water level in the spent fuel storage pool provides for absorption of water soluble fission product gases before being released to the secondary containment atmosphere.

The spent fuel storage pool water level satisfies Criteria 2 and 3 of the NRC Policy Statement.

LCO

The specified water level (232 ft 3 inches plant elevation, which is equivalent to 22 ft over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks) preserves the assumptions of the fuel handling accident analysis (Ref. 2). As such, it is the minimum required for fuel movement within the spent fuel storage pool.

(continued)

BASES (continued)

APPLICABILITY

This LCO applies during movement of fuel assemblies in the spent fuel storage 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. If moving fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of fuel assemblies is not a sufficient reason to require a reactor shutdown.

When the initial conditions for an accident cannot be met, action must be taken to preclude the accident from occurring. If the spent fuel storage pool level is less than required, the movement of fuel assemblies in the spent fuel storage pool is suspended immediately. Suspension of this activity shall not preclude completion of movement of a fuel assembly to a safe position. This effectively precludes a spent fuel handling accident from occurring.

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

This SR verifies that sufficient water is available in the event of a fuel handling accident. The water level in the spent fuel storage pool must be checked periodically. The 7 day Frequency is acceptable, based on operating experience, considering that the water volume in the pool is normally stable, and all water level changes are controlled by unit procedures.

REFERENCES

1. UFSAR, Section 10.3.

2. UFSAR, Section 14.6.4.

3. 10 CFR 50.67.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.3 (continued)

Note 1 modifies this Surveillance to indicate that diesel engine runs for this Surveillance may include gradual loading, as recommended by the manufacturer, so that mechanical stress and wear on the diesel engine are minimized.

Note 2 modifies this Surveillance by stating that momentary transients because of changing bus loads do not invalidate this test. Similarly, momentary power factor transients above the limit do not invalidate the test.

Note 3 indicates that this Surveillance should be conducted on only one DG at a time in order to avoid common cause failures that might result from offsite circuit or grid perturbations.

Note 4 stipulates a prerequisite requirement for performance of this SR. A successful DG start must precede this test to credit satisfactory performance.

To minimize testing of the DGs, Note 5 allows a single test (instead of two tests, one for each unit) to satisfy the requirements for both units, with the DG synchronized to the 4 kV emergency bus of Unit 3 for one periodic test and synchronized to the 4 kV emergency bus of Unit 2 during the next periodic test. This is allowed since the main purpose of the Surveillance, to ensure DG OPERABILITY, is still being verified on the proper frequency, and each unit's breaker control circuitry, which is only being tested every second test (due to the staggering of the tests), historically have a very low failure rate. Note 5 modifies the specified frequency for each unit's breaker control circuitry to be 62 days. If the DG fails one of these Surveillances, the DG should be considered inoperable on both units, unless the cause of the failure can be directly related to only one unit. In addition, if the test is scheduled to be performed on Unit 2, and the Unit 2 TS allowance that provides an exception to performing the test is used (i.e., when Unit 2 is in MODE 4 or 5, or moving recently irradiated fuel assemblies in the secondary containment, the Note to Unit 2 SR 3.8.2.1 provides an exception to performing this test) or if it is not preferable to perform the test on a unit due to operational concerns (however time is not to exceed 62 days plus grace), then the test shall be performed synchronized to the Unit 3 4 kV emergency bus.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.20 (continued)

The 10 year Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 8). This SR is modified by two Notes. The reason for Note 1 is to minimize wear on the DG during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations. To minimize testing of the DGs, Note 2 allows a single test (instead of two tests, one for each unit) to satisfy the requirements for both units. This is allowed since the main purpose of the Surveillance can be met by performing the test on either unit. If a DG fails one of these Surveillances, a DG should be considered inoperable on both units, unless the cause of the failure can be directly related to only one unit.

SR 3.8.1.21

With the exception of this Surveillance, all other Surveillances of this Specification (SR 3.8.1.1 through SR 3.8.1.20) are applied only to the Unit 3 AC sources. This Surveillance is provided to direct that the appropriate Surveillances for the required Unit 2 AC sources are governed by the applicable Unit 2 Technical Specifications. Performance of the applicable Unit 2 Surveillances will satisfy Unit 2 requirements, as well as satisfying this Unit 3 Surveillance Requirement. Six exceptions are noted to the Unit 2 SRs of LCO 3.8.1. SR 3.8.1.8 is excepted when only one Unit 2 offsite circuit is required by the Unit 3 Specification, since there is not a second circuit to transfer to. SR 3.8.1.12, SR 3.8.1.13, SR 3.8.1.17, SR 3.8.1.18 (ECCS load block requirements only), and SR 3.8.1.19 are excepted since these SRs test the Unit 2 ECCS initiation signal, which is not needed for the AC sources to be OPERABLE on Unit 3.

The Frequency required by the applicable Unit 2 SR also governs performance of that SR for Unit 3.

As Noted, if Unit 2 is in MODE 4 or 5, or moving recently irradiated fuel assemblies in the secondary containment, the Note to Unit 2 SR 3.8.2.1 is applicable. This ensures that a Unit 3 SR will not require a Unit 2 SR to be performed, when the

(continued)

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 4 and 5 and during movement of recently irradiated fuel assemblies in secondary containment ensures that:

a. The facility 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 an inadvertent draindown of the vessel or a fuel handling accident involving recently irradiated fuel. 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 24 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 loss of 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, and 3 have no specific analyses in MODES 4 and 5. Worst case bounding events are deemed not credible in MODES 4 and 5 because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and corresponding stresses result in the probabilities of occurrences significantly reduced or eliminated, and 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, and 3, various deviations from the analysis assumptions and design requirements are allowed within the ACTIONS. This allowance is in recognition that

(continued)

BASES

LCO (continued)

offsite circuit. In addition some equipment that may be required by Unit 3 is powered from Unit 2 sources (e.g., Containment Atmospheric Dilution System, Standby Gas Treatment System, Emergency Service Water System, and Main Control Room Emergency Ventilation System). Therefore, qualified circuits between the offsite transmission network and the Unit 2 onsite Class 1E AC electrical power distribution subsystem(s), and the DG(s) (not necessarily different DG(s) from those being used to meet LCO 3.8.2.b requirements) capable of supplying power to the required Unit 2 subsystems of each of the required components must also be OPERABLE. Together, OPERABILITY of the required offsite circuit(s) and required DG(s) ensures the availability of sufficient AC sources to operate the plant in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents involving recently irradiated fuel and reactor vessel draindown).

The qualified Unit 3 offsite circuit must be capable of maintaining rated frequency and voltage while connected to the respective Unit 3 4 kV emergency bus(es), and of accepting required loads during an accident. Qualified offsite circuits are those that are described in the UFSAR, Technical Specification Bases Section 3.8.1 and are part of the licensing basis for the unit. A Unit 3 offsite circuit consists of the incoming breaker and disconnect to the startup and emergency auxiliary transformer, the respective circuit path to the emergency auxiliary transformer and the circuit path to the Unit 3 4 kV emergency buses required by LCO 3.8.8, including feeder breakers to the required Unit 3 4 kV emergency buses. A qualified Unit 2 offsite circuit's requirements are the same as the Unit 3 circuit's requirements, except that the circuit path, including the feeder breakers, is to the Unit 2 4 kV emergency buses required to be OPERABLE by LCO 3.8.8.

The required DGs must be capable of starting, accelerating to rated speed and voltage, and connecting to their respective Unit 3 emergency bus on detection of bus undervoltage. This sequence must be accomplished within 10 seconds. Each DG must also be capable of accepting required loads within the assumed loading sequence intervals, and must continue to operate until offsite power can be restored to the 4 kV emergency buses. These capabilities are required to be met from a variety of initial conditions such as DG in standby with engine hot and DG in standby with engine at ambient conditions. Additional

(continued)

BASES

LCO (continued)

DG capabilities must be demonstrated to meet required Surveillances, e.g., capability of the DG to revert to standby status on an ECCS signal while operating in parallel test mode. Proper sequencing of loads is a required function for DG OPERABILITY. The necessary portions of the Emergency Service Water System are also required to provide appropriate cooling to each required DG.

The OPERABILITY requirements for the DG capable of supplying power to the Unit 2 powered equipment are the same as described above, except that the required DG must be capable of connecting to its respective Unit 2 4 kV emergency bus. (In addition, the Unit 2 ECCS initiation logic SRs are not applicable, as described in SR 3.8.2.2 Bases.)

It is acceptable for 4 kV emergency buses to be cross tied during shutdown conditions, permitting a single offsite power circuit to supply all required buses. No automatic transfer capability is required for offsite circuits to be considered OPERABLE.

APPLICABILITY

The AC sources are required to be OPERABLE in MODES 4 and 5 and during movement of recently irradiated fuel assemblies in the secondary containment to provide assurance that:

a. Systems providing adequate coolant inventory makeup are available for the recently irradiated fuel assemblies in the core in case of an inadvertent draindown of the reactor vessel;

b. Systems needed to mitigate a fuel handling accident involving recently irradiated fuel are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

AC power requirements for MODES 1, 2, and 3 are covered in LCO 3.8.1.

(continued)

BASES (continued)

ACTIONS

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply since the movement of recently irradiated fuel can only be performed in MODES 4 and 5.

A.1 and B.1

With one or more required offsite circuits inoperable, or with one DG inoperable, the remaining required sources may be capable of supporting sufficient required features (e.g., system, subsystem, division, component, or device) to allow continuation of CORE ALTERATIONS, recently irradiated fuel movement, and operations with a potential for draining the reactor vessel. For example, if two or more 4 kV emergency buses are required per LCO 3.8.8, one 4 kV emergency bus with offsite power available may be capable of supplying sufficient required features. By the allowance of the option to declare required features inoperable that are not powered from offsite power (Required Action A.1) or capable of being powered by the required DG (Required Action B.1), appropriate restrictions can be implemented in accordance with the affected feature(s) LCOs' ACTIONS. Required features remaining powered from a qualified offsite power circuit, even if that circuit is considered inoperable because it is not powering other required features, are not declared inoperable by this Required Action. If a single DG is credited with meeting both LCO 3.8.2.d and one of the DG requirements of LCO 3.8.2.b, then the required features remaining capable of being powered by the DG are not declared inoperable by this Required Action, even if the DG is considered inoperable because it is not capable of powering other required features.

A.2.1, A.2.2, A.2.3, A.2.4, B.2.1, B.2.2, B.2.3, B.2.4, C.1, C.2, C.3, and C.4

With an offsite circuit not available to all required 4 kV emergency buses or one required DG inoperable, the option still exists to declare all required features inoperable

(continued)

BASES

ACTIONS

A.2.1, A.2.2, A.2.3, A.2.4, B.2.1, B.2.2, B.2.3, B.2.4, C.1, C.2, C.3, and C.4 (continued)

(per Required Actions A.1 and B.1). Since this option may involve undesired administrative efforts, the allowance for sufficiently conservative actions is made. With two or more required DGs inoperable, the minimum required diversity of AC power sources may not be available. It is, therefore, required to suspend CORE ALTERATIONS, movement of recently irradiated fuel assemblies in the secondary containment, and activities that could result in inadvertent draining of the reactor vessel.

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required AC sources and to continue this action until restoration is accomplished in order to provide the necessary AC power to the plant safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required AC electrical power sources should be completed as quickly as possible in order to minimize the time during which the plant safety systems may be without sufficient power.

Pursuant to LCO 3.0.6, the Distribution System ACTIONS would not be entered even if all AC sources to it are inoperable, resulting in de-energization. Therefore, the Required Actions of Condition A have been modified by a Note to indicate that when Condition A is entered with no AC power to any required 4 kV emergency bus, ACTIONS for LCO 3.8.8 must be immediately entered. This Note allows Condition A to provide requirements for the loss of the offsite circuit whether or not a required bus is de-energized. LCO 3.8.8 provides the appropriate restrictions for the situation involving a de-energized bus.

SURVEILLANCE REQUIREMENTS

SR 3.8.2.1

SR 3.8.2.1 requires the SRs from LCO 3.8.1 that are necessary for ensuring the OPERABILITY of the Unit 3 AC sources in other than MODES 1, 2, and 3. SR 3.8.1.8 is not

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.4.8 (continued)

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required DC electrical power subsystem from service, perturb the electrical distribution system, and challenge safety systems. Credit may be taken for unplanned events that satisfy the Surveillance. The DC batteries of the other unit are exempted from this restriction since they are required to be OPERABLE by both units and the Surveillance cannot be performed in the manner required by the Note without resulting in a dual unit shutdown.

SR 3.8.4.9

With the exception of this Surveillance, all other Surveillances of this Specification (SR 3.8.4.1 through SR 3.8.4.8) are applied only to the Unit 3 DC electrical power subsystems. This Surveillance is provided to direct that the appropriate Surveillances for the required Unit 2 DC electrical power subsystems are governed by the Unit 2 Technical Specifications. Performance of the applicable Unit 2 Surveillances will satisfy Unit 2 requirements, as well as satisfying this Unit 3 Surveillance Requirement.

The Frequency required by the applicable Unit 2 SR also governs performance of that SR for Unit 3. As Noted, if Unit 2 is in MODE 4 or 5, or moving recently irradiated fuel assemblies in the secondary containment, the Note to Unit 2 SR 3.8.5.1 is applicable. This ensures that a Unit 3 SR will not require a Unit 2 SR to be performed, when the Unit 2 Technical Specifications exempts performance of a Unit 2 SR. (However, as stated in the Unit 2 SR 3.8.5.1 Note, while performance of the SR is exempted, the SR still must be met.)

REFERENCES

1. UFSAR, Chapter 14.

2. "Proposed IEEE Criteria for Class 1E Electrical Systems for Nuclear Power Generating Stations," June 1969.

3. IEEE Standard 485, 1983.

(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 UFSAR, Chapter 14 (Ref. 1), assume that Engineered Safety Feature systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the diesel generators (DGs), 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 4 and 5 and during movement of recently irradiated fuel assemblies in secondary containment ensures that:

a. The facility 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 an inadvertent draindown of the vessel or a fuel handling accident involving recently irradiated fuel.

The DC sources satisfy Criterion 3 of the NRC Policy Statement.

LCO

The Unit 3 DC electrical power subsystems, with each DC subsystem consisting of two 125 V station batteries in series, two battery chargers (one per battery), and the corresponding control equipment and interconnecting cabling supplying power to the associated bus, are required to be

(continued)

BASES

LCO (continued)

OPERABLE to support Unit 3 DC distribution subsystems required OPERABLE by LCO 3.8.8, "Distribution Systems—Shutdown." When the equipment required OPERABLE: 1) does not require 250 VDC from the DC electrical power subsystem; and 2) does not require 125 VDC from one of the two 125 V batteries of the DC electrical power subsystem, the Unit 3 DC electrical power subsystem requirements can be modified to only include one 125 V battery (the battery needed to provide power to required equipment), an associated battery charger, and the corresponding control equipment and interconnecting cabling supplying 125 V power to the associated bus. This exception is allowed only if all 250 VDC loads are removed from the associated bus. In addition, DC control power (which provides control power for the 4 kV load circuit breakers and the feeder breakers to the 4 kV emergency bus) for two of the four 4 kV emergency buses, as well as control power for two of the diesel generators, is provided by the Unit 2 DC electrical power subsystems. Therefore, the Unit 2 DC electrical power subsystems needed to support required components are also required to be OPERABLE. The Unit 2 DC electrical power subsystem OPERABILITY requirements are the same as those required for a Unit 3 DC electrical power subsystem. In addition, battery chargers (Unit 2 and Unit 3) can be powered from the opposite unit's AC source (as described in the Background section of the Bases for LCO 3.8.4, "DC Sources—Operating"), and be considered OPERABLE for the purpose of meeting this LCO.

This requirement ensures the availability of sufficient DC electrical 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 recently irradiated fuel and inadvertent reactor vessel draindown).

APPLICABILITY

The DC electrical power sources required to be OPERABLE in MODES 4 and 5 and during movement of irradiated fuel assemblies in the secondary containment provide assurance that:

a. Required features to provide adequate coolant inventory makeup are available for the recently irradiated fuel assemblies in the core in case of an inadvertent draindown of the reactor vessel;

(continued)

BASES

APPLICABILITY (continued)

b. Required features needed to mitigate a fuel handling accident involving recently irradiated fuel are available;

c. Required features necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

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 24 hours).

The DC electrical power requirements for MODES 1, 2, and 3 are covered in LCO 3.8.4.

ACTIONS

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply since the movement of recently irradiated fuel can only be performed in MODES 4 and 5.

A.1, A.2.1, A.2.2, A.2.3, and A.2.4

If more than one DC distribution subsystem is required according to LCO 3.8.8, the DC electrical power subsystems remaining OPERABLE with one or more DC electrical power subsystems inoperable may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS, recently irradiated fuel movement, and operations with a potential for draining the reactor vessel.

By allowance of the option to declare required features inoperable with associated DC electrical power subsystems inoperable, appropriate restrictions are implemented in accordance with the affected system LCOs' ACTIONS. However, in many instances, this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of recently irradiated fuel assemblies in secondary containment, and any activities that could result in inadvertent draining of the reactor vessel).

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.5.2 (continued)

As Noted, if Unit 2 is in MODE 4 or 5, or moving recently irradiated fuel assemblies in the secondary containment, the Note to Unit 2 SR 3.8.5.1 is applicable. This ensures that a Unit 3 SR will not require a Unit 2 SR to be performed, when the Unit 2 Technical Specifications exempts performance of a Unit 2 SR. (However, as stated in the Unit 2 SR 3.8.5.1 Note, while performance of an SR is exempted, the SR still must be met.)

REFERENCES

1. UFSAR, Chapter 14.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.8 Distribution Systems—Shutdown

BASES

BACKGROUND

A description of the AC and DC electrical power distribution system is provided in the Bases for LCO 3.8.7, "Distribution Systems—Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident and transient analyses in the UFSAR, Chapter 14 (Ref. 1), assume Engineered Safety Feature (ESF) systems are OPERABLE. The AC and DC 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 AC and DC 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 AC and DC electrical power sources and associated power distribution subsystems during MODES 4 and 5 and during movement of recently irradiated fuel assemblies in the secondary containment ensures that:

a. The facility 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 an inadvertent draindown of the vessel or a fuel handling accident involving recently irradiated fuel.

AC and DC electrical power 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 24 hours).

The AC and DC electrical power distribution systems satisfy Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

Various combinations of subsystems, equipment, and components are required OPERABLE by other LCOs, depending on the specific plant condition. Implicit in those requirements is the required OPERABILITY of necessary support required features. This LCO explicitly requires energization of the portions of the Unit 3 electrical distribution system necessary to support OPERABILITY of Technical Specifications required systems, equipment, and components—both specifically addressed by their own LCO, and implicitly required by the definition of OPERABILITY. In addition some components that may be required by Unit 3 receive power through Unit 2 electrical power distribution subsystems (e.g., Standby Gas Treatment System, Main Control Room Emergency Ventilation System, and DC control power for two of the four 4 kV emergency buses, as well as control power for two of the diesel generators). Therefore, Unit 2 AC and DC electrical power distribution subsystems needed to support the required equipment must also be OPERABLE.

In addition, it is acceptable for required buses to be cross-tied during shutdown conditions, permitting a single source to supply multiple redundant buses, provided the source is capable of maintaining proper frequency (if required) and voltage.

Maintaining these portions of the distribution system energized ensures the availability of sufficient power to operate the plant in a safe manner to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents involving recently irradiated fuel and inadvertent reactor vessel draindown).

APPLICABILITY

The AC and DC electrical power distribution subsystems required to be OPERABLE in MODES 4 and 5 and during movement of recently irradiated fuel assemblies in the secondary containment provide assurance that:

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core in case of an inadvertent draindown of the reactor vessel;

b. Systems needed to mitigate a fuel handling accident involving recently irradiated fuel are available;

(continued)

BASES

APPLICABILITY (continued)

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The AC and DC electrical power distribution subsystem requirements for MODES 1, 2, and 3 are covered in LCO 3.8.7.

ACTIONS

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply since the movement of recently irradiated fuel can only be performed in MODES 4 and 5.

A.1, A.2.1, A.2.2, A.2.3, A.2.4, and A.2.5

Although redundant required features may require redundant electrical power distribution subsystems to be OPERABLE, one OPERABLE distribution subsystem may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS, recently irradiated fuel movement, and operations with a potential for draining the reactor vessel. By allowing the option to declare required features inoperable with associated electrical power distribution subsystems inoperable, appropriate restrictions are implemented in accordance with the affected distribution subsystem LCO's Required Actions. However, in many instances this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made, (i.e., to suspend CORE ALTERATIONS, movement of recently irradiated fuel assemblies in the secondary containment, and any activities that could result in inadvertent draining of the reactor vessel).

(continued)

ATTACHMENT 6

PEACH BOTTOM ATOMIC POWER STATION
UNITS 2 AND 3

Docket Nos. 50-277
50-278

License Nos. DPR-44
DPR-56

License Amendment Request
"PBAPS Alternative Source Term Implementation"

List Of Commitments

The following table identifies those actions committed to by Exelon in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

Commitment	Continuing Compliance	Scheduled Completion Date
Per TSTF-51, licensees adding the term "recently" must make the following commitment which is consistent with NUMARC 93-01, Revision 3, Section 11.3.6.5, "Safety Removal for Removal of Equipment from Service During Shutdown Conditions," subheading "Containment – Primary (PWR)/Secondary (BWR)". EGC makes a commitment to the following NUMARC 93-01 section: *"In addition to the guidance in NUMARC 91-06, for plants which obtain license amendments to utilize shutdown safety administrative controls in lieu of Technical Specification requirements on primary or secondary containment operability or ventilation system operability, during fuel handling or core alterations, the following guidelines should be included in the assessment of systems removed from service:* *-During Fuel Handling/Core Alterations, ventilation system and radiation monitor availability (as defined in NUMARC 91-06) should be assessed, with respect to filtration and monitoring of releases from the fuel. Following shutdown, radioactivity in the RCS decays away fairly rapidly. The basis of the Technical Specification operability amendment is the reduction in doses due to such decay. The goal of maintaining ventilation system and radiation monitoring availability is to reduce doses even further below that provided by the natural decay and to avoid unmonitored releases.* *-A single normal or contingency method to promptly close primary or secondary containment penetrations should be developed. Such prompt methods need not completely block the penetration or be capable of resisting pressure. The purpose of this is to enable ventilation systems to draw the release from a postulated fuel handling accident in the proper direction such that it can be treated and monitored."*	X	Upon Implementation.

ATTACHMENT 7

PEACH BOTTOM ATOMIC POWER STATION
UNITS 2 AND 3

Docket Nos. 50-277
50-278

License Nos. DPR-44
DPR-56

License Amendment Request
"PBAPS Alternative Source Term Implementation"

<u>Compact Disk Containing PBAPS Meteorological Data</u>

ATTACHMENT 8

PEACH BOTTOM ATOMIC POWER STATION
UNITS 2 AND 3

Docket Nos. 50-277
50-278

License Nos. DPR-44
DPR-56

License Amendment Request
"PBAPS Alternative Source Term Implementation"

<u>UFSAR Section 5.2.4.3.2 Mark-Up</u>