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RS-08-132 10 CFR 50.90

October 23, 2008

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

LaSalle County Station, Units 1 and 2
Facility Operating License Nos. NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

Subject: Additional Information Regarding Request for License Amendment Regarding Application of Alternative Source Term

References:

1. Letter from P. R. Simpson (Exelon Generation Company, LLC) to U. S. NRC, "Request for License Amendment Regarding Application of Alternative Source Term," dated August 26, 2008
2. Technical Specifications Task Force (TSTF) Traveler, TSTF-51, "Revise Containment Requirements During Handling of Irradiated Fuel and Core Alterations," Revision 2
3. NRC Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 2000
4. Letter from J. C. Roberts (Entergy Operations, Inc.) to U. S. NRC, "GGNS Pilot Full-Scope Application of NUREG-1465 Alternative Source Term Insights, Additional Information, Supporting LDC 2000-070," dated December 22, 2000

In Reference 1, Exelon Generation Company, LLC (EGC) submitted a request for an amendment to Facility Operating License Nos. NPF-11 and NPF-18 for LaSalle County Station (LSCS), Units 1 and 2, respectively, in accordance with 10 CFR 50.67, "Accident source term," and 10 CFR 50.90, "Application for amendment of license or construction permit." Specifically, the proposed change revises Technical Specifications (TS) to support the application of alternative source term (AST) methodology with respect to the loss-of-coolant accident and the fuel handling accident.

This letter, and the Attachments contained herein, supersedes the Reference 1 submittal in its entirety. The proposed change is requested to support a full-scope application of an alternative

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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. The proposed changes to the current licensing basis for LSCS include:

- TS and associated TS Bases revisions to reflect implementation of AST assumptions;
- TS and associated TS Bases revisions to increase primary containment allowable leakage;
- TS and associated TS Bases revisions to increase the leakage limit through any one main steam isolation valve;
- TS and associated TS 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 under these conditions:
  - Standby Gas Treatment,
  - Secondary Containment, and
  - Secondary Containment Isolation Valves;
- TS and associated TS Bases revisions to reflect use of the Standby Liquid Control system to buffer suppression pool pH to prevent iodine re-evolution during a postulated radiological release; and
- TS and associated TS Bases revisions to increase the secondary containment drawdown time from the existing five minutes to 15 minutes.

The proposed changes related to the applicability requirements during movement of irradiated fuel assemblies are consistent with Technical Specifications Task Force (TSTF) Traveler, TSTF-51 (i.e., Reference 2). TSTF-51 changes the TS operability requirements for engineered safety features such that they are not applicable 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.

This request is subdivided as follows.

- Attachment 1 provides a description and evaluation of the proposed change.
- Attachment 2 provides a markup of the affected TS pages.
- Attachment 3 provides a markup of the affected TS Bases pages. The TS Bases pages are provided for information only, and do not require NRC approval.
- Attachment 4 provides a list of regulatory commitments being made in this submittal.
- Attachment 5 contains tables describing conformance with NRC Regulatory Guide 1.183 (i.e., Reference 3).
- Attachments 6 through 9 provide calculations that were completed to support application of AST methodology at LSCS. The suppression pool pH calculation provided in Attachment 8 does not include Attachments F and G, since these documents were previously submitted to the NRC in Reference 4 for Grand Gulf Nuclear Station.

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

In Reference 1, EGC requested NRC approval of the proposed change by August 26, 2009. However, in light of the resubmittal, EGC is now requesting approval of the proposed change by October 23, 2009. Once approved, the amendment will be implemented within 90 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, "Notice for public comment; State consultation," paragraph (b), EGC is notifying the State of Illinois of this application for license amendment by transmitting a copy of this letter and its attachments to the designated State Official.

Should you have any questions concerning this letter, please contact Mr. Kenneth M. Nicely at (630) 657-2803.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 23rd day of October 2008.

Respectfully,

Patrick R. Simpson

Patrick R. Simpson
Manager – Licensing

Attachments:
1. Evaluation of Proposed Change
2. Markup of Proposed Technical Specifications Pages
3. Markup of Proposed Technical Specifications Bases Pages
4. Summary of Regulatory Commitments
5. Regulatory Guide 1.183 Conformance Matrix
6. Calculation L-003063, "Alternative Source Term Onsite and Offsite X/Q Values," Revision 1
7. Calculation L-003068, "Re-analysis of Loss of Coolant Accident (LOCA) Using Alternative Source Terms," Revision 1
8. Calculation L-003064, "Suppression Pool pH Calculation for Alternative Source Terms," Revision 1
9. Calculation L-003067, "Re-analysis of Fuel Handling Accident (FHA) Using Alternative Source Terms," Revision 1

cc: NRC Regional Administrator, Region III
NRC Senior Resident Inspector – LaSalle County Station
Illinois Emergency Management Agency – Division of Nuclear Safety

ATTACHMENT 1
Evaluation of Proposed Change

1.0 SUMMARY 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. NPF-11 and NPF-18 for LaSalle County Station (LSCS), Units 1 and 2, respectively. 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," (i.e., Reference 1) will continue to be used as the radiation dose basis for equipment qualification.

EGC has performed radiological consequence analyses of the design basis accident (DBA) loss-of-coolant accident (LOCA) and fuel handling accident (FHA), to support a full-scope implementation of AST as described in Section 1.2.1 of Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (i.e., Reference 2). The AST analyses for LSCS were performed following the guidance in Reference 2, Standard Review Plan 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms" (i.e., Reference 3), and 10 CFR 50.67. Attachment 5 provides a Regulatory Guide 1.183 conformance matrix.

Approval of this change will provide a realistic source term for LSCS 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, Detailed Description.

2.0 DETAILED DESCRIPTION

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 2. Reference 2 provides NRC-accepted guidance for application of AST. The use of AST changes only the regulatory assumptions regarding the analytical treatment of the 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 as discussed in Reference 1. Since the publication of Reference 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 (i.e., Reference 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, and
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 2.

EGC has performed radiological consequence analyses of the LOCA and FHA. These analyses were performed to support full-scope implementation of AST. The implementation consisted of the following tasks:

- Identification of the AST based on plant-specific analysis of core fission product inventory,
- Application of release fractions for the LOCA and FHA DBAs that could potentially result in control room (CR) and offsite doses,
- 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 CR 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.

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 DBA, and without crediting the use of certain previously assumed safety systems/functions.

The proposed changes related to the applicability requirements during movement of irradiated fuel assemblies are consistent with Technical Specifications Task Force (TSTF) Traveler TSTF-51 (i.e., Reference 5), which 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 (i.e., Reference 6), 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. Note that at the time TSTF-51, Revision 2 was issued, a reference to Section 11.2.6 of Draft NUMARC 93-01, Revision 3, was to be made. The final version of NUMARC 93-01, Revision 3, does not have a section numbered 11.2.6. Therefore, Section 11.3.6.5 was used since the section title in TSTF-51 matches that in the final version of NUMARC 93-01, Revision 3.

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

- TS and associated TS Bases revisions to reflect implementation of AST assumptions;
- TS and associated TS Bases revisions to increase primary containment allowable leakage;
- TS and associated TS Bases revisions to increase the leakage limit through any one main steam isolation valve (MSIV);
- TS and associated TS 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 under these conditions:
  - Standby Gas Treatment,
  - Secondary Containment, and
  - Secondary Containment Isolation Valves;
- TS and associated TS Bases revisions to reflect use of the Standby Liquid Control system to buffer suppression pool pH to prevent iodine re-evolution during a postulated radiological release; and
- TS and associated TS Bases revisions to increase the secondary containment drawdown time from the existing five minutes to 15 minutes.

The proposed revisions to the LSCS TS include the following.

2.1 TS Section 1.1, "Definitions"

The proposed change adds a new definition for RECENTLY IRRADIATED FUEL. RECENTLY IRRADIATED FUEL is fuel that has occupied part of a critical reactor core within the previous 24 hours.

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 limiting condition for operation (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 C 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"

Table 3.3.6.1-1 of TS Section 3.3.6.1 lists, in part, 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 (RWCU) 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 Plenum Radiation – High), Function 4 (i.e., Fuel Pool Ventilation Exhaust Radiation – High), and Function 5 (i.e., Manual Initiation) 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 when handling recently irradiated fuel. With the application of AST, secondary containment is not credited for the FHA after a 24 hour decay period. This change is supported by TSTF-51 (i.e., Reference 5).

2.5 TS Section 3.3.7.1, "Control Room Area Filtration (CRAF) System Instrumentation"

The proposed change deletes "During CORE ALTERATIONS" from the applicability statement for TS LCO 3.3.7.1 and relaxes TS requirements to require LCO 3.3.7.1 to be applicable when handling recently irradiated fuel. These changes are being made to reflect that, with application of AST, the CRAF 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. This change is supported by TSTF-51 (i.e., Reference 5).

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

The proposed change revises Surveillance Requirement (SR) 3.6.1.3.10 to increase the leakage limit through any one main steam line. Currently, the SR requires verification that the leakage rate through any one main steam line is less than or equal to 100 standard cubic feet per hour (scfh), and that the leakage rate through all four main steam lines is less than or equal to 400 scfh, when tested at greater than or equal to 25.0 psig. The proposed change increases the leakage limit through any one main steam line from 100 scfh to 200 scfh. The combined leakage rate limit through all four main steam lines is not being changed. The revised SR 3.6.1.3.10 reads:

> Verify leakage rate through any one main steam line is ≤ 200 scfh and through all four main steam lines is ≤ 400 scfh when tested at ≥ 25.0 psig.

The Frequency for SR 3.6.1.3.10 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 when handling recently irradiated fuel. 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. This change is supported by TSTF-51 (i.e., Reference 5).

In addition, the proposed change revises SR 3.6.4.1.3 to increase the secondary containment drawdown time from less than or equal to 300 seconds to less than or equal to 900 seconds. This change reflects the application of AST assumptions.

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 when handling recently irradiated fuel. 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. These changes are being made to reflect that, with the application of AST, closure of secondary containment isolation valves is not credited for the FHA after a 24 hour decay period. This change is supported by TSTF-51 (i.e., Reference 5).

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 when handling recently irradiated fuel. 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. This change is supported by TSTF-51 (i.e., Reference 5).

2.10 TS Section 3.7.4, "Control Room Area Filtration (CRAF) 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 when handling recently irradiated fuel. The proposed change revises Condition D and Condition F, and associated Required Actions and Completion Times, to reflect the revision of the applicability requirements for LCO 3.7.4. These changes are being made to reflect that, with application of AST, the CRAF 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. This change is supported by TSTF-51 (i.e., Reference 5).

2.11 TS Section 3.7.5, "Control Room Area Ventilation Air Conditioning (AC) System"

The proposed change deletes "During CORE ALTERATIONS" from the applicability statement for TS LCO 3.7.5 and relaxes TS requirements to require LCO 3.7.5 to be applicable when handling recently irradiated fuel. The proposed change revises Condition D and Condition E, and associated Required Actions and Completion Times, to reflect the revision of the applicability requirements for LCO 3.7.5. These changes are being made to reflect that, with application of AST, the Control Room Area Ventilation AC 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. This change is supported by TSTF-51 (i.e., Reference 5).

2.12 TS Section 5.5.13, "Primary Containment Leakage Rate Testing Program"

The proposed change increases the maximum allowable primary containment leakage rate, L_a, at P_a, from 0.635% to 1.0% of primary containment air weight per day. Application of AST supports the increase in maximum allowable primary containment leakage rate.

2.13 TS Section 5.5.15, "Control Room Envelope Habitability Program"

The proposed change revises TS Section 5.5.15 to reflect that, with the adoption of AST methodology, the CR dose acceptance criterion for the LOCA and FHA are expressed in terms of TEDE.

3.0 TECHNICAL EVALUATION

3.1 Introduction

3.1.1 Attributes of the LSCS AST

The LSCS AST is based on two major accidents (i.e., LOCA and FHA), hypothesized for the purposes of design analyses or consideration of possible accidental events that could result in hazards not exceeded by those from other accidents considered credible. The AST LOCA analysis addresses events that involve a substantial meltdown of the core with the subsequent release of appreciable quantities of fission products, the times and rates of appearance of radioactive fission products released into containment, the types and quantities of the radioactive species released, and the chemical forms of iodine released.

3.1.2 Accident Source Term

The inventory of fission products in the reactor core that is available for release to the containment is based on the maximum full power operation of the core with bounding values for fuel enrichment and fuel burnup. The core power used in the analyses (i.e., 3489 MWt) is the current licensed rated thermal power limit. The period of irradiation is of sufficient duration to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values. ORIGEN 2.1 (i.e., Reference 7) based methodology was used to determine core inventory. These source terms were evaluated at end-of-cycle and at beginning-of-cycle conditions (i.e., 100 effective full power days (EFPD) to achieve equilibrium) and worst-case inventory used for the selected isotopes. These values were then divided by 3489 MWt to obtain activity in units of Ci/MWt. The Ci/MWt activities are subsequently multiplied in RADTRAD dose calculations by 3559 MWt, which is equivalent to the current licensed rated thermal power (i.e., 3489 MWt) times the Emergency Core Cooling System (ECCS) evaluation uncertainty (i.e., 1.02). Source terms are based on a two year fuel cycle with a nominal 711 EFPD per cycle. Activation products Co-58 and Co-60 used RADTRAD default library values.

Sensitivity analyses performed for other EGC plants using various enrichment levels (i.e., 3.56% to 5%) and cycle lengths (i.e., 351 EFPD up to 740 EFPD) have confirmed that the source term used produces bounding doses for CR and offsite locations for a DBA.

The DBA LOCA analysis assumes all fuel assemblies in the core are affected and the core average inventory is used. For the FHA DBA event that does 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 are applied for the FHA in determining the inventory of the damaged rods.

No adjustment to the fission product inventory is 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 the FHA event postulated to occur while the facility is shutdown, radioactive decay is modeled at 24 hours from the time of shutdown.

3.1.3 Release Fractions

The core inventory release fractions, by radionuclide groups, for the gap release and early in-vessel damage phases for the DBA LOCA listed in Table 1 of Regulatory Guide 1.183 (i.e., Reference 2) for boiling water reactors (BWRs) are used. These fractions are applied to the equilibrium core inventory developed for LSCS.

For the FHA event, the fractions of the core inventory assumed to be in the gap for the various radionuclides in Table 3 of Regulatory Guide 1.183 are used. These release fractions are used in conjunction with the fission product inventory calculated with the maximum core radial peaking factor.

These release fractions are acceptable for use given that the peak fuel burnup meets the 62,000 MWD/MTU criterion specified in Regulatory Guide 1.183 Footnote 10. EGC's core design procedures currently require peak rod burnup of the fuel to be less than 62,000 MWD/MTU.

3.1.4 Timing of Release Phases

Table 4 of Regulatory Guide 1.183 tabulates the onset and duration of each sequential release phase for DBA LOCAs. 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 is modeled as increasing in a linear fashion over the duration of the phase. For the FHA DBA in which fuel damage is projected, the release from the fuel gap and the fuel pellet is assumed to occur instantaneously with the onset of the projected damage. The LSCS AST analyses use these release phases.

3.1.5 Radionuclide Composition

The elements and radionuclide groups listed in Table 5 of Regulatory Guide 1.183 are used in the LSCS AST analyses.

3.1.6 Chemical Form

Of the radioiodine released from the reactor coolant system (RCS) to the containment in a postulated accident, which includes releases from the gap and the fuel pellets, 95% of the iodine released is assumed to be cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. With the exception of elemental and organic iodine and noble gases, fission products are assumed to be in particulate form. The same chemical form is assumed in releases from fuel pins in FHAs. However, the transport of these iodine species following release from the fuel may affect these assumed fractions. The accident-specific descriptions that follow provide additional details.

3.1.7 General AST Input Parameters

Key baseline parameters, associated changes in DBA analysis parameters, and associated license change objectives are summarized in Table 3.1-1.

TABLE 3.1-1
General AST Analysis Parameter or Method

Parameter	Pre-AST Value	AST Value	Comments
Core Power Level	3559 MWt	3559 MWt	No change. This value corresponds to the DBA power level and equals 102% of the uprated thermal power of 3489 MWt.

TABLE 3.1-1 General AST Analysis Parameter or Method			
Parameter	**Pre-AST Value**	**AST Value**	**Comments**
Core Source Terms	Based on TID-14844	ORIGEN 2.1 values for the 60 RADTRAD isotopes using the highest calculated values for each isotope	New assumption for AST justified in design analysis.
Minimum Exclusion Area Boundary (EAB) Distance Low Population Zone (LPZ) Distance Elevated Stack Release Height	423 meters 6400 meters 112.8 meters	423 meters 6400 meters 112.8 meters	No change in distances. The EAB distance is conservatively measured from the nearest corner of the Turbine Building.
Containment Purging Considerations	Containment not assumed to be in purge/vent mode at the beginning of the DBA LOCA	Containment not assumed to be in purge/vent mode at the beginning of the DBA LOCA	No change. TS SR 3.6.1.3.1 identifies purposes for containment purging at LSCS as inerting, de-inerting, pressure control, as low as reasonably achievable (ALARA) or air quality considerations for personnel entry, and surveillances that require the valves to be open. TS 3.6.3.2 provides limitations on use for inerting and deinerting at power. Containment purging at LSCS is not a routine activity.
CR Volume	117,472 ft^3	117,472 ft^3	No change.
CR Occupancy Requirements	0-24 hrs: 1.0 1-4 days: 0.6 4-30 days: 0.4	0-24 hrs: 1.0 1-4 days: 0.6 4-30 days: 0.4	No change.

3.2 Scope of Evaluation

New design analyses were prepared for the simulation of the radionuclide release, transport, and removal for the LOCA and FHA. Dose estimates associated with the postulated LOCA and FHA were calculated. Releases were evaluated for full power conditions.

The main steam line break (MSLB) accident and control rod drop accident (CRDA) radiological consequence evaluations were not re-evaluated using AST methodology. However, the existing evaluations for these accidents were reviewed, and the CR, Technical Support Center (TSC), and offsite (i.e., EAB and LPZ) dose consequences are bounded by the LOCA evaluation provided in Attachment 7.

3.2.1 Offsite Dose Consequences

The following assumptions are used in determining the TEDE for the maximum exposed individual at EAB and LPZ locations.

- The offsite dose is determined as a TEDE, which is the sum of the committed effective dose equivalent (CEDE) from inhalation and the deep dose equivalent (DDE) from external exposure from all radionuclides that are significant with regard to dose consequences and the released radioactivity. The RADTRAD computer code performs this summation to calculate the TEDE.

- The offsite dose analysis uses the CEDE dose conversion factors (DCFs) for inhalation exposure. Table 2.1 of Federal Guidance Report 11 (i.e., Reference 8) provides tables of conversion factors acceptable to the NRC. The factors in the column headed "effective" yield doses corresponding to the CEDE.

- Since RADTRAD calculates DDE using whole body submergence in a semi-infinite cloud with appropriate credit for attenuation by body tissue, the DDE can be assumed nominally equivalent to the effective dose equivalent (EDE) from external exposure. Therefore, the offsite dose analysis uses EDE in lieu of DDE DCFs in determining external exposure. Table III.1 of Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water, and Soil" (i.e., Reference 9), provides external EDE conversion factors acceptable to the NRC. The factors in the column headed "effective" yield doses corresponding to the EDE.

- The maximum EAB TEDE for any two-hour period following the start of the radioactivity release is determined and used in determining compliance with the dose acceptance criteria in 10 CFR 50.67.

- TEDE is determined for the most limiting receptor at the outer boundary of the LPZ and is used in determining compliance with the dose criteria in 10 CFR 50.67. The breathing rates specified in Regulatory Guide 1.183 and/or Standard Review Plan Section 6.4 (i.e., References 2 and 10, respectively) are used.

- No correction is made for depletion of the effluent plume by deposition on the ground.

3.2.2 Control Room Dose Consequences

The following dose contributions were considered in determining the TEDE for maximum exposed individuals located in the CR:

- Contamination of the CR atmosphere by the intake or infiltration of the radioactive material contained in the radioactive plume released from the facility,
- Contamination of the CR atmosphere by the intake or infiltration of airborne radioactive material from areas and structures adjacent to the CR envelope (i.e., via CR unfiltered inleakage),
- Radiation shine from the external radioactive plume released from the facility (i.e., external airborne cloud),
- Radiation shine from radioactive material in the reactor containment, and
- Radiation shine from radioactive material in systems and components inside or external to the CR envelope (e.g., radioactive material buildup in ventilation filters).

The radioactivity releases and radiation levels used for the CR dose are determined using the same source term, transport, and release assumptions used for determining the EAB and the LPZ TEDE values.

The most limiting X/Q values generated for the CR intake are representative for CR inleakage.

No credit for potassium iodide pills or respiratory protection is taken.

3.2.3 TSC Dose Consequences

For the TSC and other areas requiring plant personnel access, assessments contained in the LOCA analysis indicate that radiation exposures would be within regulatory limits, without credit for installed TSC filtration systems, and with no new operator actions required.

3.3 Onsite Meteorological Measurements Program

The LSCS meteorological measurement program meets the guidelines of Regulatory Guide 1.23 (i.e., Safety Guide 23), "Onsite Meteorological Programs" (i.e., Reference 11). The meteorological 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. The program consists of monitoring wind direction, wind speed, temperature, and precipitation. The method used for determining atmospheric stability is delta temperature (delta-T), which measures the vertical temperature difference. These data, referenced in ANSI/ANS-2.5-1984 (i.e., Reference 12), are used to determine the meteorological conditions prevailing at the plant site.

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 months, with component checks and adjustments performed when required.

Inspections and maintenance of all equipment is accomplished in accordance with written procedures. Qualified technicians are available who are capable of performing 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 CR and to an onsite computer for archive storage. Periodically, all digital and analog data are sent to the approved meteorological consultant for data processing and analysis. Upon receipt of the digital data, the consultant performs a 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 quality check consists of time continuity, instrument malfunction, directional switching problems, negative speeds, missing data, and digital/analog correlation.

Data are compared with other monitoring site or regional data for consistency. If deviations occur, they are evaluated and dispositioned as appropriate.

3.3.1 Meteorological Data

The LSCS meteorological tower data for the six-year period, 1998-2003, were applied in the ARCON96 modeling analyses for the CR and TSC. Wind measurements were taken at three tower elevations (i.e., 33 ft, 200 ft, and 375 ft), and the vertical temperature difference (i.e., delta-T) was measured between 200 ft and 33 ft, and between 375 ft and 33 ft on the tower. Wind speeds reported as "calm" were assigned a value of 0.3 mph (i.e., 0.13 m/s). ARCON96, however, re-assigns a default value of 0.5 m/s to each wind speed lower than 0.5 m/s.

The same meteorological data were also used in the PAVAN analysis for offsite locations (i.e., EAB and LPZ). The format of PAVAN meteorological input consists of a joint wind direction based on sixteen 22.5 degree sectors, wind speed (i.e., 14 intervals), and stability class (i.e., seven classes) occurrence frequency distribution.

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.

3.3.2 Atmospheric Dispersion Factors

Attachment 6 provides calculation L-003063, "Alternative Source Term Onsite and Offsite X/Q Values." This calculation provides the assumptions, inputs, methods, and results of the calculation used to determine X/Q values. Highlights from the calculation are summarized below.

3.3.2.1 CR and AEER

The release locations for LSCS are the stack (i.e., inclusive of the co-located SGT system stack), the Unit 1 and Unit 2 MSIV pathway through the turbine seals, the Auxiliary Building roof access (i.e., the Auxiliary Building roof access door extending north of column 8.9), the Reactor Building truck bay door, the integrated leak rate test (ILRT) penetrations, and the Reactor Building wall. The stack, Unit 1 and Unit 2 MSIV, Auxiliary Building roof access, ILRT penetrations, and Reactor Building truck bay door are modeled as point sources, and the Reactor Building wall is modeled as a diffuse area source.

An additional set of PAVAN runs was also executed for the stack to CR/AEER intake scenario in accordance with Regulatory Guide 1.194 (i.e., Reference 13) guidance to determine the distance at which the actual maximum X/Q value would occur in each given downwind sector.

3.3.2.2 Point Sources

ARCON96 was executed for several postulated release locations to each of the two CR/AEER intakes (i.e., north and south).

The stack was modeled in ARCON96 as both an elevated release and a ground-level release. Modeling the stack as an elevated release is consistent with the LSCS current licensing basis. The stack was modeled as a ground-level release to obtain X/Q values to be utilized for the FHA only.

3.3.2.3 Diffuse Area Source (i.e., Reactor Building Wall)

In accordance with Section 3.2.4.5 of Reference 13, the diffuse area source representation in ARCON96 requires the building cross-sectional area to be calculated from the maximum building dimensions projected onto a vertical plane perpendicular to the line of sight from the building center to the intake. Figure 2 of Reference 13 specified that, for a diffuse area source, only that part of the structure above grade or an enclosing building should be included in the building height. For the Reactor Building wall scenarios, the portion of the Reactor Building above the Auxiliary Building roof height was used for determining the release height, building area, and vertical diffusion coefficient.

3.3.2.4 TSC

There are two release points identified for the TSC intake X/Q analysis: (1) the stack, and (2) the Unit 2 MSIV. The stack is modeled as an elevated release. The Unit 2 MSIV, which is conservatively located at the closest point to the intake along the high and low pressure turbines, is modeled as a ground-level release.

ARCON96 was executed for each of the two release points with respect to the TSC intake. Additional PAVAN runs were executed for the stack to TSC intake scenario in accordance with Reference 13 guidance to determine the distance at which the actual maximum X/Q would occur in each given downwind sector similar to that done for the CR/AEER.

3.3.2.5 EAB and LPZ

The PAVAN model was also executed to determine the X/Q for a stack and Turbine Building release to the EAB and LPZ.

The stack was modeled as an elevated release and the Turbine Building release as a ground-level release. An EAB distance of 509 m (i.e., the shortest distance between the stack and EAB) was used for the elevated release scenarios. For all ground-level release scenarios, the worst-case EAB distance of 423 m (i.e., the shortest distance between the Turbine Building and EAB) was conservatively used. An LPZ distance of 6400 m was used for both the elevated and ground-level release scenarios.

In order to conservatively account for isolated areas of terrain higher than the plant grade, terrain elevations of 17 m (i.e., 55.8 ft) above plant grade were used for the SSW through NW sectors for all distances 1600 m and greater. Elsewhere, plant grade receptor elevation was assumed. No terrain elevations were used for the ground-level release.

3.4 NUREG-0737

EGC has determined that continued compliance will be maintained with NUREG-0737 (i.e., Reference 14), 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."

A review of the applicability of the revised AST to the current TS Bases and various commitments in accordance with NUREG-0737 was completed. NUREG-0737, Item II.B.3, "Post-Accident Sampling Capability," and Item II.F.1, "Accident Monitoring Instrumentation," will not be affected as a result of AST implementation.

NUREG-0737, Item III.D.1.1, "Leakage Control," will continue to be controlled, tested and measured in accordance with the local leak rate test program. No changes will occur as a result of AST implementation.

NUREG-0737, Item III.A.1.2, "Emergency Response Facilities," which includes the design of the TSC, has been analyzed for AST applicability. The TSC is not affected by AST. 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 would remain acceptable.

3.5 Environmental Qualification (EQ)

Regulatory Position 6 of Regulatory Guide 1.183 (i.e., Reference 2) states: "The NRC staff is assessing the effect of increased cesium releases on EQ doses to determine whether licensee action is warranted. Until such time as this generic issue is resolved, licensees may use either the AST or the TID-14844 assumptions for performing the required EQ analyses. However, no plant modifications are required to address the impact of the difference in source term characteristics (i.e., AST vs. TID-14844) on EQ doses."

Accordingly, LSCS will continue to use TID-14844 as the radiation dose basis for EQ.

Qualification of safety related equipment from the radiation environment resulting from a DBA LOCA will continue to be based on the original TID-14844 based accident treatment resulting from a DBA. This practice is recognized as acceptable because of the minimal public health and safety benefit and substantial cost of re-evaluation of radiation environment characterization with AST based assumptions of core releases and timing. The changes in plant parameters in this calculation do not impact conclusions reached or in the general underlying parameters related to primary containment sources, secondary containment airborne sources, and ECCS piping sources.

3.6 LOCA

Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 defines LOCAs as those postulated accidents that result from a loss of coolant inventory at rates that exceed the capability of the reactor coolant makeup system. Leaks up to a double-ended rupture of the largest pipe of the RCS are included. The LOCA is a conservative surrogate accident that is intended to challenge selective aspects of the facility design. Analyses are performed using a spectrum of break sizes to evaluate fuel and ECCS performance. With regard to radiological consequences, a large-break LOCA is assumed as the design basis case for evaluating the performance of release mitigation systems and the containment and for evaluating the proposed siting of a facility.

The LSCS LOCA was analyzed using a conservative set of assumptions and as-built design input parameters compatible for AST and the TEDE dose criteria. The numeric values of the critical design inputs were conservatively selected to assure an appropriate prudent safety margin against unpredicted events in the course of an accident and compensate for large uncertainties in facility parameters, accident progression, radioactive material transport, and atmospheric dispersion.

The design inputs used for the design analyses were extracted from LSCS licensing basis documents, Updated Final Safety Analysis Report (UFSAR) sections, existing calculations, design basis documents, and regulatory guidance documents. Key parameters used in the LOCA analysis are summarized in Table 3.6-1. References to figures are those figures contained within the LOCA calculation provided in Attachment 7.

TABLE 3.6-1 LOCA AST Analysis Parameter or Method			
Parameter	**Pre-AST Value**	**AST Value**	**Comments**
Primary containment volume - Drywell free volume - Wetwell airspace volume	 229,538 ft^3 164,800 ft^3	 229,538 ft^3 164,800 ft^3	No change.
Suppression pool water volume (pre-LOCA)	128,800 ft^3 (minimum) 131,900 ft^3 (maximum)	128,800 ft^3 (minimum) 131,900 ft^3 (maximum)	No change. The suppression pool volume ranges between 128,800 ft^3 at the low water level limit (i.e., -4.5 inches) and 131,900 ft^3 at the high water level limit (i.e., 3 inches).
Primary containment leak rate	0.635% per day	1.0% per day	New assumption justified in AST analysis. Value was increased for conservatism.
Secondary containment volume (i.e., Reactor Building, including the equipment access structure and a portion of the main steam tunnel)	2,875,000 ft^3	2,875,000 ft^3	No change.
Secondary containment bypass	None except for MSIV leakage	None except for MSIV leakage	No change.
SGT system flow rate	4,000 cfm (each train) ± 10%	4,000 cfm (each train) ± 10%	No change. Ventilation Filter Testing Program (VFTP) in TS 5.5.8 indicates the flow rate of $\geq$ 3600 cfm and $\leq$ 4400 cfm. The bounding flow rate of 3600 cfm was used.
SGT system filter efficiency	99% after drawdown	99% after drawdown	No change.
Reactor Building drawdown time	5 minutes	15 minutes	New assumption justified in AST analysis. A longer, more conservative drawdown time was used.

TABLE 3.6-1 LOCA AST Analysis Parameter or Method			
Parameter	**Pre-AST Value**	**AST Value**	**Comments**
MSIV leakage rates	400 scfh total 100 scfh single line	400 scfh total 200 scfh single line	New maximum single line leakage assumption justified in AST analysis. Current design basis assumes steam lines and main condenser would remain intact under post-accident conditions. Information supporting the seismic design of the LSCS steam piping and main condenser was submitted to the NRC in Reference 15. The NRC approved the seismic design in Reference 16.
ECCS leakage rate into secondary containment	10 gallons per hour (i.e., 2 times the administrative control level)	5 gallons per minute (i.e., 2 times the administrative control level)	New assumption justified in AST dose analysis. Value was increased for conservatism in AST calculations.
Emergency makeup filter unit flow	4000 ± 10% cfm	4000 ± 10% cfm	No change in nominal values. The ± 10% range is from TS 5.5.8.a. The bounding dose results have been determined to occur with the -10% value. This flow is split between the CR and the AEER. For pre-AST operations, 37.5% was directed to the CR, and the balance to the AEER. For AST, a range of fractions to the CR of 25%, 37.5%, and 50% were analyzed as shown in Figures 1, 2, and 3, with worst-case doses used.

TABLE 3.6-1 LOCA AST Analysis Parameter or Method			
Parameter	**Pre-AST Value**	**AST Value**	**Comments**
CR recirculation filter flow	18,000 cfm	18,000 cfm	This value is the TS 5.5.8.b minimum flow, which is bounding for dose purposes. This is the sum of makeup flow, inleakage flow, and actual control room air recirculation flow. CR exfiltration will be equal to the total of makeup and inleakage. For dose assessment purposes, CR filtered recirculation is the CR recirculation filter flow, less exfiltration. See Figures 1, 2, and 3.
CR recirculation filter bypass (B, as identified in Figures 1, 2, and 3)	900 cfm	900 cfm	No change in nominal values.
CR outside air unfiltered inleakage after makeup filter (g1, as identified in Figures 1, 2, and 3)	55.2 cfm	55.2 cfm	No change in nominal values.
CR outside air unfiltered inleakage into low pressure ductwork before recirculation filter (g2, as identified in Figures 1, 2, and 3)	1,200 cfm	2,400 cfm	New assumption justified in AST analysis. Value was increased for conservatism.
CR outside air unfiltered inleakage rate after recirculation filter (g3, as identified in Figures 1, 2, and 3)	7 cfm	50 cfm	New assumption justified in AST analysis. Value was increased for conservatism.

TABLE 3.6-1 LOCA AST Analysis Parameter or Method			
Parameter	**Pre-AST Value**	**AST Value**	**Comments**
CR intake filter charcoal efficiency (E1, as identified in Figures 1, 2, and 3)	95%	90%	New assumption justified in AST analysis. Values were minimized for the AST analyses to provide additional margin.
CR recirculation filter charcoal filter efficiency (E2, as identified in Figures 1, 2, and 3)	70%	70%	No change.
CR HVAC system activation times after LOCA signal - makeup filter - recirculation filter	 t = 20 minutes t = 4 hours	 t = 20 minutes t = 4 hours	No change.
CR occupancy requirements	0-24 hrs: 1.0 1-4 days: 0.6 4-30 days: 0.4	0-24 hrs: 1.0 1-4 days: 0.6 4-30 days: 0.4	No change.

TABLE 3.6-1 LOCA AST Analysis Parameter or Method			
Parameter	**Pre-AST Value**	**AST Value**	**Comments**
AEER occupancy	The LSCS CRE has historically been treated as consisting of the CR and AEER, with a shared filtered emergency makeup system and separate filtered recirculation systems. The AEER occupancy was treated as the same as the CR.	Only mission occupancy is required to start the Hydrogen Recombiner system fan for containment mixing. Worst-case time was assumed, including drawdown time when SGT system filtration is not credited. Conservative total mission time is 30 minutes. This mission was assumed to be performed by an operator not assigned full-time to the CR.	New assumption justified in AST analysis. LSCS Operations training performed a time validation. The individual who performed the actions is a non-licensed operator (NLO) instructor with recent NLO experience. The following times were recorded: - Travel time to the AEER from the CR is three minutes. - Time to perform required actions is three minutes (i.e., one minute for local actions in the AEER and two minutes for CR actions). - Travel time to return to the CR is three minutes. - The time for the entire mission is nine minutes.
AEER outside air unfiltered inleakage	1,400 cfm	100,000 cfm	New assumption justified in AST analysis. Value was increased for conservatism.
AEER filtration system consideration	AEER filter system is similar to the CR filter system.	No credit for protection by the makeup filter or the AEER recirculation filter.	While filters are not credited, parameters for this system are shown in Figures 4, 5, and 6, with the makeup filter flow splits discussed for the CR.
AEER volume	74,800 ft^3	68,800 ft^3	AST value based on updated volume calculation.

EAB, LPZ, CR, and TSC doses for LSCS were calculated using the guidance in Regulatory Guide 1.183 (i.e., Reference 2), and the TEDE dose criteria. In addition to direct shine to control room operators, the DBA LOCA calculation was performed for the following post-LOCA release paths:

- Primary containment leakage,
- ECCS leakage, and
- MSIV leakage.

In general, credit is taken only for those active accident mitigation features that are classified as safety related, are required to be operable by TS, are powered by emergency power sources, and are automatically actuated. Exceptions are the following.

- The CR emergency ventilation system is designed to automatically initiate; however, the LOCA analysis assumes manual action timing to address single failures.
- The alignment of an MSIV drain line to direct MSIV leakage to the condenser is manually initiated.
- The seismically rugged portions of steam piping and the condenser are not classified as safety related.
- The SLC system is credited for suppression pool pH control. The SLC system is manually initiated. Additional information regarding the SLC system is provided in Section 3.6.11.

The numeric values that are chosen as inputs to analyses required by 10 CFR 50.67 are compatible to AST and TEDE dose criteria and selected with the objective of maximizing the postulated dose. The use of a 10% lower makeup flow rate for the CR and a minimum CR recirculation flow rate, and use of worst-case ground release X/Q values, demonstrate the inherent conservatisms in the plant design and post-accident response analysis.

3.6.1 Assumptions on Transport in the Primary Containment

For LSCS, the radioactivity release from the reactor is assumed to mix instantaneously and homogeneously throughout the drywell. No mixing between the drywell and the wetwell is assumed for the first two hours. This is based on an assumption that the initial blowdown occurs before fuel damage commences, and that AST source terms are based on a non-mechanistic loss of ECCS flow to the reactor for two hours. After ECCS flow restoration, the rapid steaming of ECCS liquids are assumed to quickly displace significant fractions of the airborne activity in the drywell through downcomers into the suppression chamber, providing the mixing mechanism. Conservatively, no credit is taken for suppression pool scrubbing during this flow. Therefore, after two hours, complete mixing of activity in the drywell volume to the suppression chamber airspace is assumed. The RADTRAD containment compartment volume parameter and MSIV leakage flow rates implement this treatment.

With the exception of noble gases, all fission products released from the fuel to the containment are also assumed to instantaneously and homogeneously mix in the suppression pool at the time of release. RADTRAD models for ECCS leakage treat the suppression pool water as the compartment to which core activity is released.

Radioactivity in containment is reduced only by natural deposition, decay, and leakage. For LSCS, the RADTRAD computer program, including the Powers Natural Deposition algorithm based on NUREG/CR-6189 (i.e., Reference 17), is used for modeling aerosol deposition in primary containment. No natural deposition is assumed for elemental or organic iodine. The lower bound (i.e., 10%) level of deposition credit is used. Suppression pool scrubbing is not credited. Neither drywell nor wetwell spray is credited as a removal mechanism. Analyses demonstrate that suppression pool pH is maintained greater than seven, so iodine re-evolution is not assumed.

Decay of radioactivity is credited in the drywell prior to release. This is implemented in RADTRAD using the half-lives in the "nif" files. RADTRAD's decay plus daughter option is used.

Leakage from the primary containment is postulated to be released directly to the environment without mixing in the Reactor Building free air volume.

3.6.2 Post-LOCA Containment Leakage

Primary containment leakage is assumed to be controlled to an L_a rate of 1.0% per day, with no reduction after the first 24 hours.

The entire leakage is treated as being to the secondary containment. The exhaust from secondary containment is filtered through the SGT system filter train, following a 15-minute drawdown period with the filtration not credited. After drawdown, SGT system high efficiency particulate air (HEPA) and charcoal filters are available to reduce the released activity.

Other than leakage through the MSIVs, there are no other leakage pathways that bypass secondary containment at LSCS. Because of the use of the MSIV – Isolated Condenser Leakage Treatment Method (MSIV-ICLTM), MSIV leakage bypasses secondary containment and is released through the seismically rugged Turbine Condenser system, as discussed below.

3.6.3 Containment Leakage Source Term

The BWR core inventory fractions listed in Table 1 of Regulatory Guide 1.183 (i.e., Reference 2) are released into the containment at the release timing shown in Table 4 of Regulatory Guide 1.183. Since the post-LOCA minimum suppression pool water pH is greater than 7.0 for the duration of the accident, the chemical form of radioiodine released into the containment is assumed to be 95% CsI, 4.85% elemental iodine, and 0.15% organic iodide. With the exception of elemental and organic iodine and noble gases, the remaining fission products are assumed to be in particulate form. The plant-specific isotopic fission product core activities, in units of curies, were calculated and converted into Ci/MWt using the core thermal power level.

3.6.4 Containment Purging

Purging of containment is not a routine activity at LSCS. TS SR 3.6.1.3.1 identifies purposes for containment purging at LSCS as inerting, de-inerting, pressure control, ALARA or air quality considerations for personnel entry, and surveillances that require valves to be open. TS 3.6.3.2 provides limitations on use for inerting and deinerting at power.

3.6.5 Post-LOCA ECCS Leakage

The ECCS fluid systems that recirculate suppression pool water outside of the primary containment are assumed to leak during their intended operation. This release source includes leakage through valve packing glands, pump shaft seals, flanged connections, and other similar components. The radiological consequences from the postulated leakage are analyzed and combined with the radiological consequences from other fission product release paths to determine the total calculated radiological consequences from the LOCA. ECCS components are located in the Reactor Building.

3.6.6 ECCS Leakage Source Term

With the exception of noble gases, fission products released from the fuel to the containment are assumed to instantaneously and homogeneously mix in the suppression pool water at the time of release from the core. The total ECCS leakage from all components in the ECCS systems is assumed to be 5 gpm, which is assumed to start immediately after the onset of a LOCA. With the exception of iodine, remaining fission products in the recirculating liquid are assumed to be retained in the pool water. Since the post-LOCA temperature of suppression pool water recirculated through the ECCS system is less than 212°F, 10% of the iodine activity in the leaked liquid is assumed to become airborne. The reduction in ECCS leakage activity by dilution in the Reactor Building volume is not credited. The radioiodine that is postulated to be available for release to the environment due to ECCS leakage is assumed to be 97% elemental and 3% organic.

3.6.7 MSIV Leakage Release Pathway

The current MSIV leakage rate limits of 100 scfh per steam line and a total of 400 scfh for all four lines (i.e., TS SR 3.6.1.3.10) will be changed to 200 scfh for any one line and a total of 400 scfh for all four lines. These limits continue to apply at test pressures of $\geq$ 25 psig. MSIV leakage was evaluated in 1994 using NEDC-31858P-A (i.e., Reference 18) methodology. LSCS radiological effects are reanalyzed using AST, and the methodology described in NEDC-31858P-A.

Outboard MSIV failure is assumed as the single active failure since this maximizes the volume of piping in which the fluid is depressurized, minimizing deposition.

Inboard piping on one main steam line is not credited to simulate the impact of a LOCA involving a steam line break inside containment.

3.6.7.1 MSIV Leakage Source Term

The activity available for release via MSIV leakage is assumed to be that activity released into the drywell for evaluating containment leakage per Regulatory Guide 1.183 (i.e., Reference 2).

3.6.7.2 Modeling of Deposition Credit in Pipes and Condenser

LSCS has previously been analyzed and licensed to no longer credit an MSIV Leakage Control system other than the MSIV-ICLTM, and to credit seismically analyzed portions of the Turbine Condenser system. This system has previously been shown to be seismically rugged as discussed in UFSAR Section 6.8. This historical evaluation is based on methodology described in NEDC-31858P-A. That analysis was based on a design basis recirculation line break. In the AST LOCA calculation, the analysis of MSIV leakage is updated to reflect AST parameters related to release timing and chemical makeup and NRC-approved approaches regarding fission product settling and deposition, as discussed below.

3.6.7.3 Aerosol Settling

Modeling of aerosol settling is based on methodology used by the NRC in Accident Evaluation Branch (AEB)-98-03, "Assessment of Radiological Consequences for the Perry Pilot Plant Application using the Revised (NUREG-1465) Source Term," (i.e., Reference 19) with some additional conservatism based on LSCS specific parameters. For aerosol settling, only horizontal piping runs are credited, and only the horizontal projected area of horizontal piping is considered as the settling area.

This analysis implements a 20-group settling velocity distribution rather than the AEB-98-03 single, median value, model. The same settling velocity probability distribution function was applied. This is conservative because it does not consider such phenomena as thermophoresis, diffusiophoresis, flow irregularities, and hygroscopicity, which would serve to increase the rate of aerosol deposition and settling. The settling velocity distribution is a function of a randomly sampled range of the three particle parameters (i.e., density (logarithmically distributed), diameter (uniformly distributed), and shape (uniformly distributed)) and three constants (i.e., gravitational acceleration, Cunningham slip factor, and viscosity). The range of each particle parameter is referenced in AEB-98-03.

By implementing a conservative, semi-continuous, probability-weighted, 20-group step function to simulate the varied population of particulate in a given Main Steam (MS) system volume, as opposed to a single median value, this model accounts for the uneven settling of "easier to remove particles" versus "difficult to remove particles."

The analysis takes no credit for any aerosol deposition after 24 hours. This conservative aerosol deposition treatment, and the conservatisms in the AEB-98-03 conclusions, account for uncertainty in the AEB-98-03 model.

3.6.7.4 Elemental and Organic Iodine Removal

Because elemental iodine deposition is not gravity dependent, deposition is credited in both horizontal and vertical piping on all surface areas. For conservatism, no credit is taken for deposition in the drain lines that provide the previously licensed alternate drain path to the condenser. All MS drain lines are routed to the condenser at a point below the condenser tubing.

Credit is taken for deposition in the condenser, but only the deposition area of the horizontal surface of the wetwell of the high pressure condenser. The condenser tubing provides a surface area that is orders of magnitude larger than that of the credited bottom surface area. No credit is taken for any organic iodine removal in piping or the condenser.

Re-suspension of deposited elemental iodine is conservatively treated as organic iodine and immediately released. Re-suspension of iodine from steel surfaces was simulated by applying the model developed by J. E. Cline & Associates, Inc. and Science Applications International Corporation (SAIC) (i.e., Reference 20). The immediate release of re-suspended iodine, directly to the environment, in organic form is a conservative assumption due to inherent holdups in this release and tortured paths through which this activity will be transported. Therefore, this simulation conservatively models the re-suspension effects of elemental iodine.

3.6.7.5 Condenser Credit Treatment and Conservatisms

The condenser is treated as a well-mixed volume. The credited deposition area for elemental iodines includes walls and the base, which includes the wetwell. For aerosols, only the base/wetwell surfaces are credited since the removal is by gravitational settling. No organic iodine removal credit is taken.

In general, the crediting of steam line piping and the condenser results in dose contributions being dominated by noble gases and organic iodine. Even so, the treatment of aerosols and elemental iodine is conservative, for the following reasons.

1. The drain lines, which are not credited for settling or deposition, enter the condenser below the condenser tubing. Expected exhaust paths are through (1) the turbine shaft seals to the gland seal condenser exhaust (unpowered), (2) through the condenser vacuum breaker if in use prior to a Mode 3 LOCA, or (3) through shell leakage.

2. The first two paths are well above the condenser tubing. For aerosols, the direction for the first two paths requires that they "settle up" through the

condenser tubes. For elemental iodine, the neglected surface area of the condenser tubes far exceeds the credited wall and wetwell surface.

3. The condenser shell path is one of the assumed paths for non-condensable gas entry when the condenser is at vacuum. However, at loss of vacuum conditions, its out-leakage equivalent path resistance would be expected to be significantly less that the gland seal path. Therefore, general shell leakage, which could be above or below the condenser tubes, is expected to be small.

3.6.7.6 Determination of MSIV Leakage Rates in Various Main Steam Line (MSL) Volumes

The radioactivity associated with MSIV leakage is assumed to be released directly from the primary containment and into the MSLs. MSIV leakage has separate limits and a separately analyzed dose; therefore, it is not included in the L_a fraction limit and is instead separately controlled.

MSIV leakage assumed in the LOCA analysis is 400 scfh total for all MSLs and 200 scfh for any one MSL, when tested at or greater than 25 psig. The leakage rate and inboard piping flow rate associated with a 200 scfh leakage rate is adjusted for pressure and temperature differences.

Flow rates out of the condenser are similarly calculated with the assumption of a condenser air space temperature of 120°F for the accident duration. This rate applies to any condenser opening such as turbine seals, condenser shell leakage, or open vacuum breakers that may be in use under Mode 3 conditions.

Determination of inboard steam line, outboard steam line, and condenser effective filter efficiencies is determined using AEB-98-03 formulations and settling and deposition velocities.

3.6.7.7 Recirculation Line Rupture Versus MSL Rupture

10 CFR 50, Appendix A, defines LOCAs as those postulated accidents that result from a loss of coolant inventory at rates that exceed the capability of the reactor coolant makeup system. Leaks up to a double-ended rupture of the largest pipe of the RCS are included. The LOCA is a conservative surrogate accident that is intended to challenge selective aspects of the facility design. The DBA for the safety related system design is a LOCA. This LOCA leads to a specific combination of dynamic, quasi-static, and static loads in time. The thermal transients due to other postulated events, including the MSLB inside the drywell, do not impose maximum challenge to the drywell pressure boundary and fuel integrity. The LOCA results in the maximum core damage and fission product release as shown in Regulatory Guide 1.183, Table 1. Therefore, a recirculation line rupture is considered to be the limiting event with respect to radiological consequences.

Regulatory Guide 1.183, Appendix A, Section 6.5 allows reduction in MSIV releases that is due to holdup and deposition in MS piping downstream of the MSIVs and in the main condenser, including the treatment of air ejector effluent by offgas systems, 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. Although postulating a MSLB in one steam line inside the drywell would maximize the dose contribution from the MSIV leakage, the MSLB is not a credible event during a LOCA since the MS piping is designed to withstand the safe shutdown earthquake.

3.6.8 CR/AEER Model

The LSCS CR envelope has historically been treated as consisting of the CR and AEER, with a shared filtered emergency makeup system and separate filtered recirculation systems. In the AST LOCA analysis, standard continuous occupancy assumptions are applied to the CR. However, AEER occupancy is only required for the safety related action of starting the fan that provides containment air mixing as required per 10 CFR 50.44(c)(1) for combustible gas control. This mission is assumed to be performed by an operator not assigned full time to the CR, but dispatched from the CR. The total expected time for this mission outside of the CR is nine minutes. The dose analysis is based on 30 minutes. The worst-case timing for this operation would be starting at time zero because of exposure to releases during reactor enclosure drawdown. No credit is taken for any filtration provided by the makeup filter or AEER recirculation filter system. On this basis, the features that control radioactivity in the AEER, such as filtered intake, filtered recirculation, and positive pressurization are not required for this mission.

The CR and AEER share a makeup filter system, but have separate recirculation filter systems. Nominally, 37.5% of the makeup flow is directed to the CR and 62.5% is directed to the AEER. In the AST LOCA analysis, splits of 25%, 37.5%, and 50% to the CR are analyzed in this distribution with the balances directed to the AEER. The bounding values for dose analysis purposes were used to demonstrate 10 CFR 50.67 compliance.

The CR/AEER makeup filter charcoal adsorber credit is based on 90% efficiency for elemental and organic iodines, rather than the historically credited 95%. However, no changes to TS regarding filter efficiency are proposed.

Because of the presence of HEPA filtration in the makeup filter train, aerosol removal efficiency is credited at 99%. No aerosol removal is credited in the CR or AEER recirculation filter trains.

Recirculation filter bypass for the CR is assumed to be at 5% of the minimum CR supply flow. That is 900 cfm for the CR recirculation filter. Inleakage upstream of the CR recirculation filters and upstream of the supply fans are addressed separately as discussed below.

A CR recirculation filtered inleakage rate of 2400 cfm is assumed. This is 200% of the historically assumed value, and conservatively well above tracer gas testing results, to provide operational margin to be managed under the Control Room Envelope Habitability Program.

In addition to the filtered inleakage and the 5% filter bypass, another 50 cfm of unfiltered inleakage is assumed into the ductwork downstream of the CR recirculation filters and upstream of the supply fans for the CR. The allowance is based on historical estimates of maximum credible leakage, now multiplied by approximately a factor of seven.

3.6.9 Shine Doses to CR from External Sources

The pre-AST UFSAR shine doses, and supporting analyses, have been reviewed and the largest contributors re-evaluated on a conservative AST basis. These were shine from plate-out of activity on the refuel floor and control building filters. External cloud doses were also reanalyzed for possible AST effects. Attachment C of the LOCA calculation provides documentation of the review and adjustment, as necessary, of existing sources that are not reanalyzed, and the three re-analyses. Resulting external dose contributions are small.

3.6.10 Vital Area Accessibility

The LOCA analysis establishes that vital areas remain accessible. Vital areas outside of the CR are:

1. The AEER, with the associated mission dose to start fans that provide containment air mixing for post-LOCA combustible gas control purposes, and

2. The TSC, which is assumed to require occupancy equivalent to the CR.

Assessment of these analyses shows these areas to be accessible, with doses within 10 CFR 50.67 CR dose limits.

Based on evaluations in Attachment E of the LOCA calculation, the dose for occupancy of the TSC, and for the safety related mission to the AEER are within 10 CFR 50.67 CR dose limits. Existing analyses for other locations and pathways as described in UFSAR Section 12.3 were reviewed, and conservatively adjusted where merited.

3.6.11 Suppression Pool pH Control

Suppression pool pH was evaluated over the 30-day duration of the DBA LOCA and demonstrated that pH will remain above 7.0. Therefore, no iodine conversion to elemental with re-evolution is considered in the LOCA calculation. The control of pH also significantly limits the potential for airborne release from subcooled ECCS leakage inside and outside of secondary containment. Completion of the SLC system injection of its sodium pentaborate solution is required for pH control within 3.5 hours of the start of the LOCA. Injection would typically be expected sooner for an event that results in fuel damage comparable to that necessary for core radioactivity releases assumed in the DBA LOCA, both as an alternative water source, and for added subcriticality margin.

LSCS proposes to credit control of the pH in the suppression pool following a LOCA by means of injecting sodium pentaborate into the reactor core with the SLC system. The SLC system design was not previously reviewed for this safety function (i.e., pH control post-LOCA). In Reference 21, the NRC issued review guidelines for assessing the acceptability of reliance on the SLC system to control the pH of the water in a BWR suppression pool following a LOCA. Specifically, Reference 21 identifies four guidelines that the SLC system should meet. Each of the four guidelines is stated below, along with EGC's response that demonstrates the LSCS SLC system meets the guidelines.

Review Guideline 1

The SLC system should be classified as ESF grade in accordance with 10 CFR 50.34(b) or as a safety-related system as defined in 10 CFR 50.2, and satisfy the regulatory requirements for such systems.

There may be plants with an SLC system which is not classified as safety-related or as ESF grade. In such instances, the staff reviewer will determine whether the SLC system is comparable to a system classified as safety-related or ESF. A SLC system meeting items (a)-(e) below would result in its acceptance in support of a 10 CFR 50.67 request even if the system is not classified as safety-related or as ESF grade.

(a) The SLC system should be provided with standby AC power supplemented by the emergency diesel generators.

(b) The SLC system should be seismically qualified in accordance with Regulatory Guide 1.29 and Appendix A to 10 CFR Part 100.

(c) The SLC system should be incorporated into the plant's ASME Code ISI and IST Programs based upon the plant's code of record (10 CFR 50.55a).

(d) The SLC system should be incorporated into the plant's Maintenance Rule program consistent with 10 CFR 50.65.

(e) The SLC system should meet 10 CFR 50.49 and Appendix A (GDC 4) to 10 CFR 50.

EGC Response to Review Guideline 1

The LSCS SLC system is a safety related system and meets the criteria of (a)-(e) above.

Review Guideline 2

The licensee should have plant procedures for injecting the sodium pentaborate using the SLC system. This information would be reviewed by the appropriate technical review branch, as requested by the lead SPSB reviewer.

(a) A review of the procedures may be appropriate if a reliability approach is taken (4(a) below) due to timing considerations for the injection of chemicals.

(b) The SLC activation steps are placed in a safety-related plant procedure.

(c) The steps be activated by parameters that are symptoms of imminent or actual core damage.

(d) The instrumentation relied upon to provide this indication meets the quality requirements for a Type E variable as defined in RG 1.97 Tables 1 and 2.

(e) Personnel receive initial and periodic refresher training in the procedure.

(f) Other plant procedures (e.g., ERGs/SAGs) that call for termination of SLC as a reactivity control measure are appropriately revised to enable SLC injection for pH control.

<u>EGC Response to Review Guideline 2</u>

As discussed below in response to Review Guideline 4, the LSCS SLC system cannot be considered redundant with respect to its active components. Therefore, EGC proposes to demonstrate that this lack of redundancy is offset by satisfying Review Guideline 4(a). Consistent with Review Guideline 2(a), the following information is provided to describe the LSCS procedures for injecting sodium pentaborate using the SLC system.

The LSCS SLC system activation steps are in a safety related plant procedure (i.e., Emergency Operating Procedure (EOP) LGA-001, "RPV Control"). LGA-001 will be revised to ensure that SLC system injection is started from the boron solution storage tank during a DBA LOCA. In addition, LGA-001 will be revised to ensure no steps would terminate the injection during a DBA LOCA prior to emptying the SLC storage tank (i.e., injection of the full content into the RPV). This ensures complete injection upon a LOCA signal.

The steps that require activation of the SLC system are based upon symptoms of imminent or actual core damage. When RPV water level drops below -150", as read on the wide range level instruments, operator action will be to initiate SLC system injection from the SLC solution tank. This is indicative of a LOCA and that core uncovery is imminent and is symptomatic of core damage potential.

The instruments used to provide this indication are the Wide Range level instruments, which are listed in LSCS TS 3.3.3.1, "Post Accident Monitoring Instrumentation." These instruments are classified as Type A variable components as defined by Regulatory Guide 1.97 Table 1. The post accident monitoring (PAM) instrumentation LCO ensures the operability of Regulatory Guide 1.97 (i.e., Reference 22), Type A, variables so that the CR staff can: (1) perform the diagnosis specified in the EOPs; and (2) take the specified, preplanned, manually controlled actions for which no automatic control is provided, which are required for safety systems to accomplish their safety function.

Licensed Operators receive initial and periodic refresher training on the SLC system, and consequently, the steps that direct initiation of SLC.

There are three reasons for termination of the SLC system in the LSCS EOPs (i.e., LGAs). The first reason is in LGA-010, "Failure to Scram," which terminates the SLC system when all rods are inserted. However, this is an anticipated transient without scram (ATWS) mitigation procedure. The SLC system would be initiated in LGA-010 only if power is above 3% in an ATWS. This condition does not pertain to a DBA LOCA. Thus, there is no need to revise LGA-010 to remove the termination criteria.

The second reason is in LGA-001, "RPV Control." In the event of a small LOCA where level can be recovered, the SLC system, if initiated, would be shutdown. However, in this condition there are no symptoms of imminent or actual core damage. Thus there is no need to revise this guidance. The SLC system would not be needed for pH control in the suppression pool in these conditions.

Thirdly, if a large LOCA (i.e., a full DBA LOCA) were in progress, the SLC system would be used per LGA-001 both for level control and for pH control in the suppression pool. The only termination criterion for the SLC system, as related to LGA-001, is that the tank is empty.

There is also guidance in procedure LSAMG-101/201 to initiate the SLC system if it is not already running. This is done to ensure that the reactor will stay shutdown even if the control rod drives melt out of the core. In LSAMG-101/201, there are no SLC system termination criteria. It would be shutdown only when the SLC storage tank is empty. The LSAMGs would not be entered unless an event more severe than a DBA LOCA has occurred. During a DBA LOCA, EOP LGA-001 would be used.

The guidance currently in LGA-001 allows SLC injection from the boron solution storage tank or the test tank. Injection of SLC using the test tank for pH control is not appropriate and thus LGA-001 requires revision to ensure that the SLC system will be initiated from the boron solution storage tank when symptoms of imminent or actual core damage are present. In addition, the guidance currently in LGA-001 is silent on when to secure SLC. To ensure that the SLC system will not be shutdown in these conditions prior to emptying the SLC boron solution storage tank (i.e., injection of the full content into the RPV), a revision to LGA-001 is needed for AST.

Review Guideline 3

A sufficient concentration and quantity of sodium pentaborate should be available for injection into the reactor vessel to control pH in the suppression pool.

The source term analysis is tied to the plant's design basis accident, which is the large break LOCA, a break of a recirculation pipe. The licensee needs to demonstrate that within 24 hours there is adequate recirculation between the suppression pool and the reactor vessel through flow out the break to provide transport and mixing, consistent with the assumptions in the chemical analyses.

EGC Response to Review Guideline 3

Attachment 8 provides calculation L-003064, "Suppression Pool pH Calculation for Alternative Source Terms." This calculation provides the assumptions, inputs, methods, and results that demonstrate a sufficient concentration and quantity of sodium pentaborate is available for injection into the reactor vessel to control pH in the suppression pool. Section 4.5 of the calculation discusses the adequacy of recirculation between the suppression pool and the reactor vessel through flow out the break to provide transport and mixing.

Review Guideline 4

The SLC system should not be rendered incapable of performing its AST function due to a single failure of an active component. For this purpose the check valve is considered an active device for AST since the check valve must open to inject sodium pentaborate for suppression pool pH control.

If the SLC system can not be considered redundant with respect to its active components, this lack of redundancy may be offset if the licensee can satisfy (a) or (b) or (c) below:

(a) Acceptable quality and reliability of the non-redundant active components and/or compensatory actions in the event of failure of the non-redundant active components.

Under this approach, the licensee should provide the following information in justifying the lack of redundancy of active components in the SLC system:

(1) The licensee should identify the non-redundant active components in the SLC system and provide their make, manufacturer, and model number. The staff reviewer will compare this information with performance data for the component from industry data bases and other sources.

(2) The licensee should provide the design-basis conditions for the component and the environmental and seismic conditions under which the component may be required to operate during a design-basis accident. Environmental conditions include design-basis pressure, temperature, relative humidity and radiation fields. The staff reviewer will compare the environmental and seismic conditions associated with the design-basis accident to the conditions for which the component was designed to determine whether the component is capable of performing its intended function.

(3) The licensee should indicate whether the component was purchased in accordance with Appendix B to 10 CFR Part 50. If the component was not purchased in accordance with Appendix B, the licensee should provide information on the quality standards under which it was purchased. For the latter situation, information on the component would be reviewed by the appropriate

technical review branch responsible for the component, as requested by the lead SPSB reviewer.

(4) The licensee should provide the performance history of the component both at the licensee's facility and in industry databases such as EPIX and NPRDS. The staff reviewer will use this information to evaluate the reliability of the component relative to other components used in safety-related applications.

(5) The licensee should provide a description of its inspection and testing program including standards, frequency, and acceptance criteria. The staff reviewer will use this information to evaluate the licensee's activities to monitor the component's performance at the facility. The information on the component would be reviewed by the appropriate technical review branch responsible for the component, as requested by the lead SPSB reviewer.

(6) The licensee should also indicate potential compensating actions that could be taken within an acceptable time period to address the failure of the component. An example of a compensating action might be the ability to jumper a switch in the control room to overcome its failure. The staff reviewer will consider the availability of compensating actions and the likelihood of successful injection of the sodium pentaborate where non-redundant active components fail to perform their intended functions.

(b) An alternative success path for injecting chemicals into the suppression pool.

If the licensee chooses to address the SLC system's susceptibility to single failure by selecting an alternative injection path, the alternative path must be capable of performing the AST function noted above and all components which make up the alternative path should meet the same quality characteristics required of the SLC system (described in Items 1(a)-1(e), 2 and 3 above). When the staff determines that an alternative path is acceptable, the staff's safety evaluation should address the manner in which the SLC system and the alternative path met Items 1(a)-1(e), 2 and 3 above.

If the use of an alternate path is part of the EOPs, then the license amendment needs to address the following items: (1) Does the alternate injection path require actions in areas outside the control room? (2) How accessible will these areas be? (3) What additional personnel will be required?

(c) 10 CFR 50.67 and Appendix A, General Design Criterion (GDC) 19 doses are met even if pH is not controlled.

The licensee may demonstrate, through dose calculations, that 10 CFR 50.67 and GDC 19 doses are met even if pH is not controlled. The re-evolution of iodine in the particulate form from the water in the suppression pool to the elemental form for airborne iodine must be incorporated into the calculation. The calculation may take credit for the mitigating capabilities of other equipment, for example the standby gas treatment system (SGTS), if such equipment would be available. The staff will perform calculations to confirm the licensee's conclusions. If the acceptability of the facility's dose calculations was based on the utilization of certain ESF equipment, for example the SGTS, then the staff's safety evaluation should reflect this. Such a citation is necessary to assure that it is recognized and documented that there is a link between the particular ESF component's performance and the SLC system's susceptibility to single failure.

EGC Response to Review Guideline 4

The LSCS SLC system cannot be considered redundant with respect to its active components. In accordance with Review Guideline 4(a) above, the following information is provided to demonstrate that this lack of redundancy is offset such that the SLC system can be credited for post-LOCA pH control. Specifically, items (1) through (6) of Review Guideline 4(a) are addressed for the non-redundant active components in the SLC system to demonstrate acceptable quality and reliability and/or compensatory actions in the event of failure.

A. SLC System Discharge Header to RPV Outboard Check Valves – 1(2)C41-F006

(1) Manufacturer: Rockwell/Edward
Model Number: 1-1/2-3674F316T(1)

(2) Worst case accident conditions = 145 °F
Maximum accident pressure = 15 psia
Relative humidity = 100%
100 day LOCA dose = 1.0 x 10^7 rads
Seismic condition = maximum credible earthquake

(3) The components were purchased in accordance with 10 CFR 50, Appendix B.

(4) There was one LSCS Unit 1 local leak rate test (LLRT) failure that required seat refurbishment due to leakage. No failures have occurred at LSCS in the forward direction of the check valve. No valve failures for other reasons have occurred for these Unit 1 and 2 check valves.

A search of industry databases identified LLRT failures, similar to the LSCS Unit 1 LLRT failure, for the containment check valves. This type of failure does not impact the injection capability of the SLC system. No issues associated with the valves failing to open were identified.

In summary, EGC has determined that the 1(2)C41-F006 check valves have an acceptable performance history at LSCS.

(5) LSCS SR 3.1.7.8 requires verification of flow through one SLC subsystem from the pump into the RPV. The Frequency of SR 3.1.7.8 is 24 months on a staggered test basis. EGC's procedure that implements this SR

requires confirmation of flow that is $\geq$ 41.2 gpm in the forward direction of the check valve.

(6) In the unlikely event that a SLC system injection path check valve fails to open, there are means of injecting sodium pentaborate using the RWCU system. Sodium pentaborate injection via the RWCU system is currently used for other events, such as ATWS. Although the RWCU system could potentially be available for use, the AST analysis for LSCS does not credit this alternative method for pH control. Given the reliability of the non-redundant check valves of the SLC system, EGC concluded that compensating actions are not warranted.

B. SLC System Discharge Header to RPV Inboard Check Valves – 1(2)C41-F007

(1) Manufacturer: Rockwell/Edward
Model Number: 1-1/2-3674F316T(1)

(2) Maximum accident temperature = 340 °F
Maximum accident pressure = 60 psia
Relative humidity = 100%
100 day LOCA dose = 2.0 x 10^8 rads
Seismic condition = maximum credible earthquake

(3) The components were purchased in accordance with 10 CFR 50, Appendix B.

(4) There have been no LLRT failures at LSCS for inboard containment isolation purposes for these check valves. No valve failures for other reasons have occurred for these Unit 1 or 2 check valves.

A search of industry databases identified LLRT failures for the containment check valves. This type of failure does not impact the injection capability of the SLC system. No issues associated with the valves failing to open were identified.

In summary, EGC has determined that the 1(2)C41-F007 check valves have an acceptable performance history at LSCS.

(5) LSCS SR 3.1.7.8 requires verification of flow through one SLC subsystem from the pump into the RPV. The Frequency of SR 3.1.7.8 is 24 months on a staggered test basis. EGC's procedure that implements this SR requires confirmation of flow that is $\geq$ 41.2 gpm in the forward direction of the check valve.

(6) In the unlikely event that a SLC system injection path check valve fails to open, there are means of injecting sodium pentaborate using the RWCU system. Sodium pentaborate injection via the RWCU system is currently used for other events, such as ATWS. Although the RWCU system could

potentially be available for use, the AST analysis for LSCS does not credit this alternative method for pH control. Compensating actions are not warranted due to the reliability of the non-redundant check valves of the SLC system.

3.6.12 LOCA Analysis Results

Table 3.6-2 below summarizes the calculated doses and related acceptance criteria for the EAB, LPZ, and CR. All results are within regulatory limits. For the TSC and other areas requiring plant personnel access, assessments contained in the LOCA analysis indicate that radiation exposures would be within regulatory limits, without credit for installed TSC filtration systems, and with no new operator actions required.

Table 3.6-2
LOCA Dose Summary

Location			LOCA Dose Contributor
EAB (REM TEDE)	LPZ (REM TEDE)	CR (REM TEDE)	
2.10	0.21	1.76	Filtered primary containment leakage unfiltered for 15 minutes and SGT system filtered thereafter (i.e., 100% of L_a)
0.05	0.04	2.33	MSIV leakage
0.09	0.01	0.10	ECCS leakage in secondary containment
N/A	N/A	0.04	Gamma shine to CR general area
2.24	**0.26**	**4.23**	**Total Calculated Value**
25	**25**	**5**	**Regulatory Limits**

3.7 FHA

The FHA evaluation applies AST methodology to the analysis of the design basis FHA for LSCS. Dose consequences are calculated at the EAB, LPZ, and CR. This evaluation also determines the safety features required to assure that regulatory limits in 10 CFR 50.67 are met, and is performed using guidance provided in Regulatory Guide 1.183 (i.e., Reference 2).

The FHA calculation, provided in Attachment 9, evaluates the movement of fuel that has decayed a minimum of 24 hours since it occupied part of a critical reactor core such that certain available safety features are not required to maintain consequences within acceptance criteria. As a result, the analysis supports changes to the LSCS TS regarding the operability of the SGT system, secondary containment, and other systems previously required to mitigate the radiological consequences of an FHA. The NRC has generically approved changes to the standard TS based on this approach in TSTF-51 (i.e., Reference 5).

Guidance in TSTF-51 suggests that a "recently irradiated fuel" parameter be developed to identify the point in time after shutdown when certain secondary containment integrity features are not required for movement of irradiated fuel. The FHA calculation assumes that recently irradiated fuel is that which requires one or more of the following features.

1. Secondary containment integrity to assure that releases are through the plant ventilation stack, which is located on the Auxiliary Building roof and serves as a single point of release for the Reactor Building, Turbine Building, and Radwaste Building ventilation as well as off-gas, SGT, and plant gland seal exhaust system.

2. The SGT system charcoal adsorber for secondary containment release treatment.

3. The CRAF Makeup subsystem charcoal adsorbers for treated CR pressurization flow as well as the Recirculation Filter subsystem with charcoal adsorbers for airborne radioactivity removal.

TS require these systems to be operable for operating unit(s) for response to other DBAs. The principal benefit of FHA analyses is that immediate suspension of movement of irradiated fuel assemblies in the secondary containment would not be required if the fuel being moved, or potentially struck, has sufficient decay and LCOs for certain secondary containment integrity features were not met.

Based on the discussion in LSCS UFSAR Sections 15.7.4.5 and 15.7.4.5.2.1, movement of irradiated fuel will not occur less than 24 hours after the associated reactor fuel has occupied a critical reactor core; therefore, a 24-hour delay period is used as the analyzed condition. The 24-hour decay time allows time to depressurize the reactor, remove the reactor vessel head, and remove the reactor internals above the core. However, it is not expected that these operations could currently be accomplished in less than 24 hours.

3.7.1 Method Of Analysis And Acceptance Criteria

Analyses of radiological consequences resulting from a design basis FHA were performed using the guidance for application of AST in Regulatory Guide 1.183. Analyses of radiation transport and dose assessment are performed using RADTRAD version 3.03.

3.7.2 Fuel Source Term Model

The fuel source term is based on the reactor core source terms and are the same as used for the LOCA analysis. These source terms are bounding for LSCS fuel cycle designs.

The fraction of the core fuel damaged is based on the GESTAR II limiting case of damaging 172 fuel pins. This is based on a "Heavy Mast" design (i.e., the "NF500 mast" in Reference 23). The GESTAR II analysis was completed using GE12 and GE14 10x10 fuel bundle arrays with the equivalent of 87.33 pins per bundle, and with all of the damaged fuel assumed to have a limiting peaking factor of 1.7. This analysis is for an assembly and mast drop from a 34 ft maximum height from the refueling platform over the reactor well onto the reactor core, and is bounding in terms of fuel damage potential.

Based on fuel damage assessments, this bounds all currently used and historical fuel types.

Fuel bundle peak burnup will not exceed the Regulatory Guide 1.183, Footnotes 10 and 11, limit of 62 GWD/MTU. For fuel exceeding a 54 GWD/MTU burnup, the maximum linear heat generation rate will not exceed the Regulatory Guide 1.183, Footnote 11, limit of 6.3 KW/ft rod average power.

3.7.3 Gap Activity

This calculation is applicable to fuel whose burnup and power limits are bounded by those specified in Regulatory Guide 1.183, Footnote 11. This allows application of the gap activity fractions listed in Regulatory Guide 1.183, Table 3.

3.7.4 Pool Decontamination Factor (DF)

The worst-case water coverage and fuel damage for FHAs over the reactor well and the spent fuel pool were evaluated. The drop over the reactor well was determined to be more limiting due to the greater number of fuel rods damaged for the reactor well drop, and the fact that the lower iodine DF for a drop over the spent fuel pool is not significant enough to overcome the fuel damage difference.

3.7.5 Release Model

The compartments are the Reactor Building air space, the environment, and the CR. The Reactor Building exhaust rate is set artificially high at 0.1 air changes per minute to assure an essentially complete release within 2 hours.

A walkdown and related drawing review identified that there are no pathways for an FHA release from the spent fuel pool to outside the Reactor Building that could be provided by a single open door or unlocked hatch (i.e., all accesses have double door passage or security lock). For any relaxation of secondary containment integrity requirements during fuel handling, controls will be developed to ensure that no such pathway could be created by the opening of two such doors in series, by the opening of a locked hatchway, or by any intentional breach of the secondary containment. However, the Reactor Building truck bay doors have been analyzed with both doors open. The ILRT penetrations have also been analyzed.

The wall and roof surfaces above the Reactor Building refueling floor are made of sheet metal and could potentially provide an FHA leak pathway via seams and interfaces with the concrete. Due to the higher potential for leakage, compared with that of thick concrete walls, the worst-case atmospheric dispersion factor to the CR calculated for this possibility is a diffuse area source. This diffuse area source is from the wall of the Reactor Building above the refueling floor facing the closest CR air intakes as used in the dose calculation.

3.7.6 FHA Analysis Inputs

The design inputs used for the FHA analysis were extracted from LSCS licensing basis documents, UFSAR sections, existing calculations, design basis documents, and regulatory guidance documents. Key parameters used in the FHA analysis are summarized in Table 3.7-1.

TABLE 3.7-1 FHA AST Analysis Parameter or Method			
Parameter	**Pre-AST Value**	**AST Value**	**Comments**
Core Power Level	3559 MWt	3559 MWt	No change. This value corresponds to the DBA power level and equals 102% of the uprated thermal power of 3489 MWt.
Fuel assembly configuration and properties	10x10 in a 87.33 fuel pin bundle and 172 pins damaged	10x10 in a 87.33 fuel pin bundle and 172 pins damaged	No change.
Radial Peaking Factor	1.7	1.7	No change.
Allowable fuel burnup and non-LOCA gap fractions	RG 1.25	Table 3 of RG 1.183. Fuel burnup will not exceed 62 GWD/MTU. Linear heat generation rate (LHGR) for fuel >54 GWD/MTU will not exceed 6.3 KW/ft.	New assumption from RG 1.183 and justified in AST design analysis.
FHA radionuclide inventory	RG 1.25	From Attachment A of AST design FHA analysis for the 60 isotopes forming the standard RADTRAD library, with decay to 24 hours.	New assumption justified in AST analysis.

TABLE 3.7-1 FHA AST Analysis Parameter or Method			
Parameter	**Pre-AST Value**	**AST Value**	**Comments**
Underwater Decontam-ination Factor	Noble Gases: 1 Particulate (cesium and rubidium): infinity Iodine: 100	Noble Gases: 1 Particulate (cesium and rubidium): infinity Iodine: 200 (conservative value for the limiting case of a drop over the reactor well)	New assumption from RG 1.183 and justified in AST design analysis.
Dose conversion factors	ICRP-30	Federal Guidance Reports 11 and 12	New assumption from RG 1.183 and justified in AST design analysis.
Offsite dose limit	6 REM whole body 75 REM thyroid	6.3 REM TEDE	New assumption from RG 1.183 and justified in AST design analysis.
CR dose limit	5 REM whole body or its equivalent to any part of the body (30 REM thyroid)	5 REM TEDE for the duration of the accident.	New requirement per 10 CFR 50.67 and RG 1.183.
Secondary containment automatic isolation and filtration	Credited	Not credited	New assumption justified in AST analysis.
Mitigation by CRAF system	Credited	Not credited	New assumption justified in AST analysis.
Bounding CR fresh air intake	4000 ± 10% cfm	30,000 cfm, or 14% above the purge flow rate of 26,340 cfm	New assumption justified in AST analysis.
CR volume	117,472 ft^3	117,500 ft^3	New assumption justified in AST analysis.
Reactor Building normal ventilation	SGT system with elevated release credited (normal ventilation isolated)	Artificially set at an air change rate of 0.1 per minute	New assumption justified in AST analysis.

TABLE 3.7-1
FHA AST Analysis Parameter or Method

Parameter	Pre-AST Value	AST Value	Comments
CR release point basis	Main stack (elevated) through SGT system	Metal wall faces the CR intake, so a worst case "diffuse area" source to the closest (south) CR intake is assumed with relaxed Secondary Containment requirements.	New assumption justified in AST analysis.
CR dispersion factor 0 – 2 hr value (Doses calculated through 24 hours)	N/A	1.67E-03 sec/m^3 (ground-level)	New X/Q calculated for AST and used in dose analysis.
EAB release point basis	Main stack (elevated) through SGT system. Distance to EAB = 423 meters	Plant vent stack, treated as a ground-level release with relaxed Secondary Containment requirements. Distance to EAB = 423 meters	New release point for AST justified in AST analysis. No change in EAB distance.
EAB dispersion factors	1.85E-04 sec/m^3 (elevated)	5.40E-04 sec/m^3 (ground-level)	New X/Q calculated for AST and used in dose analysis.
LPZ release point basis	Main stack (elevated) through SGT system. Distance to LPZ = 6400 meters	Plant vent stack, treated as a ground-level release with relaxed Secondary Containment requirements. Distance to LPZ = 6400 meters	New release point for AST justified in AST analysis. No change in LPZ distance.

TABLE 3.7-1 FHA AST Analysis Parameter or Method			
Parameter	**Pre-AST Value**	**AST Value**	**Comments**
LPZ dispersion factors	2.3E-05 sec/m^3 (elevated)	2.26E-05 sec/m^3 (ground-level)	New X/Q calculated for AST and used in dose analysis.

3.7.7 Control Room Model

The CRAF system is determined to not be required for this event and is not credited. The intake rate is set at an extreme value of 30,000 cfm, which exceeds by about 14% the CR ventilation system purge flow rate. This is not an expected condition but conservatively maximizes the intake rate and the speed at which CR radioactivity concentrations approach outside conditions.

3.7.8 FHA Analysis Results

Table 3.7-2 below summarizes the bounding calculated doses and related acceptance criteria for the EAB, LPZ, and CR. All results are within Regulatory Guide 1.183 limits.

Table 3.7-2 FHA Dose Summary		
Location	**Limits (REM TEDE)**	**Dose (REM TEDE)**
EAB	6.3	1.50
LPZ	6.3	0.063
CR	5.0	3.35

3.8 NRC Regulatory Issue Summary 2006-04

The NRC issued Regulatory Issue Summary (RIS) 2006-04 (i.e., Reference 24) to update licensees on experience with implementation of ASTs in DBA radiological analyses. In the RIS, the NRC stated the expectation that licensees review the information for applicability to their facilities and consider actions, as appropriate. In particular, the information in the RIS should be used to support implementation of an AST through a license amendment request to aid in the reduction of requests for additional information.

EGC has evaluated the issues discussed in the RIS. Table 3.8-1 provides a summary of issues raised in the RIS, as well as EGC's comments to the issues in light of the license amendment request to adopt AST methodology at LSCS.

TABLE 3.8-1 NRC RIS 2006-04	
RIS Issue	**EGC Comments**
1. Level of Detail Contained in LARs	
The AST amendment request should provide justification for each individual proposed change to the TS.	Section 2.0 identifies each proposed change to the TS, and Section 3.0 provides justification for each of the changes.
The AST amendment request should identify and justify each change to the licensing basis accident analyses.	Section 2.0 and Section 3.0 identify each change to the licensing basis accident analyses. Tables 3.6-1 and 3.7-1 provide listings of parameters used in the AST analyses, and also identify whether there was a change from the pre-AST value. The justification for the changes is discussed in Section 3.0, and the supporting calculations provided in Attachments 6, 7, 8, and 9.
The AST amendment request should contain enough details (e.g., assumptions, computer analyses input and output) to allow the NRC staff to confirm the dose analyses results in independent calculations.	Sufficient detail in tabular format is provided in Section 3.0 to allow the NRC to confirm the dose analyses results in independent calculations. In addition, the AST calculations are provided in Attachments 6, 7, 8, and 9. These calculations contain computer input and output information to allow the NRC to confirm the dose analyses results in independent calculations.
Licensees should identify the most current analyses, assumptions, and TS changes in their submittal and supplements to the submittal.	The most current analyses, assumptions, and TS changes are identified throughout Attachment 1 of the license amendment request.
2. MSIV Leakage and Fission Product Deposition in Piping	
Any licensee who chooses to reference these AEB 98-03 assumptions should provide appropriate justification that the assumptions are applicable to their particular design.	This amendment request references the basic methodology used in AEB 98-03. However, it uses site-specific parameters and LSCS design considerations.
If appropriate justification is provided, the suppression pool free air volume may be included provided there is a mechanism to ensure mixing between the drywell and wetwell.	This mixing is discussed in the design basis LOCA calculation in Attachment 7.
For aerosol settling, only horizontal sections of piping should be credited.	Only horizontal sections of piping are credited for aerosol settling.

TABLE 3.8-1 NRC RIS 2006-04	
RIS Issue	**EGC Comments**
Given the large uncertainty associated with iodine behavior in piping, deposition of gaseous iodine in piping should be omitted unless appropriate justification is provided (including providing estimates of the thermal and hydraulic conditions in the piping).	Deposition of elemental iodine is credited based on justification provided in Attachment 7. No deposition of organic iodine in the MSIV leakage path is credited.
3. CR Habitability	
Use of non-ESF ventilation systems during a DBA should not be assumed unless the systems have emergency power and are part of the Ventilation Filter Testing Program in Section 5 of the TS.	No credit is taken for use of non-ESF ventilation systems during a DBA unless the operation of such a system (e.g., Reactor Building normal exhaust) results in increased dose.
Generic Letter (GL) 2003-01, "Control Room Habitability" requested licensees to confirm the ability of their facility's CR to meet applicable habitability regulatory requirements. The GL placed emphasis on licensees confirming that the most limiting unfiltered inleakage into the CRE was not greater than the value assumed in the DBA analyses.	The value used in the analyses for unfiltered inleakage into the CRE is more than the value measured using the tracer gas method.
Some AST amendment requests proposed operating schemes for the CR and other ventilation systems that affect areas adjacent to the CRE and are different from the manner of operation and performance described in the response to the GL without providing sufficient justification for the proposed changes in the operating scheme.	CR and other ventilation systems that affect areas adjacent to the CRE have the same operation and performance as described in the response to the GL.
4. Atmospheric Dispersion	
Licensees have the option to adopt the generally less conservative (more realistic) updated NRC staff guidance on determining X/Q values in support of design basis CR radiological habitability assessments provided in RG 1.194.	CR X/Q values for releases were calculated using the computer code ARCON96 (supplemented with using PAVAN) and the methods of RG 1.194.
Regulatory positions on X/Q values for offsite (i.e., EAB and LPZ) accident radiological consequence assessments are provided in RG 1.145.	The X/Q values for offsite locations were evaluated using PAVAN and the methods of RG 1.145.

TABLE 3.8-1 NRC RIS 2006-04	
RIS Issue	**EGC Comments**
The submittal should include a site plan showing true north and indicating locations of all potential accident release pathways and CR intake and unfiltered inleakage pathways (whether assumed or identified during inleakage testing).	A site plan showing true north is included in the design basis X/Q calculation provided in Attachment 6. Locations of accident release and CR intake pathways are identified in Attachments 7 and 9.
The submittal should include a justification for using CR intake X/Q values for modeling the unfiltered inleakage, if applicable.	The most limiting X/Q is used for CR unfiltered inleakage, with justification provided in Attachments 7 and 9.
The submittal should include a copy of the meteorological data inputs and program outputs along with a discussion of assumptions and potential deviations from staff guidelines. Meteorological data input files should be checked to ensure quality (e.g., compared against historical or other data and against the raw data to ensure that the electronic file has been properly formatted, any unit conversions are correct, and invalid data are properly identified).	The revised X/Q values used for the AST application have been developed using appropriate meteorological data as discussed in Attachment 6. The data used has been confirmed to meet Regulatory Guide 1.23, Revision 1, and a copy of the meteorological data is included within Attachment 6.
When running the CR atmospheric dispersion model ARCON96, two or more files of meteorological data representative of each potential release height should be used if X/Q values are being calculated for both ground-level and elevated releases.	Meteorological data used in the calculation of ground-level and elevated X/Qs is provided.
In addition, licensees should be aware that (1) two levels of wind speed and direction data should always be provided as input to each data file, (2) fields of "nines" (e.g., 9999) should be used to indicate invalid or missing data, and (3) valid wind direction data should range from 1° to 360°.	Two or more levels of wind speed data are used where appropriate. Invalid or missing data are correctly indicated using a field of "nines." Wind direction data is from 1 to 360 degrees.
Licensees should also provide detailed engineering information when applying the default plume rise adjustment cited in RG 1.194 to CR X/Q values to account for buoyancy or mechanical jets of high energy releases.	No such adjustments are made relative to this license amendment request.

TABLE 3.8-1 NRC RIS 2006-04	
RIS Issue	**EGC Comments**
This information should demonstrate that the minimum effluent velocity during any time of the release over which the adjustment is being applied is greater than the 95th percentile wind speed at the height of release.	Not applicable since no such adjustments are made relative to this license amendment request.
When running the offsite atmospheric dispersion model PAVAN, two or more files of meteorological data representative of each potential release height should be used if X/Q values are being calculated for pathways with significantly different release heights (e.g., ground-level versus elevated stack).	Six years of meteorological data were used in the calculation.
The joint frequency distributions of wind speed, wind direction, and atmospheric stability data used as input to PAVAN should have a large number of wind speed categories at the lower wind speeds in order to produce the best results.	A sufficiently large number of wind speed categories at the lower wind speeds were used in the offsite X/Q calculation.
5. Modeling of ESF Leakage	
The radiological consequences from the postulated ESF leakage should be analyzed and combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA.	ESF leakage is analyzed and combined with the consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA.
Licensees should account for ESF leakage at accident conditions in their dose analyses so as not to underestimate the release rate.	ESF leakage was accounted for at accident conditions.
In Appendix A to RG 1.183, Regulatory Position 5.5, the NRC staff provided a conservative value of 10 percent as the assumed amount of iodine that may become airborne from ESF leakage that is less than 212°F.	The Regulatory Guide 1.183 recommended value of 10% is used. The suppression pool pH value remains above 7.0 for the 30-day duration of the accident. Suppression pool temperature remains below 212°F.

TABLE 3.8-1 NRC RIS 2006-04	
RIS Issue	**EGC Comments**
Figure 3.1 in NUREG/CR-5950 can be used to quantify the amount of elemental iodine as a function of the sump water pH and the concentration of iodine in the solution. In some cases, however, licensees have misapplied this figure. Rather than using the total concentration of iodine (i.e., stable and radioactive), licensees based their assessment on only the radioactive iodine in the sump water. By using only the radioactive iodine, licensees have underestimated how much iodine evolves during postaccident conditions.	The calculation methodology for containment sump pH control was based on the approach outlined in NUREG-1465 and NUREG/CR-5950. Both stable and radioactive iodine were considered.
6. Release Pathways	
Changes to the plant configuration associated with a license amendment request (e.g., an "open" containment during refueling) may require a reanalysis of the design basis dose calculations. A request for TS modifications allowing containment penetrations (i.e., personnel air lock, equipment hatch) to be open during refueling cannot rely on the current dose analysis if this analysis has not already considered these release pathways. Releases from personnel air locks and equipment hatches exposed to the environment and containment purge releases prior to containment isolation need to be addressed.	The AST application reanalyzes the design basis dose calculations for an open secondary containment during refueling and following a fuel handling accident. Specific release points are discussed in Attachment 9.
Licensees are responsible for identifying all release pathways and for considering these pathways in their AST analyses, consistent with any proposed modification.	Revised CR, EAB, and LPZ atmospheric dispersion factors for applicable release paths were identified and included in Attachments 7 and 9.

TABLE 3.8-1 NRC RIS 2006-04	
RIS Issue	**EGC Comments**
7. Primary to Secondary Leakage	
Some analysis parameters can be affected by density changes that occur in the process steam. The NRC staff continues to find errors in submittals concerning the modeling of primary to secondary leakage during a postulated accident. This issue is discussed in Information Notice (IN) 88-31, "Steam Generator Tube Rupture Analysis Deficiency," and Item 3.f in RIS 2001-19. An acceptable methodology for modeling this leakage is provided in Appendix F to RG 1.183, Regulatory Position 5.2.	These specific issues are not applicable to boiling water reactors.
8. Elemental Iodine DF	
Appendix B to RG 1.183 provides assumptions for evaluating the radiological consequences of an FHA. If the water depth above the damaged fuel is 23 feet or greater, Regulatory Position 2 states that "the decontamination factors for the elemental and organic species are 500 and 1, respectively, giving an overall effective decontamination factor of 200." However, an overall DF of 200 is achieved when the DF for elemental iodine is 285, not 500.	The depth of water over the damaged fuel is greater than 23 feet for the bounding fuel handling accident in the reactor cavity. Due to the submergence of the damaged fuel, the iodine release is assumed to experience a DF of 200 per RG 1.183.
9. Isotopes Used in Dose Assessments	
For some accidents (e.g., main steamline break and rod drop), licensees have excluded noble gas and cesium isotopes from the dose assessment. The inclusion of these isotopes should be addressed in the dose assessments for AST implementation.	The standard 60-isotope RADTRAD inventory file was used for the LOCA analysis, which includes noble gas and cesium isotopes. For the FHA analysis, cesium isotopes are assumed to be retained in the water in accordance with RG 1.183.

TABLE 3.8-1 NRC RIS 2006-04	
RIS Issue	**EGC Comments**
10. Definition of Dose Equivalent Iodine-131	
In the conversion to an AST, licensees have proposed a modification to the TS definition of Dose Equivalent Iodine-131. Although different references are available for dose conversion factors, the TS definition should be based on the same dose conversion factors that are used in the determination of the reactor coolant dose equivalent iodine curie content for the main steamline break and steam generator tube rupture accident analyses.	The definition of Dose Equivalent Iodine-131 is not being modified.
11. Acceptance Criteria for Offgas or Waste Gas System Release	
As part of full AST implementation, some licensees have included an accident involving a release from their Offgas or Waste Gas system. Any licensee who chooses to implement AST for an Offgas or Waste Gas system release should base its acceptance criteria on 100 mrem TEDE. Licensees may also choose not to implement AST for this accident and continue with their existing analysis and acceptance criteria of 500 mrem whole body.	This accident is not included with this submittal.
12. Containment Spray Mixing	
Some plants with mechanical means for mixing containment air have assumed that the containment fans intake air solely from a sprayed area and discharge it solely to an unsprayed region or vice versa. Without additional analysis, test measurements or further justification, it should be assumed that the intake of air by containment ventilation systems is supplied proportionally to the sprayed and unsprayed volumes in containment.	Containment Spray is not credited in this submittal.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

The NRC's traditional methods (i.e., 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 TEDE criteria provided in 10 CFR 50.67. Regulatory Guide 1.183 provides assumptions and methods that are acceptable to the NRC 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.

Also, the NRC published SRP Section 15.0.1 (i.e., Reference 3) to address AST. SRP Section 15.0.1 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 addresses the guidance in SRP 15.0.1.

4.2 No Significant Hazards Consideration

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company, LLC (EGC) requests an amendment to Facility Operating License Nos. NPF-11 and NPF-18 for LaSalle County Station (LSCS), Units 1 and 2. Specifically, EGC is requesting a revision to the Technical Specifications (TS) 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 NRC 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 any 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.

EGC has evaluated the proposed change, using the criteria in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The following information is provided to support a finding of no significant hazards consideration.

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

   Response: No

   The implementation of AST assumptions has been evaluated in revisions to the analyses of the following limiting design basis accidents at LSCS:

   - Loss-of-Coolant Accident, and
   - Fuel Handling 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 requirements and guidance provided by the NRC for use with the AST. The regulatory requirements and guidance is presented in 10 CFR 50.67, "Accident source term," and associated NRC Regulatory Guide 1.183 and Standard Review Plan Section 15.0.1. The AST 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 change is mitigative in nature, and relied upon after an accident has been initiated. Application of the AST does not involve any physical changes to the plant design and is not an initiator of an accident. The proposed changes to the 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 AST analyses, the probability of an accident previously evaluated is not affected.

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

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

   Response: No

   The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed and there are no physical modifications to existing equipment associated with the proposed change). 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.

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

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

   Response: No

   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 change is associated with the implementation of a new licensing basis for LSCS design basis accidents. 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 design basis accidents remain within the acceptance criteria presented in 10 CFR 50.67 and Regulatory Guide 1.183.

   The proposed change continues to ensure that the doses at the exclusion area boundary and low population zone boundary, as well as the control room, are within corresponding regulatory limits.

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

4.3 Conclusions

Based on the above evaluation, EGC concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92, paragraph (c), and accordingly, a finding of no significant hazards consideration is justified.

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.

5.0 ENVIRONMENTAL CONSIDERATION

EGC has determined that the proposed amendment would change 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." However, the proposed amendment does not involve: (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion 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). Therefore, pursuant to 10 CFR 51.22, paragraph (b), no environmental impact statement or environmental assessment needs to be prepared in connection with the proposed amendment.

6.0 REFERENCES

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

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

3. NRC Standard Review Plan 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Revision 0, dated July 2000

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

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

6. Nuclear Utilities Management and Resources Council (NUMARC) 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 3

7. RSIC Code Package CCC-371, "ORIGEN 2.1, Isotope Generation and Depletion Code Matrix Exponential Method," dated May 1999

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

9. Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water, and Soil," 1993

10. NRC Standard Review Plan 6.4, "Control Room Habitability Systems," Revision 2, dated July 1981

11. NRC Regulatory Guide 1.23 (Safety Guide 23), "Onsite Meteorological Programs," Proposed Revision 1, dated February 17, 1972

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

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

14. NUREG-0737, "Clarification of TMI Action Plan Requirements," dated October 1980

15. Letter from G. G. Benes (Commonwealth Edison Company) to NRC, "Supplement to August 28, 1995 Request for Application for Amendment to Facility Operating Licenses NPF-11 and NPF-18, Appendix A, Technical Specifications, and Exemption to Appendix J of 10CFR50 Regarding Elimination of MSIV Leakage Control System and Increased MSIV Leakage Limits," dated December 15, 1995

16. Letter from M. D. Lynch (NRC) to D. L Farrar (Commonwealth Edison Company), "Issuance of Amendments (TAC Nos. M93597 and M93598)," dated April 5, 1996

17. NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments," dated July 1996

18. NEDC-31858P-A, "BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems," Revision 2

19. AEB-98-03, "Assessment of Radiological Consequences for the Perry Pilot Plant Application using the Revised (NUREG-1465) Source Term," dated December 9, 1998

20. J. E. Cline & Associates, Inc. and Science Applications International Corporation Report, "MSIV Leakage – Iodine Transport Analysis," dated August 20, 1990

21. NRC Review Guidelines, "Guidance on the Assessment of a BWR SLC System for pH Control," dated February 12, 2004 (ADAMS Accession No. ML040640364)

22. NRC Regulatory Guide 1.97, Instrumentation for Light-Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," Revision 2, dated December 1980

23. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," Revision 15, dated September 2005

24. NRC Regulatory Issue Summary 2006-04, "Experience with Implementation of Alternative Source Terms," dated March 7, 2006

ATTACHMENT 2
Markup of Proposed Technical Specifications Pages

LaSalle County Station, Units 1 and 2

Facility Operating License Nos. NPF-11 and NPF-18

REVISED TECHNICAL SPECIFICATIONS PAGES

1.1-6
3.1.7-1
3.3.6.1-9
3.3.6.2-4
3.3.7.1-1
3.6.1.3-8
3.6.4.1-1
3.6.4.1-2
3.6.4.1-3
3.6.4.2-1
3.6.4.2-3
3.6.4.3-1
3.6.4.3-2
3.6.4.3-3
3.7.4-1
3.7.4-2
3.7.4-3
3.7.5-1
3.7.5-2
3.7.5-3
5.5-13
5.5-14

1.1 Definitions (continued)

REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME

The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and method for verification have been previously reviewed and approved by the NRC.

INSERT 1.1-1

SHUTDOWN MARGIN (SDM)

SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that:

a. The reactor is xenon free;

b. The moderator temperature is 68°F; and

c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.

STAGGERED TEST BASIS

A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.

THERMAL POWER

THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

(continued)

ATTACHMENT 2
Markup of Proposed Technical Specifications Pages

INSERT 1.1-1

RECENTLY IRRADIATED FUEL	RECENTLY IRRADIATED FUEL is fuel that has occupied part of a critical reactor core within the previous 24 hours.

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.~~ , 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SLC subsystem inoperable.	A.1 Restore SLC subsystem to OPERABLE status.	7 days
B. Two SLC subsystems inoperable.	B.1 Restore one SLC subsystem to OPERABLE status.	8 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.7.1 Verify available volume of sodium pentaborate solution is within the limits of Figure 3.1.7-1.	24 hours

(continued)

Table 3.3.6.1-1 (page 4 of 4)
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
4. RWCU System Isolation (continued)					
k. Reactor Vessel Water Level–Low Low, Level 2	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5	≥ -58.0 inches
l. Standby Liquid Control System Initiation	1,2 ,3	2[b]	I	SR 3.3.6.1.5	NA
m. Manual Initiation	1,2,3	1	G	SR 3.3.6.1.5	NA
5. RHR Shutdown Cooling System Isolation					
a. Reactor Vessel Water Level–Low, Level 3	3,4,5	2[c]	J	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5	≥ 11.0 inches
b. Reactor Vessel Pressure–High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 143 psig
c. Manual Initiation	1,2,3	1	G	SR 3.3.6.1.5	NA

(b) Only inputs into one of two trip systems.

(c) Only one trip system required in MODES 4 and 5 with RHR Shutdown Cooling System integrity maintained.

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

FUNCTION	APPLICABLE MODES AND OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level—Low Low, Level 2	1,2,3,(a)	2	SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≥ -58.0 inches
2. Drywell Pressure—High	1,2,3	2	SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≤ 1.93 psig
3. Reactor Building Ventilation Exhaust Plenum Radiation—High	1,2,3, (a),(b)	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≤ 42.0 mR/hr
4. Fuel Pool Ventilation Exhaust Radiation—High	1,2,3, (a),(b)	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≤ 42.0 mR/hr
5. Manual Initiation	1,2,3, (a),(b)	1	SR 3.3.6.2.4	NA

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

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

RECENTLY IRRADIATED FUEL

3.3 INSTRUMENTATION

3.3.7.1 Control Room Area Filtration (CRAF) System Instrumentation

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

RECENTLY IRRADIATED FUEL

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 channels inoperable.	A.1	Declare associated CRAF subsystem inoperable.	1 hour from discovery of loss of CRAF subsystem initiation capability
	<u>AND</u>		
	A.2	Place channel in trip.	6 hours

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.6	Verify the isolation time of each MSIV is ≥ 3 seconds and ≤ 5 seconds.	In accordance with the Inservice Testing Program
SR 3.6.1.3.7	Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal.	24 months
SR 3.6.1.3.8	Verify each reactor instrumentation line EFCV actuates to the isolation position on an actual or simulated instrument line break signal.	24 months
SR 3.6.1.3.9	Remove and test the explosive squib from each shear isolation valve of the TIP System.	24 months on a STAGGERED TEST BASIS
SR 3.6.1.3.10 200	Verify leakage rate through any one main steam line is ≤ 100 scfh and through all four main steam lines is ≤ 400 scfh when tested at ≥ 25.0 psig.	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.3.11	Verify combined leakage rate through hydrostatically tested lines that penetrate the primary containment is within limits.	In accordance with the Primary Containment Leakage Rate Testing Program

3.6 CONTAINMENT SYSTEMS

3.6.4.1 Secondary Containment

LCO 3.6.4.1 The secondary containment shall be OPERABLE.

RECENTLY IRRADIATED FUEL

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

(continued)

ACTIONS

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

RECENTLY IRRADIATED FUEL

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1.1	Verify secondary containment vacuum is ≥ 0.25 inch of vacuum water gauge.	24 hours
SR 3.6.4.1.2	Verify one secondary containment access door in each access opening is closed.	31 days
SR 3.6.4.1.3 900	Verify the secondary containment can be drawn down to ≥ 0.25 inch of vacuum water gauge in ≤ 300 seconds using one standby gas treatment (SGT) subsystem.	24 months on a STAGGERED TEST BASIS for each SGT subsystem
SR 3.6.4.1.4	Verify the secondary containment can be maintained ≥ 0.25 inch of vacuum water gauge for 1 hour using one SGT subsystem at a flow rate ≤ 4400 cfm.	24 months on a STAGGERED TEST BASIS for each SGT subsystem
SR 3.6.4.1.5	Verify all secondary containment equipment hatches are closed and sealed.	24 months

3.6 CONTAINMENT SYSTEMS

3.6.4.2 Secondary Containment Isolation Valves (SCIVs)

LCO 3.6.4.2 Each SCIV shall be OPERABLE.

RECENTLY IRRADIATED FUEL

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

ACTIONS

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

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

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

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

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

ACTIONS

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

RECENTLY IRRADIATED FUEL

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

ACTIONS

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

RECENTLY IRRADIATED FUEL

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	C.2.1 Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
RECENTLY IRRADIATED FUEL	~~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 Be in MODE 3	12 hours
E. Two SGT subsystems inoperable during movement of irradiated fuel assemblies in the secondary containment, ~~during CORE ALTERATIONS,~~ or during OPDRVs.	E.1 --------NOTE--------- LCO 3.0.3 is not applicable. --------------------- Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	AND	
		(continued)

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.3.1	Operate each SGT subsystem for ≥ 10 continuous hours with heaters operating.	31 days
SR 3.6.4.3.2	Perform required SGT filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.6.4.3.3	Verify each SGT subsystem actuates on an actual or simulated initiation signal.	24 months

3.7 PLANT SYSTEMS

3.7.4 Control Room Area Filtration (CRAF) System

LCO 3.7.4 Two CRAF subsystems shall be OPERABLE.

-----------------NOTE-----------------
The control room envelope (CRE) boundary may be opened intermittently under administrative control.

RECENTLY IRRADIATED FUEL

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One CRAF subsystem inoperable for reasons other than Condition B.	A.1	Restore CRAF subsystem to OPERABLE status.	7 days
B. One or more CRAF subsystems inoperable due to inoperable CRE boundary in MODE 1, 2, or 3.	B.1	Initiate action to implement mitigating actions.	Immediately
	AND		
	B.2	Verify mitigating actions ensure CRE occupant exposures to radiological, chemical, and smoke hazards will not exceed limits.	24 hours
	AND		
	B.3	Restore CRE boundary to OPERABLE status.	90 days

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and Associated Completion Time of Condition A or B not met in MODE 1, 2, or 3.	C.1 Be in MODE 3.	12 hours
D. Required Action and associated Completion Time of Condition A not met during movement of ~~irradiated fuel~~ assemblies in the secondary containment~~, during CORE ALTERATIONS,~~ ~~or~~ during OPDRVs. RECENTLY IRRADIATED FUEL	------------NOTE------------ LCO 3.0.3 is not applicable. -----------------------------	
	D.1 Place OPERABLE CRAF subsystem in pressurization mode.	Immediately
	<u>OR</u>	
	D.2.1 Suspend movement of ~~irradiated fuel~~ assemblies in the secondary containment.	Immediately
	~~<u>AND</u>~~	
	~~D.2.2 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	<u>AND</u>	
	D.2.~~3~~ 2 Initiate action to suspend OPDRVs.	Immediately
E. Two CRAF subsystems inoperable in MODE 1, 2, or 3 for reasons other than Condition B.	E.1 Be in MODE 3.	12 hours

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Two CRAF subsystems inoperable during movement of ~~irradiated fuel~~ assemblies in the secondary containment~~, during CORE ALTERATIONS,~~ or during OPDRVs. <u>OR</u> One or more CRAF subsystems inoperable due to inoperable CRE boundary during movement of ~~irradiated fuel~~ assemblies in the secondary containment~~, during CORE ALTERATIONS,~~ or during OPDRVS.	------------NOTE------------ LCO 3.0.3 is not applicable. ---------------------------- F.1 Suspend movement of ~~irradiated fuel~~ assemblies in the secondary containment. ~~<u>AND</u>~~ ~~F.2 Suspend CORE ALTERATIONS.~~ <u>AND</u> F.~~3~~ 2 Initiate action to suspend OPDRVs.	 Immediately ~~Immediately~~ Immediately

RECENTLY IRRADIATED FUEL

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.4.1	Operate each CRAF subsystem for ≥ 10 continuous hours with the heaters operating.	31 days

(continued)

3.7 PLANT SYSTEMS

3.7.5 Control Room Area Ventilation Air Conditioning (AC) System

LCO 3.7.5 Two control room area ventilation AC subsystems shall be OPERABLE.

RECENTLY IRRADIATED FUEL

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One control room area ventilation AC subsystem inoperable.	A.1 Restore control room area ventilation AC subsystem to OPERABLE status.	30 days
B. Two control room area ventilation AC subsystems inoperable.	B.1 Verify control room area temperature < 90°F. AND B.2 Restore one control room area ventilation AC subsystem to OPERABLE status.	Once per 4 hours 72 hours
C. Required Action and Associated Completion Time of Condition A or B not met in MODE 1, 2, or 3.	C.1 Be in MODE 3.	12 hours

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A not met during movement of ~~irradiated fuel~~ assemblies in the secondary containment~~, during CORE ALTERATIONS, or during~~ OPDRVs.	------------NOTE------------ LCO 3.0.3 is not applicable. ----------------------------	
	D.1 Place OPERABLE control room area ventilation AC subsystem in operation.	Immediately
	<u>OR</u>	
	D.2.1 Suspend movement of ~~irradiated fuel~~ assemblies in the secondary containment.	Immediately
	~~<u>AND</u>~~	
	~~D.2.2 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	<u>AND</u>	
	D.2.~~3~~ 2 Initiate action to suspend OPDRVs.	Immediately

RECENTLY IRRADIATED FUEL

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. Required Action and associated Completion Time of Condition B not met during movement of irradiated fuel assemblies in the secondary containment, ~~during CORE ALTERATIONS,~~ or during OPDRVs. RECENTLY IRRADIATED FUEL	------------NOTE------------ LCO 3.0.3 is not applicable. ----------------------------		
	E.1	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

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	Monitor control room and auxiliary electric equipment room temperatures.	12 hours
SR 3.7.5.2	Verify correct breaker alignment and indicated power are available to the control room area ventilation AC subsystems.	7 days

5.5 Programs and Manuals

5.5.13 <u>Primary Containment Leakage Rate Testing Program</u> (continued)

2. NEI 94-01 - 1995, Section 9.2.3: The first Unit 2 Type A test performed after December 8, 1993 Type A test shall be performed prior to startup following L2R12.

3. The potential valve atmospheric leakage paths that are not exposed to reverse direction test pressure shall be tested during the regularly scheduled Type A test. The program shall contain the list of the potential valve atmospheric leakage paths, leakage rate measurement method, and acceptance criteria. This exception shall be applicable only to valves that are not isolable from the primary containment free air space.

b. The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a, is 39.9 psig.

c. The maximum allowable primary containment leakage rate, L_a, at P_a, is 0.~~6~~35% of primary containment air weight per day.
1.6%

d. Leakage rate acceptance criteria are:

1. Primary containment overall 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 combined Type B and Type C tests, and $\leq 0.75\ L_a$ for Type A tests.

2. Air lock testing acceptance criteria are:

a) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

b) For each door, the seal leakage rate is $\leq$ 5 scf per hour when the gap between the door seals is pressurized to $\geq$ 10 psig.

e. The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

(continued)

5.5 Programs and Manuals

5.5.14 Battery Monitoring and Maintenance Program

This Program provides for restoration and maintenance, which includes the following:

a. Actions to restore battery cells with float voltage < 2.13 V; and

b. Actions to equalize and test battery cells that had been discovered with electrolyte level below the top of the plates; and

c. Actions to verify that the remaining cells are ≥ 2.07 V when a cell or cells have been found to be < 2.13 V.

5.5.15 Control Room Envelope Habitability Program

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

, or 5 rem TEDE, as applicable

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

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

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

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of

ATTACHMENT 3
Markup of Proposed Technical Specifications Bases Pages

LaSalle County Station, Units 1 and 2

Facility Operating License Nos. NPF-11 and NPF-18

REVISED TECHNICAL SPECIFICATIONS BASES PAGES

B 2.1.1-5	B 3.6.4.1-4
B 2.1.2-1	B 3.6.4.1-5
B 2.1.2-2	B 3.6.4.1-6
B 2.1.2-3	B 3.6.4.2-1
B 3.1.7-1	B 3.6.4.2-2
B 3.1.7-2	B 3.6.4.2-5
B 3.1.7-3	B 3.6.4.2-7
B 3.1.7-6	B 3.6.4.3-2
B 3.1.8-1	B 3.6.4.3-3
B 3.1.8-5	B 3.6.4.3-4
B 3.2.3-1	B 3.6.4.3-5
B 3.3.6.1-8	B 3.6.4.3-6
B 3.3.6.1-12	B 3.7.4-2
B 3.3.6.1-13	B 3.7.4-3
B 3.3.6.1-14	B 3.7.4-4
B 3.3.6.1-25	B 3.7.4-5
B 3.3.6.2-2	B 3.7.4-6
B 3.3.6.2-5	B 3.7.4-7
B 3.3.6.2-6	B 3.7.4-8
B 3.3.6.2-8	B 3.7.4-10
B 3.3.6.2-10	B 3.7.5-3
B 3.3.6.2-11	B 3.7.5-5
B 3.3.6.2-12	B 3.7.5-6
B 3.3.7.1-2	B 3.7.8-1
B 3.3.7.1-3	B 3.7.8-3
B 3.6.1.1-2	B 3.9.6-1
B 3.6.1.2-2	B 3.9.6-2
B 3.6.1.3-15	B 3.9.6-3
B 3.6.4.1-1	B 3.9.7-1
B 3.6.4.1-2	B 3.9.7-2
B 3.6.4.1-3	B 3.9.7-3

ATTACHMENT 3
Markup of Proposed Technical Specifications Bases Pages

Insert 3.1.7-1

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 (Ref. 3).

Insert 3.1.7-2

Following a LOCA, offsite doses from the accident will remain within 10 CFR 50.67, "Accident Source Term," limits (Ref. 4) provided sufficient iodine activity is retained in the suppression pool. Credit for iodine deposition in the suppression pool is allowed (Ref. 3) as long as suppression pool pH is maintained at or above 7. Alternative Source Term analyses credit the use of the SLC System for maintaining the pH of the suppression pool at or above 7.

Insert 3.1.7-3

In MODES 1, 2, and 3, the SLC System must be OPERABLE to ensure that offsite doses remain within 10 CFR 50.67 (Ref. 4) 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 (Ref. 3).

Insert 3.1.7-4

3. NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants, Final Report," February 1, 1995.

4. 10 CFR 50.67.

Insert 3.3.6.1-1

. In addition, both channels are 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 3.3.6.2-1

Due to radioactive decay, these Functions are only required to isolate secondary containment during fuel handling accidents involving handling RECENTLY IRRADIATED FUEL.

Insert 3.3.7.1-1

Also due to radioactive decay, this Function is only required to be OPERABLE during fuel handling accidents involving handling RECENTLY IRRADIATED FUEL.

Insert 3.6.4.1-1

Due to radioactive decay, secondary containment is only required to be OPERABLE during fuel handling involving handling RECENTLY IRRADIATED FUEL.

Insert 3.6.4.1-2

Although secondary containment OPERABILITY is not required to move fuel that has had sufficient decay, only certain openings to the outside are allowed during fuel movement. The Reactor Building truck bay doors (between columns D14 and D15 at grade level) and penetrations MK-1RB-782 and MK-1RB-786 (ILRT opening) on the Reactor Building east wall can be open simultaneously while fuel movement is in progress during a dual unit shutdown, but must be closed if either unit is in MODES 1, 2, or 3. However, the ILRT opening may be open in MODES 1, 2, or 3 if fuel movement is not in progress and secondary containment drawdown requirements are met. Any other external opening will need to be evaluated on a case-by-case basis. Internal openings to adjacent areas are allowed; however, other DBAs such as HELB must be considered.

Insert 3.6.4.2-1

Due to radioactive decay, SCIVs are only required to be OPERABLE during fuel handling involving handling RECENTLY IRRADIATED FUEL.

Insert 3.6.4.3-1

Due to radioactive decay, the SGT System is only required to be OPERABLE during fuel handling involving handling RECENTLY IRRADIATED FUEL.

Insert 3.7.4-1

Due to radioactive decay, the CRAF System is only required to be OPERABLE during fuel handling involving handling RECENTLY IRRADIATED FUEL.

Insert 3.7.5-1

Due to radioactive decay, the Control Room Area Ventilation AC System is only required to be OPERABLE during fuel handling involving handling RECENTLY IRRADIATED FUEL.

Insert 3.7.8-1

(calculated control room operator dose and doses at the exclusion area and low population zone boundaries) are below the 10 CFR 50.67 (Ref. 3) exposure guidelines, as modified by Regulatory Guide 1.183, Table 6.

BASES (continued)

SAFETY LIMITS

The reactor core SLs are established to protect the integrity of the fuel clad barrier to prevent 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

50.67, "Accident Source Term

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. 5). Therefore, it is required to insert all insertable control rods and restore compliance with the SL within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and the probability of an accident occurring during this period is minimal.

REFERENCES

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

2. ANF-524(P)(A), Revision 2, Supplement 1 Revision 2, Supplement 2, Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors/Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors: Methodology for Analysis of Assembly Channel Bowing Effects/NRC Correspondence (as specified in Technical Specification 5.6.5).

3. EMF-2209(P)(A), SPCB Critical Power Correlation, AREVA NP (as specified in Technical Specification 5.6.5).

4. EMF-2245(P)(A), Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel, AREVA NP (as specified in Technical Specification 5.6.5).

5. 10 CFR 100. 50.67

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. According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (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 anticipated operational occurrences (AOOs).

During normal operation and AOOs, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2) for the reactor pressure vessel, and by more than 20%, in accordance with USAS B31.1-1967 Code (Ref. 3) for the RCS piping. 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. Following inception of unit operation, RCS components shall be pressure tested in accordance with the requirements of ASME Code, Section XI (Ref. 4).

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. 5). If this occurred in conjunction with a fuel cladding failure, the number of protective barriers designed to prevent radioactive releases from exceeding the limits would be reduced.

(continued)

50.67, "Accident Source Term"

BASES (continued)

APPLICABLE SAFETY ANALYSES

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

The RCS pressure SL has been selected such that it is at a pressure below which it can be shown that the integrity of the system is not endangered. The reactor pressure vessel is designed to ASME, Boiler and Pressure Vessel Code, Section III, 1968 Edition, including Addenda through the winter of 1969 for Unit 1 and winter of 1970 (excluding Appendix I) for Unit 2 (Ref. 6), which permits a maximum pressure transient of 110%, 1375 psig, of design pressure 1250 psig. The SL of 1325 psig, as measured in the reactor steam dome, is equivalent to 1375 psig at the lowest elevation of the RCS. The RCS is designed to ASME Code, Section III, 1971 Edition, including Addenda through the summer of 1971 (Ref. 7), for the reactor recirculation piping, which permits a maximum pressure transient of 120% of design pressures of 1150 psig for suction piping and 1250 psig for discharge piping. The recirculation pumps are designed to ASME Code, Section III, 1971 Edition, including Addenda through the summer of 1971 (Ref. 7). The RCS pressure SL is selected to be the lowest transient overpressure allowed by the applicable codes.

SAFETY LIMITS

The maximum transient pressure allowable in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowable in the RCS piping, valves, and fittings is 120% of design pressures of 1150 psig for suction piping and 1250 psig for discharge piping. The most limiting of these allowances is the 110% of the reactor pressure vessel design pressure; therefore, the SL on maximum allowable RCS pressure is established at 1325 psig as measured at the reactor steam dome.

APPLICABILITY

SL 2.1.2 applies in all MODES.

SAFETY LIMIT VIOLATIONS

2.2 50.67, "Accident Source Term,"

Exceeding the RCS pressure SL may cause RCS failure and create a potential for radioactive releases in excess of 10 CFR ~~100, "Reactor Site Criteria,"~~ limits (Ref. 5).

(continued)

BASES

SAFETY LIMIT VIOLATIONS

2.2 (continued)

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 this period is minimal.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 14 and GDC 15.
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000.
3. ASME, USAS, Power Piping Code, Section B31.1, 1967.
4. ASME, Boiler and Pressure Vessel Code, Section XI, Article IWB-5000.
5. 10 CFR ~~100~~. 50.67
6. ASME, Boiler and Pressure Vessel Code, Section III, 1968 Edition, Addenda, winter of 1969 (Unit 1) and winter of 1970 (Unit 2).
7. ASME, Boiler and Pressure Vessel Code, Section III, 1971 Edition, Addenda, summer of 1971.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Standby Liquid Control (SLC) System

BASES

BACKGROUND

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

INSERT 3.1.7-1

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

APPLICABLE SAFETY ANALYSES

The SLC System is manually initiated from the main control room, as directed by the emergency operating procedures, if the operator determines the reactor cannot be shut down, or kept shut down, with the control rods. The SLC System is used in the event that not enough control rods can be inserted to accomplish shutdown and cooldown in the normal manner. The SLC System injects borated water into the reactor core to compensate for all of the various reactivity effects that could occur during plant operation. To meet this objective, it is necessary to inject a quantity of boron that produces a reactivity change equivalent to a concentration of 660 ppm of enriched boron in the reactor core at 68°F. To ensure this objective is met, a sodium pentaborate solution enriched with boron-10 is used. The shutdown analysis assumes a sodium pentaborate solution with enriched boron is used (Ref. 2). A 45% enriched sodium pentaborate solution is also used to satisfy the requirements of Reference 1. To allow for potential leakage and imperfect mixing in the reactor system, an additional amount of boron equal to 25% of the amount cited above is added (Ref. 2). An additional 250 ppm is provided to

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

accommodate dilution in the RPV by the residual heat removal shutdown cooling piping. The volume versus concentration limits in Figure 3.1.7-1 are calculated such that the required concentration is achieved. This quantity of borated solution is the amount that is above the pump suction shutoff level in the boron solution storage tank. No credit is taken for the portion of the tank volume that cannot be injected.

INSERT 3.1.7-2

The SLC System satisfies ~~Criterion 4~~ of 10 CFR 50.36(c)(2)(ii). Criteria 3 and 4

LCO

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

APPLICABILITY

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

INSERT 3.1.7-3

ACTIONS

A.1

If one SLC System subsystem is inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this condition, the remaining OPERABLE subsystem is adequate to perform the shutdown function. However, the overall reliability is reduced because a single failure in

(continued)

BASES

ACTIONS

A.1 (continued)

the remaining OPERABLE subsystem could result in reduced SLC System shutdown capability and inability to meet the requirements of Reference 1. The 7 day Completion Time is based on the availability of an OPERABLE subsystem capable of performing the unit shutdown function and the low probability of a Design Basis Accident (DBA) or severe transient occurring concurrent with the failure of the Control Rod Drive System to shut down the reactor.

B.1

If both SLC subsystems are inoperable, at least one subsystem must be restored to OPERABLE status within 8 hours. The allowed Completion Time of 8 hours is considered acceptable, given the low probability of a DBA or transient occurring concurrent with the failure of the control rods to shut down the reactor.

and MODE 4 within 36 hours

C.1

Times are

the required unit conditions

If any Required Action and associated Completion Time is not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to 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.

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 volume and temperature of the borated solution in the storage tank), thereby ensuring the SLC System OPERABILITY without disturbing normal plant operation. These Surveillances ensure the proper borated solution and temperature, including the temperature (using the local indicator) of the pump suction piping up to the storage tank outlet valves, are maintained. Maintaining a minimum specified borated solution temperature is important in

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.8 and SR 3.1.7.9 (continued)

should be alternated such that both complete flow paths are tested every 48 months, at alternating 24 month intervals. The Surveillance may be performed in separate steps to prevent injecting boron into the RPV. An acceptable method for verifying flow from the pump to the RPV is to pump demineralized water from a test tank through one SLC subsystem and into the RPV. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance test when performed at the 24 month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

Demonstrating that all heat traced piping in the flow path between the boron solution storage tank and the storage tank outlet valves to the injection pumps is unblocked ensures that there is a functioning flow path for injecting the sodium pentaborate solution. An acceptable method for verifying that the suction piping up to the storage tank outlet valves is unblocked is to verify flow from the storage tank to the test tank. Upon completion of this verification, the pump suction piping between the storage tank outlet valve and pump suction must be drained and flushed with demineralized water, since the piping is not heat traced. The 24 month Frequency is acceptable since there is a low probability that the subject piping will be blocked due to precipitation of the boron from solution in the heat traced piping. This is especially true in light of the daily temperature verification of this piping required by SR 3.1.7.3. However, if, in performing SR 3.1.7.3, it is determined that the temperature of this piping has fallen below the specified minimum, SR 3.1.7.9 must be performed once within 24 hours after the piping temperature is restored within the limits of Figure 3.1.7-2.

REFERENCES

1. 10 CFR 50.62.

2. UFSAR, Section 9.3.5.3.

INSERT 3.1.7-4

B 3.1 REACTIVITY CONTROL SYSTEMS

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

BASES

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

APPLICABLE SAFETY ANALYSES

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

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

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

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

(continued)

BASES

SURVEILLANCE

SR 3.1.8.3 (continued)

reset signal, the opening of the SDV vent and drain valves is verified. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.1.1 and the scram time testing of control rods in LCO 3.1.3, "Control Rod OPERABILITY," overlap this Surveillance to provide complete testing of the assumed safety function. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 4.6.1.1.2.
2. 10 CFR ~~100~~. 50.67
3. NUREG-0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping," August 1981.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

BASES

BACKGROUND

The LHGR is a measure of the heat generation rate of a fuel rod in a fuel assembly at any axial location. Limits on the LHGR are specified to ensure that fuel design limits are not exceeded anywhere in the core during normal operation, including anticipated operational occurrences (AOOs), and to ensure that the peak cladding temperature (PCT) during the postulated design basis loss of coolant accident (LOCA) does not exceed the limits specified in 10 CFR 50.46. Exceeding the LHGR limit could potentially result in fuel damage and subsequent release of radioactive materials. Fuel design limits are specified to ensure that fuel system damage, fuel rod failure or inability to cool the fuel does not occur during the normal operations and anticipated operating conditions identified in References 1 and 2.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel system design and establish LHGR limits are presented in References 1, 2, 3, 4, 5, and 6. The fuel assembly is designed to ensure (in conjunction with the core nuclear and thermal hydraulic design, plant equipment, instrumentation, and protection system) that fuel damage will not result in the release of radioactive materials in excess of the guidelines of 10 CFR, Parts 20, ~~50,~~ ~~and~~ 100 and 50. A mechanism that could cause fuel damage during normal operations and operational transients and that is considered in fuel evaluations is a rupture of the fuel rod cladding caused by strain from the relative expansion of the UO_2 pellet.

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

Fuel design evaluations have been performed and demonstrate that the 1% fuel cladding plastic strain design limit is not exceeded during continuous operation with LHGRs up to the operating limit specified in the COLR. The analysis also includes allowances for short term transient excursions above the operating limit while still remaining within the AOO limits, plus an allowance for densification power spiking.

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

1. Main Steam Line Isolation

1.a. Reactor Vessel Water Level—Low Low Low, Level 1

Low reactor pressure vessel (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 the MSIVs and other interfaces with the reactor vessel occurs to prevent offsite dose limits from being exceeded. The Reactor Vessel Water Level—Low Low Low, Level 1 Function is one of the many Functions assumed to be OPERABLE and capable of providing isolation signals. The Reactor Vessel Water Level—Low Low Low, Level 1 Function associated with isolation is assumed in the analysis of the recirculation line break (Ref. 2). The isolation of the MSL on Level 1 supports actions to ensure that offsite dose limits are not exceeded for a DBA.

Reactor vessel water level signals are initiated from four differential pressure 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 Low Low, Level 1 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 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.

This Function isolates the Group 1 valves. 50.67

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/hour 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 event (Ref. 4). The closure of the MSIVs ensures

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

1.f. Manual Initiation

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

There are four push buttons for the logic, with two manual initiation push buttons per trip system. Four channels of Manual Initiation Function are available and are required to be OPERABLE in MODES 1, 2, and 3, since these are the MODES in which the MSL Isolation automatic Functions are required to be OPERABLE.

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

This Function isolates the Group 1 valves.

2. Primary Containment Isolation

2.a Reactor Vessel Water Level—Low Low, Level 2

Low RPV water level indicates 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 2 supports actions to ensure that offsite dose limits of 10 CFR 100 are not exceeded. The Reactor Vessel Water Level—Low Low, Level 2 Function associated with isolation is implicitly assumed in the UFSAR analysis as these leakage paths are assumed to be isolated post LOCA.

50.67

Reactor Vessel Water Level—Low Low, Level 2 signals are initiated from differential pressure transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

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

water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level—Low Low, Level 2 Function are available and are required to be OPERABLE to ensure no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level—Low Low, Level 2 Allowable Value was chosen to be the same as the ECCS Reactor Vessel Water Level—Low Low, Level 2 Allowable Value (LCO 3.3.5.1), since isolation of these valves is not critical to orderly plant shutdown.

This Function isolates the Group 2, 3, and 4 valves.

2.b Drywell Pressure—High

50.67

High drywell pressure can indicate a break in the RCPB inside the drywell. The isolation of some of the PCIVs 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 switches that sense the pressure in the drywell. Four channels of Drywell Pressure—High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be the same as the RPS Drywell Pressure—High Allowable Value (LCO 3.3.1.1), since this may be indicative of a LOCA inside primary containment.

This Function isolates the Group 2, 4, 7, and 10 valves.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

2.c. Reactor Building Ventilation Exhaust Plenum Radiation—High

High ventilation exhaust radiation is an indication of possible gross failure of the fuel cladding. The release may have originated from the primary containment due to a break in the RCPB or refueling floor due to a fuel handling accident. When Reactor Building Ventilation Exhaust Radiation—High is detected, valves whose penetrations communicate with the primary containment atmosphere are isolated to limit the release of fission products.

The Reactor Building Ventilation Exhaust Plenum Radiation-High signals are initiated from radiation detectors that are located in the reactor building return air riser above the upper area of the steam tunnel prior to the reactor building ventilation isolation dampers. The signal from each detector is input to an individual monitor whose trip outputs are assigned to an isolation channel. Four channels of Reactor Building Ventilation Exhaust Plenum Radiation—High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Values are chosen to promptly detect gross failure of the fuel cladding and to ensure offsite doses remain below 10 CFR 20 and 10 CFR 100 limits. 50.67

These Functions isolate the Group 4 valves.

2.d. Fuel Pool Ventilation Exhaust Radiation—High

High fuel pool ventilation exhaust radiation indicates increased airborne radioactivity levels in secondary containment refuel floor area which could be due to fission gases from the fuel pool resulting from a refueling accident. Since the primary and secondary containments may be in communication, the vent and purge valves for primary containment isolation are also provided with an isolation signal. Therefore, Fuel Pool Ventilation Exhaust Radiation-High Function initiates an isolation to assure timely closure of valves to protect against substantial releases of radioactive materials to the environment. While this Function is identified as initiating the Standby Gas Treatment System for a spent fuel cask drop accident

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

4.k. Reactor Vessel Water Level—Low Low, Level 2

Low RPV water level indicates the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, isolation of some reactor vessel interfaces occurs to isolate the potential sources of a break. The isolation of the RWCU System on Level 2 supports actions to ensure that fuel peak cladding temperature remains below the limits of 10 CFR 50.46. The Reactor Vessel Water Level—Low Low, Level 2 Function associated with RWCU isolation is not directly assumed in any transient or accident analysis, since bounding analyses are performed for large breaks such as MSLBs.

Reactor Vessel Water Level—Low Low, Level 2 signals are initiated from differential pressure 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 Low, Level 2 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 Low, Level 2 Allowable Value was chosen to be the same as the ECCS Reactor Vessel Water Level—Low Low, Level 2 Allowable Value (LCO 3.3.5.1), since the capability to cool the fuel may be threatened.

This Function isolates the Group 5 valves.

4.l. 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. 8). SLC System initiation signals are initiated from the two SLC pump start signals.

INSERT 3.3.6.1-1

Two channels (one from each pump) of 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, "SLC

(continued)

BASES

BACKGROUND (continued)

subsystems with both subsystems being initiated by each trip system. Automatically isolated secondary containment penetrations are isolated by two isolation valves. Each trip system initiates isolation of one of two SCIVs so that operation of either trip system isolates the associated penetrations.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

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

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

The secondary containment isolation instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii). Certain instrumentation Functions are retained for other reasons and are described below in the individual Functions discussion.

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

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

Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2. Drywell Pressure—High (continued)

safety analysis. However, the Drywell Pressure—High Function associated with isolation is not assumed in any UFSAR accident or transient analysis. It is retained for the overall redundancy and diversity of the secondary containment isolation instrumentation as required by the NRC approved licensing basis.

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

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

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

3, 4. Reactor Building Ventilation Exhaust Plenum and Fuel Pool Ventilation Exhaust Radiation—High

High secondary containment exhaust radiation is an indication of possible gross failure of the fuel cladding. The release may have originated from the primary containment due to a break in the RCPB ~~or the refueling floor due to a fuel handling accident~~. When Exhaust Radiation—High is detected, secondary containment isolation and actuation of the SGT System are initiated to limit the release of fission products as assumed in the UFSAR safety analyses (Ref~~s~~. 1 ~~and 2~~).

Secondary containment isolation is not credited for the fuel handling accident.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3, 4. Reactor Building Ventilation Exhaust Plenum and Fuel Pool Ventilation Exhaust Radiation—High (continued)

Reactor Building Ventilation Exhaust Plenum Radiation—High signals are initiated from radiation detectors that are located in the reactor building return air riser above the upper area of the steam tunnel prior to the reactor building ventilation isolation dampers. Fuel Pool Ventilation Exhaust Radiation—High signals are initiated from radiation detectors that are located in the reactor building exhaust ducting coming from the refuel floor. The signal from each detector is input to an individual monitor whose trip outputs are assigned to an isolation channel. Four channels of Reactor Building Ventilation Exhaust Plenum Radiation-High Function and four channels of Fuel Pool 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.

RECENTLY IRRADIATED FUEL

The Reactor Building Ventilation Exhaust Plenum and Fuel Pool 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 required to be OPERABLE during ~~CORE ALTERATIONS,~~ OPDRVs~~,~~ and movement of ~~irradiated fuel~~ assemblies in the secondary containment because the capability of detecting radiation releases due to fuel failures (due to fuel uncovery or dropped fuel assemblies) must be provided to ensure that offsite dose limits are not exceeded.

INSERT 3.3.6.2-1

5. Manual Initiation

The Manual Initiation push button channels introduce signals into the secondary containment isolation logic that are redundant to the automatic protective instrumentation channels, and provide manual isolation capability. There is

(continued)

BASES

ACTIONS

A.1 (continued)

Functions that have channel components common to RPS instrumentation and 24 hours for those Functions that do not have channel components common to RPS instrumentation), has been shown to be acceptable (Refs. ~~3 and 4~~ 2 and 3) to permit restoration of any inoperable channel to OPERABLE status. This out of service time is only acceptable provided the associated Function is still maintaining isolation capability (refer to Required Action B.1 Bases). If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action A.1. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an isolation), Condition C must be entered and its Required Actions taken.

B.1

Required Action B.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in a complete loss of automatic isolation capability for the associated penetration flow path(s) or a complete loss of automatic initiation capability for the SGT System. A Function is considered to be maintaining isolation capability when sufficient channels are OPERABLE or in trip, such that one trip system will generate a trip signal from the given Function on a valid signal. This ensures that one of the two SCIVs in the associated penetration flow path and the SGT subsystems can be initiated on an isolation signal from the given Function. For the Functions with two two-out-of-two logic trip systems (Functions 1, 2, 3, and 4), this would require one trip system to have two channels, each OPERABLE or in trip. The Condition does not include the Manual Initiation Function (Function 5), since it is not assumed in any accident or transient analysis. Thus, a total loss of manual initiation capability for 24 hours (as allowed by Required Action A.1) is allowed.

(continued)

BASES

ACTIONS

C.1.1, C.1.2, C.2.1, and C.2.2 (continued)

One hour is sufficient for plant operations personnel to establish required plant conditions or to declare the associated components inoperable without challenging plant systems.

SURVEILLANCE REQUIREMENTS

As noted at the beginning of the SRs, the SRs for each Secondary Containment Isolation instrumentation Function are located in the SRs column of Table 3.3.6.2-1.

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

This Note is based on the reliability analysis (Refs. ~~3~~ 2 and 3 ~~and 4~~) assumption of the average time required to perform channel surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the SCIVs will isolate the associated penetration flow paths and the SGT System will initiate when necessary.

SR 3.3.6.2.1

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.6.2.1 (continued)

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

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the channels required by the LCO.

SR 3.3.6.2.2

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the channel will perform the intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

The Frequency of 92 days is based upon the reliability analysis of References 3 ~~and~~ 4.

2 and 3

SR 3.3.6.2.3

CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.6.2.3 (continued)

The Frequency is based upon the assumption of a 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.6.2.4

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required isolation logic for a specific channel. The system functional testing, performed on SCIVs and the SGT System in LCO 3.6.4.2 and LCO 3.6.4.3, respectively, overlaps this Surveillance to provide complete testing of the assumed safety function.

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

REFERENCES

1. UFSAR, Section 15.6.5.

~~2. UFSAR, Section 15.7.4.~~

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

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

BASES

BACKGROUND (continued)

that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs a CRAF System initiation signal to the initiation logic.

APPLICABLE SAFETY ANALYSES

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

CRAF System instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

High radiation at the intake ducts of the control room outside air intakes is an indication of possible gross failure of the fuel cladding. The release may have originated from the primary containment due to a break in the RCPB or the refueling floor due to a fuel handling accident. When control room air intake high radiation is detected, the associated CRAF subsystem is automatically initiated in the pressurization mode since this radiation release could result in radiation exposure to control room personnel.

The Control Room Air Intake Radiation—High Function consists of eight independent monitors, with four monitors associated with one CRAF subsystem and the other four monitors associated with the other CRAF subsystem. Each of the four monitors associated with a CRAF subsystem are arranged in two trip systems, with each trip system containing two radiation monitors. Eight channels of the Control Room Air Intake Radiation—High Function are available and required to be OPERABLE to ensure no single instrument failure can preclude CRAF System initiation. The Allowable Value was selected to ensure protection of the control room personnel.

(continued)

BASES

LCO (continued)

Each channel must have its setpoint set within the specified Allowable Value of SR 3.3.7.1.3. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions. Nominal trip setpoints are specified in the setpoint calculations. These nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint that is less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., control room air intake radiation), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The trip setpoints are determined from the analytic limits, corrected for defined process, calibration, and instrument errors. The Allowable Values are then determined, based on the trip setpoint values, by accounting for the calibration based errors. These calibration based errors are limited to reference accuracy, instrument drift, errors associated with measurement and test equipment, and calibration tolerance of loop components. The trip setpoints and Allowable Values determined in this manner provide adequate protection because instrument uncertainties, process effects, calibration tolerances, instrument drift, and severe environment errors (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for and appropriately applied for the instrumentation.

RECENTLY IRRADIATED FUEL

APPLICABILITY

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~~ in the secondary containment to ensure that control room personnel are protected during a LOCA, fuel handling event, or a 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.

OPDRVs

(continued)

INSERT 3.3.7.1-1

BASES

BACKGROUND (continued)

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

APPLICABLE SAFETY ANALYSES

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

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

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

1.0%

The maximum allowable leakage rate for the primary containment (L_a) is ~~0.635%~~ by weight of the containment air per 24 hours at the design basis LOCA maximum peak containment pressure (P_a) of 39.9 psig (Ref. 4).

Primary containment satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Primary containment OPERABILITY is maintained by limiting leakage to ≤ 1.0 L_a, except prior to the first startup after performing a required Primary Containment Leakage Rate Testing Program leakage test. At this time, the applicable leakage limits must be met. In addition, the leakage from the drywell to the suppression chamber must be limited to ensure the primary containment pressure does not exceed

(continued)

BASES

BACKGROUND (continued)

containment leakage rate to within limits in the event of a DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the safety analysis.

APPLICABLE SAFETY ANALYSES

The DBA that postulates the maximum release of radioactive material within primary containment is a LOCA. In the analysis of this accident, it is assumed that primary containment is OPERABLE, such that release of fission products to the environment is controlled by the rate of primary containment leakage. The primary containment is designed with a maximum allowable leakage rate (L_a) of 1.0% ~~0.635%~~ by weight of the containment air mass per 24 hours at the Design Basis LOCA maximum peak containment pressure (P_a) of 39.9 psig (Ref. 2). This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air lock.

Primary containment air lock OPERABILITY is also required to minimize the amount of fission product gases that may escape primary containment through the air lock and contaminate and pressurize the secondary containment.

Primary containment air lock satisfies Criterion 3 of the 10 CFR 50.36(c)(2)(ii).

LCO

As part of the primary containment pressure boundary, the air lock safety function is related to control of containment leakage following a DBA. Thus, the air lock structural integrity and leak tightness are essential to the successful mitigation of such an event.

The primary containment air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door to be open at a time. This provision ensures that a gross breach of primary containment does not exist when primary containment is required to be OPERABLE. Closure of a single door in the air lock is sufficient to

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.10

200

The analyses in Reference 2 are based on leakage that is less than the specified leakage rate. Leakage through any one main steam line must be ≤ 100 scfh and through all four main steam lines must be ≤ 400 scfh when tested at P_t (25.0 psig). This ensures that MSIV leakage is properly accounted for in determining the overall primary containment leakage rate. The Frequency is required by the Primary Containment Leakage Rate Testing Program.

SR 3.6.1.3.11

Surveillance of hydrostatically tested lines provides assurance that the calculation assumptions of Reference 2 are met. The acceptance criteria for the combined leakage of all hydrostatically tested lines is 1 gpm times the total number of hydrostatically tested PCIVs when tested at ≥ 1.1 P_a, or other acceptable criteria based upon satisfying the acceptance criteria of 10 CFR 100, regarding the site radiological analysis. The combined leakage rates must be demonstrated in accordance with the leakage test Frequency required by the Primary Containment Leakage Rate Testing Program.

50.67

REFERENCES

1. Technical Requirements Manual.
2. UFSAR, Section 15.6.5.
3. UFSAR, Section 15.6.4.
4. UFSAR, Section 15.2.4.
5. UFSAR, Section 6.2.4.2.3.
6. NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation," June 2000

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.1 Secondary Containment

BASES

BACKGROUND	The function of the secondary containment is to contain dilute, and hold up fission products that may leak from primary containment following a Design Basis Accident (DBA). In conjunction with operation of the Standby Gas Treatment (SGT) System and closure of certain valves whose lines penetrate the secondary containment, the secondary containment is designed to reduce the activity level of the fission products prior to release to the environment and to isolate and contain fission products that are released during certain operations that take place inside primary containment, when primary containment is not required to be OPERABLE, or that take place outside primary containment. The secondary containment is a structure that completely encloses the primary containment and those components that may be postulated to contain primary system fluid. This structure forms a control volume that serves to hold up and dilute the fission products. It is possible for the pressure in the control volume to rise relative to the environmental pressure (e.g., due to pump/motor heat load additions). To prevent ground level exfiltration while allowing the secondary containment to be designed as a conventional structure, the secondary containment requires support systems to maintain the control volume pressure at less than the external pressure. Requirements for these systems are specified separately in LCO 3.6.4.2, "Secondary Containment Isolation Valves (SCIVs)," and LCO 3.6.4.3, "Standby Gas Treatment (SGT) System."
APPLICABLE SAFETY ANALYSES this	There are two principal accidents for which credit is is the taken for secondary containment OPERABILITY. These are a LOCA (Ref. 1) and a fuel handling accident (Ref. 2). The secondary containment performs no active function in response to each of these limiting events; however, its leak tightness is required to ensure that the release of radioactive materials from the primary containment is restricted to those leakage paths and associated leakage rates assumed in the accident analysis, and that fission products entrapped within the secondary containment

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

structure will be treated by the SGT System prior to discharge to the environment.

Secondary containment satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

An OPERABLE secondary containment provides a control volume into which fission products that bypass or leak from primary containment, or are released from the reactor coolant pressure boundary components located in secondary containment, can be diluted and processed prior to release to the environment. For the secondary containment to be considered OPERABLE, it must have adequate leak tightness to ensure that the required vacuum can be established and maintained, the hatches and blowout panels must be closed and sealed, the sealing mechanisms associated with each secondary containment penetration (e.g., welds, bellows, or O-rings) must be OPERABLE (such that secondary containment leak tightness can be maintained), and all inner or all outer doors in each secondary containment access opening must be closed.

INSERT 3.6.4.1-2

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~~ RECENTLY IRRADIATED FUEL assemblies in the secondary containment.

INSERT 3.6.4.1-1

(continued)

BASES (continued)

ACTIONS

A.1

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

B.1

If the secondary containment cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which overall plant risk is minimized. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours. Remaining in the Applicability of the LCO is acceptable because the plant risk in MODE 3 is similar to or lower than the risk in MODE 4 (Ref. ~~3~~ 2), because the time spent in MODE 3 to perform the necessary repairs to restore the system to OPERABLE status will be short. However, voluntary entry into MODE 4 may be made as it is also an acceptable low-risk state. The allowed Completion Time is 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 ~~irradiated fuel~~ 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 ~~irradiated fuel~~ 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.

(continued)

BASES

ACTIONS

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

and C.2

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.

RECENTLY IRRADIATED FUEL

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1.1

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

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

SR 3.6.4.1.2 and SR 3.6.4.1.5

Verifying that one secondary containment access door in each access opening is closed and each equipment hatch is closed and sealed 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. In addition, for equipment hatches that are floor plugs, the "sealed" requirement is effectively met by gravity. 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 a secondary containment barrier contains multiple inner or multiple outer doors. For these cases, the access openings share the inner door or the outer door, i.e., the access openings have

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1.2 and SR 3.6.4.1.5 (continued)

a common inner door or outer door. The intent is to not breach the 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, i.e., all inner doors closed or all outer doors closed. Thus each access opening has one door closed. However, each secondary containment access door is normally kept closed, except when the access opening is being used for entry and exit or when maintenance is being performed on the access opening. The 31 day Frequency for SR 3.6.4.1.2 has been shown to be adequate based on operating experience, and is considered adequate in view of the existing administrative controls on door status. The 24 month Frequency for SR 3.6.4.1.5 is considered adequate in view of the existing administrative controls on equipment hatches.

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 appropriate treatment equipment. Each SGT subsystem is designed to drawdown pressure in the secondary containment to ≥ 0.25 inches of vacuum water gauge in ≤ 300 seconds and maintain pressure in the secondary containment at ≥ 0.25 inches of vacuum water gauge for 1 hour at a flow rate of ≤ 4400 cfm. To ensure that all fission products released to secondary containment are treated, SR 3.6.4.1.3 and SR 3.6.4.1.4 verify that a pressure in the secondary containment that is less than the pressure external to the secondary containment boundary can rapidly be established and maintained. When the SGT System is operating as designed, the establishment and maintenance of secondary containment pressure cannot be accomplished if the secondary containment boundary is not intact. Establishment of this pressure is confirmed by SR 3.6.4.1.3, which demonstrates that the secondary containment can be drawn down to ≥ 0.25 inches of vacuum water gauge in

900

≤ 300 seconds using one SGT subsystem. SR 3.6.4.1.4 demonstrates that the pressure in the secondary containment can be maintained ≥ 0.25 inches of vacuum water gauge for 1 hour using one SGT subsystem at a flow rate ≤ 4400 cfm. This flow rate is the assumed secondary containment leak rate during the drawdown period. The 1 hour test period allows secondary containment to be in thermal equilibrium at

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1.3 and SR 3.6.4.1.4 (continued)

steady state conditions. The primary purpose of the SRs is to ensure secondary containment boundary integrity. The secondary purpose of these SRs is to ensure that the SGT subsystem being tested functions as designed. There is a separate LCO with Surveillance Requirements that serves the primary purpose of ensuring OPERABILITY of the SGT System. These SRs need not be performed with each SGT subsystem. The SGT subsystem used for these Surveillances is staggered to ensure that in addition to the requirements of LCO 3.6.4.3, either SGT subsystem will perform this test. The inoperability of the SGT System does not necessarily constitute a failure of these Surveillances relative to secondary containment OPERABILITY. Operating experience has shown the secondary containment boundary usually passes these Surveillances when performed at the 24 month Frequency. Therefore the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 15.6.5.
2. UFSAR, Section 15.7.4.

2. ~~3.~~ NEDC-32988-A, Revision 2, "Technical Justification to Support Risk-Informed Modification to Selected Required End States for BWR Plants," December 2002.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.2 Secondary Containment Isolation Valves (SCIVs)

BASES

BACKGROUND

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

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

Automatic SCIVs (i.e., dampers) 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 required to be closed during accident conditions 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 a~~ is loss of coolant accident (Ref. 1) ~~and fuel handling accident (Ref. 2)~~. The secondary containment performs no active function in response to ~~each of these~~ this limiting event~~s~~, but

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

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

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

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, automatic 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 the Technical Requirements Manual (Ref. ~~3~~). 2

The normally closed manual SCIVs are considered OPERABLE when the valves are closed and blind flanges are in place, or open under administrative controls. 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, OPERABILITY of SCIVs is required.

RECENTLY IRRADIATED FUEL

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 releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs)~~, during CORE ALTERATIONS,~~ or during movement of ~~irradiated fuel~~ assemblies in the secondary containment.

(continued)

INSERT 3.6.4.2-1

BASES

ACTIONS

B.1 (continued)

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

C.1 and C.2

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

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

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

this activity

RECENTLY IRRADIATED FUEL

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 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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.4.2.2

Verifying the isolation time of each power operated, automatic SCIV is within limits is required to demonstrate OPERABILITY. The isolation time test ensures that the SCIV will isolate in a time period less than or equal to that assumed in the safety analyses. The Frequency of this SR is 92 days.

SR 3.6.4.2.3

Verifying that each automatic SCIV closes on a secondary containment isolation signal is required to prevent leakage of radioactive material from secondary containment following a DBA or other accidents. This SR ensures that each automatic SCIV will actuate to the isolation position on a secondary containment isolation signal. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.6.2, "Secondary Containment Isolation Instrumentation," overlaps this SR to provide complete testing of the safety function. While this Surveillance can be performed with the reactor at power, operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 15.6.5.

2. ~~UFSAR, Section 15.7.4.~~

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

BASES

BACKGROUND (continued)

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

The SGT System automatically starts and operates in response to actuation signals from either Unit 1 or Unit 2 indicative of conditions or an accident that could require operation of the system. Following initiation, both supply fans start. SGT System flows are controlled automatically by flow control dampers located up stream of the supply fans.

APPLICABLE SAFETY ANALYSES

The design basis for the SGT System is to mitigate the consequences of a loss of coolant this accident ~~and fuel handling accidents~~ (Ref~~s~~. 3 ~~and 4~~). For ~~all events~~ analyzed event, 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 10 CFR 50.36(c)(2)(ii).

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.

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 OPERABLE is not required in MODE 4 or 5, except for

(continued)

BASES

APPLICABILITY (continued)

other situations under which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs)~~, during CORE ALTERATIONS,~~ or during movement of ~~irradiated fuel~~ RECENTLY IRRADIATED FUEL assemblies in the secondary containment. INSERT 3.6.4.3-1

ACTIONS

A.1

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

B.1

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 overall plant risk is minimized. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours. Remaining in the Applicability of the LCO is acceptable because the plant risk in MODE 3 is similar to or lower than the risk in MODE 4 (Ref. ~~5~~ 4) and because the time spent in MODE 3 to perform the necessary repairs to restore the system to OPERABLE status will be short. However, voluntary entry into MODE 4 may be made as it is also an acceptable low-risk state. The allowed Completion Time is 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 ~~irradiated fuel~~ 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

(continued)

BASES

ACTIONS

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

should be immediately placed in operation. This Required Action ensures that the remaining subsystem is OPERABLE, that no failures that could prevent automatic actuation will occur, and that any other failure would be readily detected.

a significant amount of

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

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.

RECENTLY IRRADIATED FUEL

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, the plant must be brought to a MODE in which the overall plant risk is minimized. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours. Remaining in the Applicability of the LCO is acceptable because the plant risk in MODE 3 is similar to or lower than the risk in MODE 4 (Ref. 3) and because the time spent in MODE 3 to perform the necessary repairs to restore the system to OPERABLE status will be short. However, voluntary entry into MODE 4 may be made as it is also an acceptable low-risk state. The allowed Completion Time is reasonable, based on operating experience,

4

(continued)

BASES

ACTIONS

D.1 (continued)

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

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

and E.2

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

RECENTLY IRRADIATED FUEL

this activity

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.4.3.2

This SR verifies that the required SGT filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The SGT System filter tests are in accordance with ANSI/ASME N510-1989 (Ref. ~~6~~ 5). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specified test frequencies and additional information are discussed in detail in the VFTP.

SR 3.6.4.3.3

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

REFERENCES

1. 10 CFR 50, Appendix A, GDC 41.

2. UFSAR, Section 6.5.1.

3. UFSAR, Section 15.6.5.

4. ~~UFSAR, Section 15.7.4~~

4. ~~5.~~ NEDC-32988-A, Revision 2, "Technical Justification to Support Risk-Informed Modification to Selected Required End States for BWR Plants," December 2002.

5. ~~6.~~ ANSI/ASME N510-1989.

BASES

BACKGROUND (continued)

an activated charcoal adsorber section, a second HEPA filter, a fan, and the associated ductwork, dampers, doors, barriers, and instrumentation and controls. Demisters remove water droplets from the airstream. The electric heater reduces the relative humidity of the air entering the EMUs. Prefilters and HEPA filters remove particulate matter that may be radioactive. The charcoal adsorbers provide a holdup period for gaseous iodine, allowing time for decay. Each Control Room and AEER Ventilation System has a charcoal recirculation filter in the supply of the system that is normally bypassed. In addition, the OPERABILITY of the CRAF System is dependent upon portions of the Control Room Area HVAC System, including the control room and auxiliary electric equipment room outside air intakes, supply fans, ducts, dampers, etc.

In addition to the safety related standby emergency filtration function, parts of the CRAF System that are shared with the Control Room Area HVAC System are operated to maintain the CRE environment during normal operation. Upon receipt of a high radiation signal from the outside air intake (indicative of conditions that could result in radiation exposure to CRE occupants), the CRAF System automatically isolates the normal outside air supply to the Control Room Area HVAC System, and diverts the minimum outside air requirement through the EMUs before delivering it to the CRE. The recirculation filters for the control room and AEER must be manually placed in service within 4 hours of receipt of any control room high radiation alarm.

The CRAF System is designed to maintain a habitable environment in the CRE for a 30 day continuous occupancy after a DBA, without exceeding a 5 rem TEDE ~~whole body~~ dose ~~or its equivalent to any part of the body~~. CRAF System operation in maintaining the CRE habitability is discussed in the UFSAR, Sections 6.4, 6.5.1, and 9.4.1 (Refs. 1, 2, and 3, respectively).

APPLICABLE SAFETY ANALYSES

The ability of the CRAF System to maintain the habitability of the CRE is an explicit assumption for the safety analyses presented in the UFSAR, Chapters 6 and 15 (Refs. 4 and 5, respectively). The pressurization mode of the CRAF System is assumed to operate following a DBA. The

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

radiological doses to CRE occupants as a result of the various DBAs are summarized in Reference 5. No single active failure will cause the loss of outside or recirculated air from the CRE.

The CRAF System provides protection from smoke and hazardous chemicals to the CRE occupants. The analysis of hazardous chemical releases demonstrates that the toxicity limits are not exceeded in the CRE following a hazardous chemical release (Ref. 1). The evaluation of a smoke challenge demonstrates that it will not result in the inability of the CRE occupants to control the reactor either from the control room or from the remote shutdown panels (Ref. 3).

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

LCO

Two redundant subsystems of the CRAF System are required to be OPERABLE to ensure that at least one is available, if a single active failure disables the other subsystem. Total CRAF System failure, such as from a loss of both ventilation subsystems or from an inoperable CRE boundary, could result in exceeding a dose of 5 rem ~~whole body or its equivalent to any part of the body~~ TEDE to the CRE occupants in the event of a DBA.

Each CRAF subsystem is considered OPERABLE when the individual components necessary to limit CRE occupant exposure are OPERABLE. A subsystem is considered OPERABLE when its associated EMU is OPERABLE and the associated charcoal recirculation filters for the control room and AEER are OPERABLE. An EMU is considered OPERABLE when its associated:

a. Fan is OPERABLE;

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

c. Heater, demister, ductwork, valves, and dampers are OPERABLE, and air circulation through the EMU can be maintained.

(continued)

BASES

LCO (continued)

Additionally, the portions of the Control Room Area HVAC System that supply the outside air to the EMUs are required to be OPERABLE. This includes the outside air intakes, associated dampers and ductwork.

In order for the CRAF subsystems to be considered OPERABLE, the CRE boundary must be maintained such that the CRE occupant dose from a large radioactive release does not exceed the calculated dose in the licensing basis consequence analysis for DBAs, and that CRE occupants are protected from hazardous chemicals and smoke.

The LCO is modified by a Note allowing the CRE boundary to be opened intermittently under administrative controls. This Note only applies to openings in the CRE boundary that can be rapidly restored to the design condition, such as doors, hatches, floor plugs, and access panels. For entry and exit through doors, the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls should be proceduralized and consist of stationing a dedicated individual at the opening who is in continuous communication with the operators in the CRE. This individual will have a method to rapidly close the opening and to restore the CRE boundary to a condition equivalent to the design condition when a need for the CRAF System to be in the pressurization mode of operation is indicated.

APPLICABILITY

In MODES 1, 2, and 3, the CRAF System must be OPERABLE to ensure that the CRE will remain habitable during and following a DBA, since the DBA could lead to a fission product release.

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

RECENTLY IRRADIATED FUEL

a. During movement of ~~irradiated fuel~~ assemblies in the secondary containment; and

(continued)

BASES

APPLICABILITY (continued)

~~b. During CORE ALTERATIONS; and~~

b. ~~c.~~ During operations with a potential for draining the reactor vessel (OPDRVs).

INSERT 3.7.4-1

ACTIONS

A.1

With one CRAF subsystem inoperable, for reasons other than an inoperable CRE boundary, the inoperable CRAF subsystem must be restored to OPERABLE status within 7 days. With the unit in this condition, the remaining OPERABLE CRAF subsystem is adequate to perform the CRE occupant protection function. However, the overall reliability is reduced because a failure in the OPERABLE subsystem could result in loss of CRAF System function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and that the remaining subsystem can provide the required capabilities.

B.1, B.2 and B.3

If the unfiltered inleakage of potentially contaminated air past the CRE boundary and into the CRE can result in CRE occupant radiological dose greater than the calculated dose of the licensing basis analyses of DBA consequences (allowed to be up to 5 rem TEDE ~~rem whole body or its equivalent to any part of the body~~), or inadequate protection of CRE occupants from hazardous chemicals or smoke, the CRE boundary is inoperable. Actions must be taken to restore an OPERABLE CRE boundary within 90 days.

During the period that the CRE boundary is considered inoperable, action must be initiated to implement mitigating actions to lessen the effect on CRE occupants from the potential hazards of a radiological or chemical event or a challenge from smoke. Actions must be taken within 24 hours to verify that in the event of a DBA, the mitigating actions will ensure that CRE occupant radiological exposures will not exceed the calculated dose of the licensing basis analyses of DBA consequences, and that CRE occupants are protected from hazardous chemicals and smoke. These mitigating actions (i.e., actions that are taken to offset the consequences of the inoperable CRE boundary) should be preplanned for implantation upon entry into the condition, regardless of whether entry is intentional or unintentional.

(continued)

BASES

ACTIONS

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

The 24 hour completion time is reasonable based on the low probability of a DBA occurring during this time period, and the use of mitigating actions. The 90 day Completion Time is reasonable based on the determination that the mitigating actions will ensure protection of CRE occupants within analyzed limits while limiting the probability that CRE occupants will have to implement protective measures that may adversely affect their ability to control the reactor and maintain it in a safe shutdown condition in the event of a DBA. In addition, the 90 day Completion Time is a reasonable time to diagnose, plan and possibly repair, and test most problems with the CRE boundary.

C.1

In MODE 1, 2, or 3, if the inoperable CRAF subsystem or the CRE boundary cannot be restored to OPERABLE status within the required Completion Time, the unit must be placed in a MODE that minimizes overall plant risk. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours. Remaining in the Applicability of the LCO is acceptable because the plant risk in MODE 3 is similar to or lower than the risk in MODE 4 (Ref. 6) and because the time spent in MODE 3 to perform the necessary repairs to restore the system to OPERABLE status will be short. However, voluntary entry into MODE 4 may be made as it is also an acceptable low-risk state. The allowed Completion Time is reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

and D.2.2

RECENTLY IRRADIATED FUEL

D.1, D.2.1, D.2.2, and D.2.3

LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since irradiated fuel assembly movement can occur in MODE 1, 2, or 3, the Required Actions of Condition D 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. Entering LCO 3.0.3 while in MODE 1, 2, or 3 would require

(continued)

BASES

ACTIONS

D.1, D.2.1, D.2.2, and D.2.3 (continued) and D.2.2

the unit to be shutdown, but would not require immediate suspension of movement of irradiated fuel assemblies. The Note to the ACTIONS, "LCO 3.0.3 is not applicable," ensures that the actions for immediate suspension of irradiated fuel assembly movement are not postponed due to entry into LCO 3.0.3.

RECENTLY IRRADIATED FUEL

During movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs, if the inoperable CRAF subsystem cannot be restored to OPERABLE status within the required Completion Time, the OPERABLE CRAF subsystem may be placed in the pressurization mode. 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 D.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require the CRAF System to be in the pressurization mode of operation. This places the unit in a condition that minimizes the accident risk.

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

this activity

E.1

If both CRAF subsystems are inoperable in MODE 1, 2, or 3, for reasons other than an inoperable CRE boundary (i.e., Condition B), the CRAF System may not be capable of performing the intended function. Therefore, the plant must be brought to a MODE in which the overall plant risk is minimized. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours. Remaining in the Applicability of the LCO is acceptable because the plant risk in MODE 3 is similar to or lower than the risk in

(continued)

BASES

ACTIONS

E.1 (continued)

MODE 4 (Ref. 6) and because the time spent in MODE 3 to perform the necessary repairs to restore the system to OPERABLE status will be short. However, voluntary entry into MODE 4 may be made as it is also an acceptable low-risk state. The allowed Completion Time is reasonable, based on operating experience, to reach the required plant conditions from full power conditions is an orderly manner and without challenging plant systems.

F.1, F.2 and F.3

and F.2

RECENTLY IRRADIATED FUEL

LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since irradiated fuel assembly movement can occur in MODE 1, 2, or 3, the Required Actions of Condition F 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. Entering LCO 3.0.3 while in MODE 1, 2, or 3 would require the unit to be shutdown, but would not require immediate suspension of movement of irradiated fuel assemblies. The Note to the ACTIONS, "LCO 3.0.3 is not applicable," ensures that the actions for immediate suspension of irradiated fuel assembly movement are not postponed due to entry into LCO 3.0.3.

During movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs, with two CRAF subsystems inoperable, or with one or more CRAF subsystems inoperable due to an inoperable CRE boundary, action must be taken immediately to suspend activities that present a potential for releasing radioactivity that might require the CRAF System to be in the pressurization mode of operation. This places the unit in a condition that minimizes the accident risk.

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

this activity

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.4.4 (continued)

signal. 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 normally pass the SR when performed at the 24 month Frequency. Therefore, the Frequency was found to be acceptable from a reliability standpoint.

SR 3.7.4.5

This SR verifies the OPERBILITY of the CRE boundary by testing for unfiltered air inleakage past the CRE boundary and into the CRE. The details of the testing are specified in the Control Room Envelope Habitability Program.

TEDE

The CRE is considered habitable when the radiological dose to CRE occupants calculated in the licensing basis analyses of DBA consequences is no more than 5 rem ~~whole body or its equivalent to any part of the body~~ and the CRE occupants are protected from hazardous chemicals and smoke. This SR verifies that the unfiltered air inleakage into the CRE is no greater than the flow rates assumed in the licensing basis analyses of DBA consequences. When the unfiltered air inleakage is greater than the assumed flow rate, Condition B must be entered. Required Action B.3 allows time to restore the CRE boundary to OPERABLE status provided mitigating actions can ensure that the CRE remains within the licensing basis habitability limits for the occupants following an accident. Compensatory measures are discussed in Regulatory Guide 1.196, Section C.2.7.3 (Ref. 8), which endorses, with exceptions, NEI 99-03, Section 8.4 and Appendix F (Ref. 9). These compensatory measures may also be used as mitigating actions as required by Required Action B.2. Temporary analytical methods may also be used as compensatory measures to restore OPERABILITY (Ref. 10). Options for restoring the CRE boundary to OPERABLE status include changing the licensing basis DBA consequence analysis, repairing the CRE boundary, or a combination of these actions. Depending upon the nature of the problem and the corrective action, a full scope inleakage test may not be necessary to establish that the CRE boundary has been restored to OPERABLE status.

(continued)

BASES

LCO (continued)

The Control Room Area Ventilation AC System is considered OPERABLE when the individual components necessary to maintain the control room and AEERs temperatures are OPERABLE in both subsystems. These components include the supply and return air fans, direct expansion cooling coils, an air-cooled condenser, a refrigerant compressor and receiver, ductwork, dampers, and instrumentation and controls.

APPLICABILITY

In MODE 1, 2, or 3, the Control Room Area Ventilation AC System must be OPERABLE to ensure that the control room and AEERs temperatures will not exceed equipment OPERABILITY limits during operation of the Control Room Area Filtration (CRAF) System in the pressurization mode.

RECENTLY IRRADIATED FUEL

In MODES 4 and 5, the probability and consequences of a Design Basis Accident are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the Control Room Area Ventilation AC 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 movement of ~~irradiated fuel~~ assemblies in the secondary containment; and

~~b. During CORE ALTERATIONS; and~~

b. ~~c.~~ During operations with a potential for draining the reactor vessel (OPDRVs).

INSERT 3.7.5-1

ACTIONS

A.1

With one control room area ventilation AC subsystem inoperable, the inoperable control room area ventilation AC subsystem must be restored to OPERABLE status within 30 days. With the unit in this condition, the remaining OPERABLE control room area ventilation AC subsystem is adequate to perform the control room air conditioning function. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in loss of the control room area ventilation air conditioning function. The 30 day Completion Time is based

(continued)

BASES

ACTIONS (continued)

D.1, D.2.1, D.2.2, and D.2.3

and D.2.2

RECENTLY IRRADIATED FUEL

LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since irradiated fuel assembly movement can occur in MODE 1, 2, or 3, the Required Actions of Condition D 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. Entering LCO 3.0.3 while in MODE 1, 2, or 3 would require the unit to be shutdown, but would not require immediate suspension of movement of irradiated fuel assemblies. The Note to the ACTIONS, "LCO 3.0.3 is not applicable," ensures that the actions for immediate suspension of irradiated fuel assembly movement are not postponed due to entry into LCO 3.0.3.

During movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs, if Required Action A.1 cannot be completed within the required Completion Time, the OPERABLE control room AC subsystem may be placed immediately in operation.

This action ensures that the remaining subsystem is OPERABLE, that no failures that would prevent actuation will occur, and that any active failure will be readily detected.

An alternative to Required Action D.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.

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. Also, if applicable, action must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Action must continue until the OPDRVs are suspended.

(continued)

BASES

ACTIONS (continued)

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

The Required Actions of Condition E.1 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 IRRADIATED FUEL

During movement of irradiated fuel assemblies in the secondary containment~~, during CORE ALTERATIONS,~~ or during OPDRVs if Required Actions B.1 and B.2 cannot be met within the required Completion Times action must be taken 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.

this activity

If applicable, ~~CORE ALTERATIONS and~~ handling of irradiated fuel 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, action must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Action must continue until the OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

This SR monitors the control room and AEER temperatures for indication of Control Room Area Ventilation AC System performance. Trending of control room area temperature will provide a qualitative assessment of refrigeration unit OPERABILITY. Limiting the average temperature of the Control Room and AEER to less than or equal to 85°F provides a threshold beyond which the operating control room area ventilation AC subsystem is no longer demonstrating capability to perform its function. This threshold provides margin to temperature limits at which equipment qualification requirements could be challenged. Subsystem operation is routinely alternated to support planned maintenance and to ensure each subsystem provides reliable service. The 12 hour Frequency is adequate considering the continuous manning of the control room by the operating staff.

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.8 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 9.1.2 (Ref. 1). The assumptions of the fuel handling accident are found in the UFSAR, Sections 9.1.2 and 15.7.4 (Refs. 1 and 2, respectively).

APPLICABLE SAFETY ANALYSES

The water level above the irradiated fuel assemblies is an explicit assumption of the fuel handling accident (Ref. 2). A fuel handling accident is evaluated to ensure that the radiological consequences (calculated whole body and thyroid doses at the exclusion area and low population zone boundaries) are ≤ 25% (NUREG-0800, Section 15.7.4, Ref. 3) of the 10 CFR 100 (Ref. 4) exposure guidelines. A fuel handling accident could release a fraction of the fission product inventory by breaching the fuel rod cladding as discussed in the Regulatory Guide ~~1.25~~ 1.183 (Ref. ~~5~~ 4).

INSERT 3.7.8-1

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 (Ref. 2). The water level in the spent fuel storage pool provides for absorption of water soluble fission product gases and transport delays of soluble and insoluble gases that must pass through the water before being released to the secondary containment atmosphere. This absorption and transport delay reduces the potential radioactivity of the release during a fuel handling accident.

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

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Section 9.1.2.

2. UFSAR, Section 15.7.4.

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

3. ~~5.~~ 10 CFR ~~100~~ 50.67.

4. Regulatory Guide 1.183, July 2000.

~~6. Regulatory Guide 1.25, March 1972.~~

B 3.9 REFUELING OPERATIONS

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

BASES

BACKGROUND

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

APPLICABLE SAFETY ANALYSES

During movement of irradiated fuel assemblies the water level in the RPV is an initial condition design parameter in the analysis of a fuel handling accident in containment postulated by Regulatory Guide ~~1.25~~ 1.183 (Ref. 1). A minimum water level of 23 ft ~~(Regulatory Position C.1.c of Ref. 1)~~ allows a decontamination factor of ~~100 (Regulatory Position C.1.g of Ref. 1)~~ 200 to be used in the accident analysis for iodine (Ref. 1). This relates to the assumption that ~~99%~~ 99.5% of the total iodine released from the pellet to cladding gap of all the ~~dropped~~ damaged fuel assembly rods is retained by the refueling cavity water. ~~The fuel pellet to cladding gap is assumed to contain 10% of the total fuel rod iodine inventory (Ref. 1).~~

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

coverage to ensure acceptable radiological consequences is specified from the RPV flange. Since the worst case event results in failed fuel assemblies seated in the core, as well as the dropped assembly, dropping an assembly on the RPV flange will result in reduced releases of fission gases.

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

LCO

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

APPLICABILITY

LCO 3.9.6 is applicable when moving irradiated fuel assemblies within the RPV. The LCO minimizes the possibility of a fuel handling accident in containment that is beyond the assumptions of the safety analysis. If irradiated fuel is not present within the RPV, there can be no significant radioactivity release as a result of a postulated fuel handling accident. Requirements for handling of new fuel assemblies or control rods (where water depth to the RPV flange is not of concern) are covered by LCO 3.9.7, "RPV Water Level - New Fuel or Control Rods." Requirements for fuel handling accidents in the spent fuel storage pool are covered by LCO 3.7.8, "Spent Fuel Storage Pool Water Level."

ACTIONS

<u>A.1</u>

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

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

Verification of a minimum water level of 22 ft above the top of the RPV flange ensures that the design basis for the postulated fuel handling accident analysis during refueling operations is met. Water at the required level limits the consequences of damaged fuel rods, which are postulated to result from a fuel handling accident in containment (Ref. 2).

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

REFERENCES

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

2. UFSAR, Section 15.7.4.

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

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

B 3.9 REFUELING OPERATIONS

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

BASES

BACKGROUND

The movement of new fuel assemblies or handling of control rods within the RPV when fuel assemblies seated within the reactor vessel are irradiated requires a minimum water level of 23 ft above the top of irradiated fuel assemblies seated within the RPV. During refueling, this maintains a sufficient water level above the irradiated fuel. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to ~~< 25% of~~ 10 CFR ~~100~~ 50.67 limits, as ~~provided~~ modified by the guidance of Reference ~~2~~ 1.

APPLICABLE SAFETY ANALYSES

During movement of new fuel assemblies or handling of control rods over irradiated fuel assemblies, the water level in the RPV is an initial condition design parameter in the analysis of a fuel handling accident in containment postulated by Regulatory Guide ~~1.25~~ 1.183 (Ref. 1). A minimum water level of 23 ft ~~(Regulatory Position C.1.c of Ref. 1)~~ allows a decontamination factor of ~~100~~ 200 ~~(Regulatory Position C.1.g of Ref. 1)~~ Ref. 1 to be used in the accident analysis for iodine. This relates to the assumption that ~~99%~~ 99.5% of the total iodine released from the pellet to cladding gap of all the ~~dropped~~ damaged fuel assembly rods is retained by the refueling cavity water. ~~The fuel pellet to cladding gap is assumed to contain 10% of the total fuel rod iodine inventory (Ref. 1).~~

Analysis of the fuel handling accident inside containment is described in Reference 2. With a minimum water level of 23 ft and a minimum decay time of 24 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water, and that offsite doses are maintained within allowable limits (Ref. ~~4~~ 3). The related assumptions include the worst case dropping of an irradiated fuel assembly onto the reactor core loaded with irradiated fuel assemblies.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

LCO

A minimum water level of 23 ft above the top of irradiated fuel assemblies seated within the RPV is required to ensure that the radiological consequences of a postulated fuel handling accident are within acceptable limits, as provided by the guidance of Reference 3. 1

APPLICABILITY

LCO 3.9.7 is applicable when moving new fuel assemblies or handling control rods (i.e., movement with other than the normal control rod drive) when irradiated fuel assemblies are seated within the RPV. The LCO minimizes the possibility of a fuel handling accident in containment that is beyond the assumptions of the safety analysis. If irradiated fuel is not present within the RPV, there can be no significant radioactivity release as a result of a postulated fuel handling accident. Requirements for fuel handling accidents in the spent fuel storage pool are covered by LCO 3.7.8, "Spent Fuel Storage Pool Water Level." Requirements for handling irradiated fuel over the RPV are covered by LCO 3.9.6, "Reactor Pressure Vessel (RPV) Water Level—Irradiated Fuel."

ACTIONS

A.1

If the water level is < 23 ft above the top of irradiated fuel assemblies seated within the RPV, all operations involving movement of new fuel assemblies and handling of control rods within the RPV shall be suspended immediately to ensure that a fuel handling accident cannot occur. The suspension of fuel movement and control rod handling shall not preclude completion of movement of a component to a safe position.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS	<u>SR 3.9.7.1</u> Verification of a minimum water level of 23 ft above the top of the irradiated fuel assemblies seated within the RPV ensures that the design basis for the postulated fuel handling accident analysis during refueling operations is met. Water at the required level limits the consequences of damaged fuel rods, which are postulated to result from a fuel handling accident in containment (Ref. 2). The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls on valve positions, which make significant unplanned level changes unlikely.

REFERENCES

1. Regulatory Guide ~~1.25, March 23, 1972~~ 1.183, July 2000.
2. UFSAR, Section 15.7.4.

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

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

ATTACHMENT 4
Summary of Regulatory Commitments

The following table identifies commitments made by Exelon Generation Company, LLC (EGC) in this document. Any other actions discussed in the submittal represent intended or planned actions. They are described to the NRC for the NRC's information and are not regulatory commitments.

COMMITMENT	COMMITTED DATE OR "OUTAGE"	COMMITMENT TYPE	
		ONE-TIME ACTION (Yes/No)	PROGRAMMATIC (Yes/No)
The guidelines of NUMARC 93-01, Revision 3, Section 11.3.6.5, will be included in the assessment of systems removed from service during movement of irradiated fuel: - During fuel handling/core alterations, the Standby Gas Treatment and Reactor Building Ventilation systems, and radiation monitor availability (as defined in NUMARC 91-06) will be assessed, with respect to filtration and monitoring of releases from the fuel. Following shutdown, radioactivity in the fuel decays away fairly rapidly. The basis of the Technical Specification operability amendment is the reduction in doses due to such decay. The goal of maintaining ventilation system and radiation monitor availability is to reduce doses even further below that provided by the natural decay. - A single normal or contingency method to promptly close primary or secondary containment penetrations will be developed. Such prompt methods need not completely block the penetration or be capable of resisting pressure. The purpose of the "prompt methods" mentioned above is to enable ventilation systems to draw the release from a postulated fuel handling accident in the proper direction such that it can be treated and monitored. Prompt in this context is defined as being accomplished within one hour.	Upon Implementation	No	Yes

ATTACHMENT 4
Summary of Regulatory Commitments

COMMITMENT	COMMITTED DATE OR "OUTAGE"	COMMITMENT TYPE	
		ONE-TIME ACTION (Yes/No)	PROGRAMMATIC (Yes/No)
Emergency Operating Procedure LGA-001, "RPV Control," will be revised to ensure that Standby Liquid Control (SLC) system injection is started from the boron solution storage tank during a design basis accident (DBA) loss-of-coolant accident (LOCA).	Upon Implementation	No	Yes
Emergency Operating Procedure LGA-001 will be revised to ensure no steps would terminate the injection during a DBA LOCA prior to emptying the SLC system boron solution storage tank (i.e., injection of the full content into the reactor pressure vessel).	Upon Implementation	No	Yes

ATTACHMENT 5
Regulatory Guide 1.183 Conformance Matrix

Table 1: Conformance with Regulatory Guide (RG) 1.183 Main Sections			
RG Section	**RG Position**	**LSCS 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. These source terms were evaluated at end-of-cycle and at beginning of cycle (100 effective full power days (EFPD), to achieve equilibrium) conditions with worst-case inventory used for the selected isotopes. This has been shown to be a conservative approach. The resulting values were converted to units of Ci/MWt. Accident analyses are based on a power level of 3559 MWt to account for two percent uncertainty (3489 x 1.02 = 3559).
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	For the DBA LOCA, all fuel assemblies in the core are assumed to be affected and the total core inventory is used. For the DBA FHA where the entire core is not affected, a radial peaking factor of 1.7 is applied in determining inventory of damaged rods.

<table>
<tr><th colspan="4">Table 1: Conformance with Regulatory Guide (RG) 1.183 Main Sections</th></tr>
<tr><th>RG Section</th><th>RG Position</th><th>LSCS Analysis</th><th>Comments</th></tr>
<tr><td>3.1</td><td>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.</td><td>Conforms</td><td>Fission product inventories used reflect full power operation plus two percent uncertainty. Radioactive decay from the time of shutdown is modeled to demonstrate and support a definition of "recently irradiated fuel" occurring after 24 hours.</td></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</td><td>Conforms</td><td>The fractions from Table 1 are used in the assessment of the DBA LOCA. The limitations of Footnote 10 are met.</td></tr>
</table>

Table 1: Conformance with Regulatory Guide (RG) 1.183 Main Sections			
RG Section	**RG Position**	**LSCS Analysis**	**Comments**
	burnup up to 62,000 MWD/MTU. The data in this section may not be applicable to core containing mixed oxide (MOX) fuel.		
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 burnups exceeding 54 GWD/MTU.	Conforms	The FHA calculation uses the fractions given in Table 3. The limitations of Footnote 11 are met.
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 is projected, the release from the fuel gap and the fuel pellet should be assumed to occur instantaneously	Conforms	The BWR durations from Table 4 are used. LOCA releases are modeled in a linear fashion using RADTRAD.

Table 1: Conformance with Regulatory Guide (RG) 1.183 Main Sections			
RG Section	**RG Position**	**LSCS Analysis**	**Comments**
	with the onset of the projected damage. **Table 4** **LOCA Release Phases** **PWRs** / **BWRs** **Phase** – PWRs **Onset**, **Duration** – BWRs **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	LSCS does not use leak-before-break methodology for DBA radiological 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	Conforms	NRC guidance on chemical

Table 1: Conformance with Regulatory Guide (RG) 1.183 Main Sections			
RG Section	RG Position	LSCS Analysis	Comments
	containment in a postulated accident, 95 percent of the iodine released should be assumed to be cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. This includes releases from the gap and the fuel pellets. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate 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.		forms for fission products is applied for all accidents as specified here and in RG 1.183 appendices.
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.	Not applicable to LOCA or FHA	
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	Conforms	Federal Guidance Report 11 dose conversion factors

Table 1: Conformance with Regulatory Guide (RG) 1.183 Main Sections			
RG Section	RG Position	LSCS Analysis	Comments
	for Intakes of Radionuclides by Workers" (Ref. 19). Table 2.1 of Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion" (Ref. 20), provides tables of conversion factors acceptable to the NRC staff. The factors in the column headed "effective" yield doses corresponding to the CEDE.		(DCFs) from the column headed "effective" are used.
4.1.3	For the first 8 hours, the breathing rate of persons offsite should be assumed to be 3.5×10^{-4} cubic meters per second. From 8 to 24 hours following the accident, the breathing rate should be assumed to be 1.8×10^{-4} cubic meters per second. After that and until the end of the accident, the rate should be assumed to be 2.3×10^{-4} cubic meters per second.	Conforms	The specified values are used in the analyses.
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 from the column headed "effective" 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	Conforms	The maximum two hour EAB dose value is determined by RADTRAD for each release path. For the LOCA calculation at LSCS, the maximum two hour period

Table 1: Conformance with Regulatory Guide (RG) 1.183 Main Sections			
RG Section	RG Position	LSCS Analysis	Comments
	"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).		effectively occurs beginning at time zero, because of the 15 minute drawdown period where no SGT is credited, which is followed by a period where SGT filtration is credited.
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	Analyses are based on X/Qs determined at the LPZ distance in conformance with Regulatory Guide 1.145.
4.1.7	No correction should be made for depletion of the effluent plume by deposition on the ground.	Conforms	No such credit is taken.
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	Conforms	The principal source of dose within the control room is due to airborne activity. For the LOCA analysis, sources that were historically addressed (UFSAR Table 6.4-2) have been reviewed, and the most significant contributors re-analyzed. These include activity accumulated on filters serving the CR and AEER; and radiation shine from activity plated out in the reactor building above the refuel floor. External Clouds are also re-evaluated to

Table 1: Conformance with Regulatory Guide (RG) 1.183 Main Sections			
RG Section	RG Position	LSCS Analysis	Comments
	components inside or external to the control room envelope, e.g., radioactive material buildup in recirculation filters.		reflect all sources. A total allowance of 0.04 REM conservatively envelopes these external source contributors.
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 assumptions are 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	RADTRAD analyses are used to evaluate transport of material into and through the control room, and to determine the resulting personnel doses. Shielding models are as discussed in Attachment 7.
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.	Exceptions taken, see comments	For the LOCA analysis, after the drawdown period, credit is taken for SGT HEPA and charcoal adsorber filtration (99% each), which includes an allowance for 0.5% filter bypass (higher than the 0.05% specified in RG 1.52). This system is automatically initiated and single failure-

Table 1: Conformance with Regulatory Guide (RG) 1.183 Main Sections			
RG Section	**RG Position**	**LSCS Analysis**	**Comments**
			proof. The CR and AEER recirculation filter trains are not designed to RG 1.52. However, TS 5.5.8.c currently credits the charcoal adsorbers at 70% efficiency. This efficiency is the established design basis and current licensing basis, and is used in the DBA analysis. No credit is taken for filtration in the DBA FHA analysis.
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 identified occupancy factors and breathing rate are used in dose analyses.
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 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,	Conforms	The equation given is utilized for finite cloud correction when calculating external doses due to the airborne activity inside the control room. This formula is also

Table 1: Conformance with Regulatory Guide (RG) 1.183 Main Sections			
RG Section	RG Position	LSCS Analysis	Comments
	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). $$DDE_{finite} = \frac{DDE_\infty V^{0.338}}{1173}$$		built into RADTRAD for use in control room dose assessments.
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	For the Technical Support Center and other areas requiring plant personnel access, assessments are contained in the LOCA analysis (i.e., Attachment 7). The radiation dose basis for environmental qualification is not changing.
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	Analyses are performed under quality assurance programs meeting Appendix B to 10 CFR Part 50.
5.1.2	Credit may be taken for accident mitigation features that are classified as safety-related, are required to be operable by technical specifications, are powered by emergency power sources, and are either automatically actuated or, in limited cases, have actuation requirements explicitly addressed in emergency operating procedures. The single active component failure that results in the most limiting radiological	Exceptions taken, see comments	The LOCA analysis generally relies on the same safety related accident mitigation features historically credited for LOCA analyses. Exceptions are discussed in

Table 1: Conformance with Regulatory Guide (RG) 1.183 Main Sections			
RG Section	RG Position	LSCS Analysis	Comments
	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.		Section 3.6 of Attachment 1. No credit is taken for mitigation factors for the FHA analysis.
5.1.3	The numeric values that are chosen as inputs to the analyses required by 10 CFR 50.67 should be selected with the objective of determining a conservative postulated dose. In some instances, a particular parameter may be conservative in one portion of an analysis but be nonconservative in another portion of the same analysis.	Conforms	Conservative assumptions are used. See input parameter tables within the LOCA and FHA calculation discussion sections for further information.
5.1.4	Licensees should ensure that analysis assumptions and methods are compatible with the AST and the TEDE criteria.	Conforms	As documented in the FHA and LOCA calculations, 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	Conforms	New atmospheric dispersion values (X/Q) for the EAB, the LPZ, and the control room were developed, using meteorology data for the years 1998 through 2003. ARCON96 and PAVAN were used with these data to determine control room and EAB/LPZ atmospheric dispersion coefficient values, respectively.

Table 1: Conformance with Regulatory Guide (RG) 1.183 Main Sections			
RG Section	RG Position	LSCS Analysis	Comments
	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.		Worst-case X/Qs for all releases are used. Review of pertinent drawings and site walkdowns have substantiated that there is no worse release pathway that could be expected to occur.

Table 2: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	LSCS 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	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

Table 2: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	LSCS Analysis	Comments
	particulate form.		event, the effects of key fission product releases, and the impact of SLC system injection. Suppression pool pH remains above 7 for at least 30 days following the LOCA.
3.1	The radioactivity released from the fuel should be assumed to mix instantaneously and homogeneously throughout the free air volume of the primary containment in PWRs or the drywell in BWRs as it is released. This distribution should be adjusted if there are internal compartments that have limited ventilation exchange. The suppression pool free air volume may be included provided there is a mechanism to ensure mixing between the drywell to the wetwell. The release into the containment or drywell should be assumed to terminate at the end of the early in-vessel phase.	Conforms	See Item 3.7 of this table below for details.
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	Not Applicable	While containment sprays are a design feature that is available at LSCS, no credit is taken for airborne activity removal by them in this LOCA

Table 2: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	LSCS Analysis	Comments
	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 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).		AST analysis. Wall deposition of elemental iodine is credited in accordance with SRP 6.5.2
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	Not applicable	No in-containment recirculation filter systems exist at LSCS.

Table 2: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	LSCS Analysis	Comments
	release associated with the revised source term should be addressed.		
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. As indicated for Item 2 above, 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	LSCS 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. For BWRs with Mark III containments, the leakage from the drywell into	Conforms	No credit is taken for the leak rate reduction after 24 hours. LSCS uses a Mark II containment, and leakage from the drywell into the suppression chamber is not credited for the first two-hour period. Rapid mixing is considered thereafter due to ECCS restoration and associated steam production

Table 2: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	LSCS Analysis	Comments
	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.		to provide the uniform distribution required, with flow from the suppression chamber air space to the drywell through vacuum breakers as steam condensation reduces drywell pressure relative to that in the suppression chamber. As noted, such mixing after two hours is contained within Regulatory Guide 1.183 for Mark III containments, and has been accepted by the NRC.
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	LaSalle Technical Specification SR 3.6.1.3.1 identifies purposes for containment purging at LaSalle as inerting, de-inerting, pressure control, ALARA or air quality considerations for personnel entry, and surveillances that require valves to be open. TS 3.6.3.2 provides limitation on use for inerting and deinerting at power.

Table 2: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)

RG Section	RG Position	LSCS Analysis	Comments
4.1	Leakage from the primary containment should be considered to be collected, processed by engineered safety feature (ESF) filters, if any, and released to the environment via the secondary containment exhaust system during periods in which the secondary containment has a negative pressure as defined in technical specifications. Credit for an elevated release should be assumed only if the point of physical release is more than two and one-half times the height of any adjacent structure.	Conforms	Secondary Containment filtered release (via the Plant Stack) credit is taken at 15 minutes after the start of gap release, (effectively 17 minutes after the initiation of the LOCA). Gap release begins at ~2 minutes after LOCA initiation. For EAB and LPZ doses, elevated stack release is assumed for primary containment and ECCS leakage to the Reactor Building. Ground-level releases are assumed for MSIV Leakage. For Control Room doses X/Qs are determined in accordance with methodology described in RG 1.194.
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	Ground-level release assumptions are used for releases during the drawdown period.
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	Conforms	The wind speed exceeded approximately 28.2 mph (200' elevation of meteorological tower) only 5% of the time at LSCS in the secondary

Table 2: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	LSCS Analysis	Comments
	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%).		containment vicinity. The X/Q calculation is provided in Attachment 6.
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 mixing in the secondary containment.
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 except for MSIV leakage has been identified which bypasses the secondary containment.
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).	Exceptions taken, see comments	The CR and AEER recirculation filter trains are not designed to RG 1.52. However, TS 5.5.8.c currently credits the charcoal adsorbers at 70% efficiency. This efficiency is the established design basis and

Table 2: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	**RG Position**	**LSCS Analysis**	**Comments**
			current licensing basis, and is used in the DBA analysis.
5.1	With the exception of noble gases, all the fission products released from the fuel to the containment (as defined in Tables 1 and 2 of this guide) should be assumed to instantaneously and homogeneously mix in the primary containment sump water (in PWRs) or suppression pool (in BWRs) at the time of release from the core. In lieu of this deterministic approach, suitably conservative mechanistic models for the transport of airborne activity in containment to the sump water may be used. Note that many of the parameters that make spray and deposition models conservative with regard to containment airborne leakage are nonconservative with regard to the buildup of sump activity.	Conforms	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 design basis 5 gpm leak rate is two times the administratively controlled acceptance criteria for the sum of the simultaneous leakage from all components in the ESF recirculation systems as addressed in the TS 5.5.2 "Primary Coolant Sources Outside Containment" program. Since certain ECCS systems take suction immediately from the suppression pool, this leak path is assumed to start at time 0. Leakage to atmospheric

Table 2: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	**RG Position**	**LSCS Analysis**	**Comments**
			tanks is 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 Core Spray / Reactor Core Isolation Cooling test lines that discharge to the Condensate Storage Tank (CST) which provides a water seal. 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	Not Applicable	The temperature of the leakage does not exceed

Table 2: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	LSCS Analysis	Comments
	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.		212°F.
5.5	If the temperature of the leakage is less than 212°F or the calculated flash fraction is less than 10%, the amount of iodine that becomes airborne should be assumed to be 10% of the total iodine activity in the leaked fluid, unless a smaller amount can be justified based on the actual sump pH history and area ventilation rates.	Conforms	ECCS leakage into secondary containment is assumed to flash such that 10% of the total iodine activity in the leaked fluid is assumed airborne.
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).	Exceptions taken, see comments	The CR and AEER recirculation filter trains are not designed to RG 1.52. However, TS 5.5.8.c currently credits the charcoal adsorbers at 70% efficiency. This efficiency is the established design basis and current licensing basis, and is used in the DBA analysis.
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	Conforms	MSIV leakage will be considered an unfiltered

Table 2: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	**RG Position**	**LSCS Analysis**	**Comments**
	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.		radioactivity release pathway, with piping and condenser deposition credit, and the radiological consequences of such a release are analyzed. The radioactivity release from the fuel is assumed to instantaneously and homogeneously mix throughout the drywell air space. Mixing of this activity into the containment air space is as discussed under Item 3.7 above.
6.2	All the MSIVs should be assumed to leak at the maximum leak rate above which the technical specifications would require declaring the MSIVs inoperable. The leakage should be assumed to continue for the duration of the accident. Postulated leakage may be reduced after the first 24 hours, if supported by site-specific analyses, to a value not less than 50% of the maximum leak rate.	Conforms	MSIV leakage assumed in this accident analysis is 400 scfh for all steam lines and 200 scfh for any one line when tested at greater than or equal to 25 psig. No reduction in leakage is assumed at 24 hours, for conservatism.
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 of deposition and plateout for MSIV piping is based on the assumption of 2 well mixed volumes for any one pipe line providing a leak path, with one node from the

Table 2: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	LSCS Analysis	Comments
			reactor pressure vessel to the inboard MSIV (except for the assumed broken line, where deposition in this node is not credited), and the other node from the inboard MSIV to the Turbine Stop Valve that provides the seismically rugged boundary of the MSIV alternate drain pathway. For aerosol settling, only horizontal piping runs are credited, and only the horizontal projected surface area is considered available. In addition, no credit is taken for aerosol settling after 24 hours, for conservatism. The condenser also continues to be credited as a node provided for deposition and plateout. Its availability is assumed to start at 20 minutes, based on manual opening of steam line drains. This system has previously been determined to be seismically rugged. The formulation for determining elemental iodine

Table 2: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	LSCS Analysis	Comments
			activity removal from a well-mixed node is based on that developed in AEB-98-03, using a 20 group probability distribution of settling velocities (based on AEB-98-03 probability descriptions) with settling efficiencies determined for each group and a net weighted average efficiency. This process is significantly more conservative than use of a median settling velocity. Resuspension of deposited elemental iodine and immediate release as organic iodine is also modeled. Other phenomena, such as effects of depletion over time of more easily settled particle sizes are considered to be adequately addressed by the above conservatisms. For elemental iodine deposition, both horizontal and vertical piping is credited on all interior surfaces, as this deposition is not gravity

Table 2: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	LSCS Analysis	Comments
			dependent. The decay heat from fission product deposition on resulting temperatures for main steam lines is negligible.
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	No ESFs such as a MSIV leakage control system are assumed to be available to collect or treat MSIV leakage. After release from the condenser system as described below, ground-level releases are assumed without credit for holdup or dilution in the turbine building.
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 between the outboard MSIVs and the turbine stop valves is credited as piping systems capable of performing their safety function during and following an SSE. This includes the condenser, which is seismically rugged and meets the requirements of 10 CFR 100, Appendix A, as discussed in LSCS UFSAR Section 6.7. For elemental iodine, RG

Table 2: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	**RG Position**	**LSCS Analysis**	**Comments**
			1.183's Reference A-9 is considered only in part since it is the basis for slug flow models. Reference A-9 provides elemental iodine deposition velocities, resuspension rates and fixation rates. The deposition velocities are used in the well-mixed model formulation in AEB-98-03 that is analogous for aerosols or elemental iodine. This modeling is described in detail in this calculation. Resuspension of deposited elemental iodine is conservatively treated as immediately released organic iodine.
7.0	The radiological consequences from post-LOCA primary containment purging as a combustible gas or pressure control measure should be analyzed. If the installed containment purging capabilities are maintained for purposes of severe accident management and are not credited in any design basis analysis, radiological consequences need not be evaluated. If the primary containment purging is required within 30 days of the LOCA, the results of this analysis should be combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA. Reduction in the amount of radioactive material released via ESF filter	Conforms	Containment purging as a combustible gas or pressure control measure is not required nor credited in any design basis analysis for 30 days following a design basis LOCA at LSCS.

Table 2: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	LSCS Analysis	Comments
	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).		

Table 3: Conformance with RG 1.183 Appendix B (Fuel Handling Accident)			
RG Section	RG Position	LSCS 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	These assumptions are used.
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	A conservative fuel damage analysis has been performed.
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	These assumptions are used.
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	Conforms	All iodine added to the pool water is assumed to dissociate.

Table 3: Conformance with RG 1.183 Appendix B (Fuel Handling Accident)			
RG Section	**RG Position**	**LSCS Analysis**	**Comments**
	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.		
2	If the depth of water above the damaged fuel is 23 feet or greater, the decontamination factors for the elemental and organic species are 500 and 1, respectively, giving an overall effective decontamination factor of 200 (i.e., 99.5% of the total iodine released from the damaged rods is retained by the water). This difference in decontamination factors for elemental (99.85%) and organic iodine (0.15%) species results in the iodine above the water being composed of 57% elemental and 43% organic species. If the depth of water is not 23 feet, the decontamination factor will have to be determined on a case-by-case method.	Exceptions taken, see comments	The decontamination factor (DF) was determined in a more conservative manner than prescribed in RG 1.183, as described in Attachment 9. The 500 DF for elemental iodine is not used. A more conservative value of 285.29 is used since it is the value that yields an overall effective DF of 200 for 23 feet of water when combined with the stated initial iodine fractions.
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	These assumptions are used.
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	This assumption is used. No credit is taken for the SGT filtration.
4.2	A reduction in the amount of radioactive material released from the fuel pool by engineered safety feature (ESF) filter systems may be taken into account provided these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02. Delays in radiation detection, actuation of the ESF filtration system, or diversion of ventilation flow to the ESF	Not Applicable	No credit is taken for filtration from the secondary containment.

Table 3: Conformance with RG 1.183 Appendix B (Fuel Handling Accident)			
RG Section	**RG Position**	**LSCS Analysis**	**Comments**
	filtration system should be determined and accounted for in the radioactivity release analyses.		
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.	Not Applicable	Two-hour release to the environment is assumed, without mixing or dilution.
5.1	If the containment is isolated during fuel handling operations, no radiological consequences need to be analyzed.	Not Applicable	Secondary containment is not isolated.
5.2	If the containment is open during fuel handling operations, but designed to automatically isolate in the event of a fuel handling accident, the release duration should be based on delays in radiation detection and completion of containment isolation. If it can be shown that containment isolation occurs before radioactivity is released to the environment, no radiological consequences need to be analyzed.	Not Applicable	Automatic 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. **Footnote 3:** The staff will generally required that technical specifications allowing such operations include administrative controls to close the airlock, hatch, or open penetrations within 30 minutes. Such administrative controls will generally require that a dedicated individual be present, with necessary equipment available, to restore containment	Exception taken to Footnote 3	This 2-hour release assumption is utilized. Administrative controls to close the secondary containment openings to the environment is assumed to be accomplished within one hour.

Table 3: Conformance with RG 1.183 Appendix B (Fuel Handling Accident)			
RG Section	**RG Position**	**LSCS Analysis**	**Comments**
	closure should a 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. 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 taken for filtration of release from the secondary containment.
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	No credit is taken for dilution or mixing of the activity released from the reactor cavity. A 2-hour release assumption is utilized.