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Exelon Nuclear

10 CFR 50.90

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U. S. Nuclear Regulatory Commission
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Limerick Generating Station, Units 1 & 2
Facility Operating License Nos. NPF-39 and NPF-85
NRC Docket Nos. 50-352 and 50-353

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

References:

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

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

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

(4) Exelon/AmerGen 180-Day Response to Generic Letter 2003-01, Control Room Habitability, December 9, 2003

Pursuant to 10 CFR 50.67, "Accident Source Term," and 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company, LLC (Exelon) hereby requests an amendment to the Facility Operating Licenses listed above. The proposed change is requested to support application of an alternative source term (AST) methodology, with the exception that Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactor Sites," will continue to be used as the radiation dose basis for equipment qualification. This submittal has used the methods described in Regulatory Guide 1.183 (Reference 1) except for the few instances where alternative methods of compliance have been proposed as allowed by the guidance in this reference. These alternative methods have been fully discussed in Tables A through E in Attachment 1 of this LAR.

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

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guidance for the implementation of AST is provided in Reference 1. This regulatory guide provides guidance on acceptable applications of ASTs. The use of AST changes only the regulatory assumptions regarding the analytical treatment of the design basis accidents (DBAs).

Exelon has performed radiological consequence analyses of the four DBAs that result in offsite exposure to support a full-scope implementation of AST as described in Reference 1. The AST analyses for Limerick Generating Station (LGS), Units 1 & 2, were performed following the guidance in References 1 and 2.

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

- TS and associated Bases revisions to increase Main Steam Isolation Valve (MSIV) maximum allowable closure time;
- TS and associated Bases revisions to reflect the changes in the laboratory testing acceptance criteria associated with the charcoal adsorbers in the Reactor Enclosure Recirculation System (RERS), Standby Gas Treatment System (SGTS), and Control Room Emergency Fresh Air System (CREFAS).
- TS change reflecting replacement of automatic initiation of the CREFAS radiation mode with a 30-minute manual isolation.
- TS and associated Bases revisions to reflect lower RERS flows associated with the dose calculation requirements.
- TS and associated Bases revisions to change the applicability requirements for the following systems during movement of recently irradiated fuel assemblies in secondary containment and to reflect that these systems are no longer required to be operable during core alterations:
  - Standby Gas Treatment System,
  - Secondary Containment, and
  - Control Room Emergency Fresh Air System
- TS and associated Bases revisions to reflect use of the Standby Liquid Control (SLC) System to buffer suppression pool pH to prevent iodine re-evolution during a postulated loss of coolant accident.

The proposed changes related to the applicability requirements during movement of irradiated fuel assemblies are consistent with Technical Specification Task Force Traveler (TSTF)-51, Revision 2. TSTF-51, Revision 2, was approved by the NRC on November 1, 1999. TSTF-51 changes the TS operability requirements for certain engineered safety features such that they are not required after sufficient radioactive decay has occurred to ensure that offsite doses remain within limits. Since a portion of this license amendment request is based on TSTF-51, Exelon 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 fuel handling accident.

Exelon has been an active participant on the NEI Control Room Habitability (CRH) Task Force and understands the NRC position regarding CRH and acknowledges the fact that Generic Letter 2003-01 has been issued. This submittal does not directly address the CRH issue other than to provide an increase in the assumed unfiltered inleakage value. However, Exelon has provided a formal response to the Generic Letter (reference 4). Although an ASTM E741 tracer gas test has not been performed to date, the assumed unfiltered inleakage value in the AST dose analyses is equal to 100% of the full Control Room pressurization airflow in the emergency modes of operation. With the assumed inleakage value this high, it is Exelon's judgment that the measured value is not reasonably expected to exceed this assumed value. Other CRH actions have been addressed via Generic Letter response.

This request is subdivided as follows.

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

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

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

4. Attachment 4 provides the Retyped Technical Specification pages.

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

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

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

8. Attachment 8 provides a discussion of the technical parameters and methodologies used in the AST calculations.

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

Exelon requests approval of the proposed amendments by February 27, 2005. Once approved, the amendments shall be implemented within 60 days. This implementation period will provide adequate time for the affected station documents to be revised using the appropriate change control mechanisms. In accordance with 10 CFR 50.91(b), Exelon is notifying the State of Pennsylvania of this application for changes to the TS by transmitting a copy of this letter and its attachments to the designated State Official.

If you have any questions or require additional information, please contact Doug Walker at (610) 765- 5726.

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

Respectfully,

Executed on 02-27-04

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

Attachments:
1. Description of Proposed Changes, Technical Analysis, and Regulatory Analysis
2. Markup of Technical Specification pages
3. Markup of Technical Specification Bases pages *(Information only)*
4. Retyped Technical Specification pages
5. Retyped Technical Specification Bases pages *(Information only)*
6. List of Commitments
7. LGS Meteorological data *(Information only)*
8. Technical Parameters for AST Calculations

cc: H. J. Miller, Administrator, Region I, USNRC
S. Hansel, USNRC Senior Resident Inspector, LGS
G. Wunder, Senior Project Manager Limerick (acting), USNRC (by FedEx)
R. R. Janati - Commonwealth of Pennsylvania

ATTACHMENT 1

Limerick Generating Station
Units 1 & 2

License Amendment Request
"LGS Alternative Source Term Implementation"

1.0 DESCRIPTION

In accordance with 10 CFR 50.67, "Accident Source Term," and 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company, LLC (Exelon) requests a change to Appendix A, Technical Specifications (TS), of Facility Operating License Nos. NPF-39 and NPF-85 for the Limerick Generating Station (LGS), Units 1 & 2. 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," will continue to be used as the radiation dose basis for equipment qualification.

Radiological consequence analyses have been performed for the four bounding Design Basis Accidents (DBAs) that result in offsite exposure (i.e., Loss of Coolant Accident (LOCA), Main Steam Line Break (MSLB), Fuel Handling Accident (FHA), and Control Rod Drop Accident (CRDA)) to support a full-scope implementation of AST. The AST analyses for LGS were performed following the guidance in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" and Standard Review Plan 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms". These analyses have been performed using NRC approved computer codes by qualified consultants and have had extensive cross-functional reviews and challenges by Exelon personnel.

The proposed changes to the TS will allow LGS to apply the results of the plant-specific AST analyses using the guidance in Regulatory Guide 1.183 and meet the requirements of 10 CFR 50.67. Approval of this change will provide a source term for LGS that will result in a more accurate assessment of the DBA radiological doses. The improved dose assessment allows relaxation of some current licensing basis requirements as described below.

This proposed change will increase allowable Main Steam Isolation Valve (MSIV) closure time from 5 to 10 seconds. Unplanned MSIV repairs are potential contributors to increased outage duration and unplanned personnel exposure.

The radiological analysis includes a leakage rate of 100 scf per hour for any main steam isolation valve and a combined maximum main steam line pathway leakage of 200 scf per hour, therefore, the current requirement to satisfy a maximum of 11.5 scf per hour for any main steam isolation valve after restoration is being removed.

To satisfy the condition of application of AST to control the suppression pool pH following a LOCA, LGS is proposing to use the Standby Liquid Control (SLC) System. This requires revising the Technical Specifications applicability requirements for the SLC system to include Operational Condition 3. Clarification is also being made to the Surveillance Requirements section for the SLC system to verify the value for the required weight of Boron-10, instead of using the currently specified weight of sodium pentaborate. This is an equivalent change.

In addition, implementation of AST will no longer require secondary containment to be established except during Operations with the Potential for Draining the Reactor

Vessel (OPDVs) and movement of recently irradiated fuel. This proposed change provides the flexibility of performing fuel floor activities (such as control rod blade exchanges and fuel movements) as well as movement of large equipment through the secondary containment boundary in support of outage activities while remaining within all safety limits.

Other benefits of AST are the cost savings that will be achieved by reducing credited charcoal efficiencies in accident analyses. This additional margin will extend available charcoal life by changing methyl iodine penetration acceptance criteria and will result in less frequent charcoal filter regeneration. HEPA efficiency credit reductions provide additional operating margin, but no reduction in test acceptance criteria are proposed. These benefits apply to the Reactor Enclosure Recirculation System (RERS), Standby Gas Treatment System (SGTS), and the Control Room Emergency Fresh Air System (CREFAS).

Adopting the AST methodology may also support future evaluations and license amendments.

2.0 PROPOSED CHANGES

The proposed changes related to the applicability requirements during movement of irradiated fuel assemblies are consistent with Technical Specification Task Force Traveler (TSTF)-51, "Revise Containment Requirements During Handling of Irradiated Fuel and Core Alterations," Revision 2. The NRC approved TSTF-51 on October 15, 1999. TSTF-51 changes the TS operability requirements for engineered safety features such that they are not required after sufficient radioactive decay has occurred to ensure that offsite doses remain within limits.

Since a portion of this license amendment request is based on TSTF-51, Exelon 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 fuel handling accident.

Proposed changes to the Technical Specifications resulting from this submittal are summarized below:

2.1 TS Section 1.0, "Definitions"

The proposed change revises the definition of DOSE EQUIVALENT I-131 in TS Definition 1.9 to replace the word "thyroid" with "inhalation committed effective dose equivalent (CEDE)" and to add a reference to "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, ORNL, 1989, as described in Regulatory Guide 1.183".

2.2 TS Section 1.0, "Definitions"

The proposed change adds the definition of RECENTLY IRRADIATED FUEL as TS Definition 1.35. RECENTLY IRRADIATED FUEL is fuel that has occupied part of a critical reactor core within the previous 24 hours. Subsequent definitions in this section are renumbered to reflect this addition.

2.3 TS Section 3/4 1.5, "Standby Liquid Control (SLC) System"

The proposed change revises the Applicability of TS Section 3.1.5 to include Operational Condition 3 for the SLC system. This change implements AST assumptions regarding the use of the SLC System to control the suppression pool pH following a LOCA involving significant fission product release. Action 3.1.5 has been revised to include action statements for inoperable SLC equipment in Operational Condition 3, which can include going to COLD SHUTDOWN. SR 4.1.5.b.2 is revised to reflect the Boron-10 weight requirement that is equivalent to the current requirements for Sodium Pentaborate at 29% enrichment.

2.4 TS Section Tables 3.3.2-1, "Isolation Actuation Instrumentation Action Statements"

The proposed change revises Table Notation (*) for TS Table 3.3.2-1 by: 1) replacing the term "irradiated fuel" with "RECENTLY IRRADIATED FUEL;" 2) removing "refueling area," since secondary containment can consist of the common refueling area and the Reactor Enclosure zones; and 3) deleting the "during CORE ALTERATIONS" criteria. The table notation applies to the applicable operation conditions for the Refueling Area Unit 1 and Unit 2 Ventilation Exhaust Duct Radiation – High and the Refueling Area Manual isolation instrumentation. These changes are consistent with TSTF-51.

2.5 TS Section Table 4.3.2.1-1, "Isolation Actuation Instrumentation Surveillance Requirements"

The proposed change revises Table Notation (*) for TS Table 4.3.2.1-1 by:
1) replacing the term "irradiated fuel" with "RECENTLY IRRADIATED FUEL;"
2) removing "refueling area," since secondary containment can consist of the common refueling area and the Reactor Enclosure zones;" and 3) deleting the "during CORE ALTERATIONS" criteria. The table notation applies to the operation conditions for which surveillance is required for the Refueling Area Unit 1 and Unit 2 Ventilation Exhaust Duct Radiation – High and the Refueling Area Manual isolation instrumentation. These changes are consistent with TSTF-51.

2.6 TS Section Table 3.3.7.1-1, "Radiation Monitoring Instrumentation"

The proposed change revises Table Notation (*) for TS Table 3.3.7.1-1 by replacing the term "irradiated fuel" with "RECENTLY IRRADIATED FUEL" and adding the criteria "or during operations with a potential for draining the reactor vessel with the vessel head removed and fuel in the vessel". The table notation applies to the applicable operation conditions for the Main Control Room Normal Fresh Air Supply Radiation Monitor. In addition, the Main Control Room Normal Fresh Air Supply Radiation Monitor is no longer applicable to Operational Condition 5 and is only required as an Alarm function only. The trip function is being removed.

2.7 TS Section Table 4.3.7.1-1, "Radiation Monitoring Instrumentation Surveillance Requirements"

The proposed change revises Table Notation (*) for TS Table 4.3.7.1-1 by replacing the term "irradiated fuel" with "RECENTLY IRRADIATED FUEL" and adding the criteria "or during operations with a potential for draining the reactor vessel with the vessel head removed and fuel in the vessel". The table notation applies to the operation conditions for which surveillance is required for the Main Control Room Normal Fresh Air Supply Radiation Monitor. In addition, the Main Control Room Normal Fresh Air Supply Radiation Monitor is no longer applicable to Operational Condition 5.

2.8 TS Section 3.4.7 and 4.4.7, "Main Steam Isolation Valves (MSIV)"

The proposed change revises Limiting Condition for Operation 3.4.7 to increase the MSIV maximum closing time from "less than or equal to 5 seconds" to "less than or equal to 10 seconds". Additionally, the proposed change also revises the Surveillance Requirement 4.4.7 to increase the MSIV full closure from "between 3 and 5 seconds" to "between 3 and 10 seconds".

2.9 TS 3.6.1.2, Restore Action c., "Primary Containment Leakage"

The proposed change revises the action statement to restore "the leakage rate to ≤100 scf per hour for any MSIV that exceeds 100 scf per hour." The current restore value is ≤ 11.5 scf per hour, for any MSIV that exceeds 100 scfh, based on the existing radiological analysis.

2.10 TS Section 3.6.5.1.2, "Refueling Area Secondary Containment Integrity"

The proposed change deletes "OPERATIONAL CONDITION *" in the Applicability section of TS 3.6.5.1.2 and the corresponding explanation is relocated from the bottom of the page. Additionally, the Applicability and Action Statements are revised by replacing the term "irradiated fuel" with "RECENTLY IRRADIATED FUEL" and deleting reference to "CORE ALTERATIONS".

2.11 TS Section 3.6.5.2.2, "Refueling Area Secondary Containment Automatic Isolation Valves"

The proposed change deletes the "OPERATIONAL CONDITION *" in the Applicability section of TS 3.6.5.2.2 and the corresponding explanation is relocated from the bottom of the page. Additionally, the Applicability and Action Statements are revised by replacing the term "irradiated fuel" with "RECENTLY IRRADIATED FUEL" and deleting reference to "CORE ALTERATIONS".

2.12 TS Section 3.6.5.3, "Standby Gas Treatment System – Common System"

The proposed change deletes the (*) in the Applicability and Action section of TS 3.6.5.3 and the corresponding explanation is relocated from the bottom of the page. Additionally, the Applicability section and Action Statements a.2 and b. are revised by replacing the term "irradiated fuel" with "RECENTLY IRRADIATED FUEL" and deleting references to "CORE ALTERATIONS". The action statement b. is being

revised to only be applicable to handling of recently irradiated fuel in the secondary containment, or during operations with a potential for draining the reactor vessel.

2.13 TS Section 4.6.5.3, "Standby Gas Treatment System – Common System"

The charcoal adsorber sample acceptance criteria for the methyl iodide penetration tests in Surveillance Requirements 4.6.5.3.b.2 and 4.6.5.3.c has been increased from less than 0.5% to less than 1.25%.

2.14 TS Section 4.6.5.4, "Reactor Enclosure Recirculation System"

The proposed change relaxes the following Surveillance Requirements (SR) related to the RERS charcoal adsorbers as shown:

- SR 4.6.5.4.a to annotate a flow range through the HEPA filters of a minimum of 30,000 cfm through the HEPA filters
- SR 4.6.5.4.b.1 to clarify that the in-place penetration test is performed at the rated flow rate (60,000 cfm ± 10%) instead of annotating a specific flow of 60,000 cfm ± 10%.
- SR 4.6.5.4.b.2 to verify at least once per 24 months, or (1) after structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any communicating ventilation zone, that a laboratory analysis of a representative carbon sample obtained shows methyl iodide penetration of less than 15% rather than 2.5%.
- SR 4.6.5.4.b.3 to verify a subsystem flow rate within a range of 30,000 to 66,000 cfm.
- SR 4.6.5.4.c to verify after 720 hours of operation, that a laboratory analysis of a representative carbon sample shows methyl iodide penetration of less than 15% rather than 2.5%.
- SR 4.6.5.4.d.1 to clarify that the in-place penetration test is performed at the rated flow rate (60,000 cfm ± 10%) instead of annotating a specific flow of 60,000 cfm ± 10%.
- SR 4.6.5.4.e to clarify that the in-place penetration test is performed at the rated flow rate (60,000 cfm ± 10%) instead of annotating a specific flow of 60,000 cfm ± 10%.
- SR 4.6.5.4.f to clarify that the in-place penetration test is performed at the rated flow rate (60,000 cfm ± 10%) instead of annotating a specific flow of 60,000 cfm ± 10%.

2.15 TS Section 3.7.1.2, "Emergency Service Water System – Common System"

The proposed change expands the definition of the (*) to include "handling RECENTLY IRRADIATED FUEL in the secondary containment and during operations with a potential for draining the reactor vessel" (consistent with TSTF-51). Additionally, the (*) in the LCO, the Applicability section and Action c. of TS 3.7.1.2 is

deleted and the corresponding explanation is relocated from the bottom of the applicable page.

2.16 TS Section 3.7.1.3, "Ultimate Heat Sink"

The proposed change expands the definition of the (*) to include "handling RECENTLY IRRADIATED FUEL in the secondary containment and during operations with a potential for draining the reactor vessel" (consistent with TSTF-51). Additionally, the reference to OPERATIONAL CONDITION (*) in the Applicability section and Action c. of TS 3.7.1.3 is deleted and the corresponding explanation is relocated from the bottom of the page.

2.17 TS Section 3.7.2, "Control Room Emergency Fresh Air Supply System – Common System "

- Applicability Section - the proposed change expands the definition of the (*) to include "when handling RECENTLY IRRADIATED FUEL in the secondary containment, or during operations with a potential for draining the reactor vessel" (consistent with TSTF-51). Additionally, the (*) in the Applicability section is deleted and the corresponding explanation is relocated from the bottom of the page.
- Action b. – the operational condition is revised to expand the definition of the (*) to include "when handling RECENTLY IRRADIATED FUEL in the secondary containment, or during operations with a potential for draining the reactor vessel" (consistent with TSTF-51). Additionally, the (*) in the Applicability section is deleted and the corresponding explanation is relocated from the bottom of the page.
- Action b.2 – the action statement is revised by replacing the term "irradiated fuel" with "RECENTLY IRRADIATED FUEL" and deleting reference to "CORE ALTERATIONS".
- Action c – the reference to Operational Condition (*) is deleted and the action has been incorporated into Action b.2.
- Notation (*) at the bottom of the page is deleted and included in the applicable sections.

2.18 TS Section 4.7.2, "Control Room Emergency Fresh Air Supply System – Common System"

The proposed change relaxes the following Surveillance Requirements (SR) related to the charcoal adsorbers as shown:

- The charcoal adsorber sample acceptance criteria for the methyl iodide penetration tests in Surveillance Requirements 4.7.2.c.2 and 4.2.7.d has been increased from less than 2.5% to less than 10%.
- The proposed change revises SR 4.7.2.e.3 to only require verification of the manual initiation of the radiation mode of CREFAS and removes reference to the outside air intake high radiation mode.

2.19 TS Section 3.8.1.2, "AC Sources – Shutdown"

The proposed change expands the definition of the (*) to include "when handling RECENTLY IRRADIATED FUEL in the secondary containment or during operations with a potential for draining the reactor vessel" (consistent with TSTF-51). Additionally, the (*) in the Applicability section of TS 3.8.1.2 is deleted and the corresponding explanation is relocated from the bottom of the page.

2.20 TS Section 3.8.2.2, "DC Sources – Shutdown"

The proposed change expands the definition of the (*) to include "when handling RECENTLY IRRADIATED FUEL in the secondary containment or during operations with a potential for draining the reactor vessel" (consistent with TSTF-51). Additionally, the (*) in the Applicability section of TS 3.8.2.2 is deleted and the corresponding explanation is relocated from the bottom of the page. The statement in TS 3.8.2.2 Action c. is revised to change "irradiated fuel" to "RECENTLY IRRADIATED FUEL."

2.21 TS Section 3.8.3.2 "Electrical Power Systems, Distribution - Shutdown"

The proposed change expands the definition of the (*) to include "when handling RECENTLY IRRADIATED FUEL in the secondary containment or during operations with a potential for draining the reactor vessel" (consistent with TSTF-51). Additionally, the (*) in the Applicability section of TS 3.8.3.2 is deleted and the corresponding explanation is relocated from the bottom of the page. The statement in TS 3.8.3.2 Actions a. and b. are revised to change "irradiated fuel" to "RECENTLY IRRADIATED FUEL."

3.0 BACKGROUND

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

The fission product release from the reactor core into containment is referred to as the "source term," and it is characterized by the composition and magnitude of the radioactive material, the chemical and physical properties of the material, and the timing of the release from the reactor core. Since the publication of U.S. Atomic Energy Commission Technical Information Document, TID-14844, Calculation of Distance Factors for Power and Test Reactor Sites (Reference 7.1), significant advances have been made in understanding the composition and magnitude, chemical form, and timing of fission product releases from severe nuclear power plant accidents. Many of these insights developed out of the major research efforts

started by the NRC and the nuclear industry after the accident at Three Mile Island. NUREG-1465 (Reference 7.4) was published in 1995 with revised ASTs for use in the licensing of future Light Water Reactors (LWRs). The NRC, in 10 CFR 50.67, later allowed the use of the ASTs described in NUREG-1465 at operating plants. This NUREG represents the result of decades of research on fission product release and transport in LWRs under accident conditions. One of the major insights summarized in NUREG-1465 involves the timing and duration of fission product releases.

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

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

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

The requested license amendment involves a full-scope application of the AST, addressing the composition and magnitude of the radioactive material, its chemical and physical form, and the timing of its release as described in Regulatory Guide 1.183.

Exelon has performed radiological consequence analyses of the four DBAs that result in the most significant offsite exposures (i.e., LOCA, MSLB, FHA, and CRDA). These analyses were performed to support full scope implementation of AST. The AST analyses have been performed in accordance with the guidance in Regulatory Guide 1.183 and NRC Standard Review Plan 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms" (Reference 7.3). The implementation consisted of the following steps:

- Identification of the AST based on plant-specific analysis of core fission product inventory,
- Calculation of the release fractions for the four DBAs that result in the most significant control room and offsite doses (i.e., LOCA, MSLB, FHA, and CRDA),
- Analysis of the atmospheric dispersion for the radiological propagation pathways,
- Calculation of fission product deposition rates and transport and removal mechanisms,
- Calculation of offsite and control room personnel Total Effective Dose Equivalent (TEDE) doses, and

- Evaluation of suppression pool pH to ensure that the iodine deposited into the suppression pool during a DBA LOCA does not re-evolve and become airborne as elemental iodine.

The analysis assumptions for the transport, reduction, and release of the radioactive material from the fuel and the reactor coolant are consistent with the guidance provided in applicable appendices of Regulatory Guide 1.183 for the four analyzed DBAs.

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

4.0 TECHNICAL ANALYSIS

4.1 Evaluation

4.1.1 Scope

4.1.1.1 Accident Radiological Consequence Analyses

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

- CRDA, UFSAR Section 15.4.9;
- MSLB, UFSAR Section 15.6.4;
- LOCA, UFSAR Section 15.6.5; and
- FHA, UFSAR Section 15.7.4.

The analyses were performed in accordance with Regulatory Guide 1.183 to confirm compliance with the acceptance criteria presented in 10 CFR 50.67.

4.1.2 NUREG-0737, Item II.B.2

Exelon has determined that continued compliance will be maintained with NUREG-0737, Item II.B.2, "Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems Which May be Used in Post-Accident Operations."

The source term associated with environmental qualification of equipment will remain consistent with previous commitments under 10 CFR 50.49.

4.2 Method of Evaluation

4.2.1 Fission Product Inventory

Pre-AST core source terms were determined based on TID-14844 methodology. That is, inventory was based on the fission product equilibrium based on U-235 fission product yields and isotopic decay constants. In accordance with Regulatory Guide 1.183, this simplified approach is replaced with ORIGEN 2.1 (Reference 7.5) methodology used to determine core inventory. This program provides a more complete and accurate simulation of isotopic buildup and depletion, including consideration of fission product yields from all isotopes, and activation as well as decay.

The current licensed thermal power level at Limerick is 3458 MWt. These source terms were evaluated at end-of-cycle and at beginning of cycle (100 effective full power days (EFPD) to achieve equilibrium) conditions and worst-case inventory used for the selected isotopes. These values were then divided by power level to obtain activity in units of Ci/MWt. Accident analyses are based on a 3527 MWt power level that includes the current accident analysis design basis allowance for instrument uncertainty.

Source terms were based on a 2-year fuel cycle with a nominal 711 EFPD per cycle. These source terms were developed using ORIGEN 2.1. The values extracted from the ORIGEN 2.1 runs generated for Peach Bottom, which are also applicable to Limerick, and are for the standard 60-isotope RADTRAD (Reference 7.6) library except that the activation products Co-58 and Co-60 used RADTRAD default library values.

The reactor coolant fission product inventory for MSLB analysis was based on the Technical Specification limits in terms of Dose Equivalent I-131 (the concentration of I-131 that alone would produce the same dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually assumed), using inhalation Committed Effective Dose Equivalent (CEDE) dose conversion factors from Federal Guidance Report 11 (Reference 7.15) as described in Regulatory Guide 1.183.

4.2.2 Radiological Consequence

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

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

Offsite exclusion area boundary (EAB) and low population zone (LPZ) atmospheric dispersion factors (χ/Qs) were calculated using the guidance of Regulatory Guide

1.145 (Reference 7.7) and the PAVAN computer code (Reference 7.8). This code has been used by the NRC in safety reviews.

The χ/Q values resulting at the Control Room Intake were calculated using the NRC-sponsored computer codes ARCON96 (Reference 7.9), consistent with the procedures in Regulatory Guide 1.194 (Reference 7.19).

Figure 1 shows the "Layout of Release Points for LGS."

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

Regulatory Guide 1.183, Section 4.1.1 states, the dose calculations should determine the TEDE and that TEDE is the sum of the committed effective dose equivalent (CEDE) from inhalation and the deep dose equivalent (DDE) from external exposure. Section 4.1.2 of the Regulatory Guide further explains that the exposure-to-CEDE factors for inhalation of radioactive material should be derived from the data provided in ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers" and that 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" provides tables of conversion factors acceptable to the NRC staff. The factors in the column headed "effective" yield doses corresponding to the CEDE.

In a similar fashion, Section 4.1.4 of the Regulatory Guide emphasizes that 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", provides external EDE conversion factors acceptable to the NRC staff. The factors in the column headed "effective" yield doses corresponding to the EDE.

LGS post-LOCA direct shine dose from sources outside of the control room is dominated by a Unit 1 14" diameter core spray pipe, located 18 inches from the 36-inch thick shield wall between the control room and the Reactor Enclosure.

- The dose from the pipe has been re-evaluated based on AST-based ECCS fluid radionuclide concentrations integrated over the accident duration with standard control room occupancy credit for the 1 to 4 day and 4 to 30 day periods. Based on a review of functions required and occupancy demand near this source, it is expected that locations within the 0.22 rem isodose line can be managed by administrative controls to within this dose criterion. The 0.22 rem value is used

to characterize direct dose for the remainder of the control room. Generally, the cabinets within or near this isodose line are only needed at most for periodic monitoring, and do not require continuous operator attention.

Other sources were examined and the only external source of significance was from an RHR line located in the reactor enclosure approximately 50 feet from the control room shield wall. This source contributes an additional 0.12 rem. Other sources such as reactor enclosure airborne and external cloud and RERS, SGTS, and CREFAS filters are negligible because of shielding, distance or both.

4.3 Inputs and Assumptions

4.3.1 Accident Radiological Consequence Analyses

Release Mode

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

Onsite Meteorological Measurements Program

The LGS meteorological measurement program meets the guidelines of Regulatory Guide 1.23 (Safety Guide 23), "Onsite Meteorological Programs" (Reference 7.11). The tower base areas are on natural surfaces (e.g., short natural vegetation) with towers free from obstructions and micro-scale influences. This ensures that data is representative of the overall site area. The program consists of monitoring wind direction, wind speed, temperature, and precipitation. The method used for determining atmospheric stability is delta temperature (T), which measures the vertical temperature difference. These data, referenced in ANSI/ANS-2.5-1984 (Reference 7.10), are used to determine the meteorological conditions prevailing at the plant site.

The main meteorological weather tower (Tower 1) located at Weather Station No. 1 is a 280-foot tower situated approximately 3000 feet NW of the LGS structure vents. Tower 1 is also located approximately 2000 feet NNW of the center of the Unit 1 cooling tower location and approximately 2400 feet NW of the center of the Unit 2 cooling tower location. Grade elevation at Weather Station No. 1 is el 250' mean sea level (MSL).

Meteorological weather tower (Tower 2) located at Weather Station No. 2 is a 310-foot tower situated approximately 2100 feet west of the LGS structure vents. Tower 2 is also located approximately 1950 feet WSW of the center of the Unit 1 cooling tower location, and approximately 2600 feet WSW of the center of the Unit 2 cooling tower location.

The wind instruments on both towers are mounted on retractable booms extending upwind 10'-0" west of Tower 1 (WNW of Tower 2). Each face of the triangular towers is 3'-6" inches wide. The temperature sensors are located in aspirators and are 2'-0" from the tower.

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

Inspections and maintenance of all equipment is accomplished in accordance with procedures based on the instrument manufacturer's manuals. This inspection occurs at least once per week by qualified technicians 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 control room and to an on-site computer for archive storage. Periodically, all digital and analog data are sent to the approved meteorological monitoring consultant for data processing and analysis. The digital data acquisition systems are remotely interrogated by the consultant to perform a daily quality check on system performance with the objective of identifying potential problems and to notify plant personnel as soon as possible in order to minimize down-time. This is performed each working day. All analog chart data are subject to a quality check by the consultant. This quality check consists of time continuity, instrument malfunction, inking problems, directional switching problems, negative speeds, missing data, and digital/analog correlation.

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

Meteorological data utilized for the calculation of new atmospheric dispersion coefficients (X/Qs) were selected from the historical record of the Station meteorological monitoring tower network. Monitoring records dating back to 1972 and extending through 2002 for Tower 1 (primary tower) and Tower 2 (backup tower) were evaluated. It was desired that this calculation be based upon the continuous 5-year period that constitutes the highest Tower 1 data recovery. The period 1996-2000 was selected because it satisfies the requirements above and it represents the most complete and accurate data set that would be representative of the site meteorological data. The data was reviewed to ensure instrumentation problems and missing or anomalous observations do not affect the validity of the data. This is consistent with the NRC staff's guidance in Regulatory Guide 1.194 that considers five years of hourly observations to be representative of long-term trends.

Tower 2 data were used only for substitution of any missing Tower 1 data as follows:

Limerick Meteorological Tower Instrument Levels
(Elevation above subject tower grade)

	Tower 1 (primary)	Tower 2 (backup)
Wind Speed:		
Elevation 1	30 ft	159 ft
Elevation 2	175 ft	304 ft
Wind Direction:		
Elevation 1:	30 ft	159 ft
Elevation 2:	175 ft	304 ft

The meteorological vendor has illustrated that the Tower 2 delta temperature data are sufficiently representative to be substituted for the Tower 1 delta temperature data; however, since the Tower 1 and Tower 2 delta temperature height intervals differ from each other somewhat, and also since for all years shown, the primary Tower 1 has data recovery rates well above the NRC's 90 percent requirements, it was deemed unnecessary to make such substitutions.

Hereinafter, the Tower 1 ARCON96 meteorological input database with applicable Tower 2 values substituted for missing Tower 1 values as indicated above will be identified as the "Tower 1 Modeling Database".

The designation of 'calm' is made to all wind speed observations 0.5 mph or less. The higher of the starting speeds of the wind vane and anemometer equipment on each of the towers (i.e. 0.5 mph) was used as the threshold for calm winds per Regulatory Guide 1.145, Section 1.1.

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.

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

Wind roses and joint frequency distributions were reviewed for meteorological and climatological reasonableness and found to be acceptable prior to use. A review

was also conducted on specific hourly data prior to the execution of the atmospheric dispersion calculations in PAVAN and ARCON96. This consisted of manual spot checks of the spreadsheet reformatted data in comparison with the raw data provided by the vendor.

The LGS North and South Stacks are executed by ARCON96 as a vent release. As depicted in Attachment B, both stacks have a height of 416 ft MSL (199 ft above grade floor elevation). The stacks are located between Reactor Enclosures 1 and 2 with the North Stack situated on the north face of the buildings and the South Stack on the south face of the buildings. These stacks are less than 2.5 times the 194.75 ft high Reactor Enclosures (i.e., the highest adjacent building), and therefore, per Regulatory Guide 1.145, they are modeled as a 'vent' release.

Both the North and South Stacks are conservatively assumed to have a zero (0) flow, for which ARCON96 requires that the exit velocity and stack diameter each be assigned an input value of zero (0). Per Regulatory Guide 1.194, Table A-2, the actual building vertical cross-sectional area perpendicular to the wind direction must be utilized; therefore, the Reactor Enclosures' combined vertical cross-sectional area of 5851 m^2 (calculated as height = 59.4 m, and w = 98.5 m), was input into ARCON96 to account for wake effects.

Transport Mode

Atmospheric dispersion coefficients were calculated, for the identified release paths, based on site-specific meteorological data collected between 1996 through 2000 based on the current Regulatory Guide 1.145. The new dispersion coefficients developed represent a change to those used in the current UFSAR analyses. The offsite location values currently in the UFSAR are based on an early draft of Regulatory Guide 1.145.

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

During the radiation mode of operation, the control room ventilation system supplies 525 cfm of filtered, outdoor air to maintain the control room at 0.1-inch water column positive pressure with respect to the adjacent areas. Intentionally admitting outdoor air into the control room facilitates reduction of infiltration through the control room boundary by assuring that air is exfiltrating from the zone at an adequate velocity (i.e., a velocity through the control room boundary to develop and maintain a pressure of 0.1-inch water column).

Automatic initiation of the radiation isolation mode is no longer credited, and manual action within 30 minutes is assumed to initiate CREFAS. During the initial 30 minutes the normal 2100 cfm of unfiltered intake is used along with an assumed additional 525 cfm of unfiltered inleakage. Since the Chlorine isolation mode could be in operation due to testing, manual transfer to the radiation isolation mode would be required. Therefore, for added conservatism, indefinite operation in the chlorine isolation mode is evaluated. This condition was demonstrated to be bounded by the results with the radiation isolation mode initiated at 30 minutes.

During the radiation isolation mode, infiltration through the control room boundary is initially negligible because the control room will be at a positive pressure at the time of system isolation. Infiltration following isolation is assumed to be 525 cfm of unfiltered inleakage, which includes impacts of ingress and egress. The opening and closing of boundary doors can induce infiltration to the control room. However, air in-flow due to ingress and egress at LGS is minimized by the use of the door seals applied by the operators in the event of a LOCA. Installation of these seals is controlled by procedure.

The infiltration through the system components located outside the control room occurs through joints and seams in the ductwork, around damper shafts, through joints and penetrations in the air-handling units, and through the dampers that isolate the control room from non-habitable areas. The inleakage has not yet been measured via tracer gas testing. However, a tracer gas test is being scheduled as indicated in the response to Generic Letter 2003-01. This AST analysis assumes a value of 525 cfm and is a conservative estimate that should easily pass a tracer gas test since no driving force greater than the supplied intake would be expected. In the event that the tracer gas test indicates inleakage greater than 525 cfm, repairs would be implemented or recalculation performed in a timely manner to ensure control room operator doses remain within the acceptable levels described in 10CFR50.67 and General Design Criteria (GDC) 19.

Potential adverse interactions between the control room and adjacent zones that could allow the transfer of radioactive gases into the control room are minimized by maintaining the control room at a positive pressure of 0.1 inch water column with respect to adjacent areas during emergency pressurized modes. In the chlorine isolation mode (toxic gas), the control room is maintained at a neutral pressure. During this mode, 525 cfm (normal emergency mode flow rate) is assumed to enter the control room unfiltered.

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

Removal Mode

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

4.3.1.1 LOCA Inputs and Assumptions

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

LOCA Release Inputs

Design basis Primary Containment leakage is assumed to be controlled to a L_a rate of 0.5% per day. For RADTRAD radioactivity transport analysis this leak rate will be

used for the first 24 hours, and then reduced by 50% thereafter as allowed by Regulatory Guide 1.183, based on containment pressure reductions.

The entire leakage is treated as being into the secondary containment as discussed in Section 6.2 of the LGS UFSAR. For analytical purposes, the secondary containment used in the LOCA analysis refers to the reactor enclosure volume. However, secondary containment could also encompass the refueling area. Using only the reactor enclosure volume provides conservatism. Due to the RERS recirculation fans operating after 3 minutes, credit is taken for 50% mixing in Secondary Containment. Conservatively, no credit is taken for RERS filtration during the drawdown period.

The exhaust from Secondary Containment is filtered through the SGTS filter trains, following a 15.5-minute drawdown period. After drawdown, SGTS HEPA and charcoal filters are available to reduce the release activity. The North Stack release point is treated as a zero velocity vent release (ground level equivalent) for Control Room X/Q determination, and as ground level release for offsite dose assessment. Therefore, effectively, no elevated release is credited.

Based on the design and operation of the Containment Atmospheric Control System, and the Primary Containment Instrument Gas (PCIG) System, LGS does not routinely purge primary containment during power operations. Therefore, releases from containment purging prior to isolation during a DBA-LOCA are not considered. High volume purging is generally only used for inerting and de-Inerting for outages. Low volume purge lines are available for pressure or oxygen concentration control, and the PCIG System draws gas from the drywell for instrument gas to minimize pressure buildup.

Secondary containment bypass leakage potential has previously been evaluated. These conclusions continue to apply with application of AST. The only bypass leakage paths are Containment Penetrations 7 (Primary Steam) and 8 (Primary Steam Line Drain). Because of the use of the MSIV Leakage Alternate Drain Pathway, MSIV leakage bypasses secondary containment and is released through the seismically rugged Turbine Condenser System.

The radioactivity associated with all MSIV leakage is assumed to be released directly from the Primary Containment and into the Main Steam Lines. MSIV leakage has separate limits and a separately analyzed dose assessment, therefore it is not included in the L_a fraction limit, and is instead separately controlled. There are no changes to L_a as a result of implementing AST.

MSIV leakage assumed in this accident analysis is 200 scfh total for all steam lines and 100 scfh for any one line, consistent with the current Technical Specifications.

At upstream conditions, this results in a flow rate of:

100 scfh/line * 14.7 psia / (14.7 psia + 22 psig) / 60 min/hr = 0.668 cfm.

MSIV leakage testing is performed at 22 psig. Containment pressures above the MSIV test pressure persist for only about the first 6.5 minutes of the DBA-LOCA.

During this limited time period very little containment air is transported into the inboard piping and even less to outboard components. Informal test runs suggest that leakage during this period results in negligible dose contributions, even if an adjustment were made to extrapolate leakage to what might be expected if MSIVs were tested at the LGS P_a of 44 psig.

However, to provide design margin, the above leak rate is increased by 25% for the first 24 hours to a value of 0.834 cfm. This margin also allows MSIV leakage to be reduced by 50% at 24 hours.

Outboard flow rates are based on expansion of this fluid from the MSIV test pressure to atmospheric pressure, and by further expansion based on worst case heating the fluid to steam line temperatures from standard temperatures. Steam line temperatures are derived based on a generic BWR evaluation crediting only conduction through pipe walls and insulation. Credit is taken for temperature reductions only at 24 hours and at 96 hours.

The proposed change deletes the action statement requiring restoration of any MSIV that exceeds the 100 scfh limit to be restored to $\leq$ 11.5 scfh.

This requirement is unnecessary because:

- The action is not necessary to assure that MSIV leakage remains within design basis limits.
- Maintenance goals such as this are not typically controlled by technical specifications.
- MSIV restoration leakage goals have not been required in other AST submittals.
- MSIV repairs will typically involve consistent best effort practices in order to avoid the need for rework or earlier future repairs and resulting cost and dose implications.

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.

Determination of inboard steam line, outboard steam line and condenser effective filter efficiencies is calculated, using AEB-98-03, "Assessment of Radiological Consequences for the Perry Pilot Plant Application using the Revised (NUREG-1465) Source Term", formulations and settling and deposition velocities.

For this AST evaluation an ECCS liquid leak rate of 5 gpm is used. Per Regulatory Guide 1.183, Appendix A, Section 5.2, this value is 2 times any administrative limits used as part of the Program for control of "Primary Coolant Sources Outside Containment".

Suppression Pool pH was evaluated over the 30-day duration of the DBA LOCA. It was demonstrated that pH would remain above 7. Therefore, no iodine conversion to elemental with re-evolution is expected or considered in this calculation. This control of pH also significantly limits the potential for airborne release from (always 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 15 hours of the start of the LOCA.

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

LOCA Transport Inputs

The Limerick Control Room is designed with one filtered air intake. The CREFAS filtration system associated with this intake is assumed to be manually initiated within 30 minutes.

The Control Room HVAC ventilated volume is 126,000 ft^3. The total flow through the CREFAS filters is 3000 cfm. In the radiation isolation mode, 525 cfm is filtered outside air, and 2475 cfm is filtered recirculation flow from the control room. In the chlorine isolation mode the entire 3000 cfm is filtered recirculation flow. The assumed intake filter efficiencies are 99% for HEPA filtration of aerosols, and 80% for the charcoal adsorber for elemental and organic iodines. This analysis assumes, and therefore provides margin for, up to 525 cfm of unfiltered intake into the control room from the control room intake vicinity.

In the radiation isolation mode, the Control Room exfiltration is 1050 cfm. In the chlorine isolation mode, the exfiltration is 525 cfm.

During the 30 minute period before manual initiation of the CREFAS radiation isolation mode, the normal intake of 2100 cfm plus an assumed 525 cfm of unfiltered inleakage is assumed, so the exfiltration rate is 2625 cfm during this period.

LOCA Removal Inputs

For LGS, the RADTRAD computer program, including the Powers Natural Deposition algorithm based on NUREG/CR-6189, is used for modeling aerosol deposition in primary containment. No natural deposition is assumed for elemental or organic iodine. The lower bound (10%) level of deposition credit is used.

Modeling of aerosol settling and elemental iodine deposition is based on methodology used by NRC in AEB-98-03. For the two steam lines modeled, two nodes are used. The first node is from the reactor pressure vessel to the inboard MSIV. The second node is from the inboard MSIV to the Turbine Stop Valve that provides the seismically designed boundary of the MSIV alternate drain pathway. For aerosol settling, only horizontal piping runs are credited, and only the bottom surface area is considered available. Per AEB-98-03, a median settling velocity is used, given the conservatism in using a well-mixed treatment. For elemental iodine deposition both horizontal and vertical piping is credited, as well as all surfaces. This is because this deposition is not gravity dependent.

For conservatism, no credit is taken for deposition in the drain lines that provide the previously licensed MSIV alternate drain pathway to the condenser.

All MS drain lines are routed to a single penetration in the HP condenser at a point below the condenser tubing. Iodine resuspension from settled or deposited iodines is not calculated. Historically, this phenomena increased organic iodine release by about a factor of two based on resuspension of TID-14844 based elemental iodine fractions. The presence of this phenomenon is questionable with aerosols with significant cesium loadings. Furthermore, while deposition on condenser tubing is not formally credited, test cases have shown that substantial removal of elemental and even organic iodine would be predicted that would more than offset any resuspension. Flow rates out of the condenser are assumed to be at 120°F and atmospheric pressure. A factor of 1.25 is applied, as is done with leakage and flow through steam lines. This leak rate is also reduced by 50% after 24 hours, consistent with the change in Containment conditions.

4.3.1.2 MSLB Accident and Assumptions

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

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

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

- There is instantaneous release from the break to the environment. No holdup in the Turbine Enclosure or dilution by mixing with Turbine Enclosure air volume is credited.
- The activity in the steam cloud is based on the total mass of water released from the break, not just the portion that flashes to steam. This assumption is conservative because it considers the maximum release of fission products.
- The fraction of liquid water contained in steam, which carries activity into the cloud is conservatively assumed to be 2.0%.

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

4.3.1.3 FHA Analysis Inputs and Assumptions

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

The postulated FHA involves the drop of a fuel assembly on top of the reactor core during refueling operations. The bounding analysis assumes that 172 GE-14 fuel rods in the full core are damaged. A radial peaking factor of 1.7 was assumed in the analysis in addition to the source term corrections discussed in Section 4.2.1. A post-shutdown 24-hour decay period was used to determine the release activity inventory. This assumption is conservative when compared to actual plant refueling outage history. The analysis assumes that gap activity in the affected rods was released instantaneously into the water in the reactor well. The bounding fuel handling accident is one in which a fuel assembly is dropped from the highest position onto the core. This produces the maximum kinetic energy, which results in the maximum damage. The drop of a fuel assembly into the spent fuel pool will not generate as much fuel damage as that due to a drop into the vessel. The analysis assumes the fuel bundle is dropped into the vessel, but only assumes a water depth of 23 feet above the assemblies seated in the reactor pressure vessel. The slightly reduced decontamination factor due to spent fuel pool water coverage of less than 23 feet is offset by less fuel damage incurred in the spent fuel pool.

In accordance with Regulatory Guide 1.183, the analysis assumes that the activity in the Reactor Enclosure environment is released within two hours, through the vent (South Stack), as a zero velocity vent release with no further credit for Reactor Enclosure holdup or dilution, or SGTS operation.

The analysis does not credit CREFAS or control room isolation.

4.3.1.4 CRDA Analysis Inputs and Assumptions

The key inputs and assumptions used in the AST CRDA analysis are included in Table 8. The design basis CRDA involves the rapid removal of the highest worth control rod resulting in a reactivity excursion that encompasses the consequences of any other postulated CRDA. For the dose consequence analysis, it was assumed that 1,200 of the fuel rods in the core were damaged, with melting occurring in 0.77 percent of the damaged rods as specified in GE Report NEDE-31152P. A conservative core average radial peaking factor of 1.7 as recommended by the fuel manufacturer was used in the analysis. For releases from the breached fuel, 10% of the core inventory of noble gases and iodines are assumed to be in the fuel gap. For releases attributed to fuel melting, 100% of the noble gases and 50% of the iodines are assumed to be released to the reactor coolant.

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

The Main Condenser is assumed to leak activity into the Turbine Enclosure at a rate of 1% per day. This activity is then released, unfiltered, to the environment by way of the North Stack, taking no credit for holdup in the Turbine Enclosure.

The forced flow path through the Steam Jet Air Ejectors (SJAE) discharges to the off-gas system. This pathway is assessed crediting elimination of iodine releases and a delay of noble gas releases by the off-gas system charcoal delay beds. This credit is as currently used and licensed in conformance with NEDO-31400A.

Forced flow from the Mechanical Vacuum Pump (MVP) is prevented by trips initiated upon detection of high radiation levels by the Main Steam Line Radiation Monitors (MSLRM). Therefore, any activity in this system is held up in the Condenser, and this forced release path need not be considered.

Unfiltered release from the Turbine Enclosure is via the North Stack at the rate of 1.0% of the condenser activity per day for 24 hours.

For this analysis, as performed using the RADTRAD code, the LGS Unit 1 & 2 Control Room is modeled as a closed volume of 126,000 ft^3. Normal maximum flow into the Control Room of 2100 cfm, plus a conservative assumption of 1050 cfm for unknown unfiltered inleakage is used. Flow into the Control Room is therefore assumed to be 3150 cfm, and to balance the system for analytical purposes, an equal flow of air is considered to leave. No credit is taken for any filtration of flows into the Control Room.

The air that enters the Control Room originates from a source that is characterized by a dispersion factor, calculated using ARCON96. Following a CRDA, the MVPs are immediately de-energized, thereby isolating this forced flow path. LGS uses a clean steam system for sealing steam, and therefore steam seal leakage is not a forced flow path. The remaining activity, all of which is assumed to have accumulated in the condenser, leaks into the Turbine Enclosure at a rate of 1% per day. The subsequent release into the environment from the Turbine Enclosure is postulated to escape through the North Stack. The total dose in the Control Room over the 24-hour period is the result of the released activities that enter through the air intake. The methodologies significant to this analysis are the dose consequence analysis in NUREG/CR-6604 Section 2.3 and the Radioactive Decay Calculations, Section 2.4.3 of this reference.

The Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) χ/Q's have been determined and are located 731 meters and 2043 meters respectively, from the postulated release points. Having determined these dispersion factors, the total dose is modeled in RADTRAD 3.03 using the same nodal breakdown as used for determining the Control Room total dose.

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

4.4 RESULTS

4.4.1 Evaluation Results

4.4.1.1. *Accident Radiological Consequence Analyses*

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

4.4.1.2 LOCA Radiological Consequence Analyses

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

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

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

The post-accident doses are the result of the following four distinct contributors:

<u>Primary Containment to Secondary Containment Leakage</u>

The leakage, captured by the secondary containment (reactor enclosure), is exhausted as an unfiltered, zero-velocity vent release as analyzed during the 15.5-minute drawdown period. After this period, this activity is collected by the SGTS, and then released to the environment through the North Stack as a zero-velocity vent (ground level equivalent) release with filter credit.

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

<u>MSIV Leakage from the Primary Containment into the Environment</u>

The MSIV leakage is released via the alternate drain pathway, condenser and turbine seals, as an unfiltered, zero-velocity vent release from the North Stack. No credit for Turbine Enclosure holdup, upward velocity, or buoyancy at the North Stack is considered.

ECCS Leakage to Secondary Containment

This leakage is assumed to start immediately after the onset of a LOCA and continue for 30 days. A flashing fraction of 1.39% determined using a methodology previously approved for use at the Clinton Power Station, is used in the analysis. The flashed activity is collected by the SGTS prior to release to the environment except during the 15.5-minute drawdown period.

Dose Assessment from Sources External to Control Room

The doses from the following external sources were evaluated:

- Direct dose resulting from ECCS Core Spray and RHR piping adjacent to the control room.
- Direct shine from RERS, SGTS, and CREFAS filters, Secondary Containment cloud, and external clouds are negligible due to available shielding and/or distances.

4.4.1.3 MSLB Accident Radiological Consequence Analysis

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

4.4.1.4 FHA Radiological Consequence Analysis

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

4.4.1.5 CRDA Radiological Consequence Analysis

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

4.4.2 Atmospheric Dispersion Factors

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

Table 13 lists χ/Q values used for the control room dose assessments. The ground level release χ/Q values (i.e., LOCA-MSIV and FHA release) were calculated by the ARCON96 computer code. The separate χ/Q results of each of these two models were then analyzed according to the methodology in RG-1.194. These results are based on site-specific hourly meteorological data in a five-year period of record.

Tables 14a lists χ/Q values for the EAB and LPZ boundaries.

4.4.3 Post-Accident Suppression Pool Water Chemistry Management

The re-evolution of elemental iodine from the Suppression Pool is strongly dependent on pool pH. The analysis assumed that the borated solution was injected within 13 hours of the onset of a DBA LOCA and mixed within the Suppression pool. The modeling of the LGS containment cabling maximized the production of hydrochloric acid. The analysis demonstrated that the Suppression Pool pH remains above 7 for the 30-day LOCA duration. The final pH and other related parameters are presented in Table 15.

4.4.4 Evaluation Conclusions

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

4.5 SUMMARY

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

Implementation of the AST provides the basis for several changes to the licensing and design bases for LGS. The principal changes affect HEPA and charcoal filtration credits, MSIV closure times, and refueling and fuel handling activities.

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

Figure 1: Layout of Release Points for LGS

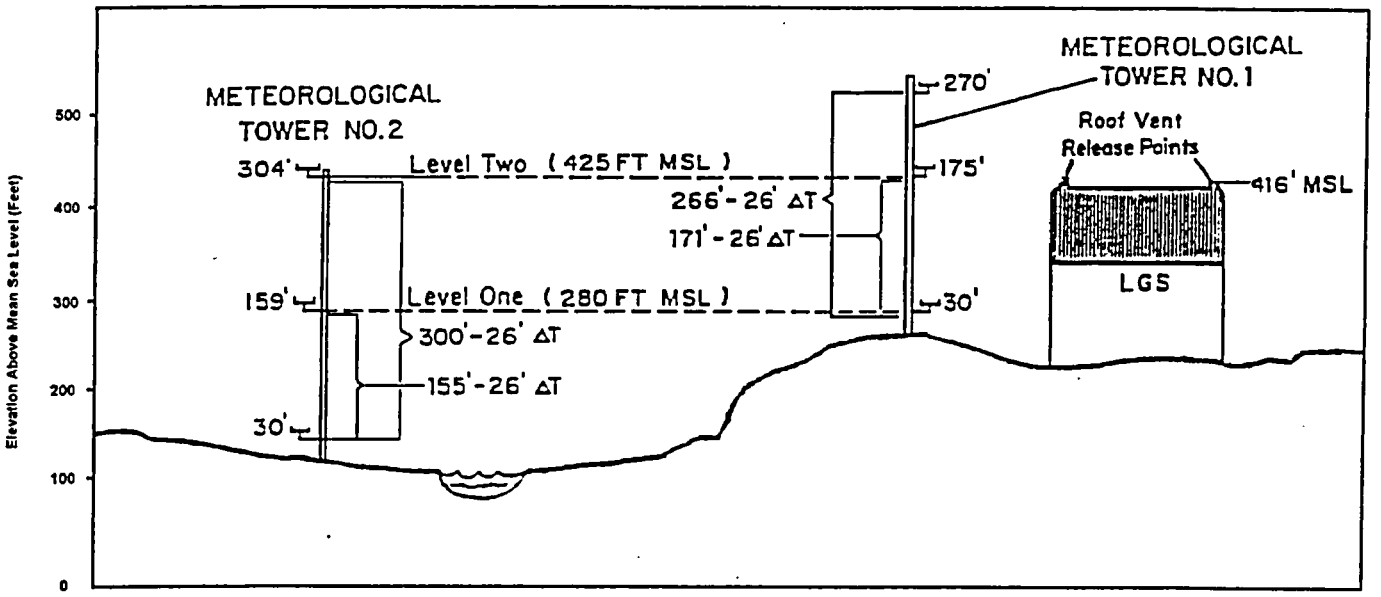

Figure 1 Parameters: Dimensional Data For Dispersion Analyses

Parameter	Value
Distance from North Stack to Control Room Intake	16.5 m
Direction, Control Room Intake to North Stack	180 degrees
Distance from South Stack to Control Room Intake	64.8 m
Direction, Control Room Intake to South Stack	180 degrees
Elevation at Plant Grade	217 feet above mean sea level (MSL) (216 feet outside of building)
Elevation at Center of Control Room Intake	340 feet above MSL
Elevation at Top of Exhaust Stacks	416 feet above MSL

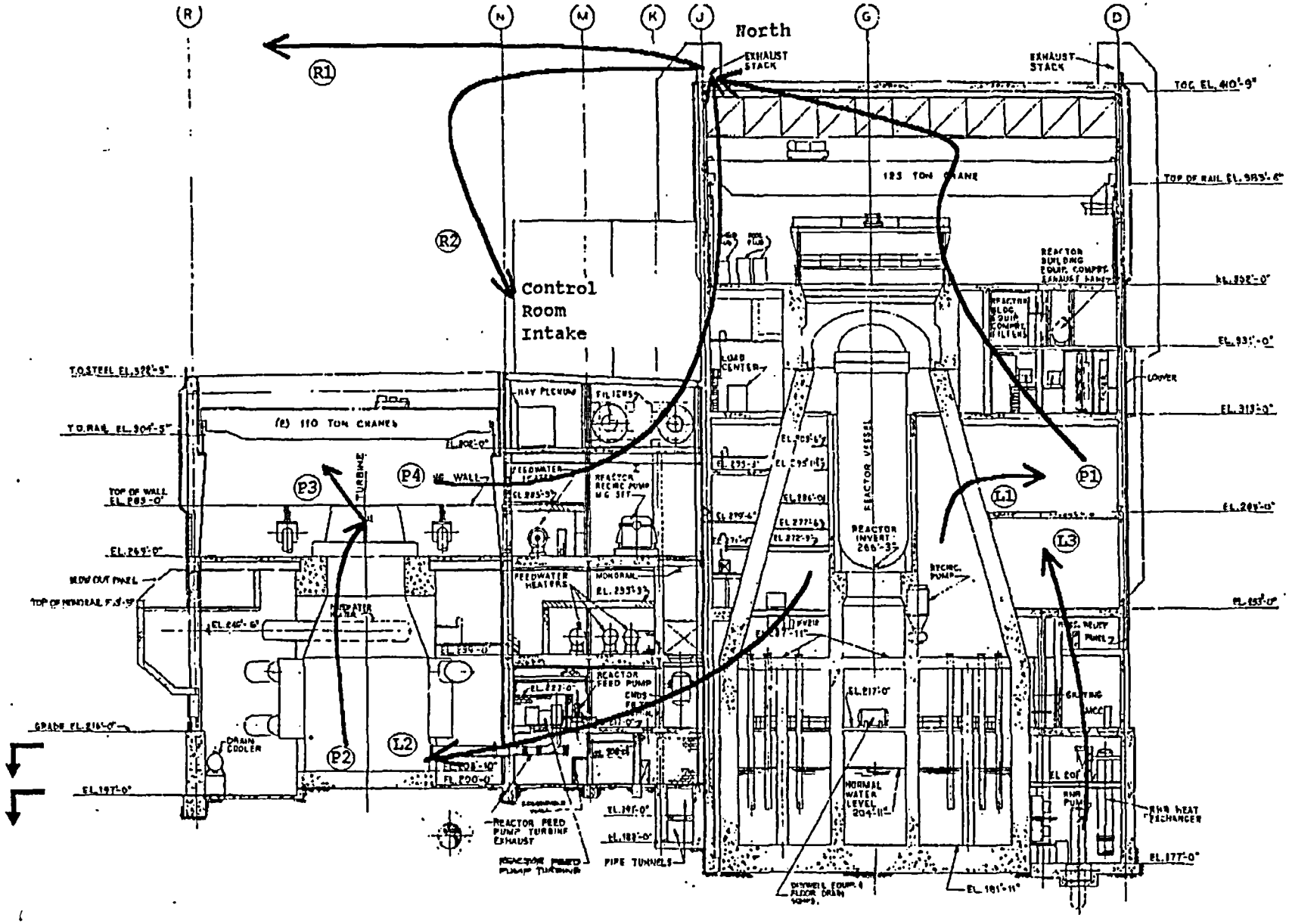
North
Control
Room
Intake
R1
R2
P1
P2
P3
P4
L1
L2
L3
R
N
M
K
J
G
D
EXHAUST STACK
EXHAUST STACK
TOG EL. 410'-9"
TOP OF RAIL EL. 383'-6"
183 TON CRANE
REACTOR BUILDING EQUIP. COMPT. EXHAUST FANS
EL. 332'-0"
EL. 331'-0"
LOUVER
EL. 313'-0"
EL. 285'-0"
EL. 255'-0"
LOAD CENTER
REACTOR VESSEL
REACTOR INVERT 266'-3"
RECIRC. PUMP
T.O. STEEL EL. 328'-9"
T.O. RAIL EL. 304'-9"
(E) 110 TON CRANES
HEAT PLENUM
FILTERS
TOP OF WALL EL. 289'-0"
EL. 265'-0"
BLOW OUT PANEL
TURBINE
FEEDWATER HEATERS
REACTOR RECIRC PUMP MG SET
MONORAIL
REACTOR FEED PUMP
CNDS FILTER
GRADE EL. 216'-0"
DRAIN COOLER
EL. 197'-0"
REACTOR FEED PUMP TURBINE EXHAUST
REACTOR FEED PUMP TURBINE
PIPE TUNNELS
NORMAL WATER LEVEL 204'-11"
EL. 181'-11"
EL. 177'-0"
RHR HEAT EXCHANGER
RHR PUMP
EL. 217'-0"
EL. 227'-0"
EL. 197'-0"
EL. 189'-0"

Figure 2

Radioactivity Transport Pathways

Figure 2 Parameters: LOCA Leakage Rates and Secondary Containment Mixing Parameters		
Path	**Description**	**Parameters & Values**
L_1	Primary Containment Leakage to Secondary Containment	Leak Rate: L_a = 0.50 %/day, 0 - 15.5 min after start of gap release. Release is unfiltered through the North Stack during drawdown period. L_a = 0.50 %/day, 15.5 min - 24hr Release is SGTS filtered through North Stack. 0.5 x L_a = 0.25 %/day, 1 - 30days Release is SGTS filtered through North Stack.
P_1	Release from Secondary Containment to Environment through SGTS Filter	Mixing by RERS in 50% of Secondary Containment volume. RERS and SGTS filtration not credited during drawdown.
L_2	MSIV Leakage to Condenser Environment	Leak Rate: Based on LLRT acceptance criterion of 200 scfh for all main steam (MS) lines: 100 scfh maximum for any one MS line when measured at greater than or equal to 22 psig; two steam lines are each treated as two-node well-mixed volumes each with 100 scfh flow, leak rate reduced by 50% after 24 hours. Release is from Turbine Enclosure, unfiltered through North Stack. Analytically, flow is direct from the condenser to the North Stack
P_2	Leakage Well Mixed in HP Turbine Shell. No credit for transport to other shells through available opening. No credit for substantial plateout potential on Condenser Tubing.	
P_3	Leak from Turbine Shaft Seals to Turbine enclosure Atmosphere	
P_4	Flow from Turbine enclosure to North Stack. No credit for mixing or holdup or deposition in Turbine enclosure. No filtration is provided for this flow.	
L_3	ECCS Leakage (Supp. Pool Water Source) to Secondary Containment	Leak Rate: 5 gpm, 0 - 15.5 min (2x administrative limit) and release is unfiltered through North Stack during drawdown period. 5 gpm, 15.5 min - 30days (2x administrative limit) and release is SGTS filtered through North Stack.
P_1	Release from Secondary Containment to Environment through SGTS Filter	Mixing by RERS in 50% of Secondary Containment volume. RERS and SGTS filtration not credited during drawdown.
R_1	Release to Environment and then Control Room Intake from the Environment	2100 cfm normal unfiltered intake, plus 525 cfm of unfiltered inleakage for the first 30 minutes; 525 cfm CREFAS filtered intake, plus 525 cfm of unfiltered inleakage thereafter. CREFAS filters credited at 99% for aerosols (based on HEPA) and 80% for charcoal absorbers (for elemental and organic iodine.)

Figure 2 Parameters: LOCA Leakage Rates and Secondary Containment Mixing Parameters		
Path	**Description**	**Parameters & Values**
	Control Room Exhaust to Environment	2625 cfm for the first 30 min of LOCA, before CREFAS is initiated, and 1050 cfm thereafter, to balance with intake and inleakage
R_2	Release to Environment for Offsite Dose Assessment Purposes	R_1 and R_2 include Primary Containment to Secondary Containment, ECCS Leakage, and MSIV Leakage related releases

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

Table 2a: Key Analysis Inputs and Assumptions	
Release Inputs - LOCA Radionuclide Source Term	
Input/Assumption	**Value**
Core Fission Product Inventory	ORIGEN-2.1 Only the 60 nuclides considered by RADTRAD are utilized in the analysis
Core Power Level	3,527 MWt
Core Burnup	711 EFPD per 2-year cycle

Fission Product Release Fractions for LOCA	**RG 1.183, Table 1** **BWR Core Inventory Fraction Released Into Containment** Group / Gap Release Phase / Early In-vessel Phase / Total Noble Gases / 0.05 / 0.95 / 1.0 Halogens / 0.05 / 0.25 / 0.3 Alkali Metals / 0.05 / 0.20 / 0.25 Tellurium Metals / 0.00 / 0.05 / 0.05 Ba, Sr / 0.00 / 0.02 / 0.02 Noble Metals / 0.00 / 0.0025 / 0.0025 Cerium Group / 0.00 / 0.0005 / 0.0005 Lanthanides / 0.00 / 0.0002 / 0.0002
Fission Product Release Timing (Per RG 1.183, the release phases are modeled sequentially)	**RG 1.183, Table 4** **LOCA Release Phases** BWRs Phase / Onset / Duration Gap Release / 2 min / 0.5 hr Early In-Vessel / 0.5 hr / 1.5 hr

Table 2b: Key Analysis Inputs and Assumptions	
Release Inputs - Non-LOCA Radionuclide Source Term	
Input/Assumption	**Value**
Core Fission Product Inventory	ORIGEN-2.1 Only the 60 nuclides considered by RADTRAD are utilized in the analysis
Core Power Level	3,527 MWt
Core Burnup	711 EFPD (per 2-year cycle)
Fission Product Gap Release Fractions for Non-LOCA Accidents	**RG 1.183, Table 3** **Non-LOCA Fraction of Fission Product Inventory in Gap** Group / Fraction I-131 / 0.08 Kr-85 / 0.10 Other Noble Gases / 0.05 Other Halogens / 0.05 Alkali Metals / 0.12

Table 3: Key LOCA Analysis Inputs and Assumptions	
Release Inputs - Primary and Secondary Containment Parameters	
Input/Assumption	**Value**
Containment Free Volume Drywell: Suppression Pool Airspace: Total Calculated Volume:	 243,580 ft^3 159,540 ft^3 403,120 ft^3
Minimum Suppression Pool Water Volume	118,655 cubic feet
Reactor Coolant Volume	13,108 cubic feet at 552.6°F or 9,663 cubic feet at 95.0°F
Primary Containment Total Leak Rate	0.5% per day for first 24 hours (L_a) 0.25% per day 24 – 720 hours
Total MSIV leak rate	200 scfh total for all 4 steam lines; 100 scfh for any single line
Secondary Containment Volume	1,800,000 cubic feet
Fraction of Secondary Containment Available for Mixing	0.5
SGTS Flow Rate	3,000 cfm pre-drawdown 2,500 cfm post-drawdown
SGTS Filter Efficiencies	HEPA: 97.5 Charcoal: 97.5
RERS Flow Rate	60,000 cfm (rated) 30,000 cfm (credited in analysis)
RERS Filter Efficiencies	HEPA: 70% Charcoal: 70%
Secondary Containment Drawdown Time	15.5 minutes
Secondary Containment Bypass	None, except MSIV
ECCS Systems Leak Rate Outside of Primary Containment (includes factor of 2)	5 gpm
ECCS Leakage Duration	0-30 days
Release Pathways ECCS/Containment Leakage MSIV Leakage	Location North Stack (ground release) North Stack (ground release)
Release Pathways ECCS/Containment Leakage MSIV Leakage	Duration 0-30 days 12 hours to 30 days

Table 4: Key LOCA Analysis Inputs and Assumptions	
Transport Inputs - Control Room Parameters	
Input/Assumption	**Value**
Nuclide Release Locations	See Figure 2
CREFAS System Initiation	Credit for the Control Room ventilation hi-radiation signal is removed. A 30-minute operator response to isolate the Control Room HVAC system is credited.
Control Room Free Volume	126,000 cubic feet
Control Room Flow Rates	Normal mode 2100 cfm unfiltered intake 525 cfm unfiltered inleakage 0 cfm recirculation flow 2625 cfm exfiltration Radiation isolation mode 525 cfm filtered intake 525 cfm unfiltered inleakage 2,475 cfm filtered recirculation 1050 cfm exfiltration Chlorine isolation mode 525 cfm unfiltered inleakage 3000 cfm filtered recirculation 525 cfm exfiltration
Elemental and Organic Iodine Removal Efficiencies	80%
Aerosols Removal Efficiency	99%

Table 5: Key LOCA Analysis Inputs and Assumptions	
Removal Inputs	
Input/Assumption	Value
Containment Spray removal	Not credited
Aerosol Natural Deposition Coefficients Used in the Containment	Credit is taken for natural deposition of aerosols based on equations for the Power's model in NUREG/CR 6189 and input directly by RADTRAD as natural deposition time dependent lambdas. No credit is assumed for natural deposition of elemental or organic iodine, or for suppression pool scrubbing.
SGTS Filter Efficiencies – Elemental and Organic Iodine Aerosols	SGTS HEPA filters and charcoal adsorbers are credited after drawdown. HEPA: 97.5% Charcoal: 97.5%
RERS Filter Efficiencies	HEPA: 70% Charcoal: 70%
Deposition/Plate-out (where credited)	Deposition based on AEB-98-03 well-mixed model and associated median settling velocity. Only horizontal piping is credited, and the bottom half as the settling area. For elemental iodine, deposition velocities from AEB-98-03 are used and all piping and surfaces credited.
Main Steam Line and Condenser Holdup Holdup Credit for MSIV Leakage	Modeling of aerosol settling and elemental iodine deposition is based on methodology used by NRC in AEB-98-03. For the two steam lines modeled, two nodes are used. The first node is from the reactor pressure vessel to the inboard MSIV. The second node is from the inboard MSIV to the Turbine Stop Valve that provides the seismically designed boundary of the MSIV Leakage Control System. For aerosol settling, only horizontal piping runs are credited, and only the bottom surface area is considered available. Per AEB-98-03, a median settling velocity is used, given the conservatism in using a well-mixed treatment. For elemental iodine deposition both horizontal and vertical piping is credited, as well as all surfaces. This is because this deposition is not gravity dependent

Table 6: Key MSLB Accident Analysis Inputs and Assumptions	
Input/Assumption	**Value**
Break Discharge Mass Release	For a 5.5 second MSIV closure time: 103,785 pounds (20,452 as steam and 83,333 as liquid) For a 10.5 second MSIV closure time: 206,933 pounds (20,452 as steam and 186,481 as liquid)
Pre-Accident Spike Iodine Concentration	4.0 μCi/gm I-131 equivalent
Maximum Equilibrium Iodine Concentration	0.2 μCi/gm I-131 equivalent
Transport Model for Control Room	Steam cloud moves past the Control Room intake at 1 m/sec
Turbine Enclosure Holdup/Control Room Filtration	Not credited

Table 7: Key FHA Analysis Inputs and Assumptions	
Input/Assumption	**Value**
Core Damage	172 fuel rods failed based on GE14 fuel and "Heavy Mast"
Radial Peaking Factor	1.7
Fuel Decay Period	24 hours
Iodine Decontamination Factor	DF = 200
Release Period	2 hours
Refuel Floor Air Removal Rate	6 air changes per hour to assure activity exhaust within 2 hours
Control Room Filtration	Not Credited
Control Room Intake Flow	2100 cfm normal intake plus 1050 cfm unfiltered inleakage
Release Location	South Stack unfiltered, zero-velocity vent release (Ground Level equivalent)
CREFAS/SGT System Initiation	Not Credited

Table 8: Key CRDA Analysis Inputs and Assumptions	
Input/Assumption	**Value**
Core Damage	1,200 fuel rods failed (66,720 fuel rods in core)
Percent of Damaged Fuel with Melt	0.77%
Radial Peaking Factor	1.7
Mechanical Vacuum Pump (MVP) Operation Before Isolation	MVP isolation by MSLRM Hi Rad signal
Condenser Leak Rate	1% of condenser activity per day throughout entire release period
Release Period	24 hours
Forced Flow Paths	None. Main Steam Line Radiation Monitor high radiation causes Mechanical Vacuum Pump trip and isolation, if required. LGS is a "clean sealing steam" plant so gland sealing steam is not a forced flow path. If no reactor isolation (MSIVs are not closed), the forced flow path is through the Steam Jet Air Ejectors (SJAE) discharge to the off-gas system. This pathway is assessed crediting elimination of iodine releases and a delay of noble gas releases by the off-gas system charcoal delay beds. This credit is as currently used and licensed in conformance with NEDO-31400A.
Release Location	See Figure 1 Zero-velocity vent release through the North Stack
CREFAS System Initiation	Not credited

Table 9: LOCA Radiological Consequence Analysis			
Location	**Duration**	**TEDE (rem)**	**Regulatory Limit TEDE (rem)**
Control Room	30 days	4.01*.	5
EAB	Maximum, 2 hours	0.90	25
LPZ	30 days	1.25	25

*The doses here are for the bounding radiation isolation mode with 525 cfm of filtered fresh air intake for pressurization and 525 cfm of unfiltered inleakage. The calculated total Control Room dose for the unpressurized chlorine isolation mode with 525 cfm of unfiltered inleakage is lower.

Table 10: MSLB Accident Radiological Consequence Analysis (10.5 Second MSIV Closure Time)				
Location	Duration	4.0 µCi/gm Dose Equivalent I-131 TEDE (rem)	0.2 µCi/gm Dose Equivalent I-131 TEDE (rem)	Regulatory Limit TEDE (rem)
Control Room	30-day integrated dose	3.61	0.18	5
EAB	Worst 2-hour integrated dose	2.82	0.14	25 (4.0 µCi/gm) 2.5 (0.2 µCi/gm)
LPZ	30-day integrated dose	1.11	0.056	25 (4.0 µCi/gm) 2.5 (0.2 µCi/gm)

Table 11: FHA Radiological Consequence Analysis			
Location	Duration	TEDE (rem)	Regulatory Limit TEDE (rem)
Control Room	30 days	2.52	5
EAB	Maximum, 2 hours	0.88	6.3
LPZ	30 days	0.32	6.3

Table 12: CRDA Radiological Consequence Analysis			
Location	Duration	TEDE (rem)	Regulatory Limit TEDE (rem)
Control Room	30 days	1.62	5
EAB	Maximum, 2 hours	0.049	6.3
LPZ	30 days	0.034	6.3

Table 13: Control Room χ/Q Values for the Different Release and Intake Combinations			
Time Period	χ/Q (sec/m³)		
	LOCA North Stack	FHA	CRDA
0 - 2 hrs	6.88E-03	1.26E-03	6.88E-03
2 - 8 hrs	5.17E-03	-	5.17E-03
8 - 24 hrs	2.04E-03	-	2.04E-03
1 - 4 days	1.29E-03	-	-
4-30 days	9.63E-04	-	-

Notes:

1. Control room intake χ/Q values are applicable for control room inleakage.
2. For MSLB, specific Control Room χ/Q values are not calculated. For conservatism, the calculated hemispherical steam plume volume is transported, without dilution, over the Control Room intake location for the duration required for plume transit at a wind speed of 1 meter per second.

Table 14a: North & South Stack (Ground Level Release) For LOCA, FHA, and CRDA χ/Q (sec/m³) Values Using RG 1.145 Methodology for the EAB and LPZ		
Time Period	EAB χ/Q (sec/m³)	LPZ χ/Q (sec/m³)
0 - 2 hrs	3.18E-04	1.15E-04
0 - 8 hrs	-	5.79E-05
8 - 24 hrs	-	4.10E-05
1 - 4 days	-	1.95E-05
4 - 30 days	-	6.68E-06

Notes:

1. For MSLB, offsite χ/Q values are determined using Regulatory Guide 1.5 methodology, and are 4.77E-04 sec/m³ for the EAB and 1.89E-04 sec/m³ for the LPZ.

Table 15: Suppression Pool pH Results (Based on a total boron weight of 240 lbm)	
Condition	Value
Initial Suppression Pool pH	5.3
SLC injection time	Required Sodium Pentaborate to be injected within 13 hours
Suppression Pool pH throughout the 30-day accident duration (with SLC injection)	Greater than 7

REGULATORY GUIDE 1.183 COMPARISON

Table A: Conformance with Regulatory Guide (RG) 1.183 Main Sections			
RG Section	RG Position	LGS 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 and worst case inventory used for the selected isotopes. These values were then converted to units of Ci/MWt. Accident analyses are based on a 3527 MWt power level, based on the current accident analysis design basis allowance for instrument uncertainty. Source terms are based on a 2 year fuel cycle with a nominal 711 EFPD per cycle.
3.1	For the DBA LOCA, all fuel assemblies in the core are assumed to be affected and the core average inventory should be used. For DBA events that do not involve the entire core, the fission product inventory of each of the damaged fuel rods is determined by dividing the total core inventory by the number of fuel rods in the core. To account for differences in power level across the core, radial peaking factors from the facility's core operating limits report (COLR) or technical specifications should be applied in determining the inventory of the damaged rods.	Conforms	Peaking factors of 1.7 are used for DBA events that do not involve the entire core, with fission product inventories for damaged fuel rods determined by dividing the total core inventory by the number of fuel rods in the core.
3.1	No adjustment to the fission product inventory should be made for events postulated to occur during power operations at less than full rated power or those postulated to occur at the beginning of core life. For events postulated to occur while the facility is shutdown, e.g., a fuel handling accident, radioactive decay from the time of shutdown may be modeled.	Conforms	No adjustments for less than full power are made in any analyses.
3.2	The core inventory release fractions, by radionuclide groups, for the gap release and early in-vessel damage phases for DBA LOCAs	Conforms	The fractions from Regulatory Position 3.1, Table 1 are used.

Table A: Conformance with Regulatory Guide (RG) 1.183 Main Sections			
RG Section	RG Position	LGS Analysis	Comments
	are listed in Table 1 for BWRs and Table 2 for PWRs. These fractions are applied to the equilibrium core inventory described in Regulatory Position 3.1. **Table 1** **BWR Core Inventory Fraction Released Into Containment** **Group** / **Gap Release Phase** / **Early In-Vessel Phase** / **Total** Noble Gases 0.05 0.95 1.0 Halogens 0.05 0.25 0.3 Alkali Metals 0.05 0.20 0.25 Tellurium Metals 0.00 0.05 0.05 Ba, Sr 0.00 0.02 0.02 Noble Metals 0.00 0.0025 0.0025 Cerium Group 0.00 0.0005 0.0005 Lanthanides 0.00 0.0002 0.0002 Footnote 10: *The release fractions listed here have been determined to be acceptable for use with currently approved LWR fuel with a peak rod burnup up to 62,000 MWD/MTU. The data in this section may not be applicable to cores containing mixed oxide (MOX) fuel.*		Footnote 10 criteria are met.

Table A: Conformance with Regulatory Guide (RG) 1.183 Main Sections			
RG Section	RG Position	LGS Analysis	Comments
3.2	For non-LOCA events, the fractions of the core inventory assumed to be in the gap for the various radionuclides are given in Table 3. The release fractions from Table 3 are used in conjunction with the fission product inventory calculated with the maximum core radial peaking factor. **Table 3** **Non-LOCA Fraction of Fission Product Inventory in Gap** **Group** / **Fraction** I-131 / 0.08 Kr-85 / 0.10 Other Noble Gases / 0.05 Other Halogens / 0.05 Alkali Metals / 0.12 Footnote 11: *The release fractions listed here have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup up to 62,000 MWD/MTU provided that the maximum linear heat generation rate does not exceed 6.3 kw/ft peak rod average power for rods with burnups that exceed 54 GWD/MTU. As an alternative, fission gas release calculations performed using NRC-approved methodologies may be considered on a case-by-case basis. To be acceptable, these calculations must use a projected power history that will bound the limiting projected plant-specific power history for the specific fuel load. For the BWR rod drop accident and the PWR rod ejection accident, the gap fractions are assumed to be 10% for iodines and noble gases.*	Conforms	Complies with Note 11 of Table 3. Peaking factor of 1.7 used for DBA events that do not involve the entire core.
3.3	Table 4 tabulates the onset and duration of each sequential release phase for DBA LOCAs at PWRs and BWRs. The specified onset is the time following the initiation of the accident (i.e., time = 0). The early in-vessel phase immediately follows the gap release phase. The activity released from the core during each release phase should be modeled as increasing in a linear fashion over the duration of the phase. For non-LOCA DBAs, in which fuel damage	Conforms	The BWR durations from Table 4 are used. LOCA is modeled in a linear fashion. Non-LOCA is modeled as an instantaneous release.

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

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

Table A: Conformance with Regulatory Guide (RG) 1.183 Main Sections			
RG Section	RG Position	LGS Analysis	Comments
	The factors in the column headed "effective" yield doses corresponding to the CEDE.		
4.1.3	For the first 8 hours, the breathing rate of persons offsite should be assumed to be 3.5×10^{-4} cubic meters per second. From 8 to 24 hours following the accident, the breathing rate should be assumed to be 1.8×10^{-4} cubic meters per second. After that and until the end of the accident, the rate should be assumed to be 2.3×10^{-4} cubic meters per second.	Conforms	The values that correspond to the rounded values in Section 4.1.3 of RG 1.183 are used.
4.1.4	The DDE should be calculated assuming submergence in semi-infinite cloud assumptions with appropriate credit for attenuation by body tissue. The DDE is nominally equivalent to the effective dose equivalent (EDE) from external exposure if the whole body is irradiated uniformly. Since this is a reasonable assumption for submergence exposure situations, EDE may be used in lieu of DDE in determining the contribution of external dose to the TEDE. Table III.1 of Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water, and Soil" (Ref. 21), provides external EDE conversion factors acceptable to the NRC staff. The factors in the column headed "effective" yield doses corresponding to the EDE.	Conforms	Federal Guidance Report 12 conversion factors are used.
4.1.5	The TEDE should be determined for the most limiting person at the EAB. The maximum EAB TEDE for any two-hour period following the start of the radioactivity release should be determined and used in determining compliance with the dose criteria in 10 CFR 50.67. The maximum two-hour TEDE should be determined by calculating the postulated dose for a series of small time increments and performing a "sliding" sum over the increments for successive two-hour periods. The maximum TEDE obtained is submitted. The time increments should appropriately reflect the progression of the accident to capture the peak dose interval between the start of the event and the end of radioactivity release (see also Table 6). Footnote 14: *With regard to the EAB TEDE, the maximum two-hour value is the*	Conforms	The maximum two-hour LOCA EAB dose starts as follows: <u>PC Leakage</u>: 3.2 hours <u>MSIV Leakage</u>: 10.4 hours <u>ECCS Leakage</u>: 3.2 hours Conservatively, the maximum 2-hour period dose was determined by adding the maximum 2-hour dose for each of the components listed above even though they do not occur simultaneously.

Table A: Conformance with Regulatory Guide (RG) 1.183 Main Sections			
RG Section	RG Position	LGS Analysis	Comments
	basis for screening and evaluation under 10 CFR 50.59. Changes to doses outside of the two-hour window are only considered in the context of their impact on the maximum two-hour EAB TEDE.		
4.1.6	TEDE should be determined for the most limiting receptor at the outer boundary of the low population zone (LPZ) and should be used in determining compliance with the dose criteria in 10 CFR 50.67.	Conforms	This guidance is applied in the analyses.
4.1.7	No correction should be made for depletion of the effluent plume by deposition on the ground.	Conforms	No such corrections made in the analyses.
4.2.1	The TEDE analysis should consider all sources of radiation that will cause exposure to control room personnel. The applicable sources will vary from facility to facility, but typically will include: Contamination of the control room atmosphere by the intake or infiltration of the radioactive material contained in the radioactive plume released from the facility, Contamination of the control room atmosphere by the intake or infiltration of airborne radioactive material from areas and structures adjacent to the control room envelope, Radiation shine from the external radioactive plume released from the facility, Radiation shine from radioactive material in the reactor containment, Radiation shine from radioactive material in systems and components inside or external to the control room envelope, e.g., radioactive material buildup in recirculation filters.	Conforms	The principal source of dose within the control room is due to airborne activity. The dose estimates from post LOCA primary containment and sources external to the control room indicate that contribution to dose is dominated by ECCS piping in the Reactor Enclosure adjacent to the Control Room.
4.2.2	The radioactive material releases and radiation levels used in the control room dose analysis should be determined using the same source term, transport, and release assumptions used for determining the EAB and the LPZ TEDE values, unless these assumptions would result in non-conservative results for the control room.	Conforms	The source term, transport, and release methodology is the same for both the control room and offsite locations.

Table A: Conformance with Regulatory Guide (RG) 1.183 Main Sections			
RG Section	RG Position	LGS Analysis	Comments
4.2.3	The models used to transport radioactive material into and through the control room, and the shielding models used to determine radiation dose rates from external sources, should be structured to provide suitably conservative estimates of the exposure to control room personnel.	Conforms	This guidance is applied in the analyses.
4.2.4	Credit for engineered safety features that mitigate airborne radioactive material within the control room may be assumed. Such features may include control room isolation or pressurization, or intake or recirculation filtration. Refer to Section 6.5.1, "ESF Atmospheric Cleanup System," of the SRP (Ref. 3) and Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Post-accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants" (Ref. 25), for guidance.	Conforms	For the LOCA (after the drawdown period), credit is taken for SGTS HEPA and charcoal adsorber filtration (97.5% each) and RERS HEPA and charcoal filtration (70% for each). Control Room and intake and recirculation filtration by CREFAS are credited in the LOCA accident analysis. Radiation isolation mode has been analyzed with manual initiation within 30 minutes. After this period, credit is taken for 99% HEPA and 80% charcoal adsorber efficiencies. No filtration credit is taken in the FHA, MSLB or CRDA accident analyses.
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	Standard occupancy factors and breathing rate are used throughout the 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	Conforms	The equation given is utilized for finite cloud correction when calculating external doses due to the airborne activity inside

Table A: Conformance with Regulatory Guide (RG) 1.183 Main Sections			
RG Section	RG Position	LGS Analysis	Comments
	difference between finite cloud geometry in the control room and the semi-infinite cloud assumption used in calculating the dose conversion factors. The following expression may be used to correct the semi-infinite cloud dose, DDE_{∞}, to a finite cloud dose, DDE_{finite}, where the control room is modeled as a hemisphere that has a volume, V, in cubic feet, equivalent to that of the control room (Ref. 22). $DDE_{finite} = \frac{DDE_{\infty} V^{0.338}}{1173}$		the control room.
4.3	The guidance provided in Regulatory Positions 4.1 and 4.2 should be used, as applicable, in re-assessing the radiological analyses identified in Regulatory Position 1.3.1, such as those in NUREG-0737 (Ref. 2). Design envelope source terms provided in NUREG-0737 should be updated for consistency with the AST. In general, radiation exposures to plant personnel identified in Regulatory Position 1.3.1 should be expressed in terms of TEDE. Integrated radiation exposure of plant equipment should be determined using the guidance of Appendix I of this guide.	Conforms	Based on an evaluation, the existing TID-14844 based analyses included in section 1.13 of the UFSAR are shown to be conservative and bounding. Given compliance with the GDC-19 limit of 5 REM when dose is based on TID-14844 source terms, compliance with 10 CFR 50.67 Control Room dose limits can be expected with the AST-based analysis. Therefore, the historically analyzed cases are sufficient and no additional analysis of vital areas of LGS are necessary.
5.1.1	The evaluations required by 10 CFR 50.67 are re-analyses of the design basis safety analyses and evaluations required by 10 CFR 50.34; they are considered to be a significant input to the evaluations required by 10 CFR 50.92 or 10 CFR 50.59. These analyses should be prepared, reviewed, and maintained in accordance with quality assurance programs that comply with Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50.	Conforms	These analyses were prepared as specified in the guidance.
5.1.2	Credit may be taken for accident mitigation features that are classified as safety-related, are required to be operable by technical specifications, are powered by emergency power sources, and are either automatically actuated or, in limited cases, have actuation	Conforms based on acceptable risk assessments	The analyses take credit for SLC System operation. The SLC System is safety-related, required to be operable by Technical Specifications, and supplied with

Table A: Conformance with Regulatory Guide (RG) 1.183 Main Sections			
RG Section	RG Position	LGS Analysis	Comments
	requirements explicitly addressed in emergency operating procedures. The single active component failure that results in the most limiting radiological consequences should be assumed. Assumptions regarding the occurrence and timing of a loss of offsite power should be selected with the objective of maximizing the postulated radiological consequences.		emergency power. The SLC System is manually initiated from the main control room, as directed by the emergency operating procedures. Due to having a common flow path with inline check valves located inside containment, SLC is not fully single-failure proof although it has a high level of redundancy regarding system flow paths and active components (e.g., multiple pumps, suction paths, and explosive injection valves).
5.1.3	The numeric values that are chosen as inputs to the analyses required by 10 CFR 50.67 should be selected with the objective of determining a conservative postulated dose. In some instances, a particular parameter may be conservative in one portion of an analysis but be non-conservative in another portion of the same analysis.	Conforms	Conservative assumptions are used.
5.1.4	Licensees should ensure that analysis assumptions and methods are compatible with the AST and the TEDE criteria.	Conforms	Analysis assumptions and methods were made per this guidance.
5.3	Atmospheric dispersion values (χ/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 χ/Q values are documented in Regulatory Guides 1.3 and 1.4, Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," and the paper, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19". The NRC computer code PAVAN implements Regulatory Guide 1.145 and its use is acceptable to the NRC staff. The methodology of the NRC computer code ARCON96 is generally acceptable to the NRC staff for use in determining control room χ/Q values.	Conforms	New atmospheric dispersion values (χ/Q) for the EAB, the LPZ, and the control room were developed, using meteorological data for the years 1996-2000. ARCON96 and PAVAN were used with these data to determine control room and EAB/LPZ atmospheric dispersion values. Control room χ/Qs from releases from the North and South Stacks were developed in conformance with RG-1.194.

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

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

Table B: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	LGS Analysis	Comments
	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).		
3.4	Reduction in airborne radioactivity in the containment by in-containment recirculation filter systems may be credited if these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. A-5 and A-6). The filter media loading caused by the increased aerosol release associated with the revised source term should be addressed.	Not Applicable	No in-containment recirculation filter systems exist at LGS.
3.5	Reduction in airborne radioactivity in the containment by suppression pool scrubbing in BWRs should generally not be credited. However, the staff may consider such reduction on an individual case basis. The evaluation should consider the relative timing of the blowdown and the fission product release from the fuel, the force driving the release through the pool, and the potential for any bypass of the suppression pool (Ref. 7). Analyses should consider iodine re-evolution if the suppression pool liquid pH is not maintained greater than 7.	Conforms	No credit is taken for suppression pool scrubbing in the LOCA AST reanalysis. Analyses have been performed that determined that the suppression pool liquid pH is maintained greater than 7, and that, therefore, iodine re-evolution is not expected.
3.6	Reduction in airborne radioactivity in the containment by retention in ice condensers, or other engineering safety features not addressed above, should be evaluated on an individual case basis. See Section 6.5.4 of the SRP (Ref. A-1).	Not Applicable	LGS 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	Conforms	The LGS Mark II primary containment leakage is assumed to be 0.5% of containment mass per day for 24 hours and 0.25% per day

Table B: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	LGS Analysis	Comments
	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 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.		from 24 to 720 hours based on containment pressure reductions.
3.8	If the primary containment is routinely purged during power operations, releases via the purge system prior to containment isolation should be analyzed and the resulting doses summed with the postulated doses from other release paths. The purge release evaluation should assume that 100% of the radionuclide inventory in the reactor coolant system liquid is released to the containment at the initiation of the LOCA. This inventory should be based on the technical specification reactor coolant system equilibrium activity. Iodine spikes need not be considered. If the purge system is not isolated before the onset of the gap release phase, the release fractions associated with the gap release and early in-vessel phases should be considered as applicable.	Conforms	The LGS primary containment is not routinely purged during power operation. Purging is limited to inerting, de-inerting and occasional short pressure control activities.
4.1	Leakage from the primary containment should be considered to be collected, processed by engineered safety feature (ESF) filters, if any, and released to the environment via the secondary containment exhaust system during periods in which the secondary containment has a negative pressure as defined in technical specifications. Credit for an elevated release should be assumed only if the point of physical release is more than two and one-half times the height of any adjacent structure.	Conforms	Secondary Containment release is via the North Stack. Filtration credit is taken after the 15.5-minute drawdown period. The Gap release begins at ~ 2 minutes after LOCA initiation. For EAB and LPZ doses, ground level releases are assumed. For Control Room doses, releases are based on zero-velocity vent

Table B: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	LGS Analysis	Comments
			release assumptions, yielding ground level release equivalent dispersion factors.
4.2	Leakage from the primary containment is assumed to be released directly to the environment as a ground-level release during any period in which the secondary containment does not have a negative pressure as defined in technical specifications.	Conforms	For EAB and LPZ doses, ground level releases are assumed. For Control Room doses, releases are based on zero-velocity vent release assumptions.
4.3	The effect of high wind speeds on the ability of the secondary containment to maintain a negative pressure should be evaluated on an individual case basis. The wind speed to be assumed is the 1-hour average value that is exceeded only 5% of the total number of hours in the data set. Ambient temperatures used in these assessments should be the 1-hour average value that is exceeded only 5% or 95% of the total numbers of hours in the data set, whichever is conservative for the intended use (e.g., if high temperatures are limiting, use those exceeded only 5%).	Conforms	The wind speed exceeded only 5% of the time at LGS in the secondary containment vicinity is approximately 19.7 mph (175' elevation of meteorological tower 2). It has been determined that a wind speed of greater than 35 mph would be required before the secondary containment pressures would be positive relative to outside air pressures at the downwind side of the reactor enclosure.
4.4	Credit for dilution in the secondary containment may be allowed when adequate means to cause mixing can be demonstrated. Otherwise, the leakage from the primary containment should be assumed to be transported directly to exhaust systems without mixing. Credit for mixing, if found to be appropriate, should generally be limited to 50%. This evaluation should consider the magnitude of the containment leakage in relation to contiguous building volume or exhaust rate, the location of exhaust plenums relative to projected release locations, the recirculation ventilation systems, and internal walls and floors that impede stream flow between the release and the exhaust.	Conforms	A 50% mixing credit is taken for dilution/mixing in secondary containment. This mixing is attributed to the RERS flow network.

Table B: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	LGS Analysis	Comments
4.5	Primary containment leakage that bypasses the secondary containment should be evaluated at the bypass leak rate incorporated in the technical specifications. If the bypass leakage is through water, e.g., via a filled piping run that is maintained full, credit for retention of iodine and aerosols may be considered on a case-by-case basis. Similarly, deposition of aerosol radioactivity in gas-filled lines may be considered on a case-by-case basis.	Conforms	No primary containment leakage, with the exception of MSIV leakage, has been identified which bypasses the secondary containment. Only the MSIV pathway leak rates are incorporated into the Technical Specifications. The AST analysis is based on an MSIV leakage limit of 200 scfh total leakage with not more that 100 scfh per line when tested at ≥ 22 psig. Modeling of aerosol settling and elemental iodine deposition is based on methodology used by NRC in AEB-98-03 for both piping and the condenser. For the two steam lines modeled, two nodes are used. The first node is from the reactor pressure vessel to the inboard MSIV. The second node is from the inboard MSIV to the Turbine Stop Valve that provides the seismically designed boundary of the MSIV alternate drain pathway. For aerosol settling, only horizontal piping runs are credited, and only the bottom surface area is considered available. A median settling velocity is used, given the conservatism in using a well-mixed treatment. For elemental iodine deposition both horizontal and vertical piping is credited, as well as all surfaces. This is because this

Table B: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	LGS Analysis	Comments
			deposition is not gravity dependent. LGS has previously been analyzed and licensed to no longer credit a MSIV Leakage Control System, and to credit holdup, settling, and deposition in seismically rugged portions of the turbine condenser system. This historical evaluation is based on methodology described in NEDC-31858P, Rev. 2, "BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems". That analysis was based on a design basis recirculation line break and TID-14844 based source terms. For this AST application, the analysis of MSIV leakage is updated to reflect AST parameters related to release timing, chemical makeup, and more recent approaches regarding fission product settling and deposition. Credit is taken for deposition in the condenser, where the deposition area is the horizontal surface of the high pressure (HP) wetwell, and the HP condenser walls. By the time that activity has reached the condenser, the aerosols are essentially depleted. Therefore, vertical wall surfaces are credited for elemental iodine removal. No credit

Table B: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	LGS Analysis	Comments
			is taken for any organic iodine removal in piping or the condenser. Also, no credit is taken for condenser tubing even though this provides approximately 40 times the deposition area in the condenser than what is credited.
4.6	Reduction in the amount of radioactive material released from the secondary containment because of ESF filter systems may be taken into account provided that these systems meet the guidance of Regulatory Guide 1.52 (Ref. A-5) and Generic Letter 99-02 (Ref. A-6).	Conforms	SGTS HEPA and charcoal adsorber filters are credited in the evaluation of a LOCA accident for onsite and offsite dose consequences. RERS HEPA filtration is also credited. Both of these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02.
5.1	With the exception of noble gases, all the fission products released from the fuel to the containment (as defined in Tables 1 and 2 of this guide) should be assumed to instantaneously and homogeneously mix in the primary containment sump water (in PWRs) or suppression pool (in BWRs) at the time of release from the core. In lieu of this deterministic approach, suitably conservative mechanistic models for the transport of airborne activity in containment to the sump water may be used. Note that many of the parameters that make spray and deposition models conservative with regard to containment airborne leakage are non-conservative with regard to the buildup of sump activity.	Conforms	With the exception of noble gases, all the fission products released from the fuel to the containment are assumed to instantaneously and homogeneously mix in the suppression pool at the time of release from the core.
5.2	The leakage should be taken as two times the sum of the simultaneous leakage from all components in the ESF recirculation systems above which the technical specifications, or licensee commitments to item III.D.1.1 of NUREG-0737 (Ref. A-8), would require declaring such systems inoperable. The leakage should be assumed to start at the earliest time the recirculation flow occurs in these systems and end at the latest time the releases from these systems are terminated. Consideration should also be given to design leakage through valves	Conforms	The 5-gpm leak rate is two times the sum of the allowed simultaneous leakage from all ECCS components. ECCS leakage is minimized at LGS through implementation of the Program committed to in T.S. 6.8.4.a, "Primary Coolant Sources Outside Containment".

Table B: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	LGS Analysis	Comments
	isolating ESF recirculation systems from tanks vented to atmosphere, e.g., emergency core cooling system (ECCS) pump minflow return to the refueling water storage tank.		Since certain ECCS systems take suction immediately from the suppression pool, this leak path is assumed to start at time 0. Leakage to atmospheric tanks is credible only for lines connecting from ECCS pump discharges to such a tank, because of relative elevations. The sole leakage paths to a tank vented to atmosphere meeting this condition are the High Pressure Coolant Injection / Reactor Core Isolation Cooling test lines that discharge to the Condensate Storage Tank (CST). These lines are isolated by two normally closed valves. Since the CST contents are demineralized water, ECCS leakage would quickly turn the water basic. Therefore, minimal elemental iodine is expected, and as a result, negligible iodine volatilization.
5.3	With the exception of iodine, all radioactive materials in the recirculating liquid should be assumed to be retained in the liquid phase.	Conforms	With the exception of iodine, all radioactive materials in ECCS liquids are assumed to be retained in the liquid phase.
5.4	If the temperature of the leakage exceeds 212°F, the fraction of total iodine in the liquid that becomes airborne should be assumed equal to the fraction of the leakage that flashes to vapor. This flash fraction, FF, should be determined using a constant enthalpy, h, process, based on the maximum time-dependent temperature of the sump water circulating outside the containment:	Not Applicable	The temperature of the leakage does not exceed 212°F.

Table B: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	LGS Analysis	Comments
	$FF = \frac{h_{f1} - h_{f2}}{h_{fg}}$ *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.		
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	An airborne release fraction of 1.39% is used and was determined using a methodology previously approved for use at the Clinton Power Station. Suppression Pool water pH is maintained above 7 for the entire 30 days of the accident dose assessment period. Under these conditions virtually none of the iodine will be in elemental form, and organic iodine formation will be inhibited. Because of the subcooled condition no flashing is expected.
5.6	The radioiodine that is postulated to be available for release to the environment is assumed to be 97% elemental and 3% organic. Reduction in release activity by dilution or holdup within buildings, or by ESF ventilation filtration systems, may be credited where applicable. Filter systems used in these applications should be evaluated against the guidance of Regulatory Guide 1.52 (Ref. A-5) and Generic Letter 99-02 (Ref. A-6).	Conforms	The credited Control Room intake charcoal and HEPA filters meet the requirements of RG 1.52 and Generic Letter 99-02. These are credited at 80% efficiency for elemental and organic iodines. Aerosol removal efficiencies are assumed to be 99% based on the HEPA/charcoal combination.
6.1	For the purpose of this analysis, the activity available for release via MSIV leakage should be assumed to be that activity determined to be in the drywell for evaluating containment leakage (see Regulatory Position 3). No credit should be assumed for activity reduction by the steam separators or by iodine partitioning in the reactor vessel.	Conforms	The activity released through the MSIVs is the same concentration as that used for evaluating Primary to Secondary Containment leakage. No credit is assumed for activity reduction by the steam separators or

Table B: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	LGS Analysis	Comments
			by iodine partitioning in the reactor vessel.
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 200 scfh for all steam lines and 100 scfh for any one line when tested at ≥ 22 psig. Reduction in leakage rates after 24 hours are based on post-accident containment pressures. No credit is taken for leakage rate reductions below 50% of the MSIV leakage limit.
6.3	Reduction of the amount of released radioactivity by deposition and plateout on steam system piping upstream of the outboard MSIVs may be credited, but the amount of reduction in concentration allowed will be evaluated on an individual case basis. Generally, the model should be based on the assumption of well-mixed volumes, but other models such as slug flow may be used if justified.	Conforms	See discussion under comments for section 4.5 of Table B.
6.4	In the absence of collection and treatment of releases by ESFs such as the MSIV leakage control system, or as described in paragraph 6.5 below, the MSIV leakage should be assumed to be released to the environment as an unprocessed, ground- level release. Holdup and dilution in the turbine building should not be assumed.	Conforms	Releases are assumed to be from the North Stack, without credit for holdup or dilution in the condenser or Turbine Enclosure. The zero-velocity vent release assumption is effectively a ground level release assumption.
6.5	A reduction in MSIV releases that is due to holdup and deposition in main steam piping downstream of the MSIVs and in the main condenser, including the treatment of air ejector effluent by offgas systems, may be credited if the components and piping systems used in the release path are capable of performing their safety function during and following a safe shutdown earthquake (SSE). The amount of reduction allowed will be evaluated on an individual case basis. References A-9 and A-10 provide guidance on acceptable models.	Conforms	See discussion under comments for section 4.5 of Table B.
7.0	The radiological consequences from post-LOCA primary containment purging as a combustible gas or pressure control measure should be	Conforms	Containment purging as a combustible gas or pressure control

Table B: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	LGS Analysis	Comments
	analyzed. If the installed containment purging capabilities are maintained for purposes of severe accident management and are not credited in any design basis analysis, radiological consequences need not be evaluated. If the primary containment purging is required within 30 days of the LOCA, the results of this analysis should be combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA. Reduction in the amount of radioactive material released via ESF filter systems may be taken into account provided that these systems meet the guidance in Regulatory Guide 1.52 (Ref. A-5) and Generic Letter 99-02 (Ref. A-6).		measure is not required nor credited in any design basis analysis for 30 days following a design basis LOCA at LGS.

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

Table C: Conformance with Regulatory Guide 1.183 Appendix B (Fuel Handling Accident)			
RG Section	RG Position	LGS Analysis	Comments
	organic species. If the depth of water is not 23 feet, the decontamination factor will have to be determined on a case-by-case method (Ref. B-1).		An overall DF of 200 is used. For a drop over the spent fuel pool, coverage over the dropped fuel assembly, as it lies across the top of the bail handles of assemblies in the fuel rack, is 21.6 feet. The coverage over the fuel pins for assemblies in the racks is 22.6 feet. The calculated df, weighted by damaged fuel pin count, is 171. The conservatively determined damage over the spent fuel pool is 70% of that over the reactor vessel. Therefore, the net effect (damage times df), is that a drop over the reactor is bounding.
3	The retention of noble gases in the water in the fuel pool or reactor cavity is negligible (i.e., decontamination factor of 1). Particulate radionuclides are assumed to be retained by the water in the fuel pool or reactor cavity (i.e., infinite decontamination factor).	Conforms	DF = 1 for noble gas isotopes.
4.1	The radioactive material that escapes from the fuel pool to the fuel building is assumed to be released to the environment over a 2-hour time period.	Conforms	The release is assumed to occur over a two hour period.
4.2	A reduction in the amount of radioactive material released from the fuel pool by engineered safety feature (ESF) filter systems may be taken into account provided these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. B-2, B-3). Delays in radiation detection, actuation of the ESF filtration system, or diversion of ventilation flow to the ESF filtration system(21) should be determined and accounted for in the radioactivity release analyses.	Conforms	No credit is taken for the Standby Gas Treatment System.

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

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

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

Table D: Conformance with Regulatory Guide 1.183 Appendix C (Control Rod Drop Accident)			
RG Section	RG Position	LGS Analysis	Comments
3.4	Of the activity that reaches the turbine and condenser, 100% of the noble gases, 10% of the iodine, and 1% of the particulate radionuclides are available for release to the environment. The turbine and condensers leak to the atmosphere as a ground- level release at a rate of 1% per day for a period of 24 hours, at which time the leakage is assumed to terminate. No credit should be assumed for dilution or holdup within the turbine building. Radioactive decay during holdup in the turbine and condenser may be assumed.	Conforms	The condenser leak rate of 1% per day for a period of 24 hours is assumed. All releases are assumed to be at ground level and based on zero-velocity vent release assumptions. Radioactive decay during holdup in the condenser is assumed.
3.5	In lieu of the transport assumptions provided in paragraphs 3.2 through 3.4 above, a more mechanistic analysis may be used on a case-by-case basis. Such analyses account for the quantity of contaminated steam carried from the pressure vessel to the turbine and condensers based on a review of the minimum transport time from the pressure vessel to the first main steam isolation (MSIV) and considers MSIV closure time.	Not Applicable	Sections 3.2 through 3.4 are used in the analysis.
3.6	The iodine species released from the reactor coolant within the pressure vessel should be assumed to be 95% CsI as an aerosol, 4.85% elemental, and 0.15% organic. The release from the turbine and condenser should be assumed to be 97% elemental and 3% organic.	Conforms	No credit for RERS, SGTS, or CREFAS filters is taken, and therefore variation in iodine species has no effect.
Foot-note 1	The activity assumed in the analysis should be based on the activity associated with the projected fuel damage or the maximum technical specification values, whichever maximizes the radiological consequences. In determining the dose equivalent I-131 (DE I-131), only the radioiodine associated with normal operations or iodine spikes should be included. Activity from projected fuel damage should not be included.	Conforms	Projected fuel damage is the limiting case.

Table D: Conformance with Regulatory Guide 1.183 Appendix C (Control Rod Drop Accident)			
RG Section	RG Position	LGS Analysis	Comments
Foot-note 2	If there are forced flow paths from the turbine or condenser, such as unisolated motor vacuum pumps or unprocessed air ejectors, the leakage rate should be assumed to be the flow rate associated with the most limiting of these paths. Credit for collection and processing of releases, such as by off gas or standby gas treatment, will be considered on a case-by-case basis.	Conforms	Upon detection of high radiation levels by the Main Steam Line Radiation Monitor system the mechanical vacuum pump trips. Air ejector flows are processed through the offgas system, which removes all iodines and substantially delays noble gases. LGS is a clean sealing steam system so gland seals are not considered a forced release pathway.

Table E: Conformance with Regulatory Guide 1.183 Appendix D (Main Steam Line Break)			
RG Section	RG Position	LGS Analysis	Comments
1	Assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide. The release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached.	Not Applicable	No fuel damage, release estimate based on coolant activity.
2	If no or minimal fuel damage is postulated for the limiting event, the released activity should be the maximum coolant activity allowed by technical specification. The iodine concentration in the primary coolant is assumed to correspond to the following two cases in the nuclear steam supply system vendor's standard technical specifications.	Conforms	No fuel damage is expected even with the extension of the MSIV closure time to 10.5 seconds. The 10.5 seconds is the same as that currently licensed for Peach Bottom, where no fuel damage is postulated. Per the current Limerick MSLB analysis, the core is uncovered for 60 seconds, which is well after the MSIV closure time. No fuel damage results during this event. By changing the MSIV closure time by 5 seconds for AST, there is no concern of uncovering the core. This is due to the swell that results from reactor depressurization that will maintain adequate core coverage until the MSIV isolation. LGS Technical Specifications limits the reactor coolant Dose Equivalent (DE) I-131 specific activity to 0.2 μCi/gm, with action to isolate all main steam lines if the reactor coolant DE I-131 specific activity exceeds 4.0 μCi/gm during Power Operation or

Table E: Conformance with Regulatory Guide 1.183 Appendix D (Main Steam Line Break)			
RG Section	RG Position	LGS Analysis	Comments
			Startup.
2.1	The concentration that is the maximum value (typically 4.0 µCi/gm DE I-131) permitted and corresponds to the conditions of an assumed pre-accident spike, and	Conforms	See Item 2 above.
2.2	The concentration that is the maximum equilibrium value (typically 0.2 µCi/gm DE I-131) permitted for continued full power operation.	Conforms	See Item 2 above.
3	The activity released from the fuel should be assumed to mix instantaneously and homogeneously in the reactor coolant. Noble gases should be assumed to enter the steam phase instantaneously.	Conforms	Mixing is per this guidance.
4.1	The main steam line isolation valves (MSIV) should be assumed to close in the maximum time allowed by technical specifications.	Conforms	See Item 2 above.
4.2	The total mass of coolant released should be assumed to be that amount in the steam line and connecting lines at the time of the break plus the amount that passes through the valves prior to closure.	Conforms	Mass of coolant released is per this guidance.
4.3	All the radioactivity in the released coolant should be assumed to be released to the atmosphere instantaneously as a ground-level release. No credit should be assumed for plateout, holdup, or dilution within facility buildings.	Conforms	This guidance was used in the analysis.
4.4	The iodine species released from the main steam line should be assumed to be 95% CsI as an aerosol, 4.85% elemental, and 0.15% organic.	Conforms	No filtration is credited, so the iodine species are irrelevant.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

Exelon Generation Company, LLC (Exelon) is requesting a revision to the Facility Operating Licenses for Limerick Generating Station, Units 1 and 2. Specifically, we are requesting a revision to the Technical Specifications and licensing and design bases to reflect the application of alternative source term (AST) assumptions.

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

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

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

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

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

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

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

The implementation of AST assumptions has been evaluated in revisions to the analyses of the following limiting design basis accidents (DBAs) at Limerick Generating Station (LGS):

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

Based upon the results of these analyses, it has been demonstrated that, with the requested changes, the dose consequences of these limiting events are within the regulatory guidance provided by the NRC for use with the AST. This guidance is presented in 10 CFR 50.67, 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.

For the proposed change to increase the MSIV closure time by 5 seconds (5.5 to 10.5 seconds), although nearly twice as much steam mass is released from the current assumptions, there is no significant net increase of dose to Control Room or offsite personnel. The effect of the increased release time is offset by the dose methodologies applied using AST. The increased closure time also impacts the volume of reactor coolant available to maintain core coverage; however, the swell that results from the reactor depressurization adequately covers the core throughout the time it takes to complete the MSIV isolation.

The AST methodology follows the guidance provided in Regulatory Guide 1.183 and conforms to the dose limits in 10 CFR 50.67. Even though these limits are not directly comparable to the previously specified whole body and thyroid requirements of GDC 19 and 10 CFR 100.11, the results of the AST analyses have demonstrated that the 10 CFR 50.67 limits are satisfied. Therefore, it is concluded that AST does not involve a significant increase in the consequences of an accident previously evaluated.

Implementation of AST provides increased operating margins for: RERS, SGTS, and CREFAS filtration efficiencies; MSIV closure time; and RERS flow. It also relaxes secondary containment integrity requirements while handling irradiated fuel that has decayed for greater than 24 hours and during core alterations. Automatic initiation of the radiation isolation mode for the control room is no longer credited in the accident analysis, which can relax some Technical Specification surveillance requirements.

The equipment affected by the proposed changes is mitigative in nature, and relied upon after an accident has been initiated. Application of the AST does result in changes to Updated Final Safety Analysis Report (UFSAR) functions (e.g., MSIV closure time, SLC system) and operation of various filtration systems. As a condition of application of AST, LGS is proposing to use the SLC system to control the Suppression Pool pH following a LOCA only. These changes have been included within the evaluations for these proposed changes. While the operation of various systems does change with the implementation of AST, the affected systems are not accident initiators. Application of the AST is not an initiator of a design basis accident. The proposed changes to the Technical Specifications (TS), while they revise certain performance requirements, do not require any physical changes to the plant.

Additionally, the proposed change to the SLC systém surveillance requirement (SR), to verify the weight of Boron-10, is equivalent to the current requirement to determine the minimum available weight of sodium pentaborate. This change performs the same chemical analysis determination as the current TS requirement, but more clearly identifies the boron weight available to mitigate the ATWS event, and, given the possible range of tank enrichments, also supports determining the minimum required weight of boron needed to control suppression pool pH for AST.

As a result, the proposed changes do not affect any of the parameters or conditions that could contribute to the initiation of any accidents. Relaxation of operability requirements during the specified conditions will not significantly increase the probability of occurrence of an accident previously analyzed. Since design basis accident initiators are not being altered by adoption of the AST, the probability of an accident previously evaluated is not affected.

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

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

The proposed changes do not involve a physical change to the plant. Implementation of AST provides increased operating margins for: RERS, SGTS, AND CREFAS filtration efficiencies; MSIV closure time; and RERS flow. It also relaxes secondary containment integrity requirements while handling irradiated fuel that has decayed for greater than 24 hours and during core alterations. Automatic initiation of the radiation isolation mode for the control room is no longer credited in the accident analysis, which can relax some Technical Specification surveillance requirements.

Similarly, the proposed changes do not require any physical changes to any structures, systems or components involved in the mitigation of any accidents. The sodium pentaborate requirement for the SLC system does not change. Therefore, no new initiators or precursors of a new or different kind of accident are created. New equipment or personnel failure modes that might initiate a new type of accident are not created as a result of the proposed changes.

As such, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

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

The proposed changes are associated with the implementation of a new licensing basis for LGS Design Basis Accidents (DBAs). Approval of the change from the original source term to a new source term taken from Regulatory Guide 1.183 is being requested. The results of the accident analyses, revised in support of the proposed changes, are subject to revised acceptance criteria. The analyses have been performed using conservative methodologies, as specified in Regulatory Guide 1.183. The dose consequences of these DBAs remain within the acceptance criteria presented in 10 CFR 50.67, "Accident Source Term", and Regulatory Guide 1.183.

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

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Conclusion

Exelon Generation Company, LLC (Exelon) concludes that the proposed changes present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of no significant hazards consideration is justified.

5.2 Applicable Regulatory Requirements/Criteria

The NRC's traditional methods (prior to the AST) for calculating the radiological consequences of design basis accidents are described in a series of regulatory guides and Standard Review Plan (SRP) chapters. That guidance was developed to be consistent with the TID-14844 source term and the whole body and thyroid dose guidelines stated in 10 CFR 100.11. Many of those analysis assumptions and methods are inconsistent with the ASTs and with the Total Effective Dose Equivalent (TEDE) criteria provided in 10 CFR 50.67. Regulatory Guide 1.183 provides assumptions and methods that are acceptable to the NRC staff for performing design basis radiological analyses using an AST. This guidance supersedes corresponding radiological analysis assumptions provided in the older regulatory guides and SRP chapters when used in conjunction with an approved AST and the TEDE criteria provided in 10 CFR 50.67.

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

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

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

6.0 ENVIRONMENTAL CONSIDERATION

Exelon Generation Company, LLC (Exelon) has evaluated the proposed changes against the criteria for identification of licensing and regulatory actions requiring

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

(i) The amendment involves no significant hazards consideration.

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

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

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

Dose Results (rem TEDE)				
Accident	EAB Doses and Limits		LPZ Doses and Limits	
	Dose	Limit	Dose	Limit
Loss of Coolant Accident	0.90	25	1.25	25
Main Steam Line Break	2.82(1) 0.14(2)	25(1) 2.5(2)	1.11(1) 0.056(2)	25(1) 2.5(2)
Control Rod Drop Accident	0.049	6.3	0.034	6.3
Fuel Handling Accident	0.88	6.3	0.32	6.3

Notes: (1) Based on a pre-accident spike concentration of 4.0 μCi/gm dose equivalent I-131.

(2) Based on a maximum equilibrium concentration of 0.2 μCi/gm dose equivalent I-131.

Adoption of the alternative source term and Technical Specification changes which implement certain conservative assumptions in the alternative source term analyses will not result in physical changes to the plant that could significantly alter the type or amounts of effluents that may be released offsite. Changes to operational parameters that could affect effluent releases have been demonstrated through analysis to satisfy regulatory requirements.

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

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

Control Room Dose Results (rem TEDE)		
Accident	**Dose**	**Limit**
Loss of Coolant Accident	4.02	5.0
Main Steam Line Break	3.61(1) 0.18(2)	5.0
Control Rod Drop Accident	1.62	5.0
Fuel Handling Accident	2.52	5.0

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

The implementation of the alternative source term has been evaluated in revisions to the analyses of the limiting design basis accidents at Limerick Generating Station, Units 1 and 2. These accidents include the control rod drop accident, fuel handling accident, loss of coolant accident, and main steam line break accident. Based upon the results of these analyses, it has been demonstrated that with the proposed changes, the dose consequences of these limiting events are within the regulatory guidance provided by the NRC for use with alternative source term (i.e., 10 CFR 50.67 and 10 CFR 50, Appendix A, General Design Criterion 19). Thus, there will be no significant increase in either individual or cumulative occupational radiation exposure.

7.0 REFERENCES

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

7.17 Generic Letter 99-02, Laboratory Testing of Nuclear-Grade Activated Charcoal, June 3,1999

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

7.19 Regulatory Guide 1.194; Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants; U.S. Nuclear Regulatory Commission; December 2001.

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

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

7.22 10 CFR 50.67, "Accident source term," December 23, 1999

7.23 10 CFR 50.90, "Application for amendment of license or construction permit", October 4, 1999

7.24 10 CFR 50, Appendix A, General Design Criterion 19

7.25 ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers,

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

7.27 NUREG/CR-6189, D.A. Powers et al, "A simplified Model of Aerosol Removal by Natural Processes in Reactor Containments," July 1996

7.28 NEDO-31400A, "Safety Evaluation for Eliminating the BWR MSIV Closure Function and Scram Function of the MSLRM", October 1992

7.29 NEDC-31858P, Rev. 2, "BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems", September 1993

ATTACHMENT 2

LIMERICK GENERATING STATION
UNITS 1 AND 2

Docket Nos. 50-352 & 50-353

License Nos. NPF-39 & NPF-85

License Amendment Request
"LGS Alternative Source Term Implementation"

Markup of Technical Specification Pages

UNIT 1	UNIT 2
1-2	1-2
1-6	1-6
1-7	1-7
3/4 1-19	3/4 1-19
3/4 1-20	3/4 1-20
3/4 3-16	3/4 3-16
3/4 3-31	3/4 3-31
3/4 3-64	3/4 3-64
3/4 3-65	3/4 3-65
3/4 3-66	3/4 3-66
3/4 3-67	3/4 3-67
3/4 4-23	3/4 4-23
3/4 6-3	3/4 6-3
3/4 6-47	3/4 6-47
3/4 6-50	3/4 6-50
3/4 6-52	3/4 6-52
3/4 6-53	3/4 6-53
3/4 6-55	3/4 6-55
3/4 6-56	3/4 6-56
3/4 7-3	3/4 7-3
3/4 7-4	3/4 7-4
3/4 7-5	3/4 7-5
3/4 7-6	3/4 7-6
3/4 7-7	3/4 7-7
3/4 8-9	3/4 8-9
3/4 8-14	3/4 8-14
3/4 8-14A	3/4 8-14A
3/4 8-20	3/4 8-19
	3/4 8-20

CORE ALTERATION

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

a) Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special moveable detectors (including undervessel replacement); and
b) Control rod movement, provided there are no fuel assemblies in the associated core cell.

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

CORE OPERATING LIMITS REPORT

1.7a The CORE OPERATING LIMITS REPORT (COLR) is the unit-specific document that provides the core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specifications 6.9.1.9 thru 6.9.1.12. Plant operation within these limits is addressed in individual specifications.

CRITICAL POWER RATIO

1.8 The CRITICAL POWER RATIO (CPR) shall be the ratio of that power in the assembly which is calculated by application of the (GEXL) correlation to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.

DOSE EQUIVALENT I-131

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

INSERT 1

DOWNSCALE TRIP SETPOINT (DTSP)

1.9a The downscale trip setpoint associated with the Rod Block Monitor (RBM) rod block trip setting.

Ē-AVERAGE DISINTEGRATION ENERGY

1.10 Ē shall be the average, weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling, of the sum of the average beta and gamma energies per disintegration, in MeV, for isotopes, with half lives greater than 15 minutes, making up at least 95% of the total noniodine activity in the coolant.

EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME

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

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PURGE - PURGING

1.31 PURGE or PURGING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

RATED THERMAL POWER

1.32 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3458 MWt.

REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY

1.33 REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY shall exist when:

a. All reactor enclosure secondary containment penetrations required to be closed during accident conditions are either:

1. Capable of being closed by an OPERABLE secondary containment automatic isolation system, or

2. Closed by at least one manual valve, blind flange, slide gate damper, or deactivated automatic valve secured in its closed position, except as provided by Specification 3.6.5.2.1.

b. All reactor enclosure secondary containment hatches and blowout panels are closed and sealed.

c. The standby gas treatment system is in compliance with the requirements of Specification 3.6.5.3.

d. The reactor enclosure recirculation system is in compliance with the requirements of Specification 3.6.5.4.

e. At least one door in each access to the reactor enclosure secondary containment is closed.

f. The sealing mechanism associated with each reactor enclosure secondary containment penetration, e.g., welds, bellows, or O-rings, is OPERABLE.

g. The pressure within the reactor enclosure secondary containment is less than or equal to the value required by Specification 4.6.5.1.1a.

REACTOR PROTECTION SYSTEM RESPONSE TIME

INSERT 2

1.34 REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

REFUELING FLOOR SECONDARY CONTAINMENT INTEGRITY

~~1.35~~ 1.36 REFUELING FLOOR SECONDARY CONTAINMENT INTEGRITY shall exist when:

a. All refueling floor secondary containment penetrations required to be closed during accident conditions are either:

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REFUELING FLOOR SECONDARY CONTAINMENT INTEGRITY (Continued)

1. Capable of being closed by an OPERABLE secondary containment automatic isolation system, or

2. Closed by at least one manual valve, blind flange, slide gate damper, or deactivated automatic valve secured in its closed position, except as provided by Specification 3.6.5.2.2.

b. All refueling floor secondary containment hatches and blowout panels are closed and sealed.

c. The standby gas treatment system is in compliance with the requirements of specification 3.6.5.3.

d. At least one door in each access to the refueling floor secondary containment is closed.

e. The sealing mechanism associated with each refueling floor secondary containment penetration, e.g., welds, bellows, or O-rings, is OPERABLE.

f. The pressure within the refueling floor secondary containment is less than or equal to the value required by Specification 4.6.5.1.2a.

REPORTABLE EVENT

1.37 1.36 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

ROD DENSITY

1.38 1.37 ROD DENSITY shall be the number of control rod notches inserted as a fraction of the total number of control rod notches. All rods fully inserted is equivalent to 100% ROD DENSITY.

SHUTDOWN MARGIN

1.39 1.38 SHUTDOWN MARGIN shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming all control rods are fully inserted except for the single control rod of highest reactivity worth which is assumed to be fully withdrawn and the reactor is in the shutdown condition; cold, i.e. 68°F; and xenon free.

SITE BOUNDARY

.40 1.39 The SITE BOUNDARY shall be that line as defined in Figure 5.1.3-1a.

~~1.40 (Deleted)~~

SOURCE CHECK

1.41 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

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3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

~~3.1.5 The standby liquid control system consisting of a minimum of two pumps and corresponding flow paths, shall be OPERABLE.~~

~~APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2~~

~~ACTION:~~

~~a. In OPERATIONAL CONDITION 1 or 2.~~

~~1. With only one pump and corresponding explosive valve OPERABLE, restore one inoperable pump and corresponding explosive valve to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.~~

~~2. With standby liquid control system otherwise inoperable, restore the system to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours~~

Insert 3

SURVEILLANCE REQUIREMENTS

4.1.5 The standby liquid control system shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying that:

1. The temperature of the sodium pentaborate solution is within the limits of Figure 3.1.5-1.

2. The available volume of sodium pentaborate solution is at least 3160 gallons.

3. The temperature of the pump suction piping is within the limits of Figure 3.1.5-1 for the most recent concentration analysis.

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SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days by:

1. Verifying the continuity of the explosive charge.

2. Determining by chemical analysis (Boron-10) and calculation* that the available weight of ~~sodium pentaborate~~ is greater than or equal to ~~3754~~ 185 lbs; the concentration of sodium pentaborate in solution is less than or equal to 13.8% and within the limits of Figure 3.1.5-1 and; the following equation is satisfied:

$$\frac{C}{13\%\ wt.} \times \frac{E}{29\ atom\ \%} \times \frac{Q}{86\ gpm} \geq 1$$

where

C = Sodium pentaborate solution (% by weight)
Q = Two pump flowrate, as determined per surveillance requirement 4.1.5.c.
E = Boron 10 enrichment (atom % Boron 10)

3. Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. Demonstrating that, when tested pursuant to Specification 4.0.5, the minimum flow requirement of 41.2 gpm per pump at a pressure of greater than or equal to 1230 ± 25 psig is met.

d. At least once per 24 months during shutdown by:

1. Initiating at least one of the standby liquid control system loops, including an explosive valve, and verifying that a flow path from the pumps to the reactor pressure vessel is available by pumping demineralized water into the reactor vessel. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch which has been certified by having one of the batch successfully fired. All injection loops shall be tested in 3 operating cycles.

2. Verify all heat-treated piping between storage tank and pump suction is unblocked.**

e. Prior to addition of Boron to storage tank verify sodium pentaborate enrichment to be added is ≥ 29 atom % Boron 10.

* This test shall also be performed anytime water or boron is added to the solution or when the solution temperature drops below the limits of Figure 3.1.5-1 for the most recent concentration analysis, within 24 hours after water or boron addition or solution temperature is restored.

** This test shall also be performed whenever suction piping temperature drops below the limits of Figure 3.1.5-1 for the most recent concentration analysis, within 24 hours after solution temperature is restored.

TABLE 3.3.2-1 (Continued)
ISOLATION ACTUATION INSTRUMENTATION
ACTION STATEMENTS

ΓION 20 - Be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ΓION 21 - Be in at least STARTUP with the associated isolation valves closed within 6 hours or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

'ION 22 - Be in at least STARTUP within 6 hours.

'ION 23 - In OPERATIONAL CONDITION 1 or 2, verify the affected system isolation valves are closed within 1 hour and declare the affected system inoperable. In OPERATIONAL CONDITION 3, be in at least COLD SHUTDOWN within 12 hours.

ION 24 - Restore the manual initiation function to OPERABLE status within 8 hours or close the affected system isolation valves within the next hour and declare the affected system inoperable or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

ION 25 - Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within 1 hour.

ION 26 - Close the affected system isolation valves within 1 hour.

TABLE NOTATIONS

Required when (1) handling RECENTLY IRRADIATED FUEL ~~irradiated fuel~~ in the ~~refueling area~~ secondary containment, or (2) ~~during CORE ALTERATIONS, or (3)~~ during operations with a potential for draining the reactor vessel with the vessel head removed and fuel in the vessel.

May be bypassed under administrative control, with all turbine stop valves closed.

During operation of the associated Unit 1 or Unit 2 ventilation exhaust system.

DELETED

A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter. Trip functions common to RPS Actuation Instrumentation are shown in Table 4.3.2.1-1. In addition, for the HPCI system and RCIC system isolation, provided that the redundant isolation valve, inboard or outboard, as applicable, in each line is OPERABLE and all required actuation instrumentation for that valve is OPERABLE, one channel may be placed in an inoperable status for up to 8 hours for required surveillance without placing the channel or trip system in the tripped condition.

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TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
7. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level Low, Low - Level 2	S	Q	R	1, 2, 3
b. Drywell Pressure## - High	S	Q	R	1, 2, 3
c.1. Refueling Area Unit 1 Ventilation Exhaust Duct Radiation - High	S	Q	R	*#
2. Refueling Area Unit 2 Ventilation Exhaust Duct Radiation - High	S	Q	R	*#
d. Reactor Enclosure Ventilation Exhaust Duct Radiation - High	S	Q	R	1, 2, 3
e. Deleted				
f. Deleted				
g. Reactor Enclosure Manual Initiation	N.A.	R	N.A.	1, 2, 3
h. Refueling Area Manual Initiation	N.A.	R	N.A.	*

RECENTLY IRRADIATED FUEL

*Required when (1) handling ~~irradiated fuel~~ in the ~~refueling area~~ secondary containment, or (2) ~~during CORE ALTERATIONS, or (3)~~ during operations with a potential for draining the reactor vessel with the vessel head removed and fuel in the vessel.

**When not administratively bypassed and/or when any turbine stop valve is open.

#During operation of the associated Unit 1 or Unit 2 ventilation exhaust system.

##These trip functions (2a, 6b, and 7b) are common to the RPS actuation trip function.

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TABLE 3.3.7.1-1

RADIATION MONITORING INSTRUMENTATION

INSTRUMENTATION	MINIMUM CHANNELS OPERABLE	APPLICABLE CONDITIONS	ALARM/TRIP SETPOINT	ACTION
1. Main Control Room Normal Fresh Air Supply Radiation Monitor	4	1,2,3,5 and *	1×10^{-5} µCi/cc (b)	70
2. Area Monitors				
a. Criticality Monitors				
1) Spent Fuel Storage Pool	2	(a)	≥ 5 mR/h and ≤20mR/h(b)	71
b. Control Room Direct Radiation Monitor	1	At All Times	N.A.(b)	73
3. Reactor Enclosure Cooling Water Radiation Monitor	1	At All Times	≤ 3 x Background(b)	72

RECENTLY IRRADIATED FUEL

TABLE 3.3.7.1-1 (Continued)

RADIATION MONITORING INSTRUMENTATION

TABLE NOTATIONS

*When ~~irradiated fuel~~ is being handled in the secondary containment.

(a) With fuel in the spent fuel storage pool.

(b) Alarm only.

ACTION STATEMENTS

ACTION 70 - With one monitor inoperable, restore the inoperable monitor to the OPERABLE status within 7 days or, within the next 6 hours, initiate and maintain operation of the control room emergency filtration system in the radiation isolation mode of operation.

With two or more of the monitors inoperable, within one hour, initiate and maintain operation of the control room emergency filtration system in the radiation mode of operation.

ACTION 71 - With one of the required monitor inoperable, assure a portable continuous monitor with the same alarm setpoint is OPERABLE in the vicinity of the installed monitor during any fuel movement. If no fuel movement is being made, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

ACTION 72 - With the required monitor inoperable, obtain and analyze at least one grab sample of the monitored parameter at least once per 24 hours.

ACTION 73 - With the required monitor inoperable, assure a portable alarming monitor is OPERABLE in the vicinity of the installed monitor or perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

or during operations with a potential for draining the reactor vessel with the vessel head removed and fuel in the vessel.

TABLE 4.3.7.1-1

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTATION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE IS REQUIRED
1. Main Control Room Normal Fresh Air Supply Radiation Monitor	S	Q	R	1, 2, 3, 5 and *
2. Area Monitors				
a. Criticality Monitors				
1) Spent Fuel Storage Pool	S	M	R	(a)
b. Control Room Direct Radiation Monitor	S	M	R	At All Times
3. Reactor Enclosure Cooling Water Radiation Monitor	S	M	R(b)	At All Times

TABLE 4.3.7.1-1 (Continued)

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATIONS

*When ~~irradiated fuel~~ is being handled in the secondary containment.

(a) With fuel in the spent fuel storage pool.

(b) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards (NBS) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

RECENTLY IRRADIATED FUEL

or during operations with a potential for draining the reactor vessel with the vessel head removed and fuel in the vessel.

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3/4.4.7 MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.4.7 Two main steam line isolation valves (MSIVs) per main steam line shall be OPERABLE with closing times greater than or equal to 3 and less than or equal to 5 seconds.

10

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one or more MSIVs inoperable:

   1. Maintain at least one MSIV OPERABLE in each affected main steam line that is open and within 8 hours, either:

      a) Restore the inoperable valve(s) to OPERABLE status, or

      b) Isolate the affected main steam line by use of a deactivated MSIV in the closed position.

   2. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.7 Each of the above required MSIVs shall be demonstrated OPERABLE by verifying full closure between 3 and 5 seconds when tested pursuant to Specification 4.0.5.

10

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

b. The combined leakage rate to be in accordance with the Primary Containment Leakage Rate Testing Program for all primary containment penetrations and all primary containment isolation valves that are subject to Type B and C tests, except for: main steam line isolation valves*, valves which are hydrostatically tested, and those valves where an exemption to Appendix J of 10 CFR 50 has been granted, and

c. The leakage rate to ~~≤11.5~~ ≤100 scf per hour for any main steam isolation valve that exceeds 100 scf per hour, and restore the combined maximum pathway leakage to ≤200 scf per hour, and

d. The combined leakage rate for all containment isolation valves in hydrostatically tested lines which penetrate the primary containment to less than or equal to 1 gpm times the total number of such valves,

prior to increasing the reactor coolant system temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The primary containment leakage rates shall be demonstrated to be in accordance with the Primary Containment Leakage Rate Testing Program, or approved exemptions, for the following:

a. Type A Test

b. Type B and C Tests (including air locks)

c. Main Steam Line Isolation Valves

d. Hydrostatically tested Containment Isolation Valves

* Exemption to Appendix "J" to 10 CFR Part 50.

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3/4.6.5 SECONDARY CONTAINMENT

REFUELING AREA SECONDARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.5.1.2 REFUELING AREA SECONDARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: ~~OPERATIONAL CONDITION *.~~ INSERT 4

ACTION:

RECENTLY IRRADIATED FUEL

Without REFUELING AREA SECONDARY CONTAINMENT INTEGRITY, suspend handling of ~~irradiated fuel~~ in the secondary containment, ~~CORE ALTERATIONS~~ and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.1.2 REFUELING AREA SECONDARY CONTAINMENT INTEGRITY shall be demonstrated by:

a. Verifying at least once per 24 hours that the pressure within the refueling area secondary containment is greater than or equal to 0.25 inch of vacuum water gauge.

b. Verifying at least once per 31 days that:

1. All refueling area secondary containment equipment hatches and blowout panels are closed and sealed.

2. At least one door in each access to the refueling area secondary containment is closed.

3. All refueling area secondary containment penetrations not capable of being closed by OPERABLE secondary containment automatic isolation dampers/valves and required to be closed during accident conditions are closed by valves, blind flanges, slide gate dampers or deactivated automatic dampers/valves secured in position.

c. At least once per 24 months:

Operating one standby gas treatment subsystem for one hour and maintaining greater than or equal to 0.25 inch of vacuum water gauge in the refueling area secondary containment at a flow rate not exceeding 764 cfm.

~~*Required when (1) irradiated fuel is being handled in the refueling area secondary containment, or (2) during CORE ALTERATIONS, or (3) during operations with a potential for draining the reactor vessel, with the vessel head removed and fuel in the vessel.~~

CONTAINMENT SYSTEMS

REFUELING AREA SECONDARY CONTAINMENT AUTOMATIC ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.5.2.2 The refueling area secondary containment ventilation system automatic isolation valves shall be OPERABLE.

APPLICABILITY: ~~OPERATIONAL CONDITION~~ INSERT 4

ACTION:

With one or more of the refueling area secondary containment ventilation system automatic isolation valves inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and within 8 hours either:

a. Restore the inoperable valves to OPERABLE status, or

b. Isolate each affected penetration by use of at least one deactivated valve secured in the isolation position, or

c. Isolate each affected penetration by use of at least one closed manual valve, blind flange or slide gate damper.

Otherwise, ~~in OPERATIONAL CONDITION*~~ suspend handling of ~~irradiated fuel~~ RECENTLY IRRADIATED FUEL in the refueling area secondary containment, ~~CORE ALTERATIONS~~ and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.2.2 Each refueling area secondary containment ventilation system automatic isolation valve shall be demonstrated OPERABLE:

a. Prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by cycling the valve through at least one complete cycle of full travel and verifying the specified isolation time.

b. At least once per 24 months by verifying that on a containment isolation test signal each isolation valve actuates to its isolation position.

c. By verifying the isolation time to be within its limit at least once per 92 days.

~~*Required when (1) irradiated fuel is being handled in the refueling area secondary containment, or (2) during CORE ALTERATIONS, or (3) during operations with a potential for draining the reactor vessel with the vessel head removed and fuel in the vessel.~~

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STANDBY GAS TREATMENT SYSTEM - COMMON SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.5.3 Two independent standby gas treatment subsystems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, and ~~*~~. INSERT 4

ACTION:

a. With one standby gas treatment subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days, or:

   1. In OPERATIONAL CONDITION 1, 2, or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   2. ~~In OPERATIONAL CONDITION *~~ INSERT 4, suspend handling of ~~irradiated fuel~~ RECENTLY IRRADIATED FUEL in the secondary containment, ~~CORE ALTERATIONS~~ and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

b. With both standby gas treatment subsystems inoperable ~~in OPERATIONAL CONDITION *~~, suspend handling of ~~irradiated fuel~~ in the secondary containment, ~~CORE ALTERATIONS~~ or operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3. are not applicable.

If in progress,

SURVEILLANCE REQUIREMENTS

4.6.5.3 Each standby gas treatment subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates with the heaters OPERABLE.

~~*Required when (1) irradiated fuel is being handled in the refueling area secondary containment, or (2) during CORE ALTERATIONS, or (3) during operations with a potential for draining the reactor vessel with the vessel head removed and fuel in the vessel.~~

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 24* months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the subsystem by:

1. Verifying that the subsystem satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 5764 cfm ± 10%.

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration of less than ~~0.175%~~ 1.25% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F), at a relative humidity of 70% and at a face velocity of 66 fpm.

3. Verify that when the fan is running the subsystem flowrate is 2800 cfm minimum from each reactor enclosure (Zones I and II) and 2200 cfm minimum from the refueling area (Zone III) when tested in accordance with ANSI N510-1980.

4. Verify that the pressure drop across the refueling area to SGTS prefilter is less than 0.25 inches water gage while operating at a flow rate of 2400 cfm ± 10%.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration of less than ~~0.175%~~ 1.25% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F), at a relative humidity of 70% and at a face velocity of 66 fpm.

d. At least once per 24 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 9.1 inches water gauge while operating the filter train at a flow rate of 8400 cfm ± 10%.

*Surveillance interval is an exception to the guidance provided in Regulatory Guide 1.52, Revision 2, March 1978.

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REACTOR ENCLOSURE RECIRCULATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.5.4 Two independent reactor enclosure recirculation subsystems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one reactor enclosure recirculation subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With both reactor enclosure recirculation subsystems inoperable, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hors.

SURVEILLANCE REQUIREMENTS

4.6.5.4 Each reactor enclosure recirculation subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates properly. (flow at a minimum of 30,000 cfm)

b. At least once per 24* months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the subsystem by:

1. Verifying that the subsystem satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, at rated ~~and the system~~ flow ~~rate is~~ (60,000 cfm ± 10%).

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration of less than ~~2.5%~~ 15% when tested in accordance with ATM D3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70%.

3. Verifying a subsystem flow rate ~~of 60,000 cfm ± 10%~~ during system operation when tested in accordance with ANSI N510-1980.

within a range of 30,000 cfm to 66,000 cfm

*Surveillance interval is an exception to the guidance provided in Regulatory Guide 1.52, Revision 2, March 1978.

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SURVEILLANCE REQUIREMENTS (Continued)

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows, the methyl iodide penetration of less than ~~2.5%~~ 15% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70%.

d. At least once per 24 months by:

1. Verifying that the pressure drop across the combined prefilter, upstream and downstream HEPA filters, and charcoal adsorber banks is less than 6 inches water gauge while operating the filter train at ~~a~~ rated flow ~~rate of~~ (60,000 cfm ± 10%) verifying that the prefilter pressure drop is less than 0.8 inch water gauge and that the pressure drop across each HEPA is less than 2 inches water gauge.

2. Verifying that the filter train starts and the isolation valves which take suction on and return to the reactor enclosure open on each of the following test signals:

a. Manual initiation from the control room, and

b. Simulated automatic initiation signal.

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter bank satisfies the inplace penetration and leakage testing acceptance criteria of less than 0.05% in ~~accordance with ANSI N510-1980~~ while operating the system at ~~a~~ rated flow ~~rate of~~ (60,000 cfm ± 10%)

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorber bank satisfies the inplace penetration and leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon ~~refrigerant test~~ gas while operating the system at ~~a~~ rated flow ~~rate of~~ (60,000 cfm ± 10%)

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EMERGENCY SERVICE WATER SYSTEM - COMMON SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least the following independent emergency service water system loops, with each loop comprised of:

a. Two OPERABLE emergency service water pumps, and

b. An OPERABLE flow path capable of taking suction from the emergency service water pumps wet pits which are supplied from the spray pond or the cooling tower basin and transferring the water to the associated Unit 1 and common safety-related equipment,

shall be OPERABLE: Two loops, in

a. ~~In~~ OPERATIONAL CONDITIONS 1, 2, and 3 ~~, two loops,~~

One loop, in

b. ~~In~~ OPERATIONAL CONDITIONS 4, 5, and ~~7, one loop.~~

INSERT 5

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, 5, and ~~7~~.

ACTION:

a. In OPERATION CONDITION 1, 2, or 3:

1. With one emergency service water pump inoperable, restore the inoperable pump to OPERABLE status within 45 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With one emergency service water pump in each loop inoperable, restore at least one inoperable pump to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3. With one emergency service water system loop otherwise inoperable, declare all equipment aligned to the inoperable loop inoperable**, restore the inoperable loop to OPERABLE status with at least one OPERABLE pump within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

~~*When handling irradiated fuel in the secondary containment.~~

**The diesel generators may be aligned to the OPERABLE emergency service water system loop provided confirmatory flow testing has been performed. Those diesel generators no aligned to the OPERABLE emergency service water system loop shall be declared inoperable and the actions of 3.8.1.1 taken.

ACTION: (Continued)

4. With three ESW pump/diesel generator pairs** inoperable, restore at least one inoperable ESW pump/diesel generator pair** to OPERABLE status within 72 hours, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

5. With four ESW pump/diesel generator pairs** inoperable, restore at least one inoperable ESW pump/diesel generator pair** to OPERABLE status within 8 hours, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 4 or 5:

1. With only one emergency service water pump and its associated flowpath OPERABLE, restore at least two pumps with at least one flow path to OPERABLE status within 72 hours or declare the associated safety related equipment inoperable and take the ACTION required by Specifications 3.5.2 and 3.8.1.2.

c. ~~In OPERATIONAL CONDITION~~ INSERT 5 :

1. With only one emergency service water pump and its associated flow path OPERABLE, restore at least two pumps with at least one flow path to OPERABLE status within 72 hours or verify adequate cooling remains available for the diesel generators required to be OPERABLE or declare the associated diesel generator(s) inoperable and take the ACTION required by Specification 3.8.1.2. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENT

4.7.1.2 At least the above required emergency service water-system loop(s) shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 24 months by verifying that:

1. Each automatic valve actuates to its correct position on its appropriate ESW pump start signal.

2. Each pump starts automatically when its associated diesel generator starts.

~~*When handling irradiated fuel in the secondary containment~~

**An ESW pump/diesel generator pair consists of an ESW pump and its associated diesel generator. If either an ESW pump or its associated diesel generator becomes inoperable, then the ESW pump/diesel generator pair is inoperable.

ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The spray pond shall be OPERABLE with:

a. A minimum pond water level at or above elevation 250' 10" Mean Sea Level, and

b. A pond water temperature of less than or equal to 88°F.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, 5, and ~~*~~. INSERT 5

ACTION:

With the requirements of the above specification not satisfied:

a. In OPERATIONAL CONDITION 1, 2, or 3, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. In OPERATIONAL CONDITION 4 or 5, declare the RHRSW system and the emergency service water system inoperable and take the ACTION required by Specifications 3.7.1.1 and 3.7.1.2.

c. ~~In OPERATIONAL CONDITION *~~, declare the emergency service water system inoperable and take the ACTION required by Specification 3.7.1.2. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.3 The spray pond shall be determined OPERABLE:

a. By verifying the pond water level to be greater than its limit at least once per 24 hours.

b. By verifying the water surface temperature (within the upper two feet of the surface) to be less than or equal to 88°F:

1. at least once per 4 hours when the spray pond temperature is greater than or equal to 80°F; and

2. at least once per 2 hours when the spray pond temperature is greater than or equal to 85°F; and

3. at least once per 24 hours when the spray pond temperature is greater than 32°F.

c. By verifying all piping above the frost line is drained:

1. within one (1) hour after being used when ambient air temperature is below 40°F; or

2. when ambient air temperature falls below 40°F if the piping has not been previously drained.

~~*When handling irradiated fuel in the secondary containment.~~

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3/4.7.2 CONTROL ROOM EMERGENCY FRESH AIR SUPPLY SYSTEM - COMMON SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.2 Two independent control room emergency fresh air supply system subsystems shall be OPERABLE.

APPLICABILITY: All OPERATIONAL CONDITIONS and ~~*~~. INSERT 5

ACTION:

a. In OPERATIONAL CONDITION 1, 2, or 3 with one control room emergency fresh air supply subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 4, 5, or ~~*~~: INSERT 5

1. With one control room emergency fresh air supply subsystems inoperable, restore the inoperable subsystem to OPERABLE status within 7 days or initiate and maintain operation of the OPERABLE subsystem in the radiation isolation mode of operation.

2. With both control room emergency fresh air supply subsystems inoperable, suspend ~~CORE ALTERATIONS,~~ handling of ~~irradiated fuel~~ RECENTLY IRRADIATED FUEL in the secondary containment and operations with a potential for draining the reactor vessel.

~~*~~ The provisions of Specification 3.0.3 are not applicable ~~in OPERATIONAL CONDITION*~~.

SURVEILLANCE REQUIREMENTS

4.7.2 Each control room emergency fresh air supply subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying the control room air temperature to be less than or equal to 85°F effective temperature.

b. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates with the heaters OPERABLE.

c. At least once per 24** months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the subsystem by:

1. Verifying that the subsystem satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 3000 cfm ± 10%.

~~* When irradiated fuel is being handled in the secondary containment.~~
** Surveillance interval is an exception to the guidance provided in Regulatory Guide 1.52, Revision 2, March 1978.

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration of less than ~~2.5%~~ 10% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70%.

3. Verifying a subsystem flow rate of 3000 cfm ± 10% during subsystem operation when tested in accordance with ANSI N510-1980.

d. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration of less than ~~2.5%~~ 10% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70%.

e. At least once per 24 months by:

1. Verifying that the pressure drop across the combined prefilter, upstream and downstream HEPA filters, and charcoal adsorber banks is less than 6 inches water gauge while operating the subsystem at a flow rate of 3000 cfm ± 10%; verifying that the prefilter pressure drop is less than 0.8 inch water gauge and that the pressure drop across each HEPA is less than 2 inches water gauge.

2. Verifying that on each of the below chlorine isolation mode actuation test signals, the subsystem automatically switches to the chlorine isolation mode of operation and the isolation valves close within 5 seconds:

   a) Outside air intake high chlorine, and

   b) Manual initiation from the control room.

3. Verifying that on ~~each of the below radiation isolation mode actuation test signals,~~ the subsystem ~~automatically~~ switches to the radiation isolation mode of operation and the control room is maintained at a positive pressure of at least 1/8 inch water gauge relative to the turbine enclosure and auxiliary equipment room and outside atmosphere during subsystem operation with an outdoor air flow rate less than or equal to 525 cfm.

   ~~a) Outside air intake high radiation, and~~

   ~~b) Manual initiation from control room.~~

Manual initiation from the Control Room,

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A.C. SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. Two diesel generators each with:

1. A day fuel tank containing a minimum of 200 gallons of fuel.

2. A fuel storage system containing a minimum of 33,500 gallons of fuel.

3. A fuel transfer pump.

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5, and ~~*~~ INSERT 5

ACTION:

a. With less than the above required A.C. electrical power sources OPERABLE, suspend CORE ALTERATIONS, handling of ~~irradiated fuel~~ RECENTLY IRRADIATED FUEL in the secondary containment, operations with a potential for draining the reactor vessel and crane operations over the spent fuel storage pool when fuel assemblies are stored therein. In addition, when in OPERATIONAL CONDITION 5 with the water level less than 22 feet above the reactor pressure vessel flange, immediately initiate corrective action to restore the required power sources to OPERABLE status as soon as practical.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.1.2 At least the above required A.C. electrical power sources shall be demonstrated OPERABLE per Surveillance Requirements 4.8.1.1.1, 4.8.1.1.2, and 4.8.1.1.3.

~~*When handling irradiated fuel in the secondary containment.~~

D.C. SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, two of the following four divisions of the D.C. electrical power sources system shall be OPERABLE with:

a. Division 1, Consisting of:
   1. 125-Volt Battery 1A1 (1A1D101).
   2. 125-Volt Battery 1A2 (1A2D101).
   3. 125-Volt Battery Charger 1BCA1 (1A1D103).
   4. 125-Volt Battery Charger 1BCA2 (1A2D103).

b. Division 2, Consisting of:
   1. 125-Volt Battery 1B1 (1B1D101).
   2. 125-Volt Battery 1B2 (1B2D101).
   3. 125-Volt Battery Charger 1BCB1 (1B1D103).
   4. 125-Volt Battery Charger 1BCB2 (1B2D103).

c. Division 3, Consisting of:
   1. 125-Volt Battery 1C (1CD101).
   2. 125-Volt Battery Charger 1BC[illegible] (1CD103).

d. Division 4, Consisting of:
   1. 125-Volt Battery 1D (1DD101).
   2. 125-Volt Battery Charger 1B[illegible] (1DD103).

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5, and [illegible] INSERT 5

ACTION:

a. With one or two required battery chargers or one required division inoperable:

   1. Restore battery terminal voltage to greater than or equal to the minimum established float voltage within 2 hours,

   2. Verify associated Division 1 or 2 float current ≤ 2 amps, or Division 3 or 4 float current ≤ 1 amp within 18 hours and once per 12 hours thereafter, and

   3. Restore battery charger(s) to OPERABLE status within 7 days.

b. With one or more required batteries inoperable due to:

   1. One or two batteries on one division with one or more battery cells float voltage < 2.07 volts, perform 4.8.2.1.a.1 and 4.8.2.1.a.2 within 2 hours for affected battery(s) and restore affected cell(s) voltage ≥ 2.07 volts within 24 hours.

~~*When handling irradiated fuel in the secondary containment.~~

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

2. Division 1 or 2 with float current > 2 amps, or with Division 3 or 4 with float current > 1 amp, perform 4.8.2.1.a.2 within 2 hours for affected battery(s) and restore battery float current to within limits within 18 hours.

3. One or two batteries on one division with one or more cells electrolyte level less than minimum established design limits, if electrolyte level was below the top of the plates restore electrolyte level to above top of plates within 8 hours and verify no evidence of leakage(*) within 12 hours. In all cases, restore electrolyte level to greater than or equal to minimum established design limits within 31 days.

4. One or two batteries on one division with pilot cell electrolyte temperature less than minimum established design limits, restore battery pilot cell temperature to greater than or equal to minimum established design limits within 12 hours.

5. Batteries in more than one division affected, restore battery parameters for all batteries in one division to within limits within 2 hours.

6. (i) Any battery having both (Action b.1) one or more battery cells float voltage < 2.07 volts and (Action b.2) float current not within limits, and/or

   (ii) Any battery not meeting any Action b.1 through b.5,

   Restore the battery parameters to within limits within 2 hours.

c. 1. With the requirements of Action a. and/or Action b. not met, or

   2. With less than two divisions of the above required D.C. electrical power sources OPERABLE for reasons other than Actions a. and/or b.,

   Suspend CORE ALTERATIONS, handling of ~~irradiated fuel~~ RECENTLY IRRADIATED FUEL in the secondary containment and operations with a potential for draining the reactor vessel.

d. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.2.2 At least the above required battery and charger shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.1.

(*) Contrary to the provisions of Specification 3.0.2, if electrolyte level was below the top of the plates, the verification that there is no evidence of leakage is required to be completed regardless of when electrolyte level is restored.

ELECTRICAL POWER SYSTEMS

LIMITING CONDITION FOR OPERATION (Continued)

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5, and ~~*~~ INSERT 5

ACTION:

a. With less than two divisions of the above required Unit 1 A.C. distribution systems energized, suspend CORE ALTERATIONS, handling of ~~irradiated fuel~~ RECENTLY IRRADIATED FUEL in the secondary containment and operations with a potential for draining the reactor vessel.

b. With less than two divisions of the above required Unit 1 D.C. distribution systems energized, suspend CORE ALTERATIONS, handling of ~~irradiated fuel~~ RECENTLY IRRADIATED FUEL in the secondary containment and operations with a potential for draining the reactor vessel.

c. With any of the above required Unit 2 and common AC and/or DC distribution system divisions not energized, declare the associated common equipment inoperable, and take the appropriate ACTION for that system.

d. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.3.2 At least the above required power distribution system divisions shall be determined energized at least once per 7 days by verifying correct breaker alignment and voltage on the busses/MCCs/panels.

~~*When handling irradiated fuel in the secondary containment.~~

Insert 1 (Page 1-2):

Table 2.1 of Federal Guidelines Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," ORNL, 1989, as described in Regulatory Guide 1.183. The factors in the column headed "effective" yield doses corresponding to the CEDE.

Insert 2 (Page 1-6):

RECENTLY IRRADIATED FUEL

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

Insert 3 (Page 3/4 1-19)

3.1.5 The standby liquid control system shall be OPERABLE and consist of a minimum of the following:

a. In OPERATIONAL CONDITIONS 1 and 2, two pumps and corresponding flow paths,

b. In OPERATIONAL CONDITION 3, one pump and corresponding flow path.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3

ACTION:

a. With only one pump and corresponding explosive valve OPERABLE, in OPERATIONAL CONDITION 1 or 2, restore one inoperable pump and corresponding explosive valve to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With standby liquid control system otherwise inoperable, in OPERATIONAL CONDITION 1, 2, or 3, restore the system to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the next 24 hours.

Insert 4 (Pages 3/4 6-47, 3/4 6-50, 3/4 6-52,)

When RECENTLY IRRADIATED FUEL is being handled in the secondary containment, or during operations with a potential for draining the reactor vessel, with the vessel head removed and fuel in the vessel

Insert 5 (Pages 3/4 7-3, 3/4 7-4, 3/4 7-5, 3/4 7-6, 3/4 8-9, 3/4 8-14, 3/4 8-20)

When RECENTLY IRRADIATED FUEL is being handled in the secondary containment, or during operations with a potential for draining the reactor vessel

CORE ALTERATION

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

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

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

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

CORE OPERATING LIMITS REPORT

1.7a The CORE OPERATING LIMITS REPORT (COLR) is the unit-specific document that provides the core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specifications 6.9.1.9 thru 6.9.12. Plant operation within these limits is addressed in individual specifications.

CRITICAL POWER RATIO

1.8 The CRITICAL POWER RATIO (CPR) shall be the ratio of that power in the assembly which is calculated by application of the (GEXL) correlation to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.

DOSE EQUIVALENT I-131

1.9 DOSE EQUIVALENT I-131 shall be that concentration of I-131, microcuries per gram, which alone would produce the same ~~thyroid dose~~ inhalation committed effective dose equivalent (CEDE) as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The ~~thyroid dose~~ conversion factors used for this calculation shall be ~~those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."~~ INSERT 1

DOWNSCALE TRIP SETPOINT (DTSP)

1.9a The downscale trip setpoint associated with the Rod Block Monitor (RBM) rod block trip setting.

$\bar{E}$-AVERAGE DISINTEGRATION ENERGY

1.10 $\bar{E}$ shall be the average, weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling, of the sum of the average beta and gamma energies per disintegration, in MeV, for isotopes, with half lives greater than 15 minutes, making up at least 95% of the total noniodine activity in the coolant.

EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME

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

PURGE - PURGING

1.31 PURGE or PURGING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

RATED THERMAL POWER

1.32 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3458 MWt.

REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY

1.33 REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY shall exist when:

a. All reactor enclosure secondary containment penetrations required to be closed during accident conditions are either:

1. Capable of being closed by an OPERABLE secondary containment automatic isolation system, or

2. Closed by at least one manual valve, blind flange, slide gate damper or deactivated automatic valve secured in its closed position, except as provided by Specification 3.6.5.2.1.

b. All reactor enclosure secondary containment hatches and blowout panels are closed and sealed.

c. The standby gas treatment system is in compliance with the requirements of Specification 3.6.5.3.

d. The reactor enclosure recirculation system is in compliance with the requirements of Specification 3.6.5.4.

e. At least one door in each access to the reactor enclosure secondary containment is closed.

f. The sealing mechanism associated with each reactor enclosure secondary containment penetration, e.g., welds, bellows, or O-rings, is OPERABLE.

g. The pressure within the reactor enclosure secondary containment is less than or equal to the value required by Specification 4.6.5.1.1a.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.34 REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

INSERT 2

REFUELING FLOOR SECONDARY CONTAINMENT INTEGRITY

~~1.35~~ 1.36 REFUELING FLOOR SECONDARY CONTAINMENT INTEGRITY shall exist when:

a. All refueling floor secondary containment penetrations required to be closed during accident conditions are either:

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REFUELING FLOOR SECONDARY CONTAINMENT INTEGRITY (Continued)

1. Capable of being closed by an OPERABLE secondary containment automatic isolation system, or

2. Closed by at least one manual valve, blind flange, slide gate damper or deactivated automatic valve secured in its closed position, except as provided by Specification 3.6.5.2.2.

b. All refueling floor secondary containment hatches and blowout panels are closed and sealed.

c. The standby gas treatment system is in compliance with the requirements of Specification 3.6.5.3.

d. At least one door in each access to the refueling floor secondary containment is closed.

e. The sealing mechanism associated with each refueling floor secondary containment penetration, e.g., welds, bellows, or O-rings, is OPERABLE.

f. The pressure within the refueling floor secondary containment is less than or equal to the value required by Specification 4.6.5.1.2a.

REPORTABLE EVENT

37 ~~1.36~~ A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

ROD DENSITY

~~1.37~~ 38 ROD DENSITY shall be the number of control rod notches inserted as a fraction of the total number of control rod notches. All rods fully inserted is equivalent to 100% ROD DENSITY.

SHUTDOWN MARGIN

~~1.38~~ 39 SHUTDOWN MARGIN shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming all control rods are fully inserted except for the single control rod of highest reactivity worth which is assumed to be fully withdrawn and the reactor is in the shutdown condition; cold, i.e. 68°F; and xenon free.

SITE BOUNDARY

40 ~~1.39~~ The SITE BOUNDARY shall be that line as defined in Figure 5.1.3-1a.

~~1.40 (Deleted)~~

SOURCE CHECK

1.41 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

~~3.1.5 The standby liquid control system, consisting of a minimum of two pumps and corresponding flow paths, shall be OPERABLE.~~

~~APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2~~

~~ACTION:~~

~~a. In OPERATIONAL CONDITION 1 or 2:~~

Insert 3

~~1. With only one pump and corresponding explosive valve OPERABLE, restore one inoperable pump and corresponding explosive valve to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.~~

~~2. With standby liquid control system otherwise inoperable, restore the system to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours.~~

SURVEILLANCE REQUIREMENTS

4.1.5 The standby liquid control system shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying that:

1. The temperature of the sodium pentaborate solution is within the limits of Figure 3.1.5-1.

2. The available volume of sodium pentaborate solution is at least 3160 gallons.

3. The temperature of the pump suction piping is within the limits of Figure 3.1.5-1 for the most recent concentration analysis.

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SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days by:

1. Verifying the continuity of the explosive charge.

2. Determining by chemical analysis Boron-10 and calculation* that the available weight of ~~sodium pentaborate~~ is greater than or equal to ~~3754~~ 185 lbs; the concentration of sodium pentaborate in solution is less than or equal to 13.8% and within the limits of Figure 3.1.5-1 and; the following equation is satisfied:

$$\frac{C}{13\%\ wt.} \times \frac{E}{29\ atom\ \%} \times \frac{Q}{86\ gpm} \geq 1$$

where

C = Sodium pentaborate solution (% by weight)
Q = Two pump flowrate, as determined per surveillance requirement 4.1.5.c.
E = Boron 10 enrichment (atom % Boron 10)

3. Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. Demonstrating that, when tested pursuant to Specification 4.0.5, the minimum flow requirement of 41.2 gpm per pump at a pressure of greater than or equal to 1230±25 psig is met.

d. At least once per 24 months during shutdown by:

1. Initiating at least one of the standby liquid control system loops, including an explosive valve, and verifying that a flow path from the pumps to the reactor pressure vessel is available by pumping demineralized water into the reactor vessel. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch which has been certified by having one of the batch successfully fired. All injection loops shall be tested in 3 operating cycles.

2. Verify all heat-treated piping between storage tank and pump suction is unblocked.**

e. Prior to addition of Boron to storage tank verify sodium pentaborate enrichment to be added is ≥ 29 atom % Boron 10.

* This test shall also be performed anytime water or boron is added to the solution or when the solution temperature drops below the limits of Figure 3.1.5-1 for the most recent concentration analysis, within 24 hours after water or boron addition or solution temperature is restored.

** This test shall also be performed whenever suction piping temperature drops below the limits of Figure 3.1.5-1 for the most recent concentration analysis, within 24 hours after solution temperature is restored.

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TABLE 3.3.2-1 (Continued)
ISOLATION ACTUATION INSTRUMENTATION
ACTION STATEMENTS

ION 20 - Be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ION 21 - Be in at least STARTUP with the associated isolation valves closed within 6 hours or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ION 22 - Be in at least STARTUP within 6 hours.

ION 23 - In OPERATIONAL CONDITION 1 or 2, verify the affected system isolation valves are closed within 1 hour and declare the affected system inoperable. In OPERATIONAL CONDITION 3, be in at least COLD SHUTDOWN within 12 hours.

ION 24 - Restore the manual initiation function to OPERABLE status within 8 hours or close the affected system isolation valves within the next hour and declare the affected system inoperable or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

ION 25 - Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within 1 hour.

ION 26 - Close the affected system isolation valves within 1 hour.

~~(RECENTLY IRRADIATED FUEL)~~ TABLE NOTATIONS

Required when (1) handling ~~irradiated fuel~~ in the ~~refueling area~~ secondary containment, or (2) ~~during CORE ALTERATIONS, or (3)~~ during operations with a potential for draining the reactor vessel with the vessel head removed and fuel in the vessel.

May be bypassed under administrative control, with all turbine stop valves closed.

During operation of the associated Unit 1 or Unit 2 ventilation exhaust system.

DELETED

A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter. Trip functions common to RPS Actuation Instrumentation are shown in Table 4.3.2.1-1. In addition, for the HPCI system and RCIC system isolation, provided that the redundant isolation valve, inboard or outboard, as applicable, in each line is OPERABLE and all required actuation instrumentation for that valve is OPERABLE, one channel may be placed in an inoperable status for up to 8 hours for required surveillance without placing the channel or trip system in the tripped condition.

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TABLE 4.3.2.1-1 (Continued)
ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRE
7. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level Low, Low - Level 2	S	Q	R	1, 2, 3
b. Drywell Pressure## - High	S	Q	R	1, 2, 3
c.1. Refueling Area Unit 1 Ventilation Exhaust Duct Radiation - High	S	Q	R	*#
2. Refueling Area Unit 2 Ventilation Exhaust Duct Radiation - High	S	Q	R	*#
d. Reactor Enclosure Ventilation Exhaust Duct Radiation - High	S	Q	R	1, 2, 3
e. Deleted				
f. Deleted				
g. Reactor Enclosure Manual Initiation	N.A.	R	N.A.	1, 2, 3
h. Refueling Area Manual Initiation	N.A.	R	N.A.	*

RECENTLY IRRADIATED FUEL

*Required when (1) handling ~~irradiated fuel~~ in the ~~refueling area~~ secondary containment, or (2) ~~during CORE ALTERATIONS, or (3)~~ during operations with a potential for draining the reactor vessel with the vessel head removed and fuel in the vessel.

**When not administratively bypassed and/or when any turbine stop valve is open.

#During operation of the associated Unit 1 or Unit 2 ventilation exhaust system.

##These trip functions (2a, 6b, and 7b) are common to the RPS actuation trip function.

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TABLE 3.3.7.1-1

RADIATION MONITORING INSTRUMENTATION

INSTRUMENTATION	MINIMUM CHANNELS OPERABLE	APPLICABLE CONDITIONS	ALARM/TRIP SETPOINT	ACTION
1. Main Control Room Normal Fresh Air Supply Radiation Monitor	4	1,2,3,~~5~~ and *	1×10^{-8} µCi/cc (b)	70
2. Area Monitors				
a. Criticality Monitors				
1) Spent Fuel Storage Pool	2	(a)	≥ 5 mR/h and ≤20mR/h(b)	71
b. Control Room Direct Radiation Monitor	1	At All Times	N.A.(b)	73
3. Reactor Enclosure Cooling Water Radiation Monitor	1	At All Times	≤ 3 x Background(b)	72

RECENTLY IRRADIATED FUEL

TABLE 3.3.7.1-1 (Continued)

RADIATION MONITORING INSTRUMENTATION

TABLE NOTATIONS

*When ~~irradiated fuel~~ is being handled in the secondary containment.

(a) With fuel in the spent fuel storage pool.

(b) Alarm only.

ACTION STATEMENTS

ACTION 70 - With one monitor inoperable, restore the inoperable monitor to the OPERABLE status within 7 days or, within the next 6 hours, initiate and maintain operation of the control room emergency filtration system in the radiation isolation mode of operation.

With two or more of the monitors inoperable, within one hour, initiate and maintain operation of the control room emergency filtration system in the radiation mode of operation.

ACTION 71 - With one of the required monitor inoperable, assure a portable continuous monitor with the same alarm setpoint is OPERABLE in the vicinity of the installed monitor during any fuel movement. If no fuel movement is being made, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

ACTION 72 - With the required monitor inoperable, obtain and analyze at least one grab sample of the monitored parameter at least once per 24 hours.

ACTION 73 - With the required monitor inoperable, assure a portable alarming monitor is OPERABLE in the vicinity of the installed monitor or perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

or during operations with a potential for draining the reactor vessel with the vessel head removed and fuel in the vessel.

TABLE 4.3.7.1-1

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTATION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE IS REQUIRED
1. Main Control Room Normal Fresh Air Supply Radiation Monitor	S	Q	R	1, 2, 3, 5 and *
2. Area Monitors				
a. Criticality Monitors				
1) Spent Fuel Storage Pool	S	M	R	(a)
b. Control Room Direct Radiation Monitor	S	M	R	At All Times
3. Reactor Enclosure Cooling Water Radiation Monitor	S	M	R(b)	At All Times

TABLE 4.3.7.1-1 (Continued)

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATIONS

*When ~~irradiated fuel~~ RECENTLY IRRADIATED FUEL is being handled in the secondary containment or during operations with a potential for draining the reactor vessel with the vessel head removed and fuel in the vessel.

(a) With fuel in the spent fuel storage pool.

(b) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards (NBS) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

3/4.4.7 MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.4.7 Two main steam line isolation valves (MSIVs) per main steam line shall be OPERABLE with closing times greater than or equal to 3 and less than or equal to ~~5~~ 10 seconds.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one or more MSIVs inoperable:

1. Maintain at least one MSIV OPERABLE in each affected main steam line that is open and within 8 hours, either:

a) Restore the inoperable valve(s) to OPERABLE status, or

b) Isolate the affected main steam line by use of a deactivated MSIV in the closed position.

2. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.7 Each of the above required MSIVs shall be demonstrated OPERABLE by verifying full closure between 3 and ~~5~~ 10 seconds when tested pursuant to Specification 4.0.5.

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LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

b. The combined leakage rate to be in accordance with the Primary Containment Leakage Rate Testing Program for all primary containment penetrations and all primary containment isolation valves that are subject to Type B and C tests, except for: main steam line isolation valves*, valves which are hydrostatically tested, and those valves where an exemption to Appendix J of 10 CFR 50 has been granted, and

c. The leakage rate to ~~≤11.5~~ ≤100 scf per hour for any main steam isolation valve that exceeds 100 scf per hour, and restore the combined maximum pathway leakage to ≤200 scf per hour, and

d. The combined leakage rate for all containment isolation valves in hydrostatically tested lines which penetrate the primary containment to less than or equal to 1 gpm times the total number of such valves,

prior to increasing reactor coolant system temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The primary containment leakage rates shall be demonstrated to be in accordance with the Primary Containment Leakage Rate Testing Program, or approved exemptions, for the following:

a. Type A Test

b. Type B and C Tests (including air locks)

c. Main Steam Line Isolation Valves

d. Hydrostatically tested Containment Isolation Valves

*Exemption to Appendix "J" to 10 CFR Part 50.

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3/4.6.5 SECONDARY CONTAINMENT

REFUELING AREA SECONDARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.5.1.2 REFUELING AREA SECONDARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: ~~OPERATIONAL CONDITION *~~. INSERT 4

ACTION:

RECENTLY IRRADIATED FUEL

Without REFUELING AREA SECONDARY CONTAINMENT INTEGRITY, suspend handling of ~~irradiated fuel~~ in the secondary containment, ~~CORE ALTERATIONS~~ and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.1.2 REFUELING AREA SECONDARY CONTAINMENT INTEGRITY shall be demonstrated by:

a. Verifying at least once per 24 hours that the pressure within the refueling area secondary containment is greater than or equal to 0.25 inch of vacuum water gauge.

b. Verifying at least once per 31 days that:

1. All refueling area secondary containment equipment hatches and blowout panels are closed and sealed.

2. At least one door in each access to the refueling area secondary containment is closed.

3. All refueling area secondary containment penetrations not capable of being closed by OPERABLE secondary containment automatic isolation dampers/valves and required to be closed during accident conditions are closed by valves, blind flanges, slide gate dampers or deactivated automatic dampers/valves secured in position.

c. At least once per 24 months:

Operating one standby gas treatment subsystem for one hour and maintaining greater than or equal to 0.25 inch of vacuum water gauge in the refueling area secondary containment at a flow rate not exceeding 764 cfm.

~~*Required when (1) irradiated fuel is being handled in the refueling area secondary containment, or (2) during CORE ALTERATIONS, or (3) during operations with a potential for draining the reactor vessel with the vessel head removed and fuel in the vessel.~~

CONTAINMENT SYSTEMS

REFUELING AREA SECONDARY CONTAINMENT AUTOMATIC ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.5.2.2 The refueling area secondary containment ventilation system automatic isolation valves shall be OPERABLE.

APPLICABILITY: ~~OPERATIONAL CONDITION *~~. INSERT 4

ACTION:

With one or more of the refueling area secondary containment ventilation system automatic isolation valves inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and within 8 hours either:

a. Restore the inoperable valves to OPERABLE status, or

b. Isolate each affected penetration by use of at least one deactivated valve secured in the isolation position, or

c. Isolate each affected penetration by use of at least one closed manual valve, blind flange or slide gate damper.

Otherwise, ~~in OPERATIONAL CONDITION*,~~ suspend handling of ~~irradiated fuel~~ RECENTLY IRRADIATED FUEL in the refueling area secondary containment, ~~CORE ALTERATIONS~~ and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.2.2 Each refueling area secondary containment ventilation system automatic isolation valve shall be demonstrated OPERABLE:

a. Prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by cycling the valve through at least one complete cycle of full travel and verifying the specified isolation time.

b. At least once per 24 months by verifying that on a containment isolation test signal each isolation valve actuates to its isolation position.

c. By verifying the isolation time to be within its limit at least once per 92 days.

~~*Required when (1) irradiated fuel is being handled in the refueling area secondary containment, or (2) during CORE ALTERATIONS, or (3) during operations with a potential for draining the reactor vessel with the vessel head removed and fuel in the vessel.~~

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STANDBY GAS TREATMENT SYSTEM - COMMON SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.5.3 Two independent standby gas treatment subsystems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, and ~~4~~. INSERT H

ACTION:

a. In OPERATIONAL CONDITION 1, 2, or 3:

1. With the Unit 1 diesel generator for one standby gas treatment subsystem inoperable for more than 30 days, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. The provisions of Specification 3.0.4 are not applicable.

2. With one standby gas treatment subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3. With one standby gas treatment subsystem inoperable and the other standby gas treatment subsystem with an inoperable Unit 1 diesel generator, restore the inoperable subsystem to OPERABLE status or restore the inoperable Unit 1 diesel generator to OPERABLE status within 72 hours, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4. With the Unit 1 diesel generators for both standby gas treatment system subsystems inoperable for more than 72 hours, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. ~~In OPERATIONAL CONDITION *~~: INSERT H

1. With one standby gas treatment subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days, or suspend handling of ~~irradiated fuel~~ RECENTLY IRRADIATED FUEL in the secondary containment, ~~CORE ALTERATIONS,~~ and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

2. With both standby gas treatment subsystems inoperable, suspend handling of ~~irradiated fuel~~ in the secondary containment, ~~CORE ALTERATIONS,~~ and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

~~*Required when (1) irradiated fuel is being handled in the refueling area secondary containment, or (2) during CORE ALTERATIONS, or (3) during operations with a potential for draining the reactor vessel with the vessel head removed and fuel in the vessel.~~

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SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 24* months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the subsystem by:

1. Verifying that the subsystem satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 5764 cfm ± 10%.

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration of less than ~~0.5%~~ 1.25% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F), at a relative humidity of 70% and at a face velocity of 66 fpm.

3. Verify that when the fan is running the subsystem flowrate is 2800 cfm minimum from each reactor enclosure (Zones I and II) and 2200 cfm minimum from the refueling area (Zone III) when tested in accordance with ANSI N510-1980.

4. Verify that the pressure drop across the refueling area to SGTS prefilter is less than 0.25 inches water gage while operating at a flow rate of 2400 cfm ± 10%.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration of less than ~~0.5%~~ 1.25% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F), at a relative humidity of 70% and at a face velocity of 66 fpm.

d. At least once per 24 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 9.1 inches water gauge while operating the filter train at a flow rate of 8400 cfm ± 10%.

*Surveillance interval is an exception to the guidance provided in Regulatory Guide 1.52, Revision 2, March 1978.

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REACTOR ENCLOSURE RECIRCULATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.5.4 Two independent reactor enclosure recirculation subsystems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one reactor enclosure recirculation subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With both reactor enclosure recirculation subsystems inoperable, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.4 Each reactor enclosure recirculation subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates properly. (flow at a minimum of 30,000 cfm)

b. At least once per 24* months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the subsystem by:

1. Verifying that the subsystem satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, (at rated) ~~and the system~~ flow ~~rate is~~ (60,000 cfm ± 10%).

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration of less than ~~2.5%~~ (15%) when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70%.

3. Verifying a subsystem flow rate ~~of 60,000 cfm ± 10%~~ during system operation when tested in accordance with ANSI N510-1980.

(within a range of 30,000 cfm to 66,000 cfm)

*Surveillance interval is an exception to the guidance provided in Regulatory Guide 1.52 Revision 2, March 1978.

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SURVEILLANCE REQUIREMENTS (Continued)

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration of less than ~~2.5%~~ 15% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) an a relative humidity of 70%.

d. At least once per 24 months by:

1. Verifying that the pressure drop across the combined prefilter, upstream and downstream HEPA filters, and charcoal adsorber banks is less than 6 inches ~~water gauge~~ while operating the filter train at ~~a~~ rated flow ~~rate of~~ (60,000 cfm ± 10%) verifying that the prefilter pressure drop is less than 0.8 inch water gauge and that the pressure drop across each HEPA is less than 2 inches water gauge.

2. Verifying that the filter train starts and the isolation valves which take suction on and return to the reactor enclosure open on each of the following test signals:

a. Manual initiation from the control room, and

b. Simulated automatic initiation signal.

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter bank satisfies the inplace penetration and leakage testing acceptance criteria of less than 0.05% in ~~accordance with ANSI N510-1980~~ while operating the system at ~~a~~ rated flow ~~rate of~~ (60,000 cfm ± 10%).

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorber bank satisfies the inplace penetration and leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon ~~refrigerant test~~ gas while operating the system at ~~a~~ rated flow ~~rate of~~ (60,000 cfm ± 10%).

3.7.1.2 At least the following independent emergency service water system loops, with each loop comprised of:

a. Two OPERABLE emergency service water pumps, and

b. An OPERABLE flow path capable of taking suction from the emergency service water pumps wet pits which are supplied from the spray pond or the cooling tower basin and transferring the water to the associated Unit 2 and common safety-related equipment,

shall be OPERABLE: Two loops in

a. ~~In~~ OPERATIONAL CONDITIONS 1, 2, and 3 ~~, two loops.~~

One loop, in

b. ~~In~~ OPERATIONAL CONDITIONS 4, 5, and ~~*, one loop.~~ INSERT 5

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, 5, and ~~*~~.

ACTION:

a. In OPERATION CONDITION 1, 2, or 3:

1. With one emergency service water pump inoperable, restore the inoperable pump to OPERABLE status within 45 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With one emergency service water pump in each loop inoperable, restore at least one inoperable pump to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3. With one emergency service water system loop otherwise inoperable, declare all equipment aligned to the inoperable loop inoperable**, restore the inoperable loop to OPERABLE status with at least one OPERABLE pump within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

~~*When handling irradiated fuel in the secondary containment.~~

**The diesel generators may be aligned to the OPERABLE emergency service water system loop provided confirmatory flow testing has been performed. Those diesel generators not aligned to the OPERABLE emergency service water system loop shall be declared inoperable and the actions of 3.8.1.1 taken.

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LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

4. With three ESW pump/diesel generator pairs** inoperable, restore at least one inoperable ESW pump/diesel generator pair** to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

5. With four ESW pump/diesel generator pairs** inoperable, restore at least one inoperable ESW pump/diesel generator pair** to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 4 or 5:

1. With only one emergency service water pump and its associated flow path OPERABLE, restore at least two pumps with at least one flow path to OPERABLE status within 72 hours or declare the associated safety related equipment inoperable and take the ACTION required by Specifications 3.5.2 and 3.8.1.2.

c. ~~In OPERATIONAL CONDITION *~~ (INSERT 5) (:)

1. With only one emergency service water pump and its associated flow path OPERABLE, restore at least two pumps with at least one flow path to OPERABLE status within 72 hours or verify adequate cooling remains available for the diesel generators required to be OPERABLE or declare the associated diesel generator(s) inoperable and take the ACTION required by Specification 3.8.1.2. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENT

4.7.1.2 At least the above required emergency service water system loop(s) shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 24 months by verifying that:

1. Each automatic valve actuates to its correct position on its appropriate ESW pump start signal.

2. Each pump starts automatically when its associated diesel generator starts.

~~* When handling irradiated fuel in the secondary containment.~~

** An ESW pump/diesel generator pair consists of an ESW pump and its associated diesel generator. If either an ESW pump or its associated diesel generator becomes inoperable, then the ESW pump/diesel generator pair is inoperable.

ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The spray pond shall be OPERABLE with:

a. A minimum pond water level at or above elevation 250'-10" Mean Sea Level, and

b. A pond water temperature of less than or equal to 88°F.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, 5, and ~~*~~ INSERT 5

ACTION:

With the requirements of the above specification not satisfied:

a. In OPERATIONAL CONDITION 1, 2, or 3, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. In OPERATIONAL CONDITION 4 or 5, declare the RHRSW system and the emergency service water system inoperable and take the ACTION required by Specifications 3.7.1.1 and 3.7.1.2.

c. ~~In OPERATIONAL CONDITION *,~~ declare the emergency service water system inoperable and take the ACTION required by Specification 3.7.1.2. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.3 The spray pond shall be determined OPERABLE:

a. By verifying the pond water level to be greater than its limit at least once per 24 hours.

b. By verifying the water surface temperature (within the upper two feet of the surface) to be less than or equal to 88°F:

1. at least once per 4 hours when the spray pond temperature is greater than or equal to 80°F; and

2. at least once per 2 hours when the spray pond temperature is greater than or equal to 85°F; and

3. at least once per 24 hours when the spray pond temperature is greater than 32°F.

c. By verifying all piping above the frost line is drained:

1. within one (1) hour after being used when ambient air temperature is below 40°F; or

2. when ambient air temperature falls below 40°F if the piping has not been previously drained.

~~*When handling irradiated fuel in the secondary containment.~~

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3/4.7.2 CONTROL ROOM EMERGENCY FRESH AIR SUPPLY SYSTEM - COMMON SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.2 Two independent control room emergency fresh air supply system subsystems shall be OPERABLE.

APPLICABILITY: All OPERATIONAL CONDITIONS and ~~*~~. INSERT 5

ACTION:

a. In OPERATIONAL CONDITION 1, 2, or 3:

1. With the Unit 1 diesel generator for one control room emergency fresh air supply subsystem inoperable for more than 30 days, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. The provisions of Specification 3.0.4 are not applicable.

2. With one control room emergency fresh air supply subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3. With one control room emergency fresh air supply subsystem inoperable and the other control room emergency fresh air supply subsystem with an inoperable Unit 1 diesel generator, restore the inoperable subsystem to OPERABLE status or restore the Unit 1 diesel generator to OPERABLE status within 72 hours, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4. With the Unit 1 diesel generators for both control room emergency fresh air supply subsystems inoperable for more than 72 hours, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 4, 5 or ~~*~~. INSERT 5

1. With one control room emergency fresh air supply subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days, or initiate and maintain operation of the OPERABLE subsystem in the radiation isolation mode of operation.

2. With both both control room emergency fresh air supply subsystem inoperable, suspend ~~CORE ALTERATIONS,~~ handling of ~~irradiated fuel~~ RECENTLY IRRADIATED FUEL in the secondary containment and operations with a potential for draining the reactor vessel.

~~c.~~ The provisions of Specification 3.0.3 are not applicable ~~in OPERATIONAL CONDITION *~~.

~~*When irradiated fuel is being handled in the Secondary Containment.~~

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration of less than ~~2.5%~~ 10% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70%.

3. Verifying a subsystem flow rate of 3000 cfm ± 10% during subsystem operation when tested in accordance with ANSI N510-1980.

d. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration of less than ~~2.5%~~ 10% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70%.

e. At least once per 24 months by:

1. Verifying that the pressure drop across the combined prefilter, upstream and downstream HEPA filters, and charcoal adsorber banks is less than 6 inches water gauge while operating the subsystem at a flow rate of 3000 cfm ± 10%; verifying that the prefilter pressure drop is less than 0.8 inch water gauge and that the pressure drop across each HEPA is less than 2 inches water gauge.

2. Verifying that on each of the below chlorine isolation mode actuation test signals, the subsystem automatically switches to the chlorine isolation mode of operation and the isolation valves close within 5 seconds:

   a) Outside air intake high chlorine, and

   b) Manual initiation from the control room.

3. Verifying that on ~~each of the below radiation isolation mode actuation test signals,~~ the subsystem ~~automatically~~ switches to the radiation isolation mode of operation and the control room is maintained at a positive pressure of at least 1/8 inch water gauge relative to the turbine enclosure and auxiliary equipment room and outside atmosphere during subsystem operation with an outdoor air flow rate less than or equal to 525 cfm.

   ~~a) Outside air intake high radiation, and~~

   ~~b) Manual initiation from control room.~~

   manual initiation from the Control Room,

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A.C. SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. Two diesel generators each with:

1. A day fuel tank containing a minimum of 200 gallons of fuel.

2. A fuel storage system containing a minimum of 33,500 gallons of fuel.

3. A fuel transfer pump.

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5, and ~~*~~ INSERT 5

ACTION:

a. With less than the above required A.C. electrical power sources OPERABLE, suspend CORE ALTERATIONS, handling of ~~irradiated fuel~~ RECENTLY IRRADIATED FUEL in the secondary containment, operations with a potential for draining the reactor vessel and crane operations over the spent fuel storage pool when fuel assemblies are stored therein. In addition, when in OPERATIONAL CONDITION 5 with the water level less than 22 feet above the reactor pressure vessel flange, immediately initiate corrective action to restore the required power sources to OPERABLE status as soon as practical.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.1.2 At least the above required A.C. electrical power sources shall be demonstrated OPERABLE per Surveillance Requirements 4.8.1.1.1, 4.8.1.1.2, and 4.8.1.1.3.

~~*When handling irradiated fuel in the secondary containment.~~

D.C. SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, two of the following four divisions of the D.C. electrical power sources system shall be OPERABLE with:

a. Division 1, Consisting of:
   1. 125-Volt Battery 2A1 (2A1D101).
   2. 125-Volt Battery 2A2 (2A2D101).
   3. 125-Volt Battery Charger 2BCA1 (2A1D103).
   4. 125-Volt Battery Charger 2BCA2 (2A2D103).

b. Division 2, Consisting of:
   1. 125-Volt Battery 2B1 (2B1D101).
   2. 125-Volt Battery 2B2 (2B2D101).
   3. 125-Volt Battery Charger 2BCB1 (2B1D103).
   4. 125-Volt Battery Charger 2BCB2 (2B2D103).

c. Division 3, Consisting of:
   1. 125-Volt Battery 2C (2CD101).
   2. 125-Volt Battery Charger 2BCC (2CD103).

d. Division 4, Consisting of:
   1. 125-Volt Battery 2D (2DD101).
   2. 125-Volt Battery Charger 2BCD (2DD103).

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5, and ~~5*~~ INSERT 5

ACTION:

a. With one or two required battery chargers on one required division inoperable:

   1. Restore battery terminal voltage to greater than or equal to the minimum established float voltage within 2 hours,

   2. Verify associated Division 1 or 2 float current ≤ 2 amps, or Division 3 or 4 float current ≤ 1 amp within 18 hours and once per 12 hours thereafter, and

   3. Restore battery charger(s) to OPERABLE status within 7 days.

b. With one or more required batteries inoperable due to:

   1. One or two batteries on one division with one or more battery cells float voltage < 2.07 volts, perform 4.8.2.1.a.1 and 4.8.2.1.a.2 within 2 hours for affected battery(s) and restore affected cell(s) voltage ≥ 2.07 volts within 24 hours.

~~*When handling irradiated fuel in the secondary containment.~~

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

2. Division 1 or 2 with float current > 2 amps, or with Division 3 or 4 with float current > 1 amp, perform 4.8.2.1.a.2 within 2 hours for affected battery(s) and restore battery float current to within limits within 18 hours.

3. One or two batteries on one division with one or more cells electrolyte level less than minimum established design limits, if electrolyte level was below the top of the plates restore electrolyte level to above top of plates within 8 hours and verify no evidence of leakage(*) within 12 hours. In all cases, restore electrolyte level to greater than or equal to minimum established design limits within 31 days.

4. One or two batteries on one division with pilot cell electrolyte temperature less than minimum established design limits, restore battery pilot cell temperature to greater than or equal to minimum established design limits within 12 hours.

5. Batteries in more than one division affected, restore battery parameters for all batteries in one division to within limits within 2 hours.

6. (i) Any battery having both (Action b.1) one or more battery cells float voltage < 2.07 volts and (Action b.2) float current not within limits, and/or

   (ii) Any battery not meeting any Action b.1 through b.5,

   Restore the battery parameters to within limits within 2 hours.

c. 1. With the requirements of Action a. and/or Action b. not met, or

   2. With less than two divisions of the above required D.C. electrical power sources OPERABLE for reasons other than Actions a. and/or b.,

   Suspend CORE ALTERATIONS, handling of ~~irradiated fuel~~ RECENTLY IRRADIATED FUEL in the secondary containment and operations with a potential for draining the reactor vessel.

d. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.2.2 At least the above required batteries and chargers shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.1.

(*) Contrary to the provisions of Specification 3.0.2, if electrolyte level was below the top of the plates, the verification that there is no evidence of leakage is required to be completed regardless of when electrolyte level is restored.

LIMITING CONDITION FOR OPERATION (Continued)

	c) 125-V DC Distribution Panels:	2PPA1	(2AD102)
		2PPA2	(2AD501)
		2PPA3	(2AD162)
2.	Unit 2 Division 2, Consisting of:		
	a) 250-V DC Fuse Box:	2FB	(2BD105)
	b) 250-V DC Motor Control Centers:	2DB-1	(20D202)
		2DB-2	(20D203)
	c) 125-V DC Distribution Panels:	2PPB1	(2BD102)
		2PPB2	(2BD501)
		2PPB3	(2BD162)
3.	Unit 2 Division 3, Consisting of:		
	a) 125-V DC Fuse Box:	2FC	(2CD105)
	b) 125-V DC Distribution Panels:	2PPC1	(2CD102)
		2PPC2	(2CD501)
		2PPC3	(2CD162)
4.	Unit 2 Division 4, Consisting of:		
	a) 125-V DC Fuse Box:	2FD	(2DD105)
	b) 125-V DC Distribution Panels:	2PPD1	(2DD102)
		2PPD2	(2DD501)
		2PPD3	(2DD162)
5.	Unit 1 and Common Division 1, Consisting of:		
	a) 250-V DC Fuse Box:	1FA	(1AD105)
	b) 125-V DC Distribution Panels:	1PPA1	(1AD102)
		1PPA2	(1AD501)
6.	Unit 1 and Common Division 2, Consisting of:		
	a) 250-V DC Fuse Box:	1FB	(1BD105)
	b) 125-V DC Distribution Panels:	1PPB1	(1BD102)
		1PPB2	(1BD501)
7.	Unit 1 and Common Division 3, Consisting of:		
	a) 125-V DC Fuse Box:	1FC	(1CD105)
	b) 125-V DC Distribution Panels:	1PPC1	(1CD102)
		1PPC2	(1CD501)
8.	Unit 1 and Common Division 4, Consisting of:		
	a) 125-V DC Fuse Box:	1FD	(1DD105)
	b) 125-V DC Distribution Panels:	1PPD1	(1DD102)
		1PPD2	(1DD501)

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5, and ~~*~~. INSERT 5

ACTION:

a. With less than two divisions of the above required Unit 2 A.C. distribution systems energized, suspend CORE ALTERATIONS, handling of ~~irradiated fuel~~ RECENTLY IRRADIATED FUEL in the secondary containment and operations with a potential for draining the reactor vessel.

~~When handling irradiated fuel in the secondary containment.~~

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

b. With less than two divisions of the above required Unit 2 D.C. distribution systems energized, suspend CORE ALTERATIONS, handling of ~~irradiated fuel~~ RECENTLY IRRADIATED FUEL in the secondary containment and operations with a potential for draining the reactor vessel.

c. With any of the above required Unit 1 and common AC and/or DC distribution system divisions not energized, declare the associated common equipment inoperable, and take the appropriate ACTION for that system.

d. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.3.2 At least the above required power distribution system divisions shall be determined energized at least once per 7 days by verifying correct breaker alignment and voltage on the busses/MCCs/panels.

Unit 2 Limerick Generating Station Technical Specification Insert

Insert 1 (Page 1-2):

Table 2.1 of Federal Guidelines Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," ORNL, 1989, as described in Regulatory Guide 1.183. The factors in the column headed "effective" yield doses corresponding to the CEDE.

Insert 2 (Page 1-6):

RECENTLY IRRADIATED FUEL

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

Insert 3 (Page 3/4 1-19)

3.1.5 The standby liquid control system shall be OPERABLE and consist of a minimum of the following:

a. In OPERATIONAL CONDITIONS 1 and 2, two pumps and corresponding flow paths,

b. In OPERATIONAL CONDITION 3, one pump and corresponding flow path.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3

ACTION:

a. With only one pump and corresponding explosive valve OPERABLE, in OPERATIONAL CONDITION 1 or 2, restore one inoperable pump and corresponding explosive valve to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With standby liquid control system otherwise inoperable, in OPERATIONAL CONDITION 1, 2, or 3, restore the system to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the next 24 hours.

Insert 4 (Pages 3/4 6-47, 3/4 6-50, 3/4 6-52,)

When RECENTLY IRRADIATED FUEL is being handled in the secondary containment, or during operations with a potential for draining the reactor vessel, with the vessel head removed and fuel in the vessel

Insert 5 (Pages 3/4 7-3, 3/4 7-4, 3/4 7-5, 3/4 7-6, 3/4 8-9, 3/4 8-14, 3/4 8-19)

When RECENTLY IRRADIATED FUEL is being handled in the secondary containment, or during operations with a potential for draining the reactor vessel

ATTACHMENT 3

LIMERICK GENERATING STATION
UNITS 1 AND 2

Docket Nos. 50-352 & 50-353

License Nos. NPF-39 & NPF-85

License Amendment Request
"LGS Alternative Source Term Implementation"

Markup of Technical Specification Bases Pages
(For information only)

<u>UNITS 1</u>
B 3/4 1-4
B 3/4 1-5
B 3/4 4-6
B 3/4 6-5
B 3/4 7-1a

<u>UNITS 2</u>
B 3/4 1-4
B 3/4 1-5
B 3/4 4-6
B 3/4 6-5
B 3/4 7-1

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

The standby liquid control system provides a backup capability for bringing the reactor from full power to a cold, Xenon-free shutdown, assuming that the withdrawn control rods remain fixed in the rated power pattern. To meet this objective it is necessary to inject a quantity of boron which produces a concentration of 660 ppm in the reactor core and other piping systems connected to the reactor vessel. To allow for potential leakage and improper mixing, this concentration is increased by 25%. The required concentration is achieved by having available a minimum quantity of 3,160 gallons of sodium pentaborate solution containng a minimum of 3,754 lbs of sodium pentaborate having the requisite ~~B-10~~ atom % enrichment of 29% as determined from Reference 5. This quantity of solution is a net amount which is above the pump suction shutoff level setpoint thus allowing for the portion which cannot be injected. The pumping rate of 41.2 gpm provides a negative reactivity insertion rate over the permissible solution volume range, which adequately compensates for the positive reactivity effects due to elimination of steam voids, increased water density from hot to cold, reduced doppler effect in uranium, reduced neutron leakage from boiling to cold, decreased control rod worth as the moderator cools, and xenon decay. The temperature requirement ensures that the sodium pentaborate always remains in solution.

With redundant pumps and explosive injection valves and with a highly reliable control rod scram system, operation of the reactor is permitted to continue for short periods of time with the system inoperable or for longer periods of time with one of the redundant components inoperable.

The SLCS system consists of three separate and independent pumps and explosive valves. Two of the separate and independent pumps and explosive valves are required to meet the minimum requirements of this technical specification and, where applicable, satisfy the single failure criterion.

The SLCS must have an equivalent control capacity of 86 gpm of 13% weight sodium pentaborate in order to satisfy 10 CFR 50.62 (Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants). As part of the ARTS/MELLL program the ATWS analysis was updated to reflect the new rod line. As a result of this it was determined that the Boron 10 enrichment was required to be increased to 29% to prevent exceeding a suppression pool temperature of 190°F. This equivalency requirement is fulfilled by having a system which satisfies the equation given in 4.1.5.b.2.

The upper limit concentration of 13.8% has been established as a reasonable limit to prevent precipitation of sodium pentaborate in the event of a loss of tank heating, which allow the solution to cool.

STANDBY LIQUID CONTROL SYSTEM (Continued)

Surveillance requirements are established on a frequency that assures a high reliability of the system. Once the solution is established, boron concentration will not vary unless more boron or water is added, thus a check on the temperature and volume once each 24 hours assures that the solution is available for use.

Replacement of the explosive charges in the valves at regular intervals will assure that these valves will not fail because of deterioration of the charges.

INSERT 2

1. C. J. Paone, R. C. Stirn and J. A. Woolley, "Rod Drop Accident Analysis for Large BWR's," G. E. Topical Report NEDO-10527, March 1972.

2. C. J. Paone, R. C. Stirn, and R. M. Young, Supplement 1 to NEDO-10527, July 1972.

3. J. M. Haun, C. J. Paone, and R. C. Stirn, Addendum 2, "Exposed Cores." Supplement 2 to NEDO-10527, January 1973.

4. Amendment 17 to General Electric Licensing Topical Report NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel".

5. "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Limerick Generating Station Units 1 and 2," NEDC-32193P, Revision 2, October 1993.

3/4.4.7 MAIN STEAM LINE ISOLATION VALVES

Double isolation valves are provided on each of the main steam lines to minimize the potential leakage paths from the containment in case of a line break. Only one valve in each line is required to maintain the integrity of the containment, however, single failure considerations require that two valves be OPERABLE. The surveillance requirements are based on the operating history of this type valve. The maximum closure time has been selected to contain fission products and to ~~ensure the core is not uncovered~~ following line breaks. The minimum closure time is consistent with the assumptions in the safety analyses to prevent pressure surges.

prevent core damage

3/4.4.8 STRUCTURAL INTEGRITY

The inspection programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant.

Components of the reactor coolant system were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code 1971 Edition and Addenda through Winter 1972.

The inservice inspection program for ASME Code Class 1, 2, and 3 components will be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a. Additionally, the Inservice Inspection Program conforms to the NRC staff positions identified in NRC Generic Letter 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping," as approved in NRC Safety Evaluations dated March 6, 1990 and October 22, 1990.

3/4.4.9 RESIDUAL HEAT REMOVAL

The RHR system is required to remove decay heat and sensible heat in order to maintain the temperature of the reactor coolant. RHR shutdown cooling is comprised of four (4) subsystems which make two (2) loops. Each loop consists of two (2) motor driven pumps, a heat exchanger, and associated piping and valves. Both loops have a common suction from the same recirculation loop. Two (2) redundant, manually controlled shutdown cooling subsystems of the RHR System can provide the required decay heat removal capability. Each pump discharges the reactor coolant, after it has been cooled by circulation through the respective heat exchangers, to the reactor via the associated recirculation loop or to the reactor via the low pressure coolant injection pathway. The RHR heat exchangers transfer heat to the RHR Service Water System. The RHR shutdown cooling mode is manually controlled.

An OPERABLE RHR shutdown cooling subsystem consists of an RHR pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path. In HOT SHUTDOWN condition, the requirement to maintain OPERABLE two (2) independent RHR shutdown cooling subsystems means that each subsystem considered OPERABLE must be associated with a different heat exhanger loop, i.e., the "A" RHR heat exchanger with the "A" RHR pump or the "C" RHR pump, and the "B" RHR heat exchanger with the "B" RHR pump or the "D" RHR pump are two (2) independent RHR shutdown cooling subsystems. Only one (1) of the two (2) RHR pumps associated with each RHR heat exchanger loop is

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3/4.6.5 SECONDARY CONTAINMENT

Secondary containment is designed to minimize any ground level release of radioactive material which may result from an accident. The Reactor Enclosure and associated structures provide secondary containment during normal operation when the drywell is sealed and in service. At other times the drywell may be open and, when required, secondary containment integrity is specified.

Establishing and maintaining a vacuum in the reactor enclosure secondary containment with the standby gas treatment system once per 24 months, along with the surveillance of the doors, hatches, dampers and valves, is adequate to ensure that there are no violations of the integrity of the secondary containment.

The OPERABILITY of the reactor enclosure recirculation system and the standby gas treatment systems ensures that sufficient iodine removal capability will be available in the event of a LOCA or refueling accident involving recently irradiated fuel (SGTS only). The reduction in containment iodine inventory reduces the resulting SITE BOUNDARY radiation doses associated with containment leakage. The operation of this system and resultant iodine removal capacity are consistent with the assumptions used in the LOCA and refueling accident analyses. Provisions have been made to continuously purge the filter plenums with instrument air when the filters are not in use to prevent buildup of moisture on the adsorbers and the HEPA filters.

Although the safety analyses assumes that the reactor enclosure secondary containment draw down time will take 930 seconds, these surveillance requirements specify a draw down time of 916 seconds. This 14 second difference is due to the diesel generator starting and sequence loading delays which is not part of this surveillance requirement.

The reactor enclosure secondary containment draw down time analyses assumes a starting point of 0.25 inch of vacuum water gauge and worst case SGTS dirty filter flow rate of 2800 cfm. The surveillance requirements satisfy this assumption by starting the drawdown from ambient conditions and connecting the adjacent reactor enclosure and refueling area to the SGTS to split the exhaust flow between the three zones and verifying a minimum flow rate of 2800 cfm from the test zone. This simulates the worst case flow alignment and verifies adequate flow is available to drawdown the test zone within the required time. The Technical Specification Surveillance Requirement 4.6.5.3.b.3 is intended to be a multi-zone air balance verification without isolating any test zone.

INSERT 3

The SGTS fans are sized for three zones and therefore, when aligned to a single zone or two zones, will have excess capacity to more quickly drawdown the affected zones. There is no maximum flow limit to individual zones or pairs of zones and the air balance and drawdown time are verified when all three zones are connected to the SGTS.

The three zone air balance verification and drawdown test will be done after any major system alteration, which is any modification which will have an effect on the SGTS flowrate such that the ability of the SGTS to drawdown the reactor enclosure to greater than or equal to 0.25 inch of vacuum water gage in less than or equal to 916 seconds could be affected.

3/4.7.2 CONTROL ROOM EMERGENCY FRESH AIR SUPPLY SYSTEM - COMMON SYSTEM

The OPERABILITY of the control room emergency fresh air supply system ensures that the control room will remain habitable for operations personnel during and following all design basis accident conditions. Constant purge of the system at 1 cfm is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less ~~whole body, or its equivalent~~ Total Effective Dose Equivalent. This limitation is consistent with the requirements of ~~General Design Criterion 19 of Appendix A,~~ 10 CFR Part ~~50~~ 50.67, Accident Source Terms.

INSERT 4

3/4.7.3 REACTOR CORE ISOLATION COOLING SYSTEM

The reactor core isolation cooling (RCIC) system is provided to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel without requiring actuation of any of the emergency core cooling system equipment. The RCIC system is conservatively required to be OPERABLE whenever reactor pressure exceeds 150 psig. This pressure is substantially below that for which low pressure core cooling systems can provide adequate core cooling.

The RCIC system specifications are applicable during OPERATIONAL CONDITIONS 1, 2, and 3 when reactor vessel pressure exceeds 150 psig because RCIC is the primary non-ECCS source of emergency core cooling when the reactor is pressurized.

With the RCIC system inoperable, adequate core cooling is assured by the OPERABILITY of the HPCI system and justifies the specified 14 day out-of-service period.

The surveillance requirements provide adequate assurance that RCIC will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage and to start cooling at the earliest possible moment.

Unit 1 Limerick AST LAR Bases Inserts

INSERT 1 (Page B3/4 1-4)

The above quantities calculated at 29% Boron-10 enrichment have been demonstrated by analysis to provide a Boron-10 weight equivalent of 185 lbs in the sodium pentaborate solution. Maintaining this Boron-10 weight in the net tank contents ensures a sufficient quantity of boron to bring the reactor to a cold, Xenon-free shutdown.

INSERT 2 (Page B3/4 1-5)

The Standby Liquid Control System also has a post-DBA LOCA safety function to buffer Suppression Pool pH in order to maintain bulk pH above 7.0. The buffering of Suppression Pool pH is necessary to prevent iodine re-evolution to satisfy the methodology for Alternative Source Term. Manual initiation is used, and the minimum amount of total boron required for Suppression Pool pH buffering is 240 lbs. Given that at least 185 lbs of Boron-10 is maintained in the tank, the total boron in the tank will be greater than 240 lbs for the range of enrichments from 29% to 62%.

ACTION Statement (a) applies only to OPERATIONAL CONDITIONS 1 and 2 because a single pump can satisfy both the reactor control function and the post-DBA LOCA function to control Suppression Pool pH since boron injection is not required until 13 hours post-LOCA. ACTION Statement (b) applies to OPERATIONAL CONDITIONS 1, 2 and 3 to address the post-LOCA safety function of the SLC system.

INSERT 3 (Page B3/4 6-5)

Based on implementation of the AST methodology, it has been demonstrated through analysis that operating the subsystem with a minimum flow rate of 30,000 cfm satisfies the dose requirements. However, based on the applicable testing standards and guidance provided in Regulatory Guide 1.52, Revision 2, and ANSI N510, 1980, the acceptance criteria must be satisfied based on the system rated flow, which provides the most conservative results. Satisfying the acceptance criteria at rated flow also demonstrates system operability at lower measured flows because the residence time is longer. Therefore, verification of subsystem flow rates between 30,000 cfm and the maximum rated flow of 66,000 cfm (includes a 10% factor) satisfies the surveillance requirements for the HEPA filters and charcoal adsorber housings based on 70% efficiencies.

INSERT 4 (Page B3/4 7-1a)

Since the Control Room Emergency Fresh Air Supply System is not credited for filtration in OPERATIONAL CONDITIONS 4 and 5, applicability to 4 and 5 is only required to support the Chlorine and Toxic Gas design basis isolation requirements.

Additionally, based on implementation of the Alternative Source Term methodology, it has been demonstrated through analysis that manual initiation of the CREFAS radiation mode within 30 minutes of the start of gap release for the limiting design basis LOCA is sufficient to assure that control room operator dose limits in 10 CFR Part 50.67 are met.

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

The standby liquid control system provides a backup capability for bringing the reactor from full power to a cold, Xenon-free shutdown, assuming that the withdrawn control rods remain fixed in the rated power pattern. To meet this objective it is necessary to inject a quantity of boron which produces a concentration of 660 ppm in the reactor core and other piping systems connected to the reactor vessel. To allow for potential leakage and improper mixing, this concentration is increased by 25%. The required concentration is achieved by having available a minimum quantity of 3,160 gallons of sodium pentaborate solution containing a minimum of 3,754 lbs of sodium pentaborate having the requisite ~~B-10~~ atom % enrichment of 29% as determined from Reference 5. This quantity of solution is a net amount which is above the pump suction shutoff level setpoint thus allowing for the portion which cannot be injected. The pumping rate of 41.2 gpm provides a negative reactivity insertion rate over the permissible solution volume range, which adequately compensates for the positive reactivity effects due to elimination of steam voids, increased water density from hot to cold, reduced doppler effect in uranium, reduced neutron leakage from boiling to cold, decreased control rod worth as the moderator cools, and xenon decay. The temperature requirement ensures that the sodium pentaborate always remains in solution.

With redundant pumps and explosive injection valves and with a highly reliable control rod scram system, operation of the reactor is permitted to continue for short periods of time with the system inoperable or for longer periods of time with one of the redundant components inoperable.

The SLCS system consists of three separate and independent pumps and explosive valves. Two of the separate and independent pumps and explosive valves are required to meet the minimum requirements of this technical specification and, where applicable, satisfy the single failure criterion.

The SLCS must have an equivalent control capacity of 86 gpm of 13% weight sodium pentaborate in order to satisfy 10 CFR 50.62 (Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants). As part of the ARTS/MELLL program the ATWS analysis was updated to reflect the new rod line. As a result of this it was determined that the Boron 10 enrichment was required to be increased to 29% to prevent exceeding a suppression pool temperature of 190°F. This equivalency requirement is fulfilled by having a system which satisfies the equation given in 4.1.5.b.2.

The upper limit concentration of 13.8% has been established as a reasonable limit to prevent precipitation of sodium pentaborate in the event of a loss of tank heating, which allow the solution to cool.

STANDBY LIQUID CONTROL SYSTEM (Continued)

Surveillance requirements are established on a frequency that assures a high reliability of the system. Once the solution is established, boron concentration will not vary unless more boron or water is added, thus a check on the temperature and volume once each 24 hours assures that the solution is available for use.

Replacement of the explosive charges in the valves at regular intervals will assure that these valves will not fail because of deterioration of the charges.

INSERT 2

1. C. J. Paone, R. C. Stirn and J. A. Woolley, "Rod Drop Accident Analysis for Large BWR's," G. E. Topical Report NEDO-10527, March 1972.

2. C. J. Paone, R. C. Stirn, and R. M. Young, Supplement 1 to NEDO-10527, July 1972.

3. J. M. Haun, C. J. Paone, and R. C. Stirn, Addendum 2, "Exposed Cores," Supplement 2 to NEDO-10527, January 1973.

4. Amendment 17 to General Electric Licensing Topical Report NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel".

5. "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Limerick Generating Station Units 1 and 2," NEDC-32193P, Revision 2, October 1993.

3/4.4.7 MAIN STEAM LINE ISOLATION VALVES

Double isolation valves are provided on each of the main steam lines to minimize the potential leakage paths from the containment in case of a line break. Only one valve in each line is required to maintain the integrity of the containment, however, single failure considerations require that two valves be OPERABLE. The surveillance requirements are based on the operating history of this type valve. The maximum closure time has been selected to contain fission products and to ~~ensure the core is not uncovered~~ prevent core damage following line breaks. The minimum closure time is consistent with the assumptions in the safety analyses to prevent pressure surges.

3/4.4.8 STRUCTURAL INTEGRITY

The inspection programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant.

Components of the reactor coolant system were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code 1971 Edition and Addenda through Winter 1972.

The inservice inspection program for ASME Code Class 1, 2, and 3 components will be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a. Additionally, the Inservice Inspection Program conforms to the NRC staff positions identified in NRC Generic Letter 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping," as approved in NRC Safety Evaluations dated March 6, 1990 and October 22, 1990.

3/4.4.9 RESIDUAL HEAT REMOVAL

The RHR system is required to remove decay heat and sensible heat in order to maintain the temperature of the reactor coolant. RHR shutdown cooling is comprised of four (4) subsystems which make two (2) loops. Each loop consists of two (2) motor driven pumps, a heat exchanger, and associated piping and valves. Both loops have a common suction from the same recirculation loop. Two (2) redundant, manually controlled shutdown cooling subsystems of the RHR System can provide the required decay heat removal capability. Each pump discharges the reactor coolant, after it has been cooled by circulation through the respective heat exchangers, to the reactor via the associated recirculation loop or to the reactor via the low pressure coolant injection pathway. The RHR heat exchangers transfer heat to the RHR Service Water System. The RHR shutdown cooling mode is manually controlled.

An OPERABLE RHR shutdown cooling subsystem consists of an RHR pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path. In HOT SHUTDOWN condition, the requirement to maintain OPERABLE two (2) independent RHR shutdown cooling subsystems means that each subsystem considered OPERABLE must be associated with a different heat exhanger loop, i.e., the "A" RHR heat exchanger with the "A" RHR pump or the "C" RHR pump, and the "B" RHR heat exchanger with the "B" RHR pump or the "D" RHR pump are two (2) independent RHR shutdown cooling subsystems. Only one (1) of the two (2) RHR pumps associated with each RHR heat exchanger loop is

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3/4.6.5 SECONDARY CONTAINMENT

Secondary containment is designed to minimize any ground level release of radioactive material which may result from an accident. The Reactor Enclosure and associated structures provide secondary containment during normal operation when the drywell is sealed and in service. At other times the drywell may be open and, when required, secondary containment integrity is specified.

Establishing and maintaining a vacuum in the reactor enclosure secondary containment with the standby gas treatment system once per 24 months, along with the surveillance of the doors, hatches, dampers and valves, is adequate to ensure that there are no violations of the integrity of the secondary containment.

The OPERABILITY of the reactor enclosure recirculation system and the standby gas treatment systems ensures that sufficient iodine removal capability will be available in the event of a LOCA or refueling accident* (SGTS only). The reduction in containment iodine inventory reduces the resulting SITE BOUNDARY radiation doses associated with containment leakage. The operation of this system and resultant iodine removal capacity are consistent with the assumptions used in the LOCA and refueling accident analyses. Provisions have been made to continuously purge the filter plenums with instrument air when the filters are not in use to prevent buildup of moisture on the adsorbers and the HEPA filters.

INVOLVING RECENTLY IRRADIATED FUEL

Although the safety analyses assumes that the reactor enclosure secondary containment draw down time will take 930 seconds, these surveillance requirements specify a draw down time of 916 seconds. This 14 second difference is due to the diesel generator starting and sequence loading delays which is not part of this surveillance requirement.

The reactor enclosure secondary containment draw down time analyses assumes a starting point of 0.25 inch of vacuum water gauge and worst case SGTS dirty filter flow rate of 2800 cfm. The surveillance requirements satisfy this assumption by starting the drawdown from ambient conditions and connecting the adjacent reactor enclosure and refueling area to the SGTS to split the exhaust flow between the three zones and verifying a minimum flow rate of 2800 cfm from the test zone. This simulates the worst case flow alignment and verifies adequate flow is available to drawdown the test zone within the required time. The Technical Specification Surveillance Requirement 4.6.5.3.b.3 is intended to be a multi-zone air balance verification without isolating any test zone.

The SGTS is common to Unit 1 and 2 and consists of two independent subsystems. The power supplies for the common portions of the subsystems are from Unit 1 safeguard busses, therefore the inoperability of these Unit 1 supplies are addressed in the SGTS ACTION statements in order to ensure adequate onsite power sources to SGTS for its Unit 2 function during a loss of offsite power event. The allowable out of service times are consistent with those in the Unit 1 Technical Specifications for SGTS and AC electrical power supply out of service condition combinations.

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3/4.7.1 SERVICE WATER SYSTEMS - COMMON SYSTEMS

The OPERABILITY of the service water systems ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of these systems, assuming a single failure, is consistent with the assumptions used in the accident conditions within acceptable limits.

The RHRSW and ESW systems are common to Units 1 and 2 and consist of two independent subsystems each with two pumps. One pump per subsystem (loop) is powered from a Unit 1 safeguard bus and the other pump is powered from a Unit 2 safeguard bus. In order to ensure adequate onsite power sources to the systems during a loss of offsite power event, the inoperability of these supplies are restricted in system ACTION statements.

RHRSW is a manually operated system used for core and containment heat removal. Each of two RHRSW subsystems has one heat exchanger per unit. Each RHRSW pump provides adequate cooling for one RHR heat exchanger. By limiting operation with less than three OPERABLE RHRSW pumps with OPERABLE Diesel Generators, each unit is ensured adequate heat removal capability for the design scenario of LOCA/LOOP on one unit and simultaneous safe shutdown of the other unit.

Each ESW pump provides adequate flow to the cooling loads in its associated loop. With only two divisions of power required for LOCA mitigation of one unit and one division of power required for safe shutdown of the other unit, one ESW pump provides sufficient capacity to fulfill design requirements. ESW pumps are automatically started upon start of the associated Diesel Generators. Therefore, the allowable out of service times for OPERABLE ESW pumps and their associated Diesel Generators is limited to ensure adequate cooling during a loss of offsite power event.

3/4.7.2 CONTROL ROOM EMERGENCY FRESH AIR SUPPLY SYSTEM - COMMON SYSTEM

Total Effective Dose Equivalent

The OPERABILITY of the control room emergency fresh air supply system ensures that the control room will remain habitable for operations personnel during and following all design basis accident conditions. Constant purge of the system at 1 cfm is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less ~~whole body, or its equivalent~~. This limitation is consistent with the requirements of ~~General Design Criterion 19 of Appendix A,~~ 10 CFR Part ~~50~~ 50.67, Accident Source Terms.

SERT 4

The CREFAS is common to Units 1 and 2 and consists of two independent subsystems. The power supplies for the system are from Unit 1 Safeguard busses, therefore, the inoperability of these Unit 1 supplies are addressed in the CREFAS ACTION statements in order to ensure adequate onsite power sources to CREFAS during a loss of offsite power event. The allowable out of service

Unit 2 Limerick AST LAR Bases Inserts

INSERT 1 (Page B3/4 1-4)

The above quantities calculated at 29% Boron-10 enrichment have been demonstrated by analysis to provide a Boron-10 weight equivalent of 185 lbs in the sodium pentaborate solution. Maintaining this Boron-10 weight in the net tank contents ensures a sufficient quantity of boron to bring the reactor to a cold, Xenon-free shutdown.

INSERT 2 (Page B3/4 1-5)

The Standby Liquid Control System also has a post-DBA LOCA safety function to buffer Suppression Pool pH in order to maintain bulk pH above 7.0. The buffering of Suppression Pool pH is necessary to prevent iodine re-evolution to satisfy the methodology for Alternative Source Term. Manual initiation is used, and the minimum amount of total boron required for Suppression Pool pH buffering is 240 lbs. Given that at least 185 lbs of Boron-10 is maintained in the tank, the total boron in the tank will be greater than 240 lbs for the range of enrichments from 29% to 62%.

ACTION Statement (a) applies only to OPERATIONAL CONDITIONS 1 and 2 because a single pump can satisfy both the reactor control function and the post-DBA LOCA function to control Suppression Pool pH since boron injection is not required until 13 hours post-LOCA. ACTION Statement (b) applies to OPERATIONAL CONDITIONS 1, 2 and 3 to address the post-LOCA safety function of the SLC system.

INSERT 3 (Page B3/4 6-5)

Based on implementation of the AST methodology, it has been demonstrated through analysis that operating the subsystem with a minimum flow rate of 30,000 cfm satisfies the dose requirements. However, based on the applicable testing standards and guidance provided in Regulatory Guide 1.52, Revision 2, and ANSI N510, 1980, the acceptance criteria must be satisfied based on the system rated flow, which provides the most conservative results. Satisfying the acceptance criteria at rated flow also demonstrates system operability at lower measured flows because the residence time is longer. Therefore, verification of subsystem flow rates between 30,000 cfm and the maximum rated flow of 66,000 cfm (includes a 10% factor) satisfies the surveillance requirements for the HEPA filters and charcoal adsorber housings based on 70% efficiencies.

INSERT 4 (Page B3/4 7-1)

Since the Control Room Emergency Fresh Air Supply System is not credited for filtration in OPERATIONAL CONDITIONS 4 and 5, applicability to 4 and 5 is only required to support the Chlorine and Toxic Gas design basis isolation requirements.

Additionally, based on implementation of the Alternative Source Term methodology, it has been demonstrated through analysis that manual initiation of the CREFAS radiation mode within 30 minutes of the start of gap release for the limiting design basis LOCA is sufficient to assure that control room operator dose limits in 10 CFR Part 50.67 are met.

ATTACHMENT 4

LIMERICK GENERATING STATION
UNITS 1 AND 2

Docket Nos. 50-352 & 50-353

License Nos. NPF-39 & NPF-85

License Amendment Request
"LGS Alternative Source Term Implementation"

Retyped Technical Specification Pages

UNIT 1	UNIT 2
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3/4 1-19	3/4 1-19
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3/4 6-52	3/4 6-52
3/4 6-53	3/4 6-53
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3/4 7-5	3/4 7-5
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3/4 8-14A	3/4 8-14A
3/4 8-20	3/4 8-19
	3/4 8-20

CORE ALTERATION

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

a) Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special moveable detectors (including undervessel replacement); and
b) Control rod movement, provided there are no fuel assemblies in the associated core cell.

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

CORE OPERATING LIMITS REPORT

1.7a The CORE OPERATING LIMITS REPORT (COLR) is the unit-specific document that provides the core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specifications 6.9.1.9 thru 6.9.1.12. Plant operation within these limits is addressed in individual specifications.

CRITICAL POWER RATIO

1.8 The CRITICAL POWER RATIO (CPR) shall be the ratio of that power in the assembly which is calculated by application of the (GEXL) correlation to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.

DOSE EQUIVALENT I-131

1.9 DOSE EQUIVALENT I-131 shall be that concentration of I-131, microcuries per gram, which alone would produce the same inhalation committed effective dose equivalent (CEDE) as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The inhalation committed effective dose equivalent (CEDE) conversion factors used for this calculation shall be those listed in Table 2.1 of Federal Guidelines Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," ORNL, 1989, as described in Regulatory Guide 1.183. The factors in the column headed "effective" yield doses corresponding to the CEDE.

DOWNSCALE TRIP SETPOINT (DTSP)

1.9a The downscale trip setpoint associated with the Rod Block Monitor (RBM) rod block trip setting.

$\bar{E}$-AVERAGE DISINTEGRATION ENERGY

1.10 $\bar{E}$ shall be the average, weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling, of the sum of the average beta and gamma energies per disintegration, in MeV, for isotopes, with half lives greater than 15 minutes, making up at least 95% of the total noniodine activity in the coolant.

EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME

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

PURGE - PURGING

1.31 PURGE or PURGING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

RATED THERMAL POWER

1.32 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3458 MWt.

REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY

1.33 REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY shall exist when:

a. All reactor enclosure secondary containment penetrations required to be closed during accident conditions are either:

1. Capable of being closed by an OPERABLE secondary containment automatic isolation system, or

2. Closed by at least one manual valve, blind flange, slide gate damper, or deactivated automatic valve secured in its closed position, except as provided by Specification 3.6.5.2.1.

b. All reactor enclosure secondary containment hatches and blowout panels are closed and sealed.

c. The standby gas treatment system is in compliance with the requirements of Specification 3.6.5.3.

d. The reactor enclosure recirculation system is in compliance with the requirements of Specification 3.6.5.4.

e. At least one door in each access to the reactor enclosure secondary containment is closed.

f. The sealing mechanism associated with each reactor enclosure secondary containment penetration, e.g., welds, bellows, or O-rings, is OPERABLE.

g. The pressure within the reactor enclosure secondary containment is less than or equal to the value required by Specification 4.6.5.1.1a.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.34 REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

RECENTLY IRRADIATED FUEL

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

REFUELING FLOOR SECONDARY CONTAINMENT INTEGRITY

1.36 REFUELING FLOOR SECONDARY CONTAINMENT INTEGRITY shall exist when:

a. All refueling floor secondary containment penetrations required to be closed during accident conditions are either:

REFUELING FLOOR SECONDARY CONTAINMENT INTEGRITY (Continued)

1. Capable of being closed by an OPERABLE secondary containment automatic isolation system, or

2. Closed by at least one manual valve, blind flange, slide gate damper, or deactivated automatic valve secured in its closed position, except as provided by Specification 3.6.5.2.2.

b. All refueling floor secondary containment hatches and blowout panels are closed and sealed.

c. The standby gas treatment system is in compliance with the requirements of specification 3.6.5.3.

d. At least one door in each access to the refueling floor secondary containment is closed.

e. The sealing mechanism associated with each refueling floor secondary containment penetration, e.g., welds, bellows, or O-rings, is OPERABLE.

f. The pressure within the refueling floor secondary containment is less than or equal to the value required by Specification 4.6.5.1.2a.

REPORTABLE EVENT

1.37 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

ROD DENSITY

1.38 ROD DENSITY shall be the number of control rod notches inserted as a fraction of the total number of control rod notches. All rods fully inserted is equivalent to 100% ROD DENSITY.

SHUTDOWN MARGIN

1.39 SHUTDOWN MARGIN shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming all control rods are fully inserted except for the single control rod of highest reactivity worth which is assumed to be fully withdrawn and the reactor is in the shutdown condition; cold, i.e. 68°F; and xenon free.

SITE BOUNDARY

1.40 The SITE BOUNDARY shall be that line as defined in Figure 5.1.3-1a.

SOURCE CHECK

1.41 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.1.5 The standby liquid control system shall be OPERABLE and consist of a minimum of the following:

a. In OPERATIONAL CONDITIONS 1 and 2, two pumps and corresponding flow paths,

b. In OPERATIONAL CONDITION 3, one pump and corresponding flow path.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3

ACTION:

a. With only one pump and corresponding explosive valve OPERABLE, in OPERATIONAL CONDITION 1 or 2, restore one inoperable pump and corresponding explosive valve to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With standby liquid control system otherwise inoperable, in OPERATIONAL CONDITION 1, 2, or 3, restore the system to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.1.5 The standby liquid control system shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying that:

1. The temperature of the sodium pentaborate solution is within the limits of Figure 3.1.5-1.

2. The available volume of sodium pentaborate solution is at least 3160 gallons.

3. The temperature of the pump suction piping is within the limits of Figure 3.1.5-1 for the most recent concentration analysis.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days by:

1. Verifying the continuity of the explosive charge.

2. Determining by chemical analysis and calculation* that the available weight of Boron-10 is greater than or equal to 185 lbs; the concentration of sodium pentaborate in solution is less than or equal to 13.8% and within the limits of Figure 3.1.5-1 and; the following equation is satisfied:

$$\frac{C}{13\%\ wt.} \times \frac{E}{29\ atom\ \%} \times \frac{Q}{86\ gpm} \geq 1$$

where

C = Sodium pentaborate solution (% by weight)
Q = Two pump flowrate, as determined per surveillance requirement 4.1.5.c.
E = Boron 10 enrichment (atom % Boron 10)

3. Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. Demonstrating that, when tested pursuant to Specification 4.0.5, the minimum flow requirement of 41.2 gpm per pump at a pressure of greater than or equal to 1230 ± 25 psig is met.

d. At least once per 24 months during shutdown by:

1. Initiating at least one of the standby liquid control system loops, including an explosive valve, and verifying that a flow path from the pumps to the reactor pressure vessel is available by pumping demineralized water into the reactor vessel. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch which has been certified by having one of the batch successfully fired. All injection loops shall be tested in 3 operating cycles.

2. Verify all heat-treated piping between storage tank and pump suction is unblocked.**

e. Prior to addition of Boron to storage tank verify sodium pentaborate enrichment to be added is ≥ 29 atom % Boron 10.

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* This test shall also be performed anytime water or boron is added to the solution or when the solution temperature drops below the limits of Figure 3.1.5-1 for the most recent concentration analysis, within 24 hours after water or boron addition or solution temperature is restored.

** This test shall also be performed whenever suction piping temperature drops below the limits of Figure 3.1.5-1 for the most recent concentration analysis, within 24 hours after solution temperature is restored.

TABLE 3.3.2-1 (Continued)
ISOLATION ACTUATION INSTRUMENTATION
ACTION STATEMENTS

ACTION 20 - Be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24-hours.

ACTION 21 - Be in at least STARTUP with the associated isolation valves closed within 6 hours or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION 22 - Be in at least STARTUP within 6 hours.

ACTION 23 - In OPERATIONAL CONDITION 1 or 2, verify the affected system isolation valves are closed within 1 hour and declare the affected system inoperable. In OPERATIONAL CONDITION 3, be in at least COLD SHUTDOWN within 12 hours.

ACTION 24 - Restore the manual initiation function to OPERABLE status within 8 hours or close the affected system isolation valves within the next hour and declare the affected system inoperable or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

ACTION 25 - Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within 1 hour.

ACTION 26 - Close the affected system isolation valves within 1 hour.

TABLE NOTATIONS

* Required when (1) handling RECENTLY IRRADIATED FUEL in the secondary containment, or (2) during operations with a potential for draining the reactor vessel with the vessel head removed and fuel in the vessel.

** May be bypassed under administrative control with all turbine stop valves closed.

\# During operation of the associated Unit 1 [illegible] ventilation exhaust system.

(a) DELETED

(b) A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter. Trip functions common to RPS Actuation Instrumentation are shown in Table 4.3.2.1-1. In addition, for the HPCI system and RCIC system isolation, provided that the redundant isolation valve, inboard or outboard, as applicable, in each line is OPERABLE and all required actuation instrumentation for that valve is OPERABLE, one channel may be placed in an inoperable status for up to 8 hours for required surveillance without placing the channel or trip system in the tripped condition.

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
7. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level Low, Low - Level 2	S	Q	R	1, 2, 3
b. Drywell Pressure## - High	S	Q	R	1, 2, 3
c.1. Refueling Area Unit 1 Ventilation Exhaust Duct Radiation - High	S	Q	R	*#
2. Refueling Area Unit 2 Ventilation Exhaust Duct Radiation - High	S	Q	R	*#
d. Reactor Enclosure Ventilation Exhaust Duct Radiation - High	S	Q	R	1, 2, 3
e. Deleted				
f. Deleted				
g. Reactor Enclosure Manual Initiation	N.A.	R	N.A.	1, 2, 3
h. Refueling Area Manual Initiation	N.A.	R	N.A.	*

*Required when (1) handling RECENTLY IRRADIATED FUEL in the secondary containment, or (2) during operations with a potential for draining the reactor vessel with the vessel head removed and fuel in the vessel.

**When not administratively bypassed and/or when any turbine stop valve is open.

#During operation of the associated Unit 1 or Unit 2 ventilation exhaust system.

##These trip functions (2a, 6b, and 7b) are common to the RPS actuation trip function.

TABLE 3.3.7.1-1

RADIATION MONITORING INSTRUMENTATION

INSTRUMENTATION	MINIMUM CHANNELS OPERABLE	APPLICABLE CONDITIONS	ALARM/TRIP SETPOINT	ACTION
1. Main Control Room Normal Fresh Air Supply Radiation Monitor	4	1,2,3, and *	1×10^{-5} μCi/cc[b]	70
2. Area Monitors				
a. Criticality Monitors				
1) Spent Fuel Storage Pool	2	(a)	≥ 5 mR/h and ≤20mR/h[b]	71
b. Control Room Direct Radiation Monitor	1	At All Times	N.A.[b]	73
3. Reactor Enclosure Cooling Water Radiation Monitor	1	At All Times	≤ 3 x Background[b]	72

TABLE 3.3.7.1-1 (Continued)

RADIATION MONITORING INSTRUMENTATION

TABLE NOTATIONS

*When RECENTLY IRRADIATED FUEL is being handled in the secondary containment or during operations with a potential for draining the reactor vessel with the vessel head removed and fuel in the vessel.

(a) With fuel in the spent fuel storage pool.

(b) Alarm only.

ACTION STATEMENTS

ACTION 70 - With one monitor inoperable, restore the inoperable monitor to the OPERABLE status within 7 days or, within the next 6 hours, initiate and maintain operation of the control room emergency filtration system in the radiation isolation mode of operation.

With two or more of the monitors inoperable, within one hour, initiate and maintain operation of the control room emergency filtration system in the radiation mode of operation.

ACTION 71 - With one of the required monitor inoperable, assure a portable continuous monitor with the same alarm setpoint is OPERABLE in the vicinity of the installed monitor during any fuel movement. If no fuel movement is being made, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

ACTION 72 - With the required monitor inoperable, obtain and analyze at least one grab sample of the monitored parameter at least once per 24 hours.

ACTION 73 - With the required monitor inoperable, assure a portable alarming monitor is OPERABLE in the vicinity of the installed monitor or perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

TABLE 4.3.7.1-1

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTATION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE IS REQUIRED
1. Main Control Room Normal Fresh Air Supply Radiation Monitor	S	Q	R	1, 2, 3, and *
2. Area Monitors				
a. Criticality Monitors				
1) Spent Fuel Storage Pool	S	M	R	(a)
b. Control Room Direct Radiation Monitor	S	M	R	At All Times
3. Reactor Enclosure Cooling Water Radiation Monitor	S	M	R(b)	At All Times

TABLE 4.3.7.1-1 (Continued)

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATIONS

*When RECENTLY IRRADIATED FUEL is being handled in the secondary containment or during operations with a potential for draining the reactor vessel with the vessel head removed and fuel in the vessel.

(a) With fuel in the spent fuel storage pool.

(b) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards (NBS) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

3/4.4.7 MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.4.7 Two main steam line isolation valves (MSIVs) per main steam line shall be OPERABLE with closing times greater than or equal to 3 and less than or equal to 10 seconds.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one or more MSIVs inoperable:

   1. Maintain at least one MSIV OPERABLE in each affected main steam line that is open and within 8 hours, either:

      a) Restore the inoperable valve(s) to OPERABLE status, or

      b) Isolate the affected main steam line by use of a deactivated MSIV in the closed position.

   2. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.7 Each of the above required MSIVs shall be demonstrated OPERABLE by verifying full closure between 3 and 10 seconds when tested pursuant to Specification 4.0.5.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

b. The combined leakage rate to be in accordance with the Primary Containment Leakage Rate Testing Program for all primary containment penetrations and all primary containment isolation valves that are subject to Type B and C tests, except for: main steam line isolation valves*, valves which are hydrostatically tested, and those valves where an exemption to Appendix J of 10 CFR 50 has been granted, and

c. The leakage rate to ≤100 scf per hour for any main steam isolation valve that exceeds 100 scf per hour, and restore the combined maximum pathway leakage to ≤200 scf per hour, and

d. The combined leakage rate for all containment isolation valves in hydrostatically tested lines which penetrate the primary containment to less than or equal to 1 gpm times the total number of such valves,

prior to increasing the reactor coolant system temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The primary containment leakage rates shall be demonstrated to be in accordance with the Primary Containment Leakage Rate Testing Program, or approved exemptions, for the following:

a. Type A Test

b. Type B and C Tests (including air locks)

c. Main Steam Line Isolation Valves

d. Hydrostatically tested Containment Isolation Valves

* Exemption to Appendix "J" to 10 CFR Part 50.

3/4.6.5 SECONDARY CONTAINMENT

REFUELING AREA SECONDARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.5.1.2 REFUELING AREA SECONDARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: When RECENTLY IRRADIATED FUEL is being handled in the secondary containment, or during operations with a potential for draining the reactor vessel, with the vessel head removed and fuel in the vessel.

ACTION:

Without REFUELING AREA SECONDARY CONTAINMENT INTEGRITY, suspend handling of RECENTLY IRRADIATED FUEL in the secondary containment and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.1.2 REFUELING AREA SECONDARY CONTAINMENT INTEGRITY shall be demonstrated by:

a. Verifying at least once per 24 hours that the pressure within the refueling area secondary containment is greater than or equal to 0.25 inch of vacuum water gauge.

b. Verifying at least once per 31 days that:

   1. All refueling area secondary containment equipment hatches and blowout panels are closed and sealed.

   2. At least one door in each access to the refueling area secondary containment is closed.

   3. All refueling area secondary containment penetrations not capable of being closed by OPERABLE secondary containment automatic isolation dampers/valves and required to be closed during accident conditions are closed by valves, blind flanges, slide gate dampers or deactivated automatic dampers/valves secured in position.

c. At least once per 24 months:

   Operating one standby gas treatment subsystem for one hour and maintaining greater than or equal to 0.25 inch of vacuum water gauge in the refueling area secondary containment at a flow rate not exceeding 764 cfm.

REFUELING AREA SECONDARY CONTAINMENT AUTOMATIC ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.5.2.2 The refueling area secondary containment ventilation system automatic isolation valves shall be OPERABLE.

APPLICABILITY: When RECENTLY IRRADIATED FUEL is being handled in the secondary containment, or during operations with a potential for draining the reactor vessel, with the vessel head removed and fuel in the vessel.

ACTION:

With one or more of the refueling area secondary containment ventilation system automatic isolation valves inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and within 8 hours either:

a. Restore the inoperable valves to OPERABLE status, or

b. Isolate each affected penetration by use of at least one deactivated valve secured in the isolation position, or

c. Isolate each affected penetration by use of at least one closed manual valve, blind flange or slide gate damper.

Otherwise, suspend handling of RECENTLY IRRADIATED FUEL in the refueling area secondary containment, and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.2.2 Each refueling area secondary containment ventilation system automatic isolation valve shall be demonstrated OPERABLE:

a. Prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by cycling the valve through at least one complete cycle of full travel and verifying the specified isolation time.

b. At least once per 24 months by verifying that on a containment isolation test signal each isolation valve actuates to its isolation position.

c. By verifying the isolation time to be within its limit at least once per 92 days.

STANDBY GAS TREATMENT SYSTEM - COMMON SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.5.3 Two independent standby gas treatment subsystems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, and when RECENTLY IRRADIATED FUEL is being handled in the secondary containment, or during operations with a potential for draining the reactor vessel, with the vessel head removed and fuel in the vessel.

ACTION:

a. With one standby gas treatment subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days, or:

   1. In OPERATIONAL CONDITION 1, 2, or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   2. When RECENTLY IRRADIATED FUEL is being handled in the secondary containment, or during operations with a potential for draining the reactor vessel, with the vessel head removed and fuel in the vessel, suspend handling of RECENTLY IRRADIATED FUEL in the secondary containment and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

b. With both standby gas treatment subsystems inoperable, if in progress, suspend handling of RECENTLY IRRADIATED FUEL in the secondary containment or operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3. are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.3 Each standby gas treatment subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates with the heaters OPERABLE.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 24* months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the subsystem by:

1. Verifying that the subsystem satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 5764 cfm ± 10%.

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration of less than 1.25% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F), at a relative humidity of 70% and at a face velocity of 66 fpm.

3. Verify that when the fan is running the subsystem flowrate is 2800 cfm minimum from each reactor enclosure (Zones I and II) and 2200 cfm minimum from the refueling area (Zone III) when tested in accordance with ANSI N510-1980.

4. Verify that the pressure drop across the refueling area to SGTS prefilter is less than 0.25 inches water gage while operating at a flow rate of 2400 cfm ± 10%.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration of less than 1.25% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F), at a relative humidity of 70% and at a face velocity of 66 fpm.

d. At least once per 24 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 9.1 inches water gauge while operating the filter train at a flow rate of 8400 cfm ± 10%.

*Surveillance interval is an exception to the guidance provided in Regulatory Guide 1.52, Revision 2, March 1978.

REACTOR ENCLOSURE RECIRCULATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.5.4 Two independent reactor enclosure recirculation subsystems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one reactor enclosure recirculation subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With both reactor enclosure recirculation subsystems inoperable, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hors.

SURVEILLANCE REQUIREMENTS

4.6.5.4 Each reactor enclosure recirculation subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates properly (flow at a minimum of 30,000 cfm).

b. At least once per 24* months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the subsystem by:

   1. Verifying that the subsystem satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, at rated flow (60,000 cfm ± 10%).

   2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration of less than 15% when tested in accordance with ATM D3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70%.

   3. Verifying a subsystem flow rate within a range of 30,000 cfm to 66,000 cfm during system operation when tested in accordance with ANSI N510-1980.

*Surveillance interval is an exception to the guidance provided in Regulatory Guide 1.52, Revision 2, March 1978.

SURVEILLANCE REQUIREMENTS (Continued)

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows, the methyl iodide penetration of less than 15% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70%.

d. At least once per 24 months by:

1. Verifying that the pressure drop across the combined prefilter, upstream and downstream HEPA filters, and charcoal adsorber banks is less than 6 inches water gauge while operating the filter train at rated flow (60,000 cfm ± 10%), verifying that the prefilter pressure drop is less than 0.8 inch water gauge and that the pressure drop across each HEPA is less than 2 inches water gauge.

2. Verifying that the filter train starts and the isolation valves which take suction on and return to the reactor enclosure open on each of the following test signals:

a. Manual initiation from the control room, and

b. Simulated automatic initiation signal.

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter bank satisfies the inplace penetration and leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 while operating the system at rated flow (60,000 cfm ± 10%).

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorber bank satisfies the inplace penetration and leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at rated flow (60,000 cfm ± 10%).

EMERGENCY SERVICE WATER SYSTEM - COMMON SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least the following independent emergency service water system loops, with each loop comprised of:

a. Two OPERABLE emergency service water pumps, and

b. An OPERABLE flow path capable of taking suction from the emergency service water pumps wet pits which are supplied from the spray pond or the cooling tower basin and transferring the water to the associated Unit 1 and common safety-related equipment,

shall be OPERABLE:

a. Two loops, in OPERATIONAL CONDITIONS 1, 2, and 3.

b. One loop, in OPERATIONAL CONDITIONS 4, 5, and when RECENTLY IRRADIATED FUEL is being handled in the secondary containment, or during operations with a potential for draining the reactor vessel.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, 5, and when RECENTLY IRRADIATED FUEL is being handled in the secondary containment, or during operations with a potential for draining the reactor vessel.

ACTION:

a. In OPERATION CONDITION 1, 2, or 3:

1. With one emergency service water pump inoperable, restore the inoperable pump to OPERABLE status within 45 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With one emergency service water pump in each loop inoperable, restore at least one inoperable pump to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3. With one emergency service water system loop otherwise inoperable, declare all equipment aligned to the inoperable loop inoperable**, restore the inoperable loop to OPERABLE status with at least one OPERABLE pump within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

**The diesel generators may be aligned to the OPERABLE emergency service water system loop provided confirmatory flow testing has been performed. Those diesel generators no aligned to the OPERABLE emergency service water system loop shall be declared inoperable and the actions of 3.8.1.1 taken.

ACTION: (Continued)

4. With three ESW pump/diesel generator pairs** inoperable, restore at least one inoperable ESW pump/diesel generator pair** to OPERABLE status within 72 hours, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

5. With four ESW pump/diesel generator pairs** inoperable, restore at least one inoperable ESW pump/diesel generator pair** to OPERABLE status within 8 hours, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 4 or 5:

1. With only one emergency service water pump and its associated flowpath OPERABLE, restore at least two pumps with at least one flow path to OPERABLE status within 72 hours or declare the associated safety related equipment inoperable and take the ACTION required by Specifications 3.5.2 and 3.8.1.2.

c. When RECENTLY IRRADIATED FUEL is being handled in the secondary containment, or during operations with a potential for draining the reactor vessel:

1. With only one emergency service water pump and its associated flow path OPERABLE, restore at least two pumps with at least one flow path to OPERABLE status within 72 hours or verify adequate cooling remains available for the diesel generators required to be OPERABLE or be in at least the associated diesel generator(s) inoperable and take the ACTION required by Specification 3.8.1.2. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENT

4.7.1.2 At least the above required emergency service water system loop(s) shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 24 months by verifying that:

1. Each automatic valve actuates to its correct position on its appropriate ESW pump start signal.

2. Each pump starts automatically when its associated diesel generator starts.

**An ESW pump/diesel generator pair consists of an ESW pump and its associated diesel generator. If either an ESW pump or its associated diesel generator becomes inoperable, then the ESW pump/diesel generator pair is inoperable.

ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The spray pond shall be OPERABLE with:

a. A minimum pond water level at or above elevation 250' 10" Mean Sea Level, and

b. A pond water temperature of less than or equal to 88°F.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, 5, and when RECENTLY IRRADIATED FUEL is being handled in the secondary containment, or during operations with a potential for draining the reactor vessel.

ACTION:

With the requirements of the above specification not satisfied:

a. In OPERATIONAL CONDITION 1, 2, or 3, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. In OPERATIONAL CONDITION 4 or 5, declare the RHRSW system and the emergency service water system inoperable and take the ACTION required by Specifications 3.7.1.1 and 3.7.1.2.

c. When RECENTLY IRRADIATED FUEL is being handled in the secondary containment, or during operations with a potential for draining the reactor vessel, declare the emergency service water system inoperable and take the ACTION required by Specification 3.7.1.2. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.3 The spray pond shall be determined OPERABLE:

a. By verifying the pond water level to be greater than its limit at least once per 24 hours.

b. By verifying the water surface temperature (within the upper two feet of the surface) to be less than or equal to 88°F:

1. at least once per 4 hours when the spray pond temperature is greater than or equal to 80°F; and

2. at least once per 2 hours when the spray pond temperature is greater than or equal to 85°F; and

3. at least once per 24 hours when the spray pond temperature is greater than 32°F.

c. By verifying all piping above the frost line is drained:

1. within one (1) hour after being used when ambient air temperature is below 40°F; or

2. when ambient air temperature falls below 40°F if the piping has not been previously drained.

3/4.7.2 CONTROL ROOM EMERGENCY FRESH AIR SUPPLY SYSTEM - COMMON SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.2 Two independent control room emergency fresh air supply system subsystems shall be OPERABLE.

APPLICABILITY: All OPERATIONAL CONDITIONS and when RECENTLY IRRADIATED FUEL is being handled in the secondary containment, or during operations with a potential for draining the reactor vessel.

ACTION:

a. In OPERATIONAL CONDITION 1, 2, or 3 with one control room emergency fresh air supply subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 4, 5, or when RECENTLY IRRADIATED FUEL is being handled in the secondary containment, or during operations with a potential for draining the reactor vessel:

   1. With one control room emergency fresh air supply subsystems inoperable, restore the inoperable subsystem to OPERABLE status within 7 days or initiate and maintain operation of the OPERABLE subsystem in the radiation isolation mode of operation.

   2. With both control room emergency fresh air supply subsystems inoperable, suspend handling of RECENTLY IRRADIATED FUEL in the secondary containment and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.2 Each control room emergency fresh air supply subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying the control room air temperature to be less than or equal to 85°F effective temperature.

b. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates with the heaters OPERABLE.

c. At least once per 24** months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the subsystem by:

   1. Verifying that the subsystem satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 3000 cfm ± 10%.

** Surveillance interval is an exception to the guidance provided in Regulatory Guide 1.52, Revision 2, March 1978.

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration of less than 10% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70%.

3. Verifying a subsystem flow rate of 3000 cfm ± 10% during subsystem operation when tested in accordance with ANSI N510-1980.

d. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration of less than 10% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70%.

e. At least once per 24 months by:

1. Verifying that the pressure drop across the combined prefilter, upstream and downstream HEPA filters, and charcoal adsorber banks is less than 6 inches water gauge while operating the subsystem at a flow rate of 3000 cfm ± 10%; verifying that the prefilter pressure drop is less than 0.8 inch water gauge and that the pressure drop across each HEPA is less than 2 inches water gauge.

2. Verifying that on each of the below chlorine isolation mode actuation test signals, the subsystem automatically switches to the chlorine isolation mode of operation and the isolation valves close within 5 seconds:

a) Outside air intake high chlorine, and

b) Manual initiation from the control room.

3. Verifying that on manual initiation from the control room, the subsystem switches to the radiation isolation mode of operation and the control room is maintained at a positive pressure of at least 1/8 inch water gauge relative to the turbine enclosure and auxiliary equipment room and outside atmosphere during subsystem operation with an outdoor air flow rate less than or equal to 525 cfm.

A.C. SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. Two diesel generators each with:

1. A day fuel tank containing a minimum of 200 gallons of fuel.

2. A fuel storage system containing a minimum of 33,500 gallons of fuel.

3. A fuel transfer pump.

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5, and when RECENTLY IRRADIATED FUEL is being handled in the secondary containment, or during operations with a potential for draining the reactor vessel.

ACTION:

a. With less than the above required A.C. electrical power sources OPERABLE, suspend CORE ALTERATIONS, handling of RECENTLY IRRADIATED FUEL in the secondary containment, operations with a potential for draining the reactor vessel and crane operations over the spent fuel storage pool when fuel assemblies are stored therein. In addition, when in OPERATIONAL CONDITION 5 with the water level less than 22 feet above the reactor pressure vessel flange, immediately initiate corrective action to restore the required power sources to OPERABLE status as soon as practical.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.1.2 At least the above required A.C. electrical power sources shall be demonstrated OPERABLE per Surveillance Requirements 4.8.1.1.1, 4.8.1.1.2, and 4.8.1.1.3.

D.C. SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, two of the following four divisions of the D.C. electrical power sources system shall be OPERABLE with:

a. Division 1, Consisting of:
   1. 125-Volt Battery 1A1 (1A1D101).
   2. 125-Volt Battery 1A2 (1A2D101).
   3. 125-Volt Battery Charger 1BCA1 (1A1D103).
   4. 125-Volt Battery Charger 1BCA2 (1A2D103).

b. Division 2, Consisting of:
   1. 125-Volt Battery 1B1 (1B1D101).
   2. 125-Volt Battery 1B2 (1B2D101).
   3. 125-Volt Battery Charger 1BCB1 (1B1D103).
   4. 125-Volt Battery Charger 1BCB2 (1B2D103).

c. Division 3, Consisting of:
   1. 125-Volt Battery 1C (1CD101).
   2. 125-Volt Battery Charger 1BCC (1CD103).

d. Division 4, Consisting of:
   1. 125-Volt Battery 1D (1DD101).
   2. 125-Volt Battery Charger 1BCD (1DD103).

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5, and when RECENTLY IRRADIATED FUEL is being handled in the secondary containment, or during operations with a potential for draining the reactor vessel.

ACTION:

a. With one or two required battery chargers or one required division inoperable:

   1. Restore battery terminal voltage to greater than or equal to the minimum established float voltage within 2 hours,

   2. Verify associated Division 1 or 2 float current $\leq$ 2 amps, or Division 3 or 4 float current $\leq$ 1 amp within 18 hours and once per 12 hours thereafter, and

   3. Restore battery charger(s) to OPERABLE status within 7 days.

b. With one or more required batteries inoperable due to:

   1. One or two batteries on one division with one or more battery cells float voltage < 2.07 volts, perform 4.8.2.1.a.1 and 4.8.2.1.a.2 within 2 hours for affected battery(s) and restore affected cell(s) voltage $\geq$ 2.07 volts within 24 hours.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

2. Division 1 or 2 with float current > 2 amps, or with Division 3 or 4 with float current > 1 amp, perform 4.8.2.1.a.2 within 2 hours for affected battery(s) and restore battery float current to within limits within 18 hours.

3. One or two batteries on one division with one or more cells electrolyte level less than minimum established design limits, if electrolyte level was below the top of the plates restore electrolyte level to above top of plates within 8 hours and verify no evidence of leakage(*) within 12 hours. In all cases, restore electrolyte level to greater than or equal to minimum established design limits within 31 days.

4. One or two batteries on one division with pilot cell electrolyte temperature less than minimum established design limits, restore battery pilot cell temperature to greater than or equal to minimum established design limits within 12 hours.

5. Batteries in more than one division affected, restore battery parameters for all batteries in one division to within limits within 2 hours.

6. (i) Any battery having both (Action b.1) one or more battery cells float voltage < 2.07 volts and (Action b.2) float current not within limits, and/or

   (ii) Any battery not meeting any Action b.1 through b.5,

   Restore the battery parameters to within limits within 2 hours.

c. 1. With the requirements of Action a. and/or Action b. not met, or

   2. With less than two divisions of the above required D.C. electrical power sources OPERABLE for reasons other than Actions a. and/or b.,

   Suspend CORE ALTERATIONS, handling of RECENTLY IRRADIATED FUEL in the secondary containment and operations with a potential for draining the reactor vessel.

d. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.2.2 At least the above required battery and charger shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.1.

(*) Contrary to the provisions of Specification 3.0.2, if electrolyte level was below the top of the plates, the verification that there is no evidence of leakage is required to be completed regardless of when electrolyte level is restored.

LIMITING CONDITION FOR OPERATION (Continued)

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5, and when RECENTLY IRRADIATED FUEL is being handled in the secondary containment, or during operations with a potential for draining the reactor vessel.

ACTION:

a. With less than two divisions of the above required Unit 1 A.C. distribution systems energized, suspend CORE ALTERATIONS, handling of RECENTLY IRRADIATED FUEL in the secondary containment and operations with a potential for draining the reactor vessel.

b. With less than two divisions of the above required Unit 1 D.C. distribution systems energized, suspend CORE ALTERATIONS, handling of RECENTLY IRRADIATED FUEL in the secondary containment and operations with a potential for draining the reactor vessel.

c. With any of the above required Unit 2 and common AC and/or DC distribution system divisions not energized, declare the associated common equipment inoperable, and take the appropriate ACTION for that system.

d. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.3.2 At least the above required power distribution system divisions shall be determined energized at least once per 7 days by verifying correct breaker alignment and voltage on the busses/MCCs/panels.

CORE ALTERATION

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

a) Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special moveable detectors (including undervessel replacement); and
b) Control rod movement, provided there are no fuel assemblies in the associated core cell.

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

CORE OPERATING LIMITS REPORT

1.7a The CORE OPERATING LIMITS REPORT (COLR) is the unit-specific document that provides the core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specifications 6.9.1.9 thru 6.9.12. Plant operation within these limits is addressed in individual specifications.

CRITICAL POWER RATIO

1.8 The CRITICAL POWER RATIO (CPR) shall be the ratio of that power in the assembly which is calculated by application of the (GEXL) correlation to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.

DOSE EQUIVALENT I-131

1.9 DOSE EQUIVALENT I-131 shall be that concentration of I-131, microcuries per gram, which alone would produce the same inhalation committed effective dose equivalent (CEDE) as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The inhalation committed effective dose equivalent (CEDE) conversion factors used for this calculation shall be those listed in Table 2.1 of Federal Guidelines Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," ORNL, 1989, as described in Regulatory Guide 1.183. The factors in the column headed "effective" yield doses corresponding to the CEDE.

DOWNSCALE TRIP SETPOINT (DTSP)

1.9a The downscale trip setpoint associated with the Rod Block Monitor (RBM) rod block trip setting.

$\bar{E}$-AVERAGE DISINTEGRATION ENERGY

1.10 $\bar{E}$ shall be the average, weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling, of the sum of the average beta and gamma energies per disintegration, in MeV, for isotopes, with half lives greater than 15 minutes, making up at least 95% of the total noniodine activity in the coolant.

EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME

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

PURGE - PURGING

1.31 PURGE or PURGING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

RATED THERMAL POWER

1.32 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3458 MWt.

REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY

1.33 REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY shall exist when:

a. All reactor enclosure secondary containment penetrations required to be closed during accident conditions are either:

1. Capable of being closed by an OPERABLE secondary containment automatic isolation system, or

2. Closed by at least one manual valve, blind flange, slide gate damper or deactivated automatic valve secured in its closed position, except as provided by Specification 3.6.5.2.1.

b. All reactor enclosure secondary containment hatches and blowout panels are closed and sealed.

c. The standby gas treatment system is in compliance with the requirements of Specification 3.6.5.3.

d. The reactor enclosure recirculation system is in compliance with the requirements of Specification 3.6.5.4.

e. At least one door in each access to the reactor enclosure secondary containment is closed.

f. The sealing mechanism associated with each reactor enclosure secondary containment penetration, e.g., welds, bellows, or O-rings, is OPERABLE.

g. The pressure within the reactor enclosure secondary containment is less than or equal to the value required by Specification 4.6.5.1.1a.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.34 REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

RECENTLY IRRADIATED FUEL

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

REFUELING FLOOR SECONDARY CONTAINMENT INTEGRITY

1.36 REFUELING FLOOR SECONDARY CONTAINMENT INTEGRITY shall exist when:

a. All refueling floor secondary containment penetrations required to be closed during accident conditions are either:

REFUELING FLOOR SECONDARY CONTAINMENT INTEGRITY (Continued)

1. Capable of being closed by an OPERABLE secondary containment automatic isolation system, or

2. Closed by at least one manual valve, blind flange, slide gate damper or deactivated automatic valve secured in its closed position, except as provided by Specification 3.6.5.2.2.

b. All refueling floor secondary containment hatches and blowout panels are closed and sealed.

c. The standby gas treatment system is in compliance with the requirements of Specification 3.6.5.3.

d. At least one door in each access to the refueling floor secondary containment is closed.

e. The sealing mechanism associated with each refueling floor secondary containment penetration, e.g., welds, bellows, or O-rings, is OPERABLE.

f. The pressure within the refueling floor secondary containment is less than or equal to the value required by Specification 4.6.5.1.2a.

REPORTABLE EVENT

1.37 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

ROD DENSITY

1.38 ROD DENSITY shall be the number of control rod notches inserted as a fraction of the total number of control rod notches. All rods fully inserted is equivalent to 100% ROD DENSITY.

SHUTDOWN MARGIN

1.39 SHUTDOWN MARGIN shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming all control rods are fully inserted except for the single control rod of highest reactivity worth which is assumed to be fully withdrawn and the reactor is in the shutdown condition; cold, i.e. 68°F; and xenon free.

SITE BOUNDARY

1.40 The SITE BOUNDARY shall be that line as defined in Figure 5.1.3-1a.

SOURCE CHECK

1.41 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.1.5 The standby liquid control system shall be OPERABLE and consist of a minimum of the following:

a. In OPERATIONAL CONDITIONS 1 and 2, two pumps and corresponding flow paths,

b. In OPERATIONAL CONDITION 3, one pump and corresponding flow path.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3

ACTION:

a. With only one pump and corresponding explosive valve OPERABLE, in OPERATIONAL CONDITION 1 or 2, restore one inoperable pump and corresponding explosive valve to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With standby liquid control system otherwise inoperable, in OPERATIONAL CONDITION 1, 2, or 3, restore the system to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.1.5 The standby liquid control system shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying that:

1. The temperature of the sodium pentaborate solution is within the limits of Figure 3.1.5-1.

2. The available volume of sodium pentaborate solution is at least 3160 gallons.

3. The temperature of the pump suction piping is within the limits of Figure 3.1.5-1 for the most recent concentration analysis.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days by:

1. Verifying the continuity of the explosive charge.

2. Determining by chemical analysis and calculation* that the available weight of Boron-10 is greater than or equal to 185 lbs; the concentration of sodium pentaborate in solution is less than or equal to 13.8% and within the limits of Figure 3.1.5-1 and; the following equation is satisfied:

$$\frac{C}{13\%\ \text{wt.}} \times \frac{E}{29\ \text{atom}\ \%} \times \frac{Q}{86\ \text{gpm}} \geq 1$$

where

C = Sodium pentaborate solution (% by weight)
Q = Two pump flowrate, as determined per surveillance requirement 4.1.5.c.
E = Boron 10 enrichment (atom % Boron 10)

3. Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. Demonstrating that, when tested pursuant to Specification 4.0.5, the minimum flow requirement of 41.2 gpm per pump at a pressure of greater than or equal to 1230±25 psig is met.

d. At least once per 24 months during shutdown by:

1. Initiating at least one of the standby liquid control system loops, including an explosive valve, and verifying that a flow path from the pumps to the reactor pressure vessel is available by pumping demineralized water into the reactor vessel. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch which has been certified by having one of the batch successfully fired. All injection loops shall be tested in 3 operating cycles.

2. Verify all heat-treated piping between storage tank and pump suction is unblocked.**

e. Prior to addition of Boron to storage tank verify sodium pentaborate enrichment to be added is ≥ 29 atom % Boron 10.

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* This test shall also be performed anytime water or boron is added to the solution or when the solution temperature drops below the limits of Figure 3.1.5-1 for the most recent concentration analysis, within 24 hours after water or boron addition or solution temperature is restored.

** This test shall also be performed whenever suction piping temperature drops below the limits of Figure 3.1.5-1 for the most recent concentration analysis, within 24 hours after solution temperature is restored.

TABLE 3.3.2-1 (Continued)
ISOLATION ACTUATION INSTRUMENTATION
ACTION STATEMENTS

ACTION 20 - Be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION 21 - Be in at least STARTUP with the associated isolation valves closed within 6 hours or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION 22 - Be in at least STARTUP within 6 hours.

ACTION 23 - In OPERATIONAL CONDITION 1 or 2, verify the affected system isolation valves are closed within 1 hour and declare the affected system inoperable. In OPERATIONAL CONDITION 3, be in at least COLD SHUTDOWN within 12 hours.

ACTION 24 - Restore the manual initiation function to OPERABLE status within 8 hours or close the affected system isolation valves within the next hour and declare the affected system inoperable or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

ACTION 25 - Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within 1 hour.

ACTION 26 - Close the affected system isolation valves within 1 hour.

TABLE NOTATIONS

* Required when (1) handling RECENTLY IRRADIATED FUEL in the secondary containment, or (2) during operations with a potential for draining the reactor vessel with the vessel head removed and fuel in the vessel.

** May be bypassed under administrative control, with all turbine stop valves closed.

\# During operation of the associated Unit 1 or Unit 2 ventilation exhaust system.

(a) DELETED

(b) A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter. Trip functions common to RPS Actuation Instrumentation are shown in Table 4.3.2.1-1. In addition, for the HPCI system and RCIC system isolation, provided that the redundant isolation valve, inboard or outboard, as applicable, in each line is OPERABLE and all required actuation instrumentation for that valve is OPERABLE, one channel may be placed in an inoperable status for up to 8 hours for required surveillance without placing the channel or trip system in the tripped condition.

TABLE 4.3.2.1-1 (Continued)
ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
7. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level Low, Low - Level 2	S	Q	R	1, 2, 3
b. Drywell Pressure## - High	S	Q	R	1, 2, 3
c.1. Refueling Area Unit 1 Ventilation Exhaust Duct Radiation - High	S	Q	R	*#
2. Refueling Area Unit 2 Ventilation Exhaust Duct Radiation - High	S	Q	R	*#
d. Reactor Enclosure Ventilation Exhaust Duct Radiation - High	S	Q	R	1, 2, 3
e. Deleted				
f. Deleted				
g. Reactor Enclosure Manual Initiation	N.A.	R	N.A.	1, 2, 3
h. Refueling Area Manual Initiation	N.A.	R	N.A.	*

*Required when (1) handling RECENTLY IRRADIATED FUEL in the secondary containment, or (2) during operations with a potential for draining the reactor vessel with the vessel head removed and fuel in the vessel.

**When not administratively bypassed and/or when any turbine stop valve is open.

#During operation of the associated Unit 1 or Unit 2 ventilation exhaust system.

##These trip functions (2a, 6b, and 7b) are common to the RPS actuation trip function.

TABLE 3.3.7.1-1

RADIATION MONITORING INSTRUMENTATION

INSTRUMENTATION	MINIMUM CHANNELS OPERABLE	APPLICABLE CONDITIONS	ALARM/TRIP SETPOINT	ACTION
1. Main Control Room Normal Fresh Air Supply Radiation Monitor	4	1,2,3, and *	1×10^{-5} μCi/cc[b]	70
2. Area Monitors				
a. Criticality Monitors				
1) Spent Fuel Storage Pool	2	(a)	≥ 5 mR/h and ≤20mR/h[b]	71
b. Control Room Direct Radiation Monitor	1	At All Times	N.A.[b]	73
3. Reactor Enclosure Cooling Water Radiation Monitor	1	At All Times	≤ 3 x Background[b]	72

TABLE 3.3.7.1-1 (Continued)

RADIATION MONITORING INSTRUMENTATION

TABLE NOTATIONS

*When RECENTLY IRRADIATED FUEL is being handled in the secondary containment or during operations with a potential for draining the reactor vessel with the vessel head removed and fuel in the vessel.

(a) With fuel in the spent fuel storage pool.

(b) Alarm only.

ACTION STATEMENTS

ACTION 70 - With one monitor inoperable, restore the inoperable monitor to the OPERABLE status within 7 days or, within the next 6 hours, initiate and maintain operation of the control room emergency filtration system in the radiation isolation mode of operation.

With two or more of the monitors inoperable, within one hour, initiate and maintain operation of the control room emergency filtration system in the radiation mode of operation.

ACTION 71 - With one of the required monitor inoperable, assure a portable continuous monitor with the same alarm setpoint is OPERABLE in the vicinity of the installed monitor during any fuel movement. If no fuel movement is being made, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

ACTION 72 - With the required monitor inoperable, obtain and analyze at least one grab sample of the monitored parameter at least once per 24 hours.

ACTION 73 - With the required monitor inoperable, assure a portable alarming monitor is OPERABLE in the vicinity of the installed monitor or perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

TABLE 4.3.7.1-1

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTATION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE IS REQUIRED
1. Main Control Room Normal Fresh Air Supply Radiation Monitor	S	Q	R	1, 2, 3, and *
2. Area Monitors				
a. Criticality Monitors				
1) Spent Fuel Storage Pool	S	M	R	(a)
b. Control Room Direct Radiation Monitor	S	M	R	At All Times
3. Reactor Enclosure Cooling Water Radiation Monitor	S	M	R(b)	At All Times

TABLE 4.3.7.1-1 (Continued)

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATIONS

*When RECENTLY IRRADIATED FUEL is being handled in the secondary containment or during operations with a potential for draining the reactor vessel with the vessel head removed and fuel in the vessel.

(a) With fuel in the spent fuel storage pool.

(b) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards (NBS) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

3/4.4.7 MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.4.7 Two main steam line isolation valves (MSIVs) per main steam line shall be OPERABLE with closing times greater than or equal to 3 and less than or equal to 10 seconds.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one or more MSIVs inoperable:

   1. Maintain at least one MSIV OPERABLE in each affected main steam line that is open and within 8 hours, either:

      a) Restore the inoperable valve(s) to OPERABLE status, or

      b) Isolate the affected main steam line by use of a deactivated MSIV in the closed position.

   2. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.7 Each of the above required MSIVs shall be demonstrated OPERABLE by verifying full closure between 3 and 10 seconds when tested pursuant to Specification 4.0.5.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

b. The combined leakage rate to be in accordance with the Primary Containment Leakage Rate Testing Program for all primary containment penetrations and all primary containment isolation valves that are subject to Type B and C tests, except for: main steam line isolation valves*, valves which are hydrostatically tested, and those valves where an exemption to Appendix J of 10 CFR 50 has been granted, and

c. The leakage rate to ≤100 scf per hour for any main steam isolation valve that exceeds 100 scf per hour, and restore the combined maximum pathway leakage to ≤200 scf per hour, and

d. The combined leakage rate for all containment isolation valves in hydrostatically tested lines which penetrate the primary containment to less than or equal to 1 gpm times the total number of such valves,

prior to increasing reactor coolant system temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The primary containment leakage rates shall be demonstrated to be in accordance with the Primary Containment Leakage Rate Testing Program, or approved exemptions, for the following:

a. Type A Test

b. Type B and C Tests (including air locks)

c. Main Steam Line Isolation Valves

d. Hydrostatically tested Containment Isolation Valves

*Exemption to Appendix "J" to 10 CFR Part 50.

3/4.6.5 SECONDARY CONTAINMENT

REFUELING AREA SECONDARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.5.1.2 REFUELING AREA SECONDARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: When RECENTLY IRRADIATED FUEL is being handled in the secondary containment, or during operations with a potential for draining the reactor vessel, with the vessel head removed and fuel in the vessel.

ACTION:

Without REFUELING AREA SECONDARY CONTAINMENT INTEGRITY, suspend handling of RECENTLY IRRADIATED FUEL in the secondary containment, and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.1.2 REFUELING AREA SECONDARY CONTAINMENT INTEGRITY shall be demonstrated by:

a. Verifying at least once per 24 hours that the pressure within the refueling area secondary containment is greater than or equal to 0.25 inch of vacuum water gauge.

b. Verifying at least once per 31 days that:

   1. All refueling area secondary containment equipment hatches and blowout panels are closed and sealed.

   2. At least one door in each access to the refueling area secondary containment is closed.

   3. All refueling area secondary containment penetrations not capable of being closed by OPERABLE secondary containment automatic isolation dampers/valves and required to be closed during accident conditions are closed by valves, blind flanges, slide gate dampers or deactivated automatic dampers/valves secured in position.

c. At least once per 24 months:

   Operating one standby gas treatment subsystem for one hour and maintaining greater than or equal to 0.25 inch of vacuum water gauge in the refueling area secondary containment at a flow rate not exceeding 764 cfm.

REFUELING AREA SECONDARY CONTAINMENT AUTOMATIC ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.5.2.2 The refueling area secondary containment ventilation system automatic isolation valves shall be OPERABLE.

APPLICABILITY: When RECENTLY IRRADIATED FUEL is being handled in the secondary containment, or during operations with a potential for draining the reactor vessel, with the vessel head removed and fuel in the vessel.

ACTION:

With one or more of the refueling area secondary containment ventilation system automatic isolation valves inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and within 8 hours either:

a. Restore the inoperable valves to OPERABLE status, or

b. Isolate each affected penetration by use of at least one deactivated valve secured in the isolation position, or

c. Isolate each affected penetration by use of at least one closed manual valve, blind flange or slide gate damper.

Otherwise, suspend handling of RECENTLY IRRADIATED FUEL in the refueling area secondary containment, and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.2.2 Each refueling area secondary containment ventilation system automatic isolation valve shall be demonstrated OPERABLE:

a. Prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by cycling the valve through at least one complete cycle of full travel and verifying the specified isolation time.

b. At least once per 24 months by verifying that on a containment isolation test signal each isolation valve actuates to its isolation position.

c. By verifying the isolation time to be within its limit at least once per 92 days.

STANDBY GAS TREATMENT SYSTEM - COMMON SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.5.3 Two independent standby gas treatment subsystems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, and when RECENTLY IRRADIATED FUEL is being handled in the secondary containment, or during operations with a potential for draining the reactor vessel, with the vessel head removed and fuel in the vessel.

ACTION:

a. In OPERATIONAL CONDITION 1, 2, or 3:

1. With the Unit 1 diesel generator for one standby gas treatment subsystem inoperable for more than 30 days, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. The provisions of Specification 3.0.4 are not applicable.

2. With one standby gas treatment subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3. With one standby gas treatment subsystem inoperable and the other standby gas treatment subsystem with an inoperable Unit 1 diesel generator, restore the inoperable subsystem to OPERABLE status or restore the inoperable Unit 1 diesel generator to OPERABLE status within 72 hours, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4. With the Unit 1 diesel generators for both standby gas treatment system subsystems inoperable for more than 72 hours, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. When RECENTLY IRRADIATED FUEL is being handled in the secondary containment, or during operations with a potential for draining the reactor vessel, with the vessel head removed and fuel in the vessel:

1. With one standby gas treatment subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days, or suspend handling of RECENTLY IRRADIATED FUEL in the secondary containment, and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

2. With both standby gas treatment subsystems inoperable, suspend handling of RECENTLY IRRADIATED FUEL in the secondary containment, and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 24* months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the subsystem by:

1. Verifying that the subsystem satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 5764 cfm ± 10%.

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration of less than 1.25% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F), at a relative humidity of 70% and at a face velocity of 66 fpm.

3. Verify that when the fan is running the subsystem flowrate is 2800 cfm minimum from each reactor enclosure (Zones I and II) and 2200 cfm minimum from the refueling area (Zone III) when tested in accordance with ANSI N510-1980.

4. Verify that the pressure drop across the refueling area to SGTS prefilter is less than 0.25 inches water gage while operating at a flow rate of 2400 cfm ± 10%.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration of less than 1.25% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F), at a relative humidity of 70% and at a face velocity of 66 fpm.

d. At least once per 24 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 9.1 inches water gauge while operating the filter train at a flow rate of 8400 cfm ± 10%.

*Surveillance interval is an exception to the guidance provided in Regulatory Guide 1.52, Revision 2, March 1978.

REACTOR ENCLOSURE RECIRCULATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.5.4 Two independent reactor enclosure recirculation subsystems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one reactor enclosure recirculation subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With both reactor enclosure recirculation subsystems inoperable, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.4 Each reactor enclosure recirculation subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates properly (flow at a minimum of 30,000 cfm).

b. At least once per 24* months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the subsystem by:

   1. Verifying that the subsystem satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, at rated flow (60,000 cfm ± 10%).

   2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration of less than 15% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70%.

   3. Verifying a subsystem flow rate within a range of 30,000 cfm to 66,000 cfm during system operation when tested in accordance with ANSI N510-1980.

*Surveillance interval is an exception to the guidance provided in Regulatory Guide 1.52 Revision 2, March 1978.

SURVEILLANCE REQUIREMENTS (Continued)

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration of less than 15% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) an a relative humidity of 70%.

d. At least once per 24 months by:

1. Verifying that the pressure drop across the combined prefilter, upstream and downstream HEPA filters, and charcoal adsorber banks is less than 6 inches water gauge while operating the filter train at rated flow (60,000 cfm ± 10%), verifying that the prefilter pressure drop is less than 0.8 inch water gauge and that the pressure drop across each HEPA is less than 2 inches water gauge.

2. Verifying that the filter train starts and the isolation valves which take suction on and return to the reactor enclosure open on each of the following test signals:

a. Manual initiation from the control room, and

b. Simulated automatic initiation signal.

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter bank satisfies the inplace penetration and leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 while operating the system at rated flow (60,000 cfm ± 10%).

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorber bank satisfies the inplace penetration and leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at rated flow (60,000 cfm ± 10%).

3.7.1.2 At least the following independent emergency service water system loops, with each loop comprised of:

a. Two OPERABLE emergency service water pumps, and

b. An OPERABLE flow path capable of taking suction from the emergency service water pumps wet pits which are supplied from the spray pond or the cooling tower basin and transferring the water to the associated Unit 2 and common safety-related equipment,

shall be OPERABLE:

a. Two loops, in OPERATIONAL CONDITIONS 1, 2, and 3

b. One Loop, in OPERATIONAL CONDITIONS 4, 5, and when RECENTLY IRRADIATED FUEL is being handled in the secondary containment, or during operations with a potential for draining the reactor vessel

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, 5, and when RECENTLY IRRADIATED FUEL is being handled in the secondary containment, or during operations with a potential for draining the reactor vessel.

ACTION:

a. In OPERATION CONDITION 1, 2, or 3:

1. With one emergency service water pump inoperable, restore the inoperable pump to OPERABLE status within 45 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With one emergency service water pump in each loop inoperable, restore at least one inoperable pump to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3. With one emergency service water system loop otherwise inoperable, declare all equipment aligned to the inoperable loop inoperable**, restore the inoperable loop to OPERABLE status with at least one OPERABLE pump within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

**The diesel generators may be aligned to the OPERABLE emergency service water system loop provided confirmatory flow testing has been performed. Those diesel generators not aligned to the OPERABLE emergency service water system loop shall be declared inoperable and the actions of 3.8.1.1 taken.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

4. With three ESW pump/diesel generator pairs** inoperable, restore at least one inoperable ESW pump/diesel generator pair** to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

5. With four ESW pump/diesel generator pairs** inoperable, restore at least one inoperable ESW pump/diesel generator pair** to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 4 or 5:

1. With only one emergency service water pump and its associated flow path OPERABLE, restore at least two pumps with at least one flow path to OPERABLE status within 72 hours or declare the associated safety related equipment inoperable and take the ACTION required by Specifications 3.5.2 and 3.8.1.2.

c. When RECENTLY IRRADIATED FUEL is being handled in the secondary containment, or during operations with a potential for draining the reactor vessel:

1. With only one emergency service water pump and its associated flow path OPERABLE, restore at least two pumps with at least one flow path to OPERABLE status within 72 hours or verify adequate cooling remains available for the diesel generators required to be OPERABLE or declare the associated diesel generator(s) inoperable and take the ACTION required by Specification 3.8.1.2. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENT

4.7.1.2 At least the above required emergency service water system loop(s) shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 24 months by verifying that:

1. Each automatic valve actuates to its correct position on its appropriate ESW pump start signal.

2. Each pump starts automatically when its associated diesel generator starts.

** An ESW pump/diesel generator pair consists of an ESW pump and its associated diesel generator. If either an ESW pump or its associated diesel generator becomes inoperable, than the ESW pump/diesel generator pair is inoperable.

ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The spray pond shall be OPERABLE with:

a. A minimum pond water level at or above elevation 250'-10" Mean Sea Level, and

b. A pond water temperature of less than or equal to 88°F.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, 5, and when RECENTLY IRRADIATED FUEL is being handled in the secondary containment, or during operations with a potential for draining the reactor vessel.

ACTION:

With the requirements of the above specification not satisfied:

a. In OPERATIONAL CONDITION 1, 2, or 3, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. In OPERATIONAL CONDITION 4 or 5, declare the RHRSW system and the emergency service water system inoperable and take the ACTION required by Specifications 3.7.1.1 and 3.7.1.2.

c. When RECENTLY IRRADIATED FUEL is being handled in the secondary containment, or during operations with a potential for draining the reactor vessel, declare the emergency service water system inoperable and take the ACTION required by Specification 3.7.1.2. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.3 The spray pond shall be determined OPERABLE:

a. By verifying the pond water level to be greater than its limit at least once per 24 hours.

b. By verifying the water surface temperature (within the upper two feet of the surface) to be less than or equal to 88°F:

1. at least once per 4 hours when the spray pond temperature is greater than or equal to 80°F; and

2. at least once per 2 hours when the spray pond temperature is greater than or equal to 85°F; and

3. at least once per 24 hours when the spray pond temperature is greater than 32°F.

c. By verifying all piping above the frost line is drained:

1. within one (1) hour after being used when ambient air temperature is below 40°F; or

2. when ambient air temperature falls below 40°F if the piping has not been previously drained.

3/4.7.2 CONTROL ROOM EMERGENCY FRESH AIR SUPPLY SYSTEM - COMMON SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.2 Two independent control room emergency fresh air supply system subsystems shall be OPERABLE.

APPLICABILITY: All OPERATIONAL CONDITIONS and when RECENTLY IRRADIATED FUEL is being handled in the secondary containment, or during operations with a potential for draining the reactor vessel.

ACTION:

a. In OPERATIONAL CONDITION 1, 2, or 3:

1. With the Unit 1 diesel generator for one control room emergency fresh air supply subsystem inoperable for more than 30 days, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. The provisions of Specification 3.0.4 are not applicable.

2. With one control room emergency fresh air supply subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3. With one control room emergency fresh air supply subsystem inoperable and the other control room emergency fresh air supply subsystem with an inoperable Unit 1 diesel generator, restore the inoperable subsystem to OPERABLE status or restore the Unit 1 diesel generator to OPERABLE status within 72 hours, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4. With the Unit 1 diesel generator and both control room emergency fresh air supply subsystems inoperable for more than 72 hours, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 4, 5 or when RECENTLY IRRADIATED FUEL is being handled in the secondary containment, or during operations with a potential for draining the reactor vessel:

1. With one control room emergency fresh air supply subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days, or initiate and maintain operation of the OPERABLE subsystem in the radiation isolation mode of operation.

2. With both both control room emergency fresh air supply subsystem inoperable, suspend handling of RECENTLY IRRADIATED FUEL in the secondary containment and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration of less than 10% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70%.

3. Verifying a subsystem flow rate of 3000 cfm ± 10% during subsystem operation when tested in accordance with ANSI N510-1980.

d. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration of less than 10% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70%.

e. At least once per 24 months by:

1. Verifying that the pressure drop across the combined prefilter, upstream and downstream HEPA filters, and charcoal adsorber banks is less than 6 inches water gauge while operating the subsystem at a flow rate of 3000 cfm ± 10%; verifying that the prefilter pressure drop is less than 0.8 inch water gauge and that the pressure drop across each HEPA is less than 2 inches water gauge.

2. Verifying that on each of the below chlorine isolation mode actuation test signals, the subsystem automatically switches to the chlorine isolation mode of operation and the isolation valves close within 5 seconds:

a) Outside air intake high chlorine, and

b) Manual initiation from the control room.

3. Verifying that on manual initiation from the control room, the subsystem switches to the radiation isolation mode of operation and the control room is maintained at a positive pressure of at least 1/8 inch water gauge relative to the turbine enclosure and auxiliary equipment room and outside atmosphere during subsystem operation with an outdoor air flow rate less than or equal to 525 cfm.

A.C. SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. Two diesel generators each with:

1. A day fuel tank containing a minimum of 200 gallons of fuel.

2. A fuel storage system containing a minimum of 33,500 gallons of fuel.

3. A fuel transfer pump.

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5, and when RECENTLY IRRADIATED FUEL is being handled in the secondary containment, or during operations with a potential for draining the reactor vessel.

ACTION:

a. With less than the above required A.C. electrical power sources OPERABLE, suspend CORE ALTERATIONS, handling of RECENTLY IRRADIATED FUEL in the secondary containment, operations with a potential for draining the reactor vessel and crane operations over the spent fuel storage pool when fuel assemblies are stored therein. In addition, when in OPERATIONAL CONDITION 5 with the water level less than 22 feet above the reactor pressure vessel flange, immediately initiate corrective action to restore the required power sources to OPERABLE status as soon as practical.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.1.2 At least the above required A.C. electrical power sources shall be demonstrated OPERABLE per Surveillance Requirements 4.8.1.1.1, 4.8.1.1.2, and 4.8.1.1.3.

D.C. SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, two of the following four divisions of the D.C. electrical power sources system shall be OPERABLE with:

a. Division 1, Consisting of:
   1. 125-Volt Battery 2A1 (2A1D101).
   2. 125-Volt Battery 2A2 (2A2D101).
   3. 125-Volt Battery Charger 2BCA1 (2A1D103).
   4. 125-Volt Battery Charger 2BCA2 (2A2D103).

b. Division 2, Consisting of:
   1. 125-Volt Battery 2B1 (2B1D101).
   2. 125-Volt Battery 2B2 (2B2D101).
   3. 125-Volt Battery Charger 2BCB1 (2B1D103).
   4. 125-Volt Battery Charger 2BCB2 (2B2D103).

c. Division 3, Consisting of:
   1. 125-Volt Battery 2C (2CD101).
   2. 125-Volt Battery Charger 2BCC (2CD103).

d. Division 4, Consisting of:
   1. 125-Volt Battery 2D (2DD101).
   2. 125-Volt Battery Charger 2BCD (2DD103).

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5, and when RECENTLY IRRADIATED FUEL is being handled in the secondary containment, or during operations with a potential for draining the reactor vessel.

ACTION:

a. With one or two required battery chargers on one required division inoperable:

   1. Restore battery terminal voltage to greater than or equal to the minimum established float voltage within 2 hours,

   2. Verify associated Division 1 or 2 float current $\leq$ 2 amps, or Division 3 or 4 float current $\leq$ 1 amp within 18 hours and once per 12 hours thereafter, and

   3. Restore battery charger(s) to OPERABLE status within 7 days.

b. With one or more required batteries inoperable due to:

   1. One or two batteries on one division with one or more battery cells float voltage < 2.07 volts, perform 4.8.2.1.a.1 and 4.8.2.1.a.2 within 2 hours for affected battery(s) and restore affected cell(s) voltage $\geq$ 2.07 volts within 24 hours.

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

2. Division 1 or 2 with float current > 2 amps, or with Division 3 or 4 with float current > 1 amp, perform 4.8.2.1.a.2 within 2 hours for affected battery(s) and restore battery float current to within limits within 18 hours.

3. One or two batteries on one division with one or more cells electrolyte level less than minimum established design limits, if electrolyte level was below the top of the plates restore electrolyte level to above top of plates within 8 hours and verify no evidence of leakage(*) within 12 hours. In all cases, restore electrolyte level to greater than or equal to minimum established design limits within 31 days.

4. One or two batteries on one division with pilot cell electrolyte temperature less than minimum established design limits, restore battery pilot cell temperature to greater than or equal to minimum established design limits within 12 hours.

5. Batteries in more than one division affected, restore battery parameters for all batteries in one division to within limits within 2 hours.

6. (i) Any battery having both (Action b.1) one or more battery cells float voltage < 2.07 volts and (Action b.2) float current not within limits, and/or

   (ii) Any battery not meeting any Action b.1 through b.5,

   Restore the battery parameters to within limits within 2 hours.

c. 1. With the requirements of Action a. and/or Action b. not met, or

   2. With less than two divisions of the above required D.C. electrical power sources OPERABLE for reasons other than Actions a. and/or b.,

   Suspend CORE ALTERATIONS, handling of RECENTLY IRRADIATED FUEL in the secondary containment and operations with a potential for draining the reactor vessel.

d. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.2.2 At least the above required batteries and chargers shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.1.

(*) Contrary to the provisions of Specification 3.0.2, if electrolyte level was below the top of the plates, the verification that there is no evidence of leakage is required to be completed regardless of when electrolyte level is restored.

LIMITING CONDITION FOR OPERATION (Continued)

	c) 125-V DC Distribution Panels:	2PPA1	(2AD102)
		2PPA2	(2AD501)
		2PPA3	(2AD162)
2.	Unit 2 Division 2, Consisting of:		
	a) 250-V DC Fuse Box:	2FB	(2BD105)
	b) 250-V DC Motor Control Centers:	2DB-1	(20D202)
		2DB-2	(20D203)
	c) 125-V DC Distribution Panels:	2PPB1	(2BD102)
		2PPB2	(2BD501)
		2PPB3	(2BD162)
3.	Unit 2 Division 3, Consisting of:		
	a) 125-V DC Fuse Box:	2FC	(2CD105)
	b) 125-V DC Distribution Panels:	2PPC1	(2CD102)
		2PPC2	(2CD501)
		2PPC3	(2CD162)
4.	Unit 2 Division 4, Consisting of:		
	a) 125-V DC Fuse Box:	2FD	(2DD105)
	b) 125-V DC Distribution Panels:	2PPD1	(2DD102)
		2PPD2	(2DD501)
		2PPD3	(2DD162)
5.	Unit 1 and Common Division 1, Consisting of:		
	a) 250-V DC Fuse Box:	1FA	(1AD105)
	b) 125-V DC Distribution Panels:	1PPA1	(1AD102)
		1PPA2	(1AD501)
6.	Unit 1 and Common Division 2, Consisting of:		
	a) 250-V DC Fuse Box:	1FB	(1BD105)
	b) 125-V DC Distribution Panels:	1PPB1	(1BD102)
		1PPB2	(1BD501)
7.	Unit 1 and Common Division 3, Consisting of:		
	a) 125-V DC Fuse Box:	1FC	(1CD105)
	b) 125-V DC Distribution Panels:	1PPC1	(1CD102)
		1PPC2	(1CD501)
8.	Unit 1 and Common Division 4, Consisting of:		
	a) 125-V DC Fuse Box:	1FD	(1DD105)
	b) 125-V DC Distribution Panels:	1PPD1	(1DD102)
		1PPD2	(1DD501)

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5, and when RECENTLY IRRADIATED FUEL is being handled in the secondary containment, or during operations with a potential for draining the reactor vessel.

ACTION:

a. With less than two divisions of the above required Unit 2 A.C. distribution systems energized, suspend CORE ALTERATIONS, handling of RECENTLY IRRADIATED FUEL in the secondary containment and operations with a potential for draining the reactor vessel.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

b. With less than two divisions of the above required Unit 2 D.C. distribution systems energized, suspend CORE ALTERATIONS, handling of RECENTLY IRRADIATED FUEL in the secondary containment and operations with a potential for draining the reactor vessel.

c. With any of the above required Unit 1 and common AC and/or DC distribution system divisions not energized, declare the associated common equipment inoperable, and take the appropriate ACTION for that system.

d. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.3.2 At least the above required power distribution system divisions shall be determined energized at least once per 7 days by verifying correct breaker alignment and voltage on the busses/MCCs/panels.

ATTACHMENT 5

LIMERICK GENERATING STATION
UNITS 1 AND 2

Docket Nos. 50-352 & 50-353

License Nos. NPF-39 & NPF-85

License Amendment Request
"LGS Alternative Source Term Implementation"

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3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

The standby liquid control system provides a backup capability for bringing the reactor from full power to a cold, Xenon-free shutdown, assuming that the withdrawn control rods remain fixed in the rated power pattern. To meet this objective it is necessary to inject a quantity of boron which produces a concentration of 660 ppm in the reactor core and other piping systems connected to the reactor vessel. To allow for potential leakage and improper mixing, this concentration is increased by 25%. The required concentration is achieved by having available a minimum quantity of 3,160 gallons of sodium pentaborate solution containng a minimum of 3,754 lbs of sodium pentaborate having the requisite Boron-10 atom % enrichment of 29% as determined from Reference 5. This quantity of solution is a net amount which is above the pump suction shutoff level setpoint thus allowing for the portion which cannot be injected.

The above quantities calculated at 29% Boron-10 enrichment have been demonstrated by analysis to provide a Boron-10 weight equivalent of 185 lbs in the sodium pentaborate solution. Maintaining this Boron-10 weight in the net tank contents ensures a sufficient quantity of boron to bring the reactor to a cold, Xenon-free shutdown.

The pumping rate of 41.2 gpm provides a negative reactivity insertion rate over the permissible solution volume range, which adequately compensates for the positive reactivity effects due to elimination of steam voids, increased water density from hot to cold, reduced doppler effect in uranium, reduced neutron leakage from boiling to cold, decreased control rod worth as the moderator cools, and xenon decay. The temperature requirement ensures that the sodium pentaborate always remains in solution.

With redundant pumps and explosive injection valves and with a highly reliable control rod scram system, operation of the reactor is permitted to continue for short periods of time with the system inoperable or for longer periods of time with one of the redundant components inoperable.

The SLCS system consists of three separate and independent pumps and explosive valves. Two of the separate and independent pumps and explosive valves are required to meet the minimum requirements of this technical specification and, where applicable, satisfy the single failure criterion.

The SLCS must have an equivalent control capacity of 86 gpm of 13% weight sodium pentaborate in order to satisfy 10 CFR 50.62 (Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants). As part of the ARTS/MELLL program the ATWS analysis was updated to reflect the new rod line. As a result of this it was determined that the Boron 10 enrichment was required to be increased to 29% to prevent exceeding a suppression pool temperature of 190°F. This equivalency requirement is fulfilled by having a system which satisfies the equation given in 4.1.5.b.2.

The upper limit concentration of 13.8% has been established as a reasonable limit to prevent precipitation of sodium pentaborate in the event of a loss of tank heating, which allow the solution to cool.

STANDBY LIQUID CONTROL SYSTEM (Continued)

Surveillance requirements are established on a frequency that assures a high reliability of the system. Once the solution is established, boron concentration will not vary unless more boron or water is added, thus a check on the temperature and volume once each 24 hours assures that the solution is available for use.

Replacement of the explosive charges in the valves at regular intervals will assure that these valves will not fail because of deterioration of the charges.

The Standby Liquid Control System also has a post-DBA LOCA safety function to buffer Suppression Pool pH in order to maintain bulk pH above 7.0. The buffering of Suppression Pool pH is necessary to prevent iodine re-evolution to satisfy the methodology for Alternative Source Term. Manual initiation is used, and the minimum amount of total boron required for Suppression Pool pH buffering is 240 lbs. Given that at least 185 lbs of Boron-10 is maintained in the tank, the total boron in the tank will be greater than 240 lbs for the range of enrichments from 29% to 62%.

ACTION Statement (a) applies only to OPERATIONAL CONDITIONS 1 and 2 because a single pump can satisfy both the reactor control function and the post-DBA LOCA function to control Suppression Pool pH since boron injection is not required until 13 hours post-LOCA. ACTION Statement (b) applies to OPERATIONAL CONDITIONS 1, 2 and 3 to address the post-LOCA safety function of the SLC system.

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3. J. M. Haun, C. J. Paone, and R. C. Stirn, Addendum 2, "Exposed Cores." Supplement 2 to NEDO-10527, January 1973.

4. Amendment 17 to General Electric Licensing Topical Report NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel".

5. "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Limerick Generating Station Units 1 and 2," NEDC-32193P, Revision 2, October 1993.

3/4.4.7 MAIN STEAM LINE ISOLATION VALVES

Double isolation valves are provided on each of the main steam lines to minimize the potential leakage paths from the containment in case of a line break. Only one valve in each line is required to maintain the integrity of the containment, however, single failure considerations require that two valves be OPERABLE. The surveillance requirements are based on the operating history of this type valve. The maximum closure time has been selected to contain fission products and to prevent core damage following line breaks. The minimum closure time is consistent with the assumptions in the safety analyses to prevent pressure surges.

3/4.4.8 STRUCTURAL INTEGRITY

The inspection programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant.

Components of the reactor coolant system were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code 1971 Edition and Addenda through Winter 1972.

The inservice inspection program for ASME Code Class 1, 2, and 3 components will be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a. Additionally, the Inservice Inspection Program conforms to the NRC staff positions identified in NRC Generic Letter 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping," as approved in NRC Safety Evaluations dated March 9, 1990 and October 22, 1990.

3/4.4.9 RESIDUAL HEAT REMOVAL

The RHR system is required to remove decay heat and sensible heat in order to maintain the temperature of the reactor coolant. RHR shutdown cooling is comprised of four (4) subsystems which make two (2) loops. Each loop consists of two (2) motor driven pumps, a heat exchanger, and associated piping and valves. Both loops have a common suction from the same recirculation loop. Two (2) redundant, manually controlled shutdown cooling subsystems of the RHR System can provide the required decay heat removal capability. Each pump discharges the reactor coolant, after it has been cooled by circulation through the respective heat exchangers, to the reactor via the associated recirculation loop or to the reactor via the low pressure coolant injection pathway. The RHR heat exchangers transfer heat to the RHR Service Water System. The RHR shutdown cooling mode is manually controlled.

An OPERABLE RHR shutdown cooling subsystem consists of an RHR pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path. In HOT SHUTDOWN condition, the requirement to maintain OPERABLE two (2) independent RHR shutdown cooling subsystems means that each subsystem considered OPERABLE must be associated with a different heat exhanger loop, i.e., the "A" RHR heat exchanger with the "A" RHR pump or the "C" RHR pump, <u>and</u> the "B" RHR heat exchanger with the "B" RHR pump or the "D" RHR pump are two (2) independent RHR shutdown cooling subsystems. Only one (1) of the two (2) RHR pumps associated with each RHR heat exchanger loop is

3/4.6.5 SECONDARY CONTAINMENT

Secondary containment is designed to minimize any ground level release of radioactive material which may result from an accident. The Reactor Enclosure and associated structures provide secondary containment during normal operation when the drywell is sealed and in service. At other times the drywell may be open and, when required, secondary containment integrity is specified.

Establishing and maintaining a vacuum in the reactor enclosure secondary containment with the standby gas treatment system once per 24 months, along with the surveillance of the doors, hatches, dampers and valves, is adequate to ensure that there are no violations of the integrity of the secondary containment.

The OPERABILITY of the reactor enclosure recirculation system and the standby gas treatment systems ensures that sufficient iodine removal capability will be available in the event of a LOCA or refueling accident involving RECENTLY IRRADIATED FUEL (SGTS only). The reduction in containment iodine inventory reduces the resulting SITE BOUNDARY radiation doses associated with containment leakage. The operation of this system and resultant iodine removal capacity are consistent with the assumptions used in the LOCA and refueling accident analyses. Provisions have been made to continuously purge the filter plenums with instrument air when the filters are not in use to prevent buildup of moisture on the adsorbers and the HEPA filters.

Although the safety analyses assumes that the reactor enclosure secondary containment draw down time will take 930 seconds, these surveillance requirements specify a draw down time of 916 seconds. This 14 second difference is due to the diesel generator starting and sequence loading delays which is not part of this surveillance requirement.

The reactor enclosure secondary containment draw down time analyses assumes a starting point of 0.25 inch of vacuum water gauge and worst case SGTS dirty filter flow rate of 2800 cfm. The surveillance requirements satisfy this assumption by starting the drawdown from ambient conditions and connecting the adjacent reactor enclosure and refueling area to the SGTS to split the exhaust flow between the three zones and verifying a minimum flow rate of 2800 cfm from the test zone. This simulates the worst case flow alignment and verifies adequate flow is available to drawdown the test zone within the required time. The Technical Specification Surveillance Requirement 4.6.5.3.b.3 is intended to be a multi-zone air balance verification without isolating any test zone.

Based on implementation of the AST methodology, it has been demonstrated through analysis that operating the subsystem with a minimum flow rate of 30,000 cfm satisfies the dose requirements. However, based on the applicable testing standards and guidance provided in Regulatory Guide 1.52, Revision 2, and ANSI N510, 1980, the acceptance criteria must be satisfied based on the system rated flow, which provides the most conservative results. Satisfying the acceptance criteria at rated flow also demonstrates system operability at lower measured flows because the residence time is longer. Therefore, verification of subsystem flow rates between 30,000 cfm and the maximum rated flow of 66,000 cfm (includes a 10% factor) satisfies the surveillance requirements for the HEPA filters and charcoal adsorber housings based on 70% efficiencies.

The SGTS fans are sized for three zones and therefore, when aligned to a single zone or two zones, will have excess capacity to more quickly drawdown the affected zones. There is no maximum flow limit to individual zones or pairs of zones and the air balance and drawdown time are verified when all three zones are connected to the SGTS.

The three zone air balance verification and drawdown test will be done after any major system alteration, which is any modification which will have an effect on the SGTS flowrate such that the ability of the SGTS to drawdown the reactor enclosure to greater than or equal to 0.25 inch of vacuum water gage in less than or equal to 916 seconds could be affected.

3/4.7.2 CONTROL ROOM EMERGENCY FRESH AIR SUPPLY SYSTEM - COMMON SYSTEM

The OPERABILITY of the control room emergency fresh air supply system ensures that the control room will remain habitable for operations personnel during and following all design basis accident conditions. Constant purge of the system at 1 cfm is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less Total Effective Dose Equivalent. This limitation is consistent with the requirements of 10 CFR Part 50.67, Accident Source Terms.

Since the Control Room Emergency Fresh Air Supply System is not credited for filtration in OPERATIONAL CONDITIONS 4 and 5, applicability to 4 and 5 is only required to support the Chlorine and Toxic Gas design basis isolation requirements.

Additionally, based on implementation of the Alternative Source Term methodology, it has been demonstrated through analysis that manual initiation of the CREFAS radiation mode within 30 minutes of the start of gap release for the limiting design basis LOCA is sufficient to assure that control room operator dose limits in 10 CFR Part 50.67 are met.

3/4.7.3 REACTOR CORE ISOLATION COOLING SYSTEM

The reactor core isolation cooling (RCIC) system is provided to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel without requiring actuation of any of the emergency core cooling system equipment. The RCIC system is conservatively required to be OPERABLE whenever reactor pressure exceeds 150 psig. This pressure is substantially below that for which low pressure core cooling systems can provide adequate core cooling.

The RCIC system specifications are applicable during OPERATIONAL CONDITIONS 1, 2, and 3 when reactor vessel pressure exceeds 150 psig because RCIC is the primary non-ECCS source of emergency core cooling when the reactor is pressurized.

With the RCIC system inoperable, adequate core cooling is assured by the OPERABILITY of the HPCI system and justifies the specified 14 day out-of-service period.

The surveillance requirements provide adequate assurance that RCIC will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage and to start cooling at the earliest possible moment.

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

The standby liquid control system provides a backup capability for bringing the reactor from full power to a cold, Xenon-free shutdown, assuming that the withdrawn control rods remain fixed in the rated power pattern. To meet this objective it is necessary to inject a quantity of boron which produces a concentration of 660 ppm in the reactor core and other piping systems connected to the reactor vessel. To allow for potential leakage and improper mixing, this concentration is increased by 25%. The required concentration is achieved by having available a minimum quantity of 3,160 gallons of sodium pentaborate solution containng a minimum of 3,754 lbs of sodium pentaborate having the requisite Boron-10 atom % enrichment of 29% as determined from Reference 5. This quantity of solution is a net amount which is above the pump suction shutoff level setpoint thus allowing for the portion which cannot be injected.

The above quantities calculated at 29% Boron-10 enrichment have been demonstrated by analysis to provide a Boron-10 weight equivalent of 185 lbs in the sodium pentaborate solution. Maintaining this Boron-10 weight in the net tank contents ensures a sufficient quantity of boron to bring the reactor to a cold, Xenon-free shutdown.

The pumping rate of 41.2 gpm provides a negative reactivity insertion rate over the permissible solution volume range, which adequately compensates for the positive reactivity effects due to elimination of steam voids, increased water density from hot to cold, reduced doppler effect in uranium, reduced neutron leakage from boiling to cold, decreased control rod worth as the moderator cools, and xenon decay. The temperature requirement ensures that the sodium pentaborate always remains in solution.

With redundant pumps and explosive injection valves and with a highly reliable control rod scram system, operation of the reactor is permitted to continue for short periods of time with the system inoperable or for longer periods of time with one of the redundant components inoperable.

The SLCS system consists of three separate and independent pumps and explosive valves. Two of the separate and independent pumps and explosive valves are required to meet the minimum requirements of this technical specification and, where applicable, satisfy the single failure criterion.

The SLCS must have an equivalent control capacity of 86 gpm of 13% weight sodium pentaborate in order to satisfy 10 CFR 50.62 (Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants). As part of the ARTS/MELLL program the ATWS analysis was updated to reflect the new rod line. As a result of this it was determined that the Boron 10 enrichment was required to be increased to 29% to prevent exceeding a suppression pool temperature of 190°F. This equivalency requirement is fulfilled by having a system which satisfies the equation given in 4.1.5.b.2.

The upper limit concentration of 13.8% has been established as a reasonable limit to prevent precipitation of sodium pentaborate in the event of a loss of tank heating, which allow the solution to cool.

STANDBY LIQUID CONTROL SYSTEM (Continued)

Surveillance requirements are established on a frequency that assures a high reliability of the system. Once the solution is established, boron concentration will not vary unless more boron or water is added, thus a check on the temperature and volume once each 24 hours assures that the solution is available for use.

Replacement of the explosive charges in the valves at regular intervals will assure that these valves will not fail because of deterioration of the charges.

The Standby Liquid Control System also has a post-DBA LOCA safety function to buffer Suppression Pool pH in order to maintain bulk pH above 7.0. The buffering of Suppression Pool pH is necessary to prevent iodine re-evolution to satisfy the methodology for Alternative Source Term. Manual initiation is used, and the minimum amount of total boron required for Suppression Pool pH buffering is 240 lbs. Given that at least 185 lbs of Boron-10 is maintained in the tank, the total boron in the tank will be greater than 240 lbs for the range of enrichments from 29% to 62%.

ACTION Statement (a) applies only to OPERATIONAL CONDITIONS 1 and 2 because a single pump can satisfy both the reactor control function and the post-DBA LOCA function to control Suppression Pool pH since boron injection is not required until 13 hours post-LOCA. ACTION Statement (b) applies to OPERATIONAL CONDITIONS 1, 2 and 3 to address the post-LOCA safety function of the SLC system.

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3/4.4.7 MAIN STEAM LINE ISOLATION VALVES

Double isolation valves are provided on each of the main steam lines to minimize the potential leakage paths from the containment in case of a line break. Only one valve in each line is required to maintain the integrity of the containment, however, single failure considerations require that two valves be OPERABLE. The surveillance requirements are based on the operating history of this type valve. The maximum closure time has been selected to contain fission products and to prevent core damage following line breaks. The minimum closure time is consistent with the assumptions in the safety analyses to prevent pressure surges.

3/4.4.8 STRUCTURAL INTEGRITY

The inspection programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant.

Components of the reactor coolant system were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code 1971 Edition and Addenda through winter 1972.

The inservice inspection program for ASME Class 1, 2, and 3 components will be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a. Additionally, the Inservice Inspection Program conforms to the NRC positions identified in NRC Generic Letter 88-01, "NRC Position on IGSCC in Austenitic Stainless Steel Piping," as approved in NRC Safety Evaluations dated March [illegible] 1990 and October 22, 1990.

3/4.4.9 RESIDUAL HEAT REMOVAL

The RHR system is required to remove decay heat and sensible heat in order to maintain the temperature of the reactor coolant. The shutdown cooling is comprised of four (4) subsystems which make two (2) loops. Each loop consists of two (2) motor driven pumps, a heat exchanger, and associated piping and valves. Both loops have a common suction from the same recirculation loop. A [illegible] redundant, manually controlled shutdown cooling subsystems of the RHR system can provide the required decay heat removal capability. Each pump discharges the reactor coolant, after it has been cooled by circulation through the respective heat exchangers, to the reactor via the associated recirculation loop or to the reactor via the low pressure coolant injection pathway. The RHR heat exchangers transfer heat to the RHR Service Water System. The RHR shutdown cooling mode is manually controlled.

An OPERABLE RHR shutdown cooling subsystem consists of an RHR pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path. In HOT SHUTDOWN condition, the requirement to maintain OPERABLE two (2) independent RHR shutdown cooling subsystems means that each subsystem considered OPERABLE must be associated with a different heat exhanger loop, i.e., the "A" RHR heat exchanger with the "A" RHR pump or the "C" RHR pump, <u>and</u> the "B" RHR heat exchanger with the "B" RHR pump or the "D" RHR pump are two (2) independent RHR shutdown cooling subsystems. Only one (1) of the two (2) RHR pumps associated with each RHR heat exchanger loop is

3/4.6.5 SECONDARY CONTAINMENT

Secondary containment is designed to minimize any ground level release of radioactive material which may result from an accident. The Reactor Enclosure and associated structures provide secondary containment during normal operation when the drywell is sealed and in service. At other times the drywell may be open and, when required, secondary containment integrity is specified.

Establishing and maintaining a vacuum in the reactor enclosure secondary containment with the standby gas treatment system once per 24 months, along with the surveillance of the doors, hatches, dampers and valves, is adequate to ensure that there are no violations of the integrity of the secondary containment.

The OPERABILITY of the reactor enclosure recirculation system and the standby gas treatment systems ensures that sufficient iodine removal capability will be available in the event of a LOCA or refueling accident involving RECENTLY IRRADIATED FUEL (SGTS only). The reduction in containment iodine inventory reduces the resulting SITE BOUNDARY radiation doses associated with containment leakage. The operation of this system and resultant iodine removal capacity are consistent with the assumptions used in the LOCA and refueling accident analyses. Provisions have been made to continuously purge the filter plenums with instrument air when the filters are not in use to prevent buildup of moisture on the adsorbers and the HEPA filters.

Although the safety analyses assumes that the reactor enclosure secondary containment draw down time will take 930 seconds, these surveillance requirements specify a draw down time of 916 seconds. This 14 second difference is due to the diesel generator starting and sequence loading delays which is not part of this surveillance requirement.

The reactor enclosure secondary containment draw down time analyses assumes a starting point of 0.25 inch of vacuum water gauge and worst case SGTS dirty filter flow rate of 2800 cfm. The surveillance requirements satisfy this assumption by starting the drawdown from ambient conditions and connecting the adjacent reactor enclosure and refueling area to the SGTS to split the exhaust flow between the three zones and verifying a minimum flow rate of 2800 cfm from the test zone. This simulates the worst case flow alignment and verifies adequate flow is available to drawdown the test zone within the required time. The Technical Specification Surveillance Requirement 4.6.5.3.b.3 is intended to be a multi-zone air balance verification without isolating any test zone.

Based on implementation of the AST methodology, it has been demonstrated through analysis that operating the subsystem with a minimum flow rate of 30,000 cfm satisfies the dose requirements. However, based on the applicable testing standards and guidance provided in Regulatory Guide 1.52, Revision 2, and ANSI N510, 1980, the acceptance criteria must be satisfied based on the system rated flow, which provides the most conservative results. Satisfying the acceptance criteria at rated flow also demonstrates system operability at lower measured flows because the residence time is longer. Therefore, verification of subsystem flow rates between 30,000 cfm and the maximum rated flow of 66,000 cfm (includes a 10% factor) satisfies the surveillance requirements for the HEPA filters and charcoal adsorber housings based on 70% efficiencies.

The SGTS is common to Unit 1 and 2 and consists of two independent subsystems. The power supplies for the common portions of the subsystems are from Unit 1 safeguard busses, therefore the inoperability of these Unit 1 supplies are addressed in the SGTS ACTION statements in order to ensure adequate onsite power sources to SGTS for its Unit 2 function during a loss of offsite power event. The allowable out of service times are consistent with those in the Unit 1 Technical Specifications for SGTS and AC electrical power supply out of service condition combinations.

3/4.7.1 SERVICE WATER SYSTEMS - COMMON SYSTEMS

The OPERABILITY of the service water systems ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of these systems, assuming a single failure, is consistent with the assumptions used in the accident conditions within acceptable limits.

The RHRSW and ESW systems are common to Units 1 and 2 and consist of two independent subsystems each with two pumps. One pump per subsystem (loop) is powered from a Unit 1 safeguard bus and the other pump is powered from a Unit 2 safeguard bus. In order to ensure adequate onsite power sources to the systems during a loss of offsite power event, the inoperability of these supplies are restricted in system ACTION statements.

RHRSW is a manually operated system used for core and containment heat removal. Each of two RHRSW subsystems has one heat exchanger per unit. Each RHRSW pump provides adequate cooling for one RHR heat exchanger. By limiting operation with less than three OPERABLE RHRSW pumps with OPERABLE Diesel Generators, each unit is ensured adequate heat removal capability for the design scenario of LOCA/LOOP on one unit and simultaneous safe shutdown of the other unit.

Each ESW pump provides adequate flow to the cooling loads in its associated loop. With only two divisions of power required for LOCA mitigation of one unit and one division of power required for safe shutdown of the other unit, one ESW pump provides sufficient capacity to fulfill design requirements. ESW pumps are automatically started upon start of the associated Diesel Generators. Therefore, the allowable out of service times for OPERABLE ESW pumps and their associated Diesel Generators is limited to ensure adequate cooling during a loss of offsite power event.

3/4.7.2 CONTROL ROOM EMERGENCY FRESH AIR SUPPLY SYSTEM - COMMON SYSTEM

The OPERABILITY of the control room emergency fresh air supply system ensures that the control room will remain habitable for operations personnel during and following all design basis accident conditions. Constant purge of the system at 1 cfm is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less Total Effective Dose Equivalent. This limitation is consistent with the requirements of 10 CFR Part 50.67, Accident Source Terms.

Since the Control Room Emergency Fresh Air Supply System is not credited for filtration in OPERATIONAL CONDITIONS 4 and 5, applicability to 4 and 5 is only required to support the Chlorine and Toxic Gas design basis isolation requirements.

Additionally, based on implementation of the Alternative Source Term methodology, it has been demonstrated through analysis that manual initiation of the CREFAS radiation mode within 30 minutes of the start of gap release for the limiting design basis LOCA is sufficient to assure that control room operator dose limits in 10 CFR Part 50.67 are met.

The CREFAS is common to Units 1 and 2 and consists of two independent subsystems. The power supplies for the system are from Unit 1 Safeguard busses, therefore, the inoperability of these Unit 1 supplies are addressed in the CREFAS ACTION statements in order to ensure adequate onsite power sources to CREFAS during a loss of offsite power event. The allowable out of service

ATTACHMENT 6

LIMERICK GENERATING STATION
UNITS 1 AND 2

Docket Nos. 50-352 & 50-353

License Nos. NPF-39 & NPF-85

License Amendment Request
"LGS Alternative Source Term Implementation"

List Of Commitments

The following table identifies those actions committed to by Exelon in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

Commitment	Continuing Compliance	Scheduled Completion Date
Per TSTF-51, licensees adding the term "recently" must make the following commitment which is consistent with NUMARC 93-01, Revision 3, Section 11.3.6.5, "Safety Removal for Removal of Equipment from Service During Shutdown Conditions," subheading "Containment – Primary (PWR)/Secondary (BWR)". Exelon makes a commitment to the following NUMARC 93-01 section: *"In addition to the guidance in NUMARC 91-06, for plants which obtain license amendments to utilize shutdown safety administrative controls in lieu of Technical Specification requirements on primary or secondary containment operability or ventilation system operability, during fuel handling or core alterations, the following guidelines should be included in the assessment of systems removed from service:* *-During Fuel Handling/Core Alterations, ventilation system and radiation monitor availability (as defined in NUMARC 91-06) should be assessed, with respect to filtration and monitoring of releases from the fuel. Following shutdown, radioactivity in the RCS decays away fairly rapidly. The basis of the Technical Specification operability amendment is the reduction in doses due to such decay. The goal of maintaining ventilation system and radiation monitoring availability is to reduce doses even further below that provided by the natural decay and to avoid unmonitored releases.* *-A single normal or contingency method to promptly close primary or secondary containment penetrations should be developed. Such prompt methods need not completely block the penetration or be capable of resisting pressure. The purpose of this is to enable ventilation systems to draw the release from a postulated fuel handling accident in the proper direction such that it can be treated and monitored."* Limerick is defining the definition of prompt in this context to mean being accomplished within 1 hour.	X	Upon Implementation.

ATTACHMENT 7

LIMERICK GENERATING STATION
UNITS 2 AND 3

Docket Nos. 50-352
50-353

License Nos. NPF-39
NPF-85

License Amendment Request
"LGS Alternative Source Term Implementation"

Compact Disk Containing LGS Meteorological Data

ATTACHMENT 8

LIMERICK GENERATING STATION
UNITS 2 AND 3

Docket Nos. 50-352
50-353

License Nos. NPF-39
NPF-85

License Amendment Request
"LGS Alternative Source Term Implementation"

Supporting Input Parameters for AST Calculations

ATTACHMENT 8

Technical Parameters for AST Calculations

This attachment serves as a supplement to provide additional technical information needed to fully understand the technical analyses that were performed.

Table of Contents

Equilibrium Two-Year Cycle Isotopic Core Inventory

The LGS core isotopic activities are based on those developed for Peach Bottom.

The Peach Bottom Source Term fuel cycle assumptions are considered bounding relative to Limerick Generating Station. The Peach Bottom batch sizes, batch average burnups, and reload bundle average enrichments are similar to those at Limerick. For example, the bundle average enrichment used in the Peach Bottom analysis was approximately 4.11 wt. % U-235. The expected bundle average enrichment for LGS is approximately 4.16 wt. % U-235. Additionally, the rated thermal power level for the Peach Bottom analyses was 3514 MWt (uprated for Appendix K Thermal Power Optimization). Source terms are developed on a curie/MWt basis, and ultimately multiplied by the maximum rated power, including the applicable instrument uncertainty. The two plants are essentially identical for the accident analysis power.

LGS has a similar core loading as PB-3 (in terms of mtU of uranium). Design cycle lengths for LGS are bounded by the 711 EFPD assumed for the PB-3 source term; shorter cycle lengths would result in reduced activities from long-lived isotopes. Since the batch sizes, enrichments, and burnup distributions are similar between PB-3 and LGS (as noted below), the PB-3 AST source term would be bounding for current LGS cycle designs.

Overall, minor LGS and PB differences such as these in initial enrichment, power levels, loading, and ultimate burnup are not significant in development of source terms from ORIGEN simulations. In general, these source terms are considered representative for two year fuel cycles for plants of this size and are not intended to be adjusted for future reloads with these general characteristics.

The ORIGEN2.1 core inventory activity and composition results for the equilibrium two-year cycle at 100 EFPD (BOC) and EOC are shown below in Table 1. The maximum of the 100 EFPD and EOC values for each isotope are selected to generate the bounding isotopic core inventory activity and composition results as shown in Table 1 below.

Table 1
Bounding Core Isotopic Activities For RADTRAD Input

Isotope	Isotopic Activity at 100 EFPD (Ci)	Isotopic Activity at EOC (Ci)	Bounding Isotopic Activity (Ci)	Bounding Isotopic Activity (Ci/MWt)
KR 83M	1.324E+07	1.158E+07	1.324E+07	3.767E+03
BR 84	2.373E+07	2.002E+07	2.373E+07	6.751E+03
BR 85	2.888E+07	2.404E+07	2.888E+07	8.216E+03
KR 85	8.806E+05	1.387E+06	1.387E+06	3.946E+02
KR 85M	2.922E+07	2.436E+07	2.922E+07	8.313E+03
RB 86	1.118E+05	2.291E+05	2.291E+05	6.518E+01
KR 87	5.739E+07	4.672E+07	5.739E+07	1.633E+04
KR 88	8.096E+07	6.570E+07	8.096E+07	2.303E+04
RB 88	8.197E+07	6.678E+07	8.197E+07	2.332E+04
SR 89	9.836E+07	8.846E+07	9.836E+07	2.798E+04
SR 90	6.982E+06	1.117E+07	1.117E+07	3.178E+03
Y 90	7.142E+06	1.150E+07	1.150E+07	3.272E+03
SR 91	1.336E+08	1.110E+08	1.336E+08	3.801E+04
Y 91	1.212E+08	1.143E+08	1.212E+08	3.448E+04
SR 92	1.412E+08	1.205E+08	1.412E+08	4.017E+04
Y 92	1.416E+08	1.210E+08	1.416E+08	4.029E+04
Y 93	1.591E+08	1.404E+08	1.591E+08	4.526E+04
ZR 95	1.521E+08	1.578E+08	1.578E+08	4.489E+04
NB 95	1.399E+08	1.586E+08	1.586E+08	4.512E+04
ZR 97	1.637E+08	1.578E+08	1.637E+08	4.657E+04
MO 99	1.771E+08	1.785E+08	1.785E+08	5.078E+04
TC 99M	1.551E+08	1.563E+08	1.563E+08	4.447E+04
RU103	1.234E+08	1.477E+08	1.477E+08	4.202E+04
RU105	7.642E+07	1.022E+08	1.022E+08	2.908E+04
RH105	7.310E+07	9.673E+07	9.673E+07	2.752E+04
RU106	3.856E+07	6.081E+07	6.081E+07	1.730E+04
SB127	8.572E+06	1.018E+07	1.018E+07	2.896E+03
TE127	8.402E+06	1.010E+07	1.010E+07	2.873E+03
TE127M	1.039E+06	1.355E+06	1.355E+06	3.855E+02
SB129	2.732E+07	3.036E+07	3.036E+07	8.638E+03
TE129	2.679E+07	2.988E+07	2.988E+07	8.501E+03
TE129M	3.875E+06	4.453E+06	4.453E+06	1.267E+03
I129	2.774E+00	4.816E+00	4.816E+00	1.370E-03
TE131M	1.269E+07	1.360E+07	1.360E+07	3.869E+03
I131	9.139E+07	9.444E+07	9.444E+07	2.687E+04
XE131M	1.015E+06	1.056E+06	1.056E+06	3.004E+02
TE132	1.322E+08	1.343E+08	1.343E+08	3.821E+04
I132	1.338E+08	1.364E+08	1.364E+08	3.881E+04

Table 1
Bounding Core Isotopic Activities For RADTRAD Input
(Continued)

Isotope	Isotopic Activity at 100 EFPD (Ci)	Isotopic Activity at EOC (Ci)	Bounding Isotopic Activity (Ci)	Bounding Isotopic Activity (Ci/MWt)
I133	1.953E+08	1.925E+08	1.953E+08	5.556E+04
XE133	1.904E+08	1.930E+08	1.930E+08	5.491E+04
XE133M	5.956E+06	6.007E+06	6.007E+06	1.709E+03
I134	2.167E+08	2.118E+08	2.167E+08	6.165E+04
CS134	1.335E+07	2.559E+07	2.559E+07	7.280E+03
I135	1.825E+08	1.806E+08	1.825E+08	5.192E+04
XE135	7.832E+07	7.086E+07	7.832E+07	2.228E+04
XE135M	3.636E+07	3.773E+07	3.773E+07	1.073E+04
CS136	3.568E+06	7.123E+06	7.123E+06	2.027E+03
CS137	9.460E+06	1.595E+07	1.595E+07	4.538E+03
BA137M	8.965E+06	1.510E+07	1.510E+07	4.296E+03
XE138	1.679E+08	1.589E+08	1.679E+08	4.777E+04
CS138	1.841E+08	1.760E+08	1.841E+08	5.238E+04
BA139	1.787E+08	1.719E+08	1.787E+08	5.084E+04
BA140	1.721E+08	1.661E+08	1.721E+08	4.896E+04
LA140	1.764E+08	1.722E+08	1.764E+08	5.019E+04
LA141	1.631E+08	1.563E+08	1.631E+08	4.640E+04
CE141	1.579E+08	1.575E+08	1.579E+08	4.492E+04
LA142	1.593E+08	1.510E+08	1.593E+08	4.532E+04
CE143	1.556E+08	1.449E+08	1.556E+08	4.427E+04
PR143	1.509E+08	1.416E+08	1.509E+08	4.293E+04
CE144	1.012E+08	1.264E+08	1.264E+08	3.596E+04
ND147	6.459E+07	6.321E+07	6.459E+07	1.838E+04
NP239	1.616E+09	1.897E+09	1.897E+09	5.397E+05
PU238	2.639E+05	6.312E+05	6.312E+05	1.796E+02
PU239	3.100E+04	4.218E+04	4.218E+04	1.200E+01
PU240	2.871E+04	4.526E+04	4.526E+04	1.288E+01
PU241	1.365E+07	2.173E+07	2.173E+07	6.182E+03
AM241	1.634E+04	3.349E+04	3.349E+04	9.528E+00
CM242	3.645E+06	8.393E+06	8.393E+06	2.388E+03
CM244	2.654E+05	9.147E+05	9.147E+05	2.602E+02

Table 2
Bounding Core Isotopic Concentrations
(Grams by Isotope)

Isotope	Isotopic Concentration (grams)
KR 83M	6.414E-01
BR 84	3.370E-01
BR 85	3.741E-02
KR 85	3.534E+03
KR 85M	3.550E+00
RB 86	2.814E+00
KR 87	2.025E+00
KR 88	6.451E+00
RB 88	6.826E-01
SR 89	3.384E+03
SR 90	8.183E+04
Y 90	2.113E+01
SR 91	3.683E+01
Y 91	4.939E+03
SR 92	1.123E+01
Y 92	1.471E+01
Y 93	4.768E+01
ZR 95	7.341E+03
NB 95	4.054E+03
ZR 97	8.560E+01
MO 99	3.720E+02
TC 99M	2.971E+01
RU103	4.575E+03
RU105	1.519E+01
RH105	1.146E+02
RU106	1.817E+04
SB127	3.811E+01
TE127	3.826E+00
TE127M	1.436E+02
I127	8.040E+03
SB129	5.397E+00
TE129	1.426E+00
TE129M	1.478E+02
I129	2.727E+04
TE131M	1.704E+01
I131	7.615E+02

Isotope	Isotopic Concentration (grams)
XE131M	1.260E+01
TE132	4.423E+02
I132	1.321E+01
I133	1.723E+02
XE133	1.031E+03
XE133M	1.339E+01
CS133	1.678E+05
I134	8.118E+00
CS134	1.977E+04
I135	5.195E+01
XE135	3.065E+01
XE135M	4.140E-01
CS135	7.841E+04
CS136	9.715E+01
CS137	1.832E+05
BA137M	2.807E-02
XE138	1.745E+00
CS138	4.348E+00
BA139	1.092E+01
BA140	2.359E+03
LA140	3.168E+02
LA141	2.883E+01
CE141	5.541E+03
LA142	1.115E+01
CE143	2.342E+02
PR143	2.241E+03
CE144	3.962E+04
ND147	8.039E+02
NP239	8.173E+03
PU238	3.685E+04
PU239	6.782E+05
PU240	1.986E+05
PU241	2.108E+05
AM241	9.755E+03
CM242	2.537E+03
CM244	1.130E+04

Maintaining Operator Doses ALARA in the Vicinity of the ECCS Pipe at the Control Room North Wall

The purpose of this attachment is to reassess gamma shine from sources external to the control room, using alternative source terms to assess contained radioactivity, and in some cases, a more detailed geometry treatment.

The dominant radiation source outside of the control room is a Unit 1 vertical run of 14 inch NPS core spray piping that is located 18 inches from the Reactor Enclosure - Control Room wall. This wall is 36 inch thick concrete. The historically determined dose contribution in the control room from this pipe and other lesser piping contributors is 4.2 rem whole body. The following give some consideration to impacted areas in the control room and the potential for use of local administrative controls:

- The core spray fluid is ECCS (Suppression Pool) water. AST source terms are more favorable because of the reduction in assumed fractions of core iodine released to the suppression pool from 50% to 30%. This is offset only partially by increases in certain other non-halogen, non-noble gas isotopes.
- Doses are evaluated with credit for Control Room occupancy per RG. 1.183 of 1.0 for the first day, 0.6 for the next 3 days, and 0.4 for the following 26 days.
- Dose assessment is performed using the point-kernel method as implemented in MicroShield 5.05.
- Doses are calculated as a function of distance into the Control Room, since doses drop significantly with distance.
- All Control Room equipment in the near vicinity has been identified and evaluated to determine occupancy required and the potential for administrative controls to reduce occupancy in the near vicinity.
    - Cabinet 1AC696 and 1BC696 are for the LOCA H_2 Recombiner Control Panels, and are expected to require minimal operator presence. If used, system startup and periodic monitoring is all that is considered to be required.
    - Cabinet 1AC464 is an ERFDS Multiplexer Enclosure requiring no operator presence. Cabinets further along this wall will have far lower dose rates than shown because of the severe angle through the shield.
    - Cabinets ODZ585 through 10C690 are for loose parts monitoring, meteorological instrumentation, and RDMS and MMDRS displays, printers and terminals. The meteorological stations are used chiefly until the Tech Support Center and (offsite) Emergency Operations Facility (EOF) is functional. Other equipment will require minimal use during a design basis accident.
    - Operator locations for Cabinets 10C626 through 10C610 are outside the 0.22 rem isodose line and therefore require no special administrative controls.
    - Some administrative control may be required in the adjacent office, although the transport angle through the wall would minimize this consideration.
- Administrative controls to maintain operator doses as low as is reasonably achievable (ALARA) will include periodic habitability surveys by Radiation Protection personnel as currently required in Emergency Operating Procedures (EOPs). In addition to these surveys, operators will need to sign in on a Radiation Work Permit (RWP) to access the plant. This RWP will contain instructions to minimize time spent in this area. If the dose rate exceeds 2.5 mR/hr, RP personnel will be required to post the area as a Radiation Area.
- A zone is identified where these controls are practical. All other areas of the control room are assumed to have a dose contribution equivalent to that calculated for the boundary of the established zone.

The potential contributions of other external sources are revisited qualitatively to assure that dose contributions are not significant. Sources considered include (a) The Unit 2 Core Spray line, (b) Reactor Enclosure airborne activity; SGTS Filter Shine; and Control Room Filter Shine.

Core Spray Line Analysis:
The steps performed this analysis were:

1. Use RADTRAD with RG 1.183, Table 1 non-noble gas release fractions and timing, with activity released to the suppression pool.
2. Have RADTRAD calculate the compartment activity as a function of time. Select a number of time steps and spacing such that source integration can be performed linearly by multiplying the step duration times the average activity over the step based on the activity at the start and end. A total of 53 time steps were used, and linear integration is conservative.
3. Extract results from the RADTRAD output into a table of the RADTRAD calculated activities and then perform the integration.
4. Use a spreadsheet to take the integrated sources through 1 day, 4 days, and 30 days, and apply occupancy factors, and unit conversions to create a MicroShield suitable time integrated source file. *{If a source in uCi/cc is input into MicroShield then doss rates are calculated (e.g. mR/hr). If the integrated source input is in uCi - hr/c, the result is a dose (e.g. in mR).}*
5. Run MicroShield using the above source file to determine doses, with design basis Control Room occupancy, at distances from the Control Room interior wall of 1, 3, 6, 9, 12, 15, 18, and 21 feet. A 20-foot piping segment is used, with doses calculated at 6 feet above the floor. Doses are calculated at these distances perpendicular to the wall. {Note that doses at the same distances from the source are also calculated at 15 and 30 degrees from perpendicular, but are not credited for conservatism, and to avoid oblique angle buildup complications.}

Results of Dose Assessment for Core Spray Pipe to Control Room
The Isodose Curve figure on the following page shows the calculated dose rates (perpendicular) overlayed on a small portion of a control room layout drawing to identify locations affected. The underlying layout is taken from LGS Drawing M-602, Rev. 29.

MicroShield Calculated Integrated Operator Doses are:

Table 3

	Integrated Dose (rem) vs. Distance and Angle							
Distance From Wall (feet)	1	3	6	9	12	15	18	21
Perpendicular	**1.081**	**0.808**	**0.571**	**0.428**	**0.334**	**0.268**	**0.220**	**0.183**
15 degrees	0.764	0.561	0.402	0.302	0.236	0.189	0.155	0.129
30 degrees potentially non-conservative	0.242	0.179	0.126	0.094	0.074	0.059	0.049	0.041

Other External Sources

1. 18 inch NPS RHR Piping: This source was conservatively modeled as two 34 foot pipe segments located at 54.5 feet from the 1 ft inside the control room dose point perpendicular to the center of the two pipe segments. This is a conservative approximation for the piping visible to the control room wall. A MicroShield result for a 34 foot segment, doubled, is approximately 0.12 rem.
2. Other sources such as reactor enclosure airborne and external cloud and RERS, SGTS, and CREFAS filters are negligible because of shielding, distance or both.

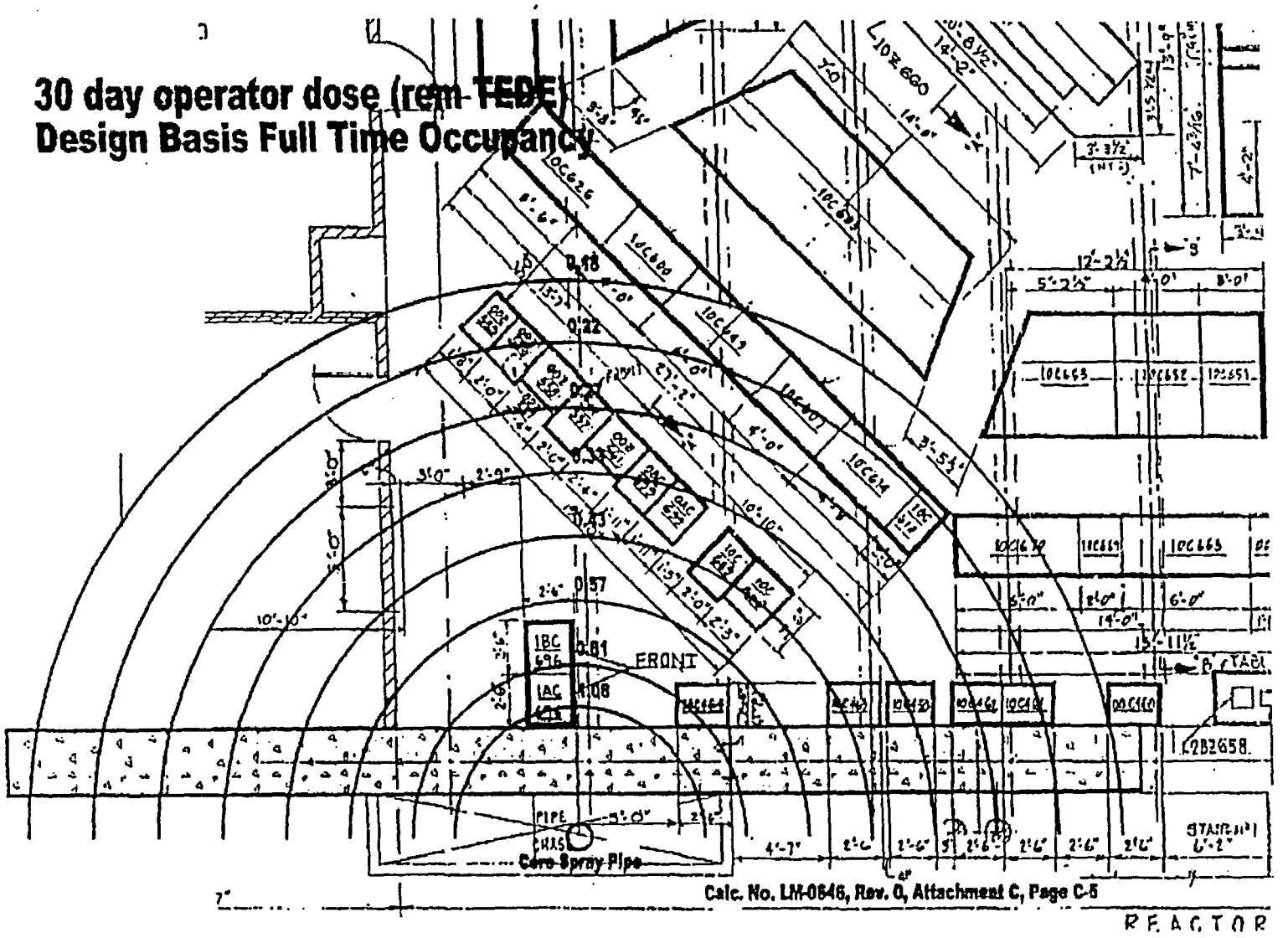
30 day operator dose (rem TEDE)
Design Basis Full Time Occupancy
0.18
0.22
0.33
0.57
0.81
1.08
FRONT
Core Spray Pipe
Calc. No. LM-0646, Rev. 0, Attachment C, Page C-5
REACTOR

Figure 1

Treatment of Alternate Drain Pathway MSIV Leakage

MSIV leakage is the only Secondary Containment bypass pathway analyzed for radiological dose consequences.

The radioactivity associated with all MSIV leakage is assumed to be released directly from the Primary Containment and into the Main Steam Lines. MSIV leakage has separate limits and a separately analyzed dose assessment, therefore it is not included in the L_a fraction limit, and is instead separately controlled.

MSIV leakage assumed in this accident analysis is 200 scfh total for all steam lines and 100 scfh for any one line.

Therefore, at upstream conditions this is a flow rate of:

*100 scfh/line * 14.7 psia / (14.7 psia+22 psig) / 60 min/hr = 0.668 cfm.*

MSIV leakage testing is performed at 22 psig. Containment pressures above the MSIV test pressure persist for only about the first 6.5 minutes of the DBA-LOCA During this limited time period very little containment air is transported into the inboard piping and even less to outboard components. Informal test runs suggest that leakage during this period, and for the first 20 minutes, results in negligible dose contributions, even if an adjustment were made to extrapolate leakage to what might be expected if MSIVs were tested at the LGS P_a of 44 psig

However, to provide design margin, the above leak rate is increased by 25% for the first 24 hours to a value of 0.834 cfm, as shown in the attached table ("Determination of MSL Decontamination Factors Due to Iodine Deposition."

Outboard flow rates are based on expansion of this fluid from the MSIV test pressure to atmospheric pressure, and by further expansion based on worst case heating the fluid to steam line temperatures from standard temperatures. Steam line temperatures are derived based on a generic BWR evaluation crediting only conduction through pipe walls and insulation and is discussed in the attached table, "Assessment of Steam Line Temperatures for Piping Deposition Credit." Credit is taken for temperature reductions only at 24 hours and at 96 hours. This results in the outboard flow rates shown.

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.

Determination of inboard steam line, outboard steam line and condenser effective filter efficiencies is shown in the attached table ("Determination of MSL Decontamination Factors Due to Iodine Deposition."

Modeling of Deposition Credit in Pipes and Condenser

LGS has previously been analyzed and licensed to no longer credit a MSIV Leakage Control System, and to credit seismically analyzed portions of Turbine Condenser System. This historical evaluation is based on methodology described in NEDC-31858P, Rev. 2, "BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems". That analysis was based on a design basis recirculation line break and TID-14844 based source terms. In this calculation the analysis of MSIV leakage is updated to reflect Alternate Source Term parameters related to release timing and chemical makeup and more recent approaches regarding fission product settling and deposition.

Modeling of aerosol settling and elemental iodine deposition is based on methodology used by NRC in AEB-98-03. For the two bounding steam lines modeled, two nodes are used. The first node is from the reactor pressure vessel to the inboard MSIV. The second node is from the inboard MSIV to the Turbine Stop Valve that provides the seismically designed boundary of the MSIV Leakage Control System. For aerosol settling, only horizontal piping runs are credited, and only the bottom surface area is credited. Per AEB-98-03, a median settling velocity is used, given the conservatism in using a well-mixed treatment. For elemental iodine deposition both horizontal and vertical piping is credited, as well as all surfaces since this deposition is not gravity dependent.

The attached table, "Main Steam Piping Summary Unit 1," shows the derivation of piping volumes, surface areas for settling and deposition, and piping effective filter efficiencies for each piping node.

For conservatism, no credit is taken for deposition in the drain lines that provide the previously licensed Alternative Drain Path to the condenser. Credit is taken for deposition in the condenser, where the deposition area is the horizontal surface of the wetwell, and the HP condenser walls. By the time that activity has reached the condenser the aerosols are essentially depleted. Therefore, vertical wall surfaces are credited for elemental iodine removal. No credit is taken for any organic iodine removal in piping or the condenser.

All MS drain lines are routed to a single penetration in the HP condenser at a point below the condenser tubing. Unlike NEDC-31858P, iodine resuspension from settled or deposited iodines is not calculated. Historically, this phenomenon increased organic iodine release by about a factor of two based on resuspension of TID-14844 based elemental iodine fractions. The presence of this phenomenon is questionable with aerosols with significant cesium loadings. Furthermore, while deposition on condenser tubing is not formally credited, test cases have shown that substantial removal of elemental and even organic iodine would be predicted that would more than offset any resuspension. These results are shown in the attached table, "LGS Condenser Characterization." The condenser tubing provides a surface area that is 40 times that of the credited wall and bottom surface areas. It should also be noted that the HP, IP, and LP condensers are interconnected by substantial openings, but flow the IP and LP condensers for further holdup is not credited.

Flow rates out of the condenser are assumed to be at 120°F and atmospheric pressure. A factor of 1.25 is applied, as is done with leakage and flow through outboard steam lines. This leak rate is also reduced by 50% after 24 hours, consistent with the change in Containment conditions.

Figure 2

Assessment of Steam Line Temperatures for Piping Deposition Credit Analysis

Main steam line wall temperatures following a LOCA have not been calculated specifically for LGS. A generic cooldown analysis developed by GE and reported in August 1990 Cline Report (Reference 28) is used. Cooling is considered independent of leakage flow. Only conduction is considered. The function to the right is from Reference 28 and fits the figure below (from Reference 29) and is used for Limerick:

$$T(^{\circ}K) = 299.7 + 265.6 * e^{04.428*10^{-6}t}$$

where

t *time*, sec.

Time (hrs)	Temperature °K	Temperature °F	Value Used °F
0	565.3	557.9	558
1	561.1	550.3	I
2	557.0	542.9	I
3	552.9	535.5	I
4	548.9	528.3	I
5	545.0	521.2	I
6	541.1	514.3	I
7	537.3	507.4	I
8	533.5	500.6	I
9	529.8	494.0	I
10	526.2	487.4	I
11	522.6	481.0	I
12	519.1	474.6	I
13	515.6	468.4	I
14	512.2	462.2	I
15	508.8	456.2	I
16	505.5	450.2	I
17	502.3	444.4	I
18	499.0	438.6	I
19	495.9	432.9	I
20	492.8	427.4	I
21	489.7	421.9	I
22	486.7	416.5	I
23	483.8	411.1	\I/
24	480.9	405.9	410
48	423.3	302.2	I
72	384.0	231.5	\I/
96	357.2	183.3	200

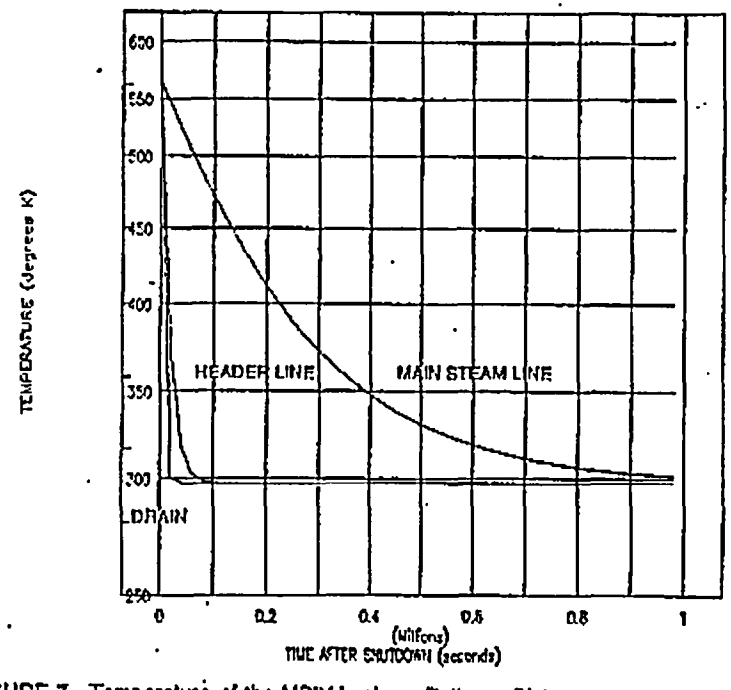

FIGURE 7. Temperature of the MSIV Leakage Pathway Piping as a Function of Time after Shutdown. The Rate of Cooling is Nearly Independent of the Flow Rate at the Low Rates Considered in the Present Analysis.

Table 4

Determination of MSL Decontamination Factors Due to Iodine Deposition

	Inboard A	Inboard B	Inboard C	Inboard D	Outboard A	Outboard B	Outboard C	Outboard D	Condenser No Tube Credit	Condenser 10% Tubes	Condenser 100% Tubes
[1]Total Pipe Surface Area (ft^2)	512	606	608	512	2353	2369	2456	2397	9022.59	44373.19	359396.59
[1]Total Pipe Volume (ft^3)	256	306	303	258	1182	1251	1236	1206	54745.5	54745.5	54745.5
[1]Horizontal Total Pipe Surface Area (ft^2)	219	315	315	219	2252	1931	2300	2259	N/A	N/A	N/A
Horizontal Settling Pipe Surface Area (ft^2)	109.68	157.60	157.60	109.68	1125.99	935.45	1179.52	1149.42	1343.06	18959.86	176584.06
[1]Horizontal Pipe Volume (ft^3)	110	159	159	110	1123	1031	1167	1156	54745.5	54745.5	54745.5
[2]Aerosol Settling Velocity (m/s)	1.173E-03	1.173E-03	1.173E-03	1.173E-03	1.173E-03	1.173E-03	1.173E-03	1.173E-03	1.173E-03	1.173E-03	1.173E-03
Aerosol Settling Velocity (ft/s)	3.839E-03	3.839E-03	3.839E-03	3.839E-03	3.839E-03	3.839E-03	3.839E-03	3.839E-03	3.839E-03	3.839E-03	3.839E-03
[5,6]Elemental Deposition Velocity 0-24hrs (m/sec)	5.359E-06	5.359E-06	5.359E-06	5.359E-06	5.359E-06	5.359E-06	5.359E-06	5.359E-06	2.289E-04	2.289E-04	2.289E-04
[5,6]Elemental Deposition Velocity 24-96hrs (m/sec)	1.248E-05	1.248E-05	1.248E-05	1.248E-05	1.248E-05	1.248E-05	1.248E-05	1.248E-05	2.289E-04	2.289E-04	2.289E-04
[5,6]Elemental Deposition Velocity 96-720hrs (m/sec)	7.945E-05	7.945E-05	7.945E-05	7.945E-05	7.945E-05	7.945E-05	7.945E-05	7.945E-05	2.289E-04	2.289E-04	2.289E-04
Elemental Deposition Velocity 0-24hrs (ft/sec)	1.759E-05	1.759E-05	1.759E-05	1.759E-05	1.759E-05	1.759E-05	1.759E-05	1.759E-05	7.507E-04	7.507E-04	7.507E-04
Elemental Deposition Velocity 24-96hrs (ft/sec)	4.095E-05	4.095E-05	4.095E-05	4.095E-05	4.095E-05	4.095E-05	4.095E-05	4.095E-05	7.507E-04	7.507E-04	7.507E-04
Elemental Deposition Velocity 96-720hrs (ft/sec)	2.607E-04	2.607E-04	2.607E-04	2.607E-04	2.607E-04	2.607E-04	2.607E-04	2.607E-04	7.507E-04	7.507E-04	7.507E-04
[5,6]Organic Deposition Velocity 0-24hrs (m/sec)	5.969E-09	5.969E-09	5.969E-09	5.969E-09	5.969E-09	5.969E-09	5.969E-09	5.969E-09	2.549E-07	2.549E-07	2.549E-07
[5,6]Organic Deposition Velocity 24-96hrs (m/sec)	1.393E-08	1.393E-08	1.393E-08	1.393E-08	1.393E-08	1.393E-08	1.393E-08	1.393E-08	2.549E-07	2.549E-07	2.549E-07
[5,6]Organic Deposition Velocity 96-720hrs (m/sec)	8.849E-08	8.849E-08	8.849E-08	8.849E-08	8.849E-08	8.849E-08	8.849E-08	8.849E-08	2.549E-07	2.549E-07	2.549E-07
Organic Deposition Velocity 0-24hrs (ft/sec)	1.959E-08	1.959E-08	1.959E-08	1.959E-08	1.959E-08	1.959E-08	1.959E-08	1.959E-08	8.361E-07	8.361E-07	8.361E-07
Organic Deposition Velocity 24-96hrs (ft/sec)	4.561E-08	4.561E-08	4.561E-08	4.561E-08	4.561E-08	4.561E-08	4.561E-08	4.561E-08	8.361E-07	8.361E-07	8.361E-07
Organic Deposition Velocity 96-720hrs (ft/sec)	2.903E-07	2.903E-07	2.903E-07	2.903E-07	2.903E-07	2.903E-07	2.903E-07	2.903E-07	8.361E-07	8.361E-07	8.361E-07
Uncorrected Flow Rate (scfh)	100	100	100	100	100	100	100	100	200	200	200
From Containment Flow Rate 0-24 hrs (cfm)	0.834	0.834	0.834	0.834	N/A	N/A	N/A	N/A	N/A	N/A	N/A
From Containment Flow Rate 24-720 hrs (cfm)	0.417	0.417	0.417	0.417	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Pipe Flow Rate 0-24 hrs (cfm)	0.835	0.835	0.835	0.835	4.017	4.017	4.017	4.017	4.577	4.577	4.577
Pipe Flow Rate 24-96 hrs (cfm)	0.417	0.417	0.417	0.417	1.716	1.716	1.716	1.716	2.289	2.289	2.289
Pipe Flow Rate 96-720 hrs (cfm)	0.417	0.417	0.417	0.417	1.302	1.302	1.302	1.302	2.289	2.289	2.289
Pipe Flow Rate 0-24 hrs (cfh)	50.095	50.095	50.095	50.095	241.034	241.034	241.034	241.034	274.621	274.621	274.621
Pipe Flow Rate 24-96 hrs (cfh)	25.034	25.034	25.034	25.034	102.993	102.993	102.993	102.993	137.311	137.311	137.311
Pipe Flow Rate 24-720 hrs (cfh)	25.034	25.034	25.034	25.034	78.125	78.125	78.125	78.125	137.311	137.311	137.311
[3]Aerosol Settling Rate Constant (hr^{-1})	1.37E+01	1.37E+01	1.37E+01	1.37E+01	1.37E+01	1.37E+01	1.37E+01	1.37E+01	3.40E-01	4.70E+00	4.46E+01
[3]Elemental Deposition Rate Constant 0-24hr (hr^{-1})	1.20E-01	1.20E-01	1.20E-01	1.20E-01	1.20E-01	1.20E-01	1.20E-01	1.20E-01	4.45E-01	2.18E+00	1.77E+01
[3]Elemental Deposition Rate Constant 24-96hr (hr^{-1})	2.93E-01	2.93E-01	2.93E-01	2.93E-01	2.93E-01	2.93E-01	2.93E-01	2.93E-01	4.45E-01	2.18E+00	1.77E+01
[3]Elemental Deposition Rate Constant 96-720hr (hr^{-1})	1.87E+00	1.87E+00	1.87E+00	1.87E+00	1.87E+00	1.87E+00	1.87E+00	1.87E+00	4.45E-01	2.18E+00	1.77E+01
[3]Organic Deposition Rate Constant 0-24hr (hr^{-1})	1.40E-04	1.40E-04	1.40E-04	1.40E-04	1.40E-04	1.40E-04	1.40E-04	1.40E-04	4.55E-04	2.42E-03	1.98E-02
[3]Organic Deposition Rate Constant 24-96hr (hr^{-1})	3.20E-04	3.20E-04	3.20E-04	3.20E-04	3.20E-04	3.20E-04	3.20E-04	3.20E-04	4.55E-04	2.42E-03	1.98E-02
[3]Organic Deposition Rate Constant 96-720hr (hr^{-1})	2.09E-03	2.09E-03	2.09E-03	2.09E-03	2.09E-03	2.09E-03	2.09E-03	2.09E-03	4.55E-04	2.42E-03	1.98E-02
[4]Aerosol Filter Efficiency (0-24 hrs)	96.80%	97.75%	97.75%	96.80%	98.47%	98.28%	98.54%	98.51%	98.92%	99.83%	99.99%
[4]Aerosol Filter Efficiency (24-96 hrs)	98.38%	98.86%	98.86%	98.38%	99.34%	99.25%	99.37%	99.38%	99.27%	99.95%	99.99%
[4]Aerosol Filter Efficiency (96-720 hrs)	98.38%	98.86%	98.86%	98.38%	99.53%	99.44%	99.62%	99.61%	99.27%	99.95%	99.99%
[4]Elemental Filter Efficiency (0-24 hrs)	39.20%	43.46%	43.46%	39.20%	38.17%	35.43%	39.23%	38.63%	98.89%	99.77%	99.97%
[4]Elemental Filter Efficiency (24-96 hrs)	73.11%	78.18%	78.18%	73.11%	77.09%	74.94%	77.87%	77.43%	99.44%	99.89%	99.99%
[4]Elemental Filter Efficiency (96-720 hrs)	95.05%	95.80%	95.80%	95.05%	96.58%	96.17%	96.72%	96.64%	99.44%	99.89%	99.99%
[4]Organic Filter Efficiency (0-24 hrs)	0.07%	0.09%	0.09%	0.07%	0.07%	0.09%	0.07%	0.07%	9.00%	32.57%	79.78%
[4]Organic Filter Efficiency (24-96 hrs)	0.33%	0.40%	0.40%	0.33%	0.37%	0.33%	0.30%	0.38%	16.51%	49.14%	88.74%
[4]Organic Filter Efficiency (96-720 hrs)	2.09%	2.48%	2.48%	2.09%	3.35%	2.72%	3.18%	3.11%	16.51%	49.14%	88.74%

[1]Pipe Wall Temperature, 0-24hr (F)	558.33
Pipe Wall Temperature, 0-24hr (K)	565.37
[1]Pipe Wall Temperature, 24-96hr (F)	410.33
Pipe Wall Temperature, 24-96hr (K)	483.15
[1]Pipe Wall Temperature, 96-720hr (F)	200.33
Pipe Wall Temperature, 96-720hr (K)	366.48
[8]Condenser Temperature, constant (F)	120.33
Condenser Temperature, constant (K)	322.04
Standard Temperature, constant (F)	68
[9]MSIV Test Pressure, constant (psig)	22
[10]Standard Test Pressure, constant (psig)	44
Atmospheric Pressure, constant (psia)	14.7
Flow Rate Conservatism Factor, constant	1.25

References:

1. Piping Take offs and Condenser Characterization (Pages B-9 and B-14)
2. USNRC Document AEB 98-03 12/01/1998, Page A-3, Median Value
3. USNRC Document AEB 98-03 12/01/1998, Page A-2, Formula 2
4. USNRC Document AEB 98-03, 12/01/1998, Page A-2, Formula 4
5. Cline J.E. "MSIV Leakage Iodine Transport Analysis", 3000-591
6. NUREG/CR-6604 RADTRAD Manual 4/1998 Supplement 1 6/1999
7. Cline J.E. 8/1990, as shown in Design Analysis LM 0545 Rev. 0, Attachment F (this Calc.)
8. Typical conservative value for HP Condenser Volume
9. LGS Technical Specification LCO 3.6.1.2.c
10. LGS Technical Specification SR 4.6.6.1.c.2

Table 5
LGS Condenser Characterization

High Pressure Condenser

Base Section (Rectangular Volume)

33.98 Height (ft.)
29.00 Length (East - West Direction) (ft.)
40.42 Width (North - South Direction) (ft.)
45710.75 Volume (ft^3)

Flue Section (Trapezoidal Volume)

18.67 Height (vertical) (ft.)
29.00 Length (East - West Direction) (ft.)
23.67 Top Width (North - South Direction) (ft.)
40.42 Bottom Width (North - South Direction) (ft.)
21.66 Slant Length (ft)
18509.22 Volume (ft^3)

Top Section (Rectangular Volume)

6.00 Height (ft.)
29.00 Length (East - West Direction) (ft.)
23.67 Width (North - South Direction) (ft.)
4118.00 Volume (ft^3)

4.33 Water Level (ft.)
1346.08 Condenser Base Area (ft^2)
5833.03 Contained Water Volume (ft^3)

350,696 Condenser Tube Surface Area (ft^2)
1,190.720 Calculated Length of Tubes (ft.)
8,219.44 Volume Displaced by Tubes (ft^3)

1,346.08 Setting Area (floor only)
9022.59 Deposition Area (floor plus Walls)

Shell Volume (ft^3) 68,767.97
Volume Above Hotwell (ft^3) 62,964.94
Free Air Volume (ft3) 54,745.50
Tube Length: 48
Condenser Tube O.D. (ft.) 0.09375
Total Condenser Tube Surface Area (ft^2) 920.003
Tube Surface Area (ft^2), by shell 350.476
Calculated Total Length of Tubes (ft.) 3,123.681
Calculated Total Volume Displaced by Tubes (ft^3) 21,562.50

Intermediate Pressure Condenser

Base Section (Rectangular Volume)

33.98 Height (ft.)
29.00 Length (East - West Direction) (ft.)
40.42 Width (North - South Direction) (ft.)
39602.00 Volume (ft^3)

Flue Section (Trapezoidal Volume)

18.67 Height (vertical) (ft.)
29.00 Length (East - West Direction) (ft.)
23.67 Top Width (North - South Direction) (ft.)
40.42 Bottom Width (North - South Direction) (ft.)
20.46 Slant Length (ft)
17345.22 Volume (ft^3)

Top Section (Rectangular Volume)

6.00 Height (ft.)
29.00 Length (East - West Direction) (ft.)
23.67 Width (North - South Direction) (ft.)
4118.00 Volume (ft^3)

4.33 Water Level (ft.)
1172.08 Condenser Base Area (ft^2)
5079.03 Contained Water Volume (ft^3)

306,096 Condenser Tube Surface Area (ft^2)
1,041,225 Calculated Length of Tubes (ft.)
7,187.48 Volume Displaced by Tubes (ft^3)

1,172.08 Setting Area (floor only)
8299.89 Deposition Area (floor plus Walls)

Shell Volume (ft^3) 61,265.22
Volume Above Hotwell (ft^3) 56,186.19
Free Air Volume (ft3) 48,998.71
Tube Length: 42

Tube Surface Area (ft^2), by shell 306.097

Low Pressure Condenser

Base Section (Rectangular Volume)

33.98 Height (ft.)
29.00 Length (East - West Direction) (ft.)
34.42 Width (North - South Direction) (ft.)
33693.25 Volume (ft^3)

Flue Section (Trapezoidal Volume)

18.67 Height (vertical) (ft.)
29.00 Length (East - West Direction) (ft.)
23.67 Top Width (North - South Direction) (ft.)
34.42 Bottom Width (North - South Direction) (ft.)
19.43 Slant Length (ft)
15721.22 Volume (ft^3)

Top Section (Rectangular Volume)

6.00 Height (ft.)
29.00 Length (East - West Direction) (ft.)
23.67 Width (North - South Direction) (ft.)
4118.00 Volume (ft^3)

4.33 Water Level (ft.)
995.08 Condenser Base Area (ft^2)
4325.03 Contained Water Volume (ft^3)

262,636 Condenser Tube Surface Area (ft^2)
891,736 Calculated Length of Tubes (ft.)
6,155.58 Volume Displaced by Tubes (ft^3)

968.08 Setting Area (floor only)
7588.40 Deposition Area (floor plus Walls)

Shell Volume (ft^3) 53,732.47
Volume Above Hotwell (ft^3) 49,407.44
Free Air Volume (ft3) 43,251.86
Tube Length: 36

Tube Surface Area (ft^2), by shell 262.857

Total Shell Volume (ft3) 183,796
Total Volume Above Hotwell (ft3) 168,559
Total Free Air Volume (ft^3) 146,998

References:
[1] LGS Dwg. No. 90-1097-5-103, Rev. F
[2] LGS Dwg. No. 90-1097-5-101, Rev. E
[3] LGS Dwg. No. 90-1097-5-102, Rev. E
[4] LGS Dwg. No. 90-1097-5-112, Rev. K
[5] LGS DEO No. L-S-21, Rev. 10, "Condensate & Condenser System"

Table 6

Main Steam Piping Summary Unit 1

24.14 Main Steam 26 Inch pipe ID — LGS Spec P-300 Rev. 44, "Piping Materials & Instrum. Piping Standards"

TOTAL MS PIPING

A	B	C	D	
5435.75	5122.09	5822.59	5524.75	26 inch piping (inches)
2863	2698	3066	2910	26 inch piping inside surface area (sq. ft.)
1440	1357	1542	1463	26 inch piping inside volume (cu. ft.)
2863	2698	3066	2910	Total inside surface area (sq. ft.)
1440	1357	1542	1463	Total inside volume (cu. ft.)
2863	2698	3066	2910	Total inside surface area (sq. ft.)
1440	1357	1542	1463	Total inside volume (cu. ft.)

HORIZONTAL MS PIPING ONLY

A	B	C	D	
4692.50	4376.84	5079.34	4781.50	26 inch piping (inches)
2471	2306	2675	2518	26 inch piping inside surface area (sq. ft.)
1243	1160	1345	1266	26 inch piping inside volume (cu. ft.)
2471	2306	2675	2518	Total inside surface area (sq. ft.)
1243	1160	1345	1266	Total inside volume (cu. ft.)

Horizontal Totals

4276.00	3780.34	4480.84	4365.00	26 inch Outboard Piping length (inches)
2252	1991	2360	2299	Total Outboard Pipe Surface Area Credit (sq. ft.)
1133	1001	1187	1156	Total Outboard Pipe Volume Credit (cu. ft.)
416.50	598.50	598.50	416.50	26 inch Inboard Piping length (inches)
219	315	315	219	Total Inboard Pipe Surface Area Credit (sq. ft.)
110	159	159	110	Total Inboard Pipe Volume Credit (cu. ft.)

Totals

4462.75	3967.09	4867.59	4551.75	26 inch Outboard Piping length (inches)
2350	2089	2458	2397	Total Outboard Pipe Surface Area Credit (sq. ft.)
1182	1051	1236	1206	Total Outboard Pipe Volume Credit (cu. ft.)
973.00	1155.00	1155.00	973.00	26 inch Inboard Piping length (inches)
512	608	608	512	Total Inboard Pipe Surface Area Credit (sq. ft.)
258	306	306	258	Total Inboard Pipe Volume Credit (cu. ft.)

Nodalization (Horizontals)						
416.50	598.50	598.50	416.50	Node 1 Length (inches)		
219	315	315	219	Node 1 Surface Area (sq. ft.)		
110	159	159	110	Node 1 Volume (cu. ft.)		
4276.00	3780.34	4480.84	4365.00	Node 2 Length (inches)		
2252	1991	2360	2299	Node 2 Surface Area (sq. ft.)		
1133	1001	1187	1156	Node 2 Volume (cu. ft.)		
Nodalization (Totals)						
973.00	1155.00	1155.00	973.00	Node 1 Length (inches)		
512	608	608	512	Node 1 Surface Area (sq. ft.)		
258	306	306	258	Node 1 Volume (cu. ft.)		
4462.75	3967.09	4867.59	4551.75	Node 2 Length (inches)		
2350	2089	2458	2397	Node 2 Surface Area (sq. ft.)		
1182	1051	1236	1206	Node 2 Volume (cu. ft.)		

Atmospheric Dispersion

Source Configuration

The North and South Stacks are executed by ARCON96 as a vent release. As depicted in Attachment B, both stacks have a height of 416 ft MSL (200 ft above grade). The stacks are located between Reactor Enclosures 1 and 2 with the North Stack situated on the north face of the buildings and the South Stack on the south face of the buildings (designated as Normal Release Point 2, "HVAC VENTS FOR REACTOR ENCLOSURES"). These stacks are less than 2.5 times the 194.75 ft high Reactor Buildings (i.e., the highest adjacent building), and therefore, per Regulatory Guide 1.145, they are modeled as a 'vent' release.

Both the North and South Stacks are conservatively assumed to have a zero (0) flow, for which ARCON96 requires that the exit velocity and stack diameter each be assigned an input value of zero (0). Per Regulatory Guide 1.194, Table A-2, the actual building vertical cross-sectional area perpendicular to the wind direction must be utilized; therefore, the Reactor Enclosures' combined vertical cross-sectional area of 5851 m^2 (calculated as height = 59.4 m, and w = 98.5 m), was input into ARCON96 to account for wake effects.

Receptors

The model ARCON96 was executed for X/Q at the Control Room Intake, which is centered on the north face of the Control Structure at a height of 124 ft above grade.

The direction, relative to true north (assumed 0°) of a straight line extending from the Control Room Intake towards the stack source location, is also an input parameter required by ARCON96. Attachment D depicts the two (2) separate intake-to-stack direction scenarios analyzed in this calculation. They are as follows:

	Direction (degrees) Intake to Stack	Distance (m) Intake to Stack
• North Stack	180	16.5
• South Stack	180	64.8

Meteorological Data

The Station's meteorological database for the five-year period from 1996 through 2000 was applied in the ARCON96 modeling analysis. Data measured at two (one primary and one backup) meteorological towers were used.

Meteorological Tower 1 is the primary tower and is located approximately 0.6 miles north-northwest of the North and South Stacks, whereas Tower 2 is the backup tower

and is located west at approximately 0.4 miles from the North and South Stacks. Tower 2 data were used only for substitution of any missing Tower 1 data as follows:

Limerick Meteorological Tower Instrument Levels

(Elevations in reference to tower grade)

	Tower 1 (primary)	Tower 2 (backup)
Wind Speed:		
Elevation 1	30 ft	159 ft
Elevation 2	175 ft	304 ft
Wind Direction:		
Elevation 1:	30 ft	159 ft
Elevation 2:	175 ft	304 ft

The meteorological vendor illustrated that the Tower 2 delta temperature data are sufficiently representative to be substituted for the Tower 1 delta temperature data; however, since the Tower 1 and Tower 2 delta temperature height intervals differ from each other somewhat, and also since for all years shown, the primary Tower 1 has data recovery rates well above the NRC's 90 percent requirements, it was deemed unnecessary to make such substitutions.

Hereinafter, the Tower 1 ARCON96 meteorological input database with applicable Tower 2 values substituted for missing Tower 1 values as indicated above will be identified as the "Tower 1 Modeling Database".

The designation of 'calm' is made to all wind speed observations 0.5 mph or less. The higher of the starting speeds of the wind vane and anemometer equipment on each of the towers (i.e. 0.5 mph) was used as the threshold for calm winds, per Regulatory Guide 1.145, Section 1.1.

Attachment 7 to this License Amendment Request contains the lower and upper level joint wind direction, wind speed, and stability class distribution tables, based on the five-year lower and upper level Tower 1 Modeling Database, as used for the ARCON96 modeling analysis. (These data are provided both in the format of number of observations and percent occurrence frequency.)

ARCON96 Run Scenarios

Control Room Intake X/Q values were calculated by ARCON96 for various source/receptor scenarios. These two scenarios were analyzed using the five-year hourly meteorological joint wind and stability database, as identified below:

ARCON96 RELEASE SCENARIO	METEOROLOGICAL DATABASE SCENARIOS		
	Wind Speed and Direction		Stability Class (Delta Temperature)
	Primary	Secondary*	
1: North Stack	Tower 1: 175 ft	Tower 1: 30 ft	Tower 1: 171 – 26'
2: South Stack	Tower 1: 175 ft	Tower 1: 30 ft	Tower 1: 171 – 26'

* Secondary data used only for those hours when primary data are missing.

The upper level of the Tower 1 Modeling database is the obvious most representative monitoring location of choice for wind data representing the North and South Stack release points.

The North and South Stacks are not tall enough to avoid building-induced downwash; therefore, with zero (0) exit velocity having been assumed, ARCON96 treats their releases as a 'ground-level' type release.

Calculations

The X/Q values resulting from the ARCON96 modeling analysis of each release and meteorological database scenario for the required time intervals are presented as follows:

Table 7
ARCON96 X/Q (sec/m^3) RESULTS

RELEASE / INTAKE & METEOROLOGICAL SCENARIO	0-2 hour	2-8 hour	8-24 hour	1-4 day	4-30 day
1. North Stack to Control Room Intake: • Wind: Tower 1 175' Stability: Tower 1 171 – 26'	6.88E-03	5.17E-03	2.04E-03	1.29E-03	9.63E-04
2. South Stack To Control Room Intake: • Wind: Tower 1 175' Stability: Tower 1 171 – 26'	1.26E-03	9.64E-04	3.80E-04	2.39E-04	1.80E-04

PAVAN MODELING ANALYSES OF CONTROL ROOM, EAB AND LPZ X/Q

The model PAVAN is a commercial software package designated by Washington Group, International as MC-131, an "active" program applicable to nuclear safety related analyses as well as non-safety related studies and evaluations. The PAVAN code Revision 1 verification was performed for the 0-2 hour, 0-8 hour, 8-24, 1-4 day, and 4-30 day 0.5-percentile, and annual average direction-specific X/Q values, and the overall site 95-percentile maximum X/Q for each of the 0-2 hour, 0-8 hour, 8-24 hour, 1-4 day, and 4-30 day time-averaging periods. This verification was performed with WGI (formerly Raytheon Engineers & Constructors, Inc.) corporate standards, and is consistent with Computer Software Control, NEP-09. Revision 1 of MC-131 was verified for ground-level (i.e., non-elevated) releases, as well as elevated releases, with zero (0) vertical exit velocity assumed.

Methodology and Acceptance Criteria

The computer code PAVAN is a straight line Gaussian dispersion model utilized to estimate relative ground-level air concentrations (X/Q) for potential accidental releases of radioactive material from nuclear facilities. Such assessment is required by 10 CFR 100 and 10 CFR 50. The program implements the NRC guidance provided in Regulatory Guide 1.145. The technical basis for the program is presented by Snell and Jubach. Utilizing joint frequency of occurrence distributions of wind direction, wind speed and Pasquill atmospheric stability class, PAVAN calculates X/Q values as a function of direction for various time-averaging periods at the EAB and the outer boundary of the LPZ. Calculations are made from assumed ground-level (i.e., non-elevated) releases (such as vents and building penetrations), which are less than 2.5 times the height of adjacent solid structures, and from elevated releases (i.e., stacks). Three (3) procedures are utilized for calculating X/Q: a direction-dependent approach, a direction-independent approach, and an overall site X/Q approach.

The PAVAN model contains certain model options for executing the program. The table below summarizes the options invoked for the EAB and LPZ X/Q calculations.

Table 8

Option No.	Description	Option Invoked?
1	Calculate σ_y and σ_x based on desert diffusion.	No
2	X/Q values include evaluation for no building wake.	No
3	ENVLOP calculations printed which describe upper envelope curve.	No
4	Print points used in upper envelope curve and calculation.	Yes
5	Null	---
6	Joint frequency distribution in % frequency format.	No
7	Print X/Q calculation details	Yes
8	Distribute calm winds observations into first wind speed category.	Yes
9	Use site-specific terrain adjustment factors for the annual average calculations.	Yes*
10	Assume a default terrain adjustment factor for the average annual calculations. Option 10 is applied, which together with application of Option 9 means that site specific terrain factors will be used.	Yes

* A uniform value of 1.0 is used.

Source Configuration

Releases for Control Room Intake X/Q Evaluation

The North and South Stacks are the assumed release points. Because these stacks do not qualify as 'elevated' releases as defined by Regulatory Guide 1.145, in accordance with Regulatory Guide 1.194 methodology no PAVAN modeling (i.e., only ARCON96 modeling) is appropriate for the Control Room assessment.

Releases for EAB and LPZ X/Q Evaluation
As previously stated, the North and South Stacks have a physical height of 200 ft and are located to the south of the Control Room Intake. These stacks do not quality as elevated releases per Regulatory Guide 1.145. Therefore, the stacks were executed by PAVAN as 'ground' type releases requiring that each of these stack heights be assigned an input value of 10 m. The Reactor Building height of 59.4 m and smallest calculated Reactor Enclosure vertical cross-sectional area of 2426 m^2 was used for each of the scenarios.

Receptors
For the North and South Stack to the EAB and LPZ scenarios, PAVAN was executed in ground release mode with stack-to-intake horizontal distances of 731 m for the EAB and 2043 m for the LPZ.

Meteorological Data
As described in Section 1.0, Limerick meteorological data from the five-year period, 1996-2000 for two meteorological towers (one primary and one backup), was used in the PAVAN analysis.

The format of PAVAN meteorological input consists of a joint wind direction (based on sixteen 22.5 degree sectors), wind speed (7 intervals). and stability class (7 classes) occurrence frequency distribution.

Each such meteorological joint frequency distribution for input to PAVAN was prepared by using the WGI pre-qualified program ARCONtoPAVANMETrev1 (Program Number NU-840) to transform the data to a joint wind-stability occurrence frequency distribution. The seven wind speed categories were defined according to Regulatory Guide 1.23 with the first category identified as "calm". The higher of the starting speeds of the wind vane and anemometer equipment on each of the towers (i.e. 0.50 mph) was used as the threshold for calm winds, per Regulatory Guide 1.145. Section 1.1. A midpoint was also assumed between each of the Regulatory Guide 1.23 wind speed categories, Nos. 2-6, as to be inclusive of all monitored wind speeds. The Regulatory Guide 1.23 wind speed categories have, therefore, been refined as follows:

DEFINED WIND SPEED CATEGORY RANGES FOR PAVAN MODELING

Category No.	Regulatory Guide 1.23 Speed Interval (mph)	PAVAN-Assumed Speed Interval (mph)
1 (Calm)	0 to < 1	0 to<0.50
2	1 to 3	>=0.50 to <3.5
3	4 to 7	>=3.5 to <7.5
4	8 to 12	>=7.5 to <12.5
5	13 to 18	>=12.5 to <18.5
6	19 to 24	>=18.5 to <24
7	>24	>=24

The same delta temperature stability class database utilized for the ARCON96 analysis described above was also adopted for the PAVAN analysis.

The joint lower level wind-stability class occurrence frequency distribution, based on the five-year Tower 1 Modeling Database, is included in the PAVAN input files.

PAVAN Run Scenarios

Stack release scenarios were identified for the purpose of applying the PAVAN model using the selected representative meteorological wind and stability class databases to predict the X/Q values that result at the EAB and LPZ. They are shown as follows:

PAVAN X/Q SCENARIOS		
RELEASE/RECEPTOR SCENARIO	METEOROLOGICAL DATABASE SCENARIOS (Tower ID: Measurement Height above Tower Grade)	
	Wind Speed and Direction	Stability Class (Delta Temperature)
EAB (731 m):		
• North and South Stack	Tower 1: 30'	Tower 1: 171 – 26'
LPZ (2043 m):		
• North and South Stack	Tower 2: 30'	Tower 2: 171 – 26'

The Tower 1 Modeling Database is representative for deriving all required meteorological input for the PAVAN modeling of the North and South Stack release X/Q for each subject receptor.

The EAB and LPZ are located at distances of 731 m and 2403 m, respectively. It should be noted that the lower (30 ft) level wind speeds contained in the Tower 1 Modeling Database were used instead of the upper (175 ft) winds, even though it might be otherwise expected that the 175 ft level winds would better represent the 200 ft North and South stack tops. This is because PAVAN requires that any non-elevated release be assumed as a 'ground level' release, which accordingly requires that whatever the release elevation may actually be, it is reassigned a value of 10 meters above station grade. Thus, using actual 10-meter monitored data (i.e., data from the 30 ft level on Tower 1) is considered to be superior to using data from another level (i.e., 175 feet) that PAVAN would subsequently adjust (but imprecisely so, by power law extrapolation) down to 10 meters.

Calculations

The X/Q values for the EAB and LPZ calculated by the PAVAN modeling analysis of each release scenario are presented below for each time interval required by NRC Regulatory Guide 1.145.

Table 9
PAVAN X/Q (sec/m^3) Results
North And South Stacks to EAB and LPZ

RELEASE LOCATION	X/Q PARAMETER (sec/m^3)	0-2 hour	0-8 hour	8-24 hour	1-4 day	4-30 day
EAB (731 m)						
North and South Stacks*	Direction-Specific Max	3.18E-04 (ESE)	1.76E-04 (ESE)	1.31E-04 (ESE)	6.89E-05 (ESE)	2.74E-05 (ESE)
	Site Limit	2.79E-04	1.58E-04	1.19E-04	6.39E-05	2.63E-05
LPZ (2043 m)						
North and South Stacks*	Direction-Specific Max	1.15E-04 (ESE)	5.79E-05 (ESE)	4.10E-05 (ESE)	1.95E-05 (ESE)	6.68E-06 (ESE)
	Site Limit	1.01E-04	5.18E-05	3.71E-05	1.81E-05	6.41E-06

* The same PAVAN results apply to the North and South Stacks individually.
**The higher of the direction specific and the site limit values are indicated in bold.

SUMMARY AND CONCLUSIONS

The ARCON96 and PAVAN X/Q modeling calculation results are summarized below for the Control Room, EAB and LPZ for the regulated time-averaging periods. Control Room intake results are calculated using the ARCON96 model and the EAB and LPZ results have been calculated using the PAVAN model. All input files for ARCON96 and PAVAN are provided in Attachment 7.

Table 10
X/Q RESULTS SUMMARY
(sec/m^3)

RECEPTOR	RELEASE POINT	0-2 hour	2-8 hour*	8-24 hour	1-4 day	4-30 day
Control Room Intake	North Stack	6.88E-03	5.17E-03	2.04E-03	1.29E-03	9.63E-04
	South Stack	1.26E-03	9.64E-04	3.80E-04	2.39E-04	1.80E-04
EAB (731 m)	North and South Stacks**	3.18E-04 (ESE)	1.76E-04 (ESE)	1.31E-04 (ESE)	6.89E-05 (ESE)	2.74E-05 (ESE)
LPZ (2,043 m)	North and South Stacks**	1.15E-04 (ESE)	5.79E-05 (ESE)	4.10E-05 (ESE)	1.95E-05 (ESE)	6.68E-06 (ESE)

PAVAN result representing 0-8 hour time period.
** The same PAVAN results apply to the North and South Stacks individually.

RADTRAD Input Information

Table 11a

Primary Containment Leakage Pathway (LOCA) RADTRAD Compartments				
Compartment Number:	**1**	**2**	**3**	**4**
Name:	Containment	Reactor Enclosure	Environment	Control Room
Type:	Other	Other	Environment	Control Room
Volume:	403,120 ft^3 Includes Primary Containment and Wetwell Airspace	1,800,000 ft^3 900,000 cfm used in analysis to account for 50% mixing credit	N/A	126,000 ft^3
Source Term Fraction:	1.0	0.0	0.0	0.0
Compartment Features:	Natural Deposition: Powers BWR model 10% (lower bound)	Recirculation Filter: RERS 0 to 15.5-min: Flow: 30,000 cfm No filtration credited. 15.5 min to 720 hrs: Flow: 30,000 cfm HEPA: 70% Charcoal: 70%	N/A	Recirculation Filter: CREFAS Radiation mode: 2475 cfm. Manual start in 30 min or less HEPA: 80% Charcoal: 99% Chlorine mode: 3000 cfm. Auto Starts HEPA: 80% Charcoal: 99%
Comments:	Volume determined via the containment leakage program. No elemental iodine removal coefficient. Instantaneous and homogeneous mixing is conservatively assumed.	No filter credit during the 15.5-minute drawdown period. 30,000 cfm RERS flow provides mixing in 50% of the RE volume.		

Table 11b

Primary Containment Leakage Pathway (LOCA) RADTRAD Active Transfer Pathways				
Pathway Number and Name:	**1 Containment to Reactor Encl.**	**2 RE Exhaust to SGTS Node**	**3 CR Unfiltered Inleakage**	**4 CR Filtered Intake**
From Compartment:	Containment (1)	Reactor Enclosure (2)	Environment (3)	Environment (3)
To Compartment:	Reactor Enclosure (2)	Environment (3)	Control Room (4)	Control Room (4)
Transfer Mechanism:	Air Leakage	Filtered Exhaust	Unfiltered inleakage	Filtered intake
Transfer Mechanism Details:	Leak rate: 0-24 hr, 0.50% per day; 24-720 hr, 0.25% per day	Flows: 0-1 min: 9.00E6 cfm 1 min to 15.5 min: 3000 cfm (drawdown period); 15.5 min to 720 hrs, 2500 cfm. (Post-drawdown). RERS Filter Efficiencies: 0% during 0 to 15.5 min drawdown period. 70% for aerosol (HEPA), elemental, and organic iodines (Charcoal)	Flow rate: 0-720 hours, 525 cfm – models the assumed bounding unfiltered inleakage. 0% efficiency for all	Radiation Mode: 0-30 min, 2100 cfm; 30 min-720 hours, 525 cfm Models the normal CR filtered intake flow rate. Filter efficiencies: 99% for aerosol (HEPA) and; 80% for elemental and organic iodines (charcoal). Chlorine Mode has no filtered intake.
Comments:		The 0-1 min 9.00E6 cfm flow corresponds to 10 air changes per minute to simulate the last minute of RERS startup (first two minutes are before gap release phase). The 1 min to 15.5 min 3000 cfm is during the drawdown period (no filtration).		

Table 11c

Primary Containment Leakage Pathway (LOCA) RADTRAD Active Transfer Pathways (Continued)				
Pathway Number and Name:	**5** **CR Exhaust (Equilibrium)**	**6** **SGTS Node to Environment**		
From Compartment:	Control Room (4)	SGTS Node (5)		
To Compartment:	Environment (3)	Environment (3)		
Transfer Mechanism:	Filtered Exhaust (Maintains equilibrium)	Filter		
Transfer Mechanism Details:	Rad Mode Flows: 0-30 min, 2625; 30 min-720 hrs, 1050 cfm Chlorine Mode Flows: 0-720 hours, 525 cfm Filter Efficiency Panel – Filter efficiency is entered as 100.0% for all chemical forms of iodine, for all time periods.	Flow rate 0-1 minute, 9.00E+6 cfm Flow rate 1 minute to 15.5 minutes - 3000 cfm for drawdown period. Flow rate for 15.5 minutes until 720 hours, 2500 cfm SGTS Filter Efficiency: 0% during 0 to 15.5 min drawdown period 97.5% for aerosol (HEPA), elemental, and organic iodines (Charcoal)		
Comments:	This equals the total flow that was taken into the CR volume, which includes inleakage (525 + 2100, for manual initiation time; 525 + 525 cfm, after initiation). This is the exit from the control room to the environment; the filtration prevents a double counting of the iodine release, although RADTRAD 3.03 documentation indicates that this effect has been eliminated	0-1 minute, 9.00E+6 cfm corresponds to 10 air changes per minute to simulate the last minute of RERS startup (first two minutes are before gap release phase).		

Table 11d

Primary Containment Leakage Pathway (LOCA) RADTRAD Dose Receptor Locations			
Dose Location Number:	1	2	3
In Compartment:	Control Room	EAB (Distance: 731 Meters)	LPZ (Distance: 2043 Meters)
Breathing Rate (m^3/sec):	0 to 720 hrs: 3.5E-04	0 to 8 hrs: 3.5E-04 8 to 24 hrs: 1.8E-04 24 to 720 hrs: 2.3E-04	0 to 8 hrs: 3.5E-04 8 to 24 hrs: 1.8E-04 24 to 720 hrs: 2.3E-04
Occupancy Fractions:	0 to 24 hrs: 1.0 24 to 96 hrs: 0.6 24 to 720 hrs: 0.4	Highest 2-hr window: 1.0	0 to 720 hrs: 1.0
X/Q (sec/m^3):	0 to 2 hr: 6.88E-03 2 to 8 hr: 5.17E-03 8 to 24 hr: 2.04E-03 1 to 4 day: 1.29E-03 4 to 30 day: 9.63E-04	3.2-5.2 hrs: 3.18E-04 (Highest two hours)	0 to 8 hr: 5.79E-05 8 to 24 hr: 4.10E-05 1 to 4 day: 1.95E-05 4 to 30 day: 6.68E-06

Table 11e

Primary Containment Leakage Pathway (LOCA) RADTRAD Source Terms & Dose Conversion Factors		
Nuclide Inventory:	3527 MWt LGS Specific NIF for 60 MACCS isotopes (See Table 1 of this document)	C0-58 & Co-60 values are RADTRAD default values
Release Fractions & Timing:	RADTRAD Standard BWR-DBA values, No delay	
Dose Conversion Factors:	RADTRAD Library of FGR 11&12 values for 60 MACCS isotopes	
Decay & Daughter Products:	Enabled - Decay / Daughter Products considered	
Iodine Chemical Fractions:	NUREG-1465 based Iodine Chemical Form Fractions Aerosol: 0.95 Elemental: 0.0485 Organic: 0.0015	

Table 12a

MSIV Leakage Pathway (LOCA) RADTRAD Compartments				
Compartment Number:	1	2	3	4
Name:	Containment	(Node 1) Inboard MSL "A" Volume	(Node 1) Inboard MSL "D" Volume	(Node 2) Outboard MSL "A" Volume
Type:	Other	Other	Other	Other
Volume:	403,120 ft^3 Includes Primary Containment and Wetwell Airspace	258 ft^3	258 ft^3	1182 ft^3
Source Term Fraction:	1.0	0.0	0.0	0.0
Compartment Features:	Natural Deposition: Powers BWR model 10% (lower bound) No elemental iodine removal coefficient.	None	None	None
Comments:	Volume determined via containment leakage program. Instantaneous and homogeneous mixing is assumed.	Minimum steam line piping volume from RPV to inboard MSIV.	Minimum steam line piping volume from RPV to inboard MSIV.	Minimum steam line piping volume from inboard MSIV to turbine stop valve.

Table 12b

MSIV Leakage Pathway (LOCA) RADTRAD Compartments (Continued)					
Compartment No.:	5	6	7	8	9
Compartment Name:	(Node 2) Outboard MSL "D" Volume	HP Condenser	Environment	Control Room	"Hold"
Type:	Other	Other	Environment	Control Room	Other
Volume:	1206 ft^3	54,750 ft^3	N/A	126,000 ft^3	N/A
Source Term Fraction:	0.0	0.0	0.0	0.0	0.0
Compartment Features:	None	None	N/A	Recirculation Filter: CREFAS Radiation mode: 2475 cfm. Manual start at 30 min or less HEPA: 99% Charcoal: 80% Chlorine mode: 3000 cfm. Auto Starts HEPA: 99% Charcoal: 80%	None
Comments:	Minimum steam line piping volume from outboard MSIV to turbine stop valve.	HP Condenser shell free-air volume. No credit taken for deposition in condenser or on substantial surface of condenser tubing.			This compartment holds Primary Containment leakage to prevent "double-counting" of releases.

Table 12c

MSIV Leakage Pathway (LOCA) RADTRAD Active Transfer Pathways				
Pathway Number:	1	2	3	4
From Compartment:	Containment	Containment	Containment	Inboard MSL "A" piping (node 1)
To Compartment:	Inboard piping MSL "A" volume (node 1)	Inboard piping MSL "D" volume (node 1)	"Hold"	Outboard MSL "A" piping (node 2)
Transfer Mechanism:	Filter	Filter	Air leakage	Filter
Transfer Mechanism Details:	Flow rates/Efficiencies: 0-24hours, 0.834cfm Aerosol: 0% Elemental: 0% Organic: 0% 24 to 720hours, 0.417cfm Aerosol: 0% Elemental: 0% Organic: 0%	Flow rates/Efficiencies: 0-24hours, 0.834cfm Aerosol: 0% Elemental: 0% Organic: 0% 24 to 720hours, 0.417cfm Aerosol: 0% Elemental: 0% Organic: 0%	Leakage rate: 0-24hrs, 0.5% per day 24-720hrs, 0.25% per day	Flow rate: 0-24hrs, 0.834cfm Aerosol: 96.8% Elemental: 39.31% Organic: 0% 24-96hrs, 0.417cfm; Aerosol: 98.38% Elemental: 75.11% Organic: 0% 96-720hrs, 0.417cfm Aerosol: 98.38% Elemental: 95.05% Organic: 0%
Comments:	50% of first day value based on a conservatively assumed containment pressure at 24 hours. There is no filtration of leakage out of Containment into this first piping node.	50% of first day value based on a conservatively assumed containment pressure at 24 hours. There is no filtration of leakage out of Containment into this first piping node.	Flow from PC leakage other than through MSIVs is sent to HOLD compartment to prevent dose contribution in this run, as it is not a contributor to this MSIV release model.	Flow rates were derived from the table "Determination of MSL Decontamination Factors Due to Iodine Deposition."

Table 12d

MSIV Leakage Pathway (LOCA) RADTRAD Active Transfer Pathways (Continued)				
Pathway Number:	5	6	7	8
From Compartment:	Inboard MSL "D" Piping Volume (node 1)	Outboard MSL A piping volume (node 2)	Outboard MSL D piping volume (node 2)	Condenser
To Compartment:	Outboard MSL "D" Piping (node 2)	Condenser	Condenser	Environment
Transfer Mechanism:	Filter	Filter	Filter	Filter
Transfer Mechanism Details:	Flow rates/Efficiencies: 0-24hr, 0.834 cfm: Aerosol: 96.8% Elemental: 39.3% Organic: 0% 24-96 hr, 0.417 cfm: Aerosol: 98.38% Elemental: 75.11% Organic: 0% 96-720 hr, 0.417 cfm Aerosol: 98.38% Elemental: 95.05% Organic: 0%	Flow rates/Efficiencies: 0-24hr, 0.834 cfm: Aerosol: 98.47% Elemental: 38.17% Organic: 0% 24-96 hr, 0.417 cfm: Aerosol: 77.00% Elemental: 75.11% Organic: 0% 96-720 hr, 0.417 cfm Aerosol: 96.58% Elemental: 95.05% Organic: 0%	Flow rates/Efficiencies: 0-24hrs, 4.017 cfm; Aerosol: 98.51% Elemental: 38.63% Organic: 0% 24-96 hrs, 1.716 cfm; Aerosol: 99.36% Elemental: 77.43% Organic: 0% 96-720 hrs, 1.302 cfm Aerosol: 99.51% Elemental: 96.64% Organic: 0%	Flow rates/Efficiencies: 0-24hr, 4.577cfm Aerosol: 98.55% Elemental: 98.89% Organic: 0% 24 to 96hr, 2.289cfm Aerosol: 99.27% Elemental: 99.44% Organic: 0% 96-720hr, 2.289cfm Aerosol: 99.27% Elemental: 99.44% Organic: 0% :
Comments:	Flow rates were derived from the table "Determination of MSL Decontamination Factors Due to Iodine Deposition."	Flow rates per attached spreadsheet	Flow rates per attached spreadsheet	Three different cases run. No condenser credit, 10% of condenser tube credit, 100% tube credit. See attached spreadsheet

Table 12e

MSIV Leakage Pathway (LOCA) RADTRAD Active Transfer Pathways (Continued)				
Pathway Number:	**9**	**10**	**11**	
From Compartment:	Environment	Environment	Control Room	
To Compartment:	Control Room	Control Room	Environment	
Transfer Mechanism:	Filter	Filter	Filter (Exhaust)	
Transfer Mechanism Details:	Rad Mode Flow rate: 0-30min, 2100cfm (0% for all); 30min-720hrs, 525cfm 99% HEPA; 80% Charcoal Chlorine Mode Flow Rate: 0 cfm (No filtered intake)	Flow Rate: 0-720hrs, 525cfm for Rad & Chlorine Modes Efficiencies: 0% for all	Rad. Mode Flow Rate: 0.30min, 2625cfm, 30min-720hrs, 1050cfm Chlorine Mode Flows: 0-720hrs, 525cfm Efficiencies: 100% for all iodines for all periods	
Comments:	Flow rates were derived from the table "Determination of MSL Decontamination Factors Due to Iodine Deposition."	Unfiltered Inleakage	Flows balance inputs for each time period. 100% Filtration prevents double counting of iodine release.	

Table 12f

MSIV Leakage Pathway (LOCA) RADTRAD Dose Receptor Locations			
Dose Location Number:	**1**	**2**	**3**
In Compartment:	Control Room	EAB (Distance: 731 Meters)	LPZ (Distance: 2043 Meters)
Breathing Rate (m^3/sec):	0 to 720 hrs: 3.5E-04	0 to 8 hrs: 3.5E-04 8 to 24 hrs: 1.8E-04 24 to 720 hrs: 2.3E-04	0 to 8 hrs: 3.5E-04 8 to 24 hrs: 1.8E-04 24 to 720 hrs: 2.3E-04
Occupancy Fractions:	0 to 24 hrs: 1.0 24 to 96 hrs: 0.6 24 to 720 hrs: 0.4	Highest 2-hr window: 1.0	0 to 720 hrs: 1.0
X/Q (sec/m^3):	0 to 2 hr: 6.88E-03 2 to 8 hr: 5.17E-03 8 to 24 hr: 2.04E-03 1 to 4 day: 1.29E-03 4 to 30 day: 9.63E-04	10.4 to 12.4 hrs: 3.18E-04 (Highest two hours)	0 to 8 hr: 5.79E-05 8 to 24 hr: 4.10E-05 1 to 4 day: 1.95E-05 4 to 30 day: 6.68E-06

Table 12g

MSIV Leakage Pathway (LOCA) RADTRAD Source Terms & Dose Conversion Factors		
Nuclide Inventory:	3527 MWt LGS Specific NIF for 60 MACCS isotopes (See Table 1 of this document)	C0-58 & Co-60 values are RADTRAD default values.
Release Fractions & Timing:	RADTRAD Standard BWR-DBA values, No delay	
Dose Conversion Factors:	RADTRAD Library of FGR 11&12 values for 60 MACCS isotopes	
Decay & Daughter Products:	Enabled - Decay / Daughter Products considered	
Iodine Chemical Fractions:	NUREG-1465 based Iodine Chemical Form fractions Aerosol: 0.95 Elemental: 0.0485 Organic: 0.0015	

Table 13a

ECCS Leakage Pathway (LOCA) RADTRAD Compartments					
Containment Number:	**1**	**2**	**3**	**4**	**5**
Name:	ECCS Fluid	Reactor Enclosure	Environment	Control Room	SGTS Node
Type:	Other	Other	Environment	Control Room	Control Room
Volume:	959,900 gallons	900,000 ft^3	N/A	126,000 ft^3	1 ft^3
Source Term Fraction:	1.0	0.0	N/A	0.0	0.0
Compartment Features:	None	Recirc. Filter: RERS Flow Rate: 0-15.5min, 30,000 cfm, no filter credit 15.5min-720hr: 30,000cfm at 70% efficiency for aerosols and all iodines.	N/A	Recirc. Filter: Rad Mode CREFAS Flow Rate: 0-30min, no filter credit; 30min-720hrs, 2475cfm at 99% for aerosols, 80% for all iodines. Chlorine Mode: 3000cfm for all periods at 99% for aerosols, 80% for all iodines.	None
Comments:	118,655 ft^3 Supp. Pool minimum + 9,663 ft^3 reactor coolant = 959,900 gallons				Compartment used to model condition of SGTS filter train in series with RERS filter train.

Table 13b

ECCS Leakage Pathway (LOCA) RADTRAD Active Transfer Pathways				
Pathway Number and Name:	**1 ECCS Fluid to RE**	**2 CR Filtered Intake**	**3 CR Exhaust**	**4 CR Unfiltered Inleakage**
From Compartment:	ECCS Fluid (1)	Environment (3)	Control Room (4)	Environment (3)
To Compartment:	Reactor Enclosure (2)	Control Room (4)	Environment (3)	Control Room (4)
Transfer Mechanism:	Filter Flow Rate: 5gpm	Filter (CREFAS)	Filter	Filter
Transfer Mechanism Details:	98.57% for iodines. The "filter is used to simulate a 1.43% flashing fraction.	CREFAS Flow rates/Efficiencies: 0-30 min, 2100 cfm: Aerosol: 0% Elemental: 0% Organic: 0% 30 min-720 hours, 525 cfm – Aerosol: 99% Elemental: 80% Organic: 0%	Rad Mode Flow rates/Efficiencies: 0-30 min, 2625cfm; Aerosol: 100% Elemental: 100% Organic: 100% 30 min-720 hours, 1050 cfm Aerosol: 100% Elemental: 100% Organic: 100%	Flow rate: 0-720 hours, 525 cfm for Radiation Mode and Chlorine Mode CR. Aerosol: 0% Elemental: 0% Organic: 0%
Comments:	Since the gallon volume value was entered as the ECCS Fluid volume, entering a gallon/minute value here is correct.	Models the normal CR filtered intake flow rate. Filter efficiencies are 99% for aerosol (HEPA) and 80% for elemental and organic iodines (charcoal).	This equals the total flow that was taken into the CR volume, which includes inleakage (525 + 2100, for manual initiation time; 525 + 525 cfm, after initiation). Chlorine Mode Flow Rate – 0-720 hours, 525 cfm – which balances the inleakage.	Unfiltered inleakage flow rate.

Table 13c

ECCS Leakage Pathway (LOCA) RADTRAD Active Transfer Pathways (Continued)				
Pathway Number and Name:	**5** **RE Exhaust to SGTS**	**6** **SGTS to Environment**		
From Compartment:	Reactor Enclosure (2)	SGTS Node (5)		
To Compartment:	SGTS Node (5)	Environment (3)		
Transfer Mechanism:	Filter (RERS)	Filter (SGTS)		
Transfer Mechanism Details:	Flow rate: 0-1 minute, 9.00E6 cfm Flow rate 1 minute to 15.5 minutes, 3000 cfm for drawdown period. Flow rate for 15.5 minutes until 720 hours, 2500 cfm. RERS Filter Efficiency is 70% for aerosol (HEPA), elemental, and organic iodines (Charcoal), for accident duration.	Flow rate: 0-1 minute, 9.00E6 cfm Flow rate 1 minute to 15.5 minutes, 3000 cfm for drawdown period. Flow rate for 15.5 minutes until 720 hours, 2500 cfm SGTS Filter Efficiency 0% during 0 to 15.5 minute drawdown period. 97.5% for aerosol (HEPA), elemental, and organic iodines (Charcoal)		
Comments:	0-1 minute, 9.00E6 cfm – 10 air changes per minute to simulate last minute of RERS startup (first two minutes are before gap release phase). This is an intermediary pathway to effectively model SGTS filtration in series.	0-1 minute, 9.00E6 cfm – 10 air changes per minute to simulate last minute of RERS startup (first two minutes are before gap release phase).		

Table 13d

ECCS Leakage Pathway (LOCA) RADTRAD Dose Receptor Locations			
Dose Location Number:	**1**	**2**	**3**
In Compartment:	**Control Room**	**EAB** (Distance: 731 Meters)	**LPZ** (Distance: 2043 Meters)
Breathing Rate (m^3/sec):	0 to 720 hrs: 3.5E-04	0 to 8 hrs: 3.5E-04 8 to 24 hrs: 1.8E-04 24 to 720 hrs: 2.3E-04	0 to 8 hrs: 3.5E-04 8 to 24 hrs: 1.8E-04 24 to 720 hrs: 2.3E-04
Occupancy Fractions:	0 to 24 hrs: 1.0 24 to 96 hrs: 0.6 24 to 720 hrs: 0.4	Highest 2-hr window: 1.0	0 to 720 hrs: 1.0
X/Q (sec/m^3):	0 to 2 hr: 6.88E-03 2 to 8 hr: 5.17E-03 8 to 24 hr: 2.04E-03 1 to 4 day: 1.29E-03 4 to 30 day: 9.63E-04	3.2-5.2 hrs: 3.18E-04 (Highest two hours)	0 to 8 hr: 5.79E-05 8 to 24 hr: 4.10E-05 1 to 4 day: 1.95E-05 4 to 30 day: 6.68E-06

Table 13e

ECCS Leakage Pathway (LOCA) RADTRAD Source Terms & Dose Conversion Factors		
Nuclide Inventory:	3527 MWt LGS Specific NIF for 60 MACCS isotopes	Iodines only for ECCS leakage. I-131: 0.2687E+05 I-132: 0.3881E+05 I-133: 0.5556E+05 I-134: 0.6165E+05 I-135: 0.5192E+05
Release Fractions & Timing:	RADTRAD Standard BWR-DBA values, No delay	
Dose Conversion Factors:	RADTRAD Library of FGR 11&12 values for 60 MACCS isotopes	
Decay & Daughter Products:	Enabled - Decay / Daughter Products considered	
Iodine Chemical Fractions:	NUREG-1465 based Iodine Chemical fractions Aerosol: 0.00 Elemental: 0.97 Organic: 0.03	Since iodines produced by flashing, treated as 97% elemental and 3% organic.

RADTRAD LOCA Results Summary

The following tables are tabulations of doses from the various activity leakage pathways associated with the analyzed DBA-LOCA. The first table shows results for the Radiation Isolation Mode, while the second shows the Chlorine Isolation Mode dose consequences.

Table 14a
Radiation Isolation Mode Runs

Activity Leakage Pathway	Dose Location			RADTRAD Run Output Filename
	Control Room (rem TEDE)	EAB (rem TEDE)	LPZ (rem TEDE)	
Primary Containment Leakage	2.9363E+00	8.7994E-01	1.1356E+00	LGS LOCA PC Leak - 70% RERS Filt Credit - 97,5% SGTS Filter - 99% Aerosol, 80% E and O CR Filter - 30min CREFAS Delay - Rad Mode 525cfm Unfilt Inleak.o0
MSIV Leakage (No Condenser Tube Credit)	6.9688E-01	1.6823E-02	1.0943E-01	LGS LOCA MSIV Leak - HP Condenser Rem Credit (No Tubes) - 80% CR Charcoal Eff with 30min CREFAS Delay - Rad Mode 525cfm Unfilt Inleak.o0
MSIV Leakage (10% Condenser Tube Credit)	4.4196E-01	1.6089E-02	9.4223E-02	LGS LOCA MSIV Leak - HP Condenser Rem Credit (10th of Tubes - Organic Removal) - 80% CR Charcoal Eff with 30min CREFAS Delay - Rad Mode 525cfm Unfilt Inleak.o0
MSIV Leakage (100% Condenser Tube Credit)	2.3471E-01	1.5249E-02	8.1909E-02	LGS LOCA MSIV Leak - HP Condenser Rem Credit (All of Tubes - Organic Removal) - 80% CR Charcoal Eff with 30min CREFAS Delay - Rad Mode 525cfm Unfilt Inleak.o0
ECCS Leakage	4.6595E-02	1.0167E-03	2.8091E-03	LGS LOCA ECCS Leak - 70% RERS Filt Credit - 97,5% SGTS Filter - 99% Aerosol, 80% E and O CR Filter - 30min CREFAS Delay - Rad Mode 525cfm Unfilt Inleak.o0

Table 14b
Chlorine Isolation Mode Runs

Activity Leakage Pathway	Dose Location			RADTRAD Run Output Filename
	Control Room (rem TEDE)	EAB (rem TEDE)	LPZ (rem TEDE)	
Primary Containment Leakage	2.4950E+00	8.7994E-01	1.1356E+00	LGS LOCA PC Leak - 70% RERS Filt Credit - 97,5% SGTS Filter - 99% Aerosol, 80% E and O CR Filter - Chlorine Mode 525cfm Unfilt Inleak.o0
MSIV Leakage (No Condenser Tube Credit)	6.1828E-01	1.6823E-02	1.0943E-01	LGS LOCA MSIV Leak - HP Condenser Rem Credit (No Tubes) - 80% CR Charcoal Eff - Chlorine Mode 525cfm Unfilt Inleak.o0
MSIV Leakage (10% Condenser Tube Credit)	3.9826E-01	1.6089E-02	9.4223E-02	LGS LOCA MSIV Leak - HP Condenser Rem Credit (10th of Tubes - Organic Removal) - 80% CR Charcoal Eff - Chlorine Mode 525cfm Unfilt Inleak.o0
MSIV Leakage (100% Condenser Tube Credit)	2.1939E-01	1.5249E-02	8.1909E-02	LGS LOCA MSIV Leak - HP Condenser Rem Credit (All of Tubes - Organic Removal) - 80% CR Charcoal Eff - Chlorine Mode 525cfm Unfilt Inleak.o0
ECCS Leakage	3.9572E-02	1.0167E-03	2.8091E-03	LGS LOCA ECCS Leak - 70% RERS Filt Credit - 97,5% SGTS Filter - 99% Aerosol, 80% E and O CR Filter - Chlorine Mode 525cfm Unfilt Inleak.o0

Table 15
RADTRAD Bounding LOCA Results By Pathway

LOCATION			DOSE CONTRIBUTOR
Control Room (Rem TEDE)	EAB (Rem TEDE)	LPZ (Rem TEDE)	
2.936	0.880	1.136	Filtered Primary Containment (PC) Leakage (unfiltered for 15.5 minutes, SGTS filtered thereafter) [100% of L_A], Control Room in Rad Mode
0.697	0.017	0.109	MSIV Leakage, without LCS but with piping deposition credit. *[200 scfh total all MS lines, 100 scfh max/line]*
0.047	0.001	0.003	ECCS Leakage in Secondary Containment (SC) *[5 gpm]*
0.34			Gamma Shine to Control Room General Area
4.02	**0.90**	**1.25**	**Total Calculated Value**
5	**25**	**25**	**Regulatory Limits**