PSEG Nuclear LLC
P.O. Box 236, Hancocks Bridge, New Jersey 08038-0236

JUL 2 9 2002

LR-N02-0244
LCR S02-010

PSEG
Nuclear LLC

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

**REQUEST FOR CHANGES TO TECHNICAL SPECIFICATIONS
REFUELING OPERATIONS - RELAXATION OF REQUIREMENTS APPLICABLE DURING MOVEMENT OF IRRADIATED FUEL
SALEM NUCLEAR GENERATING STATION, UNITS 1 AND 2
FACILITY OPERATING LICENSES DPR-70 AND DPR-75
DOCKET NOS. 50-272 AND 50-311**

Gentlemen:

Pursuant to 10 CFR 50.90, PSEG Nuclear LLC (PSEG) hereby requests a revision to Appendix A of the Technical Specifications for the Salem Nuclear Generating Station, Units 1 and 2. In accordance with 10CFR50.91(b)(1), a copy of this submittal has been sent to the State of New Jersey.

The purpose of this License Amendment Request is to provide flexibility in scheduling outage tasks and to modify unnecessarily restrictive containment closure and Fuel Handling Area Ventilation requirements. TSTF-51, Revision 2 is used as guidance in the preparation of this change request. This request revises the requirements for containment closure associated with the equipment hatch and personnel airlocks during Core Alterations and movement of irradiated fuel within the containment. This proposed change would allow the equipment hatch and the personnel airlocks to remain open during fuel movement in the containment provided administrative controls are developed and implemented, ensuring the closure of the equipment hatch and personnel airlock following a fuel handling accident within the containment building. Appropriate Bases changes are included to reflect the proposed changes.

The basis for the proposed changes is the reanalysis of the limiting design basis Fuel Handling Accident (FHA) using the guidelines contained in 10CFR 50.67 and Regulatory Guide 1.183, Alternative Source Term. The proposed changes are based on NRC approval of selective implementation of Alternative Source Term methodology for the Salem Units 1 & 2 Fuel Handling Accident submitted by PSEG, letter LR-N02-0231 dated June 28, 2002. The Salem Units 1 & 2 UFSAR will be updated to reflect the enclosed amendment and analysis following NRC approval.

PSEG has evaluated the proposed changes in accordance with 10CFR50.91(a)(1), using the criteria in 10CFR50.92(c), and has determined this request involves no significant hazards considerations. An evaluation of the requested changes is provided in Attachment 1 to this letter. The marked up Technical Specification pages affected by the proposed changes are provided in Attachment 2.

PSEG requests approval of the proposed License Amendment by September 12, 2002 to be implemented within 30 days. This will support the start of Salem Nuclear Generating Station Unit 1, fifteenth (1R15) refueling outage that is scheduled for October 12, 2002.

A001

JUL 2 9 2002

Below is a summary of the commitments made in this submittal:

1. Develop and implement administrative controls to ensure that containment closure is accomplished within 1 hour following a Fuel Handling Accident within Containment.

Should you have any questions regarding this request, please contact Mr. Brian Thomas at 856-339-2022.

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

Sincerely,

Executed on 7/29/02

D. F. Garchow
Vice President – Operations

Attachments (2)

C Mr. H. J. Miller, Administrator - Region I
U. S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

U. S. Nuclear Regulatory Commission
ATTN: Mr. R. Fretz, Project Manager - Salem
Mail Stop 08B1
Washington, DC 20555-0001

USNRC Senior Resident Inspector - Salem (X24)

Mr. K. Tosch, Manager IV
Bureau of Nuclear Engineering
PO Box 415
Trenton, NJ 08625

SALEM NUCLEAR GENERATING STATION, UNITS 1 AND 2
FACILITY OPERATING LICENSES DPR-70 AND DPR-75
DOCKET NOS. 50-272 AND 50-311

EVALUATION OF REVISIONS TO THE TECHNICAL SPECIFICATIONS REFUELING OPERATIONS – RELAXATION OF REQUIREMENTS APPLICABLE DURING MOVEMENT OF IRRADIATED FUEL

REQUEST FOR CHANGE TO TECHNICAL SPECIFICATIONS RELAXATION OF REQUIREMENTS APPLICABLE DURING MOVEMENT OF IRRADIATED FUEL

Table of Contents

**REQUEST FOR CHANGE TO TECHNICAL SPECIFICATIONS
RELAXATION OF REQUIREMENTS APPLICABLE DURING
MOVEMENT OF IRRADIATED FUEL**

1. DESCRIPTION

This letter is a request to amend Operating Licenses 50-272 and 50-311 for Salem Units 1 & 2 respectively.

The purpose of this License Amendment Request is to provide flexibility in scheduling outage tasks and to modify unnecessarily restrictive containment closure and Fuel Handling Area Ventilation System requirements. TSTF-51, Revision 2 is used as guidance.

The proposed amendment would allow movement of sufficiently decayed irradiated fuel within the containment building with the equipment hatch, personnel air locks and containment penetrations open. Operation of the Containment Purge Exhaust System (CPES) is not required during movement of sufficiently decayed fuel provided that the Auxiliary Building Ventilation System is in operation and taking suction from the containment atmosphere via the open containment airlocks. The automatic isolation of the Containment Purge System, resulting from the containment radiation monitors, is being relaxed in order to maintain monitoring of post-Fuel Handling Accident (FHA) containment activity until the closure of the containment penetrations is accomplished. Eliminating the automatic actuation allows the containment purge and exhaust system to remain in operation during refueling when the equipment hatch is open. It also assures that negative pressure in containment is maintained by either the containment purge system or the auxiliary building ventilation system with suction from the containment atmosphere via the open personnel airlocks. This action complies with the requirements listed in General Design Criteria (GDC 64), which requires monitoring the reactor containment atmosphere following postulated accidents. Manual capability for isolation of the containment purge system, if required, would remain unchanged. The amendment also would allow movement of irradiated fuel assemblies within the Fuel Handling Building with the Fuel Handling Area Ventilation System (FHAVS) in operation with no credit taken for filtration.

2. PROPOSED CHANGE

The proposed changes to the Technical Specifications would revise the following sections:

a. Technical Specification Section Definition 1.8, previously not used, is being used for the revised definition of CORE ALTERATIONS, previously Definition 1.9. The proposed definition is identical to the Standard Technical Specifications, NUREG 1431 and better reflects the assumptions used for the (FHA) as described in UFSAR Section 15.4.6. The analyzed event is the drop of a fuel assembly and the subsequent rupture of the cladding of all the fuel rods in the assembly. Other off-normal events described during CORE ALTERATIONS are bounded by the FHA.

b. The current Definition 1.9a, Core Operating Limits Report, is relocated to Definition 1.9, previously used by the CORE ALTERATIONS definition. This is considered an administrative change and no relaxation of current requirements is proposed.

c. Deletion of the requirements for automatic isolation of the Containment Purge System during fuel movement within containment. TS Table 3.3-6, Item 2.a.1.a and 2.a.2.a are modified allowing the continuous monitoring of the containment atmosphere until containment closure is accomplished following a FHA. The containment ventilation provided by either the Containment Purge System or the Auxiliary Building Ventilation System (with personnel airlocks open) maintains outside airflow into containment and minimizes potential releases out of the open equipment hatch. The Auxiliary Building Ventilation System exhausts via the plant vent which, has installed radiation monitors providing for the continuous monitoring of Fuel Handling Building and containment atmosphere following a FHA with containment hatch and personnel airlocks open.

The applicable surveillances of Table 4.3-3, items 2.a.1.a and 2.a.2.a are modified to reflect the relaxations of automatic isolation of containment purge to allow for continuous monitoring of containment atmosphere following the FHA. In addition to the deletion of the Limited Condition for Operation (LCO) and Surveillance Requirement (SR) for the air particulate activity radiation monitor for containment purge & pressure vacuum relief isolation in Mode 6, this change also corrects an oversight during the submittal and approval of Amendment 79 (Unit 1) and Amendment 53 (Unit 2). Amendment 79/53 was issued on April 10, 1987 to revise accident monitoring instrumentation and radiation monitoring instrumentation requirements. As part of this change, the requirements to have the containment air particulate monitor (R11A) operable in Modes 1, 2, 3, and 4 was deleted from TS Table 3.3-6 Item 2.a.2.a requiring the R11A only to be operable in Mode 6 for purge & pressure vacuum relief isolation. Although the mode applicability was revised for LCO Table 3.3-6 to eliminate the Mode 1-4 requirement, an oversight was made in the request in that the corresponding surveillance requirement TS Table 4.3-3 Item 2.a.2.a was not revised to reflect the above change as well. Currently there is a mis-match between the surveillance table and the LCO table. This proposed TS change is correcting this oversight by deleting the requirement to perform the surveillance requirement for TS Table 4.3-3 item 2.a.2.a in modes 1-4.

d. The proposed change would revise Limiting Condition for Operation 3.9.4, Containment Building Penetrations, to allow the containment equipment hatch and the personnel airlocks to be open during movement of irradiated fuel assemblies within containment, provided they are capable of being closed within 1 hour under administrative controls. A new surveillance Requirement (SR) would be added to verify the capability to install the containment equipment hatch within the required time, if the hatch is open, prior to the start of irradiated fuel movement within the containment building.

e. The Bases for this TS are being revised to define the capability to close containment following a FHA and describe the administrative controls required to comply with LCO implementation.

f. Deletion of Core Alterations from the Applicability and Actions in TS 3/4.7.6, Control Room Emergency Air Conditioning System. The analyzed event is the drop of a fuel assembly and the subsequent rupture of the cladding of all the fuel rods in the assembly. Other off-normal events described during CORE ALTERATIONS are bounded by the FHA.

f. Relocating TS 3/4 3.9.9 Refueling Operations, Containment Purge and Pressure-Vacuum Relief Isolation System to be combined with TS 3.9.4, Containment Building Penetrations. The purpose of this relocation is two-fold; 1) Implementing consistency with the Improved Technical Specifications and, 2) expanding and clarifying the requirements for the Containment Purge System during movement of irradiated fuel assemblies within the containment.

g. Deletion of the SFP Filtration System and it's associated surveillances from TS 3/4 3.9.12 Fuel Handling Area Ventilation. In addition, Action a. is being revised to delete the requirement for suspending crane operation with loads over the storage pool. This action is not required since the FHA considers the drop of a fuel assembly to be the limiting scenario. The Control of Heavy Loads Program contains restrictions that do not allow movement of loads exceeding 2200 pounds (weight of a fuel assembly and associated handling device) over the spent fuel pool. Additionally, TS 3/4 9.7 prohibits travel of loads in excess 2200 pounds over fuel assemblies in the storage pool.

The marked up Technical Specification pages are included in Attachment 2.

3. BACKGROUND

In December 1999, the NRC issued a new regulation, 10 CFR 50.67, which provides a means for power reactors to replace their existing accident source term with Alternative Source Term (AST). Regulatory Guide 1.183 provides guidance for the implementation of alternate source terms. 10 CFR 50.67 requires licensees seeking to use AST to apply for a license amendment and include an evaluation of the consequences of the affected design-basis accidents. PSEG submitted a License Change Request via letter LR-N02-0231, dated June 28, 2002, which addresses these requirements by proposing the selective application of AST described in RG 1.183 in evaluating the radiological consequences of a FHA. As part of the implementation of the AST, the TEDE acceptance criterion of 10 CFR 50.67 (b)(2) replaces the previous whole body and thyroid dose guidelines of 10CFR 100.11 and 10 CFR 50, Appendix A, GDC 19 for the FHA. This LCR follows in part the guidance provided by TSTF-51, rev 2.

The proposed amendment is similar to amendments issued by the NRC (Reference 8m) allowing containment closure relaxations during fuel movement by applying the dose guidelines of 10 CFR 50.67 and Regulatory Guide 1.183, Alternative Source Term.

4. TECHNICAL ANALYSIS

Each containment at Salem Nuclear Station is equipped with two personnel air locks, an equipment hatch, and a Containment Purge System. Technical Specification (TS) 3.9.4 requires that during Core Alterations or movement of irradiated fuel assemblies within containment, the associated equipment hatch be closed and secured with at least four bolts and at least one of the two doors in each personnel air lock be closed. TS 3.9.4 also requires that each penetration providing direct access from the containment atmosphere to the environment either be closed by an isolation valve, blind flange, or equivalent. The applicable design basis event is the Fuel Handling Accident inside containment. During Core Alterations or movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident, involving dropping of a spent fuel assembly resulting in the rupture of the cladding of all the fuel rods in the assembly. In the re-analysis of this design basis event, airborne activity resulting from the initiating event (FHA) is assumed to be released to the environment over a 2-hour time period via the open equipment hatch and the plant vent not taking credit for filtration.

Following reactor shutdown, decay of the short-lived fission products greatly reduces the fission product inventory present in irradiated fuel. Following sufficient decay time, the primary success path for mitigating the FHA no longer includes the functioning of the active containment systems. Therefore, water level and decay time are the primary success paths for mitigating a FHA.

CORE ALTERATIONS, as presently defined in the Technical Specifications, is the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. As described in TSTF-51, Revision 2, accidents postulated to occur during core alterations include inadvertent criticality, fuel handling accident, and the loading of a fuel assembly or control component in an incorrect location. Generically, it was concluded that of these off normal occurrences, only the fuel handling accident results in cladding damage and potential radiological release. Consequently, it is being proposed that the definition of CORE ALTERATIONS be modified as shown in the TS mark-up. This change is consistent with the Improved Technical Specifications (ITS).

DOSE CALCULATIONS

Calculations were performed to determine atmospheric dispersion factors (χ/Qs) at the Salem Nuclear Generating Station (SNGS) control room (CR) air intake due to the FHA releases from the Containment Equipment Hatch and Plant Vent (PV).

Analyses were performed to determine the Exclusion Area Boundary (EAB), Low Population Zone (LPZ) and Control Room (CR) doses due to a FHA occurring in the containment building with containment equipment hatch and Personnel Locks opened. The FHA analysis was performed using the AST, the guidance in the Regulatory Guide 1.183, Appendix B, and TEDE

dose criteria. The results of these calculations are within the regulatory acceptance criteria and are summarized below.

FUEL HANDLING ACCIDENT ANALYSIS DOSE RESULTS

Results	CONTROL ROOM TEDE (rem)	EAB TEDE (rem)	LPZ TEDE (rem)
Salem FHA in Containment Building	2.93E+00	4.15E+00	5.94E-02
Salem FHA in Fuel Handling Building	1.90E+00	4.15E+00	5.93E-02
Regulatory Acceptance Criteria	5.00E+00	6.30E+00	6.30E+00

FUEL HANDLING BUILDING

The Fuel Building of each nuclear unit at Salem Units 1 and 2 Nuclear Station is equipped with a Fuel Handling Area Ventilation System (FHAVS) with two exhaust and one supply fans. TS 3/4 9.12 presently requires that the FHAVS be operable and in operation while irradiated fuel is in the storage pool. There is one germane design basis event, the Fuel Handling Accident in the Fuel Building.

For movement of irradiated fuel, within the Fuel Handling Building, operation of the FHAVS ventilation is credited. TS 3/4 9.12 requires that the FHAVS be operable. If the FHAVS is inoperable, movement of irradiated fuel assemblies within the Fuel Building is to be immediately suspended. The LCO applicability is being modified to reflect the conditions under which a FHA could occur. The dose calculations performed, using AST, result in acceptable doses without crediting the FHAVS Filtration System thus; the Surveillances Requirements associated with the Filtration System are deleted. The Filtration System is not required for operability of the FHVAS.

Revised analyses of the FHA have been performed in support of this License Change Request. The analyses were performed pursuant to Regulatory Guide 1.183 using the Alternative Source Term (AST) methodology. The source terms were calculated pursuant to R.G. 1.183. Releases from the reactor cavity or spent fuel pool were modeled pursuant to R.G. 1.183. The results were included as part of the selective application of Alternative Source Term applicable to the FHA, in the PSEG submittal LR-N02-0231, dated June 28, 2002.

ADMINISTRATIVE CONTROLS

NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants", provides guidance to reduce resulting doses and to avoid unmonitored releases. This document recommends the development of contingency methods to promptly close the containment penetrations following Fuel Handling Accidents 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.

Even though containment closure is not credited in the dose calculations, PSEG proposes to develop administrative controls to provide reasonable assurance that containment closure, as a defense-in-depth measure, can be reestablished promptly following a Fuel Handling Accident to limit the releases much lower than assumed in the dose calculations. It is not necessary that the specific actions contained in these administrative controls be contained in the TS Bases. Plant procedures and changes thereto fall under the requirements of 10 CFR 50.59. The proposed administrative controls are shown below:

Containment Building Closure:

The following requirements shall be maintained to ensure defense-in-depth. Closure Controls are in effect whenever the affected Containment is open during operations within containment involving movement of irradiated fuel assemblies. The definition of an open containment penetration is a penetration that provides direct access from the containment atmosphere to the outside environment.

1. The equipment necessary to implement containment closure shall be appropriately staged prior to maintaining any containment penetration open including airlock doors and the containment equipment hatch.

2. Hoses and cables running through any open penetration, airlock, or equipment hatch should be configured to facilitate rapid removal in the event that containment closure is required.

3. The containment personnel airlock may be open provided the following conditions exist:

   a. One door in each airlock is capable of being closed.
   b. Hoses and cables running through the airlock shall employ a means to allow safe, quick disconnection or severance.
   c. The airlock door is not blocked in such a way that it cannot be expeditiously closed. Protective covers used to protect the seals/airlock doors or devices to keep the door open/supported do not violate this provision.
   d. Personnel are designated and available with the responsibility for expeditious closure (within 1 hour) of at least one door on the containment airlocks following the FHA.

4. The containment equipment hatch may be open provided the following conditions exist:

   a. The containment equipment hatch is capable of being closed or an equivalent closure device is available and can be closed within 1 hour.
   b. Hoses and cables running through the equipment hatch shall employ a means to allow safe, quick disconnection or severance.
   c. The equipment hatch is not blocked in such a way that it cannot be expeditiously closed. Protective covers used to protect the seals/equipment hatch or devices to keep the hatch open supported do not violate this provision.
   d. Necessary tools to install the equipment hatch and tighten at least four equipment hatch closure bolts are available or other methods to close the equipment hatch opening (i.e., restrict air flow out of the containment), such as a refueling hatch closure device, is staged at the work area along with the necessary installation tools.
   e. A sufficient number of personnel are designated and available with the responsibility for expeditious closure (within 1 hour) of the containment equipment hatch opening following the FHA.

5. If containment closure would be hampered by an outage activity, compensatory actions will be developed.

6. Either the Containment Purge system or the Auxiliary Building Ventilation System with suction from the containment atmosphere, with associated radiation monitoring will be available whenever movement of irradiated fuel is in progress in the containment building and the equipment hatch is open. If for any reason, this ventilation requirement can not be met, movement of fuel assemblies within the containment building shall be discontinued until the flow path(s) can be reestablished, or close the equipment hatch (or an equivalent closure device is installed) and personnel airlocks. Periodic verification (once per shift) of this administrative control will ensure that air flow will be directed from containment to the Auxiliary Building or the Plant Vent where continuous monitoring will be in effect thus minimizing the potential for unmonitored releases out the open containment hatch following the FHA.

7. Personnel responsible for Containment Building Closure shall be trained and knowledgeable in using the procedure for executing containment closure. Walkdowns should be considered to demonstrate the closure capability including compensatory actions in the event of loss of electrical power.

Fuel Handling Building Closure:

The following requirements shall be maintained to ensure defense-in-depth. Closure Controls are in effect during operations within the Fuel Handling Building involving movement of irradiated fuel assemblies.

1. The Fuel Handling Building doors shall be maintained closed except for normal entry and exit unless a designated person is available to close the open door(s) should a FHA occur within the Fuel Handling Building.
2. The FHAVS, with associated radiation release monitoring will be available for the release flow path. If for any reason operation of the fuel handling area ventilation system flow path must be discontinued and the fuel building is open to the outside environment, fuel movement within the Fuel Handling Building shall be discontinued until the flow path can be reestablished, or until the openings to the outside environment are closed.
3. If the Fuel Handling Building closure would be hampered by an outage activity, compensatory actions will be developed.

Control Room Emergency Air Conditioning System (CREACS)

During movement of irradiated fuel assemblies, both CREACS normal outside air intakes should normally be open. If one intake is closed, movement of irradiated fuel assemblies will be suspended until the intake is reopened. These controls are governed by existing action requirements under Technical Specification 3.7.6.1. The actuation of CREACS during a FHA is performed by the radiation monitors located in the normal outside air intakes. Exceeding the setpoints of these radiation monitors will cause dampers to reposition to isolate the normal ventilation system from the Control Room Envelope, start the CREACS fans and open the appropriate outside emergency air intake. The radiation monitors in the Control Room normal outside intake are required to be OPERABLE during movement of irradiated fuel assemblies as governed by TS Table 3.3-6.

5. REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration

PSEG Nuclear LLC (PSEG) has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment" as discussed below:

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

   Response: No.

   An alternate source term calculation has been performed for Salem Nuclear Station that demonstrates that offsite and control room dose consequences of a postulated fuel handling accident remain within the limits provided sufficient decay has occurred prior to the movement of irradiated fuel without taking credit for certain mitigation features such as ventilation filter systems and containment closure. Fuel movement is allowed provided that irradiated fuel has undergone the required decay time.

   The proposed amendment would allow movement of sufficiently decayed irradiated fuel within the containment building with the equipment hatch and personnel air locks open provided that administrative controls are implemented to promptly (within 1 hour) close the containment penetrations.

Either the Containment Purge system or the Auxiliary Building Ventilation System with suction from the containment atmosphere, with associated radiation monitoring will be available whenever movement of irradiated fuel is in progress in the containment building and the equipment hatch is open. If for any reason, this ventilation requirement can not be met, movement of fuel assemblies within the containment building shall be discontinued until the flow path(s) can be reestablished or close the equipment hatch and personnel airlocks. The amendment also would allow movement of irradiated fuel assemblies within the Fuel Handling Building with the Fuel Handling Area Ventilation System (FHAVS) in operation but no credit taken for filtration.

This amendment does not alter the methodology of the FHA or equipment used directly in fuel handling operations. Neither ventilation filter systems, the CPES nor the FHAVS, is used to actually handle fuel. Therefore neither of these systems is an "accident initiator". Similarly, neither the equipment hatch, the personnel air locks, nor any other containment penetration, nor any component thereof is an accident initiator.

In the postulated Fuel Handling Accident, the revised dose calculations, performed using 10 CFR 50.67 and Regulatory Guide 1.183, Alternative Source Term, do not take credit for automatic containment purge isolation thus allowing for continuous monitoring of containment activity until containment closure is achieved. If required, containment purge isolation can be initiated manually from the control room.

Actual fuel handling operations are not affected by the proposed changes. Therefore, the probability of a Fuel Handling Accident is not affected with the proposed amendment. No other accident initiator is affected by the proposed changes.

The FHA in the Fuel Handling Building has been analyzed without credit for filtration by the FHAVS. The analyses of these design basis events were conducted with the Alternative Source Term Methodology in accordance with 10 CFR 50.67 and Regulatory Guide 1.183. These analyses show that the resultant radiation doses are within the limits specified in these documents.

The TEDE radiation doses from the analyses supporting this LCR have been compared to equivalent TEDE radiation doses estimated with the guidelines of R.G. 1.183. The new values are shown to be within the regulatory guidelines.

The revision to the definition of Core Alterations simply reflects the definition in the Standard Technical Specifications, NUREG 1431 for Westinghouse Plants and is supported by the bounding effects of the Fuel Handling Accident analysis.

The deletion of Core Alterations from the APPLICABILITY section of the affected LCO's is based on the fact that, during Core Alterations only the FHA results in cladding damage and potential radiological release. Consequently, the deletion of Core Alterations is consistent with industry approved practice and guidance documents (ex: TSTF-51, revision 2).

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

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

Response: No.

The proposed change does not involve addition or modification to any plant system, structure, or component. The proposed amendment would permit the equipment hatch and personnel air locks to be open during movement of irradiated fuel. The proposed amendment does not involve any change in the operation of these containment

penetrations. Having these penetrations open does not create the possibility of a new accident.

The proposed amendment also would remove the requirements for operability of the FHAVS Filtration System during movement of sufficiently decayed irradiated fuel. It does not alter the operation of these systems. Therefore, the system is not an accident initiator. Modification of the requirements of operability for the system from the plant Technical Specifications does not create the possibility of a new accident.

The revision to the definition of Core Alterations simply reflects the industry position supported by the definition in the Standard Technical Specifications, NUREG 1431 for Westinghouse Plants and is supported by the bounding effects of the Fuel Handling Accident analysis.

The deletion of Core Alterations from the APPLICABILITY section of the affected LCO's is based on the fact that, during Core Alterations only the FHA results in cladding damage and potential radiological release. Consequently, the deletion of Core Alterations is consistent with industry approved practice and guidance documents (ex: TSTF-51, revision 2).

The proposed amendment does not create the possibility of a new or different kind of accident than any previously evaluated.

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

Response: No.

The assumptions and input used in the analysis are conservative as noted below. The design basis Fuel Handling Accidents have been defined to identify conservative conditions. The source term and radioactivity releases have been calculated pursuant to Regulatory Guide 1.183 and with conservative assumptions concerning prior reactor operation. The control room atmospheric dispersion factors have been calculated with conservative assumptions associated with the release. The conservative assumptions and input noted above ensure that the radiation doses cited in this License Change Request are the upper bound to radiological consequences of a Fuel Handling Accident either in Containment or the Fuel Handling Building. The analyses show that there is a significant margin between the TEDE radiation doses calculated for the postulated Fuel Handling Accident using the Alternative Source Term and the acceptance limits of 10 CFR 50.67 and Regulatory Guide 1.183.

The revision to the definition of Core Alterations simply reflects the industry position supported by the definition in the Standard Technical Specifications, NUREG 1431 for Westinghouse Plants and is supported by the bounding effects of the Fuel Handling Accident analysis.

The deletion of Core Alterations from the APPLICABILITY section of the affected LCO's is based on the fact that, during Core Alterations only the FHA results in cladding damage and potential radiological release. Consequently, the deletion of Core Alterations is consistent with industry approved practice and guidance documents (ex: TSTF-51, revision 2).

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

Based on the above, PSEG 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

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

The NRC's traditional methods 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. This guide 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 other regulatory documents when used in conjunction with an approved AST and the TEDE criteria provided in 10 CFR 50.67.

PSEG used this regulatory guide extensively in the preparation of this "selective implementation". This application and the supporting analyses comply with this guidance as it applies to a Fuel Handling Accident.

Title 10, Code of Federal Regulations, Part 50 Section 67, "Accident Source Term".

10 CFR 50.67 permits licensees to voluntarily revise the accident source term used in design basis radiological consequences analyses. This document is part of a 10 CFR 50.90 license amendment application and evaluates the consequences of a design basis fuel handling accident as previously described in the Salem UFSAR.

Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors".

RG 1.183 supersedes corresponding radiological assumptions provided in other regulatory guides and standard review plan chapters when used in conjunction with an approved alternative source term and the TEDE provided in 10 CFR 50.67.

10 CFR 100, "Determination of Exclusion Area, Low Population Zone and Population Center Distance".

10 CFR 100.11 provides criteria for evaluating the radiological aspects of reactor sites. A footnote to 10 CFR 100.11 states that the fission product release assumed in these evaluations should be based on a major accident involving substantial meltdown of the core with subsequent release of appreciable quantities of fission products. A similar footnote appears in 10 CFR 50.67. In accordance with the provisions of 10 CFR 50.67(a), PSEG applied the dose reference values in 10 CFR 50.67 (b) (2) in the analyses in lieu of 10 CFR 100 for the Fuel Handling Accident.

NUREG-0800, Standard Review Plan, Section 15.7.4, "Radiological Consequences of Fuel Handling Accidents".

The SRP Section 15.7.4 describes the radiological effects of a postulated Fuel Handling Accident. The SRP does not directly refer to the guidance of RG 1.183 or 10 CFR 50.67. Instead, it refers to regulatory documents, which are superseded by the selective application of the Alternative Source Term for the FHA.

10 CFR 50 Appendix A, General Design Criteria 19, Control Room

PSEG has applied the guidelines provided by 10 CFR 50.67 and RG 1.183, which supersede the current requirements of GDC 19 for the Fuel Handling Accident.

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

6. ENVIRONMENTAL CONSIDERATION

ENVIRONMENTAL ASSESSMENT/IMPACT STATEMENT

Pursuant to 10 CFR 51.22(b), an evaluation of this license change request has been performed to determine whether or not it meets the criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) of the regulations.

Implementation of this amendment will have no adverse impact upon the Salem units; neither will it contribute to any significant additional quantity or type of effluent being available for adverse environmental impact or personnel exposure. The change does not introduce any new effluents or significantly increase the quantities of existing effluents. As such, the change cannot significantly affect the types or amounts of any effluents that may be released offsite. The new consequences of the revised Fuel Handling Accident analysis remain well below the acceptance criteria specified in 10 CFR 50.67 and Regulatory Guide 1.183.

It has been determined there is:

1. No significant hazards consideration,

2. No significant change in the types, or significant increase in the amounts, of any effluents that may be released offsite, and

3. No significant increase in individual or cumulative occupational radiation exposures involved.

Therefore, this amendment to the Salem TS meets the criteria of 10 CFR 51.22(c)(9) for categorical exclusion from an environmental impact statement.

7. RISK SIGNIFICANCE

Based on the results of the conservative dose calculations provided in the previous submittal (LR-N02-0231, dated June 28, 2002) and to support this submittal, the risk to the health and safety of the public as a result of a Fuel Handling Accident inside the containment with the equipment hatch open is minimal. Actual fuel handling accidents, which have occurred in the past, have resulted in minimal or no releases, which supports that the assumptions and methodology utilized in the radiological dose calculations, in accordance with Regulatory Guide 1.183, are very conservative. Other safety factors supporting the minimal risk significance involved with this request are: 1) Fuel Decay Time prior to start of fuel movement, 2) The water level (reactor cavity and spent fuel pool) that covers the fuel assemblies, 3) administrative controls for containment closure, and 4) ventilation and radiation monitoring availability.

In summary, based on the considerations discussed above:

a. There is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner,
b. Such activities will be conducted in compliance with the Commission's regulations, and
c. The issuance of the amendment will not be inimical to the common defense and security or the health and safety of the public.

REFERENCES

a. PSEG Submittal to the NRC, LCR S02-03 (Decay Time), dated June 28, 2002.
b. UFSAR Section 9.4.3.1, Fuel Handling Area Ventilation.
c. UFSAR Section 15.4.6, Fuel Handling Accident.
d. PSEG Calculation S-C-ZZ-MDC-1920, Revision 1, Fuel Handling Accidents Occurring in Fuel Handling Building and Containment – AST Analysis for Relaxation of Containment Integrity.
e. PSEG Calculation S-C-ZZ-MDC-1912, Revision 0, CR χ/Qs Using ARCON96 Code- Equipment Hatch, Plant Vent Releases.
f. NUMARC 93-01, Revision 3, " Industry Guidelines for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants", July 2000.
g. NUREG-0800, Standard Review Plan, Section 15.7.4, Rev.1, July 1981.
h. USNRC Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, July 2000.
i. 10 CFR 50.67, "Accident Source Term"
j. Salem Units 1 and 2 Technical Specifications.
k. USNRC Safety Guide 25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for BWRS and PWRS".
l. NUREG 1431, Standard Technical Specifications for Westinghouse PWRS, TSTF-51, Rev 2.
m. NRC Issued License Amendments involving Containment Closure Relaxations During Fuel Movement:
  - Turkey Point, SER Issued September 27, 2001
  - Brunswick, SER Issued March 14, 2002
  - Shearon Harris, SER issued July 30, 2001
  - Watts Bar, SER issued January 22, 2002
  - TMI Unit 1, SER issued October 2, 2001
  - Fort Calhoun, SER issued March 26,2002

**SALEM NUCLEAR GENERATING STATION, UNITS 1 AND 2
FACILITY OPERATING LICENSES DPR-70 AND DPR-75
DOCKET NOS. 50-272 AND 50-311
REFUELING OPERATIONS – RELAXATION OF REQUIREMENTS APPLICABLE DURING MOVEMENT OF IRRADIATED FUEL**

TECHNICAL SPECIFICATION PAGES WITH PROPOSED CHANGES

The following Technical Specifications for Facility Operating License No. DPR-70 are affected by this change request:

Technical Specification	Page
1.8 1.9	1-2
3/4.3.3.1	3/4 3-36 3/4 3-37 3/4 3-38
3/4.7.6.1	3/4 7-18 3/4 7-19
3/4.9.4	3/4 9-4
3/4.9.9	3/4 9-9
3/4.9.12	3/4 9-12 3/4 9-13 3/4 9-14
B 3/4.3.3.1	B 3/4 3-1a B 3/4 3-2
B 3/4.7.7	B 3/4 7-5c
B 3/4.9.4	B 3/4 9-1 B 3/4 9-2 B 3/4 9-3
B 3/4.9.9 B 3/4.9.12	B 3/4 9-4

The following Technical Specifications for Facility Operating License No. DPR-75 are affected by this change request:

Technical Specification	Page
1.8 1.9	1-2
3/4.3.3.1	3/4 3-39 3/4 3-40 3/4 3-41
3/4.7.6	3/4 7-15 3/4 7-16
3/4.9.4	3/4 9-4
3/4.9.9	3/4 9-10
3/4.9.12	3/4 9-13 3/4 9-14 3/4 9-15
B 3/4.3.3.1	B 3/4 3-1a B 3/4 3-2
B 3/4.7.7	B 3/4 7-5c
B 3/4.9.4	B 3/4 9-1 B 3/4 9-2 B 3/4 9-3
B 3/4.9.9 B 3/4.9.12	B 3/4 9-4

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

1.7.1 All penetrations required to be closed during accident conditions are either:

a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open under administrative control as permitted by Specification 3.6.3.1.

1.7.2 All equipment hatches are closed and sealed,

1.7.3 Each air lock is OPERABLE pursuant to Specification 3.6.1.3,

1.7.4 The containment leakage rates are within the limits of Specification 3.6.1.2, and

1.7.5 The sealing mechanism associated with each penetration (e.g., welds, bellows or 0-rings) is OPERABLE.

CORE ALTERATION

1.8 ~~NOT USED~~ 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. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

~~CORE ALTERATION~~

~~1.9 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.~~

CORE OPERATING LIMITS REPORT

1.9a The CORE OPERATING LIMITS REPORT (COLR) is the unit-specific document that provides 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 Specification 6.9.1.9. Unit operation within these operating limits is addressed in individual specifications.

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of 1-131 (microcuries per gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The

TABLE 3.3-6
RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Fuel Storage Area	1	*	≤15 mR/hr	10^{-1}-10^{4} mR/hr	19
b. Containment Area	2	1,2,3&4	≤10^{3} R/hr	1-10^{7} R/hr	23
2. PROCESS MONITORS					
a. Containment					
1) Gaseous Activity	1#	~~6~~	~~Set at less than or equal~~	~~10^{1}-10^{6} cpm~~	~~22 & 23~~
a) Purge & Pressure - Vacuum Relief Isolation		and	to 50% of the 10CFR20 ~~concentration limits for gaseous effluents released to unrestricted areas.~~		
		1,2,3,4&5	per ODCM Control 3.3.3.9		
b) RCS Leakage Detection	1	1,2,3&4	N/A	10^{1}-10^{6} cpm	20
2) Air Particulate Activity					
a) ~~Purge & Pressure-Vacuum Relief Isolation~~ (NOT USED)	~~1~~	~~6~~	~~≤ 2 x background~~	~~10^{1}-10^{6} cpm~~	~~22~~
b) RCS Leakage Detection	1	1,2,3&4	N/A	10^{1}-10^{6} cpm	20

* With fuel in the storage pool or building.
\# The plant vent noble gas monitor may also function in this capacity when the purge/pressure-vacuum relief isolation valves are open.

TABLE 3.3-6 (Continued)

TABLE NOTATION

ACTION 19 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

ACTION 20 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

ACTION 22 - (Not Used) ~~With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.9.~~

ACTION 23 - With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirements, initiate the preplanned alternate method of monitoring the appropriate parameter(s), within 72 hours, and:

1) either restore the inoperable Channel(s) to OPERABLE status within 7 days of the event, or

2) prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

ACTION 24 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel(s) to OPERABLE status within 7 days or initiate and maintain operation of the Control Room Emergency Air Conditioning System (CREACS) in the pressurization or recirculation mode of operation. CORE ALTERATIONS and movement of irradiated fuel assemblies will be suspended during operation in the recirculation mode.

ACTION 25 - With no channels OPERABLE in a Control Room air intake, immediately initiate and maintain operation of the CREACS in the pressurization or recirculation mode of operation. CORE ALTERATIONS and movement of irradiated fuel assemblies will be suspended during operation in the recirculation mode.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNELS CHECKS	SOURCE CHECKS	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. AREA MONITORS					
a. Fuel Storage Area	S	M	R	Q	*
b. Containment Area	S	M	R	Q	1, 2, 3 & 4
2. PROCESS MONITORS					
a. Containment Monitors					
1) Gaseous Activity					
a) Purge & Pressure Vacuum Relief Isolation	S	M	R	Q	1, 2, 3, 4~~,&~~ 5 ~~&6~~
b) RCS Leakage Detection	S	M	R	Q	1, 2, 3 & 4
2) Air Particulate Activity					
a) ~~Purge & Pressure Vacuum Relief Isolation~~ (NOT USED)	~~S~~	~~M~~	~~R~~	~~Q~~	~~1, 2, 3, 4, & 6~~
b) RCS Leakage Detection	S	M	R	Q	1, 2, 3 & 4

*With fuel in the storage pool or building.

3/4.7.6 CONTROL ROOM EMERGENCY AIR CONDITIONING SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.6.1 The common control room emergency air conditioning system (CREACS)* shall be OPERABLE with:

a. Two independent air conditioning filtration trains (one from each unit) consisting of:

1. Two fans and associated outlet dampers,
2. One cooling coil,
3. One charcoal adsorber and HEPA filter array,
4. Return air isolation damper.

b. All other automatic dampers required for operation in the pressurization or recirculation modes.

c. The control room envelope intact.

APPLICABILITY: ALL MODES and during movement of irradiated fuel assemblies ~~and during CORE ALTERATIONS.~~

ACTION: MODES 1, 2, 3, and 4

a. With one filtration train inoperable, align CREACS for single filtration train operation within 4 hours, and restore the inoperable filtration train to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With CREACS aligned for single filtration train operation and with one of the two remaining fans or associated outlet damper inoperable, restore the inoperable fan or damper to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With the Control Room Envelope inoperable, restore the Control Room Envelope to OPERABLE status within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With one or both series isolation damper(s) on a normal Control Area Air Conditioning System (CAACS) outside air intake or exhaust duct inoperable, close the affected duct within 4 hours by use of at least one isolation damper secured in the closed position or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. (Refer to ACTION 24 of Table 3.3-6.)

* The CREACS is a shared system with Salem Unit 2

e. With one or both isolation damper(s) on an outside emergency air conditioning air intake duct inoperable, close the affected duct within 4 hours by use of at least one isolation damper secured in the closed position and restore the damper(s) to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

f. With any isolation damper between the normal CAACS and the CREACS inoperable, secure the damper in the closed position within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

MODES 5 and 6 or during movement of irradiated fuel assemblies ~~and during CORE ALTERATIONS.~~

a. With one filtration train inoperable, align CREACS for single filtration train operation within 4 hours, or suspend ~~CORE ALTERATIONS and~~ movement of irradiated fuel assemblies.

b. With CREACS aligned for single filtration train operation with one of the two remaining fans or associated outlet damper inoperable, restore the fan or damper to OPERABLE status within 72 hours, or suspend ~~CORE ALTERATIONS and~~ movement of irradiated fuel assemblies.

c. With two filtration trains inoperable, immediately suspend ~~CORE ALTERATIONS and~~ movement of irradiated fuel assemblies.

d. With the Control Room Envelope inoperable, immediately suspend ~~CORE ALTERATIONS and~~ movement of irradiated fuel assemblies.

e. With one or both series isolation damper(s) on a normal CAACS outside air intake or exhaust duct inoperable, immediately suspend ~~CORE ALTERATIONS and~~ movement of irradiated fuel assemblies until the affected duct is closed by use of at least one isolation damper secured in the closed position. (Refer to ACTION 24 of Table 3.3-6.)

f. With one or both series isolation damper(s) on an outside emergency air conditioning air intake duct inoperable, immediately suspend ~~CORE ALTERATIONS and~~ movement of irradiated fuel assemblies until the affected duct is closed by use of at least one isolation damper secured in the closed position. To resume ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies, at least one emergency air intake duct must be operable on each unit.

g. With any isolation damper between the CAACS and the CREACS inoperable, immediately suspend ~~CORE ALTERATIONS and~~ movement of irradiated fuel assemblies until the damper is closed and secured in the closed position.

REFUELING OPERATIONS

CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The containment building penetrations shall be in the following status:

a. The equipment hatch inside door is capable of being closed and held in place by a minimum of four bolts, or an equivalent closure device installed and capable of being closed,

b. A minimum of one door in each airlock is capable of being closed

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or
2. capable of being closed by the Containment Purge and Pressure-Vacuum Relief Isolation System.

Note: Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls.

APPLICABILITY: During ~~CORE ALTERATIONS or~~ movement of irradiated fuel within the containment.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving ~~CORE ALTERATIONS or~~ movement of irradiated fuel in the containment building. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.4.1 Each of the above required containment building penetrations shall be determined to be either in its required condition or capable of being closed by a manual or automatic containment isolation valve at least once per 7 days.

4.9.4.2 Once per refueling prior to the start of movement of irradiated fuel assemblies within the containment building, verify the capability to install, within 1 hour, the equipment hatch. Applicable only when the equipment hatch is open during movement of irradiated fuel in the containment building.

4.9.4.3 Verify, once per 18 months, each required containment purge isolation valve actuates to the isolation position on a manual actuation signal.

~~REFUELING OPERATIONS~~

~~CONTAINMENT PURGE AND PRESSURE-VACUUM RELIEF ISOLATION SYSTEM~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.9.9 The Containment Purge and Pressure-Vacuum Relief isolation system shall be OPERABLE.~~

~~APPLICABILITY: MODE 6.~~

~~ACTION:~~

~~With the Containment Purge and Pressure-Vacuum Relief isolation system inoperable, close each of the Purge and Pressure-Vacuum Relief penetrations providing direct access from the containment atmosphere to the outside atmosphere. The provisions of Specification 3.0.3 are not applicable.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.9.9 The Containment Purge and Pressure-Vacuum Relief isolation system shall be demonstrated OPERABLE within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS by verifying that containment Purge and Pressure-Vacuum Relief isolation occurs on manual initiation and on a high radiation test signal from each of the containment radiation monitoring instrumentation channels.~~

REFUELING OPERATIONS

FUEL HANDLING AREA VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.12 The Fuel Handling Area Ventilation System shall be OPERABLE with:

a. Two exhaust fans and one supply fan OPERABLE and operating, and
b. Capable of maintaining slightly negative pressure in the Fuel Handling Building.

APPLICABILITY: ~~Whenever irradiated fuel is in the storage pool.~~ During movement of irradiated fuel within the Fuel Handling Building

ACTION:

a. With no Fuel Handling Area Ventilation System OPERABLE, suspend all operations involving movement of fuel within the storage pool ~~or crane operation with loads over the storage pool~~ until the Fuel Handling Area Ventilation System is restored to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12 The above required ventilation system shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that, the Fuel Handling Building is maintained at a slightly negative pressure with respect to atmospheric pressure.

b. At least once per 31 days by verifying both exhaust fans and one supply fan start and operate for at least 15 minutes, if not operating already.

c. At least once per 18 months by verifying a system flowrate of 19,490 cfm ± 10% during system operation.

~~a.1 At least once per 31 days by initiating flow through the HEPA filter and charcoal adsorber train and verifying that the train operates for at least 15 minutes.~~

~~a.2 Prior to an during movement of irradiated fuel assemblies or crane operation over the storage pool in the Fuel Handling Building:~~

~~1.Both exhaust fans and one supply fan must be OPERABLE and operating with flow being directed through the HEPA and charcoal filters.~~

~~2.All dampers required to divert the entire airflow through the HEPA/charcoal filter train are OPERABLE and in the position required to divert full exhaust flow through the HEPA/charcoal filter train.~~

~~3.Ductwork, dampers and housings which will ensure all post-accident exhausted air is processed through the HEPA/charcoal filter train are intact.~~

~~4.The fuel handling area is maintained at a negative pressure equal to or more negative than 1/8 inch water gauge relative to the outside atmosphere, and~~

~~5.At least once per 24 hours thereafter verify both exhaust fans and one supply fan operating with the entire flow being directed through the HEPA and charcoal filters.~~

~~b. At least once per 18 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 system, by:~~

~~1. Verifying that with the ventilation system operating at a flow rate of 19,490 cfm ± 10% and exhausting through the HEPA filters and charcoal adsorbers, the total bypass flow of the ventilation system to the facility vent, including leakage through the ventilation system diverting valves, is ≤ 1% when the ventilation system is tested by admitting cold DOP at the storage pool ventilation system intake.~~

~~2. Verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place while operating the ventilation system at a flow rate of 19,490 cfm ± 10%.~~

~~3. Verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place while operating the ventilation system at a flow rate of 19,490 cfm ± 10%.~~

~~4. Verifying within 31 days after removal from the FHV unit, that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than 5.0% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and a relative humidity of 95%.~~

~~5. Verifying a system flow rate of 19,490 cfm, ± 10% during system operation.~~

~~c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal from the FHV unit, that a laboratory analysis of a representative carbon sample, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows a methyl iodide penetration less than 5.0% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and a relative humidity of 95%.~~

~~d. At least once per 18 months by:~~

~~1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is ≤ 4 inches Water Gauge while operating the ventilation system at a flow rate of 19,490 cfm ± 10%.~~

~~2. Verifying that the air flow distribution is uniform within 20% across HEPA filters and charcoal adsorbers.~~

~~3. Deleted.~~

~~4. Verifying that the ventilation system maintains the spent fuel storage pool area at a negative pressure of ≥ 1/8 inches Water Gauge relative to the outside atmosphere during system operation.~~

~~e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place while operating the filter train at a flow rate of 19,490 cfm ± 10%.~~

~~f. After each complete or partial replacement of a charcoal absorber bank by verifying that the charcoal absorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place while operating the filter train at a flow rate of 19,490 cfm ± 10%.~~

field sensors and signal processing equipment for these channels are assumed to operate within the allowances of these uncertainty magnitudes.

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability. Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," and Supplements to that report. Surveillance intervals and out of service times were determined based on maintaining an appropriate level of reliability of the Reactor Protection System and Engineered Safety Features instrumentation.

The measurement of response time at the specified frequencies provides assurance that the protective and ESF action function associated with each channel is completed within the time limit assumed in the accident analyses.
No credit was taken in the analyses for those channels with response times indicated as not applicable.

Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either 1) in place, onsite or offsite test measurements or 2) utilizing replacement sensors with certified response times.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that 1) the radiation levels are continually measured in the areas served by the individual channels and 2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

~~The isolation alarm/trip setpoint for the Containment Purge and Pressure Relief system during MODE 6 is established to ensure that in the event of a fuel handling accident inside containment, prompt isolation will occur to ensure calculated offsite doses remain below 10CFR100 limits. Prompt isolation will also ensure that Control Room doses following a fuel handing accident will remain below GDC-19 limits. The alarm/trip setpoint value of Table 3.3-6 for the R12A while in Mode 6 will be established based upon isolating the Containment Purge and Pressure Relief System when containment gaseous activity levels reach 50% of the more conservative 10CFR20 concentration limits for release to unrestricted areas. These concentration limits are specified in 10CFR20, Appendix B, Table II, Column 1. A setpoint based on 50% of the 10CFR20 concentration limits will be low enough to ensure that prompt Containment Purge and Pressure Relief system isolation occurs during a fuel handling accident and high enough to prevent unnecessary Containment Purge and Pressure Relief system isolations caused by routine outage activities.~~

In the postulated Fuel Handling Accident, the revised dose calculations, performed using 10 CFR 50.67 and Regulatory Guide 1.183, Alternative Source Term, do not take credit for automatic containment purge isolation thus allowing for continuous monitoring of containment activity until containment closure is achieved. If required, containment purge isolation can be initiated manually from the control room.

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION (Continued)

CROSS REFERENCE - TABLES 3.3-6 AND 4.3-3

T/S Table Item No.	Instrument Description	Acceptable RMs Channels
1a	Fuel Storage Area	1R5 or 1R9
1b	Containment Area	1R44A and B
2a1a	Containment Gaseous Activity Purge & Pressure/Vacuum Relief Isolation	1R12A or 1R41A, and D(1) (2)
2a1b	Containment Gaseous Activity RCS Leakage Detection	1R12A
2a2a	~~Containment Air Particulate Activity Purge & Pressure/Vacuum Relief Isolation~~ (NOT USED)	~~1R11A~~
2a2b	Containment Air Particulate Activity RCS Leakage Detection	1R11A
2b1	Noble Gas Effluent Medium Range Auxiliary Building Exhaust System (Plant Vent)	1R41B & D (1)(3)(5)
2b2	Noble Gas Effluent High Range Auxiliary Building Exhaust System (Plant Vent)	1R41C &D (1)(4)(5)
2b3	Noble Gas Effluent Main Steamline Discharge - Safety Valves and Atmospheric Steam Dumps	1R46
2b4	Noble Gas Effluent Condenser Exhaust System	1R15
3a	Unit 1 Control Room Intake Channel 1 (to Unit 1 Monitor) Unit 1 Control Room Intake Channel 2 (to Unit 2 Monitor) Unit 2 Control Room Intake Channel 1 (to Unit 2 Monitor) Unit 2 Control Room Intake Channel 2 (to Unit 1 Monitor)	1R1B-1 2R1B-2 2R1B-1 1R1B-2

Immediate action(s), in accordance with the LCO Action Statements, means that the required action should be pursued without delay and in a controlled manner.

(1) The channels listed are required to be operable to meet a single operable channel for the Technical Specification's "Minimum Channels Operable" requirement.

(2) The setpoint applies to 1R41D. The measurement range applies to 1R41A and B which display in uCi/cc using the appropriate channel conversion factor form cpm to uCi/cc.

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 50 except for the Fuel Handling Accident, where the allowable doses to Control Room personnel are provided by 10CFR 50.67 and RG 1.183, Alternative Source Term.

3/4.7.7 AUXILIARY BUILDING EXHAUST AIR FILTRATION SYSTEM

The Auxiliary Building Ventilation System (ABVS) consists of two major subsystems. They are designed to control Auxiliary Building temperature during normal and emergency modes of operation, and to contain Auxiliary Building airborne contamination during Loss of Coolant Accidents (LOCA). The two subsystems are:

1. A once through filtration exhaust system, designed to contain particulate and gaseous contamination and prevent it from being released from the building in accordance with 10CFR20, and

2. A once through air supply system, designed to deliver outside air into the building to maintain building temperatures within acceptable limits. For the purposes of satisfying the Technical Specification LCO, one supply fan must be administratively removed from service such that the fan will not auto-start on an actuation signal; however, the supply fan must be OPERABLE with the exception of this administrative control.

These systems operate during normal and emergency plant modes. Additionally, the system provides a flow path for containment purge supply and exhaust during Modes 5 and 6. Either the Containment Purge system or the Auxiliary Building Ventilation System with suction from the containment atmosphere, with associated radiation monitoring will be available whenever movement of irradiated fuel is in progress in the containment building and the equipment hatch is open. If for any reason, this ventilation requirement can not be met, movement of fuel assemblies within the containment building shall be discontinued until the flow path(s) can be reestablished or close the equipment hatch and personnel airlocks.

The exhaust system consists of three 50% capacity fans that are powered from vital buses. These fans exhaust from a common plenum downstream from three High Efficiency Particulate Air (HEPA) filter banks, two of which, 11 & 12 can be interchangeably aligned to discharge to a single carbon adsorber bed. Filter unit 11 is limited in capacity and can only be aligned to the ECCS areas of the Auxiliary Building for HEPA only or HEPA + Carbon modes of filtration. Filter unit 12 can be used to ventilate the normal areas of the Auxiliary Building in HEPA only, or when used in conjunction with 13, may be used to ventilate the ECCS areas of the Auxiliary Building in HEPA + Carbon. Filter unit 13 does not communicate with the carbon adsorber housing and is used for exhausting air from the normal areas of the Auxiliary Building during any plant Mode or purging the Containment Building during Modes 5&6. The fans are designed for continuous operation, to control the Auxiliary Building pressure at -0.10" Water Gauge with respect to atmosphere.

The supply system consists of two 100% capacity fans that are powered from vital buses, and distribute outdoor air to the general areas and corridors of the building through associated ductwork.

AUXILIARY BUILDING VENTILATION ALIGNMENT MATRIX

Unit 11 from ECCS HEPA only, with
Unit 12 from Aux. Normal HEPA only; or

3/4.9.1 BORON CONCENTRATION

The limitations on minimum boron concentration (2000 ppm) ensure that: 1) the reactor will remain subcritical during CORE ALTERATIONS, and 2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. The limitation on K_{eff} of no greater than 0.95 which includes a conservative allowance for uncertainties, is sufficient to prevent reactor criticality during refueling operations.

The sampling and analysis required by surveillance requirement 4.9.1.2 ensures the boron concentration required by Limiting Condition of Operation 3.9.1 is met. Sampling and analysis of the refueling canal is required if water exists in the refueling canal, regardless of the amount.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the source range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short-lived fission products. This decay time is consistent with the assumptions used in the accident analyses.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

During ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies within containment the requirements for containment building penetration closure capability and OPERABILITY ensure that a release of fission product radioactivity within containment will not exceed the guidelines and dose calculations described in Reg. Guide 1.183, Alternative Radiological Source Term for Evaluating Design Basis Accidents at Nuclear Power Reactors. In MODE 6, the potential for containment pressurization as a result of an accident is not likely. Therefore, the requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements during ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies within containment are referred to as "containment closure" rather than containment OPERABILITY. For the containment to be OPERABLE, CONTAINMENT INTEGRITY must be maintained. Containment closure means that all potential containment atmosphere release paths are closed or capable of being closed. Closure restrictions include the administrative controls to allow the opening of both airlock doors and the equipment hatch during fuel movement provided that: 1) the equipment inside door or an equivalent closure device installed is capable of being closed with four bolts within 1 hour by a designated personnel; 2) the airlock door is capable of being closed within 1 hour by a designated personnel, 3) either the Containment Purge System or the Auxiliary Building Ventilation System taking suction from the containment atmosphere are operating and 4) the plant is in Mode 6 with at least 23 feet of water above the reactor pressure vessel flange.

Administrative requirements are established for the responsibilities and appropriate actions of the designated personnel in the event of a Fuel Handling Accident inside containment. These requirements include the responsibility to be able to communicate with the control room, to ensure that the equipment hatch is capable of being closed, and to close the equipment hatch and personnel airlocks within 1 hour in the event of a fuel handling accident inside containment. These administrative controls ensure containment closure will be established in accordance with and not to exceed the dose calculations performed using guidelines of Regulatory Guide 1.183.

The containment serves to limit the fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained well within the requirements of 10CFR100 and Reg. Guide 1.183, Alternative Source Term, as applicable. Additionally, the containment provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The Containment Equipment Hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into or out of containment. During ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies within containment, the Containment Equipment Hatch inside door can be open provided that: 1) It is capable of being closed with four bolts within 1 hour by designated personnel, 2) either the Containment Purge System or the Auxiliary Building Ventilation System taking suction from the containment atmosphere are operating and 3) The plant is in Mode 6 with at least 23 feet of water above the reactor pressure vessel flange. Good engineering practice dictates that the bolts required by the LCO are approximately equally spaced.

An equivalent closure device may be installed as an alternative to installing the Containment Equipment Hatch inside door with a minimum of four bolts. Such a closure device may provide penetrations for temporary services used to support maintenance activities inside containment at times when containment closure is required; and may be installed in place of the Containment Equipment Hatch inside door or outside door. Penetrations incorporated into the design of an equivalent closure device will be considered a part of the containment boundary and as such will be subject to the requirements of Technical Specification 3/4.9.4. Any equivalent closure device used to satisfy the requirements of Technical Specification 3/4.9.4.a will be designed, fabricated, installed, tested, and utilized in accordance with established procedures to ensure that the design requirements for the mitigation of a fuel handling accident during refueling operations are met. In case that this equivalent closure device is installed in lieu of the equipment hatch inside door, the same restrictions and administrative controls apply to ensure closure will take place within 1 hour following a FHA inside containment.

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during operation in MODES 1, 2, 3, and 4 as specified in LCO 3.6.1.3, "Containment Air Locks". Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of unit shutdown, when containment closure is not required and frequent containment entry is necessary, the air lock interlock mechanism may be disabled. This allows both doors of an airlock to remain open for extended periods.

During ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies within containment, containment closure may be required; therefore, the door interlock mechanism may remain disabled, and both doors of each containment airlock may be open if: 1) At least one door of each airlock is capable of being closed within 1 hour by a dedicated individual, 2) either the Containment Purge System or the Auxiliary Building Ventilation System taking suction from the containment atmosphere are operating and 3) the plant is in Mode 6 with at least 23 feet of water above the reactor pressure vessel flange.

In the postulated Fuel Handling Accident, the revised dose calculations, performed using 10 CFR 50.67 and Regulatory Guide 1.183, Alternative Source Term, do not take credit for automatic containment purge isolation thus allowing for continuous monitoring of containment activity until containment closure is achieved. If required, containment purge isolation can be initiated manually from the control room.

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side. Isolation may be achieved by an OPERABLE automatic isolation valve, or by a manual isolation valve, blind flange, or equivalent. Equivalent isolation methods may include the use of a material that can provide a temporary atmospheric pressure, ventilation barrier. Any equivalent method used to satisfy the requirements of Technical Specification 3/4.9.4.c.1 will be designed, fabricated, installed, tested, and utilized in accordance with established procedures to ensure that the design requirements for the mitigation of a fuel handling accident during refueling operations are met.

The surveillance requirement 4.9.4.2 demonstrates that the necessary hardware, tools, and equipment are available to close the equipment hatch. The surveillance is performed prior to movement of irradiated fuel assemblies within the containment. This surveillance is only required to be met when the equipment hatch is to be open during fuel movement.

~~The OPERABILITY of the containment purge system ensures that the containment vent and purge penetrations will be automatically isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.~~

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during CORE ALTERATIONS.

3/4.9.6 MANIPULATOR CRANE

The OPERABILITY requirements for the manipulator cranes ensure that: 1) manipulator cranes will be used for movement of control rods and fuel assemblies, 2) each crane has sufficient load capacity to lift a control rod or fuel assembly, and 3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE BUILDING

The restriction on movement of loads in excess of the nominal weight of a fuel and control rod assembly and associated handling tool over other fuel assemblies in the storage pool ensures that in the event this load is dropped (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the accident analyses.

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirements that at least one residual heat removal loop be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification. A minimum flow rate of 1000 gpm is required. Additional flow limitations are specified in plant procedures, with the design basis documented in the Salem UFSAR. These flow limitations address the concerns related to vortexing and air entrapment in the Residual Heat Removal system, and provide operational flexibility by adjusting the flow limitations based on time after shutdown. The requirement to have two RHR loops OPERABLE when there is less than 23 feet of water above the reactor vessel flange ensures that a single failure of the operating RHR loop will not result in a complete loss of residual heat removal capability.

For support systems: Service Water (SW) and Component Cooling (CC), component redundancy is necessary to ensure no single active component failure will cause the loss of Decay Heat Removal. One piping path of SW and CC is adequate when it supports both RHR loops. The support systems needed before entering into the desired configuration (e.g., one service water loop out for maintenance in Modes 5 and 6) are controlled by procedures, and include the following:

- A requirement that the two RHR, two CC and two SW pumps, powered from two different vital buses be kept operable
- A listing of the active (air/motor operated) valves in the affected flow path to be locked open or disabled.

Note that four filled reactor coolant loops, with at least two steam generators with at least their secondary side water level greater than or equal to 5% (narrow range), may be substituted for one residual heat removal loop. This ensures that a single failure does not cause a loss of decay heat removal.

With the reactor vessel head removed and 23 feet of water above the reactor pressure vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core.

3/4.9.9 (NOT USED) ~~CONTAINMENT PURGE AND PRESSURE-VACUUM RELIEF ISOLATION SYSTEM~~

~~The OPERABILITY of this system ensures that the containment vent and purge penetrations will be automatically isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.~~

3/4.9.10 and 3/4/9/11 WATER LEVEL - REACTOR VESSEL AND STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the accident analysis.

3/4.9.12 FUEL HANDLING AREA VENTILATION SYSTEM

~~The limitations on the fuel handling area ventilation system ensure that all radioactive material released from a dropped irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. The OPERABILITY of this system is consistent with the assumptions of the accident analyses. Laboratory testing of the carbon adsorber is performed in accordance with ASTM D3803-1989 with an acceptance criteria that is determined by applying a minimum safety factor of 2 to the charcoal filter removal efficiency credited in the design basis dose analysis as specified in Generic Letter 99-02.~~

The operability of the Fuel Handling Area Ventilation System during movement of irradiated fuel ensures that a release of fission product radioactivity within the Fuel Handling Building will not exceed the guidelines and dose calculations described in Reg. Guide 1.183, Alternative Radiological Source Term for Evaluating Design Basis Accidents at Nuclear Power Reactors.~~ensures all building exhaust flow is processed through the HEPA/charcoal filter train whenever a Fuel Handling Accident is possible. This will minimize offsite doses following the postulated Fuel Handling Accident.~~

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

1.7.1 All penetrations required to be closed during accident conditions are either:

a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are opened under administrative control as permitted by Specification 3.6.3.1.

1.7.2 All equipment hatches are closed and sealed,

1.7.3 Each air lock is OPERABLE pursuant to Specification 3.6.1.3,

1.7.4 The containment leakage rates are within the limits of Specification 3.6.1.2, and

1.7.5 The sealing mechanism associated with each penetration (e.g., welds, bellows or 0-rings) is OPERABLE.

CORE ALTERATION

1.8 ~~NOT USED~~ 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. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position

~~CORE ALTERATION~~

~~1.9 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.~~

CORE OPERATING LIMITS REPORT

1.9~~a~~ The CORE OPERATING LIMITS REPORT (COLR) is the unit-specific document that provides 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 Specification 6.9.1.9. Unit operation within these operating limits is addressed in individual specifications.

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The

TABLE 3.3-6
RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Fuel Storage Area	1	*	≤15 mR/hr	10^{-1}-10^{4} mR/hr	23
b. Containment Area	2	1,2,3&4	≤10^{3} R/hr	1-10^{7} R/hr	26
2. PROCESS MONITORS					
a. Containment					
1) Gaseous Activity	1#	~~6~~	~~Set at less than or equal~~	~~10^{1}-10^{6} cpm~~	~~26~~
a) Purge & Pressure - Vacuum Relief Isolation		~~and~~	~~to 50% of the 10CFR20 concentration limits for gaseous effluents released to unrestricted areas~~.		
		1,2,3,4&5	per ODCM Control 3.3.3.9		
b) RCS Leakage Detection	1	1,2,3&4	N/A	10^{1}-10^{6} cpm	24
2) Air Particulate Activity					
~~a)~~a) ~~Purge & Pressure-Vacuum Relief Isolation~~ (NOT USED)	~~1~~	~~6~~	~~≤ 2 x background~~	~~10^{1}-10^{6} cpm~~	~~25~~
b) RCS Leakage Detection	1	1,2,3&4	N/A	10^{1}-10^{6} cpm	24

* With fuel in the storage pool or building.
\# The plant vent noble gas monitor may also function in this capacity when the purge/pressure-vacuum relief isolation valves are open.

TABLE 3.3-6 (Continued)

TABLE NOTATION

ACTION 23 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

ACTION 24 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.7.1.

ACTION 25 - (Not Used) ~~With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.9.~~

ACTION 26 - With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirements, initiate the preplanned alternate method of monitoring the appropriate parameter(s), within 72 hours, and:

1) either restore the inoperable Channel(s) to OPERABLE status within 7 days of the event, or

2) prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

ACTION 27 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel(s) to OPERABLE status within 7 days or initiate and maintain operation of the Control Room Emergency Air Conditioning System (CREACS) in the pressurization or recirculation mode of operation. CORE ALTERATIONS and movement of irradiated fuel assemblies will be suspended during operation in the recirculation mode.

ACTION 28 - With no channels OPERABLE in a Control Room air intake, immediately initiate and maintain operation of the CREACS in the pressurization or recirculation mode of operation. CORE ALTERATIONS and movement of irradiated fuel assemblies will be suspended during operation in the recirculation mode.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNELS CHECKS	SOURCE CHECKS	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. AREA MONITORS					
a. Fuel Storage Area	S	M	R	Q	*
b. Containment Area	S	M	R	Q	1, 2, 3 & 4
2. PROCESS MONITORS					
a. Containment Monitors					
1) Gaseous Activity					
a) Purge & Pressure Vacuum Relief Isolation	S	M	R	Q	1, 2, 3, 4~~, 5 &6~~
b) RCS Leakage Detection	S	M	R	Q	1, 2, 3 & 4
2) Air Particulate Activity					
a) ~~Purge & Pressure Vacuum Relief Isolation~~ (NOT USED)	~~S~~	~~M~~	~~R~~	~~Q~~	~~1, 2, 3,& 4, & 6~~
b) RCS Leakage Detection	S	M	R	Q	1, 2, 3 & 4

*With fuel in the storage pool or building.

3/4.7.6 CONTROL ROOM EMERGENCY AIR CONDITIONING SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.6 The common control room emergency air conditioning system (CREACS)* shall be OPERABLE with:

a. Two independent air conditioning filtration trains (one from each unit) consisting of:

1. Two fans and associated outlet dampers,
2. One cooling coil,
3. One charcoal adsorber and HEPA filter array,
4. Return air isolation damper.

b. All other automatic dampers required for operation in the pressurization or recirculation modes.

c. The control room envelope intact.

APPLICABILITY: ALL MODES and during movement of irradiated fuel assemblies ~~and during CORE ALTERATIONS.~~

ACTION: MODES 1, 2, 3, and 4

a. With one filtration train inoperable, align CREACS for single filtration train operation within 4 hours, and restore the inoperable filtration train to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With CREACS aligned for single filtration train operation and with one of the two remaining fans or associated outlet damper inoperable, restore the inoperable fan or damper to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With the Control Room Envelope inoperable, restore the Control Room Envelope to OPERABLE status within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With one or both series isolation damper(s) on a normal Control Area Air Conditioning System (CAACS) outside air intake or exhaust duct inoperable, close the affected duct within 4 hours by use of at least one isolation damper secured in the closed position or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. (Refer to ACTION 27 of Table 3.3-6.)

*The CREACS is a shared system with Salem Unit 1

LIMITING CONDITION FOR OPERATION (Continued)

e. With one or both isolation damper(s) on an outside emergency air conditioning air intake duct inoperable, close the affected duct within 4 hours by use of at least one isolation damper secured in the closed position and restore the damper(s) to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

f. With any isolation damper between the normal CAACS and the CREACS inoperable, secure the damper in the closed position within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

MODES 5 and 6 or during movement of irradiated fuel assemblies ~~and during CORE ALTERATIONS.~~

a. With one filtration train inoperable, align CREACS for single filtration train operation within 4 hours, or suspend ~~CORE ALTERATIONS and~~ movement of irradiated fuel assemblies.

b. With CREACS aligned for single filtration train operation with one of the two remaining fans or associated outlet damper inoperable, restore the fan or damper to OPERABLE status within 72 hours, or suspend ~~CORE ALTERATIONS and~~ movement of irradiated fuel assemblies.

c. With two filtration trains inoperable, immediately suspend ~~CORE ALTERATIONS and~~ movement of irradiated fuel assemblies.

d. With the Control Room Envelope inoperable, immediately suspend ~~CORE ALTERATIONS and~~ movement of irradiated fuel assemblies.

e. With one or both series isolation damper(s) on a normal CAACS outside air intake or exhaust duct inoperable, immediately suspend ~~CORE ALTERATIONS and~~ movement of irradiated fuel assemblies until the affected duct is closed by use of at least one isolation damper secured in the closed position. (Refer to ACTION 27 of Table 3.3-6.)

f. With one or both series isolation damper(s) on an outside emergency air conditioning air intake duct inoperable, immediately suspend ~~CORE ALTERATIONS and~~ movement of irradiated fuel assemblies until the affected duct is closed by use of at least one isolation damper secured in the closed position. To resume ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies, at least one emergency air intake duct must be operable on each unit.

g. With any isolation damper between the CAACS and the CREACS inoperable, immediately suspend ~~CORE ALTERATIONS and~~ movement of irradiated fuel assemblies until the damper is closed and secured in the closed position.

REFUELING OPERATIONS

CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The containment building penetrations shall be in the following status:

a. The equipment hatch inside door is capable of being closed and held in place by a minimum of four bolts, or an equivalent closure device installed and capable of being closed,

b. A minimum of one door in each airlock is capable of being closed

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or
2. capable of being closed by the Containment Purge and Pressure-Vacuum Relief Isolation System.

Note: Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls.

APPLICABILITY: During ~~CORE ALTERATIONS or~~ movement of irradiated fuel within the containment.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving ~~CORE ALTERATIONS or~~ movement of irradiated fuel in the containment building. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.4.1 Each of the above required containment building penetrations shall be determined to be either in its required condition or capable of being closed by a manual or automatic containment isolation valve at least once per 7 days.

4.9.4.2 Once per refueling prior to the start of movement of irradiated fuel assemblies within the containment building, verify the capability to install, within 1 hour, the equipment hatch. Applicable only when the equipment hatch is open during movement of irradiated fuel in the containment building.

4.9.4.3 Verify, once per 18 months, each required containment purge isolation valve actuates to the isolation position on a manual actuation signal.

~~REFUELING OPERATIONS~~

~~CONTAINMENT PURGE AND PRESSURE-VACUUM RELIEF ISOLATION SYSTEM~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.9.9 The Containment Purge and Pressure-Vacuum Relief isolation system shall be OPERABLE.~~

~~APPLICABILITY: MODE 6.~~

~~ACTION:~~

~~With the Containment Purge and Pressure-Vacuum Relief isolation system inoperable, close each of the Purge and Pressure-Vacuum Relief penetrations providing direct access from the containment atmosphere to the outside atmosphere. The provisions of Specification 3.0.3 are not applicable.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.9.9 The Containment Purge and Pressure-Vacuum Relief isolation system shall be demonstrated OPERABLE within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS by verifying that containment Purge and Pressure-Vacuum Relief isolation occurs on manual initiation and on a high radiation test signal from each of the containment radiation monitoring instrumentation channels.~~

REFUELING OPERATIONS
3/4.9.12 FUEL HANDLING AREA VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.12 The Fuel Handling Area Ventilation System shall be OPERABLE with:

a. Two exhaust fans and one supply fan OPERABLE and operating, and

b. Capable of maintaining slightly negative pressure in the Fuel Handling Building.

APPLICABILITY: ~~Whenever irradiated fuel is in the storage pool.~~ During movement of irradiated fuel within the Fuel Handling Building

ACTION:

a. With no Fuel Handling Area Ventilation System OPERABLE, suspend all operations involving movement of fuel within the storage pool ~~or crane operation with loads over the storage pool~~ until the Fuel Handling Area Ventilation System is restored to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12 The above required ventilation system shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the Fuel Handling Building is maintained at a slightly negative pressure with respect to atmospheric pressure.

b. At least once per 31 days by verifying both exhaust fans and one supply fan start and operate for at least 15 minutes, if not operating already.

c. At least once per 18 months by verifying a system flowrate of 19,490 cfm ± 10% during system operation.

~~a.1. At least once per 31 days by initiating flow through the HEPA filter and charcoal adsorber train and verifying that the train operates for at least 15 minutes.~~

~~a.2 Prior to and during movement of irradiated fuel assemblies or crane operation over the storage pool in the Fuel Handling Building:~~

~~1.Both exhaust fans and one supply fan must be OPERABLE and operating with flow being directed through the HEPA and charcoal filters.~~

~~2.All dampers required to divert the entire airflow through the HEPA/charcoal filter train are OPERABLE and in the position required to divert full exhaust flow through the HEPA/charcoal filter train.~~

~~3.Ductwork, dampers and housings which will ensure all post-accident exhausted air is processed through the HEPA/charcoal filter train are intact.~~

~~4.The fuel handling area is maintained at a negative pressure equal to or more negative than 1/8 inch water gauge relative to the outside atmosphere, and~~

~~5.At least once per 24 hours thereafter verify both exhaust fans and one supply fan operating with the entire flow being directed through the HEPA and charcoal filters.~~

~~b. At least once per 18 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 system, by:~~

1. Verifying that with the ventilation system operating at a flow rate of 19,490 cfm ± 10% and exhausting through the HEPA filters and charcoal adsorbers, the total bypass flow of the ventilation system to the facility vent, including leakage through the ventilation system diverting valves, is ≤ 1% when the ventilation system is tested by admitting cold DOP at the storage pool ventilation system intake.

2. Verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas and that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place using the test procedure guidance of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978 (except for the provisions of ANSI N510 Sections 8 and 9), and the system flow rate is 19,490 cfm ± 10%.

3. Verifying within 31 days after removal from the FHV unit, that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b. of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than 5.0% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and a relative humidity of 95%.

4. Verifying a system flow rate of 19,490 cfm ± 10% during system operation.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal from the FHV unit, that a laboratory analysis of a representative carbon sample, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows a methyl iodide penetration less than 5.0% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and a relative humidity of 95%.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than or equal to 4 inches Water Gauge while operating the system at a flow rate of 19,490 cfm ± 10%.

2. Deleted.

3. Verifying that the system maintains the spent fuel storage pool area at a negative pressure of greater than or equal to 1/8 inches Water Gauge relative to the outside atmosphere during system operation.

~~e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place while operating the system at a flow rate of 19,490 cfm ± 10%.~~

~~f. After each complete or partial replacement of a charcoal absorber bank by verifying that the charcoal absorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are in-place tested while operating the system at a flow rate of 19,490 cfm ± 10%.~~

these uncertainties are factored into the determination of each Trip Setpoint. All field sensors and signal processing equipment for these channels are assumed to operate within the allowances of these uncertainty magnitudes.

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability. Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," and Supplements to that report. Surveillance intervals and out of service times were determined based on maintaining an appropriate level of reliability of the Reactor Protection System and Engineered Safety Features instrumentation.

The measurement of response time at the specified frequencies provides assurance that the protective and ESF action function associated with each channel is completed within the time limit assumed in the accident analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable.

Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either 1) in place, onsite or offsite test measurements or 2) utilizing replacement sensors with certified response times.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that 1) the radiation levels are continually measured in the areas served by the individual channels and 2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

~~The isolation alarm/trip setpoint for the Containment Purge and Pressure Relief system during MODE 6 is established to ensure that in the event of a fuel handling accident inside containment, prompt isolation will occur to ensure calculated offsite doses remain below 10CFR100 limits. Prompt isolation will also ensure that Control Room doses following a fuel handing accident will remain below GDC-19 limits. The alarm/trip setpoint value of Table 3.3-6 for the R12A while in Mode 6 will be established based upon isolating the Containment Purge and Pressure Relief System when containment gaseous activity levels reach 50% of the more conservative 10CFR20 concentration limits for release to unrestricted areas. These concentration limits are specified in 10CFR20, Appendix B, Table II, Column 1. A setpoint based on 50% of the 10CFR20 concentration limits will be low enough to ensure that prompt Containment Purge and Pressure Relief system isolation occurs during a fuel handling accident and high enough to prevent unnecessary Containment Purge and Pressure Relief system isolations caused by routine outage activities.~~

In the postulated Fuel Handling Accident, the revised dose calculations, performed using 10 CFR 50.67 and Regulatory Guide 1.183, Alternative Source Term, do not take credit for automatic containment purge isolation thus allowing for continuous monitoring of containment activity until containment closure is achieved. If required, containment purge isolation can be initiated manually from the control room.

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION (Continued)

CROSS REFERENCE - TABLES 3.3-6 and 4.3-3

T/S Table Item No.	Instrument Description	Acceptable RMs Channels
1a	Fuel Storage Area	2R5 or 2R9
1b	Containment Area	2R44A and B
2a1a	Containment Gaseous Activity Purge & Pressure/Vacuum Relief Isolation	2R12A or 2R41A, B and D(1) (2)
2a1b	Containment Gaseous Activity RCS Leakage Detection	2R12A
2a2a	~~Containment Air Particulate Activity Purge & Pressure/Vacuum Relief Isolation~~ (NOT USED)	~~2R11A~~
2a2b	Containment Air Particulate Activity RCS Leakage Detection	2R11A
2b1	Noble Gas Effluent Medium Range Auxiliary Building Exhaust System (Plant Vent)	2R45B(3)
2b2	Noble Gas Effluent High Range Auxiliary Building Exhaust System (Plant Vent)	2R45C(3)
2b3	Noble Gas Effluent Main Steamline Discharge - Safety Valves and Atmospheric Steam Dumps	2R46
2b4	Noble Gas Effluent Condenser Exhaust System	2R15
3a	Unit 2 Control Room Intake Channel 1 (to Unit 2 Monitor) Unit 2 Control Room Intake Channel 2 (to Unit 1 Monitor) Unit 1 Control Room Intake Channel 1 (to Unit 1 Monitor) Unit 1 Control Room Intake Channel 2 (to Unit 2 Monitor)	2R1B-1 1R1B-2 1R1B-1 2R1B-2

(1) The channels listed are required to be operable to meet a single operable channel for the Technical Specification's "Minimum Channels Operable" requirement.

(2) ~~For Mode 6, the setpoint applies to 2R41D using 2 x Background from 2R41A.~~ For Modes 1, 2, 3, 4 & 5, the setpoint applies to 2R41D per Specification 3.3.3.9. The measurement range applies to 2R41A and B which display in uCi/cc using the appropriate channel conversion factor form cpm to uCi/cc.

(3) If 2R45 is out of service 2R41 may be used to meet the technical specification action requirement.

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 (GDC) 19 of Appendix "A", 10 CFR Part 50. The re-analysis of the Fuel Handling Accident is based on the criteria of 10 CFR 50.67 and Reg. Guide 1.183, Alternative Source Term, which replace GDC 19 for Control Room doses.

3/4.7.7 AUXILIARY BUILDING EXHAUST AIR FILTRATION SYSTEM

The Auxiliary Building Ventilation System (ABVS) consists of two major subsystems. They are designed to control Auxiliary Building temperature during normal and emergency modes of operation, and to contain Auxiliary Building airborne contamination during Loss of Coolant Accidents (LOCA). The two subsystems are:

1. A once through filtration exhaust system, designed to contain particulate and gaseous contamination and prevent it from being released from the building in accordance with 10CFR20, and

2. A once through air supply system, designed to deliver outside air into the building to maintain building temperatures within acceptable limits. For the purposes of satisfying the Technical Specification LCO, one supply fan must be administratively removed from service such that the fan will not auto-start on an actuation signal; however, the supply fan must be OPERABLE with the exception of this administrative control.

These systems operate during normal and emergency plant modes. Additionally, the system provides a flow path for containment purge supply and exhaust during Modes 5 and 6. Either the Containment Purge system or the Auxiliary Building Ventilation System with suction from the containment atmosphere, with associated radiation monitoring will be available whenever movement of irradiated fuel is in progress in the containment building and the equipment hatch is open. If for any reason, this ventilation requirement can not be met, movement of fuel assemblies within the containment building shall be discontinued until the flow path(s) can be reestablished or close the equipment hatch and personnel airlocks.

The exhaust system consists of three 50% capacity fans that are powered from vital buses. These fans exhaust from a common plenum downstream from three High Efficiency Particulate Air (HEPA) filter banks, two of which, 21 & 22 can be interchangeably aligned to discharge to a single carbon adsorber bed. Filter unit 21 is limited in capacity and can only be aligned to the ECCS areas of the Auxiliary Building for HEPA only or HEPA + Carbon modes of filtration. Filter unit 22 can be used to ventilate the normal areas of the Auxiliary Building in HEPA only, or when used in conjunction with 23, may be used to ventilate the ECCS areas of the Auxiliary Building in HEPA + Carbon. Filter unit 23 does not communicate with the carbon adsorber housing and is used for exhausting air from the normal areas of the Auxiliary Building during any plant Mode or purging the Containment Building during Modes 5&6. The fans are designed for continuous operation, to control the Auxiliary Building pressure at -0.10" Water Gauge with respect to atmosphere.

The supply system consists of two 100% capacity fans that are powered from vital buses, and distribute outdoor air to the general areas and corridors of the building through associated ductwork.

3/4.9.1 BORON CONCENTRATION

The limitations on minimum boron concentration (2000 ppm) ensure that: 1) the reactor will remain subcritical during CORE ALTERATIONS, and 2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. The limitation on K_{eff} of no greater than 0.95 which includes a conservative allowance for uncertainties, is sufficient to prevent reactor criticality during refueling operations.

The sampling and analysis required by surveillance requirement 4.9.1.2 ensures the boron concentration required by Limiting Condition of Operation 3.9.1 is met. Sampling and analysis of the refueling canal is required if water exists in the refueling canal, regardless of the amount.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the source range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the accident analyses.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

During ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies within containment the requirements for containment building penetration closure capability and OPERABILITY ensure that a release of fission product radioactivity within containment will ~~be restricted from leaking to the environment~~ not exceed the guidelines and dose calculations described in Reg Guide 1.183, Alternative Radiological Source Term for Evaluating Design Basis Accidents at Nuclear Power Plants. In MODE 6, the potential for containment pressurization as a result of an accident is not likely. Therefore, the requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements during ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies within containment are referred to as "containment closure" rather than containment OPERABILITY. For the containment to be OPERABLE, CONTAINMENT INTEGRITY must be maintained. Containment closure means that all potential release paths are closed or capable of being closed. Closure restrictions ~~must be sufficient to provide an atmospheric ventilation barrier to restrict radioactive material released from a fuel element rupture during refueling operations.~~ include the administrative controls to allow the opening of both airlock doors and the equipment hatch during fuel movement provided that: 1) the equipment inside door or an equivalent closure device installed is capable of being closed with four bolts within 1 hour by a designated personnel; 2) the airlock doors are capable of being closed within 1 hour by designated personnel ,3) either the Containment Purge System or the Auxiliary Building Ventilation System taking suction from the containment atmosphere are operating and 4) the plant is in Mode 6 with at least 23 feet of water above the reactor pressure vessel flange.

Administrative requirements are established for the responsibilities and appropriate actions of the designated personnel in the event of a Fuel Handling Accident inside containment. These requirements include the responsibility to be able to communicate with the control room, to ensure that the equipment hatch is capable of being closed, and to close the equipment hatch and personnel airlocks within 1 hour in the event of a fuel handling accident inside containment. These administrative controls ensure containment closure will be established in accordance with and not to exceed the dose calculations performed using guidelines of Regulatory Guide 1.183.

The containment serves to limit the fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained well within the requirements of 10CFR100 and Reg Guide 1.183, Alternative Source Term, as applicable. Additionally, the containment provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The Containment Equipment Hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into or out of containment. During ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies within containment can be open provided that: 1) it is capable of being closed with four bolts within 1 hour by designated personnel, 2) either the Containment Purge System or the Auxiliary Building Ventilation System taking suction from the containment atmosphere are operating and 3) the plant is in Mode 6 with at least 23 feet of water above the reactor pressure vessel flange.~~the Containment Equipment Hatch inside door must be held in place by at least four bolts.~~ Good engineering practice dictates that the bolts required by the LCO are approximately equally spaced.

An equivalent closure device may be installed as an alternative to installing the Containment Equipment Hatch inside door with a minimum of four bolts. Such a closure device may provide penetrations for temporary services used to support maintenance activities inside containment at times when containment closure is required; and may be installed in place of the Containment Equipment Hatch inside door or outside door. Penetrations incorporated into the design of an equivalent closure device will be considered a part of the containment boundary and as such will be subject to the requirements of Technical Specification 3/4.9.4. Any equivalent closure device used to satisfy the requirements of Technical Specification 3/4.9.4.a will be designed, fabricated, installed, tested, and utilized in accordance with established procedures to ensure that the design requirements for the mitigation of a fuel handling accident during refueling operations are met. In case that this equivalent closure device is installed in lieu of the equipment hatch inside door, the same restrictions and administrative controls apply to ensure closure will take place within 1 hour following a Fuel Handling Accident inside containment.

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during operation in MODES 1, 2, 3, and 4 as specified in LCO 3.6.1.3, "Containment Air Locks". Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of unit shutdown, when containment closure is not required and frequent containment entry is necessary, the air lock interlock mechanism may be disabled. This allows both doors of an airlock to remain open for extended periods. During ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies within containment, containment closure may be required; therefore, the door interlock mechanism may remain disabled, ~~but one air lock door must always remain closed~~ and both doors of each containment airlock may be open if: 1) At least one door of each airlock is capable of being closed within 1 hour by dedicated personnel,) 2) either the Containment Purge System or the Auxiliary Building Ventilation System taking suction from the containment atmosphere are operating and 3) The plant is in Mode 6 with at least 23 feet of water above the reactor pressure vessel flange.

In the postulated FHA, the revised dose calculations performed using RG 1.183 criteria, do not assume automatic containment purge isolation thus allowing for continous monitoring of containment acitivity until the release pathways are isolated. If required, manual isolation of containment purge can be initiated from the control room.

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side. Isolation may be achieved by an OPERABLE automatic isolation valve, or by a manual isolation valve, blind flange, or equivalent. Equivalent isolation methods may include the use of a material that can provide a temporary atmospheric pressure, ventilation barrier. Any equivalent method used to satisfy the requirements of Technical Specification 3/4.9.4.c.1 will be designed, fabricated, installed, tested, and utilized in accordance with established procedures to ensure that the design requirements for the mitigation of a fuel handling accident during refueling operations are met.

The surveillance requirement 4.9.4.2 demonstrates that the necessary hardware, tools, and equipment are available to close the equipment hatch. The surveillance is performed once per refueling prior to the start of movement of irradiated fuel assemblies within the containment. This surveillance is only required to be met when the equipment hatch is open.

~~The OPERABILITY of the containment purge system ensures that the containment vent and purge penetrations will be automatically isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment~~

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during CORE ALTERATIONS.

3/4.9.6 MANIPULATOR CRANE

The OPERABILITY requirements for the manipulator cranes ensure that: 1) manipulator cranes will be used for movement of control rods and fuel assemblies, 2) each crane has sufficient load capacity to lift a control rod or fuel assembly, and 3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE BUILDING

The restriction on movement of loads in excess of the nominal weight of a fuel and control rod assembly and associated handling tool over other fuel assemblies in the storage pool ensures that in the event this load is dropped (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the accident analyses.

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirements that at least one residual heat removal loop be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification. A minimum flow rate of 1000 gpm is required. Additional flow limitations are specified in plant procedures, with the design basis documented in the Salem UFSAR. These flow limitations address the concerns related to vortexing and air entrapment in the Residual Heat Removal system, and provide operational flexibility by adjusting the flow limitations based on time after shutdown. The requirement to have two RHR loops OPERABLE when there is less than 23 feet of water above the reactor vessel flange ensures that a single failure of the operating RHR loop will not result in a complete loss of residual heat removal capability.

For support systems: Service Water (SW) and Component Cooling (CC), component redundancy is necessary to ensure no single active component failure will cause the loss of Decay Heat Removal. One piping path of SW and CC is adequate when it supports both RHR loops. The support systems needed before entering into the desired configuration (e.g., one service water loop out for maintenance in Modes 5 and 6) are controlled by procedures, and include the following:

- A requirement that the two RHR, two CC and two SW pumps, powered from two different vital buses be kept operable
  A listing of the active (air/motor operated) valves in the affected flow path to be locked open or disable.

Note that four filled reactor coolant loops, with at least two steam generators with at least their secondary side water level greater than or equal to 5% (narrow range), may be substituted for one residual heat removal loop. This ensures that single failure does not cause a loss of decay heat removal.

With the reactor vessel head removed and 23 feet of water above the reactor pressure vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core.

3/4.9.9 (Not Used)~~CONTAINMENT PURGE AND PRESSURE-VACUUM RELIEF ISOLATION SYSTEM~~

~~The OPERABILITY of this system ensures that the containment vent and purge penetrations will be automatically isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.~~

3/4.9.10 and 3/4/9/11 WATER LEVEL - REACTOR VESSEL AND STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the accident analysis.

3/4.9.12 FUEL HANDLING AREA VENTILATION SYSTEM

~~The limitations on the fuel handling area ventilation system ensure that all radioactive material released from a dropped irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. The OPERABILITY of this system is consistent with the assumptions of the accident analyses. Laboratory testing of the carbon adsorber is performed in accordance with ASTM D3803-1989 with an acceptance criteria that is determined by applying a minimum safety factor of 2 to the charcoal filter removal efficiency credited in the design basis dose analysis as specified in Generic Letter 99-02.~~

The operability of the Fuel Handling Area Ventilation System during movement of irradiated fuel ensures that a release of fission product radioactivity within the Fuel Handling Building will not exceed the guidelines and dose calculations described in Reg. Guide 1.183, Alternative Radiological Source Term for Evaluating Design Basis Accidents at Nuclear Power Reactors~~all building exhaust flow is processed through the HEPA/charcoal filter train whenever a Fuel Handling Accident is possible. This will minimize offsite doses following the postulated Fuel Handling Accident.~~