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November 13, 2009

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

Oyster Creek Nuclear Generating Station
Renewed Facility Operating License No. DPR-16
Docket No. 50-219

Subject: Response to Request for Additional Information – Application to Revise Technical Specifications Regarding Secondary Containment Operability Requirements During Refueling

References:

1) Letter from Pamela B. Cowan to U.S. Nuclear Regulatory Commission Technical Specification Change Request 338 - Secondary Containment Operability Requirements During Refueling, dated November 2, 2007

2) U.S. Nuclear Regulatory Commission facsimile dated March 25, 2008, Draft Request for Additional Information (RAI) Regarding Proposed License Amendment - Secondary Containment Operability Requirements During Refueling, Oyster Creek Nuclear Generating Station (Docket No. 50-219)

3) Letter from Pamela B. Cowan to U.S. Nuclear Regulatory Commission Response to Draft Request for Additional Information - AmerGen Application to Revise Technical Specifications Regarding Secondary Containment Operability Requirements During Refueling, dated May 5, 2008

4) Letter from Pamela B. Cowan to U.S. Nuclear Regulatory Commission Supplemental Response - AmerGen Application to Revise Technical Specifications Regarding Secondary Containment Operability Requirements During Refueling, dated July 3, 2008

5) U.S. Nuclear Regulatory Commission facsimile dated September 11, 2008, Draft Request for Additional Information (RAI) Regarding Proposed License Amendment - Secondary Containment Operability Requirements During Refueling, Oyster Creek Nuclear Generating Station (Docket No. 50-219)

6) Letter from Pamela B. Cowan to U.S. Nuclear Regulatory Commission Response to Request for Additional Information Supplemental Response - AmerGen Application to Revise Technical Specifications Regarding Secondary Containment Operability Requirements During Refueling, dated September 22, 2008

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Attachment 8 transmitted herewith contains Sensitive Unclassified Non-Safeguards Information (SUNSI). When separated from Attachment 8, this transmittal document is decontrolled.

7) Letter from G. Edward Miller, U.S. Nuclear Regulatory Commission to Mr. Charles G. Pardee, Exelon Generation Company, LLC, dated October 20, 2009, Oyster Creek Nuclear Generating Station – Request for Additional Information Regarding Secondary Containment Operability License Amendment Request (TAC No. MD7261)

By letter dated November 2, 2007 (Reference 1), Exelon Generation Company, LLC (Exelon) (formerly AmerGen Energy Company, LLC) submitted a License Amendment Request (LAR) to revise the Oyster Creek Nuclear Generating Station (OCNGS) Technical Specifications (TS) to modify the requirements for Secondary Containment operability during handling of irradiated fuel with sufficient decay.

Exelon provided additional information by letters dated May 5, 2008 (Reference 3) and September 22, 2008 (Reference 6), in response to U.S. Nuclear Regulatory Commission (NRC) requests for additional information (References 2 and 5) concerning this LAR. In addition, Exelon also provided supplemental information in a letter dated July 3, 2008 (Reference 4).

Subsequently, by letter dated October 20, 2009 (Reference 7), the NRC requested additional information in order to complete the review of the LAR. Attachment 1 to this letter restates the NRC's questions followed by Exelon's response.

Attachment 2 to this letter includes a revised Section 2.0, *"Proposed Changes,"* which supersedes the information previously submitted in Section 2.0 of the November 2, 2007, submittal (Reference 1). Attachment 3 to this letter contains the revised TS mark-ups, and Attachment 4 includes the revised TS Bases mark-ups. These mark-ups were modified to address the NRC concern with needing additional definition in the TS. Attachment 5 includes a copy of Calculation C-1302-822-E310-081, Revision 1, *"Oyster Creek Onsite Atmospheric Dispersion (X/Q) for Fuel Handling Accident (FHA)."* Attachment 6 includes a copy of Calculation C-1302-822-E310-082, Revision 2, *"Oyster Creek Analysis of Fuel Handling Accident (FHA) Using Alternative Source Terms (AST)."* Attachment 7 includes a revised Section 5.0, "Regulatory Analysis," which supersedes the information previously submitted in Section 5.0 of the November 2, 2007, submittal (Reference 1). Attachment 8 includes annotated drawings showing the evaluated release locations. The information included in Attachment 8 is considered *"Sensitive Unclassified Non-Safeguards Information"* (SUNSI) and should be withheld in accordance with 10 CFR 2.390.

Exelon has determined that the information provided in this response further supplements the information provided in the Technical Analysis in the original submittal (Reference 1) and supporting supplemental responses (References 3, 4, and 6), and it does not impact the Environmental Consideration previously submitted. However, as indicated above, Exelon has revised Section 2.0, "Proposed Changes," and Section 5.0, "Regulatory Analysis," of the Reference 1 submittal to reflect the analysis provided in this response.

There are no regulatory commitments contained in this submittal.

If any additional information is needed, please contact Mr. Richard Gropp at 610-765-5557.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 13th day of November 2009.

Respectfully,

Pamela B. Cowan
Director, Licensing and Regulatory Affairs
Exelon Generation Company, LLC

Attachments:
1) Response to NRC Request for Additional Information
2) Revised License Amendment Request Section 2.0, “Proposed Changes”
3) Revised Mark-ups of Technical Specifications Pages
4) Revised Mark-ups of Technical Specifications Bases Pages
5) Calculation C-1302-822-E310-081, Revision 1, *"Oyster Creek Onsite Atmospheric Dispersion (X/Q) for Fuel Handling Accident (FHA)"*
6) Calculation C-1302-822-E310-082, Revision 2, *"Oyster Creek Analysis of Fuel Handling Accident (FHA) Using Alternative Source Terms (AST)"*
7) Revised License Amendment Request Section 5.0, “Regulatory Analysis”
8) Annotated Drawings Showing Evaluated Release Locations (*contains SUNSI*)

cc:		
	Regional Administrator - NRC Region I	w/o Attachment 8
	NRC Senior Resident Inspector - OCNGS	“
	NRC Project Manager, NRR - OCNGS	“
	Director, Bureau of Nuclear Engineering, New Jersey Department of Environmental Protection	“

ATTACHMENT 1

Oyster Creek Nuclear Generating Station

Response to NRC Request for Additional Information
Secondary Containment Operability Requirements During Refueling

Background

By letter dated November 2, 2007 (Reference 1), Exelon Generation Company, LLC (Exelon) (formerly AmerGen Energy Company, LLC) submitted a License Amendment Request (LAR) to revise the Oyster Creek Nuclear Generating Station (OCNGS) Technical Specifications (TS) to modify the requirements for Secondary Containment operability during handling of irradiated fuel with sufficient decay.

Subsequently, by letter dated October 20, 2009 (Reference 2) the U.S. Nuclear Regulatory Commission (NRC) requested additional information in order to complete the review of the License Amendment Request (LAR). The specific questions are restated below followed by Exelon's response.

NRC Question 1

In the November 2, 2007, submittal, Attachment 2, Table 1, "Conformance with Regulatory Guide (RG) 1.183 Main Sections," states that the submittal "conforms" to Regulatory Position 5.1.2 and that "[t]he analysis takes no credit for safety related features." Later correspondence contained in calculation C-1302-822-E31 0-082, Revision 1, continues to state conformance to RG 1.183.[1]

Regulatory Position 5.1.2 states:

> *Credit may be taken for accident mitigation features that are classified as safety-related, are required to be operable by technical specifications, are powered by emergency power sources, and are either automatically actuated or, in limited cases, have actuation requirements explicitly addressed in emergency operating procedures.*

In the September 22, 2008, response to a Request for Information, Exelon stated:

> *AmerGen [Exelon] will update the UFSAR [Updated Final Safety Analysis Report] to include the acceptable secondary containment penetrations and openings that could be breached/opened while moving irradiated fuel with sufficient decay ... Any additional penetrations and openings not included in the UFSAR (as outlined in Table 1 below in response to NRC Question 2) must be analyzed in accordance with applicable regulatory requirements (i.e., 10CFR50.59) before relaxation of secondary containment requirements for movement of irradiated fuel with sufficient decay. The method of evaluation used will demonstrate that radiological consequences associated with the Fuel Handling Accident (FHA) do not exceed applicable regulatory dose limits.*

Based upon the above, the statement that, "[t]he analysis takes no credit for safety related features," and the stated conformance to Regulatory Position 5.1.2 appear to be inconsistent. By proposing a limited number of acceptable penetrations and openings that can be breached, Exelon credits the capability of any remaining secondary containment accident mitigation features as being capable of performing their safety functions for the analyzed conditions for the duration of their mission times.[2] *However, the licensee's proposed technical specification (TS) changes remove all requirements for all secondary containment accident mitigative features after 24 hours. Instead, Exelon proposes that the secondary containment mitigative features are to be established in the UFSAR. Exelon's proposed deletion of TSs associated with secondary* *containment operability and incorporation of controls in the UFSAR is not consistent with Regulatory Position 5.1.2. In accordance with 10 CFR 50.36, "Technical specifications," Exelon's proposed continued reliance on some safety-related features of secondary containment to function or actuate to mitigate a design-basis accident necessitates their inclusion in the TSs.*

The TSs proposed in the original LAR, which were not amended in the July 3, 2008, supplement or the September 22, 2008, RAI response, are insufficient for the NRC staff to find that the licensee has provided the lowest functional capability or performance level of equipment for safe operation of the facility that would provide reasonable assurance that, in the event of an FHA when secondary containment is INOPERABLE, the dose consequences will meet NRC regulatory requirements.

Therefore, the NRC staff requests that the licensee provide revised TS changes, consistent with its proposed revised analysis of record, that ensure the lowest functional capability or performance level of equipment credited for functioning or actuating to mitigate the design basis fuel handling accident.

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[1] *According to calculation, C-1302-822-E31 0-082, Revision 1: 1) "This calculation determines the safety features required to assure that regulatory limits in 1OCFR50.67 are met, and is performed in conformance with guidance for analysis of this event provided in Regulatory Guide (RG) 1.183, Appendix B." 2) "Dose models for both onsite and offsite are simplified and meet RG 1.183 requirements," and 3) "This analysis uses Alternative Source Term (AST) assumptions per guidance in RG 1.183."*

[2] *Per page 14 of calculation C-1302-822-E31 0-082, Revision 1 several structures and components are part of the primary success path and function to mitigate the Fuel Handling Accident. Specifically, Exelon states that the Commodities Penetration on the RB North Wall, MAC Facility Personnel Airlock, MAC Facility Entrance, Trunion Room Door to Turbine Building are credited in the analysis.*

<u>Response – NRC Question 1</u>

Exelon has determined it appropriate to revise the Technical Specifications (TS) and TS Bases pages to reflect that the locations (i.e., doors and hatches that do not involve disassembly of the Secondary Containment) as described in Table 1 of this attachment can be opened during movement of irradiated fuel provided that there is sufficient decay of the irradiated fuel. This decay period has been demonstrated in a design analysis to be 168 hours, the time at which Secondary Containment and Standby Gas Treatment System (SGTS) are no longer required to meet dose limitations for the Control Room (CR) and offsite locations. However, one SGTS circuit will be available to be used to draw any potential release from a postulated fuel handling accident in the proper direction such that it can be treated and monitored. The only function required during this shutdown condition with fuel movement in progress with sufficient decay is for monitoring purposes and as a defense-in-depth measure, thereby further mitigating any possible dose to the public in the event of a fuel handling accident. Local grab samples at Secondary Containment openings may be employed as additional monitoring for radioactive airborne activity within the Secondary Containment. Provisions will be provided to ensure Secondary Containment openings as described in Table 1 of this attachment can be closed within one hour.

Exelon considers that these proposed changes are consistent with Regulatory Position 5.1.2. The proposed TS and TS Bases changes are provided in Attachments 3 and 4 of this submittal.

NRC Question 2

In NRC Regulatory Issue Summary 2006-04, "Experience with Implementation of Alternative Source Terms," the NRC reiterated its regulatory position that "Licensees are responsible for identifying all release pathways and for considering these pathways in their AST analyses, consistent with any proposed modification." During the course of the review, which includes its supplements and RAI responses, the licensee has provided three separate lists of analyzed or considered release points and pathways. In reviewing these lists, the staff has identified variations that raise concerns regarding whether the licensee has analyzed all potential release points and pathways to ensure that regulatory dose limits would be met in the event of an FHA. Consistent with NRC's established regulatory position, the NRC staff requests that the licensee provide a comprehensive list of all analyzed and unanalyzed secondary containment potential release points and pathways to the environment and control room. These pathways should include those pathways to adjacent buildings that could lead to the environment or to the control room (i.e. Secondary Containment HVAC ductwork, structural openings etc.). Additionally, the licensee's evaluation of the pathways should consider the effects of operability or inoperability of other safety systems such as the Secondary Containment Isolation Valves and Standby Gas Treatment System (SGTS). For each potential release point or pathway, the licensee should provide the following:

a. The results of its dose analysis demonstrating that 10 CFR 50.67 regulatory limits are met;

b. If a dose analysis has not been performed, a technically sound basis for why this release point or pathway is bounded by other analyzed release points; and

c. An explanation for how the existing proposed TS changes or, as necessary, new revised TSs will ensure that the dose limits are met.

Response – NRC Question 2

Exelon proposes to allow doors, penetrations, and hatch openings which do not require disassembly of the Secondary Containment to be opened during shutdown conditions with fuel movement in progress. These opening are described in Table 1 of this attachment, with one exception (i.e., Northwest RB Personnel Airlock to Office Building at Elev. 51'-3"). With respect to those doors and penetrations, these items are, and have always been, credited as being part of Secondary Containment. There are no new systems or components being credited and qualified as such in this analysis. One SGTS circuit will be available to be used to draw any potential release from a postulated fuel handling accident in the proper direction such that it can be treated and monitored. The only function required during this shutdown condition with fuel movement in progress with sufficient decay is for monitoring purposes and as a defense-in-depth measure, thereby further mitigating any possible dose to the public in the event of a fuel handling accident. Local grab samples at Secondary Containment openings may be employed as additional monitoring for radioactive airborne activity within the Secondary Containment. Provisions will be provided to ensure Secondary Containment openings can be closed within one hour.

The results of the dose analysis are dependent on many bounding analysis factors and assumptions as indicated in Exelon's July 3, 2008, supplemental response and included in Calculation C-1302-822-E310-082, Revision 1, *"Oyster Creek Analysis of Fuel Handling Accident (FHA) Using Alternative Source Terms (AST),"* and Revision 2 (Attachment 6). The varying factor for potential results is based on release pathway X/Q values, which is a function of location orientation in respect to the receptor points of the CR, Exclusion Area Boundary (EAB), and Low Population Zone (LPZ). The worst-case X/Q for any opening permitted to be open is used in the revised Calculation C-1302-822-E310-081, Revision 1, *"Oyster Creek Onsite Atmospheric Dispersion (X/Q) for Fuel Handling Accident (FHA)"* (Attachment 5). The calculation has determined that a 168-hour decay time is required for permitting openings, without the requirement of Secondary Containment or SGTS filtration. After this time, penetrations can be opened in accordance with Table 1 of this attachment with acceptable dose consequences.

The airlock on the northwest corner (51'-3" elevation) of the Reactor Building provides for a means of ingress and egress from the Reactor Building to the Office complex area (46'-6" elevation). Both doors are closed to provide the boundary except during transit. In addition, even during transit, the airlock is provided with two doors that have interlocks so that only one door will be opened at a time. This ensures that the Secondary Containment boundary is maintained, as required. The airlock enclosure is completely surrounded by concrete (walls, floors, and ceiling). The airlock is shown in more detail on one of the drawings (BR 4054) contained in Attachment 8. The CR is located on the 46'-6" elevation in the Turbine Building (northeast corner) adjacent to the Office complex. The CR has its own Heating, Ventilation and Air Conditioning (HVAC) system and is designed to maintain the CR at a pressure slightly higher than atmospheric pressure. This is a requirement specified in current OCNGS TS 4.17 B.

Those release locations considered in the analysis are described in Table 1 of this attachment.

NRC Question 3

In Table 4-3, "Parameters Applicable to AST Fuel Handling Accident Dose Considerations for Oyster Creek Nuclear Generating Station," of the July 3, 2008, supplement, Exelon states that no credit is taken for filtration by the SGTS. However, in the same supplement, Exelon provides a commitment to provide "...prompt methods...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." This appears to be a reference to the SGTS. Based on the proposed TS changes from the original LAR, it does not appear that the SGTS will be required to be operable during non-recently irradiated fuel handling operations (i.e., after the reactor has been subcritical for 24 hours). As such, it would not be available for performing the safety function described in the commitment. With the SGTS inoperable, the licensee would be unable to create the differential pressure inside the secondary containment necessary for the purposes of directing the radioactive release from a fuel handling accident through the SGTS filtration and the Main Stack. As such, other potential release points or pathways, such as smaller secondary containment penetrations and pathways the licensee has determined must remain closed even after 24 hours of decay time, that have not been analyzed or considered may become relevant dose contributors. Therefore, the NRC staff requests the licensee provide the following information:

a. *An analysis demonstrating that the SGTS can perform its safety function under all possible plant configurations related to secondary containment operability during fuel handling operations. This analysis should consider the potential impacts of differential pressures caused by local wind conditions.*

b. *Appropriate TS related to SGTS operability during periods when it is credited for performing a safety function related to the mitigating the consequences of an FHA.*

Response – NRC Question 3

The proposed TS require that one SGTS circuit be available within one hour after a fuel handling accident; therefore, the "prompt methods" as committed in Exelon's LAR submitted by letter dated November 2, 2007, can be satisfied. However, Exelon will not credit for the filtration or elevated release if any Secondary Containment penetration (e.g., door, hatch, or otherwise) is open during fuel movement.

Given the 168-hour delay prior to allowing penetrations to be open with the exception of the airlock as described (Northwest RB Personnel Airlock to Office Building at Elev. 51'-3"), SGTS is no longer required, as the exhaust is assumed to exit the building at the worst permitted release locations. This includes the diffuse area of the reactor building wall.

NRC Question 4

Based on the differences identified between the licensee's analyses and its statements regarding conformance with Regulatory Position 5.1.2 discussed in question #1, the NRC staff requests that the licensee reevaluate its conformance with Regulatory Position 5.1.2 and provide additional justification that all credited accident mitigation features are classified as safety-related, are required to be operable by TSs, are powered by emergency power sources, and are either automatically actuated or, in limited cases, have actuation requirements explicitly addressed in emergency operating procedures.

Response – NRC Question 4

There are no new accident mitigation features credited as a result of this LAR. The credited accident mitigation features are classified as safety-related, are required to be operable by TS, are powered by emergency power sources, and are automatically actuated. Secondary Containment openings as described in Table 1 of this attachment are permitted to be open during movement of irradiated fuel with a minimum of 168 hours decay. These Secondary Containment openings will be capable of being closed within one hour after a fuel handling accident. The exception identified in Table 1 of this attachment (i.e., Item 15 – Northwest RB Personnel Airlock to Office Building at Elev. 51'-3") is a safety-related Secondary Containment boundary and will remain closed.

NRC Question 5

Exelon has proposed the following commitment in the July 3, 2008, supplement:

> *Plant procedures will continue to require that secondary containment integrity be maintained when handling heavy loads (greater than one fuel assembly), such as the reactor vessel head or dryer/separator assembly, over the reactor cavity with fuel in the reactor vessel.*

Currently, TS requirements 3.5.B.1.c, 3.5.B.1.d and 3.5.B.1.e (consistent with 10 CFR 50.36, Criterion 3) exist to require secondary containment integrity when heavy loads could cause a release of radioactive materials (i.e. reactor vessel head is on, operations are not being performed in, above, or around the spent fuel pool that could cause release of radioactive materials, etc.). Exelon proposes to delete or modify these TS requirements. However, the licensee has not provided a technical justification for why these controls are no longer required to account for an FHA resulting from the potential drop of a heavy load. Such an accident has the potential to result in greater fuel damage and radioactive release than that assumed in the license amendment. As such, the NRC staff request that the licensee provide a technical justification for why these limiting conditions for operation are not required to establish the lowest functional capability or performance levels for equipment required for safe operation of the facility (in accordance with 10 CFR 50.36) for movement of heavy loads over the reactor cavity or spent fuel pool.

<u>Response – NRC Question 5</u>

Exelon modified TS Section 3.5.B, *"Secondary Containment,"* regarding the need to maintain Secondary Containment integrity during movement of heavy loads in, above, or around the reactor cavity and spent fuel pool. Refer to the discussion in Attachment 2 of this submittal regarding the proposed TS changes for controlling heavy loads over the reactor cavity and spent fuel pool. Attachment 3 contains the proposed TS mark-ups.

<u>NRC Question 6</u>

In the November 2, 2007, submittal, Section 2 provides the proposed changes. No justification is provided for the change labeled 2.4 and additional justification is needed for changes 2.3, 2.5, 2.6 and 2.8. Many of the proposed changes cite a conformance with TSTF-51, Rev. 2 as the justification. However, the licensee's subsequent revisions to the original LAR reduce its consistency with TSTF-51, Rev. 2. Therefore, the NRC staff requests that the licensee provide further detailed justification for each proposed change.

Note: The NRC staff recognizes that the licensee's response to the RAIs above may result in significant changes to the TSs proposed in the November 2, 2007, submittal. Substantial changes could invalidate or render moot the original justification for the proposed changes. In that case, the NRC staff encourages the licensee to completely revise Section 2.0, "Proposed Changes" and submit a new Section 2.0 which includes a clear technical justification for each proposed change.

<u>Response – NRC Question 6</u>

An updated Section 2.0 describing and justifying the proposed TS changes is provided in Attachment 2. The information contained in Attachment 2 of this submittal supersedes the information provided in Section 2.0 of Exelon's November 2, 2007, LAR submittal.

The justification for the proposed changes is based on the revised analysis submitted in Exelon's supplemental response dated July 3, 2008, and included in Calculation C-1302-822-E310-082, Revision 2, *"Oyster Creek Analysis of Fuel Handling Accident (FHA) Using Alternative Source Terms (AST),"* and Calculation C-1302-822-E310-081, Revision 1, *"Oyster Creek Onsite Atmospheric Dispersion (X/Q) for Fuel Handling Accident (FHA)."* These revised calculations are included in Attachments 5 and 6 of this submittal.

NRC Question 7

In its amendment request, Exelon assumed that all radioactivity will enter the control room through the HVAC intake ductwork. However, no technical basis for that assumption was provided. The NRC staff requests that the licensee provide a justification for this assumption. As part of its justification, the staff requests that the licensee reevaluate whether release pathways exist from the secondary containment into buildings connected to the control room. If such pathways exist, the staff requests that the licensee justify why the atmospheric dispersion factors used in its analysis are limiting. Finally, the staff requests that Exelon provide scale drawings showing the relationship of the secondary containment to the control room.

Response – NRC Question 7

All radioactivity released during the FHA event is assumed to enter the CR through the CR HVAC intake louvers. This is conservative since the intake louvers are closer to the sources (than the CR itself), having higher X/Q values for any permitted opening. Therefore, it is conservative to use these X/Q values. There is one exception identified in Table 1 of this attachment (i.e., Item 15 – Northwest RB Personnel Airlock to Office Building at Elev. 51'-3"). This airlock must remain closed and this Secondary Containment boundary maintained during fuel movement.

As requested, Attachment 8 contains drawings that show the relationship of the Secondary Containment to the CR.

NRC Question 8

The NRC staff requests that the licensee explain how the monitoring of radioactive releases resulting 'from an FHA or "inadvertent release of radioactive material" (GDC 63 and 64) will be accomplished with the secondary containment open. The current licensing basis for Oyster Creek assumes the secondary containment is operable during fuel handling operations and by extension would be operable during an FHA or an "inadvertent release of radioactive material." As such, any radiation monitoring and filtering equipment inside secondary containment would have been designed, located, and calibrated based on the current design and licensing basis. The proposed changes could impact the effectiveness of that monitoring equipment. For example, the timing of proceduralized operator actions related to indications or alarms from this equipment could potentially be delayed or prevented by a reduced effectiveness of this equipment. The staff believes the ability to effectively monitor the radioactive release is critical to the protection of the public and plant personnel.

Response – NRC Question 8

Since one SGTS circuit will be available and running within one hour and all Secondary Containment openings will be capable of being closed within one hour, this will ensure that any release from a postulated fuel handling accident will be directed in such a manner that it can be treated and monitored. Local grab samples at Secondary Containment openings may be employed as additional monitoring for radioactive airborne activity within the Secondary Containment. Provisions will be provided to ensure Secondary Containment openings can be closed within one hour.

NRC Question 9

Regulatory Guide 1.194, "Atmospheric Relative Concentrations for Control Room Habitability Assessments at Nuclear Power Reactors," states that:

> *Diffuse source modeling should be used only for those situations in which the activity being released is homogenously distributed throughout the building and when the assumed release rate from the building surface would be reasonably constant over the surface of the building.*

The release from the reactor cavity and spent fuel pool is to the area in the reactor building that is above elevation (El.) 119'-3". The reactor building is constructed entirely of reinforced concrete to the refueling floor level at El. 119'-3". Above the refueling floor, the structure is steel framework with insulated, corrosion-resistant metal siding. Because of the differences in construction, leakage appears to be more likely from the secondary containment above the refueling floor than from the secondary containment below the refueling floor. Therefore, the NRC staff requests that the licensee justify the use of the entire exposed area of the reactor building for calculation of the reactor building diffuse source rather than using only the area of the building above the refueling floor where the materials of construction would be more likely to have release pathways to the environment. Additionally, the staff requests that the licensee provide a technical basis for its assumption that the activity being released will be homogenously distributed throughout the building and that the release rate from the building surface will be reasonably constant over the surface of the building.

Response – NRC Question 9

A new diffuse area source has been determined using part of the metal siding surface area facing the CR HVAC intakes. Since a reasonable amount of mixing is anticipated, considering that temperature differences within the volume will cause turbulence, only 50% of this surface area is considered. Furthermore, this surface area is selected such that the resulting X/Q is maximized with respect to the worst-case CR intake location.

Due to the normal ventilation in effect during fuel movement the activity is conservatively distributed throughout the fuel handling area in determination of the diffuse area source X/Q. However, for the purposes of this analysis, the released activity is only distributed into a volume of 100 ft^3.

The metal siding above the 119'-3" elevation is sealed, insulated, routinely tested via drawdown testing, and not expected to leak. The previous analysis included the entire surface area of the reactor building west wall in the consideration of a diffuse area source. A revised diffuse area

<u>NRC Question 10</u>

Are any non-safety related systems and components credited in the alternate source term analyses? If so:

a. *Describe how this system will be electrically separated from the safety-related system (provide a detailed discussion on how a fault on the non-Class 1E electrical circuit will not propagate to the Class 1E electrical circuit).*

b. *Describe the independence (e.g., electrical and physical separation) and redundancy of these systems.*

c. *Describe how these systems meet the single failure criterion.*

d. *Describe how the operators will be notified in the event that these systems and components would become inoperable (e.g., control room annuciators).*

e. *Describe any impacts on seismic qualifications of these systems and components.*

<u>Response – NRC Question 10</u>

No non-safety related systems or components are credited as part of this LAR to revise the Technical Specifications regarding Secondary Containment operability requirements during refueling.

<u>NRC Question 11</u>

Are any loads being added to the Oyster Creek emergency diesel generators (EDGs)? If so, describe how the loads being added to the EDGs affect the capability and capacity of the EDGs (e.g., describe the impact of the proposed change on the EDG ratings).

<u>Response – NRC Question 11</u>

No loads are required to be added to the OCNGS Emergency Diesel Generators (EDGs) as part of this LAR.

<u>NRC Question 12</u>

Provide the loading sequence for each EDG at Oyster Creek. In your response, describe the changes that have been made to the EDG loading sequence to support this LAR.

<u>Response – NRC Question 12</u>

See response to Question 11 above.

<u>NRC Question 13</u>

Provide a list and description of components being added to your 10 CFR 50.49 program due to this LAR. Confirm that these components are qualified for the environmental conditions they are expected to be exposed to.

Response – NRC Question 13

No additional equipment or components are being added to the 10 CFR 50.49 program as a result of this LAR.

NRC Question 14

Are there any changes in the chemical composition of the chemical spray solution as a result of this LAR? If so, provide the chemical composition and provide a detailed evaluation to show the components are qualified for the environmental conditions they are expected to be exposed to. Also, describe, if any, changes in the operation of the chemical spray system and its impact on the environment.

Response – NRC Question 14

No chemical spray solution is credited as part of this LAR.

NRC Question 15

Confirm that Oyster Creek environmental qualification (EQ) analyses will continue to be based on Technical Information Document (TID)-14844, "Calculation of Distance Factors for Power and Test Reactor Sites" in the EQ program. Otherwise, provide the EQ analyses to support this LAR.

Response – NRC Question 15

Environmental Qualification (EQ) analyses will continue to be based on TID-14844 (Reference 3). No changes related to EQ are sought as part of this LAR.

References

1. Letter from Pamela B. Cowan to U.S. Nuclear Regulatory Commission Technical Specification Change Request 338 - Secondary Containment Operability Requirements During Refueling, dated November 2, 2007

2. Letter from G. Edward Miller, U.S. Nuclear Regulatory Commission to Mr. Charles G. Pardee, Exelon Generation Company, LLC, dated October 20, 2009, Oyster Creek Nuclear Generating Station – Request for Additional Information Regarding Secondary Containment Operability License Amendment Request (TAC No. MD7261)

3. TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," U.S. Atomic Energy Commission, March 23, 1962

Table 1
Evaluated Release Locations

Item	Release Location	Results of Dose Analysis With 168 Hours Decay[(1)] (Rem TEDE)	Maximum Calculated (or Assumed) 0-2 Hour X/Q (sec/m³)	Comments
1	Reactor Building West Wall (Modeled as a Diffuse Area)	CR: <2.72 [(1)] EAB: 1.16 LPZ: 0.11	3.73E-03	Since no specific point-source leakage is expected on this wall, the potential release source through this wall is best characterized as a diffuse area source. As this is a modeling method used to identify a limiting release location, the TS does not allow any specific penetrations associated with this modeling. Since only the metal siding portion of the east wall of the RB has the potential for leakage, the surface area is limited to this section of the wall. Furthermore, since complete mixing in the refueling area volume cannot be assumed, only 50% of the metal siding area is assumed in the determination of the diffuse area X/Q. This limited area is assumed to be at the worst case location with relationship to either CR intake location to maximize the calculated dose.
2	Drywell Access Facility	CR: <1.41 [(1)] EAB: 1.16 LPZ: 0.11	West Wall: 1.61E-03 South Wall: 1.93E-03	This is a temporary structure used as the Drywell Support Center and Outage Command Center. It is connected to the east Reactor Building personnel air lock through a series of temporary tunnels. These tunnels are not safety related; however, since they are connected to the RB personnel air lock, any air discharged through the air lock could be forced through the tunnel, being discharged at the doors to the temporary facility. These doors are closer to the CR intakes than the RB personnel air lock itself. There are four (4) doors associated with the D/W Access Facility. All 4 doors are located in roughly the same direction relative to the CR intakes. The X/Q for the door on the northern wall would be

Table 1
Evaluated Release Locations

Item	Release Location	Results of Dose Analysis With 168 Hours Decay(1) (Rem TEDE)	Maximum Calculated (or Assumed) 0-2 Hour X/Q (sec/m^3)	Comments
				bounded by the door on the west wall, which is closer to CR intakes A and B. Similarly, the two doors on the southern wall facing the RB, would bound the door on the west wall. The X/Q for the south eastern-most door would be bounded by the south western-most door, which is closer to CR intakes A and B.
3	Commodities penetration on RB South Wall Elevation 23'-6"	CR: <1.29 (1) EAB: 1.16 LPZ: 0.11	1.77E-03	This is a flanged penetration, for which procedures are currently in place for routing commodities through the penetration without violating the secondary containment boundary. However, for the purpose of determining the maximum dose to CR personnel, this penetration is considered to be open during shutdown conditions with movement of fuel with sufficient decay in progress.
4	Commodities penetration on RB North Wall Elevation 23'-6"	CR: <3.81 (1) EAB: 1.16 LPZ: 0.11	5.21E-03	This is a flanged penetration, for which procedures are currently in place for routing commodities through the penetration without violating the secondary containment boundary. However, for the purpose of determining the maximum dose to CR personnel, this penetration is considered to be open during shutdown conditions with movement of fuel with sufficient decay in progress.
5	Commodities penetration on RB East Wall, Elevation 23'-6"	CR: <1.02 (1) EAB: 1.16 LPZ: 0.11	[*1.40E-03*]	This is a flanged penetration, for which procedures are currently in place for routing commodities through the penetration without violating the secondary containment boundary. However, for the purpose of determining the maximum dose to CR personnel, this penetration is considered to be open during shutdown conditions with movement of fuel with sufficient decay in progress. This penetration is adjacent to the RB east personnel airlock and is considered bounded by

Table 1
Evaluated Release Locations

Item	Release Location	Results of Dose Analysis With 168 Hours Decay[(1)] (Rem TEDE)	Maximum Calculated (or Assumed) 0-2 Hour X/Q (sec/m^3)	Comments
				the airlock (which is significantly less than the bounding X/Q used in the dose analysis. Therefore, a specific X/Q was not calculated. .
6	RB Roof Hatch	CR: <1.33 [(1)] EAB: 1.16 LPZ: 0.11	1.82E-03	The roof hatch is a personnel access way to the RB roof. Although not opened routinely, it is considered to be a potential release location in the event that it is open during shutdown conditions with movement of fuel with sufficient decay in progress.
7	Stack Tunnel Door	CR: <0.62 [(1)] EAB: 1.16 LPZ: 0.11	8.55E-04	This is the access location to certain SGTS dampers and equipment to the east of the Reactor Building. It is modeled as a potential release location if opened during shutdown conditions with movement of fuel with sufficient decay in progress.
8	East RB Airlock Door	CR: <1.02 [(1)] EAB: 1.16 LPZ: 0.11	1.40E-03	The east RB Airlock door is connected to the temporary tunnel leading to the Drywell Access Facility. However, since this temporary facility is not safety related, leakage from this location is postulated during movement of fuel with sufficient decay in progress.
9	South East RB Airlock Door	CR: <1.29 [(1)] EAB: 1.16 LPZ: 0.11	[*1.77E-03*]	Since the south east RB Airlock door is in the same relative direction as the south RB commodities penetration and farther away, it is considered bounded by the south RB commodities penetration. Therefore, a specific X/Q was not calculated.
10	Reactor Building Truck Airlock	CR: <1.33 [(1)] EAB: 1.16 LPZ: 0.11	[*1.82E-03*]	Since the RB Truck Airlock door is in the same relative direction and farther away from other calculated openings, this penetration is considered bounded by the south RB commodities penetration and the RB roof hatch. Therefore, a specific X/Q was not calculated.
11	Isolation Condenser Vents	CR: <1.33 [(1)] EAB: 1.16 LPZ: 0.11	[*1.82E-03*]	Since these vents (on the RB east wall) are in the same general direction as the RB roof hatch (and farther away) with respect to the CR intakes, a release

Table 1
Evaluated Release Locations

Item	Release Location	Results of Dose Analysis With 168 Hours Decay[(1)] (Rem TEDE)	Maximum Calculated (or Assumed) 0-2 Hour X/Q (sec/m^3)	Comments
				from this location is considered bounded by the RB roof Hatch. Therefore, a specific X/Q was not calculated.
12	MAC Facility Entrance	CR: <4.84 [(1)] EAB: 1.16 LPZ: 0.11	6.62E-03	These double doors are modeled as a single penetration.
13	**(MAC Facility Personnel Airlock) Northwest RB Personnel Airlock at Elevation 23'-6"**	**CR: 4.93** **EAB: 1.16** **LPZ: 0.11**	**6.75E-03**	**The MAC Facility Personnel Airlock exits out of tornado/missile protection area located on the north RB wall (23'-6" elev.). This is the location with the maximum X/Q for the dose analysis and results in the maximum dose for all penetrations allowed to be open during shutdown conditions with movement of fuel with sufficient decay.**
14	Trunnion Room Door to Turbine Building	CR: <2.72 [(1)] EAB: 1.16 LPZ: 0.11	3.73E-03	The Trunnion Room is a subset of secondary containment and houses the outboard MSIVs. The single access door is not an airlock. Access to this room is permitted (via TS) during operation through intermittent opening of the door under administrative controls.
15	Northwest RB Personnel Airlock to Office Building at Elevation 51'-3"	CR: **>5.0** EAB: 1.16 LPZ: 0.11	X/Q Not Calculated Based on Inspection of Closeness to CR Access Door **<u>NOT</u> Permitted to Be Open During Fuel Movement**	The Northwest RB Personnel Airlock leads from the RB to the Office Building on the 51' 3" elevation (Columns RF and R6). Its closure is credited in the analysis of the FHA. The upper containment personnel air lock performs no active function in response to the postulated accident; however, its leak tightness is required to ensure that the release of radioactive materials from primary containment is restricted to those leakage paths assumed in the accident analysis, and the fission products released by the FHA will be treated by the SGT System. It was not originally on the list of penetrations allowed to be open. Since this airlock opens into the Office Building (close to the CR entrance), it is **<u>NOT</u>** permitted to

Table 1
Evaluated Release Locations

Item	Release Location	Results of Dose Analysis With 168 Hours Decay[(1)] (Rem TEDE)	Maximum Calculated (or Assumed) 0-2 Hour X/Q (sec/m^3)	Comments
				be open due to its proximity to the CR HVAC intakes and the CR access door. Based on this closeness, a specific X/Q was not calculated.
16	Southwest RB Personnel Airlock to TB at Elevation 6'-5"	CR: <2.72 [(1)] EAB: 1.16 LPZ: 0.11	[3.73E-03]	This airlock leads from the southwest RB to the Turbine Building on the 3'-6"elevation (Columns J and R6). It was not originally on the list of penetrations allowed to be open. However, it is assumed to be open during shutdown conditions with movement of fuel with sufficient decay in progress. Since this location is in the same general direction as the Trunnion Room door and is farther away, the X/Q is considered bounded by the Trunnion Room door. Therefore, a specific X/Q was not calculated.
17	Northwest RB Personnel Airlock to TB at Elevation -1'-11"	CR: <4.93 [(1)] EAB: 1.16 LPZ: 0.11	[6.75E-03]	This airlock leads from the northwest RB to the Turbine Building on the 3'-6" elevation (Columns RG and R6). It was not originally on the list of penetrations allowed to be open. However, it is assumed to be open during shutdown conditions with movement of fuel with sufficient decay in progress. Since this location is in the same general direction as the MAC Facility personnel airlock and is farther away, the X/Q is considered bounded by the MAC Facility personnel airlock. Therefore, a specific X/Q was not calculated.
18	Floor Plug to SW RB Corner Room Elevation 23'-6"	CR: <2.72 [(1)] EAB: 1.16 LPZ: 0.11	[3.73E-03]	Since this location is in the same general direction as the Trunnion Room door and is farther away, the X/Q is considered bounded by the Trunnion Room door. Therefore, a specific X/Q was not calculated.
19	Floor Plug to NW RB Corner Room Elevation 23'-6"	CR: <4.93 [(1)] EAB: 1.16 LPZ: 0.11	[6.75E-03]	Since this location is in the same general direction as the MAC Facility personnel airlock and is farther away,

Table 1
Evaluated Release Locations

Item	Release Location	Results of Dose Analysis With 168 Hours Decay[1] (Rem TEDE)	Maximum Calculated (or Assumed) 0-2 Hour X/Q (sec/m³)	Comments
				the X/Q is considered bounded by the MAC Facility personnel airlock. Therefore, a specific X/Q was not calculated.
20	Service Water Pipe Penetration	CR: <1.02 EAB: 1.16 LPZ: 0.11	1.40E-03	At elevation 41'-6" and located approximately 64' south of the North face of the Reactor Building

[1] CR Dose estimated using a ratio to the maximum X/Q of 6.75E-03 sec/m^3 allowed.

ATTACHMENT 2

Revised License Amendment Request Section 2.0, “Proposed Changes”

Background

By letter dated November 2, 2007, Exelon (formerly AmerGen) submitted Technical Specification Change Request 338, *"Secondary Containment Operability Requirements During Refueling,"* requesting changes to the Oyster Creek TS that would modify Secondary Containment requirements during refueling operations. Exelon provided additional information concerning this proposed LAR by letters dated May 5, 2008, July 3, 2008, and September 22, 2008.

Subsquently, in the NRC's Request for Additional Information (RAI) dated October 20, 2009, the NRC indicated that based on the information provided, significant changes to the TS may have resulted that could invalidate or render moot the original justification for the proposed changes. Therefore, the NRC is requesting that Exelon completely revise Section 2.0, *"Proposed Changes,"* and submit a new Section 2.0 which includes a clear technical justification for each proposed change.

Accordingly, the information provided below contains a new Section 2.0, *"Proposed Changes,"* for the proposed LAR. This Section 2.0 supersedes the information provided in Section 2.0 of Exelon's November 2, 2007, submittal. The new Section 2.0 describes each of the proposed changes along with the supporting justification.

2.0 Proposed Changes

The proposed changes to the Oyster Creek TS are being requested to allow certain Secondary Containment operability requirements to be modified during fuel handling operations when handling fuel that has not occupied part of a critical reactor core within the previous 168 hours. The proposed changes have been evaluated and are supported by reanalysis of the radiological consequences of a Fuel Handling Accident (FHA) utilizing AST methodology previously reviewed and approved by NRC for use at Oyster Creek in TS Amendment No. 262, dated April 26, 2007. The movement of sufficiently decayed irradiated fuel is consistent with TSTF-51, Revision 2 to NUREG-1433, Volume 1, Revision 2, *"Standard Technical Specifications - General Electric Plants."*

Existing Secondary Containment integrity requirements specified in TS 3.5.B will remain applicable when fuel handling operations involve fuel that has occupied part of a critical reactor core within the previous 168 hours.

2.1 TS Definition 1.52, RECENTLY IRRADIATED FUEL, is being added to specify when the existing TS provisions for Secondary Containment integrity remain applicable. The proposed TS definition specifies a minimum decay time of 168 hours before Secondary Containment requirements can be altered.

2.2 TS Section 3.5.B.1 is being revised to include an exception for maintaining Secondary Containment integrity at all times. This exception is intended to permit movement of irradiated fuel during refueling operations without Secondary Containment integrity when the reactor has been subcritical for greater than 168 hours provided certain provisions are met. These provisions include: 1) only certain evaluated release locations can be opened, 2) one Standby Gas Treatment System (SGTS) circuit shall be available, and 3) in the event of fuel handling accident the available SGTS circuit shall be running within one hour and all Secondary Containment openings must be closed within one hour. This will ensure that, in the event of a radiological release from a fuel handling accident, air flow is directed such that it can be treated and monitored. This proposed change has been evaluated and is supported by reanalysis of the radiological consequences of a FHA utilizing AST methodology.

2.3 TS Section 3.5.B.1.c and d are revised to allow the handling of irradiated fuel without Secondary Containment integrity when the reactor has been subcritical for greater than 168 hours. Secondary Containment integrity is required when handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 168 hours), or when work is being performed on the reactor or its connected systems in the reactor building which has the potential to drain the reactor vessel. Secondary Containment is not required with the reactor vessel head or drywell head in place, as allowed by current TS.

TS Section 3.5.B.1.c is being revised to read: *"No work is being performed on the reactor or its connected systems in the reactor building which has the potential to drain the reactor vessel."* This replaces the existing requirement that: *"The reactor vessel head or the drywell head are in place,"* which is being relocated to Section 3.5.B.1.d.

TS Section 3.5.B.1.d is being revised to stipulate that one of the following must be met: *1) "The reactor vessel head or the drywell head are in place,"* or *2) The reactor has been subcritical for greater than 168 hours."* This replaces the existing requirement that: *"No work is being performed on the reactor or its connected systems in the reactor building which could result in inadvertent releases of radioactive material."*

These proposed changes are considered conservative and consistent with TSTF-51, Revision 2.

2.4 TS Section 3.5.B.1.e is being revised. The existing wording: *"No operations are being performed in, above, or around the spent fuel pool or reactor cavity that could cause release of radioactive materials,"* will be revised to restrict movement of heavy loads over the reactor cavity or spent fuel pool when Secondary Containment is not available. The proposed changes will still limit the type of activities that can be performed in, above, or around the spent fuel pool or reactor cavity that could cause release of radioactive materials.

2.5 TS Section 3.5.B.4.a action statement for loss of SECONDARY CONTAINMENT INTEGRITY or inoperable secondary containment isolation valves (SCIVs) during power operation is being revised to include a restriction on the movement of heavy loads in, above, or around the spent fuel pool.

2.6 TS Section 3.5.B.4.b.(1) action statement for loss of SECONDARY CONTAINMENT INTEGRITY or inoperable secondary containment isolation valves (SCIVs) during refueling is being revised to include a restriction on the movement of RECENTLY IRRADIATED FUEL. The original request to delete the restriction on activities that could reduce the shutdown margin has been re-evaluated and is no longer being considered. This specific requirement will be retained in the TS.

2.7 TS Section 3.5.B.4.b.(2) action statement for loss of SECONDARY CONTAINMENT INTEGRITY or inoperable secondary containment isolation valves (SCIVs) during refueling is being revised to read as follows: *"Cease all work on the reactor or its connected systems in the reactor building which could result in potential to drain the* *reactor vessel."* The existing phrase *"...inadvertent releases of radioactive materials"* was replaced by the phrase *"...potential to drain the reactor vessel."* This proposed change is considered conservative and consistent with TSTF-51, Revision 2.

2.8 TS Section 3.5.B.4.b.(3) action statement for loss of SECONDARY CONTAINMENT INTEGRITY or inoperable secondary containment isolation valves (SCIVs) during refueling is being revised to include a restriction on the movement of heavy loads in, above, or around reactor cavity or the spent fuel pool.

2.9 TS Section 3.5.B.5 is a new action statement for loss of SECONDARY CONTAINMENT INTEGRITY or inoperable secondary containment isolation valves (SCIVs) during refueling that has been added to control the movement of irradiated fuel without Secondary Containment integrity if the reactor has been subcritical for greater than 168 hours under certain conditions. These conditions include: 1) only certain evaluated release locations are permitted to be open, 2) one SGTS circuit shall be available, and 3) in the event of a fuel handling accident the available SGTS circuit shall be running within one hour and all Secondary Containment openings must be closed within one hour. This will ensure that, in the event of a radiological release from a fuel handling accident, air flow is directed such that it can be treated and monitored. This proposed change has been evaluated and is supported by reanalysis of the radiological consequences of a FHA utilizing AST methodology. These new requirements were also discussed in 2.2 above.

2.10 TS Section 3.5.B.6.b.(3), which has been renumbered as 3.5.B.7.b.(3), is revised to specify that during refueling the availability requirements for the SGTS apply when: 1) fuel handling operations involve RECENTLY IRRADIATED FUEL, 2) when performing operations with the potential to drain the reactor vessel, or 3) for movement of heavy loads in, above, or around the reactor cavity or spent fuel pool. The original request to delete the restriction on activities that could reduce the shutdown margin has been re-evaluated and is no longer being considered. This specific requirement will be retained in the TS.

2.11 TS Section 3.5.B.7.c, which is being renumbered from 3.5.B.6 to 3.5.B.7, is a new section that was added related to SGTS operability requirements during refueling operations. This new section would permit SGTS operability requirements to be altered when moving irradiated fuel provided the reactor has been subcritical for greater than 168 hours and all release locations from the Secondary Containment can be closed within one hour.

2.12 TS Section 3.5.B.9 is a new section that was added related to Secondary Containment requirements specifying that in the event of a fuel handling accident, SGTS must be running within one hour and all Secondary Containment openings must be closed within one hour following the fuel handling accident.

2.13 Table 3.5-1 is a new table that has been added for TS Section 3.5.B.5 describing release locations that are acceptable to be opened once the reactor has been subcritical for greater than 168 hours. There is one exception identified in the table (i.e., Item 15 - Northwest RB Personnel Airlock to Office Building at Elev. 51' 3") and this penetration must remain closed. The release locations identified in the table are the only release pathways evaluated and acceptable to be opened during movement of irradiated fuel. Other potential release pathways may be opened provided that any potential release from the pathway will be bounded by a pathway previously evaluated as identified in the table.

2.14 TS Section 3.17 C and D, are being revised to specify that operability requirements for the Control Room Heating, Ventilating, and Air-Conditioning (HVAC) System during refueling apply when handling RECENTLY IRRADIATED FUEL, and to substitute the wording "...the potential to drain the reactor vessel...," in lieu of "...inadvertent release of radioactive material...," consistent with TSTF-51, Revision 2. In addition, these sections are being revised to include restrictions on the movement of heavy loads in, above, or around the reactor cavity or spent fuel pool.

2.15 TS Bases Sections 3.5 and 4.5 are revised to incorporate the basis for the proposed changes, including the basis for the term RECENTLY IRRADIATED FUEL assuming a minimum decay time of 168 hours, the need for maintaining SGTS operability or availability, the permitted release locations, and to incorporate the defense-in-depth guidelines contained in TSTF-51, Revision 2.

ATTACHMENT 3

Revised Technical Specifications Page Mark-ups

Affected Pages

1.0-9
3.5-5
3.5-6
3.5-7
3.5-7a (new page)
3.5-7b (new page)
3.5-7c (new page)
3.17-1

1.49 RATED THERMAL POWER (RTP)

RTP shall be a total reactor core heat transfer rate to the reactor coolant of 1930 MWt.

1.50 THERMAL POWER

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

1.51 PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 6.23.

1.52 RECENTLY IRRADIATED FUEL

RECENTLY IRRADIATED FUEL is fuel that has occupied part of a critical reactor core within the previous 168 hours. This 168-hour period may be reduced to 24 hours if all secondary containment openings are closed and the Standby Gas Treatment System is OPERABLE.

8. Shock Suppressors (Snubbers)

a. All safety related snubbers are required to be operable whenever the systems they protect are required to be operable except as noted in 3.5.A.8.b and c below.

b. With one or more snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to operable status.

c. If the requirements of 3.5.A.8.a and 3.5.A.8.b cannot be met, declare the protected system inoperable and follow the appropriate action statement for that system.

d. An engineering evaluation shall be performed to determine if the components protected by the snubber(s) were adversely affected by the inoperability of the snubber prior to returning the system to operable status.

B. Secondary Containment

1. Secondary containment integrity shall be maintained at all times unless all of the following conditions are met **except as specified in Specification 3.5.B.5**:

a. The reactor is subcritical and Specification 3.2.A is met.

b. The reactor is in the cold shutdown condition.

c. **No work is being performed on the reactor or its connected systems in the reactor building which has the potential to drain the reactor vessel**~~The reactor vessel head or the drywell head are in place.~~

d. ~~No work is being performed on the reactor or its connected systems in the reactor building which could result in inadvertent releases of radioactive material~~**One of the following are met:**

1) The reactor vessel head or the drywell head are in place.

2) The reactor has been subcritical for greater than 168 hours.

e. No movement of heavy loads in, above, or around the reactor cavity or spent fuel pool.

~~e. No operations are being performed in, above, or around the spent fuel storage pool that could cause release of radioactive materials~~

2. Upon the accidental loss of SECONDARY CONTAINMENT INTEGRITY, restore, SECONDARY CONTAINMENT INTEGRITY within 4 hours, except as provided in specification 3.5.B.3.

3. With one or more of the automatic secondary containment isolation valves inoperable:

   a. Maintain at least one automatic secondary containment isolation valve in each affected penetration OPERABLE.

   b. Within 8 hours restore the inoperable automatic secondary containment isolation valve(s) to OPERABLE status or isolate each affected penetration with at least one valve secured in the closed position.

4. If Specifications 3.5.B.2 or 3.5.B.3 cannot be met:

   a. During Power Operation:

      (1) Have the reactor mode switch in the shutdown mode position within the following 24 hours.

      (2) Cease all work on the reactor or its connected systems in the reactor building which could result in inadvertent releases of radioactive materials.

      (3) Cease all operations in, above or around the Spent Fuel Storage Pool that could cause release of radioactive materials.

      (4) Cease movement of heavy loads in, above, or around the spent fuel pool.

   b. During refueling:

      (1) Cease fuel handling operations **involving RECENTLY IRRADIATED FUEL** or activities which could reduce the shutdown margin (excluding reactor coolant temperature changes).

      (2) Cease all work on the reactor or its connected systems in the reactor building which could result in ~~inadvertent releases of radioactive materials~~**potential to drain the reactor vessel.**

      (3) **Cease movement of heavy loads in, above, or around the reactor cavity or spent fuel pool.**~~**Cease all operations in, above or around the Spent Fuel Storage Pool that could cause release of radioactive materials.**~~

5. During refueling, movement of irradiated fuel is permitted without secondary containment integrity if the reactor has been subcritical for greater than 168 hours when:

a. Any of the evaluated release locations listed in Table 3.5-1 are open, with one noted exception[1];

and;

b. One standby gas treatment circuit is available[2].

c. As specified in Specification 3.5.B.9, in the event of a fuel handling accident, one standby gas treatment circuit shall be running within one hour and all secondary containment openings must be closed within one hour.[2]

~~5~~6. Two separate and independent standby gas treatment system circuits shall be operable when secondary containment is required except as specified by Specification 3.5.B.~~6~~7.

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(1) The release locations listed in Table 3.5-1 are the only release locations that have been evaluated and are acceptable to be opened during movement of irradiated fuel, with one noted exception (Item 14 - Northwest RB Personnel Airlock to Office Building at elev. 51'-3"). Other potential release pathways may be opened provided that any potential release from the pathway will be bounded by a pathway previously evaluated as identified in the table.

(2) One standby gas treatment circuit shall be running within one hour to direct flow in the proper direction in the event of a radiological release from a fuel handling accident such that it can be treated and monitored.

~~6~~**7**. With one standby gas treatment system circuit inoperable:

a. During Power Operation:

(1) Verify the operability of the other standby gas treatment system circuit within 2 hours. If testing is required to demonstrate operability and significant painting, fire, or chemical release has taken place in the reactor building within the previous 12 hours, then demonstration by testing shall take place within 1 hour of the expiration of the 12 hour period, and

(2) Continue to verify the operability of the standby gas treatment system circuit once per 24 hours until the inoperable standby gas treatment circuit is returned to operable status.

(3) Restore the inoperable standby gas treatment circuit to operable status within 7 days.

b. During Refueling:

(1) Verify the operability of the other standby gas treatment system within 2 hours. If testing is required to demonstrate operability and significant painting, fire, or chemical release has taken place in the reactor building within the previous 12 hours, then demonstration by testing shall take place within 1 hour of the expiration of the 12 hour period, and

(2) Continue to verify the operability of the redundant standby gas treatment system once per 7 days until the inoperable system is returned to operable status.

(3) Restore the inoperable standby gas treatment system to operable status within 30 days or cease all ~~spent fuel handling, core alterations or operation that could reduce the shutdown margin (excluding reactor coolant temperature changes.~~**of the following:**

(a) Activities that could reduce shutdown margin (excluding reactor coolant temperature changes).

(b) Fuel handling operations involving RECENTLY IRRADIATED FUEL.

(c) Operations with the potential to drain the reactor vessel.

(d) Movement of heavy loads in, above, or around the reactor cavity or spent fuel pool.

(e) Opening penetrations in accordance with Table 3.5-1 and any penetrations that are open shall be closed.

c. Standby gas treatment system availability can be modified during movement of irradiated fuel when:[1]

(1) The reactor has been subcritical for greater than 168 hours, and

(2) All release locations from the secondary containment can be closed[1].

~~7.~~8 If Specifications 3.5.B.~~5~~ 6 and 3.5.B.~~6~~ 7 are not met, reactor shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours and the condition of Specification 3.5.B.1 shall be met.

9. In the event of a fuel handling accident, one standby gas treatment circuit must be running within one hour, and all secondary containment openings shall be closed within one hour following the fuel handling accident.[1]

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(1) One standby gas treatment circuit shall be capable of running within one hour to direct flow in the proper direction in the event of a radiological release from a fuel handling accident such that it can be treated and monitored.

Table 3.5-1[(1)]
Evaluated Release Locations

Item	Release Location	Release Location Permitted to be Open During Movement of Irradiated Fuel
1	Drywell Access Facility	Yes[(2)]
2	Commodities penetration on RB South Wall Elev. 23'-6"	Yes
3	Commodities penetration on RB North Wall Elev. 23'-6"	Yes
4	Commodities penetration on RB East Wall, Elev. 23'-6"	Yes
5	RB Roof Hatch	Yes
6	Stack Tunnel Door	Yes
7	East RB Airlock Door	Yes
8	South East RB Airlock Door	Yes
9	Reactor Building Truck Airlock	Yes
10	Isolation Condenser Vents	Yes
11	MAC Facility Entrance	Yes[(3)]
12	(MAC Facility Personnel Airlock) Northwest RB Personnel Airlock at 23'-6"	Yes
13	Trunnion Room Door to Turbine Building	Yes
14	Northwest RB Personnel Airlock to Office Building at Elev. 51'-3"	No[(1)]
15	Southwest RB Personnel Airlock to TB at Elev. - 6'-5"	Yes
16	Northwest RB Personnel Airlock to TB at Elev. -1'-11"	Yes
17	Floor Plug to SW RB Corner Room Elev. 23'-6"	Yes
18	Floor Plug to NW RB Corner Room Elev. 23'-6"	Yes
19	Service Water Pipe Penetration	Yes

(1) The release locations listed in Table 3.5-1 are the only release locations that have been evaluated and are acceptable to be opened during movement of irradiated fuel, with one noted exception (Item 14 - Northwest RB Personnel Airlock to Office Building at Elev. 51'-3"). Other potential release pathways may be opened provided that any potential release from the pathway will be bounded by a pathway previously evaluated as identified in the table.

(2) At any time the access points to the Drywell Access Facility (Item 1 above) can be open provided the East RB Airlock (Item 7 above) is closed. After the 168-hour decay period during refueling operations, the requirement to have the airlock closed can be relaxed.

(3) At any time the access points to the MAC Facility Entrance (Item 11 above) can be open provided the Northwest RB Personnel Airlock 23'-6" (MAC Facility Personnel Airlock) is closed. After the 168-hour decay period during refueling operations, the requirement to have the airlock closed can be relaxed.

3.17 Control Room Heating, Ventilating, and Air-Conditioning System

Applicability: Applies to the operability of the control room heating, ventilating, and air conditioning (HVAC) system and Control Room Envelope (CRE) boundary.

---NOTE---
The CRE boundary may be opened intermittently under administrative control.

Objective: To assure the capability of the control room HVAC system and CRE boundary to minimize the amount of radioactivity, hazardous chemicals, or smoke from entering the control room in the event of an accident.

Specifications:

A. The control room HVAC system shall be operable during all modes of plant operation.

B. With one control room HVAC system determined inoperable for reasons other than specification D:

1. Verify once per 24 hours the partial recirculation mode of operation for the operable system, or place the operable system in the partial recirculation mode; and

2. Restore the inoperable system within 7 days, or prepare and submit a special report to the Commission in lieu of any other report required by Section 6.9, within the next 14 days, outlining the action taken, the cause of the inoperability and the plans/schedule for restoring the HVAC system to operable status.

C. With both control room HVAC systems determined inoperable for reasons other than specification D:

1. During Power Operation: place the reactor in the cold shutdown condition within 30 hours

2. During Refueling:

(a) Cease **RECENTLY IRRADIATED FUEL** ~~irradiated fuel~~ handling operations; and

(b) ~~(b)~~ Cease all work on the reactor or its connected systems in the reactor building which could result in ~~inadvertent releases of radioactive materials~~**the potential to drain the reactor vessel**~~.~~**; and**

(c) Cease movement of heavy loads in, above, or around the reactor cavity or spent fuel pool.

D. When one or both control room HVAC systems are determined inoperable due to an inoperable CRE boundary:

1. During Power Operation: actions to implement mitigating actions shall be performed immediately, verification that the mitigating actions are in place shall be performed within 24 hours, and the CRE boundary shall be restored to operable status within 90 days.

2. During **Refueling:**~~movement of irradiated fuel assemblies in the containment or during operations with a potential for draining the reactor vessel:~~

(a) Immediately suspend movement of ~~irradiated fuel~~**RECENTLY IRRADIATED FUEL** assemblies in the containment; and

(b) ~~(b)~~ Immediately initiate action to suspend operations with the potential to drain the reactor vessel~~.~~**; and**

(c) Immediately suspend movement of heavy loads in, above, or around the reactor cavity and spent fuel pool.

ATTACHMENT 4

Revised Technical Specifications Bases Page Mark-ups

3.5-8
3.5-11
3.5-11a (new page)
3.5-11b (new page)
3.5-11c (new page)
3.5-11d (new page)
3.5-12
3.5-12a
3.5-12b (new page)
4.5-13

Bases:

Specifications are placed on the operating status of the containment systems to assure their availability to control the release of any radioactive materials from irradiated fuel in the event of an accident condition. The primary containment system[1] provides a barrier against uncontrolled release of fission products to the environs in the event of a break in the reactor coolant systems.

Whenever the reactor coolant water temperature is above 212°F, failure of the reactor coolant system would cause rapid expulsion of the coolant from the reactor with an associated pressure rise in the primary containment. Primary containment is required, therefore, to contain the thermal energy of the expelled coolant and fission products which could be released from any fuel failures resulting from the accident. If the reactor coolant is not above 212°F, there would be no pressure rise in the containment. In addition, the coolant cannot be expelled at a rate which could cause fuel failure to occur before the core spray system restores cooling to the core. Primary containment is not needed while performing low power physics tests since procedures and the Rod Worth Minimizer would limit rod worth such that a rod drop would not result in any fuel damage. In addition, in the unlikely event that an excursion did occur, the reactor building and standby gas treatment system, which shall be operational during this time, offer a sufficient barrier to keep off-site doses well below ~~10 CFR 100~~ **applicable 10 CFR** limits.

The absorption chamber water volume provides the heat sink for the reactor coolant system energy released following the loss-of-coolant accident. The core spray pumps and containment spray pumps are located in the comer rooms and due to their proximity to the torus, the ambient temperature in those rooms could rise during the design basis accident. Calculations[7] made, assuming an initial torus water temperature of 100°F and a minimum water volume of 82,000 ft^3, indicate that the comer room ambient temperature would not exceed the core spray and containment spray pump motor operating temperature limits and, therefore, would not adversely affect the long-term core cooling capability. The maximum water volume limit allows for an operating range without significantly affecting accident analyses with respect to free air volume in the absorption chamber. For example, the containment capability[8] with a maximum water volume of 92,000 $ft.^3$ is reduced by not more than 5.5% metal-water reaction below the capability with 82,000 $ft.^3$.

Experimental data indicate that excessive steam condensing loads can be avoided if the peak temperature of the suppression pool is maintained below 160°F during any period of relief valve operation with sonic conditions at the discharge exit. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.

Snubbers are designed to prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or severe transient, while allowing normal thermal motion during startup and shutdown. The consequence of an inoperable snubber is an increase in the probability of structural damage to piping as a result of a seismic or other event initiating dynamic loads. It is, therefore, required that all snubbers required to protect the primary coolant system or any other safety system or component be OPERABLE whenever the systems they protect are required to be OPERABLE.

The purpose of an engineering evaluation is to determine if the components protected by the snubber were adversely affected by the inoperability of the snubber. This ensures that the protected component remains capable of meeting the designed service. A documented visual inspection will usually be sufficient to determine system OPERABILITY.

Because snubber protection is required only during low probability events, a period of 72 hours is allowed for repairs or replacements.

Secondary containment[5] is designed to minimize any ground level release of radioactive materials which might result from a serious accident. The reactor building provides secondary containment during reactor operation when the drywell is sealed and in service and provides primary containment when the reactor is shutdown and the drywell is open, as during refueling. Because the secondary containment is an integral part of the overall containment system, it is required at all times that primary containment is required. Moreover, secondary containment is required during fuel handling operations **involving RECENTLY IRRADIATED FUEL,** and whenever work is being performed on the reactor or its connected systems in the reactor building **which could result in the potential to drain the reactor vessel.**

During refueling operations, when the reactor has been subcritical for greater than 168 hours movement of irradiated fuel is permitted with the release locations listed in Bases Table 3.5-1 open (with one specified exception – Northwest RB Personnel Airlock to Office Building at Elev. 51'-3") without secondary containment integrity provided one standby gas treatment circuit is available. These are the only release locations that have been evaluated and are acceptable to be opened during movement of irradiated fuel. No other release locations can be opened during movement of irradiated fuel unless they have been appropriately evaluated and determined that any release would be bounded by one of the previously evaluated release locations listed in the table. ~~since their operation could result in inadvertent release of radioactive material.~~

When secondary containment is not maintained, the additional restrictions on operation and maintenance give assurance that the probability of inadvertent releases of radioactive material will be minimized. Maintenance will not be performed on systems which connect to the reactor vessel lower than the top of the active fuel unless the system is isolated by at least one locked closed isolation valve.

Bases Table 3.5-1[1]
Evaluated Release Locations

Item	Release Location	Release Locations Permitted to be Open During Movement of Irradiated Fuel	Comments
1	Drywell Access Facility[2]	Yes	This is a temporary structure used as the Drywell Support Center and Outage Command Center. It is connected to the east Reactor Building personnel air lock through a series of temporary tunnels. These tunnels are not safety related; however, since they are connected to the RB personnel air lock, any air discharged through the air lock could be forced through the tunnel, being discharged at the doors to the temporary facility. These doors are closer to the CR intakes than the RB personnel air lock itself. There are four (4) doors associated with the D/W Access Facility. All 4 doors are located in roughly the same direction relative to the CR intakes. The X/Q for the door on the northern wall would be bounded by the door on the west wall, which is closer to CR intakes A and B. Similarly, the two doors on the southern wall facing the RB, would bound the door on the west wall. The X/Q for the south eastern-most door would be bounded by the south western-most door, which is closer to CR intakes A and B.
2	Commodities penetration on RB South Wall Elev. 23'-6"	Yes	This is a flanged penetration, for which procedures are currently in place for routing commodities through the penetration without violating the secondary containment boundary. However, for the purpose of determining the maximum dose to CR personnel, this penetration is considered to be open during shutdown conditions with movement of fuel with sufficient decay in progress.
3	Commodities penetration on RB North Wall Elev. 23'-6"	Yes	This is a flanged penetration, for which procedures are currently in place for routing commodities through the penetration without violating the secondary containment boundary. However, for the purpose of determining the maximum dose to CR personnel, this penetration is considered to be open during shutdown conditions with movement of fuel with sufficient decay in progress.

Bases Table 3.5-1(1)
Evaluated Release Locations

4	Commodities penetration on RB East Wall, Elev. 23'-6"	Yes	This is a flanged penetration, for which procedures are currently in place for routing commodities through the penetration without violating the secondary containment boundary. However, for the purpose of determining the maximum dose to CR personnel, this penetration is considered to be open during shutdown conditions with movement of fuel with sufficient decay in progress. This penetration is adjacent to the RB east personnel airlock and is considered bounded by the airlock (which is significantly less than the bounding X/Q used in the dose analysis. Therefore, a specific X/Q was not calculated.
5	RB Roof Hatch	Yes	The roof hatch is a personnel access way to the RB roof. Although not opened routinely, it is considered to be a potential release location in the event that it is open during shutdown conditions with movement of fuel with sufficient decay in progress.
6	Stack Tunnel Door	Yes	This is the access location to certain SGTS dampers and equipment to the east of the Reactor Building. It is modeled as a potential release location if opened during shutdown conditions with movement of fuel with sufficient decay in progress.
7	East RB Airlock Door	Yes	The east RB Airlock door is connected to the temporary tunnel leading to the Drywell Access Facility. However, since this temporary facility is not safety related, leakage from this location is postulated during movement of fuel with sufficient decay in progress.
8	South East RB Airlock Door	Yes	Since the south east RB Airlock door is in the same relative direction as the south RB commodities penetration and farther away, it is considered bounded by the south RB commodities penetration. Therefore, a specific X/Q was not calculated.
9	Reactor Building Truck Airlock	Yes	Since the RB Truck Airlock door is in the same relative direction and farther away from other calculated openings, this penetration is considered bounded by the south RB commodities penetration and the RB roof hatch. Therefore, a specific X/Q was not calculated.
10	Isolation Condenser Vents	Yes	Since these vents (on the RB east wall) are in the same general direction as the RB roof hatch (and farther away) with respect to the CR intakes, a release from this location is considered bounded by the RB roof Hatch. Therefore, a specific X/Q was not calculated.
11	MAC Facility Entrance	Yes	These double doors are modeled as a single penetration.

Bases Table 3.5-1[1]
Evaluated Release Locations

12	(MAC Facility Personnel Airlock) Northwest RB Personnel Airlock at Elev. 23'-6"[3]	Yes	The MAC Facility Personnel Airlock exits out of tornado/missile protection area located on the north RB wall (23'-6" elev.). This is the location with the maximum X/Q for the dose analysis and results in the maximum dose for all penetrations allowed to be open during shutdown conditions with movement of fuel with sufficient decay.
13	Trunnion Room Door to Turbine Building	Yes	The Trunion Room is a subset of secondary containment and houses the outboard MSIVs. The single access door is not an airlock. Access to this room is permitted (via TS) during operation through intermittent opening of the door under administrative controls.
14[1]	Northwest RB Personnel Airlock to Office Building at Elev. 51'-3"	No	The Northwest RB Personnel Airlock leads from the RB to the Office Building on the 51' 3" elevation (Columns RF and R6). Its closure is credited in the analysis of the FHA. The upper containment personnel air lock performs no active function in response to the postulated accident; however, its leak tightness is required to ensure that the release of radioactive materials from primary containment is restricted to those leakage paths assumed in the accident analysis, and the fission products released by the FHA will be treated by the SGT System. It was not originally on the list of penetrations allowed to be open. Since this airlock opens into the Office Building (close to the CR entrance), it is **<u>NOT</u>** permitted to be open due to its proximity to the CR HVAC intakes and the CR access door. Based on this closeness, a specific X/Q was not calculated.
15	Southwest RB Personnel Airlock to TB at Elev. - 6'-5"	Yes	This airlock leads from the southwest RB to the Turbine Building on the 3'-6" elevation (Columns J and R6). It was not originally on the list of penetrations allowed to be open. However, it is assumed to be open during shutdown conditions with movement of fuel with sufficient decay in progress. Since this location is in the same general direction as the Trunnion Room door and is farther away, the X/Q is considered bounded by the Trunnion Room door. Therefore, a specific X/Q was not calculated.
16	Northwest RB Personnel Airlock to TB at Elev. -1'-11"	Yes	This airlock leads from the northwest RB to the Turbine Building on the 3'-6" elevation (Columns RG and R6). It was not originally on the list of penetrations allowed to be open. However, it is assumed to be open during shutdown conditions with movement of fuel with sufficient decay in progress. Since this location is in the same general direction as the MAC Facility personnel airlock and is farther away, the X/Q is considered bounded by the MAC Facility personnel airlock. Therefore, a specific X/Q was not calculated.

Bases Table 3.5-1[1]
Evaluated Release Locations

17	Floor Plug to SW RB Corner Room Elev. 23'-6"	Yes	*Since this location is in the same general direction as the Trunnion* Room door and is farther away, the X/Q is considered bounded by the Trunnion Room door. Therefore, a specific X/Q was not calculated.
18	Floor Plug to NW RB Corner Room Elev. 23'-6"	Yes	Since this location is in the same general direction as the MAC Facility personnel airlock and is farther away, the X/Q is considered bounded by the MAC Facility personnel airlock. Therefore, a specific X/Q was not calculated.
19	Service Water Pipe Penetration	Yes	At elevation 41'-6" and located approximately 64' south of the North face of the Reactor Building

(1) The release locations listed in Bases Table 3.5-1 are the only release locations that have been evaluated and are acceptable to be opened during movement of irradiated fuel, with the noted exception (Item 14 – Northwest RB Personnel Airlock to Office Building at Elev. 51'-3"). Other potential release pathways may be opened provided that any potential release from the pathway will be bounded by a pathway previously evaluated as identified in the table.

(2) At any time, the access points to the Drywell Access Facility (Item 1 above) can be open provided the East RB Airlock (Item 7 above) is closed. After the 168-hour decay period during refueling operations, the requirement to have the airlock closed can be relaxed.

(3) At any time, the access points to the MAC Facility Entrance (Item 11 above) can be open provided the Northwest RB Personnel Airlock 23'-6" (MAC Facility Personnel Airlock) is closed. After the 168-hour decay period during refueling operations, the requirement to have the airlock closed can be relaxed.

With regard to Diffuse Area release modeling, the following evaluation is provided.

Reactor Building West Wall (Modeled as a Diffuse Area)	Since no specific point-source leakage is expected on this wall, the potential release source through this wall is best characterized as a diffuse area source. As this is a modeling method used to identify a limiting release location, the TS does not allow any specific penetrations associated with this modeling. Since only the metal siding portion of the east wall of the RB has the potential for leakage, the surface area is limited to this section of the wall. Furthermore, since complete mixing in the refueling area volume cannot be assumed, only 50% of the metal siding area is assumed in the determination of the diffuse area X/Q. This limited area is assumed to be at the worst case location with relationship to either CR intake location to maximize the calculated dose.

Plant procedures direct that secondary containment integrity be maintained when handling heavy loads (greater than one fuel assembly) such as the reactor vessel head or dryer/separator) in, above, or around the reactor cavity with fuel in the reactor vessel, to provide additional protection. Plant procedures also direct that secondary containment integrity be maintained when handling heavy loads in, above, or around the spent fuel pool.

The trunnion room door is not an access opening for the passage of personnel and equipment into the reactor building. During all modes of operation, the trunnion room is a low traffic area and momentary openings of the door would be limited and administratively controlled and have little effect on SGTS and HVAC.

The standby gas treatment system[6] filters and exhausts the reactor building atmosphere to the stack during secondary containment isolation conditions, with a minimum release of radioactive materials from the reactor building to the environs. Due to radioactive decay, during fuel handling operations these controls are only required when handling RECENTLY IRRADIATED FUEL; or during operations with the potential to drain the reactor vessel; or when handling heavy loads in, above, or around the reactor cavity or spent fuel pool.

In Section 3.5.B.5 and 3.5.B.6 of the Technical Specification, the use of the word "Circuits" actually *means "Trains" as the word trains is used in the following paragraph.*

Two separate filter trains are provided, each having 100% capacity[6]. There is a section of ductwork upstream and downstream that is common to both filter trains. If one filter train becomes inoperable, there is no immediate threat to secondary containment and reactor operation may continue while repairs are being made. Since the test interval for this system is one month (Specification 4.5), the time out-of-service allowance of 7 days is based on considerations presented in the Bases in Specification 3.2 for a one-out-of-two system.

There is also only one vital power supply to the SGTS automatic initiation controls and for the operation of the heating coils for both filter trains.

Therefore, the SGTS is not mechanically nor electrically single failure proof. However, manual actuation of the SGTS is not vulnerable to single failures and is an acceptable backup to automatic initiation.

Two automatic secondary containment isolation valves are installed in each reactor building ventilation system supply and exhaust duct penetration. Both isolation valves for each supply duct penetration are located inside the secondary containment boundary, and the two exhaust duct penetration isolation valves are located outside of the secondary containment boundary. Removal of an inboard supply or exhaust valve (closest to the boundary) is permitted only when secondary containment is not required. The outboard isolation supply or exhaust valve can be removed when secondary containment is required as long as the inboard valve is secured in the closed position. Due to radioactive decay, during fuel handling operations the secondary containment isolation valves are only required to be OPERABLE when handling RECENTLY IRRADIATED FUEL; during operations with the potential to drain the reactor vessel; or when handling heavy loads in, above, or around the reactor cavity or spent fuel pool.

The addition of the term RECENTLY IRRADIATED FUEL associated with handling irradiated fuel in all secondary containment function Technical Specification requirements is applicable since analysis has demonstrated that after sufficient radioactive decay has occurred, off-site and control room operator doses resulting from a fuel handling accident remain below the limits of 10 CFR 50.67.

The following guidelines are included in the assessment of systems removed from service during movement of irradiated fuel:

- During fuel handling/core alterations, ventilation system and radiation monitor availability (as defined in NUMARC 91-06) should be assessed, with respect to filtration and monitoring of releases from fuel. Following shutdown, radioactivity in the fuel decays away fairly rapidly. The basis of the Technical Specification operability amendment is the reduction in doses due to such decay. The goal of maintaining ventilation system and radiation monitor availability is to reduce doses even further below that provided by the natural decay.
- A single normal or contingency method to promptly close secondary containment penetrations should be developed for those times when secondary containment is not required. Such prompt methods need not completely block the penetration or be capable of resisting pressure. The purpose of the "prompt methods" mentioned above are 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.

In addition to the above, in the event of a fuel handling accident, one standby gas treatment circuit shall be running within one hour and all secondary containment openings must be closed within one hour. These actions will 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 as discussed above.

A fuel handling accident is an event that could result in the release of significant quantities of fission products from the accidental dropping of a equipment (including fuel bundles) onto irradiated fuel in the reactor core or spent fuel.

References:
(1) FDSAR, Volume I, Section V-1
(2) FDSAR, Volume I, Section V-1.4.1
(3) FDSAR, Volume I, Section V-1.7
(4) Licensing Application, Amendment 11, Question III-25
(5) FDSAR, Volume I, Section V-2
(6) FDSAR, Volume I, Section V-2.4
(7) Licensing Application, Amendment 42
(8) Licensing Application, Amendment 32, Question 3
(9) Robbins, C. H., "Tests on a Full Scale 1/48 Segment of the Humboldt Bay Pressure Suppression Containment," GEAP-3596, November 17, 1960.
(10) Bodega Bay Preliminary Hazards Summary Report, Appendix I, Docket 50-205, December 28, 1962.
(11) Report H. R. Erickson, Bergen-Paterson To K. R. Goller, NRC, October 7, 1974. Subject: Hydraulic Shock Sway Arrestors.
(12) General Electric NEDO-22155 "Generation and Mitigation of Combustible Gas Mixtures in Inerted BWR Mark I Containment" June 1982.
(13) Oyster Creek Nuclear Generating Station, Mark I Containment Long-Term Program, Plant Unique Analysis Report, Suppression Chamber and Vent System, MPR-733; August, 1982.

(14) Oyster Creek Nuclear Generating Station, Mark I Containment Long-Term Program, Plant Unique Analysis Report, Torus Attached Piping, MPR-734; August, 1982.

(15) AmerGen Calculation C-1302-243-E170-087, "Wetwell-to-Drywell Vacuum Breaker Sizing."

(16) General Electric NEDE-24802, "Mark I Containment Program Mark I Wetwell-to-Drywell Vacuum Breaker Functional Requirements, Task 9.4.3," April, 1980.

(17) Technical Specification Task Force (TSTF) Improved Standard Technical Specifications Traveler TSTF-51-A, Rev. 2

(18) NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management"

During each refueling outage, four suppression chamber-drywell vacuum breakers will be inspected to assure components have not deteriorated. Since valve internals are designed for a 40-year lifetime, an inspection program which cycles through all valves in about 1/10th of the design lifetime is extremely conservative. The alarm systems for the vacuum breakers will be calibrated during each refueling outage. This frequency is based on experience and engineering judgement.

Initiating reactor building isolation and operation of the standby gas treatment system to maintain a 1/4 inch of water vacuum, tests the operation of the reactor building isolation valves, leakage tightness of the reactor building and performance of the standby gas treatment system. Checking the initiating sensors and associated trip channels demonstrates the capability for automatic actuation. Performing the reactor building in leakage test prior to refueling demonstrates secondary containment capability prior to ~~extensive~~ fuel handling operations **involving RECENTLY IRRADIATED FUEL** associated with the outage. Verifying the efficiency and operation of charcoal filters once per 18 months gives sufficient confidence of standby gas treatment system performance capability. A charcoal filter efficiency of 99% for halogen removal is adequate.

The in-place testing of charcoal filters is performed using halogenated hydrocarbon refrigerant which is injected into the system upstream of the charcoal filters. Measurement of the refrigerant concentration upstream and downstream of the charcoal filters is made using a gas chromatograph. The ratio of the inlet and outlet concentrations gives an overall indication of the leak tightness of the system. Although this is basically a leak test, since the filters have charcoal of known efficiency and holding capacity for elemental iodine and/or methyl iodide, the test also gives an indication of the relative efficiency of the installed system. The test procedure is an adaptation of test procedures developed at the Savannah River Laboratory which were described in the Ninth AEC Cleaning Conference.*

High efficiency particulate filters are installed before and after the charcoal filters to minimize potential releases of particulates to the environment and to prevent clogging of the iodine filters. An efficiency of 99% is adequate to retain particulates that may be released to the reactor building following an accident. This will be demonstrated by testing with DOP at testing medium.

The 95% methyl iodide removal efficiency is based on the formula in GL 99-02 for allowable penetration [(100% - 90% credited in DBA analysis) divided by a safety factor of 2]. If the allowable penetration is ≤5%, the required removal efficiency is ≥95%. If laboratory tests for the adsorber material in one circuit of the Standby Gas Treatment System are unacceptable, all adsorber material in that circuit shall be replaced with adsorbent qualified according to Regulatory Guide 1.52. Any HEPA filters found defective shall be replaced with those qualified with Regulatory Position C.3.d of Regulatory Guide 1.52.

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* D.R. Muhabier. "In Place Nondestructive Leak Test for Iodine Adsorbers." Proceedings of the Ninth AEC Air Cleaning Conference. USAEC Report CONF-660904, 1966