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10 CFR 50.90

November 2, 2007
2130-07-20501

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

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

Subject: Technical Specification Change Request No. 338 – Secondary Containment Operability Requirements During Refueling

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," AmerGen Energy Company, LLC (AmerGen) proposes changes to Appendix A, Technical Specifications (TS), of the Oyster Creek Nuclear Generating Station (Oyster Creek) Facility Operating License. Enclosure 1 contains AmerGen's description and assessment of the change. Enclosure 2 contains the proposed TS changes.

The proposed change would revise Oyster Creek TS 3.5.B, "Secondary Containment," to eliminate the requirement for secondary containment to be operable during handling of irradiated fuel. Related changes to Technical Specifications 1.0, 3.5, 3.5 Bases, 3.17, and 4.5 Bases are also proposed. The Fuel Handling Accident (FHA) has been analyzed using 10 CFR 50.67, "Accident source term." The results of that analysis show that the postulated Exclusion Area Boundary (EAB), Low Population Zone (LPZ), and Control Room (CR) dose consequences remain well within the limits specified by 10 CFR 50.67 and NRC Regulatory Guide 1.183.

The proposed amendment has been reviewed by the Oyster Creek Plant Operations Review Committee and approved by the Nuclear Safety Review Board in accordance with the requirements of the AmerGen Quality Assurance Program.

Using the standards in 10 CFR 50.92, AmerGen has concluded that these proposed changes do not constitute a significant hazards consideration, as described in the enclosed analysis performed in accordance with 10 CFR 50.91(a)(1). Pursuant to 10 CFR 50.91(b)(1), AmerGen is notifying the State of New Jersey of this application for changes to the TS by transmitting a copy of this letter and its attachments to the designated State Official.

A001
NRR

We request approval of the proposed change by September 28, 2008, with the amendment being implemented within 30 days of issuance. This will allow an orderly implementation of these changes following approval to support the 1R22 refueling outage scheduled for the Fall of 2008.

Regulatory commitments established by this submittal are identified in Enclosure 3. If you have any questions or require additional information, please contact Mr. David Distel at (610) 765-5517.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 2nd day of November, 2007.

Respectfully,

Pamela B. Cowan
Director - Licensing and Regulatory Affairs
AmerGen Energy Company, LLC

Enclosures:
1) Oyster Creek Nuclear Generating Station Technical Specification Change Request No. 338 - Secondary Containment Operability Requirements During Refueling
2) Oyster Creek Nuclear Generating Station Technical Specification Change Request No. 338 - Markup of Proposed Technical Specification and Bases Page Changes
3) List of Commitments

cc: S. J. Collins, Administrator, USNRC Region I
M. S. Ferdas, USNRC Senior Resident Inspector, Oyster Creek
G. E. Miller, USNRC Project Manager, Oyster Creek
P. Baldauf, Assistant Director, New Jersey Bureau of Nuclear Engineering
Oyster Creek File No. 07042

ENCLOSURE 1

Oyster Creek Technical Specification Change Request No. 338

Secondary Containment Operability Requirements During Refueling

Description and Assessment

Subject: Secondary Containment Operability Requirements During Refueling

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ATTACHMENTS:

Attachment 1: AmerGen/Exelon Calculation C-1302-822-E310-082, Revision 0, "Oyster Creek analysis of Fuel Handling Accident (FHA) using Alternative Source Terms (AST)"

Attachment 2: Regulatory Guide 1.183 – Conformance Tables

Attachment 3: AmerGen/Exelon Calculation C-1302-822-E310-081, Revision 0, "Oyster Creek Onsite Atmospheric Dispersion (X/Q) for Fuel Handling Accident (FHA)"

ENCLOSURE 1

DESCRIPTION AND ASSESSMENT

1.0 Description

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," AmerGen Energy Company, LLC (AmerGen) is requesting an amendment to Facility Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station (Oyster Creek). The proposed change would revise Oyster Creek Technical Specification (TS) 3.5.B, "Secondary Containment" to eliminate the requirement for secondary containment when handling irradiated fuel. The proposed change provides flexibility in scheduling outage tasks and to modify unnecessarily restrictive containment closure and ventilation system requirements, and is consistent with TSTF-51, Revision 2 to NUREG-1433 Vol. 1, Rev. 2, Standard Technical Specifications, General Electric Plants. Secondary Containment operability requirements remain applicable for work being performed on the reactor or its connected systems in the reactor building which could result in the potential to drain the reactor vessel or when handling recently irradiated fuel. The Fuel Handling Accident has been analyzed using a revised accident source term in accordance with 10 CFR 50.67. Alternative Source Term (AST) methodology has been previously approved by NRC for use at Oyster Creek in TS Amendment No. 262, dated April 26, 2007.

Related changes to Technical Specifications 1.0, 3.5, 3.5 Bases, 3.17, and 4.5 Bases are also proposed.

AmerGen requests that the following changed replacement pages be inserted into the existing Technical Specifications:

> Revised Oyster Creek TS Pages: 1.0-8, 3.5-5, 3.5-6, 3.5-7, 3.5-8, 3.5-11, 3.5-12, 3.5-12a, 3.17-1, and 4.5-13.

The proposed changed replacement TS pages are provided in Enclosure 2.

2.0 Proposed Changes

The proposed revisions to the Oyster Creek TS are being made to allow relaxation of secondary containment operability requirements during fuel handling operations when handling fuel that has not occupied part of a critical reactor core within the previous 24 hours. This change is evaluated 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 Vol. 1, Rev. 2, Standard Technical Specifications, General Electric Plants.

Existing secondary containment integrity requirements specified in TS 3.5.B remain applicable when fuel handling operations involve fuel that has occupied part of a critical reactor core within the previous 24 hours.

2.1 TS Definition 1.49, RECENTLY IRRADIATED FUEL, is being added to specify when the existing TS provisions for secondary containment integrity remain applicable.

2.2 TS Section 3.5.B.1.d is revised to allow the handling of irradiated fuel without secondary containment integrity when the reactor has been subcritical for greater than 24 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 24 hours), or when work is being performed on the reactor or its connected systems in the reactor building which have 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.

2.3 TS Section 3.5.B.1.d (revised TS Section 3.5.B.1.c) and TS Section 3.5.B.4.b.(2) are revised to substitute the wording "...the potential to drain the reactor vessel...", in lieu of "...inadvertent release of radioactive material...", consistent with TSTF-51, Rev. 2.

2.4 TS Sections 3.5.B.1.e and 3.5.B.4.b.(3) are deleted in order to allow the movement of sufficiently decayed irradiated fuel in, above, or around the spent fuel storage pool.

2.5 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 revised to specify that fuel handling operations involving recently irradiated fuel shall cease. The current restriction on activities that could reduce the shutdown margin is being deleted since the existing TS 3.5.B.1.a requirement ensures shutdown margin requirements will be met if secondary containment integrity will not be maintained.

2.6 TS Section 3.5.B.6.b.(3) is revised to specify that during refueling the operability requirements for the Standby Gas Treatment system apply when fuel handling operations involve recently irradiated fuel, or when performing operations with the potential to drain the reactor vessel. The current restriction on activities that could reduce the shutdown margin is being deleted since the existing TS 3.5.B.1.a requirement ensures shutdown margin requirements will be met if secondary containment integrity will not be maintained.

2.7 TS Sections 3.5 and 4.5 Bases are revised to incorporate the basis for the proposed changes, including the basis for the term "recently irradiated fuel," and to incorporate the defense-in-depth guidelines contained in TSTF-51, Rev. 2.

2.8 TS Section 3.17 is 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, Rev. 2. The proposed markup of TS Section 3.17 provided in Enclosure 2 also contains the previously submitted markups incorporating the provisions of TSTF-448, Control Room Envelope Habitability, which are currently under NRC review. These changes were submitted in Exelon/AmerGen letter to the NRC dated April 12, 2007 (2130-07-20457).

3.0 Background

3.1 The Alternative Source Term Methodology (AST) for Oyster Creek has been approved for the Loss-of-coolant accident (LOCA) analysis of radiological dose consequences. AST is now being applied to the analysis of the design basis Fuel Handling Accident (FHA) for Oyster Creek. This analysis considers normal (unfiltered) exhaust through the reactor building ventilation stack and other openings, identified in Table 4-1 below, in support of the proposed changes to the current Oyster Creek Technical Specifications to allow that at certain times secondary containment integrity and the operability of emergency filtration system (SGTS) and subsystems are not required to mitigate the radiological consequences of fuel handling accidents. The secondary containment openings evaluated consist of normal building airlock and personnel access points, equipment and commodities penetrations, and SGTS ductwork maintenance locations.

Not having to consider secondary containment integrity and filtration requirements of SGTS in support of refueling activities has the potential to significantly improve the flexibility and duration of scheduled plant outage activities.

3.2 Release points for the potential normal openings in secondary containment (Table 4-1) were considered and evaluated. Plant walk-downs and review of plant general arrangement and layout drawings were completed to ensure that the most limiting of the release locations identified in Table 4-1 were analyzed. The most limiting X/Q values associated with these openings were utilized in the radiological dose analysis supporting fuel movement after the 24-hour decay period.

3.3 As described in TSTF-51, Revision 2, accidents postulated to occur during core alterations, in addition to fuel handling accidents, are inadvertent criticality (due to control rod removal error or continuous control rod withdrawal error during refueling) and the inadvertent loading of, and subsequent operation with, a fuel assembly in an improper location. These events are not postulated to result in fuel cladding integrity damage. Since the only accident postulated to occur during core alterations that results in a significant radioactive release is the fuel handling accident, the proposed TS changes apply when "core alterations" involve recently irradiated fuel.

3.4 Guidance in TSTF-51 suggests that "recently irradiated fuel" parameters be developed to identify the point in time after shutdown when secondary containment features are required. Therefore, this evaluation supports the proposed definition of RECENTLY IRRADIATED FUEL as being:

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

TSTF-51, Revision 2, requires licensees incorporating this change to commit to NUMARC 93-01, Revision 3, Section 11.2.6, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," subheading "Containment-Primary (PWR)/Secondary (BWR)."

The commitment in TSTF-51, Revision 2, was based on a draft version of NUMARC 93-01, Revision 3. When NUMARC 93-01, Revision 3 was approved in July 2000, the guidelines

referred to in TSTF-51, Revision 2, were designated as Section 11.3.6.5. Section 11.3.6.5 of NUMARC 93-01 states:

> *Maintenance activities involving the need for open containment should include evaluation of the capability to achieve containment closure in sufficient time to mitigate potential fission product release. This time is dependent on a number of factors, including the decay heat level and the amount of RCS inventory available.*
>
> *For BWRs, Technical Specifications may require secondary containment to be closed under certain conditions, such as during fuel handling and operations with a potential to drain the vessel.*
>
> *In addition to the guidance in NUMARC 91-06, for plants which obtain license amendments to utilize shutdown safety administrative controls in lieu of Technical Specification requirements on primary or secondary containment operability and ventilation system operability during fuel handling or core alterations, the following guidelines should be included in the assessment of systems removed from service:*
>
> *During fuel handling and core alterations, ventilation systems and radiation monitor availability (as defined in NUMARC 91-06) should be assessed with respect to filtration and monitoring of releases from the fuel. Following shutdown, radioactivity in the RCS decays 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, and to avoid unmonitored releases.*
>
> *A single normal or contingency method to promptly close primary or secondary containment penetrations should be developed. Such prompt methods need not completely block the penetration or be capable of resisting pressure. The purpose is to enable ventilation systems to draw the release from a postulated fuel handling accident in the proper direction such that it can be treated and monitored.*

Oyster Creek will follow the guidelines in Section 11.3.6.5 of NUMARC 93-01, Revision 3, during refueling within containment. Plant procedures will be revised, as appropriate, to implement these guidelines. This is also being placed in the Bases for TS Section 3.5.

3.5 Analyses of radiation transport and dose assessment are performed using RADTRAD v. 3.03. RADTRAD is a simplified model of RADionuclide Transport and Removal And Dose Estimation developed for the NRC and endorsed by the NRC as an acceptable methodology for reanalysis of the radiological consequences of design basis accidents. The technical basis for the RADTRAD code is documented in Section 2 of NUREG/CR-6604 (Reference 7.1). The methodologies significant to this analysis are the dose

consequence analysis (NUREG/CR-6604, Section 2.3) and the Radioactive Decay Calculations (NUREG/CR-6604, Section 2.4).

3.6 Although not related to the licensing basis for secondary containment, in order to maintain the level of protection currently provided, plant procedures will continue to require secondary containment integrity during the handling of 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.

4.0 Technical Analysis

4.1 Accident Source Term

The source term nuclide inventory used for the Fuel Handling Accident (FHA) is the same as that previously used in the LOCA AST analysis. This inventory is considered to be bounding with respect to expected fuel types in use at Oyster Creek.

Regulatory Guidance for DBA fuel handling accidents is such that the fission product inventory of each of the damaged fuel rods is determined by dividing the total core inventory by the number of fuel rods in the core. To account for differences in power level across the core, radial peaking factors are applied in determining the inventory of the worst-case damaged rods.

Consistent with the current Oyster Creek UFSAR accident analysis, the fraction of the core fuel damaged is based on the GESTAR II limiting case of damaging 140 fuel rods (based on a "Heavy Mast" design; i.e., the "NF500 mast") from GE11 or GE13 9x9 fuel bundle arrays with the analyzed equivalent of 74 rods per bundle, and with all of the damaged fuel assumed to have a limiting Radial Peaking Factor (PF) of 1.7. This analysis is for an assembly and mast drop from the maximum height allowed by the refueling platform over the reactor well onto fuel in the reactor, and bounds all locations in terms of fuel damage potential. Oyster Creek core loads consist of all GE11 9x9 fuel assemblies, and therefore, the results of the GESTAR II limiting case are consistent with Oyster Creek core loading.

Bundle Type	Fuel Array	A Rods in Bundle	B Failed Rods	C Damaged Core Fraction Assuming Core is All Specified Bundle Type **[C=B/(560*A)]**	D Radial Peaking Factor (PF)	E Damaged Core Fraction with PF **[E=C*D]**
GE11, GE13	9x9	74	140	0.003378	1.7	0.005743
Various	8x8	62	124	0.003571	1.5	0.005356
Various (Fuel Pool only)	7x7	49	111	0.004045	1.5	0.006068

Only the GE11 and GE13, 9x9 array, is analyzed because the 9x9 assembly core damage fraction bounds the 8x8 assembly, and the 7x7 assemblies are only located in the fuel pool and have experienced sufficient iodine decay. With a 1.7 radial peaking factor, the associated power of the damaged fuel = 1969 MWth * 0.003378 * 1.7 = 11.31 MWth.

Because of radioactive decay, the worst-case fuel handling accident is that associated with handling fuel that has most recently been part of a critical core operating at full power. Movement of irradiated fuel will not occur less than 24 hours after the associated reactor shutdown, and therefore, a 24-hour delay period is the minimum decay period assumed. This value continues to be a very conservative assumption for BWRs, given the operations necessary before commencing fuel movement. Radioactive decay from the time of shutdown is modeled for 24 hours to support the opening of the most restrictive (i.e., highest X/Q) secondary containment potential release point.

4.2 Gap Activity

This calculation is applicable to fuel whose burnup and power limits are bounded by those specified in RG 1.183, footnote 11 (Reference 7.4). This allows application of the gap activity fractions for LOCA events per RG 1.183, Table 3, which are as follows:

5% of the noble gases (excluding Kr-85)
10% of the Kr-85
5% of the iodine inventory (excluding I-131)
8% of the I-131
12% of the Alkali metal inventory

Because RADTRAD does not allow for application of isotope specific release fractions, the "OC AST Source Term.nif" file is modified to accommodate the differential gap activities among the halogen (I-131) and noble gas (Kr-85) gap fractions dictated by RG 1.183 (Ref. 7.4), shown above. Therefore, the initial activity of isotope I-131 and Kr-85 are multiplied by 1.6 and 2.0, respectively, in order to accommodate the respective 8% and 10% release fractions directed by regulatory guidance.

4.3 Decontamination Factor (DF)

An assessment of water coverage and effective Decontamination Factors (DFs) considering FHAs over the reactor well and in the spent fuel pool was performed. This assessment identified that the drop over the reactor well is more limiting. Water coverage over the reactor well is approximately 30 feet.

For a drop over the spent fuel pool, coverage over the dropped assemblies is slightly less than 23 feet, assuming that the dropped bundle is laying across the tops of fuel bundles within the spent fuel racks. There is always greater than 23 feet of water above top of active fuel for bundles within the racks. No additional credit is taken for water depth greater than 23 feet in order to maintain conservatism and consistency with regulatory guidance.

The slight reduction in DF due to slightly less than 23 feet water coverage in the event of a drop in the fuel pool is offset by the reduction in fuel damage due to a much shorter drop (i.e., less than approximately 4 feet) over fuel in the spent fuel pool.

4.4 Release Model

Release modeling uses the RADTRAD V3.03 computer program. The compartments are the Refuel Floor Air Space (in the Reactor Building), the Environment (EAB and LPZ), and

the Control Room. The refuel floor exhaust rate is set artificially high. This results in 99.9999% of the contained radioactivity being exhausted within two hours.

The Release Fraction and Timing files for this event provides for a rapid release (1.0E-04 hours) of gap activity to the fuel pool and subsequently to the refuel floor air space. The nominal gap fraction for noble gas is 5%, and for iodine is also 5%. The iodine release is further reduced to reflect the pool DF of 200. As discussed above, the normal Nuclide Inventory File representing an Oyster Creek core is artificially adjusted to account for the higher than average gap fractions for I-131 and Kr-85 provided by Regulatory Guide (RG) 1.183.

This accident analysis evaluates the movement of fuel that has decayed a minimum of 24 hours since it occupied part of a critical reactor core. The plant vent stack is treated as a ground level (rather than elevated) release and no SGTS is credited.

4.5 Control Room Model

The Oyster Creek Control Room HVAC system contains no HEPA or Charcoal filters. All intake air into the control room is unfiltered. As a result, the radionuclide concentration inside the control room reaches equilibrium with the concentration of the plume at the Control Room HVAC System air intake within a short period of time after the postulated release. Equilibrium occurs whether the ventilation system was in purge or partial recirculation mode of operation. In the purge mode (14,000 cfm system flow) the control room volume is changed once every 2 minutes. In the partial recirculation mode (2,000 cfm system intake flow) the control room volume is changed once every 14 minutes. It would require only one hour for the control room to be at radiological equilibrium with the outside environment while in the partial recirculation mode. The dose to control room operators is calculated over a total duration of 720 hours. Oyster Creek LAR No. 315, "Application of Alternative Source Term, submitted to NRC on March 28, 2005 (2130-05-20040), reanalyzed the design basis LOCA radiological dose consequences for Oyster Creek using Alternative Source Term methodology. The control room operator dose consequences are conservatively analyzed using the full Control Room HVAC System flow seen during the purge mode of operation. It should be noted that the purge mode (14,000 cfm intake) is not expected to be in operation during a radiological event. Therefore, the assumption that the Control Room HVAC system is operating in the purge mode for the duration of the FHA results in a conservative radiological dose to control room operators. In both modes of operation (purge and partial recirculation) the radiation exposure to personnel in the control room remains within the regulatory limits. The partial recirculation mode (2,000 cfm) produces a slightly lower dose than the purge mode. The sensitivity study included in the Oyster Creek LOCA analysis bounds the range of flow between the 2,000 cfm minimum flow in the partial recirculation mode, and the maximum flow of approximately 30,000 cfm for the unlikely event both fans are operating in the full purge mode. Therefore, since any increase in flow beyond 4,000 cfm does not cause an increase in dose, the amount of unfiltered inleakage does not affect the total calculated control room operator dose (calculated at 14,000 cfm). The analysis model and assumptions described above have been previously accepted by the NRC (Reference 7.3).

4.6 Dispersion Model

Dispersion factors at given increments in time have been calculated (Attachment 3). The X/Q's are based on RG 1.194 methodology (Reference 7.5) as implemented by ARCON96 for onsite locations (Control Room) and on the RG 1.145 methodology as implemented by PAVAN (Reference 7.6) for offsite locations (EAB & LPZ).

The list of evaluated release points is included in Table 4-1, below. The release points evaluated consist of openings related to normal personnel and equipment access points such as doors and hatches (the diffuse reactor building wall surfaces are also identified in the calculation and modeled in accordance with RG 1.194). This evaluation does not include breaches in the secondary containment (except for the HVAC ductwork at the base of the main stack shown in Table 4-1) for purposes of plant modifications or maintenance for which the boundary is opened in non-designed locations unless specifically analyzed. Secondary containment integrity, in this analysis, is in reference to normal personnel/equipment doors and hatches identified in Table 4-1. Except for the stack tunnel door (for which disassembly was planned and evaluated) and the flanged commodities penetrations (typically opened during outages and which are evaluated here), the secondary containment boundary cannot be breached in other locations without further evaluation.

The evaluated release points listed in Table 4-1 will be identified in the Oyster Creek Updated Final Safety Analysis Report (UFSAR) and in appropriate plant procedures governing secondary containment integrity and refueling activities. Evaluation of other locations will be administratively controlled in accordance with the requirements of 10 CFR 50.59. These controls will ensure that future evaluations of any additional openings in secondary containment will remain bounded by this analysis.

The meteorological data used in the X/Q determination is the same as that used for the Oyster Creek LOCA evaluation previously submitted to the NRC in AmerGen letter dated March 16, 2007 (2130-07-20473), and approved by the NRC in Amendment No. 262, dated April 26, 2007.

The X/Q values resulting at the Control Room HVAC Intakes are calculated using the NRC-sponsored computer code ARCON96, consistent with the procedures in RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," (Reference 7.5).

A five-year (i.e., 1995 -1999) onsite hourly meteorological database was utilized in the ARCON96 modeling of X/Q.

<u>ARCON96 Modeling Analysis of Control Room X/Q</u>

Assumptions

The following assumptions have been made relative to specific release point locations:

- The Reactor Building (RB) East Airlock Door release height is assumed to be at the vertical center of an 8 ft high doorway.

- Consistent with the application of the "taut string method" described in RG 1.194 (Reference 7.5), the intake height is set equal to the release height for the X/Q evaluations of point sources.
- The Drywell (D/W) Access Facility and the Monitor and Control (MAC) Facility Entrance release heights are assumed to be at the vertical center of the doorways
- The MAC Facility Personnel Airlock release is assumed to be at the vertical center of the tornado/missile shield.

Source/Receptor Scenarios and Configurations

Table 4-1 outlines each of the potential release locations evaluated after plant walkdowns and reviews. Of these, the following were selected for ARCON96 modeling.

- RB Roof Hatch
- Stack Tunnel Door
- D/W Access Facility (door on west wall and western-most door on south wall)
- East Airlock Door
- RB Diffuse Release
- Commodities Penetrations on the South RB Wall
- Commodities Penetrations on the North RB Wall
- MAC Facility Personnel Airlock
- MAC Facility Entrance

Table 4-1 also provides the justification for each of the evaluated release points not selected for modeling. Specifically, for each such point, Table 4-1 identifies the nearest bounding modeled release point that is at approximately the same elevation, and a shorter distance from both Intakes A and B.

The X/Q values resulting from the ARCON96 modeling analysis of each source/intake scenario are presented in Attachment 3. The most limiting X/Q results used in the FHA analysis are shown in Table 4-3, below.

Table 4-1
Potential Release Points Considered For Oyster Creek Fuel Handling Accident (With Respect To CR Intakes)

Description	Comments	X/Q Modeling Required?
RB Roof hatch NOTE: This hatch has an airlock associated with it so roof access is possible during plant operation.	It is feasible that the airlock could be defeated during an outage if significant roof maintenance is required.	YES (Values calculated)
RB grade-level access at south east corner of RB: This access is a security barrier and will not normally be opened.	It is possible that this access point could be open with appropriate security measures in place during an outage with fuel movement.	NO X/Q bounded by RB commodities penetration on south RB wall (Item 10)[1]
Main stack exhaust fans and ductwork at(base of main stack (also known as the Stack tunnel Door)	This ductwork could be dismantled during an outage with fuel movement in progress. Support for this work is the reason for this evaluation.	YES (Values calculated)
RB entrance (D/W Access Facility) There is an airlock associated with this access point.	If the airlock is defeated, the actual point of release could be at the entrance to the D/W Access Facility (northeast of RB), which is closer to the CR intake.	YES[2] (Values calculated)
RB personnel access airlock on east wall of 23' 6" elevation RB wall (near columns RA and R5).	It may be beneficial if the airlock could be defeated during an outage with fuel movement. The actual point of release is directly in front of the airlock.	YES (Values calculated)
RB Truck Airlock (at column RA, between columns R2 and R3):	This airlock could be open during an outage with fuel movement.	NO X/Q bounded by RB commodities penetration on south RB wall (Item 10) and/or RB personnel airlock on east wall (Item 6A)[1]
RB diffuse release	This area source is from exposed RB walls "visible" to the CR intakes.	YES (Values calculated)
RB commodities penetration on south RB wall[3] (23.5' elev.)	This is a flanged connection through which, air, electric, and water connections are run. Existing procedures allow for use.	YES (Values calculated)
RB commodities penetration on north RB wall (23.5' elev.)	This is a flanged connection through which, air, electric, and water connections are run. Existing procedures allow for use.	YES (Values calculated)
MAC Facility Personnel Airlock (exits out of tornado/missile protection area located on the north RB wall (23.5' elev.)		YES (Values calculated)
MAC Facility Entrance (double doors)		YES (Values calculated)
MAC Facility Entrance (single door)		NO X/Q bounded by the MAC Facility Personnel Airlock (Item 12)[1]
Trunion Room Door (at Column Lines R4 and RG)	This room, which could be opened during an outage, contains the Outboard MSIVs. The door opens into the Turbine building. The room is serviced by Reactor Building HVAC systems.	NO X/Q values for Turbine Building releases were previously calculated for MSIV leakage during a LOCA. Therefore, the same X/Q values will be used for this release location for the FHA.

[1] Bounding modeled release point most proximate to subject release point.
[2] There are four (4) doors associated with the D/W Access Facility (Item 6). The X/Q for the door on the northern wall would be bounded by the door on the west wall, which is closer to Intakes A and B. Similarly, on the southern wall, the X/Q for the eastern-most door would be bounded by the western-most door, which is closer to Intakes A and B.
[3] There are two commodities penetrations located on the south RB wall (Item 10). The taut string length was calculated from each penetration to both Intake A and B, and the penetration of shortest length to Intake A and to Intake B was selected to be modeled.

Each of the Oyster Creek calculated release points has two (2) associated receptors, Intake A, with a vertical center height of above grade level of 13.7 m, and Intake B, with a vertical center height of 18.1 m.

All scenarios are modeled as point sources in ARCON96, except the Reactor Building Wall scenarios, which are modeled as a "diffuse area" source.

All point source scenarios are conservatively assumed to have vertical velocity, exhaust flow and stack/vent radius values equal to zero (0).

Point Sources

ARCON96 requires an input of horizontal source-receptor distance, which is defined in RG 1.194, Section 3.4, as "the shortest horizontal distance between the release point and the intake". However, for releases in building complexes, a "taut string length" can be utilized as justifiable. For each applicable point source, the "taut string length" distances to Intakes A and B were utilized to account for the intervening Reactor Building. When the "taut string length" is utilized, the intake and release height are set equal to each other so as not to also take undue advantage of the slant distance that ARCON96 calculates. Therefore, for each of the scenarios, the intake height was set equal to the release height.

The height of each of the release points are all less than 2.5 times the height of their adjacent buildings; and therefore, per RG 1.194 they are modeled as ground-level releases. Aerodynamic building plume downwash effects are present for ground-level releases; therefore, in accordance with RG 1.194, Table A-2, the building cross-sectional area perpendicular to the wind direction is utilized. Attachment 3 contains calculations of the projected area of the Reactor Building for each of the point sources and, where applicable, calculations of the "redirected" intake-to-source direction and the associated projected area of the Reactor Building. The redirected projected area is derived based on an intake-to-source direction that is adjusted to account for the redirected flow from the nearest taut string building edge to the intake per RG 1.194, Table A-2.

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

Per RG 1.194, Section 3.2.4.5, the diffuse area source representation in ARCON96 requires the building cross-sectional area to be calculated from the maximum building dimensions projected onto a vertical plane perpendicular to the line of sight from the building center to the intake.

RG 1.194, Figure 2, specifies that, for a diffuse area source, "only that part of the structure above grade or an enclosing building should be included in the building height." For the Reactor Building Wall scenarios, the portion of the Reactor Building above the Office Building roof height was utilized for determining the release height, building area and vertical diffusion coefficient (σ_z). However, since the Office Building only borders one side of the Reactor Building, a set of "alternate" values for these parameters based on the entire height of the Reactor Building were also derived.

RG 1.194 also requires the diffuse area source release height to be assumed at the vertical center of the projected area, and initial lateral (σ_y) and vertical (σ_z) diffusion coefficients to be specified.

A summary of ARCON96 input parameters is provided for each source/intake scenario in Table 2 of Attachment 3.

4.7 Dose Modeling

Dose models for both onsite and offsite meet RG 1.183 requirements. Dose conversion factors are based on Federal Guidance Reports 11 and 12, as referenced in RG 1.183. RADTRAD uses the following formulations, integrated numerically over the accident duration:

EAB and LPZ

Doses at the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) for the FHA are based on the following formulas:

$$\text{Dose}_{\text{CEDE}}\ (\text{rem}) = \text{Release (Curies)} * \frac{\chi}{Q}(\text{sec/m}^3) * \text{Breathing Rate (m}^3\text{/sec)} * \text{Inhalation DCF (rem}_{\text{CEDE}}\text{/Ci inhaled)}$$

and

$$\text{Dose}_{\text{EDE}}\ (\text{rem}) = \text{Release (Curies)} * \frac{\chi}{Q}(\text{sec/m}^3) * \text{Submersion DCF (rem}_{\text{EDE}}\text{ - m}^3\text{/ Ci - sec)}$$

and finally,

$$Dose_{TEDE}\ (rem) = Dose_{CEDE}\ (rem) + Dose_{EDE}(rem)$$

Control Room

The formulas used by RADTRAD, by time increment, are:

$$\text{Dose}_{\text{CEDE}}\ (\text{rem}) = \text{Time Dependent CR Air Concentration (Ci/m}^3) * \text{Time Increment Duration (sec)} * \text{Breathing Rate (m}^3\text{/sec)} * \text{Inhalation DCF (rem}_{\text{CEDE}}\text{/Ci inhaled)} * \text{Occupancy Factor of 1}$$

and

$$\text{Dose}_{\text{EDE}}\ (\text{rem}) = \text{Time Dependent CR Air Concentration (Ci/m}^3) * \text{Time Increment Duration (sec)} * \text{Submersion DCF (rem}_{\text{EDE}}\text{ - m}^3\text{/ Ci - sec)} * \text{Occupancy Factor of 1} * \text{CR Geometry Factor}$$

and finally,

$$\text{Dose}_{\text{TEDE}}\ (\text{rem}) = \text{Dose}_{\text{CEDE}}\ (\text{rem}) + \text{Dose}_{\text{EDE}}\ (\text{rem})$$

4.8 Dose Acceptance Criteria

Dose acceptance criteria are per 10 CFR 50.67 and RG 1.183 guidance.

Table 4-2 lists the regulatory limits for accidental dose due to a postulated Fuel Handling Accident to: 1) a control room operator, 2) a person at the EAB, and 3) a person at the LPZ boundary.

Table 4-2
Regulatory Dose Limits (Rem TEDE)

CR (30 days)	EAB (2 hours)	LPZ (30 days)
5	6.3	6.3

4.9 Design Inputs

The design inputs used for this evaluation are based on Oyster Creek licensing basis documents, existing calculations, and regulatory guidance documents. These parameters are summarized in the Table 4-3 below:

Table 4-3
Parameters Applicable to AST Fuel Handling Accident Dose Considerations for Oyster Creek Nuclear Generating Station

Parameter or Method	AST Value	Comments
Reactor Power	1969 MWth	Includes 2% margin for instrument uncertainty relative to the rated thermal power of 1930 MWt.
Fuel Assembly Configuration and properties	9 x 9 with 140 rods damaged	Bounding assumptions for current Oyster Creek licensing basis FHA. Movement of irradiated fuel without secondary containment will not occur less than 24 hours after the associated reactor shutdown.
Radial Peaking Factor	1.7	Conservative bounding assumption
Allowable Fuel Burnup and non-LOCA gap fractions	RG 1.183, Table 3. Fuel bundle peak burnup will not exceed 62 GWD/MTU. For fuel exceeding 54 GWD/MTU, the maximum linear heat generation rate will not exceed 6.3 kW/ft.	The GE 11 fuel burnup is 27.6 GWD/MTU. Therefore, the 6.3 kW/ft linear heat generation rate limit does not apply.

Parameter or Method	AST Value	Comments
FHA Radionuclide Inventory	From the 60 isotopes forming the RADTRAD library used in the LOCA analysis, with decay to 24 hours. Gap activities are per R.G. 1.183.	Spent fuel source terms are based on the same bounding reactor core source terms as was used for the LOCA analysis.
Underwater Decontamination Factor	Noble Gases: 1 Particulate (cesiums and rubidiums): infinity Iodine: 200, corresponding to a 23-ft water depth for an assembly drop into the reactor vessel	For conservatism, the effective minimum depth of 23 feet is assumed to be the water coverage over the reactor core. This is the worst-case location for a fuel drop FHA to take place (significantly more damage is produced than a drop in the fuel pool).
Iodine chemical distribution	From RG 1.183	95% CsI, instantaneously dissociating in the pool water and re-evolving as elemental iodine. Since the pH of the pool water is not maintained above 7, iodine is assumed to be 97% elemental and 3% organic in the air space above the pool.
Activity Transport to the Environment	Activity reaching the refuel floor airspace will essentially be all exhausted within 2 hours by using an artificially high exhaust rate.	This also provides an allowance for uneven mixing in the refuel floor airspace.
Release Pathways	The release pathways are described in Table 4-1.	No credit is taken for filtration by the SGTS, or the elevated release resulting from exhaust through the Oyster Creek Main Stack.
Dose Conversion Factors	EPA Federal Guidance Reports 11 and 12	
Offsite Dose Limit	6.3 rem TEDE	After 2 hours per 10CFR50.67 and RG 1.183
Control Room Dose Limit	5 rem TEDE for the duration of the accident	Per 10CFR50 App. A, GDC 19 and 10CFR50.67
CR Volume CR Intake Rate	Volume 27,500 ft^3 14,000 cfm	As previously demonstrated for the LOCA analysis, the CR dose is maximized when an intake rate of 4,000 cfm is achieved. However, the maximum intake of 14,000 cfm in the purge mode is used for conservatism. This also eliminates the need to address the unfiltered inleakage rate, since additional intake produces no additional dose.

Parameter or Method	AST Value	Comments
Refuel Floor Ventilation Rate and Volume	For this analysis, an artificial volume of 100 ft^3 with an artificially high exhaust rate is assumed for simplicity.	This evacuates 99.9999% of all activity within 2-hours.
CR Potential Release Points	Limiting X/Qs (0 – 2 hr)	The most limiting 0-2 hr potential release point was determined to be the MAC facility entrance.
Drywell Access Facility (West Wall)	1.61E-03 sec/m^3	
Drywell Access facility (South Wall)	1.93E-03 sec/m^3	
Commodities Penetration on the RB South Wall	1.77E-03 sec/m^3	
Commodities Penetration on the RB North Wall	5.21E-03 sec/m^3	
MAC Facility Personnel Airlock	6.75E-03 sec/m^3	The most limiting 0-2 hr potential release point was determined to be the MAC Facility Personnel Airlock.
MAC Facility Entrance	6.62E-03 sec/m^3	
RB Roof Hatch	1.82E-03 sec/m^3	
Stack Tunnel Door	8.55E-04 sec/m^3	
East Airlock Door	1.40E-03 sec/m^3	
Reactor Building Wall (Diffuse Area)	2.15E-03 sec/m^3	
Trunion Room Door to Turbine Building	3.73E-03 sec/m^3	Previously approved for MSIV leakage in TS Amendment No. 262, dated April 26, 2007 (Reference 7.3).
EAB Release Point Basis and Distance to EAB	Normal RB exhaust stack and 414 m (considered as applicable to all release locations)	All releases considered as a ground level release. Previously approved for TS Amendment No. 262, dated April 26, 2007 (Reference 7.3).
Limiting Dispersion Factors (0 – 2 hr)	1.41E-03 sec/m^3	
LPZ Release Point Basis and Distance to LPZ	Normal RB exhaust stack and 3218 m (considered as applicable to all release locations)	All releases considered as a ground level release. Previously approved for TS Amendment No. 262, dated April 26, 2007 (Reference 7.3).
Limiting Dispersion Factors (0 – 2 hr)	1.35E-04 sec/m^3	

4.10 Summary of Results - FHA

The RADTRAD code was used to examine the effects of the alternative source term release on offsite and CR doses. Table 4-4 below shows the results as calculated in Attachment 1. Calculated doses are with a decay time of 24 hours.

Table 4-4
Results (Rem TEDE)

Location	Dose (Rem TEDE)
LIMITS	CR = 5.0; EAB & LPZ = 6.3
EAB	0.741
LPZ	0.071
CR	1.45

4.11 Conclusions

For postulated releases through Reactor Building openings described in Table 4-1, movement of irradiated fuel that has NOT occupied part of a critical reactor core in the previous 24 hours can safely be accomplished with doses within regulatory limits without secondary containment integrity or SGTS operability.

5.0 Regulatory Analysis

5.1 No Significant Hazards Consideration

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

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

   Response: No.

   The proposed change is related to a postulated fuel handling accident inside the Reactor Building occurring during fuel loading and refueling activities. The proposed change does not involve a change to structures, components, or systems that would affect the probability of an accident previously evaluated in the Oyster Creek Updated Final Safety Analysis Report. (UFSAR). Oyster Creek Alternative Source Term (AST) methodology has been previously reviewed and approved by the NRC. AST is used to evaluate the dose consequences of the postulated fuel handling accident. The postulated fuel handling accident has been analyzed without credit for Secondary Containment integrity and Standby Gas Treatment system operation. The resultant radiological consequences are within the acceptance criteria set forth in 10 CFR 50.67 and RG 1.183. Therefore, the proposed changes do not significantly increase the consequences of an accident previously evaluated.

This amendment does not alter the methodology or equipment used directly in fuel handling operations. The Secondary Containment structure and the Standby Gas Treatment system, and any component thereof, are not accident initiators. Actual fuel handling operations are not affected by the proposed changes. Therefore, the probability of a fuel handling accident is not affected with the proposed amendment. No other accident initiator is affected by the proposed changes.

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

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

Response: No.

The proposed amendment will not create the possibility for a new or different type of accident from any accident previously evaluated. Equipment important to safety will continue to operate as designed. Component integrity is not challenged. The changes do not result in any event previously deemed incredible being made credible. The changes do not result in more adverse conditions or result in any increase in the challenges to safety systems. The systems affected by the changes are used to mitigate the consequences of an accident that has already occurred. The proposed Technical Specification changes do not allow reduction of the mitigative function of these systems.

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

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

Response: No.

Safety margins and analytical conservatisms have been evaluated and have been found acceptable. The analyzed event has been carefully selected and margin has been retained to ensure that the analysis adequately bounds the postulated event scenario. The dose consequences due to the postulated event comply with the requirements of 10 CFR 50.67 and the guidance of RG 1.183.

The proposed amendment is associated with the implementation of a new licensing basis for the Oyster Creek Fuel Handling Accident. The change from the original source term to a new source term taken from RG 1.183 has been previously approved by the NRC for Oyster Creek. The results of the accident analysis, revised in support of the proposed license amendment, are subject to revised acceptance criteria. The analysis has been performed using conservative methodologies, as specified in RG 1.183. Safety margins have been evaluated and analytical conservatism has been utilized to ensure that the analysis adequately bounds the postulated limiting event scenario. The dose consequences of this design basis accident remain within the acceptance criteria presented in 10 CFR 50.67, "Accident source term", and RG 1.183.

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

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

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

5.2 Applicable Regulatory Requirements/Criteria

The fuel handling accident dose analysis has been performed using AST and TEDE dose criteria in accordance with the guidance provided in RG 1.183. AST methodology has been previously reviewed and approved by NRC for use at Oyster Creek. The revised fuel handling accident analysis demonstrates that the radiological dose consequences remain within the requirements of 10 CFR 50.67 and GDC 19.

AmerGen has determined that the proposed changes do not require any exemptions or relief from regulatory requirements and does not affect conformance with any General Design Criteria.

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

5.3 Precedent

The proposed amendment is consistent with the approach used for relaxation of secondary containment operability requirements by Hope Creek Generating Station in an application dated June 28, 2002, and approved by NRC in Amendment no. 146, dated April 15, 2003.

6.0 Environmental Consideration

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 References

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

7.2 NEDE–24011–P–A, "General Electric Standard Application for Reactor Fuel (GESTAR II)," and NEDE–24011–P–A–US, and "General Electric Standard Application for Reactor Fuel (Supplement for United States)," the latest revision being applicable.

7.3 Oyster Creek Technical Specification Amendment No. 262, dated April 26, 2007.

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

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

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