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10 CFR 50.90

RA-08-045

May 5, 2008

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: Response to Draft Request for Additional Information - AmerGen Application to Revise Technical Specifications Regarding Secondary Containment Operability Requirements During Refueling

Reference: AmerGen Letter to USNRC, "Technical Specification Change Request 338 - Secondary Containment Operability Requirements During Refueling," dated November 2, 2007

This letter provides additional information in response to an NRC request for additional information (RAI) received via NRC facsimile, dated March 25, 2008, regarding Oyster Creek Nuclear Generating Station (Oyster Creek) Technical Specification (TS) Change Request 338 – "Secondary Containment Operability Requirements During Refueling," submitted to the NRC for review on November 2, 2007 (Reference). The additional information is provided in the Enclosure.

AmerGen Energy Company, LLC (AmerGen) has reviewed the information supporting a finding of no significant hazards consideration that was previously provided to the NRC in the referenced document. The additional information provided in this submittal does not affect the bases for concluding that the proposed license amendment does not involve a significant hazards consideration.

No new regulatory commitments are established by this submittal. If any additional information is needed, please contact Mr. David Robillard at (610) 765-5952.

A001
NRR

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 5th day of May, 2008.

Sincerely,

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

Enclosure: Response to Draft Request for Additional Information

cc: S. J. Collins, USNRC Administrator, 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

ENCLOSURE

OYSTER CREEK NUCLEAR GENERATING STATION

RESPONSE TO DRAFT REQUEST FOR ADDITIONAL INFORMATION

TECHNICAL SPECIFICATION CHANGE REQUEST No. 338
"SECONDARY CONTAINMENT OPERABILITY REQUIREMENTS DURING REFUELING"

ENCLOSURE

OYSTER CREEK NUCLEAR GENERATING STATION

RESPONSE TO DRAFT REQUEST FOR ADDITIONAL INFORMATION TECHNICAL SPECIFICATION CHANGE REQUEST No. 338 "SECONDARY CONTAINMENT OPERABILITY REQUIREMENTS DURING REFUELING"

By letter dated November 2, 2007, AmerGen Energy Company, LLC (AmerGen) submitted an amendment request for the Oyster Creek Nuclear Generating Station (Oyster Creek). The proposed amendment would revise the Oyster Creek Technical Specifications regarding secondary containment operability requirements during refueling.

The Nuclear Regulatory Commission staff has reviewed the information provided in support of the proposed amendment and finds that the following information is required to complete its review:

INTEGRITY OF FACILITY DESIGN BASIS

1. **NRC Question**

   For the Oyster Creek revised Fuel Handling Design Basis Accident (DBA) source term analysis done in accordance with 10 CFR 50.67, Alternative Source Term (AST) methodology, provide the current licensing basis (CLB) parameters along with the revised values where applicable, as listed in Table 4-3 of the November 2, 2007 submittal. In addition, provide the basis for any changes to the CLB parameters as a result of the proposed AST application for the fuel handling accident (FHA). The NRC staff requests that the licensee include CLB information and the revised parameters in a table whether or not the individual parameter changed for this amendment request (See regulatory positions 1.3.2 and 1.3.4 of Regulatory Guide (RG) 1.183 and NRC Regulatory Issue Summary (RIS) 2006-04).

   Response

   The AST parameters of Table 4-3 of the November 2, 2007 submittal are shown below with additional columns for the current licensing basis values and the basis for the changes from the current licensing basis to AST values provided in the November 2, 2007 submittal.

Parameter or Method	AST Value	Pre-AST and Current Licensing Basis	Justification and Comments
Reactor Power	1969 MWth	1930 MWth nominal 1969 MWth Appendix K	This value is unchanged and includes 2% margin for instrument uncertainty relative to the rated thermal power of 1930 MWth.
Fuel Assembly Configuration and properties	9 x 9 with 140 rods damaged	8 x 8 with 124 rods damaged	The 9x9 array is the bounding assumption for current Oyster Creek licensing basis FHA. (Taken from GESTAR) There are 22 of the 8 x 8 fuel assemblies remaining in the reactor core (to be replaced during the next refuel outage).
Radial Peaking Factor	1.7	1.5	Conservative bounding assumption. RG 1.183 justifies this change.
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.	N/A	Some peak rod-average burnups can exceed 54 GWD/MT. However, the 6.3 kW/ft linear heat generation rate limit will not be exceeded. This data is reviewed for each cycle.
FHA Radionuclide Inventory	From the 60 isotopes forming the RADTRAD library used in the Loss of Coolant Accident (LOCA) analysis, with decay to 24 hours. Gap activities are per RG 1.183.	TID-14844	Spent fuel source terms are based on the same bounding reactor core source terms as was used for the approved AST LOCA analysis. The Radionuclide inventory from the previous analysis was based on TID 14844 assumptions.
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	Noble Gases: 1 Particulates: infinity Iodine: 100	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 shorter drop in the fuel pool).
Iodine chemical distribution	The chemical form of radioiodine released from the fuel to the spent fuel pool is assumed to be 95% CsI, 4.85% elemental iodine, and 0.15% organic iodide. 95% CsI, instantaneously dissociates in the pool water. All iodine is re-evolved as elemental iodine.	TID-14844	From RG 1.183. 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. The design basis FHA calculation uses only gaseous iodine (no particulate).
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.	Elevated release via filtered SGTS to Main Stack	All (99.9999%) activity is exhausted to the environment within 2 hours per RG 1.183. This also provides an allowance for uneven mixing in the refuel floor airspace.
Release Pathways	Activity reaching the refuel floor airspace will essentially be all exhausted within 2 hours by using an artificially high exhaust rate. The release pathways are	Filtration by SGTS with elevated release through the Main Stack.	For conservatism, no credit is taken for filtration by the SGTS, or the elevated release resulting from exhaust through the Oyster Creek Main Stack.

Parameter or Method	AST Value	Pre-AST and Current Licensing Basis	Justification and Comments
	described in Table 4-1.		
Dose Conversion Factors	EPA Federal Guidance Reports 11 and 12	ICRP60 Dose conversion Factors, RG 1.109	As recommended in RG 1.183 and implemented in RADTRAD.
Offsite Dose Limit	6.3 rem TEDE	25 rem whole body 300 rem thyroid	After 2 hours per 10 CFR50.67 and RG 1.183
Control Room Dose Limit	5 rem TEDE for the duration of the accident	5 Rem Whole Body, or its equivalent to any part.	Per 10 CFR50 App. A, GDC 19 and 10 CFR50.67
CR Volume CR Intake Rate	Volume 27,500 ft^3 14,000 cfm	Volume 27,500 ft^3 14,000 cfm	For the AST values, as previously demonstrated for the approved LOCA analysis, the CR dose is maximized when an intake rate of 4,000 cfm is achieved (parametric evaluation previously performed). 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.
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.	Exhaust flow via SGTS	This new AST assumption evacuates 99.9999% of all activity within 2 hours.
CR Potential Release Points Drywell Access Facility (West Wall) Drywell Access facility (South Wall) Commodities Penetration on the RB South Wall Commodities Penetration on the RB North Wall MAC Facility Personnel Airlock MAC Facility Entrance RB Roof Hatch Stack Tunnel Door East Airlock Door Reactor Building Wall (Diffuse Area) Trunion Room Door to Turbine Building	Limiting X/Qs (0 – 2 hr) 1.61E-03 sec/m^3 1.93E-03 sec/m^3 1.77E-03 sec/m^3 5.21E-03 sec/m^3 6.75E-03 sec/m^3 6.62E-03 sec/m^3 1.82E-03 sec/m^3 8.55E-04 sec/m^3 1.40E-03 sec/m^3 2.15E-03 sec/m^3 3.73E-03 sec/m^3	Limiting X/Qs with Murphy Campe method (FHA) are inappropriate for AST.	The most limiting 0-2 hr potential release point was determined to be the MAC Facility Personnel Airlock Pre-AST values used Murphy-Campe methodology. New AST values utilize ARCON96 assumptions per RG 1.194.
EAB Release Point Basis and Distance to EAB Limiting Dispersion Factors (0 – 2 hr)	Normal RB exhaust stack and 414 m (considered as applicable to all release locations) 1.41E-03 sec/m^3	Main Stack 414 m	AST FHA releases are considered as ground level releases. Previously approved for TS Amendment No. 262, dated April 26, 2007 (Reference 7.3).

Parameter or Method	AST Value	Pre-AST and Current Licensing Basis	Justification and Comments
LPZ Release Point Basis and Distance to LPZ Limiting Dispersion Factors (0 – 2 hr)	Normal RB exhaust stack and 3218 m (considered as applicable to all release locations) 1.35E-04 sec/m^3	Main Stack 3218 m Normal RB exhaust stack and 3218 m (considered as applicable to all release locations) 1.35E-04 sec/m^3	AST FHA releases are considered as ground level releases. Previously approved for TS Amendment No. 262, dated April 26, 2007 (Reference 7.3).

Note: References cited in the Table are from the November 2, 2007 submittal.

ACCIDENT SOURCE TERM

2. **NRC Question**

In the FHA AST assumptions AmerGen stated that the General Electric (GE)-11 fuel burnup is 27.6 giga-watt-days (GWD)/Metric Ton Uranium (MTU) so that the 6.3 kilo-watt (kW) / foot (ft) restriction outlined in footnote 11 of RG 1.183 does not apply. Please clarify that the GE 11 9X9 fuel, in the course of its projected power history for any specific fuel load, will not exceed the value of 54 GWD/MTU (See Regulatory Position 3.2 and footnote 11 of RG 1.183).

Response

Since peak rod-average burnups can exceed 54 GWD/MTU, AmerGen has incorporated alternative source term limits in its procedures and processes. Procedure NF-AB-110-2210, Core Loading Pattern Development, requires that the core loading be developed in accordance with NRC Regulatory Guide 1.183. There is a procedural step to ensure that 1) peak rod-average burnup of the fuel is less than 62 GWD/MTU, 2) that the maximum linear heat generation rate does not exceed 6.3 kilo-watt/ft for any bundle exceeding a peak rod burnup of 54 GWD/MTU, 3) that the cycle length not exceed 711 EFPD and 4) that the maximum radial peaking factor does not exceed 1.7.

3. **NRC Question**

Given the information above to be correct, does Oyster Creek have other fuel types that would exceed 54 GWD/MTU and be more limiting for a FHA at Oyster Creek as far as the maximum fuel inventory at the end of fuel life (See Regulatory Position 3.2 and footnote 11 of RG 1.183).

Response

AmerGen has processes in place that require checks to be performed on peak rod-average exposures for all fuel types. The Oyster Creek core design engineer utilizes the 3D simulator code, PANACEA, to confirm compliance with NRC Regulatory Guide 1.183. For any fuel rod that exceeds 54 GWD/MTU rod average exposure, the kW/ft value is checked to confirm that the linear hear generation rate is less than 6.3 kW/ft. For Oyster Creek's current operating cycle (Cycle 21), maximum rod average exposure at end-of-cycle is 55.1

GWd/MTU. The maximum linear heat generation rate for fuel rods projected to exceed 54 GWd/MTU is 5.1 kW/ft.

4. **NRC Question**

Please clarify the term "Reactor Well" discussed in your submittal in Section 4.3 under the heading "Decontamination Factor."

Response

The term reactor well refers to the reactor cavity, which includes the reactor vessel and the volume of water that connects the reactor vessel to the spent fuel pool during refueling operations.

5. **NRC Question**

In Section 4.3 of the submittal, AmerGen determined that "A drop over the reactor well is more limiting than accidents in the spent fuel pool (SFP)." Provide the reference or detail that shows the drop in the well or as described in the Oyster Creek final safety analysis report (FSAR), "a drop onto the reactor core from the maximum height allowed by the refueling equipment," as the limiting accident (See Regulatory Position 2.4 of RG 1.183).

Response

The Oyster Creek UFSAR Section 15.7.4.4 describes the number of fuel rods that are damaged during a design basis fuel handling accident. Dropping a fuel assembly onto the reactor core from the maximum height allowed by the refueling equipment (less than 30 feet) results in an impact velocity of 40 ft/sec. The kinetic energy acquired by the falling assembly is less than 17,000 ft-lb and is dissipated in one or more impacts. The first impact is expected to dissipate most (80%) of the energy and cause the largest number of cladding failures. This first impact is calculated to cause 105 fuel rods to fail. The second impact is calculated to cause 19 rods to fail. No rods are expected to fail during the third impact.

For a fuel assembly drop onto the spent fuel storage racks within the spent fuel pool, the number of cladding failures due to the initial impact is expected to be much less than the 105 calculated for the 30-foot drop. This is due to the fact that the kinetic energy due to a drop of 4 feet (a conservative height based on plant procedure restrictions of 2 feet) is less than that for a drop of about 30 feet.

Using the standard kinematics equation,

$$s = v^2/2a$$

Where
s = distance
v = velocity
a = acceleration due to gravity (32 ft/sec^2 in air, for conservatism),

Solving for v;

$$v = \text{sqrt}(2as) = \text{sqrt}(2 * 32 \text{ ft/sec}^2 * 4\text{ft}) = 16 \text{ ft/sec}$$

This is the velocity achieved for a 4-foot drop (16 ft/sec). Since the velocity at the bottom of the fall in the spent fuel pool is 40% of that for a drop in the vessel, the number of rods damaged in the first impact would be 105 * 0.4 = 42. Since the second and third impacts are when the assembly falls from the vertical (to the horizontal position), this damage amount (19 fuel rods) remains unchanged. Therefore, the total fuel rods damaged would be 42 + 19 = 61. This amounts to 49.2% of the fuel rods damaged during a drop in the fuel pool.

6. **NRC Question**

Appendix B of RG 1.183 allows an overall decontamination factor of 200 if the depth of water above damaged fuel is 23 feet or greater. If the depth of the water is less than 23 feet, the decontamination factor will have to be determined on a case-by-case method (Ref. B-1) of RG 1.183. Provide either a detailed analysis that proves the statement in Section 4.3 that a shorter drop in the fuel pool would result in less radiological release to the containment or conservatively adjust the decontamination factor used for the FHA to account for water depth less than 23 feet (See Regulatory Position 2.4 of RG 1.183 and Regulatory Position 2 of Appendix B of RG 1.183).

Response

When a fuel handling accident occurs in the spent fuel pool, it is assumed that the dropped bundle will rest on the tops of the bail handles of the fuel in the storage racks. This puts the highest point of the (now horizontal) dropped assembly at 22.27 feet below the surface of the water when the fuel pool is at the minimum allowed water level.

Using the equations from the Reference B-1 of Regulatory guide 1.183, Burley's "Evaluation of Fission Product Release and Transport", the DF for 22.5 feet water coverage is determined to be 183.2 compared to the value of 200 determined for 23 feet. This is an 8.4% decrease in overall assumed DF. This slight decrease in DF is compared to the approximately 50% increase in damaged fuel rods assumed for the drop over the reactor vessel compared to the lesser fuel damage that would occur for a drop over the spent fuel pool. For conservatism, the accident with the most fuel damage produced was assumed (i.e., the drop over the reactor well, approximately 30 feet) while simultaneously limiting the decontamination factor assumed for a 23-foot drop to 200, and is therefore bounding for either possible accident.

ANALYSIS ASSUMPTIONS AND METHODOLOGY

7. **NRC Question**

In Table 4-3 of the submittal, the Exclusion Area Boundary (EAB) and the Low Population Zone (LPZ) limiting dispersion factors are listed as "Normal RB exhaust stack" with limiting atmospheric dispersion factor (X/Q values) of 1.41E-03 second per cubic meter (sec/m^3) for the 414 m EAB and 1.35E-4 sec/m^3 for the 3218 m LPZ. The reference to stack release and the dispersion values are in conflict with a worst-case ground level release as well as the values listed in Attachment 1 (Calculation C-1302-822-E310-082) of the submittal Table 6.1.3 which shows ground level release EAB X/Q value of 1.10E-3 sec/m^3 and LPZ X/Q value of 5.60E-5 sec/m^3. Please clarify the above apparent discrepancies in your submittal.

Response

The EAB and LPZ values printed in Table 6.1.3 of Calculation C-1302-822-E310-082 are incorrect and not used in the analysis. The values of 1.41E-03 sec/m^3 for the 414 m EAB and 1.35E-4 sec/m^3 for the 3218 m LPZ are used correctly in the RADTRAD runs (see Page B-4 of the calculation) to calculate the overall dose. This editorial discrepancy did not impact any of the conclusions reached in the referenced submittal, and has been entered into the corrective action program.

METEOROLOGY ASSUMPTIONS

8. **NRC Question**

In Table 4-3, on page 15 of Enclosure 1 to the submittal dated November 02, 2007, the licensee states:

> "*The most limiting 0-2 hr potential release point was determined to be the MAC facility entrance.*"

However, the table later lists the most limiting 0-2 hr (i.e., highest X/Q value) potential release point for the FHA was the Monitor and Control (MAC) Facility Personnel Airlock source location with a value of 6.75E-03 sec/m^3. The 0-2 hr X/Q value for the MAC Facility Entrance source location is 6.62E-03 sec/m^3. Please clarify this statement and confirm which 0-2 hr value was used in calculating the control room dose estimate for the Oyster Creek Nuclear Generating Station.

Response

The intent was to indicate the limiting X/Q in Table 4-3 as the MAC Facility Personnel Airlock. However, the first such indication incorrectly stated that it was for the MAC facility entrance instead of the MAC Facility Personnel Airlock (as stated on the line where the X/Q value is listed). The correct X/Q values were used in the RADTRAD runs in the calculation (see Page B-4 of the calculation). This editorial discrepancy did not impact any of the conclusions reached in the referenced submittal, and has been entered into the corrective action program.

9. **NRC Question**

Two control room Heating, Ventilating, and Air-Conditioning System intakes (Intake A and Intake B) were used as the respective receptor points for each source location for the onsite X/Q analyses. ARCON96 was used to perform these calculations of estimated atmospheric dispersion values. The guidance for ARCON96, RG 1.194, notes that certain considerations should be evaluated in identifying the control room outside air intakes for which X/Q values should be considered.

Please clarify if the two intakes to the control room used to model the onsite X/Q values are: (1) in the same wind direction window or not and (2) share an imbalance of the intake flow rate to the control room. Based on these clarifications, please identify which ARCON96 X/Q equation was used for each source/receptor pair analyzed in this submittal and confirm that this was the appropriate methodology for the relative calculation. For example, if the Oyster Creek control room has dual intakes (e.g., Intake A and Intake B) with equal intake flow rates and share a wind direction window, then equation (5a) is deemed appropriate for this particular onsite X/Q analysis.

Response

(1) Each CR intake was analyzed individually for each source-receptor pair. The intake with the highest X/Q for each pair was used in the dose analysis. Therefore, there is no need to address wind direction window issues. Only one intake is used at a time. The accident analysis assumes it is the one with the highest X/Q.

(2) The ARCON96 point source calculation, which maximizes the X/Q, was used. There is no intake imbalance since we assume that the intake with the highest individual X/Q is used in the calculation.

10. **NRC Question**

ARCON96 was used to compute the onsite X/Q values, which is supported by the guidance of RG 1.194. On page 14 of this regulatory guide (i.e., 1.194-14), it states that:

"If the distance to the receptor is less than about 10 meters, the ARCON96 code and the procedures in Regulatory Position 4 should not be used to assess X/Q values."

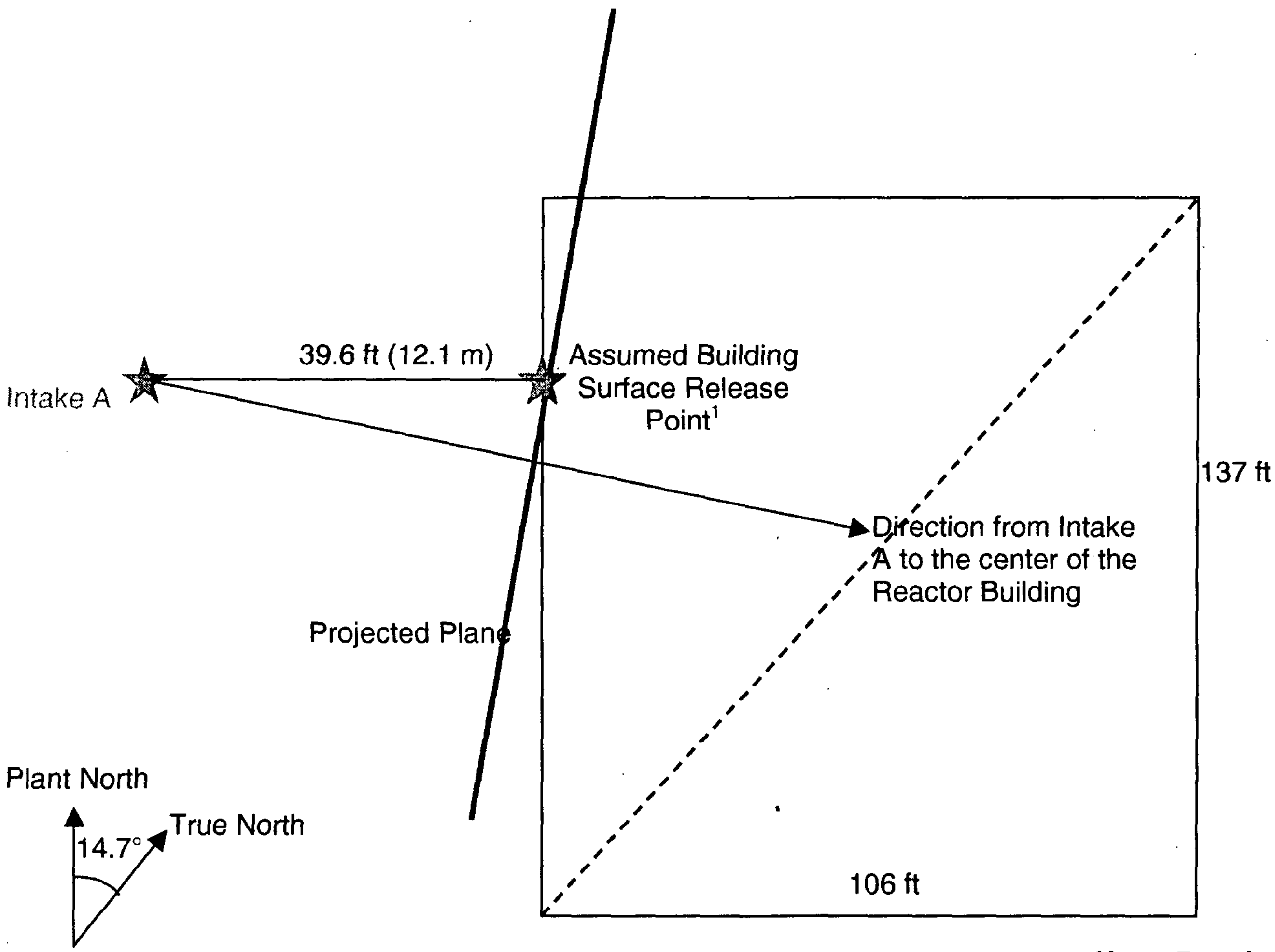
Considering this statement, please justify the use of the ARCON96 computer code to estimate the X/Q value for the Reactor Building Wall source location to the Control Room Intake A receptor point. The horizontal distance for this particular source/receptor pair is noted as 7.9 meters (25.8 feet), which is considerably less than 10 meters (as specified appropriate for use of the ARCON96 computer code in RG 1.194). Please indicate what impact this may have on the resulting X/Q values if used inappropriately.

Response

The subject diffuse area source-to-receptor distance of 7.9 meters that was assumed in Calculation C-1302-822-E310-081 is not the closest distance from Intake A to the actual physical Reactor Building wall, but instead to the much nearer vertical planar surface projected perpendicular to the line of sight from Intake A to the center of the Reactor Building. This plane was positioned to intersect the Reactor Building at the point that minimizes the plane's horizontal distance from Intake A (as shown in Attachment G, page 5 of 6 of the Calculation). This distance is less than 10 meters, the approximate source-to-receptor distance less than which Regulatory Guide 1.194 indicates that ARCON96 may not be appropriate. However, if the X/Q for this scenario were to be recalculated by ARCON96 with a horizontal distance of 12.1 m (i.e., the distance from the closest point on the actual Reactor Building surface to Intake A, as depicted in the sketch below) in accordance with the diffuse area source method described in Section 3.2.4.5 of R.G 1.194, the resulting X/Q values would be smaller.

Oyster Creek RAI #10 Assumed Diffuse Area Building Surface Release Point

Scenario: Reactor Building Wall to Intake A



Note: Drawing not to scale.

<u>Direction from Intake to Assumed Building Surface Release Point:</u>
90° from Plant North
90° - 14.7° = 75.3° from True North

[1] Per RG 1.194, Section 3.2.4.5, the release point is defined as the closest point on the building surface to the intake.

11. **NRC Question**

Numerous assumptions were made relative to the building wake area and distance calculations for source/receptor pairs used in the assessment of control room X/Q values at the Oyster Creek Nuclear Generating Station. In order to justify these assumptions, a more graphically detailed schematic is needed than the plant layout provided in Attachment A to the amendment request.

Please provide a legible and suitably scaled schematic of the Oyster Creek Nuclear Generating Station site area showing true north. The drawing should indicate site boundaries and the location and orientation of principal plant structures within the site area representative of those used in the FHA most limiting X/Q analyses as presented in Table 4-3 of this submittal. Highlight postulated release and receptor points or locations and include relative straight-line or taut string distances, as deemed appropriate, on the drawing.

Response

Attachment E of the previously provided calculation C-1302-822-E310-081 contains all pertinent details with sufficient clarity with respect to all release points and receptors needed for ARCON96 input.

ADDITIONAL INFORMATION

12. **NRC Question**

Provide the Oyster Creek proposed FSAR Markup, as suggested by Regulatory Position 1.5 of RG 1.183, or revised updated final safety analysis report pages outlining the Oyster Creek revised AST licensing basis for the FHA. Regulatory Position 1.6 of RG 1.183 outlines the FSAR update requirements including a reference to 10 CFR 50.71.

Response

NRC RG 1.183 recommends submitting the affected FSAR pages annotated with changes that reflect the revised analysis or submitting the actual calculation. To meet this recommendation, AmerGen provided the actual calculation documentation with the original license amendment request submittal (Reference). The affected FSAR pages are typically revised as part of the AmerGen process for implementing the NRC approved TS amendment, and as such, have not yet been developed.