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10 CFR 50.90
10 CFR 50.67

January 24, 2007
2130-07-20448

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

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

Subject: Response To Request For Additional Information – License Amendment Request No. 315, "Application of Alternative Source Term" (TAC No. MC6519)

This letter provides additional information in response to NRC request for additional information (RAI), dated December 19, 2006, regarding Oyster Creek License Amendment Request No. 315, submitted to NRC for review on March 28, 2005. The additional information is provided in Enclosure 1. Response to the NRC RAI Question Numbers 2.c, 3, and 5 will be provided by separate submittal, as noted in the RAI dated December 19, 2006.

The NRC request for additional information, dated December 19, 2006, requested the enclosed response by January 18, 2007. The extension of the due date to January 24, 2007 for this response was discussed with the NRC staff on January 17, 2007 and was found to be acceptable.

Enclosure 2 provides a summary of the regulatory commitments made in this submittal. If any additional information is needed, please contact David J. Distel at (610) 765-5517.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 24 day of January, 2007.

Respectfully,

DJD Pamela B. Cowan

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

Enclosures: 1) Response to Request for Additional Information
2) Regulatory Commitments
3) CD – Oyster Creek Meteorological Data 1995, 1996, 1997, 1998, 1999 (ASCII Format)

A001

cc: S. J. Collins, USNRC Administrator, Region I
G. E. Miller, USNRC Project Manager, Oyster Creek
M. S. Ferdas, USNRC Senior Resident Inspector, Oyster Creek
File No. 03079

ENCLOSURE 1

OYSTER CREEK

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

LICENSE AMENDMENT REQUEST No. 315

APPLICATION OF ALTERNATIVE SOURCE TERM

1. Justify the use of MAAP4 for the containment accident thermal-hydraulics. Describe the phenomena occurring in containment as the accident progresses and show that MAAP4 can adequately model these phenomena in terms of any benchmarking to data or other computer codes for each phenomena. Since MAAP4 is being used for design basis calculations, show that the MAAP4 calculations bound the expected response.

Response

MAAP4 is able to generate the primary containment thermal-hydraulic (T-H) response to an accident involving degraded emergency core cooling and consequent damage to the core. Such a thermal-hydraulic response would be expected for the release of activity specified in Regulatory Guide (RG) 1.183 for a Loss of Coolant Accident (LOCA).

In the course of its development for more than twenty years, MAAP4 has been benchmarked extensively against separate-effects experiments, integrated experiments, and other computer models. A specific example of the latter (a comparison of computer models that is directly relevant to the application of MAAP4 T-H to LOCA dose analysis) is the parallel MAAP4 and MELCOR analyses for AP1000 that is discussed further below.

The core damage thermal-hydraulic response is one in which little or no decay heat is transferred from the core to the containment following blowdown. Heat transfer is minimized because the reactor core is not being cooled. If core cooling were successful, core damage would not result and the activity release to the containment would be minimal. Therefore, the release to the containment of large fractions of the core inventory of volatile fission products (e.g., 100% of the core inventory of fission product noble gas) suggests that the decay heat, at least up to the end of the release, is being largely retained within the reactor vessel. This condition is assumed to persist until the assumed restoration of core cooling at the end of the release phase.

The consequence of limited heat transfer is a drywell pressure that is sufficiently low in the presence of drywell sprays to force the sprays to be periodically secured. Once secured, the drywell pressure increases, the spray re-initiation pressure is reached, and the sprays are restarted. This periodic cessation of drywell sprays during the release phase has a negative effect on activity removal and is therefore conservative. When the core decay power is being transferred to the containment (as in LOCA models prepared for the purpose of containment analysis), this intermittent spray behavior is not observed; rather, a quasi-steady state condition is reached in which the steam partial pressure in the drywell corresponds to the temperature increase in the combined core cooling and spray flow necessary to remove the core decay power which changes only slowly with time. Therefore, using containment analysis T-H conditions as boundary conditions for the Oyster Creek spray removal analysis would be both (1) inconsistent with expected T-H conditions for a core damage event and (2) nonconservative.

MAAP4 has been used in other LOCA dose assessments in which it has been recognized that containment T-H conditions play an important role. In the Design Certification of AP600, a similar potential for nonconservative results existed using containment analysis T-H conditions. The T-H conditions play an important role in phoretic removal (natural aerosol removal processes associated with heat transfer); and early in the assessment of phoretic deposition for AP600, it was determined that using containment analysis T-H

conditions (in which condensation rates were high because steam the generation rates were high) exaggerated the rate of activity removal.

In the Design Certification of both AP600 and AP1000, MAAP4 core damage accident T-H conditions were used so that activity removal associated with steam generation and condensation rates would not be overstated. This is reflected in the following statements from the Final Safety Evaluation Report (Chapter 15) for AP1000:

> *In the past, the staff and industry evaluated aerosol removal through well-established models of spray removal or condensation. The AP1000 application relies on natural deposition processes that depend strongly on local T-H conditions. While gravitational settling is relatively easy to understand, aerosol removal through diffusiophoresis and thermophoresis is much more complex. Diffusiophoresis is associated with steam condensation on the heat sinks and depends on the condensation steam mass flux. Thermophoresis relies only on the temperature gradient close to the surface on which the particles would be deposited. Thermophoresis is more subtle than the other two natural deposition processes. Because the temperature gradient cannot be measured or easily calculated, its model uses the heat flux at the surface divided by the thermal conductivity of the gas adjacent to the surface as an equivalent measure of the driving force. Simultaneous occurrence of the two phoretic processes introduces an additional level of complexity.*
>
> *The Westinghouse methodology includes industry's Modular Accident Analysis Program (MAAP) code, an integrated accident analysis program, to establish T-H boundary conditions as an input to an aerosol code (STARNAUA). To determine the acceptability of the Westinghouse modeling, the staff audited Westinghouse calculations of the containment removal coefficients. The audit revealed that the heat flux used by Westinghouse included the convection, the thermal radiation, and the decay heat from airborne fission products. Thermal radiation and decay heat do not contribute to the temperature gradient that drives thermophoresis and their use caused the overall aerosol removal to be nonrealistic and nonconservative. Westinghouse recalculated the overall aerosol removal coefficients by correcting this error.*
>
> *In its independent evaluation of aerosol removal coefficients, the staff considered the same natural processes for removing aerosols from the containment atmosphere over the entire period of an accident (30 days). These processes include the sedimentation mechanism of gravitational settling, such as aerosol agglomeration, and the phoretic mechanisms of diffusiophoresis and thermophoresis.*
>
> *The Westinghouse calculation of aerosol removal coefficients is based on an analysis of a single T-H scenario and uses a single aerosol model without providing an uncertainty analysis. The staff believes that the Westinghouse approach, though potentially acceptable, represents a single BE [Best Estimate] result. Westinghouse used T-H conditions associated with the 3BE-1 severe accident sequence. The staff concludes that using the T-H conditions associated with the 3BE-1 severe accident sequence represents the spectrum of accidents evaluated for the AP1000 ... and that ... the 3BE-1 accident sequence is appropriate for determining the amount of credit to give to the natural aerosol removal processes in the AP1000 containment.*

The following was also pointed out with respect to the selection of the 3BE-1 sequence:

> *The use of a fully depressurized, low-pressure accident sequence in conjunction with the source term described in NUREG-1465 is appropriate because the release fractions for the source terms presented in NUREG-1465 are intended to be representative or typical of those associated with a low-pressure core melt accident.*

To test the potential impacts of uncertainty on what appeared to the NRC to be a "single BE result", the MELCOR code (a severe accident code similar to MAAP4) was used by the NRC to provide the T-H conditions for an uncertainty analysis of activity removal. Variations in the T-H conditions were not included in the uncertainty analysis, although 11 other parameters were varied; i.e.,

> *The staff's uncertainty analysis did not include differences between the staff and Westinghouse calculations with respect to containment T-H and containment modeling as variables for study.*

The results of the uncertainty analysis were positive.

> *The staff finds the radiological consequence analysis of the postulated DBA LOCA acceptable, based on the Westinghouse DCD Tier 2, Chapter 15, plant parameters used in the staff's analysis, the staff-calculated aerosol removal coefficient estimates (50th percentile, 95 percent confidence), and the latest revision to the AP1000 χ/Qs, as documented in the Revision 5 response to DSER Open Item 15.3-1, dated June 21, 2004. Westinghouse will include the revised information, including χ/Qs, in revision 12 of the AP1000 DCD and the staff will confirm. With this basis, the doses meet the regulatory criteria of 10 CFR 50.34 and GDC 19 and are, therefore, acceptable.*

Specifically, with respect to the T-H conditions:

> *Because of the unique nature of the AP1000 design, which enhances natural aerosol removal phenomena (such as the enhanced condensation of steam by external cooling of the containment vessel instead of an internal containment spray), the staff has approved the use of this [MAAP4] T-H profile specifically for the AP1000. The NRC does not intend credit for aerosol removal because of diffusiophoresis and thermophoresis to be generic for other plant designs, and this practice must be approved on a case-by-case basis.*

To summarize the application of MAAP4 T-H conditions to AP600 and AP1000:

- MAAP4 was shown to be able to adequately model the important phenomena associated with core damage accident progression for purposes of calculating aerosol removal rates. Specifically, the 3BE-1 event (a fully depressurized, low-pressure accident sequence of the kind considered by NUREG-1465 for release fractions and timing) was modeled.

- NRC Staff performed similar calculations using MELCOR using the same event for the same purpose. Although 11 parameters were used by the NRC Staff in an uncertainty analysis to confirm the acceptability of the aerosol removal coefficients provided by Westinghouse, the MAAP4 T-H conditions were accepted as is.

MAAP4 has been used in a similar way for Oyster Creek. A recirculation loop large LOCA was used as the initiating event with an assumed delay in core spray operation to produce the RG 1.183-appropriate degree of core damage. Because of the BWR-2 design, the recirculation loop large LOCA produced very little steaming without core spray, and consequently very little hydrogen. The small amount of hydrogen generation (and the small partial pressure of hydrogen that was the result) contributed to periodic interruption of spray operation as described above. Therefore, the combination of MAAP4 and the selection of the event to be analyzed bound the expected response in terms of minimizing the effectiveness of sprays.

2. RG 1.183 Position 4.5 states that technical specification values should be used. Position 6.2 states a similar position for the main steam isolation valves. Given that secondary bypass leakage rate is calculated as a function of pressure:

   a. Provide justification that leakage through narrow, ill-defined clearances that may change with pressure, like the stem and seat areas of valves, can be modeled as isentropic nozzles.

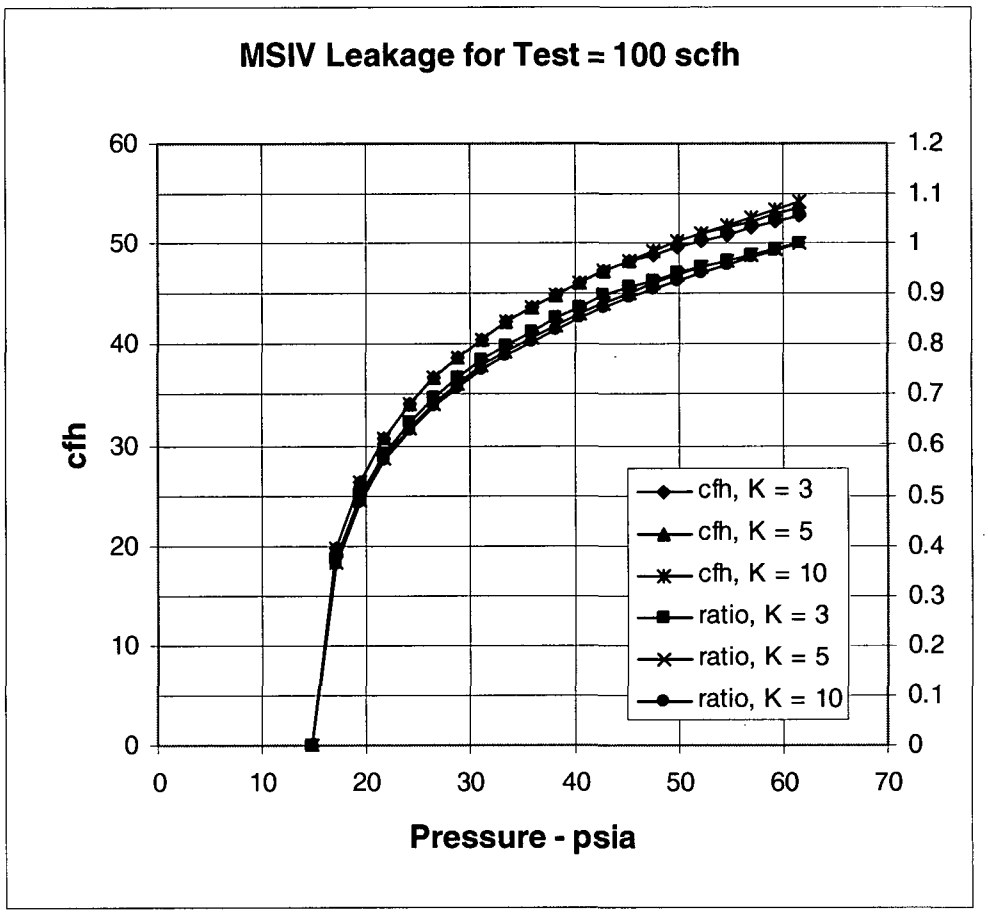
   Response

   The volumetric flow through any flowpath is basically a function of four parameters: the upstream pressure, the fluid density, the flowpath area, and the flowpath resistance. For a given fluid (e.g., air or nitrogen at valve test conditions corresponding to a given pressure and temperature), a given flow rate can be reproduced analytically by an infinite number of specific combinations of flowpath area and flowpath resistance. One such combination of flowpath area and flowpath resistance is the assumption of an isentropic nozzle. Such an assumption minimizes resistance and, therefore, minimizes flow area for a given leak rate, gas test pressure, and gas test temperature.

   There is no assurance that the flow area obtained from the assumption of an isentropic nozzle is correct; in fact, it is almost certain to be greater (as would be, also, the resistance to flow). The key assumption is that the "actual" flow area and flow resistance would produce a ratio of accident volumetric flow to test volumetric flow that matches that produced by the assumed isentropic nozzle.

   It is technically the case that the ratio of accident flow to test flow would be weakly affected by the assumption of a specific flow resistance. This effect is shown in the following study of Main Steam Isolation Valve (MSIV) volumetric flow vs. pressure with resistance as a parameter for a given mass flow at test conditions (100 scfh):

Figure 1



The plotted ratio in Figure 1 (right-hand vertical axis) is that of volumetric flow at 61.6 psia (the peak accident pressure for the case studied – not Oyster Creek) to volumetric flow at a given pressure less than the peak accident pressure. One can see that the ratio is not greatly affected by the assumed head loss coefficient, K; and this observation tends to confirm the assumption that any specific value of assumed flow resistance would not produce ratios of accident flow to test flow greatly different from another. (Note that the head loss coefficient for an isentropic nozzle is effectively zero since all velocity head is recovered. While that case was not specifically included in this study, the trend of the study is clear: lower flow resistances tend to create slightly greater ratios of accident flow to test flow for pressures less than the test pressure. This means that for Oyster Creek, the assumption of an isentropic nozzle tends to result in slightly higher containment leakage rates as the containment pressure decreases than would be the case for $K > 0$. However, in any case, the effect is small.)

The assumption of an isentropic nozzle comes directly from the pre-alternative source term (AST) licensing basis for Oyster Creek, but it is as valid as any other flowpath assumption since the impact of accident conditions vs. test conditions (fluid pressure and temperature) on volumetric leak rate is assessed on a <u>relative</u> basis (not on an absolute basis) as just described.

In modeling the valve leakage on a relative basis, the assumption is made that the flowpath area and flowpath resistance under accident conditions remain the same as they were under test conditions. That is, the change in going from test conditions to accident conditions is one only of fluid properties.

Part of the NRC Staff's question, however, deals with the flow area (i.e., is it possible that the flow area would actually become greater as pressure decreases?). If so, then the volumetric leak rate may be higher at low pressure than the assumption of constant flow area would indicate. This question is independent of any assumed relationship between flow area and flow resistance since rather than having both increase or decrease to match the measured flow under test conditions, it is possible that the flow area could increase and the resistance either remain the same or actually decrease as the upstream pressure decreases.

It is noted that the leak areas are very small in absolute terms to achieve the required leak tightness. In a response to previous NRC RAIs regarding the Oyster Creek AST application (Oyster Creek letter to the NRC, dated February 9, 2001), the impact of passing large quantities of particulate through such leak paths was discussed. The following is excerpted from that earlier response:

Impaction DF as Determined by the Potential for Leak Path Plugging

It is known that aerosol approaching an abrupt contraction of a flowpath (especially if the carrier gas is accelerating to a very high velocity and the streamlines are exhibiting substantial curvature) will tend to deviate from the carrier gas streamlines and will impact on the area around the abrupt contraction. Particles being collected around the leak path contraction will tend to plug the contraction if the leak path is sufficiently small. From this perspective, it is interesting to study the mass of aerosol leaked out of the Oyster Creek drywell through each of the inboard MSIVs as a function of time to get an estimate of when the leak paths would be plugged in terms of timing and quantity of aerosol leaked prior to plugging.

According to the Vaughan/Morewitz plugging model ..., leak path plugging is predicted when the "suspended mass carried to or past plug" amounts to KD^3 where D is the diameter of the leak path and K equals 30 $\pm$ 20 g/cm^3.

The equivalent orifice diameter corresponding to the Oyster Creek MSIV leak test is 0.049 cm. Note that this assumption is conservative as the MSIV leak path is represented here as a single orifice, while the real leakage is believed to occur at different locations, each location being characterized by a much smaller (and more easily plugged) leak path.

Using the above expression with the most conservative value for K, the leak path would be plugged when 5.9E-3 grams of aerosol has leaked through the single orifice. ... It is noteworthy that the leak path would be plugged very early into the event, as 5.9E-3 grams of aerosol would have leaked through the hypothetical MSIV leak orifice in only 180 seconds.

This discussion indicates that in as little as three minutes, sufficient aerosol could have been brought to the entrance of the MSIV leak path to bring about complete plugging of the leak path. Complete plugging is not credited in the analysis, but such a consideration does put into context any concern regarding a small increase in the flow area that may occur at low pressures. Even if the flow area were to double, for example, the amount of aerosol needed to plug the leak path completely would increase by less than a factor of three; and instead of plugging in three minutes, plugging might require nine or ten minutes. This is still a very small amount of time relative to the duration of the dose analysis.

b. Provide a reference to an NRC approval supporting page 36/45 of Attachment 1, which states that this modeling approach is consistent with the current licensing basis.

Response

NRC letter to Oyster Creek, dated September 2, 1982 (LS05-82-09-011), "Safety Evaluation of SEP Topic XV-19, "Radiological Consequences of a Loss Of Coolant Accident"," provided the staff's final evaluation of this Integrated Plant Safety Assessment Systematic Evaluation Program (SEP) Topic for Oyster Creek. This SER states that because of the uncertainties in the calculation of the doses and because the estimated thyroid doses could exceed the 10 CFR 100 dose guideline by 14%, it is recommended that a more realistic analysis be performed for MSIV doses factoring in the effects of drywell pressure vs. MSIV leakage rate as a function of time. The NRC final evaluation of the Oyster Creek Integrated Plant Safety Assessment Systematic Evaluation Program (SEP), NUREG-0822, January 1983, Section 4.38, Topic XV-19, "Loss-of–Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary," documented the results of the NRC staff's assessment of dose consequences for the postulated design basis accident and closed this SEP Topic.

The NRC SER, dated July 15, 1986, License Amendment No. 105, reviewed the Oyster Creek control room habitability analysis, including the principal assumptions, methodology, and results regarding design basis radiation doses to the Oyster Creek control room operators, as described in Oyster Creek submittals dated June 4, 1985 and June 17, 1985. The analysis supporting these submittals was based on realistic MSIV leak rates determined by considering the effect of the post-LOCA pressure/temperature response of primary containment on the MSIV leakage. This SER stated that the assumptions used in the analysis were reasonable and acceptable.

Additionally, Oyster Creek letter to the NRC, dated May 16, 1989, entitled, "Control Room Habitability," provided final responses to NRC Request for Additional Information, dated February 22, 1984. These responses described the MSIV leakage reassessment considering MSIV leakage as a function of accumulator and containment pressures. This RAI response addressed the design objectives identified in the Oyster Creek June 4, 1985 letter to NRC, Control Room Habitability (NUREG-0737, Item III.D.3.4), which was the basis for the Oyster Creek License Amendment No. 105, dated July 15, 1986, referenced above, and also supported NRC issuance of Oyster Creek License Amendment No. 139, dated May 29, 1990, for control room habitability Technical Specification modifications.

Based on the above, Oyster Creek Updated Final Safety Analysis Report (UFSAR) Section 6.4.4.1 describes the modeling approach consistent with the current licensing basis assumption that the MSIV leakage was reassessed considering MSIV leakage as a function of accumulator and containment pressures, and that the pressure profile utilized was considered conservative for the assessment.

c. Provide the results of a sensitivity study to show the difference between the time-dependent leakage assumption and the results using the technical specification leakage.

Response

Response to be provided by separate submittal.

3. On page 33 of Attachment 1 to the March 28,2005, submittal, AmerGen states that the current licensing basis for Oyster Creek includes an assumption of full mixing credit for dilution/mixing in the secondary containment. Please provide a reference for the NRC staff approval of this assumption. If applicable, provide a sensitivity study supportive of assuming full mixing credit.

Response

Response to be provided by separate submittal.

4. With regard to the assumptions for drywell iodine removal:

a. Justify the use of the STARNAUA removal models for this application. One way that would be acceptable to the NRC staff would be to compare STARNAUA to the models referred to in RG 1.183 as acceptable spray removal and natural deposition models.

Response

The spray removal model for aerosol is basically the same for both STARNAUA and Standard Review Plan (SRP) 6.5.2 (the latter being identified in RG 1.183 as being a model acceptable to the NRC). The SRP 6.5.2 states the following:

The first-order removal coefficient, λ_P, for particulates may be estimated by

$$\lambda_P = 3hFE / 2VD$$

where h is the fall height of the spray drops, V is the containment building net free volume, F is the spray flow, and (E/D) is the ratio of a dimensionless collection efficiency E to the average spray drop diameter D. Since the removal of particulate material depends markedly upon the relative sizes of the particles and the spray drops, it is convenient to combine parameters that cannot be known (Ref. 13). It is conservative to assume (E/D) to be 10 per

meter initially (i.e., 1% efficiency for spray drops of one millimeter in diameter), changing abruptly to one per meter after the aerosol mass has been depleted by a factor of 50 (i.e., 98% of the suspended mass is ten times more readily removed than the remaining 2%).

Reference 13 above is: A. K. Postma, R. R. Sherry, and P. S. Tam, "Technological Bases for Models of Spray Washout of Airborne Contaminants in Containment Vessel", U.S. Nuclear Regulatory Commission Report, NUREG/CR-0009, October 1978.

The difference between STARNAUA and SRP 6.5.2 is in the quantification of E/D. As noted above, the E/D is conservatively estimated to be 10 per hour (1% efficiency for a one mm drop) until 98% of the mass has been removed and then one per hour (0.1% efficiency for one mm drop) after that. The removal efficiency calculated rigorously by STARNAUA is similar as seen on Figure 2.

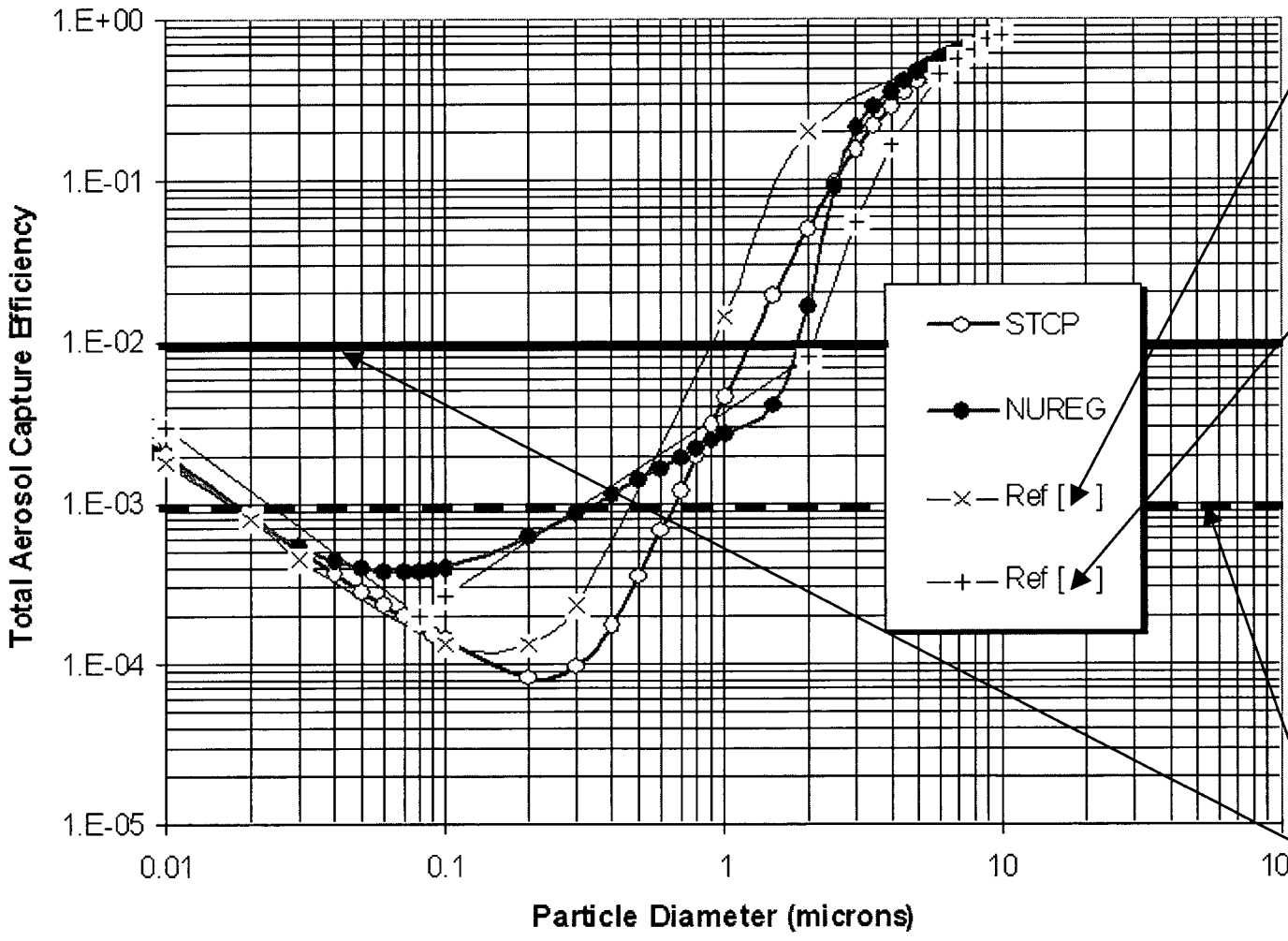
In Figure 2, both STARNAUA modeling options (STCP and NUREG) are presented. The NUREG option is preferred and has been exercised for Oyster Creek. Figure 2 shows the removal efficiency for particles of different sizes. It may be noted that for sub-micron particles, the rigorously calculated removal efficiency may be substantially lower than 1% and even 0.1% (the removal efficiencies used for the SRP 6.5.2 model). The key, therefore, is having an accurate estimate of the size distribution of the airborne particulate.

The source particle size specified for Oyster Creek is r_g = 0.23 microns with a logarithmic standard deviation of 0.593 and a specific gravity of 3.23 during the gap release and 5.14 during the early in-vessel release. Therefore, the aerodynamic geometric diameter of the distribution is very close to one micron (+/-), the point where the efficiency of the SRP 6.5.2 model begins to fall below (or exceed) that of STARNAUA. The greater removal rate of large particles competes with particle agglomeration to establish a given particle size distribution in the course of the analysis, the average particle aerodynamic diameter becoming some what greater than one micron as the analysis proceeds.

If the SRP 6.5.2 model were used, the spray lambda for Oyster Creek would be approximately 17 per hour during the release phase. The STARNAUA results vary between 48 per hour (immediately after one of the spray restarts at t = 68 minutes) to 16.5 per hour (quasi-steady state at the end of the early in-vessel release phase). During the time the sprays are off, agglomeration tends to increase the airborne particle size so that when sprays are restarted, there is a short-duration surge in removal rate. The "steady-state" removal rates are very similar (SRP vs. STARNAUA), as the diagram below would suggest for particle aerodynamic diameters between one and two microns. Therefore, for Oyster Creek, the best way to characterize the spray removal rates is that they are comparable to those derived from NRC methods, but not as conservative. They are more physical in that when sprays are secured, the growth in the airborne particles by agglomeration is reflected in the brief increase in the removal rate when the sprays are restarted.

Figure 2

STARNAUA Spray Droplet Efficiency for 1000 um Droplet Compared to SRP 6.5.2



R.K. Hilliard, A.K. Postma, J.D. McCormack, and L.F. Coleman, "Removal of Iodine and Particles by Sprays in the Containment Systems Experiment", Nucl. Tech. 10, 499 (1971).

S.L. Magruder, "Minutes of Public Meeting of January 26, 1994, Regarding In-Containment Refueling Water Storage Tank (IRWST) PH Control and the Use of the New Source Term for the ABB-CE System 80+ Standard Plant Design", U.S. NRC, April 12, 1994.

Note: "NUREG" is D.A. Powers and S.B. Burson, "A Simplified Model of Aerosol Removal by Containment Sprays", NUREG/CR-5966, SAND92-2689, U.S. NRC, June 1993

SRP 6.5.2: heavy solid line for DF < 50, heavy dashed line for DF > 50

For steam line deposition, the lambda is related to the steam line internal diameter and deposition velocity, u, by the expression $u = \pi D\lambda/4$. Since the steam line ID is 21.56" or 0.55 m, the deposition velocity corresponding to the maximum steam line lambda of 2.6 per hour is approximately 3.1E-4 mps. This is approximately the 15th percentile sedimentation velocity from NRC AEB-98-03, dated December 9, 1998, entitled, "Assessment of Radiological Consequences for the Perry Pilot Plant Application using the Revised (NUREG-1465) Source Term." During the gap release phase and for the first two hours or so after the in-vessel release phase, the deposition velocity is approximately half this value, decreasing even further (to 4.4E-5 mps) late on the first day of the accident. The decrease is the result of the decreasing particle size of the activity remaining airborne.

b. On page 8 of Attachment 3 AmerGen gives justification for assuming that the aerosol and elemental iodine removal rates are the same in the drywell, stating that it is believed that the elemental iodine will adhere to the aerosol, and if that is not so that the elemental iodine would be removed from the containment at a rate greater than the particulate. What is the basis for the statement that elemental iodine would be removed from the containment at a rate greater than particulate?

Response

In SRP 6.5.2, the elemental iodine removal rate calculation is described as follows:

> *During injection, the effectiveness of the spray against elemental iodine vapor is chiefly determined by the rate at which fresh solution surface area is introduced into the containment building atmosphere. The rate of solution surface created per unit gas volume in the containment atmosphere may be estimated as (6F/VD), where F is the volume flow rate of the spray pump, V is the containment building net free volume, and D is the mass-mean diameter of the spray drops. The first-order removal coefficient by spray, λ_S, may be taken to be*
>
> $$\lambda_S = 6K_gTF / VD$$
>
> *where K_g is the gas-phase mass-transfer coefficient, and T is the time of fall of the drops, which may be estimated by the ratio of the average fall height to the terminal velocity of the mass-mean drop (Ref. 14). The above expression represents a first-order approximation if a well-mixed droplet model is used for the spray efficiency. The expression is valid for λ_S values equal to or greater than ten per hour. λ_S is to be limited to 20 per hour to prevent extrapolation beyond the existing data for boric acid solutions with a pH of 5 (Refs. 8 and 11). For λ_S values less than ten per hour, analyses using a more sophisticated expression are recommended.*

References 8, 11, and 14 above are, respectively:

R. K. Hilliard, A. K. Postma, J. D. McCormack, L. F. Coleman, and C. E. Lunderman, "Removal of Iodine and Particulates From Containment Atmospheres – Containment Systems Experiments", Pacific Northwest Laboratories Report, BNWL-1244, February 1970.

A. K. Postma, L. F. Coleman, and R. K. Hilliard, "Iodine Removal From Containment Atmospheres by Boric Acid Spray," Pacific Northwest Laboratories Report, BNP-100, July 1970.

G. B. Wallis, "The Terminal Speed of Single Drops or Bubbles in an Infinite Medium", International Journal of Multiphase Flow, 1, pages 491-511 (1974).

A typical value for K_g is 6E-2 mps and a typical droplet exposure time (given the relatively small fall height of Oyster Creek) would be about three seconds. Therefore, the $6K_gT$ part of the expression is roughly equal to one meter for Oyster Creek.

The spray flow divided by the volume of the drywell is approximately 0.13 per hour, and the mass mean spray droplet size is about 3E-3 m. Therefore, the elemental iodine removal lambda for a "clean" spray would exceed 40 per hour. Given that the post-accident suppression pool pH for Oyster Creek is much greater than 5, one would not expect to have to limit the elemental iodine removal lambda to 20 per hour; and on average, therefore, the elemental iodine removal rate would greatly exceed that of the particulate.

Even if the elemental iodine lambda were limited to 20 per hour by the SRP convention, the integrated fraction of the core's elemental iodine airborne during spray operation would be about 1.0%-hour using the particulate lambdas and about 1.1%-hour using the elemental iodine removal lambda of 20 per hour. Given that the elemental iodine is only 4.85% of the iodine release, these integrated airborne fractions are essentially the same.

Because the pH of the suppression pool is controlled post-accident, the equilibrium iodine in elemental form in the gas phase when the minimum pH is reached after 30 days is only ~1E-4 of that released to the containment. Since about five percent of the radioiodine release is in elemental form to begin with, the ultimate DF for elemental iodine that can be justified is about 500 (i.e., 5E-2/1E-4). Based on such a large DF and the fact that the 0.15% of the radioiodine that is released in organic form is assumed to never be removed from the gas phase (equivalent to a DF of ~30 relative to the ~5% elemental iodine release), there is no practical need to limit the removal of elemental iodine. For this reason (and the fact that treating the elemental iodine as particulate is conservative from the standpoint of removal rate), the assumption that elemental iodine behaves as particulate is not only technically correct, it also has either no effect on the analysis or is actually somewhat conservative.

5. Address the aggregated effects of the assumptions discussed in questions 2 through 4, above.

Response

Response to be provided by separate submittal.

6. Credit is proposed for control of the pH in the suppression pool following a loss-of coolant accident (LOCA) by means of injecting sodium pentaborate into the reactor core with the standby liquid control (SLC) system. The SLC system design was not previously reviewed for this safety function (pH control post-LOCA). Licensees proposing such credit need to demonstrate that the SLC system is capable of performing the pH control safety function assumed in the AST LOCA dose analysis.

    a. Identify whether the SLC system is classified as a safety-related system as defined in Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.2, and whether the system satisfies the regulatory requirements for such systems. If the SLC system is not classified as safety related, please provide the information requested in items (i) through (v) below to show that the SLC system is comparable to a system classified as safety related. If any item is answered in the negative, please explain why the SLC system should be found acceptable for pH control agent injection.

        (i) Is the SLC system provided with standby AC power supplemented by the emergency diesel generators?

        (ii) Is the SLC system seismically qualified in accordance with RG 1.29 and Appendix A to 10 CFR Part 100 (or equivalent used for original licensing)?

        (iii) Is the SLC system incorporated into the plant's American Society of Mechanical Engineers Boiler and Pressure Vessel Code inservice inspection and inservice testing programs based upon the Oyster Creek's code of record in accordance with 10 CFR 50.55a?

        (iv) Is the SLC system incorporated into the Oyster Creek's Maintenance Rule program consistent with 10 CFR 50.65?

        (v) Does the SLC system meet the requirements of 10 CFR 50.49? Describe how the SLC system design addresses General Design Criterion 4, or equivalent used for original licensing?

Response

The SLC system at Oyster Creek is classified as a Safety Related system as defined in 10 CFR 50.2. The Oyster Creek SLC System satisfies the regulatory requirements for Safety Related Structures, Systems, and Components (SSCs) specified in 10 CFR Part 50.2. The SLC System is powered by the 480 VAC distribution system. In the event of loss of power, the 480 VAC distribution system, including the SLC System, is powered by the emergency diesel generators. The SLC System is seismically designed as described in Updated Final Safety Analysis Report (UFSAR) Section 9.3.5. The SLC System is designed to be operable during and after a seismic event and is classified as Seismic Category I. Some support functions, such as heat tracing, are classified as anti-fall down.

The SLC System is incorporated into the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Inservice Inspection (ISI) and Inservice Testing (IST) Programs as required by 10 CFR 50.55a, "Codes and standards." The

SLC System function (i.e., injection of sodium pentaborate during LOCA conditions) and the components described herein are scoped into the Maintenance Rule Program in accordance with 10 CFR 50.65.

The SLC System is not currently subject to the requirements of 10 CFR 50.49, "Environmental qualification of electric equipment important to safety for nuclear power plants," since it does not provide any design function to accidents that would cause a harsh environment. Oyster Creek was designed and built prior to the implementation of 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants." As part of the application for a Full Term Operating License, the design of Oyster Creek, was evaluated against the requirements of 10 CFR 50.34, Appendix A, "General Design Criteria for Nuclear Power Plants," in effect on July 7, 1971. This evaluation was submitted as Amendment No. 68 to the original Facility Description and Safety Analysis Report (FDSAR). Conformance with NRC General Design Criteria for Oyster Creek has also been established as part of the Systematic Evaluation Program detailed in NRC NUREG-0822, and summarized in the UFSAR Sections 1.10 and 3.1. The design capability of the Oyster Creek SLC System has been reviewed and accepted by the NRC in the Safety Evaluation Report for License Amendment No. 124, dated July 14, 1988. The design basis of the SLC System is described in Section 9.3.5 of the Oyster Creek UFSAR.

b. Describe proposed changes to plant procedures that implement SLC sodium pentaborate injection as a pH control additive and associated operator training.

Response

The Emergency Operating Procedures (EOPs) and Severe Accident Management Guidelines (SAMGs) at Oyster Creek will be changed to direct the Operators to manually initiate the Liquid Poison System under either of the following conditions:

- Non-ATWS Events – Reactor Pressure Vessel (RPV) water level cannot be restored and maintained above Minimum Steam Cooling RPV Water Level (MSCRWL)
- ATWS Events - RPV water level cannot be restored and maintained above the Minimum Steam Cooling RPV Water Level (MSCRWL)

Oyster Creek EOP Support Procedure 7 provides specific guidance to the Operators on operation of the Liquid Poison System for RPV water level control. This procedure will be revised to direct the Operator to inject the entire contents of the Liquid Poison tank in the event that a LOCA is in progress.

No changes are required to the Severe Accident Management Guidelines since entry into the SAMGs already requires the Liquid Poison System to be initiated.

The Oyster Creek EOPs are entered any time that RPV water level cannot be maintained above the scram setpoint of 137 inches above Top of Active Fuel (TAF). Under LOCA conditions where RPV water level cannot be maintained above the lowest level that ensures adequate core cooling, the changes above will ensure that SLC is injected into the reactor for suppression pool pH control.

In the event that SLC is being injected for ATWS mitigation concurrent with a LOCA, the changes above will ensure that SLC will continue to be injected even if no longer required to mitigate an ATWS.

Operator Training

All changes to the EOPs are specifically included in Licensed Operator Training and the changes above and their basis will be included as part of the EOP change implementation.

Operation of the Liquid Poison System for RPV water level control is already included in the Oyster Creek EOPs and SAMGs, and is already trained on as part of the 2-year training program for Licensed Operators. Since no changes in the method of operating the Liquid Poison System are being made, only the basis/reason for SLC injection and EOP step sequencing need to be added to the Licensed Operator Requalification (LOR) training program.

The use of sodium pentaborate for pH control of the suppression pool under LOCA conditions will also be included in the EOP User's Guide developed for use by the operators.

c. How is transport of the sodium pentaborate to the suppression pool assured to occur? Is a low-pressure safety injection pump injecting coolant at the time of SLC injection?

Response

The SLC System at Oyster Creek is manually initiated and injected into the bottom of the Reactor Pressure Vessel (RPV). Under the postulated scenario, the low pressure Core Spray System will be running, taking suction from the suppression pool and injecting into the RPV. Therefore the sodium pentaborate will be transported from the RPV to the Drywell through the break in the Reactor Coolant System. Overflow from the Drywell will go to the suppression pool thereby transporting sodium pentaborate from the RPV to the suppression pool.

d. Show that the SLC system has suitable redundancy in components and features to assure that, for operation from onsite or offsite electric power, its safety function of injecting sodium pentaborate for the purpose of suppression pool pH control can be accomplished assuming a single failure. For this purpose, the check valve is considered an active device since the check valve must open to inject sodium pentaborate.

For reference, the following three options are listed as ways to justify taking credit for the SLC system if it can not be considered redundant with respect to its active components.

Option 1: Show acceptable quality and reliability of the non-redundant active components and/or compensatory actions in the event of failure of the non-redundant active components. If you choose this option, please provide the following information to justify the lack of redundancy of active components in the SLC system:

(1) Identify the non-redundant active components in the SLC system and provide their make, manufacturer, and model number.

Response

The Oyster Creek SLC System has the following non-redundant active components:

TABLE 1			
Non-Redundant Component	**Component Description**	**Component Manufacturer**	**Component Model Number**
V-19-16	Liquid poison inlet check valve to reactor outside drywell	Velan Valve Corp	2A34B
V-19-20	Liquid poison inlet check valve to reactor in the Drywell	Velan Valve Corp	2A34B
TIC-1106-32	Control for liquid poison pump suction line heat coil	Chromalox	AR2529
H-19-1	Liquid poison pump suction line heater	Chemelex Division / Raychem Corp	8BTV1
TIS-IL0009	Poison tank temperature indicating control switch	Fenwal Electronics Div / Kidde Inc	55101140-340
T-19-1	Liquid poison tank immersion heater	Saracco Tank Manufacturing Company	73-4157
Liquid Poison Control Switch	Liquid poison pump NP02-A/ NP02-B control switch (14S1) in Control Room panel 4F	General Electric	CR2940UN200E

(2) Provide the design-basis conditions for the component and the environmental and seismic conditions under which the component may be required to operate during a DBA. Environmental conditions include design-basis pressure, temperature, relative humidity and radiation fields.

Response

Table 2 below summarizes the design basis conditions for the non-redundant active components, for both normal operation and during an accident.

TABLE 2										
Component	Component Location	Seismic	Pressure		Temperature		Relative Humidity		Radiation fields	
			Normal	Accident	Normal	Accident	Normal	Accident	Normal	Accident
V-19-16	Reactor Building elevation 95'	Operable during and after seismic event, seismic class I.	14.7 psia	15.5 psia	79°F	271.3°F	Varies Not a concern	100%	8.77×10^3 Rads	2×10^4 R gamma 6.7×10^5 R beta
V-19-20	Drywell elevation 82'	Operable during and after seismic event, seismic class I.	16 psia	53.1 psia	184°F	317°F (local area temperature)	50%	100%	2×10^7 Rads	3.2×10^7 R gamma 9.6×10^8 R beta
TIC-1106-32	Reactor Building elevation 95'	Reg. Guide 1.29 (anti fall down)	14.7 psia	15.5 psia	79°F	271.3°F	Varies Not a concern	100%	8.77×10^3 Rads	2×10^4 R gamma 6.7×10^5 R beta
H-19-1	Reactor Building elevation 95'	Reg. Guide 1.29 (anti fall down)	14.7 psia	15.5 psia	79°F	271.3°F	Varies Not a concern	100%	8.77×10^3 Rads	2×10^4 R gamma 6.7×10^5 R beta
TIS-IL0009	Reactor Building elevation 95'	Operable after seismic event, seismic class I.	14.7 psia	15.5 psia	79°F	271.3°F	Varies Not a concern	100%	8.77×10^3 Rads	2×10^4 R gamma 6.7×10^5 R beta
T-19-1	Reactor Building elevation 95'	Operable during and after seismic event, seismic class I.	14.7 psia	15.5 psia	79°F	271.3°F	Varies Not a concern	100%	8.77×10^3 Rads	2×10^4 R gamma 6.7×10^5 R beta
Liquid Poison Control Switch	Control Room panel 4F	Operable during and after seismic event, seismic class I.	14.7 psia	Mild Environ-ment	80°F	Mild Environ-ment	Varies Not a concern	N/A	Mild Environ-ment	Mild Environ-ment

(3) Indicate whether the component was purchased in accordance with Appendix B to 10 CFR Part 50. If the component was not purchased in accordance with Appendix B, provide information on the quality standards under which it was purchased.

Response

The SLC system was designed and installed by General Electric as a Safety-Related system prior to development of 10 CFR 50, Appendix B. The NRC Safety Evaluation Report relating to the Full-Term Operating License for Oyster Creek (NUREG-1382), issued January 29, 1991, confirmed the design of the Oyster Creek SLC System to be acceptable.

The following non-redundant active components are classified as Safety-Related in accordance with the AmerGen Quality Assurance Topical Report (QATR). Any replacement, repairs, work and modifications are performed in accordance with Appendix B of 10 CFR 50.

1. V-19-16: Liquid poison inlet check valve to reactor outside Drywell
2. V-19-20: Liquid poison inlet check valve to reactor in the Drywell
3. TIS-IL0009: Control for liquid poison pump suction line heat coil
4. T-19-1: Liquid poison tank immersion heater
5. Switch 14S1: Liquid poison pump NP02-A/ NP02-B control switch in Control Room panel 4F

The following non-redundant active components are classified as Augmented Quality in accordance with the AmerGen QATR.

1. H-19-1: Liquid poison pump suction line heater
2. TIS-IL0009: Poison tank temperature indicating control switch

(4) Provide the performance history of the component both at the licensee's facility and in industry databases such as EPIX and NPRDS.

Response

An internal and external performance history search for the above non-redundant active components was performed using the following databases: Oyster Creek PIMS, Oyster Creek CAP, EPIX, INPO, and NPRDS.

The search results documented no instances of the SLC check valves (V-19-16 and V-19-20) failing to open and no instances of liquid poison tank immersion heater (T-19-1) failure to control temperature. No internal and external failures have been experienced in regards to the SLC pump control switch used at Oyster Creek.

Several internal and external operating experiences were identified in regards suction line heater (H-19-1), the tank temperature indicating control switch (TIS-IL0009), and control for liquid poison pump suction line heat coil

(TIC-1106-32). During these events the temperatures were found out of tolerance. However, in none of these events were the out of tolerance values significant enough to affect the condition of the sodium pentaborate solution.

(5) Provide a description of the component's inspection and testing program, including standards, frequency, and acceptance criteria.

Response

V-19-16 and V-19-20:

The two check valves mentioned above have an open function to support the injection of sodium pentaborate and a close function for primary containment isolation. The open function is tested under the Inservice Test (IST) Program every refueling outage. Oyster Creek Technical Specifications require verification of flow through one SLC subsystem from a pump into the RPV. This allows the system to be tested for complete continuity during a shutdown when demineralized water can be pumped into the RPV. During the test, one of the subsystems, including an explosive valve, is initiated, and it is verified that a flow path from the pump to the RPV is available. This test verifies proper operation of both check valves. In order to meet the acceptance criteria for this test, all components tested need to perform their intended function.

The close function is tested under the Local Leak Rate Test (LLRT) Program every refueling outage (i.e., every 2 years). This leak test is performed to verify containment isolation capability. Performance of this test is in accordance with 10 CFR 50, Appendix J, and satisfies the requirements of the ASME Code to perform a low pressure seat leak test. Acceptance criteria for V-19-16 and V-19-20 are 3.0 SCFH (1.376 SLM) and 7.0 SCFH (2.904 SLM), respectively.

TIC-1106-32, H-19-1, TIS-IL0009 and T-19-1:

Components TIC-1106-32 and TIS-IL0009 are calibrated every 365 days. Acceptance criteria for TIC-1106-32 and TIS-IL009 are 125 +/-10 °F and 100 +/- 1°F respectively. Performance of components H-19-1 and T-19-1 is verified by successful completion of the Preventive Maintenance described above.

In addition, these heating and temperature control components are monitored by the Operators, once per shift. The readings taken by the Operators have a monitoring acceptance criterion. The acceptance criterion for TIC-1106-32 is a reading no lower than 95°F and no greater than 125°F. The acceptance criterion for TIS-IL0009 is equal to or greater than 90°F. Actions are taken, as needed, if the readings obtained are outside of the acceptance criteria.

Liquid Poison Control Switch:

The SLC System Liquid Poison Control Switch is tested every Refueling Outage under the IST program mentioned above, which verifies proper operation of the system.

(6) Indicate potential compensating actions that could be taken within an acceptable time period to address the failure of the component. An example of a compensating action might be the ability to jumper a switch in the control room to overcome its failure. In the response please consider the availability of compensating actions and the likelihood of successful injection of the sodium pentaborate when non-redundant active components fail to perform their intended functions.

Response

The control switch for both of the SLC pumps has been determined to be highly reliable based on the performance history review performed. However, if the SLC control switch was to fail, a jumper can be installed to bypass the switch and initiate the SLC injection. Based on the strategic location of this control switch and given a time range of 24 hours, there is a high likelihood of successful injection of the sodium pentaborate even if this switch is to fail.

In case of a single failure of any of the components of the SLC System (including either or both of the non-redundant check valves, the tank heater coil and temperature controller, or the suction pipe heater coil and temperature controller), alternate chemical injection could be accomplish by using the Feedwater System at Oyster Creek. This alternate method is contained in the Oyster Creek Emergency Operating Procedures (EOPs). There is a high likelihood for success of this alternate method for the following reasons:

- All actions taken and materials required are located outside the Reactor Building and away from the severe environment and high radiation levels associated with a LOCA that is progressing to the conditions evaluated by the AST submittal.
- The alternate method for injecting SLC is already contained in the EOPs and SAMGs.
- Use of the alternate method found in the EOPs is included in the LOR training program and is included as part of the job performance measures used to train and evaluate the licensed operators at Oyster Creek.
- Sufficient time exists in the postulated scenario to implement the alternate method of SLC injection.

7. Pages 4 and 5 of Attachment 1 to the March 28, 2005 AST LAR state that previously-approved licensing basis atmospheric dispersion factors (x/Q values) were used for the radiological propagation pathways.

   Table 2 on page 151 of Attachment 3 provides control room x/Q values for postulated releases from the Oyster Creek stack, yard, and turbine building. Exclusion area boundary and low population zone x/Q values for ground level and elevated releases are listed in Table 3. Confirm that the stack, yard, and turbine building are the only release locations that need to be considered to substantiate that the LOCA is the limiting DBA at Oyster Creek when applying the AST.

The turbine building control room x/Q values were discussed in the safety evaluation associated with Oyster Creek License Amendment No. 225 (Agencywide Documents Access and Management System accession number ML020320579) dated February 2, 2002. By what licensing action(s) were the other x/Q values previously approved? If any of the x/Q values were not previously approved, provide the input files (electronic files for data input into computer codes) used to generate the x/Q values, summary output files.

Response

These release locations (stack, yard, and turbine building) were chosen as bounding since they are the closest release locations to the worst Control Room (CR) air intake location. Other release locations would have lower x/Q values due to further distances from the CR air intake.

The turbine building control room x/Q values reviewed and approved in the referenced Oyster Creek License Amendment No. 225 were used in the analysis (Calculation PSAT 05201H.08, Revision 2). Other x/Q values (using the same meteorological data) were also provided in the calculation supporting Amendment 225. The hourly meteorological data files previously provided are also being re-transmitted herein (Enclosure 3).

8. Page 2 of Attachment 1 to the March 28, 2005, LAR states that AmerGen has performed a radiological consequence analysis for the Oyster Creek DBA that results in the most limiting offsite and control room operator exposure (i.e., LOCA). The analysis is presented to support full-scope implementation of the AST; although, it is further stated that Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactor Sites," will continue to be used as the radiation dose basis for the main steamline break, control rod drop and fuel handling accidents and for equipment qualification. Page 2 states that adopting the AST methodology may support future evaluations and license amendments. Page 8 states that the x/Q values for other accident situations are similar to those used in the LOCA dose assessment, which was given as one reason that the LOCA remained the limiting DBA. Provide a list of the other x/Q values to show that they are similar to the LOCA x/Q values. Justify why releases from these other scenarios as well as from bypass during secondary containment drawdown, loss of offsite power, or other single failure would not result in more limiting doses than those estimated for the LOCA DBA.

Response

The CR x/Q values for the main steam line break (MSLB) are as listed below. Although these values are somewhat higher than those determined and used for the LOCA (see values listed in the response to Question #9 below), the source term for this accident is orders of magnitude less than that for the LOCA. Therefore, the radiological consequence of a LOCA remains bounding.

- 0-8 hours: 8.37E-03 sec/m^3
- 8-24 hours: 6.36E-03 sec/m^3
- 24-96 hours: 4.44E-03 sec/m^3
- 96-720 hours: 2.68E-03 sec/m^3

The x/Q values for the CRDA and FHA release points are the same as those for the LOCA. Since the magnitudes of the releases associated with the Control Rod Drop Accident (CRDA) and the Fuel Handling Accident (FHA) are much lower than the LOCA, the radiological consequence of a LOCA remains bounding.

9. Control room AST dose assessments are typically made for the 0-2 hour, 2-8 hour, 8-24 hour, 24-96 hour, and 96-720 hour time periods. Table 2 combines the first two time periods into a single time interval presenting a single 0-8 hour x/Q value for each postulated release location. What are the 0-2 hour and 2-8 hour control room x/Q values? How does use of the 0-8 hour x/Q values impact the dose assessment when compared with inputting the 0-2 hour and 2-8 hour x/Q values?

Response

The 0-2 hour and 2-8 hour x/Q values are as follows (in sec/m^3 with the original t=0 to t=8 hour x/Q values shown in *italics* for comparison):

	Stack	Yard	Turbine Building
From t=0 to t=8 hours	*1.80E-4*	*2.59E-3*	*2.71E-3*
From t=0 to t=2 hours	1.80E-4	2.88E-3	3.73E-3
From t=2 to t=8 hours	1.80E-4	2.49E-3	2.37E-3

If the 0-2 hour control room x/Qs are actually applied from 0.508 hours to 2.508 hours (the worst time period for Exclusion Area Boundary (EAB) dose accumulation), the total inhalation + immersion dose for the control room increases from 2.99 rem TEDE to 3.27 rem TEDE (about 9%) with this change. The addition of the 0.62 rem external shine contribution (which would not be greatly affected by the x/Q changes) would bring the total control room dose to 3.89 rem TEDE with these changes (compared to 3.61 rem TEDE using the 0-8 hour values, about an 8% increase). Therefore, while the increase in dose is noticeable, its significance is limited since the margin to the five rem limit would be decreased by only 20% (i.e., from 1.39 rem TEDE to 1.11 rem TEDE) by the adoption of the 0-2 hour x/Q.

ENCLOSURE 2

REGULATORY COMMITMENTS

SUMMARY OF COMMITMENTS

The following table identifies commitments made in this document. (Any other actions discussed in the submittal represent intended or planned actions. They are described to the NRC for the NRC's information and are not regulatory commitments.)

COMMITMENT	COMMITTED DATE OR "OUTAGE"	COMMITMENT TYPE	
		ONE-TIME ACTION (Yes/No)	PROGRAMMATIC (Yes/No)
The Emergency Operating Procedures (EOPs) and Severe Accident Management Guidelines (SAMGs) at Oyster Creek will be changed to direct the Operators to manually initiate the Liquid Poison System under either of the following conditions: • Non-ATWS Events – Reactor Pressure Vessel (RPV) water level cannot be restored and maintained above Minimum Steam Cooling RPV Water Level (MSCRWL) • ATWS Events - RPV water level cannot be restored and maintained above the Minimum Steam Cooling RPV Water Level (MSCRWL)	Implement with amendment.	No	Yes
Oyster Creek EOP Support Procedure 7 will be revised to direct the Operator to inject the entire contents of the Liquid Poison tank in the event that a LOCA is in progress. Include these EOP changes and their basis in Licensed Operator Training, and update the EOP User's Guide to include the use of sodium pentaborate for pH control of the suppression pool under LOCA conditions.	Implement with amendment.	No	Yes

ENCLOSURE 3

CD – Oyster Creek Meteorological Data 1995, 1996, 1997, 1998, 1999 (ASCII Format)