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An Exelon/British Energy Company

RS-03-239 10 CFR 50.90

December 23, 2003

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

Clinton Power Station, Unit 1
Facility Operating License No. NPF-62
NRC Docket No. 50-461

Subject: Additional Information Supporting the Request for License Amendment Related to Application of the Alternative Source Term

References: (1) Letter from Michael J. Pacilio (AmerGen Energy Company, LLC) to U. S. NRC, "Request for License Amendment Related to Application of Alternative Source Term," dated April 3, 2003

(2) Letter from U. S. NRC to John L. Skolds (AmerGen Energy Company, LLC), "Clinton Power Station, Unit 1 – Request for Additional Information Regarding Alternate Source Term Submittal (TAC No. MB8365)," dated October 30, 2003

(3) Letter from U. S. NRC to John L. Skolds (AmerGen Energy Company, LLC), "Clinton Power Station, Unit 1 – Corrected Request for Additional Information Regarding Alternative Source Term Submittal (TAC No. MB8365)," dated November 18, 2003

In Reference 1, AmerGen Energy Company (AmerGen), LLC submitted a request for a change to Appendix A, Technical Specifications (TS), of Facility Operating License No. NPF-62 for Clinton Power Station (CPS). Specifically, the proposed change is requested to support application of an alternative source term (AST) methodology, in accordance with 10 CFR 50.67, "Accident source term," with the exception 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 equipment qualification.

The NRC, in Reference 2, provided AmerGen with a request for additional information.

A001

Reference 3 provided a correction to the request provided in Reference 2. As indicated in the response to Question 1, AmerGen is revising the piping deposition calculation. Therefore, the attachment to this letter provides the requested information for all questions in Reference 3 except for the responses to Questions 4, 5, and 12. Responses to these remaining NRC questions will be provided separately once the calculation revision is complete. The response to Question 2 indicates that an exemption request is required to take exception to 10 CFR 50 Appendix J, Option B and allow separate evaluation of the leakage through the Containment Purge System penetrations. This exemption request will be provided separately. In addition, as described in the attachment, the enclosed CD-ROM contains the information requested in Questions 13 and 15.

Should you have any questions related to this information, please contact Mr. Timothy A. Byam at (630) 657-2804.

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

Respectfully,

December 23, 2003
Executed on

Keith R. Jury
Keith R. Jury
Director – Licensing and Regulatory Affairs
AmerGen Energy Company, LLC

Attachment: Additional Information Supporting the Request for License Amendment Related to Application of the Alternative Source Term

Enclosure: CD-ROM Containing χ/Q Input Files and Suppression Pool pH Calculation IP-M-0726, Revision 0

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Clinton Power Station
Office of Nuclear Facility Safety – IEMA Division of Nuclear Safety

ATTACHMENT

Additional Information Supporting the Request for License Amendment Related to Application of the Alternative Source Term

Question 1

On Page 8 of Attachment 2 to the April 3, 2003, submittal, the last paragraph states that the leakage of air from the feedwater isolation valve (FWIV) would be 10.98 cfm for a 1-hour period until the feedwater piping is filled with water. However, Table 4 on Page 18 of 35, states the leakage as 10.98 cfm for "each of two penetrations" from 21.15 minutes to 1-hour, a period of less than 40 minutes. The table on Page 27 of 35 states that the leakage is 10.98 cfm total. Similar confusion exists over the 2 gpm value after 1-hour. Please clarify the appropriate leakage value, onset, and duration. Please confirm that the analyses were performed using the correct values. If the 21.15 minute leakage onset is correct, please provide the basis for this onset timing.

Response 1

The wording associated with the FWIV leak rate in Table 4 is incorrect. The correct description is "FWIV leak rate (Total for Two Penetrations)". The 10.98 cfm gaseous path leak rate corresponds to a liquid leak rate of 2 gpm. For the purposes of this analysis, this total flow rate is conservatively assumed to be through a single feedwater penetration. The statement in Table 4 will be corrected and a revised Table 4 will be provided with the responses to Questions 4, 5, and 12. In addition, the analysis of piping deposition during the Feedwater (FW) piping gaseous path phase is being revised to use the well-mixed modeling assumptions identified in NRC Staff Report AEB-98-03, "Assessment of the Radiological Consequences for the Perry Pilot Plant Application Using the Revised (NUREG-1465) Source Term." The 21.15 minute leakage onset was based on a plug flow model delay time and will no longer be credited. The 10.98 cfm gaseous leak rate will be used for the entire first hour.

Question 2

On page 9 of Attachment 2 to the submittal, the third paragraph states that since a separate dose analysis has been performed for the primary containment purge lines, the leakage from these penetrations no longer need to be considered in determining compliance with the secondary containment bypass leakage or primary containment leakage rate acceptance criteria in technical specifications. This is also shown on Page 18 in Attachment 5. The staff finds this argument to be technically correct but believes 10 CFR Part 50 Appendix J (e.g., III.B.3) requires the leakage from all pathways subject to testing to be summed. Please provide an explanation of how your proposed protocol will meet the requirements of Appendix J.

Response 2

The referenced 10 CFR Part 50 Appendix J, Option B, paragraph states in part that a site is required to "demonstrate that the sum of the leakage rates at accident pressure of Type B tests, and pathway leakage rates from Type C tests, is less than the performance criterion (La) with margin, as specified in the Technical Specification."

As stated in the proposed Alternative Source Term (AST) amendment request (see Reference 1 to this Attachment), Clinton Power Station (CPS) will test and evaluate the leakage of the Containment Purge lines (penetrations 1MC-101 and 1MC-102) by implementation of Technical Specification (TS) Surveillance Requirement (SR) 3.6.1.3.5.

The leakage of the 1MC-101 and 1MC-102 penetrations is subject to a separate dose analysis. CPS will continue to perform Type C leakage testing on 1MC-101 and 1MC-102 and will evaluate the leakage result per the specified 0.02 L_a criteria. This methodology is consistent with that of the CPS Main Steam line penetration leakage, which is also evaluated separately from the overall L_a calculation by virtue of having a separate dose analysis and separate TS surveillance criteria as documented in CPS Updated Safety Analysis Report (USAR) Table 6.2-1. CPS agrees that 10 CFR Part 50 Appendix J, Option B, requires all atmospheric pathway leakage to be summed and evaluated against L_a. Any exception to this requirement would necessitate the CPS Operating License to be amended to reflect a revised criterion. Therefore, AmerGen is preparing an exemption request to take exception to the requirement of 10 CFR 50 Appendix J and allow separate evaluation of the leakage of penetrations 1MC-101 and 1MC-102. This exemption request will be provided separately.

Question 3
On Page 10 of Attachment 2 to the submittal, the control room unfiltered leakage is established at 600 cfm. Please provide an explanation of the basis or derivation of this value. Include in your explanation any testing results that confirm the assumed value.

Response 3
The 600 cfm unfiltered value is not a derived value. The control room dose calculation was performed assuming an unfiltered inleakage of 600 cfm. This value was intended to maximize the inleakage, bound what is expected to be the actual inleakage, and still meet the acceptance criteria for the control room dose. Since the CPS control room is maintained at a positive pressure, it is expected that the actual unfiltered inleakage is bounded by the assumption of 600 cfm. Therefore, the use of 600 cfm was determined to be an allowance that was acceptable for control room personnel protection. AmerGen has not performed testing to determine the actual control room unfiltered inleakage. Unfiltered inleakage will be measured and controlled as described in the AmerGen response to NRC Generic Letter 2003-01, "Control Room Habitability," (see Reference 2 to this attachment). As described in the Generic Letter response, AmerGen plans to perform a test in 2004 using ASTM Standard E741 methodology to verify that the current surveillances for filtered and unfiltered inleakage are conservative and adequate.

Question 6
Section 4.3 of Attachment 2 to the submittal addresses the main steamline break accident analysis. AmerGen has proposed a transport model that is based on thermo-hydraulic rather that the meteorological processes addressed in regulatory guidance. AmerGen's approach appears to maximize the volume of the assumed hemisphere which minimizes its concentration, exposing the control room intake to a lower concentration for a longer period. The description in the submittal doesn't provide sufficient information for the staff to conclude that this is an adequately conservative approach. Please provide the following information:

a. *Whether AmerGen performed a sensitivity analysis to determine if the maximum hemisphere volume yields the highest control room intake? Did AmerGen consider* *heat losses during expansion that could reduce the size of the expanded hemisphere?*

b. *The pressure and temperature of the steam at the point of release (prior to expansion to atmospheric pressure and temperature).*

c. *A clarification of whether "atmospheric pressure and temperature" is to be interpreted as 14.7 psia and the associated saturation temperature. If another temperature or pressure is assumed, please identify the values and their bases.*

d. *The assumption regarding control room intake during the puff transit. For example, a particular flow rate for the duration of the hemisphere movement.*

Response 6

The methodology used to assess the radiological consequences of a Main Steam Line Break (MSLB) accident is the same as that used by Exelon in previous MSLB accident assessments. This approach has been used for Dresden Nuclear Power Station Units 2 and 3 and Quad Cities Nuclear Power Station Units 1 and 2 AST submittals (see Reference 3 to this attachment) and is currently being reviewed by the NRC.

As described in Section 4.3 of Attachment 2 to the original submittal (see Reference 1 to this attachment), the postulated MSLB accident assumes a double-ended break of one main steam line outside the primary containment with displacement of the pipe ends that permits maximum blowdown rates. The mass released following the break includes the amount of steam in the steam line and connecting lines at the time of the break, plus the amount of steam that passes through the valves prior to closure.

The analysis assumes the MSLB accident to be an instantaneous ground level release. Two models are considered for assessment of MSLB accident radiological consequences. One is for assessing control room dose and the other is for assessing offsite consequences.

In the control room model, the released reactor coolant and steam at operating temperature and pressure is conservatively assumed to expand to a hemispheric volume at atmospheric pressure and temperature. No credit is taken for dilution of the steam cloud by the air into which the steam is ejected. Neither the Turbine Building structure nor its ventilation system is assumed to have an effect on the cloud resulting from the MSLB accident. This hemisphere is then assumed to move at a speed of 1 meter per second downwind past the control room intake. No credit is taken for buoyant rise of the steam cloud or for decay in transit. Dilution (i.e., dispersion) of the activity in the plume in transit was also conservatively ignored.

For offsite locations, the buoyant rise of the steam cloud is similarly ignored, and the ground level dispersion is based on the conservative and simplified Regulatory Guide (RG) 1.5, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors," methodology. Use of the RG

1.5 methodology is discussed in further detail in the response to Question 7 below. The "instantaneous release" of the MSLB accident is converted to an equivalent curie release. Since no credit is taken for decay over this release time, or in transit, the calculation accurately models an instantaneous release.

In summary, the following assumptions were used in the control room and offsite dose evaluations for a MSLB accident.

- The release from the break to the environment is assumed to be instantaneous. No holdup in the Turbine Building or dilution by mixing with Turbine Building air volume is credited.

- The steam cloud is assumed to consist solely of the initial steam blowdown and that portion of the liquid reactor coolant release that flashed to steam.

- The released reactor coolant and steam is assumed to expand to a hemispheric volume at atmospheric pressure and temperature consistent with an assumption of no Turbine Building credit.

- This hemisphere is then assumed to move at a speed of 1 meter per second downwind past the control room intake.

- No credit is taken for buoyant rise of the steam cloud or for decay, and dispersion of the activity of the plume was conservatively ignored.

- For offsite locations, the buoyant rise of the steam cloud is similarly ignored, and the ground level dispersion is based on the conservative and simplified Regulatory Guide 1.5 methodology.

The specific information requested in Question 6 is provided below.

a. A sensitivity analysis was not performed since the model maximizes the available concentration and thus yields the highest control room intake. The modeling did not consider heat losses in the expansion.

b. The normal operating reactor vessel steam outlet temperature of 548°F and corresponding pressure of 1028.5 psia were used as the temperature and pressure of the steam prior to expansion to atmospheric temperature and pressure.

c. Atmospheric pressure and temperature were assumed to be 14.7 psia and its associated saturation temperature.

d. The dose assessment modeling conservatively assumed that an unprotected individual is located at the control room intake location for the duration of the accident (i.e., during the time of cloud passage). No protection is provided by the control room envelope, except in terms of the geometry factor for external exposure.

Question 7

Section 4.3 of Attachment 2 to the submittal also states that the meteorological dispersion model of RG 1.5 is used for offsite doses. The methodology of RG 1.5 requires the release rate to be expressed in terms of release rate. Please explain how the release quantity for an instantaneous release has been converted to a release rate.

Response 7

As noted in response to Question 6 above, the "instantaneous release" associated with a MSLB accident is converted to an equivalent curie release. Since no credit is taken for decay over this release time, or in transit, the calculation accurately models an instantaneous release.

The relationship between the release quantity for an instantaneous release and release rate is as follows.

$$\text{Release Rate}\left[\frac{\text{Ci}}{\text{sec}}\right] * \frac{\text{X}}{\text{Q}}\left[\frac{\text{sec}}{\text{m}^3}\right] * \text{Breathing Rate}\left[\frac{\text{m}^3}{\text{sec}}\right] * \text{DCF}\left[\frac{\text{rem}}{\text{Ci inhaled}}\right] * \text{Duration}[\text{sec}]$$

$$= \text{Dose}[\text{rem}] =$$

$$\text{Integrated Release}[\text{Ci}] * \frac{\text{X}}{\text{Q}}\left[\frac{\text{sec}}{\text{m}^3}\right] * \text{Breathing Rate}\left[\frac{\text{m}^3}{\text{sec}}\right] * \text{DCF}\left[\frac{\text{rem}}{\text{Ci inhaled}}\right]$$

where:

DCF = dose conversion factor

Question 8

Section 4.3 of Attachment 2 to the submittal provides three bullet items related to establishing the magnitude of the release activity. Please explain the relationship of the second and third bullet items as they apply to the statement in the first bullet that the activity in the steam cloud is based on the total mass of water released from the break, not just that which flashes to steam. These last two bullets appear to conflict with the first of the three bullets and the break discharge mass entry in Table 8.

Response 8

The three bullet items in Section 4.3 of Attachment 2 to the submittal (Reference 1 to this attachment) attest to the conservatisms in the assessment which, simply stated, is that 100% of the radioiodine carried in both the water and steam becomes airborne and then moves downwind within the hemispherical volume, concentrated within the mass of steam in the cloud. No credit is taken for partitioning of radioiodine remaining in any condensed liquid. Table 8 of Attachment 2 to the submittal (Reference 1 to this attachment) cites a total mass of 96,250 pounds being released. This total mass released is comprised of 42,500 pounds of steam and 53,750 pounds of liquid. The radiological assessment conservatively assumes that the total radioiodine inventory present in the 96,250 pounds released becomes airborne. The assessment assumes that the reactor coolant radioiodine inventory consists of a concentration as evaluated for the following two cases; (1) 0.2 µCi/gm I-131 dose equivalent, and (2) 4 µCi/gm I-131 dose equivalent.

Question 9

In Table 4 of Attachment 2 to the submittal, the emergency core cooling system (ECCS) water component of the FWIV leak rate is reduced at 24 hours. It appears that this assumption is predicated on the RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," assumption that containment leakage may be reduced by 50 percent at 24 hours. As the staff understands the Clinton design, the ECCS water is a forced flow intended to seal the penetration. The staff believes that the pressure associated with this forced flow is that of the ECCS pump discharge pressure, less system pressure drops, and is independent of the containment pressure. Please explain the basis for your assumed reduction at 24 hours.

Response 9

In response to this question, AmerGen completed a review of the design of this system and determined that configurations could occur where ECCS pump discharge pressure controls the FW penetration seal pressure. Based on this determination, it was concluded that the assumption of a 50% reduction in the FWIV leak rate after 24 hours was not appropriate. Therefore, the analysis is being revised to no longer credit a 50% reduction in ECCS water leakage due to containment pressure reduction. This change will be reflected in the revised Table 4 provided in the subsequent submittal providing the responses to Questions 4, 5, and 12. The responses to these questions will provide the results of the revised analysis.

Question 10

In Table 4 of Attachment 2 to the submittal, the ECCS system leakage flash fraction is set at 1.36 percent. Entry 5.5 in Attachment 5 indicates that the value of 1.36 percent is the current design-basis value derived from ORNL-TM-2412. However, the third paragraph on Page 9 in Attachment 2 states that the ECCS system leakage is a <u>new release path</u> for CPS analyzed to comply with RG 1.183. Thus, it would appear that there is no current licensing basis for the flash fraction.

Paragraph 5.5 of Appendix A of RG 1.183 states that the flash fraction should be assumed to be 10 percent, unless a smaller value can be justified on the actual sump pH history and area ventilation rates. Please explain why ORNL-TM-2412 is an acceptable alternative to the guidance in RG 1.1983.

Response 10

The ECCS system leakage path is a new release pathway for CPS. However, the ECCS system leakage flash fraction of 1.36% is the historical design basis value derived from ORNL-TM-2412 in CPS design-basis calculation C-020, "Feedwater Leakage Control System (LCS) Fill Time and Dose Study," Revision 0 and used for ECCS releases resulting from the use of the Feedwater Leakage Control System. The following excerpt (in italics) from calculation C-020 describes the basis for the use of a flash fraction of 1.36%.

Iodine Partition Coefficient (IPC) and POSTDBA Purge Filter Efficiency

The IPC is used to calculate the fraction of the iodine in the suppression pool water that becomes airborne upon reaching the environment. This is modeled in POSTDBA by converting the IPC to a filter efficiency and using this as input for the purge filter efficiency in POSTDBA.

The IPC is defined, at equilibrium, as:

IPC = (concentration in liquid) / (concentration in gas)

The IPC has been calculated as a function of molar concentration, pH, and temperature using ORNL-TM-2412. The molar concentration of iodine, IC_M*, in the suppression pool water is determined from the following expression (1 gram-atom = 1 mole)*

IC_M *= (dissolved iodine mass in gram-atoms)/(sup pool water volume in liters)*

The mass of iodine in the core at shutdown is 182.3 gram-atoms. 50% of this iodine is dissolved in the suppression pool water. Therefore the mass of dissolved iodine is 91.15 gram atoms. The suppression pool volume is 146,400 ft^3 *which is 4.14559E+6 liters.*
Therefore the molar concentration IC_M*, is:*

IC_M *= (91.15 gram-atoms)/4.14559E+6*
IC_M *= 2.1987E-5 moles/liter*

The pH of the suppression pool is 7 (neutral) because no chemicals have not been added to increase the pH to increase iodine retention. The temperature of the suppression pool is assumed to be 80°C (176°F). At a pH of 7.0 and 80°C ORNL-TM-2412 provides the following values for the IPC.

IC_M *= 3.46E-5* *IPC = 59.7*
IC_M *= 1.73E-5* *IPC = 78.5*

Using linear interpolation, the IPC is calculated to be 73.4 at an IC_M *of 2.1987E-5 moles/liter. This means that the ratio of iodine in the air to that in the water is 1/73.4* *which in turn means that 1.3624% of the iodine is released to the air (1/73.4=1.3624E-2). Therefore the iodine released via the suppression pool leakage must be reduced by a factor of 1.3624%. This corresponds to a filter efficiency of 98.6376 (i.e. 1.3624E-2 of the incoming source is transmitted through the filter). This is rounded up to 98.64 and is input as the purge filter efficiency in the POSTDBA case FWLCSSUP.*

The use of a flash fraction of 1.36% was previously reviewed and found to be acceptable by the NRC in CPS Amendment 127 (see Reference 4 to this attachment). This historic treatment is very conservative for the AST analyses because: (1) Only 30 percent of core iodine is released to the suppression pool rather than the 50 percent assumed previously (i.e., calculation C-020); and (2) the suppression pool pH stays above 7.0 throughout the duration of the accident, virtually eliminating the elemental iodine that contributes to iodine flashing. Therefore, the historic 1.36% flash fraction is considered acceptable in accordance with paragraph 5.5 of Appendix A to RG 1.183 for ECCS releases from the Feedwater Leakage Control System as well as potential ECCS leakage into the Auxiliary Building.

Question 11
In Table 5 of Attachment 2 to the submittal, the control room volume is set at 324,000 cubic feet. Final safety analysis report 6.4.2.1 states that the volume is 405,134 cubic feet. Please resolve this discrepancy.

Response 11
As stated in CPS USAR Section 6.4.2.1, the total control room envelope has a volume of 405,134 cubic feet. However, in the dose analysis the control room free air volume is conservatively considered to be 324,000 cubic feet. This value is reflected in USAR Section 15.6.5.5.3. The difference in volume is accounted for in the space occupied by components such as panels, equipment, and ductwork.

Question 13
On Page 5 of Attachment 2 to the submittal, the last bullet states that AmerGen developed new offsite and control room atmospheric dispersion factors. Please provide the following information needed for staff confirmation of these values:

a. *The information, including the joint frequency data file, that was input into the PAVAN code to generate χ/Q values. An electronic copy or a paper copy of the PAVAN input file(s) would be an acceptable approach to providing these data.*

b. *The information, including the meteorological data files, that were used as input into the ARCON96 code to generate χ/Q values for the control room. The meteorological data files should be submitted on electronic media in the format readable by the ARCON96 code. For the remaining data, tabular data or paper copies of the ARCON96 input files would be an acceptable approach to submitting this information.*

Response 13
The requested information is provided as electronic data files on the enclosed CD-ROM. The specific files requested have been provided.

a. The following PAVAN input files are included.

PAVAN Input Files for the following building areas (for selection of the highest resulting χ/Q values):

- Building Area 1: largest vertically projected area of the control building is equal to 2402 m^2
- Building Area 2: smallest vertically projected area of the control building is equal to 1094 m^2
- Building Area 3: projected semi-circle area of the containment dome plus projected containment rectangular area from grade to the bottom of the dome is equal to 2065 m^2

SEL_LL1.inp - Stack to EAB and LPZ, Building Area 1
SEL_LL2.inp - Stack to EAB and LPZ, Building Area 2
SEL_LL3.inp - Stack to EAB and LPZ, Building Area 3

b. The following ARCON96 input files are included.

ARCON96 Input Files (for each of the above building areas and each of the three control room intakes):

STOE_1.RSF - Stack to East Intake, Building Area 1
STOE_2.RSF - Stack to East Intake, Building Area 2
STOE_3.RSF - Stack to East Intake, Building Area 3
STON_1.RSF - Stack to Normal Intake, Building Area 1
STON_2.RSF - Stack to Normal Intake, Building Area 2
STON_3.RSF - Stack to Normal Intake, Building Area 3
STOW_1.RSF - Stack to West Intake, Building Area 1
STOW_2.RSF - Stack to West Intake, Building Area 2
STOW_3.RSF - Stack to West Intake, Building Area 3

ARCON96 Meteorological Files (data provided by ABS Consulting):

Year 2000 Meteorological Data: CL2000ARCONREV1.met
Year 2001 Meteorological Data: CL2001ARCONREV1.met
Year 2002 Meteorological Data: CL2002ARCONrev2.met

Question 14
The licensee stated in its submittal, as part of the loss-of-coolant accident (LOCA) suppression pool pH evaluation, that 4246 pounds of sodium pentaborate is delivered to the suppression pool to ensure that the particulate iodine deposited in the suppression

pool following a LOCA does not re-evolve and become airborne as elemental iodine. The staff is interested in the details of the analysis used for calculating the buffering effect of the standby liquid control system sodium pentaborate solution on the suppression pool pH. Of special interest to the staff are:

a. *Formation of nitric acid in the containment.*

b. *Generation of hydrochloric acid from radiolysis of the containment cables.*

c. *Amounts of Hydriodic Acid and Cesium Hydroxide release from the damaged fuel after a LOCA.*

Response 14

The response to Question 15 below provides the LOCA suppression pool pH calculation. This calculation provides the details of the analysis used for calculating the buffering effect of the Standby Liquid Control System sodium pentaborate solution on the suppression pool pH. The following information is included in this calculation.

a. Formation of nitric acid in the containment (calculated in the Attachment C spreadsheets Column I using the equation in section 6.3 of the main body of the calculation)
b. The generation of hydrochloric acid from radiolysis of the containment cables (calculated in the Attachment C spreadsheets Columns J, K, L, and M using the equations in section 6.4 of the main body of the calculation)
c. The amounts of Hydriodic Acid (calculated in the Attachment C spreadsheets Column H using the equation in section 6.2 of the main body of the calculation) and Cesium Hydroxide (calculated in the Attachment C spreadsheets Column O using the equation in section 6.5 of the main body of the calculation) released from the damaged fuel after a LOCA.

Question 15

In order to complete its evaluation, the staff needs to appraise the general assumptions and methodologies used by the licensee to prove that the suppression pool pH will be maintained above 7 throughout the duration of the accident. Please describe the procedure utilized, including sample calculations, for calculating pH of the suppression pool water during the 30 day period after a LOCA.

Response 15

Included on the enclosed CD-ROM is the full LOCA suppression pool pH calculation, IP-M-0726, "Suppression Pool pH Calculation for Alternative Source Term," Revision 0 in lieu of sample or partial calculations. This calculation provides details on the assumptions and methodologies used to demonstrate that the suppression pool pH will be maintained above 7.0 throughout the 30-day duration of the accident. The methodology used in this calculation is consistent with the methodology reviewed and approved by the NRC for Grand Gulf (see Reference 5 of this attachment).

ATTACHMENT

Additional Information Supporting the Request for License Amendment Related to Application of the Alternative Source Term

References

1. Letter from Michael J. Pacilio (AmerGen Energy Company, LLC) to U. S. NRC, "Request for License Amendment Related to Application of Alternative Source Term," dated April 3, 2003
2. Letter from Michael P. Gallagher (Exelon Generation Company, LLC) to U. S. NRC, "Exelon/AmerGen 180-Day Response to NRC Generic Letter 2003-01, 'Control Room Habitability,'" dated December 9, 2003
3. Letter from Keith R. Jury (Exelon Generation Company, LLC) to U. S. NRC, "Request for License Amendments Related to Application of Alternative Source Term," dated October 10, 2002
4. Letter from U. S. NRC to Mike Reandeau (Illinois Power Company), "Issuance of Amendment – Clinton Power Station, Unit 1 (TAC No. MA3888)," dated April 25, 2000
5. Letter from U. S. NRC to William A. Eaton (Entergy Operations, Inc.) "Grand Gulf Nuclear Station (GGNS), Unit 1 – Issuance of Amendment Re: Full-Scope Implementation of an Alternative Accident Source Term (TAC No. MA8065)," dated March 14, 2001