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10 CFR 50.90

October 10, 2005

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

Limerick Generating Station, Units 1 & 2
Facility Operating License Nos. NPF-39 and NPF-85
NRC Docket Nos. 50-352 and 50-353

Subject: Supplement to the Request for License Amendment Related to Application of Alternative Source Term, dated February 27, 2004

References:
(1) Letter from M. P. Gallagher (Exelon Generation Company, LLC) to US NRC, dated February 27, 2004

(2) Letter from R. J. DeGregorio (Exelon Generation Company, LLC) to US NRC, dated October 25, 2004

(3) Letter from T. Tate (U. S. Nuclear Regulatory Commission) to Christopher M. Crane (Exelon Generation Company, LLC), dated August 18, 2005

In Reference 1, Exelon Generation Company, LLC (Exelon) submitted a request for a change to Appendix A, Technical Specifications (TS), of Facility Operating License Nos. NPF-39 and NPF-85 for Limerick Generating Station (LGS). 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". Exelon provided additional information in Reference (2) to support the U. S. Nuclear Regulatory Commission's (NRC's) review of the proposed change.

In Reference (3), the NRC requested additional information regarding the Limerick AST License Amendment Request (LAR). Attachment 1 to this supplemental letter provides the requested information. As described in Attachment 1, Exelon is submitting revised supporting calculations for the Loss of Coolant Accident Analysis (LOCA), the Fuel Handling Accident (FHA) Analysis, the Control Rod Drop Accident (CRDA) Analysis and the Main Steam Line Break (MSLB) Accident Analysis.

Exelon has developed a set of tables documenting the changes in parameters and methods used in each version of the supporting calculations. These tables are provided as Attachment 2. A summary of the revised analysis results is also included in Attachment 2. The tables provided in Attachment 2 to this letter supersede Tables 3 through 15 provided in Attachment 8 to Reference 1.

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As a result of the revised analyses, Exelon is rescinding the proposed changes to TS pages 3/4 6-53, 3/4 6-55, 3/4 6-56, and 3/4 7-7 for Units 1 & 2. These changes would have revised the allowed methyl iodide penetration value for the laboratory testing of the ESF ventilation systems filters. Additionally, pages 3/4 7-3, 3/4 7-4, and 3/4 7-5, have also been rescinded regarding the Emergency Service Water System and the Ultimate Heat Sink for both Units 1 & 2. The changes were initially implemented to provide consistency with TSTF-51, Revised Containment Requirements During Handling of Irradiated Fuel and Core Alterations, Revision 2; however, they provide no perceived benefit to the station.

In addition, Unit 1 TS Bases pages B 3/4 6-5 and B 3/4 7-1a and Unit 2 TS Bases pages B 3/4 6-5 and B 3/4 7-1 have been revised to reflect the TS page changes and are being re-submitted for your information as Attachment 4. Attachment 4 contains requested plant drawings from the Request for Additional Information questions.

There is no adverse impact to the No Significant Hazards Consideration submitted in the Reference (1) letter. Additional commitments are contained within this letter.

If you have any questions or require additional information, please contact Doug Walker at (610) 765-5726.

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

Respectfully,

10/10/05
Executed On

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

Attachments:

1. Response to Request for Additional Information
2. Limerick Generating Station (LGS) Plant Parameters Tables
3. TS Bases Pages Markups
4. Requested Plant Drawings
5. Additional Commitments

Enclosure: CD-ROM Containing the following Calculations:
LM-0641, Rev 0 (X/Q)
LM-0642, Rev 1 (Suppression Pool pH)
LM-0643, Rev 1 (CRDA)
LM-0644, Rev 1 (MSLB)
LM-0645, Rev 1 (FHA)
LM-0646, Rev 1 (LOCA)

cc: S. J. Collins, Regional Administrator, Region I, USNRC
S. Hansell, USNRC Senior Resident Inspector, LGS
T. Tate, Senior Project Manager [LGS], USNRC
R. R. Janati - Commonwealth of Pennsylvania

bcc: Sr. Vice President, Mid-Atlantic Operations
Vice President, Licensing & Regulatory Affairs
Sr. Vice President, Operations Support
Plant Manager-LGS
Director, Operations-LGS
Director, Engineering
Director, Site Engineering-LGS
Director, Licensing & Regulatory Affairs
Manager, Regulatory Assurance-LGS
Manager, Licensing
Manager, LGS Nuclear Oversight
PA DEP BRP Inspector - LGS, SSB2-4
Commitment Coordinator - KSA 3-E
Correspondence Control Desk - KSA 1-N-1
Records Management - KSA 1-N-1

ATTACHMENT 1

LIMERICK GENERATING STATION
UNITS 1 AND 2

Docket Nos. 50-352
50-353

License Nos. NPF-39
NPF-85

Supplement to License Amendment Request for
"LGS Alternative Source Term Implementation"

Response to Request for Additional Information

ATTACHMENT 1

REQUEST FOR ADDITIONAL INFORMATION

REGARDING PROPOSED LICENSE AMENDMENT REQUEST

FOR IMPLEMENTATION OF ALTERNATIVE SOURCE TERM (AST)

LIMERICK GENERATING STATION, UNITS 1 AND 2

DOCKET NOS. 50-352 AND 50-353

The following questions apply to both units unless otherwise noted. References to attachments of the cover letter refer to your February 27, 2004, application.

1. Please provide all design-basis accident calculations, including all design-basis parameters, assumptions, or methodologies, that were changed in the radiological design-basis accident analyses as a result of the proposed change. If there are many changes, it would be helpful to compare and contrast them in a table. Also, please provide a justification for any changes.

   EXELON RESPONSE
   All radiological dose calculations regarding AST are being provided in this response in an Enclosure to this letter. Calculations and their subject matter include:

   LM-0641, Rev 0 (X/Q)
   LM-0642, Rev 1 (Suppression Pool pH)
   LM-0643, Rev 1 (CRDA)
   LM-0644, Rev 1 (MSLB)
   LM-0645, Rev 1 (FHA)
   LM-0646, Rev 1 (LOCA)

   Parameter Tables are also provided as Attachment 2. These tables include values for the current licensing basis, values as submitted in the original LAR, and final supplemented AST values. Justification for all changes is provided in the parameter tables.

2. Attachment 8 to the application, Table 11b, does not appear to include a leakage pathway currently in the LGS design basis. Per the LGS Updated Final Safety Analysis Report (UFSAR), Section 15.6.5.5.1.2, "Fission Product Transport to the Environment", states that:

   "In accordance with this guidance, and as explained in Section 6.5.3, the LGS evaluation assumes that the mechanisms discussed above will ensure the assumed 50% mixing within the large reactor enclosure at all times during the period when the reactor enclosure pressure is above minus ¼ inch, as well as when it is below. However, it will also be conservatively assumed that there is unfiltered exfiltration at 2500 cfm, in addition to the SGTS exhaust, during periods when the pressure is above minus ¼ inch wg."

   Please include this pathway or provide adequate justification for not including it.

EXELON RESPONSE

The LGS Standby Gas Treatment System (SGTS) is an engineered safety feature system whose primary safety related function is to isolate and drawdown the secondary containment(s), filter the airborne iodine and aerosol activities prior to discharge to the environment, and maintain a negative pressure of at least ¼ inches W.G. following a Design Basis Accident (DBA).

The LGS secondary containment consists of three distinct isolation zones. Zones I and II are the Unit 1 & 2 reactor enclosures respectively, and Zone III is the common refueling area. The SGTS is designed to isolate all 3 zones simultaneously or any combination of the zones.

Drawdown time is defined as the time it takes for the SGTS to restore a negative pressure of at least ¼ inch W.G. in the affected reactor enclosure secondary containment isolation(s) zone following receipt of an isolation signal. At Limerick, the drawdown time is 15.5 minutes. The existing Exelon DBA-LOCA Dose analysis conservatively assumes that following a DBA at time equals 0 seconds, total fuel damage occurs and all leakage from the primary containment goes to, and mixes in 50% of the reactor enclosure secondary containment volume. Additionally, no credit is taken for any SGTS filtering of the secondary containment release during the entire drawdown period, and secondary containment via SGTS is assumed to exhaust 3000 cfm during the drawdown period. After the reactor enclosure secondary containment negative pressure has been re-established to at least ¼ inch W.G., the SGTS will maintain that pressure at a flow rate not exceeding 2500 cfm per Reactor Enclosure zone.

Table 15.6-13 of the LGS UFSAR specifies the design basis assumptions and pathway utilized for the Limerick LOCA analysis. This table clearly indicates that during the drawdown period a flow rate of 3000 cfm is used and after drawdown, a flow rate of 2500 cfm is used for the remaining period of the LOCA.

The pathway as specified in UFSAR Section 15.6.5.5.1.2 is an erroneous description of the SGTS and is not the correct design basis pathway nor is it supported by any design or licensing basis analysis. A review of the existing Exelon DBA-LOCA Dose analysis showed that this analysis assumed an unfiltered flow rate of 3000 cfm during the entire 15.5 minute drawdown period and a filtered flow rate of 2500 cfm for the remaining event duration which is consistent with UFSAR Table 15.6-13 and is the design and licensing bases.

The Table 15.6-13 design basis assumptions were developed in support of a Unit 1 & 2 Technical Specification change in 1997 to change the drawdown time and maximum allowable leakage rate following a DBA. A review of the NRC Safety Evaluation Report for these licensing amendments 122 (Unit 1) and 86 (Unit 2) implies that the Table 15.6-13 design basis assumptions were used for their assessment.

This issue has been captured in the Exelon Corrective Action Program and a UFSAR change will be prepared to revise Section 15.6.5.5.1.2 to reflect the appropriate design basis description.

3. Attachment 1, page 42 of 76, Table A, indicates that the LGS analysis conforms to a list of sections within Regulatory Guide (RG) 1.183, including RG 1.183, Section 4.1.2. A review of the proposed changes indicates that strict conformance with the RG for this section does not appear to be correct. The RG states:

   "4.1.2 The exposure-to-CEDE factors for inhalation of radioactive material should be derived from the data provided in ICRP Publication 30, 'Limits for Intakes of Radionuclides by Workers' (Ref. 19). Table 2.1 of Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion" (Ref. 20), provides tables of conversion factors acceptable to the NRC [Nuclear Regulatory Commission] staff. The factors in the column headed "effective" yield doses corresponding to the CEDE."

   Your application references the report: K. F. Eckerman, et al., "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," Federal Guidance Report 11, EPA-520/1-88-020, Environmental Protection Agency, 1988 (Reference 20).

   The licensee's proposed new definition is:

   "DOSE EQUIVALENT I-131 shall be that concentration of I-131, microcuries per gram, which alone would produce the same inhalation committed effective dose equivalent (CEDE) as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The inhalation committed effective dose equivalent (CEDE) conversion factors used for this calculation shall be those listed in Table 2.1 of Federal Guidelines Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," ORNL [Oak Ridge National Laboratory], 1989, as described in Regulatory Guide 1.183. The factors in the column headed "effective" yield doses corresponding to the CEDE."

   Please explain how the proposed new definition conforms to the RG if the 1989 ORNL report is different from Reference 20 in the RG. Please provide the values used and the justification.

   EXELON RESPONSE

   Exelon has confirmed that ORNL, 1989 and FGR-11 are the same document with two different names. Since there are no differences, there is no need to provide justification for values used. Note that FGR-11 and FGR-12 were issued in 1988 and shown in the ORNL distribution to have been corrected (specifics not discussed) in 1989.

4. Attachment 1, page 57, Section 6.1, states that the activity released through the main steam isolation valves (MSIVs) is the same concentration as that used for evaluating primary to secondary containment leakage. RG 1.183, Section 6.1, states that the leakage should be assumed to be that activity determined to be in the drywell. These assumptions appear to be inconsistent. Previously, when using the Technical Information Document (TID) source term an assumption that the mixing between the drywell and wetwell is instantaneous and not mechanistically modeled may have been found acceptable. Using the AST non-mechanistic modeling is likely not to be found acceptable.

The staff does not believe the explanation provided in the comments section of page 57 are compatible with the timing assumptions modeled with the AST. Please provide information sufficient to model the time dependent activity used as a source term for the MSIV leakage.

a. Explain whether the free space in the suppression pool is used to dilute this activity. If so, provide justification for using this volume and also provide the drywell-to-suppression pool free space flow rates versus time and the basis for the flow rate used.

EXELON RESPONSE

As noted in Revision 1 to LOCA Calculation LM-0646, dilution of activity by the free space in the suppression pool is only credited after the first two hours of the accident, with no such credit prior to this time. Only the minimum suppression pool free air space volume is utilized. The large steam flow out of the reactor vessel, at the assumed two-hour ECCS restoration time, is caused by the injection of colder ECCS water onto the hot surfaces within the vessel, which then flashes to steam. The resulting steam flow to the drywell will serve to mix with and rapidly displace activity accumulated in the drywell, through the suppression pool, and into the suppression chamber air space. These phenomena provide for the assumed instantaneous mixing at 2 hours. Drywell to wetwell vacuum breakers are available and will open periodically to aid in the air return. No credit is taken for suppression pool scrubbing. This is the same mechanism previously approved in SERs for Fermi (ref. accession #ML042430179) and Vermont Yankee (ref. accession # ML041280490). This methodology is consistent with RG 1.183 Appendix A Sections 3.7 and 6.1.

b. If any drywell-to-wetwell flow is based on the results of thermal-hydraulic analyses performed for the duration of the release, provide a summary of the analyses for staff review, or

EXELON RESPONSE

No specific thermal-hydraulic analyses were performed.

c. Provide justification for this assumption for the duration of the release.

EXELON RESPONSE

The release from the core to the drywell is terminated by the restoration of ECCS flow at 2 hours. The large steam flow out of the reactor vessel, at the assumed two-hour ECCS restoration time, is caused by the injection of colder ECCS water onto the hot surfaces within the vessel, which then flashes to steam. The resulting steam flow to the drywell will serve to mix with and rapidly displace activity accumulated in the drywell, through the suppression pool, and into the suppression chamber air space. These phenomena provide for the assumed instantaneous mixing at 2 hours. Drywell to wetwell vacuum breakers are available and will open periodically to aid in the air return. No credit is taken for suppression pool scrubbing. This is the same mechanism previously approved in SERs for Fermi (ref. accession #ML042430179)

and Vermont Yankee (ref. accession # ML041280490). This methodology is consistent with RG 1.183 Appendix A Sections 3.7 and 6.1.

5. Attachment 1, page 25, Section 4.4.3, states that the suppression pool pH is maintained greater than 7. Page 37, Table 15, states that the initial suppression pool pH is 5.3 and that the standby liquid control (SLC) injection is assumed to occur within 13 hours. Please justify how, with an initial pH at 5.3 and SLC initiation at 13 hours, the suppression pool pH is greater than 7 throughout the 30-day accident.

EXELON RESPONSE

The suppression pool pH during the LOCA is derived in enclosed Calculation LM-0642 using methodology used and approved for Grand Gulf (ref. accession #ML010780172). The methodology determined that the pH increases, because the released elemental cesium core activity is roughly 10 times that of iodine, thus quickly driving the initial pH above 7 during the gap release. The initial suppression pool pH value of 5.3 is based upon the lower bounding value as described in UFSAR section 6.1.1.2. During the gap release and early in-vessel phases of the first 2 hours, the core cesium and iodine releases are the primary components driving the pH. The calculation methodology explicitly derives both the production of the strong base CsOH and the strong hydriodic acid (HI) produced from the released core iodine.

The effects of radiolysis of the water in the suppression pool and the cable jacketing in containment produce additional acids that tend to lower the pH over time. The calculation determined 13 hours to be the time at which the pH would fall below 7 after the initial rise from the CsOH. This provided the time limit for introduction of the Sodium Pentaborate solution to maintain the pH above 7 for the remainder of the event.

6. Attachment 1, page 11 of 76, gives conflicting information. It states that the LGS post-LOCA (loss-of-coolant accident) direct-shine dose from the Unit 1 14-inch diameter core spray pipe can be managed using administrative controls within the 0.22 rem. Attachment 1, Page 12 of 76 states that "other sources such as reactor enclosure airborne and external cloud and RERS [Reactor Enclosure Recirculation System], SGTS [Standby Gas Treatment System], and CREFAS [Control Room Emergency Fresh Air System] filters are negligible because of shielding, distance or both." Please provide the assumptions, methods, inputs for these analyses, a quantified value for what is considered negligible, and the results of the shielding analyses.

EXELON RESPONSE

The use of administrative controls for access to portions of the control room is no longer being considered for this revision of the LOCA calculation LM-0646. The assumptions, methods, inputs, and results of the shielding analyses for direct shine dose to the control room are provided in Attachment C of the LOCA calculation. In general, if a dose contributor is not expected to impact the overall dose analysis in a perceptible manner (in the second decimal place or about 0.5% of the regulatory limit), then it is considered negligible. At this level such minor contributors are well within the conservatism used in the analyzed dose.

7. In Attachment 6, page 1, Exelon makes a commitment to NUMARC 93-01, Revision 3, Section 11.3.6.5, rather than Section 11.2.6, as specified in technical specification task force (TSTF)-51, Revision 2. Please provide a justification for why Section 11.3.6.5 is a valid substitution for the section stated within the TSTF.

EXELON RESPONSE

TSTF-51, Revision 2 cites Section 11.2.6 of draft Revision 3 of the NUMARC 93-01 document. The Revision 3, July 2000 NUMARC 93-01 document does not have a section 11.2.6. Therefore, Section 11.3.6.5 was used since the section title in TSTF-51 matches that in the July 2000 NUMARC 93-01 Revision 3 document.

8. Attachment 1, page 12, Section 4.3.1, states that the releases for the radiological consequences analyses are evaluated at full-power conditions. Please confirm that full-power conditions are most limiting or provide justification for why other conditions were not evaluated to determine the most-limiting release conditions.

EXELON RESPONSE

The fuel damage is maximized for the CRDA at low power, but full power isotopics are used for conservatism. For the FHA, conservatively, the assemblies are assumed to have operated at full power prior to shutdown with a radial peaking factor of 1.7 applied to the dropped and struck assemblies. Additionally, the source term considers isotopic activities at 100 Effective Full Power Days (EFPD) and 711 EFPD. The isotope activities used in the analysis are the higher of the values of either of the two time periods. This yields a composite source term that is applicable to all periods of core life. The historically generated full power steam line blowdowns before isolation remain for the MSLB licensing basis. However, a bounding value of 140,000 lbs reactor coolant discharge is used in the MSLB analysis for conservatism. For the LOCA analysis, only full-power conditions are considered, as this is the most limiting release condition for a LOCA.

9. Appendix B to Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, establishes quality assurance requirements for the design, construction, and operation of those structure, system, and components (SSCs) that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. Appendix B, Criterion III, "Design Control," requires that design control measures be provided for verifying or checking the adequacy of a design. Appendix B, Criterion XVI, "Corrective Action," requires measures to be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, defective material and equipment, and nonconformances are promptly identified and corrected. Generic Letter (GL) 2003-01, "Control Room Habitability," addresses current issues with respect to previously assumed values of unfiltered inleakage. Generally, these issues can only be resolved by inleakage testing.

Exelon requested a change in the design basis of the control room heating, ventilation, and air conditioning (HVAC) system. This request no longer takes credit for the automatic initiation of the radiation isolation mode. With no credit for this initiation, during the initial 30 minutes of the accident, the control room HVAC operates in the normal mode rather than in the radiation isolation mode. The licensee assumed that in this mode 525 cfm of unfiltered inleakage in addition to the normal 2100 cfm of unfiltered

inleakage is transferred into the control room. According to the LGS response to GL 2003-01, this mode of operation does not appear to have been tested for inleakage. In light of the Appendix B requirements and GL 2003-01, provide justification to explain why the value assumed for the control room's unfiltered inleakage is appropriate. Please provide details regarding control room, design, maintenance and assessments, or tests to justify the use of this number. Please note that because of the high percentage of control rooms that have historically been unable to successfully predict the amount of unfiltered inleakage, the staff will generally only accept a measured value.

EXELON RESPONSE

Exelon will no longer pursue a change in the design basis of the control room HVAC system to remove the Control Room Emergency Fresh Air System (CREFAS) automatic start feature in the radiation isolation mode from the Unit 1 & 2 Technical Specifications and replace it with a manual start. The removal of the automatic isolation feature from TS provided additional flexibility to address potential system issues. Upon further review of the Generic Letter 2003-01 testing requirements, LGS has decided to rescind this request. The control room HVAC system will remain as originally designed and licensed: automatic isolation of the control room and initiation of the CREFAS immediately upon receipt of a radiation isolation signal to maintain control room habitability.

While in the radiation isolation mode of operation, the CREFAS system is designed to maintain the control room at a positive pressure of at least 1/8 inch W.G. relative to its surrounding areas with a filtered outside air flow rate of less than or equal to 525 cfm. Verification and testing of the CREFAS system operation in the radiation isolation mode was performed as required by GL 2003-01. Tracer gas test results determined that the filtered outside air flow rate was less than 525 cfm and the unfiltered inleakage into the control room was less than 100 cfm for both trains of CREFAS. The tracer gas test results were formally submitted to the NRC in a letter dated December 10, 2004 (ref. accession #ML043510210).

10. Starting on page 17 of Attachment 1 of the application, Exelon describes the methodology used to calculate the leakage from the primary containment into the main steam lines. At upstream conditions, the flow rate out of the MSIVs is adjusted by the MSIV surveillance pressures. This method does not appear to consider the accident conditions in the drywell. Methods acceptable for calculating the accident pressures in the drywell typically use the design pressure for this calculation. Please justify the methodology proposed.

EXELON RESPONSE

Section 4.5.3.1 of LOCA calculation LM-0646 describes the consideration of accident conditions in the drywell on MSIV flow rates. The maximum drywell temperature after 2 minutes was incorporated into this flow rate determination.

The method took the leakage acceptance criterion and converted it to an MSIV leakage rate in cubic feet per hour (cfh) by the formulation shown below.

MSIV leakage assumed in this accident analysis is 200 scfh total for all steam lines and 100 scfh for any one line.

The leak rate and inboard piping flow rate associated with a 100 scfh Leak Rate Acceptance Criterion is:

Leak Rate Acceptance Criterion (scfh) $* [14.7 / (P_{MSIVtest} + 14.7)] * (276+460) / (68+460)$

Where:

$P_{MSIVtest}$ = 22 psig MSIV Test Pressure (It is more conservative to use Test Pressure in this equation than Peak Drywell Pressure or P_a)
276 °F is the peak drywell temperature at 2 minutes, per Attachment E of calc. LM-0646.

Associated outboard piping flow rates are:

Leak Rate Acceptance Criterion (scfh) $* (550+460) / (68+460)$

Where:

550 °F is the normal steam line operating temperature.
Credit is taken for temperature reductions to 410 °F at 24 hours, and to 200 °F at 96 hours in determining later flow rates.

Flow rates out of the condenser are similarly calculated with the assumption of a condenser air space temperature of 120 °F for the accident duration.

In RADTRAD analyses, the containment volume is input as the combined volume of the drywell and suppression chamber air space. Since only the drywell is credited for the first two hours, the leak rates from containment to the two steam lines are increased by the ratio of (the drywell plus suppression pool volume) / (the drywell volume).

11. Page 18 of Attachment 1 states that the inboard steamline, outboard steamline and condenser effective filter efficiencies are calculated using AEB 98-03 formulations and settling and deposition velocities. The discussion and the data provided are insufficient to support an NRC staff confirmation. Please provide the following information.

a. On page 19 of Attachment 1 of your submittal, you state that your submittal is based upon the methodology used in AEB 98-03.

If the analysis did not use the entire methodology, please describe differences between the model used and the AEB 98-03 model. Please provide justification for any differences between the two models.

EXELON RESPONSE

Section 4.5.3 of calculation LM-0646 describes the consideration of effective filter efficiencies for iodine deposition for the MSIV pathway. A parameter table (Attachment 2) is also provided to indicate features of AEB-98-03 that were used, those that were not used, and the bases in either case. The spreadsheet implementing the model is also submitted and can be found in Calculation LM-0646, Attachment B.

b. A single-line sketch of the four main steamlines and the isolation valves. Annotate this sketch to identify each of the control volumes assumed by Exelon in the deposition model.

EXELON RESPONSE

The requested figure is provided as Figure 4.2 of LOCA Calculation LM-0646.

c. A tabulation of all of the parameters input into the proposed AEB 98-03 model for each control volume shown in the sketch (and time step) for which you are crediting deposition. This includes:

- Flow rate
- Gas pressure
- Gas temperature
- Volume
- Inner surface area
- Total pipe bend angle

Note: Attachment 8, Table 4, provides some of this information but neither the paper nor the electronic copy of this file is legible.

EXELON RESPONSE

The applicable portions of the request are provided in Attachment B of calculation LM-0646, as amplified in section 4.5.3 of the calculation.

A table of AEB 98-03 parameters is provided as Table 7 of Attachment 2 to this letter, except for total pipe bend angle. The total pipe bend angle is a RADTRAD Brockmann model parameter that is not used in AEB-98-03.

d. For each of the bulleted parameters in this question, please provide a brief derivation and an explanation of why that assumption is conservative for a design basis calculation. Address changes in parameters over time, e.g., plant cooldown.

EXELON RESPONSE

A table of AEB 98-03 parameters is provided in Attachment 2 of this letter.

Attachment B, Tables 2.1 and 2.2, and Section 4.5.3 of Calculation LM-0646 provides the consideration of the bases and the conservatisms in the assumptions for iodine deposition for the MSIV pathway.

e. Page 46 of Attachment 1, Table A, indicates that the Exelon analysis conforms to RG 1.183, Section 5.1.2. Please clarify whether your analysis addresses a single failure of one of the MSIVs. Such a failure could change the control volume parameters that are input in the deposition model. Previous implementations of main steam deposition have been found acceptable only if the licensee modeled a limiting single failure. Please confirm that the limiting MSIV single failure has been modeled and describe which failure was utilized and justify why this is the limiting failure.

EXELON RESPONSE

A failure of the outboard MSIV is assumed in order to maximize the flow rate in penetration piping to minimize deposition. In contrast, a single failure of an inboard MSIV does not impact the availability of deposition in penetration piping.

Calculation LM-0646 Attachment B, Table 2.2 and Section 4.5.3 of this calculation provide this consideration of the limiting single failure for iodine deposition for the MSIV pathway.

f. Since crediting the main steamline deposition effectively establishes the main steam piping as a fission product mitigation system, the staff expects the piping to meet the requirements of an engineered safety feature (ESF) system, including seismic and single failure considerations. Please confirm that the main steam piping, condenser and the isolation valves that establish the control volumes for the modeling of deposition were designed and constructed to maintain integrity in the event of the safe shutdown basis earthquake for LGS. If the design basis for the piping and components does not include integrity during earthquakes, please provide an explanation of how the LGS design satisfies the prerequisites of the staff-approved NEDC-31858P-A, "BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems."

   If piping systems and components at LGS were previously found by the staff to be seismically qualified using the methodology of this Boiling-Water Reactor Owners Group (BWROG) report, please provide a specific reference to the staff's approval.

   EXELON RESPONSE

   The piping systems and components at LGS credited for iodine deposition for the MSIV pathway were previously found by the NRC to be seismically rugged to withstand a Safe Shutdown Earthquake (SSE) using the methodology of this Boiling Water Reactor Owners Group (BWROG) report, as referenced in the LGS UFSAR Section 6.7 on the MSIV Leakage Alternate Drain Pathway. The related license amendments approving this design and addressing seismic ruggedness review for credited components are: LGS Unit 1 Amendment 107 (Adams Accession No. ML0115602140), and LGS Unit 2 Amendment 53 (Adams Accession No. ML0115607070).

g. Page 19 of Attachment 1 states, "For aerosol settling, only horizontal piping runs are credited, and only the bottom surface area is considered available." If only horizontal piping runs are credited, please justify using the surface area of the bottom half of the pipe for aerosol deposition when the cross-sectional edges of this piping are essentially vertical or inclined.

EXELON RESPONSE

In the current revision of the LOCA calculation (LM-0646), the submittal has been conservatively revised to consider the projected horizontal area of the piping which is calculated to be the pipe diameter times the length of the pipe. For aerosol settling, only horizontal pipe lengths were considered.

h. Page 10 of Attachment 8 states, "For the two bounding steamlines modeled, two nodes are used." Please specify which two steamlines are bounding and specify how they were chosen and why they are bounding.

EXELON RESPONSE

The bounding lines are the two main steam lines (A & B) that demonstrate the lowest iodine removal efficiency (based on the NRC's AEB-98-03 nodalized deposition model that was applied to this analysis). Typically, the lowest iodine removal efficiency can simply be correlated to the main steam lines of shortest length. However, due to flow expansion in the node modeled as outboard, piping in the respective outboard nodes becomes less efficient in removing iodine.

It was initially (Rev. 0 of LM-0646) found that the two lines, whose nodes combine for the lowest iodine removal efficiencies, were the two bounding lines (Lines A & D for LGS). However, for the revised analysis it is conservatively, non-mechanistically, and artificially assumed that the postulated LOCA is initiated by an inboard main steam line break. Further, it is conservatively assumed that this break results in the complete loss of iodine removal credit associated with the inboard piping node of the affected steam line. It is found that the worst-case such break is that of the line whose remaining credited penetration and outboard pipe combines for the least efficient iodine removal; for LGS this line is MSL B. Therefore, for the revised LGS LOCA analysis (LM-0646, Rev. 1), MSL B will replace the second worst overall steam line, MSL D, when this artificial MSLB LOCA is considered.

i. Table B contained on page 53 of Attachment 1 states that a previous analysis based upon Technical Information Document (TID)-14844-based source terms assumed a recirculation line break. The design loss-of-coolant accident (LOCA) analyses are required by regulation to consider a spectrum of break locations and break sizes. Proposals to credit deposition in the main steamlines need to consider the impact of the break location on steamline deposition. In light of crediting this deposition, please justify why a break of a main steamline is not considered and why the recirculation line remains bounding or consider the break in the most-limiting reactor coolant system location. Note that, although thermodynamic analyses may show that significant core damage is unlikely for a reactor coolant system break in the steamline, a LOCA involving a recirculation piping break is similarly unlikely to cause significant core damage. Nonetheless, the regulatory guidance for a design-basis LOCA assumes a substantial release of fission products as a means of assessing the ability of the containment design to mitigate the consequences of a LOCA in the unlikely event the emergency core cooling system (ECCS) should fail. As such, the break location and size are not determinants for the amount of fuel damage assumed to occur in the stylized design-basis analysis.

EXELON RESPONSE

A steam line break inside containment is now assumed, so that the broken line's volume and areas are not credited for aerosol settling or elemental iodine deposition. The re-analysis (calculation LM-0646) now includes the broken steam line as the least favorable line for deposition in the remaining pipe, thereby maximizing the dose.

j. Page 20 of Attachment 1 states, "Iodine resuspension from settled or deposited iodines is not calculated. Historically, this phenomena increased the organic iodine release by about a factor of two based on resuspension of TID-14844-based elemental iodine fractions. The presence of this phenomenon is questionable with aerosols with significant cesium loadings. Furthermore, while deposition on condenser tubing is not formally credited, test cases have shown that substantial removal of elemental and even organic iodine would be predicted that would more than offset any resuspension. Flow rates out of the condenser are assumed to be at 120 degrees F and atmospheric pressure. A factor of 1.25 is applied, as is done with leakage and flow-through steamlines. This leak rate is also reduced by 50% after 24 hours, consistent with the change in containment conditions."

The staff believes that the above information does not provide adequate justification for changing the historical basis for organic iodine resuspension. Please provide additional information to justify not utilizing the historical resuspension, including the mechanics for changing the current methodology. As a minimum, the information previously used to determine the factor of two should be examined, and LGS should provide a complete assessment of why the previous assessment is no longer applicable.

If reliance on the condenser tubing is being used to offset the change in methodology, then provide a justification that this is conservative. Please confirm that the condenser piping credited is designed and constructed to maintain integrity in the event of the safe shutdown basis earthquake for LGS. If the design basis for the piping and components does not include integrity during earthquakes, please provide an explanation of how the LGS design satisfies the prerequisites of the staff-approved NEDC-31858P-A, "BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems." If the condenser piping systems and components at LGS were previously found by the NRC staff to be seismically qualified using the methodology of this BWROG report, please provide a specific reference to the staff's approval.

EXELON RESPONSE

Organic Iodine Resuspension is now considered for this revision of the LOCA calculation (LM-0646, section 4.5.3.2.2), using the J. E. Cline methodology as shown in the LOCA calculation, Attachment B spreadsheet and Attachment A Resuspension RADTRAD runs.

Exelon does not rely on aerosol settling or elemental iodine deposition on condenser tubing, although LM-0646 does test the value of such credit as a measure of conservatism. Test cases have shown that substantial removal of elemental and even organic iodine would be predicted that would more than offset any

resuspension. These results are shown in Attachment A of the revised LGS LOCA analysis, LM-0646. As shown in Attachment B of the same analysis, the condenser tubing provides a surface area that is about 40 times that of the credited wall and bottom surface areas. No credible seismic failure would render these surfaces unavailable. It should also be noted that the HP, IP, and LP condensers are interconnected by substantial openings, but flow to the IP and LP condensers for further holdup is not credited.

The piping systems and components at LGS credited for iodine deposition for the MSIV pathway were previously found by the NRC to be seismically rugged to withstand a Safe Shutdown Earthquake (SSE) using the methodology of this Boiling Water Reactor Owners Group (BWROG) report, as referenced in the LGS UFSAR Section 6.7 on the MSIV Leakage Alternate Drain Pathway. The related license amendments approving this design and addressing seismic ruggedness review for credited components are: LGS Unit 1 Amendment 107 (Adams Accession No. ML0115602140), and LGS Unit 2 Amendment 53 (Adams Accession No. ML0115607070).

12. Page 46 of Attachment 1, Table A, states that the Exelon analysis conforms to Regulatory Position 5.1.2. For each design-basis accident analyzed please provide:

a. The single active component failure that results in the most-limiting radiological consequences and justify why it is the most-limiting.

EXELON RESPONSE

Most single failures do not impact the consequences of design basis accidents because credited systems are single failure proof. For each accident the specific single failure impacts of note are:

LOCA: No single failures have been identified that impact the releases or mitigation systems or analysis for the pathways dealing with primary containment leakage, ECCS leakage release paths, or external dose assessment. The MSIV leakage pathway has been determined to be impacted by the selected MSIV failure. The worst case failure is the failure of an outboard MSIV because it minimized deposition in penetration piping as discussed in LM-0646.

FHA: This event credits no safety systems in the determination of releases and consequences. Therefore, single failures have no adverse effects.

MSLB: This event only credits the impact of MSIV closure to terminate the reactor coolant releases. Since the MSIVs are redundant, the release is not impacted by a single failure.

CRDA: The event scenarios for the CRDA are the design basis scenarios of the UFSAR of a 1% per day condenser leak or operation of the Steam Jet Air Ejectors (UFSAR section 15.4.9). These event scenarios remain in accordance with NRC approved guidance in NEDO 31400A. These scenarios take no credit for safety systems and no single failures were considered.

b. The assumptions regarding the occurrence and timing of a loss of offsite power (LOOP) and justify why it provides the most limiting radiological consequences.

EXELON RESPONSE

No credit is taken for the loss of offsite power if it is beneficial. No accident was determined to be aggravated by the presence of offsite power. Specifically, for each accident:

LOCA: The availability of offsite power would not adversely impact the postulated release to containment, the release mechanism from containment, and the behavior of credited mitigative systems.

FHA: The availability of offsite power would not adversely impact the release from the spent fuel pool or releases from the refuel floor airspace. Control room dose modeling with an artificial one air change per minute envelopes all possible control room ventilation and related power supply conditions.

MSLB: The analysis of this event does not take credit for the presence of offsite power or the lack thereof. The sole plant feature credited is the MSIV closure instrumentation, whose loss would result in valve closure. Again, control room dose modeling with an artificial one air change per minute envelopes all possible control room ventilation and related power supply conditions.

CRDA: Two CRDA scenarios are addressed. The path associated with the 1%/day release from the condenser is passive and requires no power. The path through the offgas system assumes offsite power availability. Loss of offsite power for this path would either revert to the first path, or would trap release activity in the charcoal delay beds, providing additional uncredited delay.

13. Based on information provided in the application, the licensee has assumed an MSIV leakage rate of 0.668 cfm for the 100 scfh lines (0.834 cfm with 25% design-margin included). The leakage rate is reduced after 24 and 96 hours based upon changing steamline temperatures. When the proposed MSIV leakage, in scfh, at test conditions (typically 70 degrees and 25 psig) are scaled to peak drywell pressure and temperature (typically 40-50 psig and about 340 degrees) the TS leakage past the inboard MSIV has been shown to be 2.0 cfm, about double the value assumed. However, the temperature of the fluid in the steamlines is based on the steam piping temperatures, typically 500-600 degrees (558 degrees F for 0-24 hours for LGS). At the steam piping conditions, the flow is even higher. Likewise a pressure gradient will exist from the first closed MSIV to the end of the last deposition. The gradient would depend on the actual leakage through each MSIV. As such, the deposition nodes downstream of the first MSIV conservatively may be assumed to be at atmospheric pressure. Therefore, these flow rates would be even higher. While the trend of increasing flowrates is reflected in Table 4 of the submittal (Attachment 8, page 12), the absolute values calculated by the licensee are smaller than expected when compensating for the changes in temperature and pressure. The equation provided in Attachment 8, page 9, does not adequately compensate for the leakages in the steamline nodes. Likewise, the arbitrary 25% design-margin added, while conservative, does not compensate for the expected flow rates.

a. Please provide the methodology used to calculate the flow rates in each steamline node and the parameters used. Justify how these parameters conservatively model the changing conditions in the steamline or provide calculations that conservatively account for these steamline-condition changes.

b. Attachment 1, page 18, states, "However, to provide design-margin, the above leak rate is increased by 25% for the first 24 hours to a value of 0.834 cfm. This margin also allows MSIV leakage to be reduced by 50% at 24 hours." Please explain how the design-margin allows the MSIV leakage to be reduced by 50% at 24 hours.

c. Page 11 of Attachment 8 provides a generic assessment of the steamline temperatures following a LOCA. Please provide justification for why this generic assessment is applicable and conservative for LGS. Provide references 28 and 29 from the amendment request.

EXELON RESPONSE

LOCA calculation LM-0646 Section 4.5.3.1 and Attachment B of the calculation describes the considerations on MSIV flow rates. Calculation Attachments E and F provide the justifications for MSIV leakage reduction (by less than 50%, for this revision) and steam line temperature reductions with time, on a conservative basis.

The method used was to take the leakage acceptance criterion and convert it to a containment leakage rate by:

The radioactivity associated with all MSIV leakage is assumed to be released directly from the Primary Containment and into the Main Steam Lines. MSIV leakage has separate limits and a separately analyzed dose assessment; therefore, it is not included in the L_a fraction limit, and is instead separately controlled.

MSIV leakage assumed in this accident analysis is 200 scfh total for all steam lines and 100 scfh for any one line.

The leak rate and inboard piping flow rate associated with a 100 scfh Leak Rate Acceptance Criterion is:

Leak Rate Acceptance Criterion (scfh) $* [14.7 / (P_{MSIVtest} + 14.7)] * (276+460) / (68+460)$

Where:

$P_{MSIVtest}$ = 22 psig MSIV Test Pressure
276 °F is the peak drywell temperature at 2 minutes, per Attachment E.

Associated outboard piping flow rates are:

Leak Rate Acceptance Criterion (scfh) $* (550+460) / (68+460)$

Where:

550 °F is the normal steam line operating temperature.
Credit is taken for temperature reductions to 410 °F at 24 hours, and to 200 °F at 96 hours in determining later flow rates.

Flow rates out of the condenser are similarly calculated with the assumption of a condenser air space temperature of 120 °F for the accident duration.

The main steam line wall temperatures utilized for LGS, following a LOCA, were developed as shown in Attachment F to Calculation LM-0646 Rev. 1, as a conservative bounding approximation to the generic cooldown analysis developed by GE and reported in the August 20, 1990 SAIC Report "MSIV Leakage - Iodine Transport Analysis" by J. E. Cline. Only conduction was considered, with cooldown independent of MSIV leakage flow for the relatively low flow rates. This conservative approximation considers cooldown from an initial 550 degrees F initial pipe temperature only at 1 day and 4 days after the assumed LOCA, with no credit for cooldown between these times. The Cline assessment is generically applicable to all BWR main steam lines, and the conservative cooldown approximation utilized assures its conservative application to LGS.

The Cline generic cooldown curve is based on 4 inches of insulation on the main steam lines and is conservative for the 3.5 inches of insulation on LGS main steam line piping from the reactor vessel to the turbine stop valve.

References 28 (MSIV Leakage Iodine Transport Analysis, August 20, 1990) and 29 (MSIV Leakage Iodine Transport Analysis March 26, 1991) can be found in NRC ADAMS system under ascension numbers 9009120302 and ML003683718, respectively.

14. In Attachment 1, page 24, a value for the emergency core cooling system (ECCS) flash fraction is given as 1.39% as opposed to 10% in RG 1.183, Section 5.5. LGS states that a smaller amount (than the RG) was determined using a method approved for the Clinton Power Station, Unit I. If this value is not in your current licensing basis, please explain why this method is acceptable for LGS. If the value is new, please provide the details used to calculate this value, including the following:

    a. Although the analysis includes a limiting pH, no specific details regarding the pH history versus time are provided. Please provide the iodine concentration in the sump versus time. Please provide the pH vs. time or the pH assumed for the duration of the accident, including justification for the pH and iodine concentration used. Please provide the area ventilation rates that the ECCS leakages are exposed to.

    b. The ORNL study cited in the Clinton AST submittal is based upon theoretical calculations for the design of reactor containment spray systems. Many of the release mechanisms and other plant-specific issues have not been addressed. This creates notable uncertainties in how much iodine is available for release. Major uncertainties exist to what extent the chemicals within the leakage will interact when their release to the environment leads to a great reduction in vapor pressure.

       The production of elemental iodine is related to the pH of the water pools. A major uncertainty in fixing the production of volatile iodine chemical forms is due to uncertainty in the extent of evaporation to dryness. Experts believe that up to

20% of the iodine in water pools that has evaporated would be converted to a volatile form (most likely as elemental iodine). Uncertainties also depend upon the environment where the fluid is leaked and the way the fluid is leaked (misting, etc.). Fluid pH shifts may occur due to interactions with components, cable jackets, concrete and radiation. Please include a discussion of these issues to support the proposed value.

EXELON RESPONSE

A 10% flashing fraction is now used in the AST LOCA analysis (calculation LM-0646), which is the same value used in the existing licensing basis analysis and recommended in Reg Guide 1.183. Therefore, responses to parts a. and b. are no longer necessary.

15. From the LGS UFSAR, Table 6.2-4a, (stated as Rev. 11), the minimum suppression pool free airspace is given as 147.670 cubic feet. Typically, a Mark II suppression pool free volume is on the order of hundreds of thousands of cubic feet. Please clarify whether the UFSAR number is a typographical error and whether the decimal should be a comma.

Please provide justification for the use of 159,540 cubic feet provided in Table 3, on page 31 of Attachment 1. Why is the more conservative UFSAR value not valid for the LOCA analysis?

EXELON RESPONSE

The 147.670 cubic feet referenced in the UFSAR was a typographical error and should have been 147,670 cubic feet. The UFSAR has been corrected through Exelon's design change process.

In addition, the minimum suppression chamber airspace (147,670 cubic feet) as specified in the UFSAR, should have been used in the LOCA analysis. The LOCA analysis has been revised to reflect this change.

16. Page 61 of Attachment 1, Table C, contains a comparison of the LGS analysis to Section 2.0 of RG 1.183. The LGS analysis column of this table states that it conforms with RG 1.183, but this RG does not find the use of a Decontamination Factor (DF) of 200 acceptable for less than 23 feet of water covering a damaged fuel assembly.

a. Please provide the DF used for 21.6 and 22.6 feet of water and the parameters, methodology and justification used to calculate this value.

EXELON RESPONSE

The calculation LM-0645, Rev 1 for FHA is enclosed and provides the requested information. The DF for 21.6 feet of water is 155.7 and the DF for 22.6 feet of water is 186.9. Appendix E of LM-0645 provides justification for use of these values.

b. Considering the statement "The conservatively determined damage over the spent fuel pool is 70% of the reactor vessel," please provide the analysis used to justify this statement.

EXELON RESPONSE

The calculation LM-0645, Rev 1 for FHA is enclosed. Section 2.1 of the calculation provides the bases for determination of damage resulting from an FHA over the spent fuel pool. These analyses are consistent with the current LGS design basis as associated with GE14 fuel.

c. The UFSAR fuel handling analysis states that 212 are assumed damaged as the result of the fuel-handling accident. Attachment 1, page 60, states that based upon a generic evaluation of GE11 and GE14 fuel, such an accident "yields 172 failed rods." Is this a change? If so, please justify. If not, where is it substantiated?

EXELON RESPONSE

The design basis for the FHA has been revised. The details for these changes and justification for the revised basis is described in Section 2.1 of Calculation LM-0645.

17. Attachment 1, page 45, Section 4.2.3, states that the models used to transport radioactive material into and through the control room, and the shielding models used to determine radiation dose rates from external sources, should be structured to provide suitably conservative estimates of the exposure to control room personnel. It states that the LGS analysis conforms to this guidance.

a. Attachment 8, page 7, states that RADTRAD was used to determine the core spray line dose rates. The 60 radionuclides that are contained in the RADTRAD code were selected based upon a study that determined that those 60 radionuclides have the greatest impact on offsite dose. Please confirm that the most conservative radionuclides were used to determine the source for the LGS shielding studies for the shine doses from external sources to the control room. Provide the source terms used and the geometry and materials used in these shielding studies.

EXELON RESPONSE

A re-evaluation for this revision of the LOCA calculation of direct shine doses to the control room is provided in calculation LM-0646 Attachment C. The source terms, the geometry and the materials used are contained in the calculation. A total of 110 isotopes (including applicable isotopes from the 60 considered in the RADTRAD code) are modeled, to assure conservative shielding results.

b. Attachment 8, page 6, states that a zone is identified where controls are practical and suggests that the maximum boundary dose (at the inside control room wall) from outside sources should not be used to determine the limiting control room dose. Administrative controls and occupancy factors within zones seem to be credited. The value added to the control room dose from gamma shine is .22 rem, which appears to correlate to a dose 18 feet from the wall.

The above described methods and assumptions are inconsistent with your current licensing bases. UFSAR, Section 6.4.2.5, states that shielding is designed for continuous occupancy. Section 12.3.2.3 states:

"The shielding thicknesses are selected to reduce the aggregate radiation level from all contributing sources below the upper limit of the radiation zone specified for each plant area. Shielding requirements are evaluated at the point of maximum radiation dose through any wall. Therefore, the actual anticipated radiation levels in the greater region of each plant area are below this maximum dose and, therefore, below the radiation zone upper limit."

The NRC staff does not find the proposed practice acceptable. Access is needed to these locations. Administrative controls within the control room boundary are not an adequate substitute for potentially inadequate shielding. The staff believes that this is not consistent with the licensee's stated conformance with Regulatory Position 4.2.3 of RG 1.183. Please include the maximum doses from these external sources consistent with your current licensing basis or provide additional justification why such deviations from standard shielding practices are unavoidable and necessary.

EXELON RESPONSE

As noted in the Response to RAI question 6, the use of administrative controls for access to portions of the control room is no longer being considered and the maximum dose rates in the control room from external sources are used to characterize the entire control room.

c. The licensee states that MicroShield was used to determine the doses from the external piping. Please explain how the impact of scattering is considered. Please justify why modeling of scattering off piping, air, etc., is not considered, or include the impact of scattering in your assessment.

EXELON RESPONSE

Analysis of scatter is based on the use of the point-source infinite medium buildup factors used by MicroShield, which includes modeling of scattering off piping, air, etc. The impact of scattering has been included in the dose determination. See Attachment C of LM-0646.

d. Please provide a copy of the calculation or the information necessary to model the shine from this pipe. Include the geometry (drawings, piping, etc.), source term, materials, and assumptions used to determine the doses given on page 7 of Attachment 8.

EXELON RESPONSE

LOCA calculation LM-0646 is provided for review. The requested information is provided in Attachment C of the calculation.

e. UFSAR, Section 6.4.4.1, states, "Control room shielding design, based on the most limiting radiological accident (design basis LOCA) is discussed in Section 12.3. The evaluations in Chapter 12 demonstrate that radiation exposures to control room personnel originate from containment shine, external cloud shine, and containment airborne radioactivity sources. Total exposures resulting from the worst radiological accident are below the dose limits specified by [General Design Criterion] GDC 19; the portion contributed by containment shine and external cloud shine is reduced to a small fraction of the walls which surround the control room."

Page 6 of Attachment 8 to the application states that historically the dose due to the core spray piping and other lesser piping contributors is 4.2 rem whole body. The licensee also states that the "Other sources such as reactor enclosure airborne and external cloud and RERS, SGTS, and CREFAS filters are negligible because of shielding, distance or both."

Please clarify whether the proposed change involves a change to the bases for current shine analysis for piping and sources other than the containment spray piping. If parameters or assumptions have changed, please provide the bases for the sources used, the parameters used for this reevaluation, any assumptions used, and the results of the analyses.

EXELON RESPONSE

The physical sources addressed quantitatively and qualitatively in Attachment C of LM-0646 are consistent with the original plant design analyses. The radioactivity in these sources is changed to reflect AST assumptions. These changes are due to ECCS fluid activities based on the extended isotope list, with the releases assumed to be directly from the core with the RG 1.183 release fractions and timing.

The quoted UFSAR Section 6.4.4.1 statement will be revised to acknowledge the contribution of the other sources. Contributions from other sources are appropriately discussed and assessed in section 4.7 of LM-0646 Rev 1.

18. Table 11c, page 24, of Attachment 8 to the application indicates that "pathway 6" provides a flow path from node 5 to node 3. Table 11a does not provide a description of node 5. Please provide a description of node 5 (as is done with nodes 1 through 4) and describe how this is different from node 2. If node 5 is the same as the node 5 described in Table 13a of Attachment 8, justify the use of a node for the SGTS. Typically, the SGTS is modeled as a transfer pathway rather than a node. Confirm this model yields conservative results.

EXELON RESPONSE

Node 5 is a compartment that receives the portion of RERS exhaust that is directed to the environment through the SGTS (titled SGTS Node). Pathway 6 is the transfer of the activity from the SGTS Node 5 to the environment. Therefore, the SGTS is modeled as a transfer pathway and is consistent with normal RADTRAD use.

Reference 1, Attachment 8, Tables 3 – 15 have been superseded. Refer to calculation LM-0646 (LOCA), section 6.1 for information regarding node and pathway descriptions.

19. More detail regarding the main steam line break (MSLB), fuel handling, and control rod drop accidents is needed. Please provide all assumptions, inputs, models and methodologies (CRDAs) used to calculate the offsite and control room doses. Please include answers to the following questions:

What is the reactor coolant system (RCS) activity used for the MSLB analysis? Provide the assumptions, input, and methods used to determine this activity.

The second bulleted item on page 20 of Attachment 1 to the application states that the activity in the steam cloud is based on the total mass of water released from the break. Confirm that the total activity released for this accident is the RCS-specific activity times the break discharge mass (103, 785 lbm). If this is not the methodology used, please provide more detail regarding the model utilized. Also, provide the input parameters used to calculate and justify the fraction of liquid water contained in the steam and the flashing fraction of liquid water released.

EXELON RESPONSE

The following calculations are enclosed.

MSLB: LM-0644
FHA: LM-0645
CRDA: LM-0643

Additionally, parameter tables are provided in Attachment 2 of this letter, showing the currently licensed values, the values submitted in the original AST submittal, and the final supplemented values.

The reactor coolant system (RCS) activities used for the MSLB analysis are 0.2 uCi/gm and 4.0 uCi/gm per TS limits and Reg Guide 1.183.

The value of 103,785 lbm should have been 108,785 (typographical error). The correct value was used in the calculation. The MSLB analysis now uses a bounding 140,000 lbs reactor coolant released. The dose due to cesium activity has also been considered in the analysis.

The justification for the fraction of liquid water contained in the steam and the flashing fraction of liquid water released is contained in Section 6.1 of Calculation LM-0644.

20. In Attachment 1, page 35, Table 8, a value of 0.77% damaged fuel with melt is provided for the CRDA. The value typically used for fuel melt with General Electric 14 fuel is 1% for the CRDA. Please confirm this value of 0.77% and justify the value if this is a change to your licensing basis.

EXELON RESPONSE

The 1% value is a conservative roundup rather than GE's (used at Clinton). The value of 0.77% is the correct non-rounded value based on the current LGS design basis described in UFSAR section 15.4.9.5.1.1.

21. Attachment 1, page 16, states that, "Infiltration following isolation is assumed to be 525 cfm of unfiltered inleakage, which includes impacts of ingress and egress." Please confirm that the 525 cfm value includes 10 cfm for the ingress and egress into the control room after a LOCA.

EXELON RESPONSE

The allowance of 525 cfm unfiltered inleakage included consideration for ingress/egress. However, as a result of recalculation, a new unfiltered inleakage value of 275 cfm is now assumed in the analysis. This value bounds the 79 cfm maximum as measured by tracer gas testing.

The unfiltered inleakage value assumes that there will be 0 cfm for ingress/egress. This is based on installation of a main control room door seal as discussed in UFSAR Sections 1.13, 6.4, and 15.10. The installation of the door seal is directed through existing plant procedures (SE-10, LOCA) and is based on receipt of the Turbine Enclosure Radiation monitor Common Area High Rad Alarm.

22. Comments provided for Section 5.1.3 in Attachment 1, page 47, state that, "conservative assumptions are used."

Please confirm that the control room and SGTS HVAC flow rates assumed in the accident analysis (including control room doses) are conservative based on the range of flow rates allowable by the TSs.

EXELON RESPONSE

The SGTS system flowrate is the Technical Specification maximum and is conservative because a higher flow rate maximizes the release rate and resulting dose to the Main Control Room (MCR) and offsite.

For the Control Room Emergency Fresh Air System (rad isolation mode), the filtered intake flow rate is assumed as the TS maximum which has been determined to maximize control room dose. The total CREFAS recirculation flow rate is based on Technical Specification minimum since this minimizes activity removal from the control room atmosphere.

23. The proposed change to TS 3.6.5.1.2, "Refueling Area Secondary Containment Integrity," will no longer require that the secondary containment be operable during the movement of fuel assemblies that have a decay period of at least 24 hours. The fuel-handling accident (FHA) analysis assumes the release to the control room intake and the environment is through the turbine building/reactor building (TB/RB) ventilation south stack. Please justify that an FHA release through the TB/RB ventilation south stack is an appropriately conservative assumption given that the secondary containment may be inoperable. Include general arrangement drawings in your response showing the potential release points.

EXELON RESPONSE

The North Stack, which is used for releases filtered by the SGTS, is located closer to the Control Room intake and therefore has higher X/Qs. However, the SGTS is designed to remove at least 99% of the iodine that would otherwise be released; this filtration more than overcomes the effect of the higher X/Qs, as demonstrated in LM-0645; therefore, the South Stack unfiltered release is bounding.

Site walkdowns and specific reviews of the LGS General Arrangement Drawings such as M-102, the Plan at El. 217' - 0" (one foot above Grade) and M-107, the Section showing the North and South Stack, confirmed that there are no potential release pathways that could be worse with respect to the Control Room intake than the analyzed stacks (these drawings are included in Attachment 4).

Hatches and Reactor Enclosure openings leading directly to the outside, as well as grade openings are considered to have X/Qs that are bounded by the South Stack release point X/Q. This is based on the South Stack release point having the shortest distance of travel to the Control Room intake. This includes the large railroad doors at grade elevation, which could be postulated to be open at the same time as the equipment hatch cover on the refueling floor to support future spent fuel cask moves. Any new opening would be evaluated for its effects on this accident before such opening would be allowed.

24. Please explain in detail the methodology used to model steam cloud transport for the MSLB accident. Please also describe the methodology (e.g., inputs and assumptions) used to determine the control room doses for the MSLB accident.

EXELON RESPONSE

Calculation LM-0644 is submitted for review, which contains the discussion of the cloud model used for the MSLB. A plant parameter table for AST (reference attachment 2 of this letter) is also provided to assist in the review.

The steam cloud model is only used for MCR dose analysis. Offsite dose analyses use X/Qs.

25. The inleakage of unfiltered air into the control room (which can occur through the control room boundary, system components, and backflow at the control room doors) was modeled using the control room intake χ/Q values. Please verify that there are no potential unfiltered inleakage pathways during the normal operation mode, radiation isolation mode, and chlorine isolation mode that could result in χ/Q values that are higher than the control room intake χ/Q values.

EXELON RESPONSE

Prior to the performance of the inleakage testing of the control room envelope in accordance with the requirements of GL 2003-01, numerous walkdowns were performed to identify any potential unfiltered leak pathways that could exist during any of the control room operating modes (normal, radiation isolation and chlorine isolation). These

walkdowns included the normal control room HVAC system, CREFAS system and all associated ductwork as well as the actual control room boundary. All potential unfiltered inleakage pathways identified were addressed prior to the performance of the inleakage testing. The actual test results demonstrated that the unfiltered inleakage into the control room envelope during either the radiation isolation mode or chlorine isolation mode of operation was less than 100 cfm. As a result, the control room intake X/Q values currently being used are appropriate and there are no unfiltered inleakage pathways that would necessitate changing the values being used.

26. Provide a curve of containment pressure as a function of time for the large break LOCA to verify that the containment pressure decreases to less than 50% of its peak value within 24 hours.

EXELON RESPONSE

In lieu of a curve of containment pressure as a function of time for the large break LOCA, a tabulation is provided as LOCA Calculation LM-0646, Attachment E, verifying that the containment pressure decreases to less than 50% of its peak value within 24 hours.

27. In TS Table 3.3.2-1, "Isolation Actuation Instrumentation Action Statements," and TS Table 4.3.2.1-1, "Isolation Actuation Instrumentation Surveillance Requirements," does the instrumentation referenced in the proposed change provide protection for an area that is common to both units and as such would it still be required when either unit was operating even though the other unit is in refueling? Please explain whether the alarm capability of this instrument would be available even if the actuation function were removed? Please explain whether the removal of this function would support the monitoring requirements of GDC-64? Please explain whether procedures are available that would manually isolate in lieu of the automatic isolation that is to be removed?

EXELON RESPONSE

The LGS secondary containment consists of three distinct isolation zones. Zones I and II are the Unit 1 & 2 reactor enclosures, respectively, and Zone III is the common refueling area. The SGTS is designed to isolate all 3 zones simultaneously or any combination of the zones. The common refueling area has a separate Unit 1 and Unit 2 HVAC system and is designed to operate during normal operation and also during fuel handling operations. Each HVAC system is capable of providing refueling area secondary containment zone isolation as required.

This License Amendment Request is not removing nor changing the function of any instrumentation associated with the Refueling Area HVAC system. The instrumentation discussed in TS Tables 3.3.2-1 and 4.3.2.1-1 (Refueling Area Ventilation Exhaust Duct Radiation – High) is associated only with the common refueling area HVAC systems and only provides protection for the refueling area environment in the event of a FHA. This instrumentation does not impact, nor will it cause an isolation of either the Unit 1 or Unit 2 reactor enclosure secondary containments, nor is it required to support operability of either Unit 1 or Unit 2. This instrumentation is needed to support the refueling area operability during Operational Condition *" and only when the associated Refueling Area exhaust system is operating and is physically separated from the operating unit(s).

Additionally, in order to support the removal of the drywell shield plugs prior to the start of a refueling outage, LGS has the capability to procedurally combine either reactor enclosure secondary containment zone (I or II) to the common refueling area secondary containment zone (III). During this limited period of time, when the two zones are combined as one zone, the refueling area isolation actuation instrumentation is needed to support the operating unit as well as the refueling area. The ability of the refueling area isolation instrumentation to provide the required isolation protection and alarming function in this combined mode is not being altered or changed as currently designed by this License Amendment Request.

During periods when the Refueling Area secondary containment integrity is not required, but movement of irradiated fuel or Core Alterations need to be performed, procedures will be in place to assure that the appropriate refueling area HVAC system is in operation, which includes the exhaust radiation monitors, in order to assure that proper protection is provided in the event of a FHA.

28. Please explain whether the instrumentation in TS Table 3.3.7.1-1, "Radiation Monitoring Instrumentation," and TS Table 4.3.7.1-1, "Radiation Monitoring Instrumentation Surveillance Requirements," would be operable if either unit were operating during an accident in one unit since the control room is common to both units? Explain whether the alarm and isolation functions would still be required since an accident at the operating unit could affect the habitability of the main control room?

EXELON RESPONSE

The Limerick control room is a common facility used to support both Unit 1 and Unit 2. The control room habitability system is designed to provide safety and comfort for operating personnel during normal operation and postulated design bases accident (DBA) conditions. A feature of this design is the Control Room Isolation System, which is provided in part to isolate the control room in the event of a high radiation condition and to provide control room annunciation. Radiation levels are monitored in the control room outside air intake using four separate channels to provide the appropriate radiation isolation signal and initiate the system isolation actions.

Although there is a separate Technical Specification for each unit that describes the radiation monitoring instrumentation, the surveillance requirements, and applicable operating conditions, the four radiation monitors and isolation channels are common to both Units 1 & 2. As a result, whenever a single unit (either Unit 1 or Unit 2) is operating, the control room isolation system is required to be operational, which includes all four radiation monitors and associated isolation channels being in full compliance with the Technical Specification requirements, to assure continued plant operation. This design is provided to assure continued control room habitability.

The proposed changes to TS Tables 3.3.7.1-1 and 4.3.7.1-1 will not affect or impact the ability of the control room habitability system to perform as designed and as described above.

29. Considering TS Section 3.6.5.2.2, "Refueling Area Secondary Containment Automatic Isolation Valves." TSTF-51 allows certain engineered safety feature (ESF) functions to be inoperable, such as the automatic isolation feature; however, it still requires the ability to isolate the secondary containment in order to meet the objectives of NUMARC 93-01. Will the ability to isolate the containment be retained if the automatic feature is disabled? If the secondary containment cannot be isolated, please explain how the station will meet the intent of GDC-64 in monitoring releases and the GDC 61 intent of controlling releases through containment, confinement, or filtering.

EXELON RESPONSE

The FHA was analyzed without benefit of filtration. Closure of the refueling area secondary containment is a non-credited defense-in-depth action. Manual closure will occur upon alarm actuation. The automatic isolation function is not physically being removed. The TS requirement for it is being removed.

Provisions and contingency actions to restore appropriate plant features in the event of a fuel handling accident will be included in plant procedures, as committed to in the original Limerick AST submittal, attachment 6. The restoration actions will assure that the intent of GDC-61 for controlling releases and GDC-64 for monitoring releases is achieved. The contingency actions will be identified in plant procedures and/or the Barrier Breach Program.

When automatic isolation valves are removed from service, the contingency actions will provide for a prompt method to sufficiently block the penetration such that the Standby Gas Treatment System or Refueling Area ventilation system can be utilized to process effluent from the affected area and that releases can be monitored.

The refueling area ventilation system contains radiation monitors on the exhaust to atmosphere and will be in-service during fuel movement or core alterations. These radiation monitors will alarm upon detection of high radiation.

Therefore, plant procedures will assure that the intent of GDC-61 and GDC-64 is satisfied.

30. In TS Section 4.6.5.3, "Standby Gas Treatment System - Common System," the staff notes that the TS cited references RG 1.52, Revision 2. Revision 2 states the maximum penetration for a 2-inch carbon adsorber should be less than 1%. The staff has issued Revision 3 which allows a penetration of 2.5% for a 2-inch bed filter. Please provide the appropriate RG Revision proposed for the penetration and include any extenuating circumstances where the conditions of the RG are not being met.

Discuss whether the filter is larger than a 2-inch bed filter. Discuss any specific need to retain RG 1.52, Revision 2, and exceed the maximum penetration limits shown in Table 2.

EXELON RESPONSE

Exelon has rescinded the proposed request to increase the allowable methyl iodide penetration for the laboratory testing of the Standby Gas Treatment System (TS Sections 4.6.5.3.b.2 and 4.6.5.3.c) charcoal adsorber samples from 0.5% to 1.25%. Therefore, since the design and operation of the Standby Gas Treatment System (SGTS) relative to charcoal adsorber sample testing is unchanged and still within existing Technical Specifications, Reg Guide 1.52, Rev. 2 and GL 99-02 requirements and commitments, no further action on this RAI is required. Note that the SGTS charcoal adsorber is 8 inches thick.

31. In SR 4.6.5.4.a, explain how this reactor enclosure recirculation system flow rate compares to the design flow rate of the system used in the evaluation of design-basis accidents. Are there any reasons why the design flow rate (rated flow) of the system with an appropriate tolerance should not be specified? Explain whether the proposed change would allow testing at a flow rate that was significantly lower than the design flow rate for its intended service.

EXELON RESPONSE

Exelon has rescinded the proposed request to change the acceptable Reactor Enclosure Recirculation System subsystem flow rate from 60,000 cfm +/- 10% to 30,000 cfm to 66,000 cfm. Therefore, since the design and operation of the Reactor Enclosure Recirculation System relative to subsystem flow rate is unchanged, no further action on this RAI is required.

32. Please explain whether the changes proposed in SR 4.6.5.4.b.1, SR 4.6.5.4.d.1, SR 4.6.5.4.e, and SR 4.6.5.4.f, provide for doing anything different from the way it is done now.

EXELON RESPONSE

The proposed changes have been rescinded in conjunction with question 31 above.

33. In SR 4.6.5.4.b.2 and SR 4.6.5.4.c, the existing penetration of 2.5% is the maximum allowable penetration for a 2-inch filter based on the conditions of RG 1.52, Revision 3. The TS references Revision 2. The proposed 15% penetration indicates that the carbon adsorber is in a degraded state. The RG 1.52 values are based on clean carbon adsorbers. The staff does not have data to show how quickly carbon adsorbers degrade once they are in a degraded state. Although the analysis may show that a 15% penetration would be acceptable, there is an increased uncertainty that the filters would still be acceptable at the end of the inspection interval. The fact that the filters have reached the degraded state may indicate that some operational changes need to be made to prevent filter degradation. Please provide data to justify the filter performance from a 15%-degraded state for the entire inspection interval or justify this change by other information.

EXELON RESPONSE

Exelon has rescinded the proposed request to increase the allowable methyl iodide penetration for the laboratory testing of the Reactor Enclosure Recirculation System (TS Sections 4.6.5.4.b.2 and 4.6.5.4.c) charcoal adsorber samples from 2.5% to 15%. Therefore, since the design and operation of the Reactor Enclosure Recirculation System relative to charcoal adsorber sample testing is unchanged and still within existing Technical Specifications, Reg Guide 1.52, Rev. 2 and GL 99-02 requirements and commitments, no further action on this RAI is required.

34. In SR 4.6.5.4.b.3, the subsystem flow rate affects the clean-up rate for filtration and should be established at the flow rate credited for the subsystem in any analyses. Please clarify why a large range is needed and why the flow rate cannot be closely tied to the values used in the design-basis analyses.

EXELON RESPONSE

Exelon has rescinded the proposed request to change the acceptable Reactor Enclosure Recirculation System subsystem flow rate from 60,000 cfm +/- 10% to 30,000 cfm to 66,000 cfm. Therefore, since the design and operation of the Reactor Enclosure Recirculation System relative to subsystem flow rate is unchanged, no further action on this RAI is required.

35. In TS Section 3.7.1.2, "Emergency Service Water System - Common System," and TS Section 3.7.1.3, "Ultimate Heat Sink," the staff is concerned that the proposed change does not "expand the definition" as stated. The relaxations that have been granted through TSTF-51 were based on satisfying the requirements of the FHA. Please provide additional justification for this change. Discuss whether other potential transients that would require the use of either the emergency service water or ultimate heat sink have been evaluated to assure that eliminating this applicability is justified with respect to two unit operability in which one unit is at full power.

EXELON RESPONSE

Exelon has rescinded the proposed request to change TS Section 3.7.1.2, "Emergency Service Water System - Common System," and TS Section 3.7.1.3, "Ultimate Heat Sink." The changes were requested to be consistent with TSTF-51; however, upon further review, the change was rescinded to avoid possible future confusion in the interpretation of the TS intent.

36. In TS Section 4.7.2, "Control Room Emergency Fresh Air Supply System - Common System," the existing penetration of 2.5% is the maximum allowable penetration for a 2- inch filter based on the conditions of RG 1.52, Revision 3. The TS references Revision 2. The proposed 10% penetration indicates that the carbon adsorber is in a degraded state. RG 1.52 values are based on clean carbon adsorbers. The staff does not have data to show how quickly carbon adsorbers degrade once they are in a degraded state. Although the analysis may show that a 10% penetration would be acceptable, there is an increased uncertainty that the filters would still be acceptable at the end of the inspection interval. The fact that the filters have reached the degraded

state may indicate that some operational changes need to be made to prevent filter degradation. Please provide data to justify the filter performance from a 10%-degraded state for the entire inspection interval or justify this change by other information. Please provide additional justification for changing to a manual initiation of the radiation mode of the control room emergency fresh air system. RG 1.183 states that "modifications proposed for the facility generally should not create a need for compensatory programmatic activities, such as reliance on manual operator actions." Please discuss the impact of this change on one unit when the other unit is at full power.

EXELON RESPONSE

Exelon has rescinded the proposed request to increase the allowable methyl iodide penetration for the laboratory testing of the Control Room Emergency Fresh Air Supply System (TS Section 4.7.2.c.2 and 4.7.2.d) charcoal adsorber samples from 2.5% to 10%. Therefore, since the design and operation of Control Room Emergency Fresh Air Supply System is unchanged and still within existing Technical Specifications, Reg Guide 1.52, Rev. 2 and GL 99-02 requirements and commitments, no further action on the RAI is required. Additionally, as stated in response to RAI question #9, Exelon will no longer pursue a change to the Control Room Emergency Fresh Air Supply System design basis, which would have allowed removal of the automatic start feature in the radiation isolation mode.

37. Please provide a description of the analysis assumptions, inputs, methods, and results that show that a sufficient quantity of sodium pentaborate can be injected to raise and maintain the suppression pool greater than pH 7 within 24 hours of the start of the event. (See also Position 2 of Appendix A to RG 1.183.) In your response, please discuss the adequacy of recirculation of suppression pool liquid via ECCS through the reactor vessel and the break location and back to the suppression pool in meeting the transport and mixing assumptions in the chemical analyses. Assume a large break LOCA.

In responding to this question, please indicate the source and volume flow rate of water that mixes with the sodium pentaborate and washes it from the vessel to the suppression pool. A diagram showing the injection point of the sodium pentaborate, the flow path through the core, and the exit path from the vessel would be helpful. Please discuss how the proposed change would continue to ensure that the core ECCS flow does not bypass the region of the vessel that contains sodium pentaborate and that sufficient sodium pentaborate will be transported to the suppression pool.

EXELON RESPONSE

Enclosed calculation LM-0642 provides the derivation of the suppression pool pH using the calculation methodology approved by the NRC for the Grand Gulf Nuclear Station. Calculation LM-0642 contains the complete description of the analysis assumptions, inputs, methods, and results requested above. The following major assumptions and inputs are included in the calculation:

- The Suppression Pool is assumed to be well mixed so that the pH at any time can be represented by a single value.
- All exposed cables located in the drywell are considered, with 10% of the total cable lengths and 10% of the total cable surface area added to the computed quantities to account for any potential missed cables.
- The coolant inventory volume was selected to bound the total volume of the suppression pool at high water level, the Reactor vessel, and the ECCS piping, with additional volume added for further conservatism.

LM-0642 (Attachment C) calculated suppression pool pH as a function of time after accident initiation for Beginning of Cycle (BOC) and End of Cycle (EOC) conditions, both with and without SLC injection, to determine the minimum amount of boron required to maintain pH above 7.0 for 30 days. LM-0642 concluded that 1500 gallons containing 1313 lbs of sodium pentaborate (and 240 lbs of total boron) would provide sufficient buffering to maintain pH above 7.0 for the 30-day event duration. Technical Specification 4.1.5 requires 3160 gallons in the SLC tank to perform the reactivity control function. This volume contains greater than 240 pounds total boron under all allowed concentrations and enrichments. Note, per the Reference 1 submittal, TS 4.1.5.b.2 is changed to require greater than 185 lbs of boron-10 instead of 3754 lbs of Sodium Pentaborate.

The SLC system flow is injected into the RPV through B Core Spray (CS) line. The SLC system connection to CS is inboard of the CS isolation valves. In the scenario postulated following a large break loss-of-coolant accident (LOCA), the sodium pentaborate solution is not required until well after the gap release and early in-vessel release phases of the event (after the first 2 hours). Since this is in the ECCS long-term core cooling phase (after 10 minutes), the ECCS design core spray system minimum required flow inside the shroud is 6250 gpm at 105 psid, in order to provide sufficient core cooling. Thus, when the sodium pentaborate is injected during the long-term core cooling phase of the accident, the sodium pentaborate will either be entrained with B Core Spray flow prior to entering the vessel or, if B Core Spray were not available, mixed with the alternate ECCS flows once it enters inside the core shroud.

The initial mixing of the sodium pentaborate with the ECCS flow as it enters the core will ensure that the sodium pentaborate will be transported with the ECCS flow. The core coolant (with the sodium pentaborate) would flow down through the core volume into the lower plenum. Due to the effects of coolant addition inside the core shroud, lower plenum fluid would be forced upward through the jet pumps, and into the vessel annulus via the jet pump suction chamber. Coolant flow would flow out of the break location via the applicable Recirculation Pump suction nozzle. The flow would exit the reactor coolant pressure boundary via the break location, mix with coolant in the drywell, and flow through the downcomers into the suppression pool. See Attachment 4 for a schematic representation of the flowpath.

38. The submittal states that LGS is committing to NUMARC 93-01 which requires prompt closure of containment and control of releases from FHAs. NUMARC 93-01 states, in part, that, "these prompt methods need not completely block the penetrations nor be capable of resisting pressure, but are to enable the ventilation systems to draw from the postulated FHA such that it can be treated and monitored." Please describe the prompt

methods and the degree of closure that will be achieved. How much of an open area to the environment would be permitted? Also, please describe the ventilation systems that would be used to draw the release from the postulated FHA. Specifically, are the ventilation systems ESF systems, do they have carbon adsorber filters and high-efficiency particulate air (HEPA) filters, are they tested in accordance with RG 1.52 or other standards, and do they have sufficient drawing capacity to assure that air flow is going from the environment to the containment?

Will there be a test to determine that all air flow was going into the containment in the event that the LGS procedure allows partial closure?

EXELON RESPONSE

The LGS secondary containment consists of three distinct isolatable zones. Zones I and II consist of the Unit 1 and Unit 2 reactor enclosures, respectively and Zone III is the common refueling area. Each zone is provided with an independent HVAC system designed to operate during normal plant operation. The common refueling area has a separate Unit 1 and Unit 2 HVAC system and is designed to operate during normal operation and also during fuel handling operations. Additionally, each HVAC system is capable of providing secondary containment zone isolation as required.

During periods when the Refueling Area Secondary Containment integrity is required (TS Section 3/4.6.5.1.2) to be established, the normally operating refueling area HVAC system will automatically initiate a Zone III secondary containment isolation upon receipt of one of the following isolation signals; 1) high radiation in the refueling area exhaust ducts or 2) low zone differential pressure. Either of these isolation signals will result in the following automatic sequence for Zone III:

1. Close (within 3-5 seconds) all normally open isolation valves (two in series) separating the safety related from the non-safety related portions of Zone III.
2. Trip all running ventilation fans and prevent standby fans from operating within Zone III.
3. Open (within 3-5 seconds) normally closed isolation valves (two in parallel) in the duct that connects Zone III to the Standby Gas Treatment System (SGTS) fans and filters.
4. Start the SGTS.

The SGTS is a safety related, Engineered Safety Feature (ESF) system that is designed to reduce airborne iodine and aerosol activities potentially present following a postulated FHA before the gases are discharged to the environment. This system is designed to exhaust a sufficient amount of air from the Zone III secondary containment to re-establish and maintain a negative pressure of at least 0.25 inch W.G for the entire duration of the FHA event. The SGTS will maintain the re-established negative pressure of at least 0.25 inch W.G. at a flow rate not exceeding 764 cfm.

All Zone III secondary containment exhaust air is processed thru the SGTS which consists of an 8 inch thick bed of activated/impregnated charcoal adsorber and two

banks of HEPA filters (upstream and downstream of the charcoal adsorber), prior to being released thru a monitored exhaust path (north stack) to the environment.

The SGTS was designed and is periodically tested in accordance with the requirements specified in Reg Guide 1.52, Revision 2 and Generic Letter (GL) 99-02. Specific testing requirements are denoted in the Limerick Technical Specifications. Additionally, a review of periodic surveillance test results that have been performed on the SGTS clearly demonstrates that the SGTS is capable of re-establishing and maintaining a negative pressure within the Zone III secondary containment and at a flow rate not exceeding 764 cfm.

When secondary containment integrity is required, the capability of the Zone III secondary containment integrity, the isolation capability and the SGTS will not be physically challenged or compromised as a result of any proposed changes associated with the Alternative Source Term license amendment.

A program will be in place to ensure prompt action(s) will be taken, if necessary, to isolate the refueling area and control/monitor any potential releases as a result of a FHA during periods when the Refueling Area Secondary Containment integrity is not required and movement of irradiated fuel or Core Alterations need to be performed. The degree of closure and the size of the open area to the environment will be controlled through the Barrier Breach Program to ensure that the ability exists to contain the release of a postulated fuel handling accident and to send it in the proper direction such that it can be monitored through either the SGTS or the normal Refuel Floor HVAC system. The SGTS is an ESF system that contains carbon adsorber filters and high-efficiency particulate air (HEPA) filters. The normal Refueling Area HVAC is a non-filtered and a non-ESF system; however, the Refueling Area HVAC system directs exhaust to the South Stack, which provides a monitored release path and has been evaluated to meet dose limits.

Additionally, these prompt actions are capable of being performed within a one hour period following the FHA. These prompt actions are not credited in the FHA analysis that was created for this scenario, but are only considered as additional defense-in-depth measures. Testing, other than what is currently required by the Technical Specifications, is not needed.

39. Limerick has proposed to credit control of the pH in the suppression pool following a LOCA by means of injecting sodium pentaborate into the reactor core with the SLC system. The SLC system design was not previously reviewed for this safety function (pH control post-LOCA). Please demonstrate that the SLC system is capable of performing the pH control safety function assumed in the AST LOCA dose analysis.

The following questions are from a set of generic questions developed by the staff and are being provided to all BWR licensees with pending AST license amendment requests. In responding to questions regarding the SLC system, please focus on the proposed pH control safety function. The reactivity control safety function is not in question. For example, the SLC system may be redundant with regard to the reactivity control safety function, but lack redundancy for the proposed pH control safety function. If you believe that the information was previously submitted to support the license amendment request

to implement AST, you may refer to where that information may be found in the documentation.

EXELON RESPONSE

The capability of the SLC system to perform the function to control suppression pool pH is described in the answers to the questions 40, 41, 42, and 43 below.

40. Please state whether or not the SLC system is classified as a safety-related system as defined in Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Section 50.2, and whether or not 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 1.1 to 1.5 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.

    a. Is the SLC system provided with standby AC power supplemented by the emergency diesel generators?

    b. 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)?

    c. 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 plant's code of record (10 CFR Part 50.55a)?

    d. Is the SLC system incorporated into the plant's Maintenance Rule program consistent with 10 CFR 50.65?

    e. Does the SLC system meet 10 CFR 50.49 and Appendix A to 10 CFR Part 50 (GDC-4, or equivalent used for original licensing)?

EXELON RESPONSE

The Limerick SLC system is a safety related system and the components required to perform the function of backup capability for reactivity control meet criteria of (a) through (d) above.

Limerick equipment requiring environmental qualification was qualified in accordance with NUREG-0588, Category II as described in Sections 8.1.6.1 and 3.11 of the UFSAR. Several SLC components in the Reactor Enclosure including the SLC Tank level transmitters, PCIV operators and the Motor Control Centers for the pumps are environmentally qualified for the post-LOCA environment. Some components including the squib valve operator and the pump motor were not previously environmentally qualified for the post-LOCA environment. These components have been determined to be capable of performing the new post-LOCA function in the accident environment. The component records will be revised to address the Post LOCA EQ requirements prior to AST implementation.

41. Please describe proposed changes to plant procedures that implement SLC sodium pentaborate injection as a pH control additive. In addition, please address Items 2.1 to 2.5 below in your response. If any item is answered in the negative, please explain why the SLC system should be found acceptable for pH control additive injection.

EXELON RESPONSE

Limerick Special Event procedure SE-10, "LOCA", which provides direction for operator action after a design basis loss of coolant accident initiation signal has been received, will be revised to include direction to inject sodium pentaborate in accordance with system operating procedure S48.1.B, "Standby Liquid Control System Manual Initiation".

a. Are the SLC injection steps part of a safety-related plant procedure?

EXELON RESPONSE

The procedures that implement SLC sodium pentaborate injection are controlled procedures that are prepared, reviewed, approved, and used in accordance with the Exelon Generation Co, LLC, AmerGen Energy Co, LLC Quality Assurance Topical Report, NO-AA-10.

b. Are the entry conditions for the SLC injection procedure steps symptoms of imminent or actual core damage?

EXELON RESPONSE

Yes. Currently SLC is injected as directed in Limerick Transient Response Implementation Plan (TRIP) procedures and Severe Accident Management Plan (SAMP) procedures. These procedures implement the generic BWROG Emergency Procedure Guidelines (EPG)/Severe Accident Guidelines (SAG).

SLC injection is directed in LGS procedures T-101, "RPV Control", as an Anticipated Transient without Scram (ATWS) mitigation strategy and can be utilized as an alternate injection system for RPV level control in T-101, T-111, "Level Restoration/Steam Cooling" and T-116, "RPV Flooding". Additionally, upon transition to the Severe Accident Management procedures, SAMP-1, "RPV And Primary Containment Flooding Control", requires SLC injection to prevent core re-criticality. Injection of SLC in SAMP-1 is performed regardless as to whether an ATWS condition exists or not.

T-101, "RPV Control," is entered with reactor pressure vessel (RPV) water level below the scram setpoint, RPV pressure above the high pressure scram setpoint, drywell pressure above the scram setpoint, or reactor power above the low power alarm with a scram signal present. The entry conditions for T-101 are indicative of a plant condition that could degrade to imminent or actual core damage. The RPV low level and the drywell high pressure entry conditions ensure that T-101 is entered for a LOCA.

When conditions defined in the TRIP procedures indicate that adequate core cooling cannot be restored and maintained, for any reason, then SAMP entry is

directed. Adequate core cooling is defined as core submersion, steam cooling with injection, or steam cooling without injection.

In parallel to performance of the TRIP procedures, procedure SE-10, "LOCA," is entered based on the design basis entry condition for a LOCA of:

Low Low Low RPV water level (-129")
OR
High Drywell Pressure (1.68 psig)
AND Low RPV Pressure (455 psig)

The appropriate TRIP procedure steps currently acknowledge parallel performance of SE-10, by incorporation of a NOTE associated with performance steps related to RPV level, drywell pressure and RPV pressure.

The specific entry conditions described above that alert the operator to take actions to inject SLC are symptoms of imminent core damage.

c. Does the instrumentation cited in the procedure entry conditions meet the quality requirements for a Type E variable as defined in RG 1.97, Tables 1 and 2?

EXELON RESPONSE

No, although the drywell high pressure, reactor pressure, and reactor water level instruments used for procedure entry conditions are not Type E instruments, they do meet the quality requirements for Type A or B variables as defined in Regulatory Guide 1.97, Tables 1 and 2. Type E variables are for post-accident radiation monitoring, whereas the instruments used to determine initiating conditions for SLC injection are based upon a LOCA signal rather than radiation.

d. Have plant personnel received initial and periodic refresher training in the SLC injection procedure?

EXELON RESPONSE

Yes, Licensed operators have received initial training on the TRIP and SAMP procedures, and will continue to receive periodic refresher training. Additionally, training will be provided to operators and appropriate Emergency Response Organization personnel for procedures that will be revised prior to implementation of the AST amendment to specifically direct boron injection for pH control following a LOCA.

e. Have other plant procedures (e.g., emergency response guidelines/senior advisory groups (ERGs/SAGs)) that call for termination of SLC as a reactivity control measure been appropriately revised to prevent blocking of SLC injection as a pH control measure. (For example, the override before Step RC/Q-1, *"If while executing the following steps: ....it has been determined that the reactor will remain shutdown under all conditions without boron, terminate boron injection and...."*)

EXELON RESPONSE

BWROG EPG/SAG step RC/Q-1 and the associated override apply to ATWS events only, as part of the reactor power control strategy. If an ATWS was in progress and it had been subsequently determined that the reactor would remain shutdown under all conditions, without boron, it is appropriate to terminate boron injection.

Procedure SE-10, "LOCA," is entered and performed in parallel to the Limerick TRIP procedures based on the design basis entry condition for a LOCA. The appropriate TRIP procedure steps currently acknowledge parallel performance of SE-10. No change to the reactor power control leg of the Limerick TRIP procedures is required.

The change described above to SE-10, "LOCA", to include direction to inject sodium pentaborate in accordance with system operating procedure S48.1.B, "Standby Liquid Control System Manual Initiation", will assure that sufficient sodium pentaborate is injected into the RPV to maintain suppression pool pH above 7.0 for the duration of the LOCA.

42. Please provide a description of the analysis assumptions, inputs, methods, and results that show that a sufficient quantity of sodium pentaborate can be injected to raise and maintain the suppression pool greater than pH 7 within 24 hours of the start of the event. (See also Position 2 of Appendix A to RG 1.183.) In your response, please discuss the adequacy of recirculation of suppression pool liquid via ECCS through the reactor vessel and the break location and back to the suppression pool in meeting the transport and mixing assumptions in the chemical analyses. Assume a large break LOCA.

EXELON RESPONSE

The response to question 37 above addresses this question.

43. Please show that the SLC system has suitable redundancy in components and features to assure that for onsite <u>or</u> offsite electric power operation 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. If the SLC system cannot be considered redundant with respect to its active components, the licensee should implement one of the three options described below, providing the information specified for that option for staff review.

EXELON RESPONSE

The LGS SLC system is redundant with respect to its active components, except as outlined below. The non-redundant active components on the LGS SLC system are the two series check valves 048-*F007 and 048-*027 on the injection line inside containment (* represents unit designation). These two non-redundant active components are addressed in accordance with Option 1.

a. 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:

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

EXELON RESPONSE

The non-redundant active components on the LGS SLC system are the two series check valves 048-*F007 and 048-*027 on the injection line inside containment.

Unit	Component	Description	Manufacturer	Model
1	048-1F007	Lift-Check	Borg-Warner	77930
1	048-1027	Lift-Check	Borg-Warner	77930
2	048-2F007	Lift-Check	Borg-Warner	77930
2	048-2027	Lift-Check	Borg-Warner	77930

a.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 design-basis accident. Environmental conditions include design-basis pressure, temperature, relative humidity and radiation fields.

EXELON RESPONSE

The vendor design ratings for these check valves are 2570 psi at 100 degrees F. The check valves are designed for pressure and temperature conditions commensurate with the piping system in which they are installed. Valve 048-*F007 is designed to 1500 psig and 598 degrees F. Valve 048-*027 is designed to 1250 psig and 582 degrees F. These valves are both located inside the primary containment. The valves are designed to Seismic Category 1 requirements. Additionally, the components are metallic and do not contain "soft" parts, and therefore would not be subject to an adverse impact due to the radiation environment and humidity in the primary containment.

a.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.

EXELON RESPONSE

The above valves are safety related and were purchased ASME section III, class 1, 1974 edition, Winter 1976 addenda and 10CFR50 Appendix B.

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

EXELON RESPONSE

There have been no failures of the above valves in LGS plant history. No failures of these lift check valves were identified in the EPIX or NPRDS databases.

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

EXELON RESPONSE

The 048-*027 valves are in the IST program due to their safety function in the open direction to provide a flowpath. These valves are exercise tested during each refueling outage by the SLC injection test in accordance with the IST program requirements. This test verifies the SLC system can inject a minimum of 41.2 gpm to the reactor. (Ref. ST-6-048-320-*, SLC Operability Verification and Valve Test)

The 048-*F007 valves are in the IST program due to the safety functions in both the open direction (provide flowpath) and in the closed direction as a primary containment isolation valve (PCIV). The open direction is tested by the SLC injection test each refueling outage as described above. The closed direction is tested by a seat leakage rate test in accordance with the Appendix J, Option B Local Leak Rate testing (LLRT) program. The LLRT ensures a measurable leak rate of check valve 048-*F007 to verify that it closes properly and prevents reverse flow. (Ref. ST-4-LLR-421-*, Standby Liquid Control)

a.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 your 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.

EXELON RESPONSE

The LGS SLC system has sufficient redundancy to ensure a very high level of reliability. There are no known instances when the SLC injection test required elevated pump discharge pressure to inject flow to the reactor. The potential risk of the lift check valve(s) not opening is

considered extremely low since the positive displacement pumps discharge pressure will increase as needed to push the nominal 43 gpm per pump flowrate through the line. The maximum pressure developed by the pumps is 1400 psig as this is the relief valve lift setpoint. Therefore, this can result in a differential pressure across the check valve(s) in excess of 1000 psig.

The alternate methods to inject SLC solution to the reactor require access to the Reactor Enclosure. Given the high radiation environment in the building expected post-LOCA, it is not expected that these alternate methods would be available should the extremely unlikely failure of the check valves occur.

b. <u>Option 2</u> Provide for an alternative success path for injecting chemicals into the suppression pool. If you chose this option, please provide the following information:

b.1 Provide a description of the alternative injection path, its capabilities for performing the pH control function, and its quality characteristics.

b.2 Do the components which make up the alternative path meet the same quality characteristics required of the SLC system as described in Items 1.1 to 1.5, 2, and 3 above?

b.3 Does the alternate injection path require actions to be taken in areas outside the control room? How accessible will these areas be? What additional personnel would be required?

EXELON RESPONSE

Option 2 was not chosen. See Exelon's response to Option 1 above

c. <u>Option 3</u> Show that 10 CFR 50.67 dose criteria are met even if pH is not controlled. If you choose this option, demonstrate through analyses that the projected accident doses will continue to meet the criteria of 10 CFR 50.67 assuming that the suppression pool pH is not controlled. The dissolution of cesium iodide (CsI) and its re-evolution from the suppression pool as elemental iodine must be evaluated by a suitably conservative methodology. The analysis of iodine speciation should be provided for staff review. The analysis documentation should include a detailed description and justification of the analysis assumptions, inputs, methods, and results. The resulting iodine speciation should be incorporated into the dose analyses. The calculation may take credit for the mitigating capabilities of other equipment, for example the SGTS, if such equipment would be available. A description of the dose analysis assumptions, inputs, methods, and results should be provided. Licensees proposing this approach should recognize that this option will incur longer staff review times and will likely involve fee-billable support from national laboratories.

EXELON RESPONSE

Option 3 was not chosen. See Exelon's response to Option 1 above

44. Page 16 of Attachment 1 states "the transfer of radioactive gases into the control room are minimized by maintaining the control room at a positive pressure of 0.1-inch water column with respect to adjacent areas during emergency pressurized modes." Unit 1 TS 3.7.2 states that the control room is maintained at 1/8-inch water gauge positive pressure. This is equivalent to 0.125-inch water column. Verify that the 0.1-inch water gauge was inadvertently truncated and that LGS is not requesting to change its license basis to 0.1-inch water gauge or provide justification for your proposed change.

EXELON RESPONSE

This was a typographical error. The 0.125 value was truncated to 0.1 in the text.

ATTACHMENT 2

LIMERICK GENERATING STATION
UNITS 1 AND 2

Docket Nos. 50-352
50-353

License Nos. NPF-39
NPF-85

Supplement to License Amendment Request for
"LGS Alternative Source Term Implementation"

Limerick Generating Station (LGS) Plant Parameters Tables

Limerick Generating Station (LGS) Plant Parameters Tables

(Includes Changes Made Since February 27, 2004 Submittal)

NOTE: If a conflict exists between these tables and the associated Design Basis Calculation, the information from the calculation is to be used.

Table 1: Limerick General Parameters and Methods Applicable to Design Basis Accidents				
General AST Analysis Parameter or Method	**Currently Licensed Value or Method**	**Value or Method Submitted in February 27, 2004 AST Application**	**Final Supplemented Value or Method Used in AST Analysis**	**Justification for Change**
Core Power Level	3458 MWth (current) 100% power is 3458 MWth, 102% power with RG 1.49 instrument uncertainty allowance is 3527 MWth.	3458 MWth (current) 100% power is 3458 MWth, 102% power with RG 1.49 instrument uncertainty allowance is 3527 MWth.	3458 MWth (current) 100% power is 3458 MWth, 102% power with RG 1.49 instrument uncertainty allowance is 3527 MWth.	No changes since 02/27/04 submittal.
Core Source Terms	TID-14844	ORIGEN2-based inventory includes the 60 isotopes in the standard RADTRAD library.	ORIGEN2-based inventory includes the 60 isotopes in the standard RADTRAD library.	The ORIGEN 2.1 based source terms developed for AST application are applied to Limerick AST accident analysis, on a Curie per MWt basis.
Dose Conversion Factors for Thyroid Inhalation, Whole Body, and TEDE Dose	Thyroid – TID-14844 Whole Body – TID-14844 TEDE – N/A	Federal Guidance Reports 11 & 12 (RADTRAD input file) (TEDE only)	Federal Guidance Reports 11 & 12 (RADTRAD input file) (TEDE only)	RG 1.183 No changes since 02/27/04 submittal.
Distance to EAB	731 meters	731 meters	731 meters	No changes
EAB Dispersion Factors – Ground Level Release (LOCA) 0 – 2 hr (or worst two hours)	N/A – New X/Qs needed for AST	<u>(sec/m^3)</u> 3.18E-04	<u>(sec/m^3)</u> 3.18E-04	(PAVAN) Changed from currently licensed to AST basis. No changes since 02/27/04 submittal.
Distance to LPZ	2043 meters	2043 meters	2043 meters	No changes

Table 1: Limerick General Parameters and Methods Applicable to Design Basis Accidents				
General AST Analysis Parameter or Method	**Currently Licensed Value or Method**	**Value or Method Submitted in February 27, 2004 AST Application**	**Final Supplemented Value or Method Used in AST Analysis**	**Justification for Change**
LPZ Dispersion Factors – Ground Level Release (LOCA) 0 – 8 hr 8 – 24 hr 1 – 4 day 4 – 30 day	N/A – New X/Qs needed for AST	(sec/m^3) 5.79E-05 4.10E-05 1.95E-05 6.68E-06	(sec/m^3) 5.79E-05 4.10E-05 1.95E-05 6.68E-06	(PAVAN) Changed from currently licensed to AST basis. No changes since 02/27/04 submittal.
CR Dispersion Factors – Ground Level Release, conservatively via the North Vent Stack (LOCA) 0 – 2 hr 2 – 8 hr 8 – 24 hr 1 – 4 day 4 – 30 day	N/A – New X/Qs needed for AST	(sec/m^3) 6.88E-03 5.17E-03 2.04E-03 1.29E-03 9.63E-04	(sec/m^3) 6.88E-03 5.17E-03 2.04E-03 1.29E-03 9.63E-04	(ARCON96, RG 1.194) Changed from currently licensed to AST basis. No changes since 02/27/04 submittal.
Control Room Volume	126,000 ft^3	126,000 ft^3	126,000 ft^3	No changes
Control Room Filtration System Initiation (Automatic)	Auto initiation upon high radiation signal in upstream ductwork.	Manual initiation assumed at 30 minutes.	Auto initiation upon high radiation signal in upstream ductwork.	Supported by design basis analysis and testing. The requested change in the 02/27/04 Application to no longer take credit for automatic initiation during the initial 30 minutes of the accident is rescinded.
Filtered Intake Rate - Rad Mode	Maximum of 525 cfm	Maximum of 525 cfm	Maximum of 525 cfm	No Changes

Table 1: Limerick General Parameters and Methods Applicable to Design Basis Accidents				
General AST Analysis Parameter or Method	**Currently Licensed Value or Method**	**Value or Method Submitted in February 27, 2004 AST Application**	**Final Supplemented Value or Method Used in AST Analysis**	**Justification for Change**
CR Intake Filter Efficiency - Rad Mode	95% for all iodines	99% for aerosols; 80% for elemental and organic iodines	99% for aerosols; 95% for elemental and organic iodines	The 02/27/04 proposed TS change is rescinded. New values are consistent with TS Testing requirements.
Filtered Intake Rate – Chlorine Isolation Mode	0 cfm	0 cfm	0 cfm	No changes
CR Recirculation Filter Efficiency (Chlorine Isolation Mode)	95% for all iodines	99% for aerosols; 80% for elemental and organic iodines	99% for aerosols; 95% for elemental and organic iodines	The 02/27/04 proposed TS change is rescinded. New values are consistent with TS Testing requirements.
CR Unfiltered Inleakage	50 cfm (rad mode); 525 cfm (chlorine isolation mode)	525 cfm (rad mode); 525 cfm (chlorine isolation mode)	275 cfm (rad mode); 275 cfm (chlorine isolation mode)	Supported by design basis analysis and tracer gas testing.

Table 2: Limerick Parameters and Methods Applicable to Loss Of Coolant Accident (LOCA)				
LOCA AST Analysis Parameter or Method	**Currently Licensed Value or Method**	**Value or Method Submitted in February 27, 2004 AST Application**	**Final Supplemented Value or Method Used in AST Analysis**	**Justification for Change**
Releases to Containment	TID-14844 Instantaneous release into containment at T = 0	No Core Activity release for first 2 minutes. Release Fractions and Timing per NUREG-1465.	No Core Activity release for first 2 minutes. Release Fractions and Timing per NUREG-1465.	RG 1.183 No changes since 02/27/04 submittal.
Natural Deposition in Containment	TID-14844	Powers' algorithm built into RADTRAD 10 percentile value	Powers' algorithm built into RADTRAD 10 percentile value	No changes since 02/27/04 submittal.
Containment Spray Removal Mechanism	Not credited	Not credited	Not credited	No changes
Drywell Volume	243,580 ft^3	243,580 ft^3	0.95 x 243,580 ft^3	Supported by design basis analysis. 5% reduction in drywell volume for future conservatism.
Minimum Suppression Pool Air Volume	147,670 ft^3	147,670 ft^3	147,670 ft^3	No changes
Minimum Suppression Pool Water Volume	118,655 ft^3	118,655 ft^3	118,655 ft^3	No changes
Reactor Coolant Volume (for dilution of ECCS water)	13,108 ft^3 @ 552.6 °F = 9,663 ft^3 @ 95.0 °F	13,108 ft^3 @ 552.6 °F = 9,663 ft^3 @ 95.0 °F	13,108 ft^3 @ 552.6 °F = 9,663 ft^3 @ 95.0 °F	No changes
Secondary Containment Volume	1,800,000 ft^3 (below refueling floor)	1,800,000 ft^3 (below refueling floor)	1,800,000 ft^3 (below refueling floor)	No changes since 02/27/04 submittal. Analysis uses 900,000 ft^3 due to 50% mixing.
Fraction of Secondary Containment Available for Mixing	0.5	0.5	0.5	Secondary containment mixing is provided by RERS. Only ½ of this volume is

Table 2: Limerick Parameters and Methods Applicable to Loss Of Coolant Accident (LOCA)				
LOCA AST Analysis Parameter or Method	**Currently Licensed Value or Method**	**Value or Method Submitted in February 27, 2004 AST Application**	**Final Supplemented Value or Method Used in AST Analysis**	**Justification for Change**
				assumed for mixing. No changes since 02/27/04 submittal.
Reactor Enclosure Recirculation System (RERS) Flow Rate	60,000 cfm, available after 3 minutes (with an assumed Loss of Offsite Power)	30,000 cfm to 66,000 cfm, available after 3 minutes (with an assumed Loss of Offsite Power)	60,000 cfm +/- 10%, available after 3 minutes (with an assumed Loss of Offsite Power) 54,000 cfm used in analysis	The 02/27/04 proposed TS change is rescinded. Analysis considers the tolerance instead of the nominal value.
Reactor Enclosure Recirculation System (RERS) Filter Efficiency	99% Particulate 30% Organic 95% Elemental	70% Aerosol 70% Elemental/Organic Iodine	99% Aerosol 95% Elemental/Organic Iodine	The 02/27/04 proposed TS change is rescinded. New values are consistent with TS Testing requirements.
Secondary Containment Drawdown Time	15.5 minutes	15.5 minutes	15.5 minutes	No changes
SGTS Flow Rate	3,000 cfm (pre-drawdown) 2,500 cfm (post drawdown)	3,000 cfm (pre-drawdown) 2,500 cfm (post drawdown)	3,000 cfm (pre-drawdown) 2,500 cfm (post drawdown)	No changes
SGTS Filter Efficiency HEPA: Charcoal:	 99% 99%	 97.5% 97.5%	 99% 99%	The 02/27/04 proposed TS change is rescinded.

Table 2: Limerick Parameters and Methods Applicable to Loss Of Coolant Accident (LOCA)				
LOCA AST Analysis Parameter or Method	**Currently Licensed Value or Method**	**Value or Method Submitted in February 27, 2004 AST Application**	**Final Supplemented Value or Method Used in AST Analysis**	**Justification for Change**
Primary Containment Total Leak Rate, excluding MSIV Leakage (L_a)	0.5% per day	0.5% / day, 0 – 1 day 0.25% / day, 1 – 30 days	0.5% / day, 0 – 1 day 0.25% / day, 1 – 30 days	No changes since 02/27/04 submittal. Reduction after 24 hours supported by design analysis.
Aerosol Natural Deposition Coefficients Used in the Containment	TID-14844	Credit is taken for natural deposition of aerosols based on equations for the Powers' model in NUREG/CR 6189 and input directly into RADTRAD as natural deposition time dependent lambdas. No credit is assumed for natural deposition of elemental or organic iodine, or for suppression pool scrubbing.	Credit is taken for natural deposition of aerosols based on equations for the Powers' model in NUREG/CR 6189 and input directly into RADTRAD as natural deposition time dependent lambdas. No credit is assumed for natural deposition of elemental or organic iodine, or for suppression pool scrubbing.	No changes since 02/27/04 submittal. Containment aerosol deposition per RADTRAD Powers' Model using 10% uncertainty. No elemental or organic iodine deposition credited.
Main Steam Line Deposition/Plateout Total leakage for all lines: Maximum for any one line: Test Pressure:		200 scfh 100 scfh Reduced to 55.1% of these values after 24 hours 22 psig	200 scfh 100 scfh Reduced to 55.1% of these values after 24 hours 22 psig	MSL A, B, C, and D piping volumes and inside surface areas were derived from system isometric drawings. Lines A and B were found to be the worst-case lines, with inboard line B determined to be the worst case postulated break location. These lines have a leak acceptance criteria of 100 scfh per line, with 200 total for all lines.

Table 2: Limerick Parameters and Methods Applicable to Loss Of Coolant Accident (LOCA)				
LOCA AST Analysis Parameter or Method	Currently Licensed Value or Method	Value or Method Submitted in February 27, 2004 AST Application	Final Supplemented Value or Method Used in AST Analysis	Justification for Change
				Although a Recirculation-Suction Line Break (RSLB) is historically assumed to be the most limiting non-mechanistic source of a LOCA, a Main Steam Line Break (MSLB) of the worst-case inboard line is conservatively assumed, in order to artificially limit deposition credit. Outboard MSIV failure is assumed as the Single Active Failure since this maximizes the volume of piping in which the fluid is depressurized, minimizing deposition. Only horizontal piping is credited, and only the horizontal projected area of the pipe is used as the settling area for aerosols. For elemental iodine, all piping and surfaces are credited. Two nodes, one for inboard, one for penetration and outboard pipe, are used to

Table 2: Limerick Parameters and Methods Applicable to Loss Of Coolant Accident (LOCA)				
LOCA AST Analysis Parameter or Method	**Currently Licensed Value or Method**	**Value or Method Submitted in February 27, 2004 AST Application**	**Final Supplemented Value or Method Used in AST Analysis**	**Justification for Change**
				model each assessed steam line. Only seismically rugged or qualified piping/equipment is credited. This piping is from the reactor vessel to the turbine stop valves and Condenser.
ECCS Leakage into Secondary Containment Leak Rate	5 gpm	5 gpm (2 times the maximum allowable admin limit of 2.5 gpm)	5 gpm (2 times the maximum allowable admin limit of 2.5 gpm)	No changes
ECCS Fraction Flashed ECCS Leakage Filtered by SGTS: Leakage Duration:	10% Yes 0 – 30 days	1.36% Yes 0 – 30 days	10% Yes 0 – 30 days	The change proposed in 02/27/04 submittal is rescinded. 10% now used in design analysis. No changes No changes
Control Room Operator Shine Dose (LOCA)	Shine dose includes TID-14844 isotopes.	Shine dose includes TID-14844 isotopes (Same dose assumed for conservatism).	Shine dose due to airborne activity includes the 60 RG 1.183 isotopes. Shine doses due to water-borne activity in piping include these isotopes plus additional isotopes with potentially significant CR dose consequences (excluding noble gases).	Supported by design basis analysis.

Table 3: Limerick Parameters and Methods Applicable to Suppression Pool pH Transient Analysis				
Suppression Pool pH Transient AST Analysis Parameter or Method	**Currently Licensed Value or Method**	**Value or Method Submitted in February 27, 2004 AST Application**	**Final Supplemented Value or Method Used in AST Analysis**	**Justification for Change**
Maximum Suppression Pool Water Volume	134,600 ft^3 at 95°F	134,600 ft^3 at 95°F	134,600 ft^3 at 95°F	No changes
Maximum Dilution Water Volume Post-LOCA (Analysis Value)	Not previously required for pH control	175,000ft^3 at 95°F	175,000 ft^3 at 95°F	No changes since 02/27/04 submittal.
Minimum Mass of Sodium Pentaborate Delivered Within 13 hours Post-LOCA	Not previously required for pH control	1313 lb	1313 lb	No changes since 02/27/04 submittal.
Minimum Volume of Sodium Pentaborate Solution Delivered	Not previously required for pH control	1500 gal	1500 gal	No changes since 02/27/04 submittal.
SLCS credit	Not previously required for pH control	Credited for pH control full sodium pentaborate injection into suppression pool within 13 hours of accident initiation and pool is well mixed.	Credited for pH control full sodium pentaborate injection into suppression pool within 13 hours of accident initiation and pool is well mixed.	SLCS credited per new design basis analysis performed for the 02/27/04 submittal. No change since 02/27/04 submittal.
Iodine Re-evolution from Containment Liquids (none if pH maintained above 7)	N/A – Not required for TID-14844	None. Suppression pool pH stays above 7 during entire accident.	None. Suppression pool pH increases to above 7 in the gap phase and remains above 7 during the remaining period of the accident.	New design basis analysis performed for the 02/27/04 submittal. Clarification provided since 02/27/04 submittal.

Table 4: Limerick Parameters and Methods Applicable to Control Rod Drop Accident (CRDA)				
CRDA AST Analysis Parameter or Method	**Currently Licensed Value or Method**	**Value or Method Submitted in February 27, 2004 AST Application**	**Final Supplemented Value or Method Used in AST Analysis**	**Justification for Change**
Damaged Fuel Releases to Reactor Coolant Fraction of failed fuel that melts Fuel Bundles in Core Fuel peaking factor	850 fuel rods failed - (47,368 fuel rods in core) 0.77% Value used in NEDO-31400A (Section 2.1, Page 2) and UFSAR 15.4.9.5.1.1 764 1.5	1200 fuel rods failed - (66,720 fuel rods in core) 0.77% Value used in NEDO-31400A (Section 2.1, Page 2) 764 1.7	1200 fuel rods failed - (66,720 fuel rods in core) 0.77% Value used in NEDO-31400A (Section 2.1, Page 2) 764 1.7	No changes since 02/27/04 submittal. No changes. No changes No changes since 02/27/04 submittal.
Radioactivity transport pathway	Carryover with steam to condenser and leakage from condenser to the environment.	Carryover with steam to condenser and leakage from condenser to the environment.	Carryover with steam to condenser and leakage from condenser to the environment.	No changes
Fuel released activity carried to condenser	Noble Gases: 100% Iodines: 10% Remaining radionuclides: 1%	Noble Gases: 100% Iodines: 10% Remaining radionuclides: 1%	Noble Gases: 100% Iodines: 10% Remaining radionuclides: 1%	No changes
CREFAS System Initiation	Not credited, normal CR HVAC operation assumed.	Not credited, normal CR HVAC operation assumed.	Not credited. A CR unfiltered intake rate of 1 air change per minute is assumed for conservatism. This bounds any expected	Supported by design basis analysis.

Table 4: Limerick Parameters and Methods Applicable to Control Rod Drop Accident (CRDA)				
CRDA AST Analysis Parameter or Method	**Currently Licensed Value or Method**	**Value or Method Submitted in February 27, 2004 AST Application**	**Final Supplemented Value or Method Used in AST Analysis**	**Justification for Change**
			amount of unfiltered inleakage in the normal mode of operation.	
Activity in condenser release to environment	Noble Gases 100% Iodines 10% Remaining radionuclides 1%, with released Iodine species as follows: 97% elemental and 3% organic	Noble Gases 100% Iodines 10% Remaining radionuclides 1%, with released Iodine species as follows: 97% elemental and 3% organic	Noble Gases 100% Iodines 10% Remaining radionuclides 1%, with released Iodine species as follows: 97% elemental and 3% organic	No changes
Airborne condenser activity leakrate to environment	1% per day for 24 hours, then 0.	1% per day for 24 hours, then 0.	1% per day for 24 hours, then 0.	No changes
Credit for activity decay during condenser residence	Yes	Yes	Yes	No changes
Credit for activity decay after release to environment	No	No	No	No changes
Credit for accident termination features	Yes, isolation of Mechanical Vacuum Pumps by main steam line rad monitor upon CRDA initiation.	Yes, isolation of Mechanical Vacuum Pumps by main steam line rad monitor upon CRDA initiation.	Yes, isolation of Mechanical Vacuum Pumps by main steam line rad monitor upon CRDA initiation.	No changes
Credit for Holdup in Turbine Building	No	No	No	No changes
Atmospheric Dispersion	N/A – New X/Qs calculated for AST	See Table 1	See Table 1	No changes since 02/27/04 submittal.

Table 5: Limerick Parameters and Methods Applicable to the Main Steam Line Break (MSLB) Accident				
MSLB AST Analysis Parameter or Method	**Currently Licensed Value or Method**	**Value or Method Submitted in February 27, 2004 AST Application**	**Final Supplemented Value or Method Used in AST Analysis**	**Justification for Change**
MSIV closure time	5.5 seconds	10.5 seconds	5.5 seconds	Closure time increase was rescinded in a supplement dated October 25, 2004.
Break Characteristic and Coolant Release Discharged mass	108,785 lb (88,333 lb liquid and 20,452 lb steam)	206,933 lb (186,481 lb liquid and 20,452 lb steam) for 10.5 seconds MSIV closure time	140,000 lb liquid (no steam) (40% flashing)	This change ensures that the discharged mass is maximized (conservative application of SRP 15.6.4, Paragraph III.2.a). The value of 140,000 lb liquid is only used for control room and offsite dose analysis. The values for coolant released listed in the current licensing basis remain the basis for no fuel damage.
Release characteristics	All radioactivity in the released coolant is assumed to be released to the atmosphere instantaneously as a ground level release. No credit is assumed for plateout, holdup, or dilution within facility buildings.	All radioactivity in the released coolant is assumed to be released to the atmosphere instantaneously as a ground level release. No credit is assumed for plateout, holdup, or dilution within facility buildings.	All radioactivity in the released coolant is assumed to be released to the atmosphere instantaneously as a ground level release. No credit is assumed for plateout, holdup, or dilution within facility buildings.	No changes
Amount of Fuel Damage Projected	None	None	None	No changes

Table 5: Limerick Parameters and Methods Applicable to the Main Steam Line Break (MSLB) Accident				
MSLB AST Analysis Parameter or Method	**Currently Licensed Value or Method**	**Value or Method Submitted in February 27, 2004 AST Application**	**Final Supplemented Value or Method Used in AST Analysis**	**Justification for Change**
Analyzed Cases and Activity Releases	Two cases analyzed, one without and one with postulated iodine spike, using TS reactor coolant activity limits and reactor coolant iodine activity concentrations. Maximum equilibrium value 0.2 uCi/gm Pre-accident spike activity 4.0 uCi/gm	Two cases analyzed, one without and one with postulated iodine spike, using TS reactor coolant activity limits and reactor coolant iodine activity concentrations. Maximum equilibrium value 0.2 uCi/gm Pre-accident spike activity 4.0 uCi/gm	Two cases analyzed, one without and one with postulated iodine spike, using TS reactor coolant activity limits and reactor coolant iodine activity concentrations. Maximum equilibrium value 0.2 uCi/gm Pre-accident spike activity 4.0 uCi/gm Cesium activities are now considered in addition to iodine.	The activity of the cloud is based on the total mass of water released from the break. Supported by design basis analysis. Cesium iodide (CsI) constitutes 95% of the total (radioactive and stable) iodine release
Iodine Species Released	RG 1.5	95% CsI as aerosol 4.85% elemental 0.15% organic	95% CsI as aerosol 4.85% elemental 0.15% organic	RG 1.183 No changes since 02/27/04 submittal.
Offsite Atmospheric Dispersion: EAB: LPZ:	 Updated for AST Updated for AST	<u>(sec/m^3)</u> 4.77E-04 1.89E-04	<u>(sec/m^3)</u> 4.77E-04 1.89E-04	No changes since 02/27/04 submittal. Used method described in RG 1.5

Table 6: Limerick Parameters and Methods Applicable to the Fuel Handling Accident (FHA)				
FHA AST Analysis Parameter or Method	**Currently Licensed Value or Method**	**Value or Method Submitted in February 27, 2004 AST Application**	**Final Supplemented Value or Method Used in AST Analysis**	**Justification for Change**
Minimum time after shutdown required before movement of recently irradiated fuel	24 hours	24 hours	24 hours	No changes
Fuel bundle peak burnup	N/A	Not to exceed 62 GWD/MTU.	Not to exceed 62 GWD/MTU.	No changes since 02/27/04 submittal.
Maximum linear heat generation rate.	N/A	6.3 kw/ft average power for fuel exceeding 54 GWD/MTU	6.3 kw/ft average power for fuel exceeding 54 GWD/MTU	RG 1.183 No changes since 02/27/04 submittal.
Spent fuel source terms are based on reactor core source terms for LOCA	TID-14844	ORIGEN2-based inventory Includes the 60 isotopes in the standard RADTRAD library (same as for LOCA).	ORIGEN2-based inventory Includes the 60 isotopes in the standard RADTRAD library (same as for LOCA).	Supported by design basis analysis. No changes since 02/27/04 submittal.
Radial Peaking Factor (Bounding dose basis)	1.5	1.7	1.7	Supported by design basis analysis. No changes since 02/27/04 submittal. Increase from licensing basis in UFSAR for conservatism in current dose analysis.
Bounding Analyzed Water Depths above damaged fuel for drop in reactor well	23 feet	23 feet	23 feet	Supported by design basis analysis. No changes since 02/27/04 submittal.

Table 6: Limerick Parameters and Methods Applicable to the Fuel Handling Accident (FHA)				
FHA AST Analysis Parameter or Method	**Currently Licensed Value or Method**	**Value or Method Submitted in February 27, 2004 AST Application**	**Final Supplemented Value or Method Used in AST Analysis**	**Justification for Change**
Accident Release Duration	2 hours	2 hours Activity reaching the refuel floor airspace will effectively be all (99.97%) exhausted within two hours by using an artificially high exhaust rate. This also provides an allowance for uneven mixing in the refuel floor airspace.	2 hours Activity reaching the refuel floor airspace will effectively be all (99.97%) exhausted within two hours by using an artificially high exhaust rate. This also provides an allowance for uneven mixing in the refuel floor airspace.	No changes since 02/27/04 submittal. Supported by design analysis.
Exhaust pathway	Filtration through SGTS as elevated release	Through the normal exhaust to the Reactor Enclosure South Stack. No credit is taken for filtration by the SGTS.	Through the normal exhaust to the Reactor Enclosure South Stack. No credit is taken for filtration by the SGTS.	Supported by design basis analysis. No changes since 02/27/04 submittal.
Credit for operation of CREFAS	Credited	No credit is taken for the operation of the CREFAS during the FHA. Normal intake and outflow are credited.	No credit is taken for the operation of the CREFAS during the FHA. To eliminate uncertainty regarding unfiltered inleakage in the normal mode of CR HVAC operation, unfiltered intake and outflow at 1 CR air change per minute (126,000 cfm) is assumed.	Supported by design basis analysis. An artificially high unfiltered intake rate is now assumed.
Fuel Assembly Configuration (Bounding dose basis)	8x8 in a 62-pin bundle	10x10 in a 87.33 equivalent-pin bundle	8x8 in a 62-pin bundle	No changes from current licensing dose basis in UFSAR. The change back to

Table 6: Limerick Parameters and Methods Applicable to the Fuel Handling Accident (FHA)				
FHA AST Analysis Parameter or Method	**Currently Licensed Value or Method**	**Value or Method Submitted in February 27, 2004 AST Application**	**Final Supplemented Value or Method Used in AST Analysis**	**Justification for Change**
				an 8x8 62 pin bundle provides additional margin.
Number of pins Damaged (Bounding dose basis)	212 total pins damaged (bounding)	172 total pins damaged (bounding)	212 total pins damaged (bounding)	No changes from current licensing dose basis in UFSAR. The change back to 212 pins provides additional margin.
FHA Radionuclide Inventory		ORIGEN 2.1 analysis. The 60 isotopes forming the standard RADTRAD library, with decay to 24 hours are used. Gap activities are per R.G. 1.183.	ORIGEN 2.1 analysis. The 60 isotopes forming the standard RADTRAD library, with decay to 24 hours are used. Gap activities are per R.G. 1.183.	Supported by design basis analysis. No changes since 02/27/04 submittal.
Underwater Decontamination Factor	Iodine DF = 100 Noble Gas DF = 1 Particulate DF = infinity	Iodine DF = 200, corresponding to a 23-ft water depth for bounding accident of drop over reactor. Noble Gas DF = 1 Particulates (cesiums and rubidiums) DF = infinity	Iodine DF = 200, corresponding to a 23-ft water depth for bounding accident of drop over reactor. Noble Gas DF = 1 Particulates (cesiums and rubidiums) DF = infinity	Supported by design basis analysis and RG 1.183. No changes since 02/27/04 submittal.
Iodine chemical distribution	97% Elemental 3% organic	95% CsI, instantaneously dissociating in the pool water and re-evolving as elemental iodine, 4.85% elemental; 0.15% organic	95% CsI, instantaneously dissociating in the pool water and re-evolving as elemental iodine 4.85% elemental; 0.15% organic	RG 1.183 No changes since 02/27/04 submittal.

Table 6: Limerick Parameters and Methods Applicable to the Fuel Handling Accident (FHA)				
FHA AST Analysis Parameter or Method	**Currently Licensed Value or Method**	**Value or Method Submitted in February 27, 2004 AST Application**	**Final Supplemented Value or Method Used in AST Analysis**	**Justification for Change**
Secondary Containment automatic isolation and filtration	Credited	Not credited for all fuel with 24 hours decay.	Not credited for all fuel with 24 hours decay.	Supported by design basis analysis. RG 1.183, TSTF-51 No changes since 02/27/04 submittal.
Mitigation by CREFAS system	Credited	Not credited for all fuel with 24 hours decay.	Not credited for all fuel with 24 hours decay. An unfiltered intake rate of 1 control room volume per minute is assumed (126,000 cfm) to conservatively bound any amount of control room unfiltered inleakage.	Supported by design basis analysis.
Refuel Floor Normal Ventilation rate and volume	Secured upon initiation of SGTS	Approximately 6 air changes per hour and an artificial value of 100 ft^3 is used for simplicity. This evacuates 99.9994% of all activity within 2 hours.	Approximately 6 air changes per hour and an artificial value of 100 ft^3 is used for simplicity. This evacuates 99.9994% of all activity within 2 hours.	Supported by design basis analysis. No changes since 02/27/04 submittal.
CR Worst Case Release Point Basis Dispersion Factors 0 – 2 hr:	SGTS	Normal South Stack (ground level, unfiltered) 1.26E-03 sec/m3	Normal South Stack (ground level, unfiltered) 1.26E-03 sec/m3	Supported by design basis analysis. No changes since 02/27/04 submittal.

Table 6: Limerick Parameters and Methods Applicable to the Fuel Handling Accident (FHA)				
FHA AST Analysis Parameter or Method	**Currently Licensed Value or Method**	**Value or Method Submitted in February 27, 2004 AST Application**	**Final Supplemented Value or Method Used in AST Analysis**	**Justification for Change**
EAB Worst Case Release Point Basis and Dispersion Factors 0 – 2 hr:	SGTS	Normal RB exhaust stack (ground level, unfiltered) 3.18E-04 sec/m3	Normal RB exhaust stack (ground level, unfiltered) 3.18E-04 sec/m3	Supported by design basis analysis. No changes since 02/27/04 submittal.
LPZ Worst Case Release Point Basis Dispersion Factors 0 – 2 hr:	SGTS	Normal RB exhaust stack 1.15E-04 sec/m3	Normal RB exhaust stack 1.15E-04 sec/m3	Supported by design basis analysis. No changes since 02/27/04 submittal.

Table 7: Limerick MSIV Leakage, AEB 98-03, and NEDC-31858P Parameters and Methods		
AEB 98-03 or NEDC-31858P Parameters and Methods	**Final Supplemented Value or Method Used in AST Analysis**	**Justification**
Leakage Distribution	50% in MS Line A (Shortest Unbroken line) 50% in MS Line B (Shortest line when inboard piping assumed broken and unavailable)	Leakage limits are 200 scfh total, and 100 scfh for any one line. Maximizing flow through shorter lines minimizes piping deposition credit. Lines selected for minimization of effective filter efficiencies.
Nodalization for AEB-98-03; Single Active Failure Assumptions	Two-node treatment is used for each steam line in which flow occurs. The first node is from the reactor vessel to the inboard MSIV. The second node is from the inboard MSIV to the Condenser. Steam line piping is seismically rugged through this boundary as per the current UFSAR Alternate Drain Path discussion. The outboard MSIV is assumed to fail on both steam lines in which flow is assumed. See note at end of this table for discussion of condenser modeling.	Outboard MSIV failure is selected since this maximizes the volume of piping in which the fluid is depressurized. This in turn minimizes deposition and this treatment is used for all steam lines. Although the LGS design basis line break associated with the non-mechanistic R.G. 1.183 source terms is a recirculation line break, a steam line break inside containment is assumed in order to ensure conservative treatment of deposition. That is assumed to result in the unavailability for deposition of inboard piping on the broken steam line. The inboard node for the broken steam line is not credited. For conservatism and for consistency with AEB-98-03, only two nodes are used in any one steam line.
Well-mixed modeling for aerosol gravitational settling and elemental iodine deposition.	The formulations for effective filter efficiencies in piping segments that are used in the LGS AST application are from AEB-98-03, Appendix A "Use of Plug-Flow and Well-Mixed Models for Fission Product Deposition in the Main Steam Line for the Perry Assessment".	For conservatism, a well-mixed model is used for both inboard and outboard piping nodes. The AEB-98-03 Appendix A formulas are for a well-mixed treatment of a piping segment or node and are presented for use with gravitational settling of aerosols. This formulation would also be applicable for elemental iodine removal when the appropriate deposition velocities and deposition areas are used.
Aerosol Settling Velocities	Settling Velocities used for gravitational settling use a 20-group probability distribution based on the AEB-98-03 recommended distribution parameters.	NRC's AEB-98-03 indicates *"The staff believes that, at this time, a well-mixed model is more appropriate than a plug flow model for settling in the main steam line. However complete mixing may not occur along the entire length of the pipe and, in some pipe segments,*

Table 7: Limerick MSIV Leakage, AEB 98-03, and NEDC-31858P Parameters and Methods		
AEB 98-03 or NEDC-31858P Parameters and Methods	**Final Supplemented Value or Method Used in AST Analysis**	**Justification**
		plug flow may exist. Given the conservatism associated with using a well-mixed model for the entire length of the pipe and a number of additional conservatisms inherent in the piping depositions analysis, use of a 10th percentile settling velocity with a well-mixed model is not appropriate. Additional conservatisms include additional deposition by thermophoresis, diffusiophoresis, and flow irregularities; addition deposition as a result of hygroscopicity and possible plugging of the leaking MSIV by aerosols. Given the conservatism of the well-mixed assumption, we believe it is acceptable then to utilize median values (as compared to more conservative values) for deposition parameters." LGS is somewhat different than Perry in that piping downstream from the outboard MSIV to the condenser is credited at LGS. The NRC staff has questioned whether crediting the same settling velocities throughout the piping system, and treatment of the downstream piping as a third node was adequately conservative. In response, LGS: 1. Has combined penetration piping and downstream piping into a single outboard node; 2. Uses a 20 group probability distribution on settling velocities with efficiencies determined for each group and a net weighted average efficiency used, a process that is significantly more conservative than use of a median settling velocity; 3. Takes no credit for aerosol settling after 24 hours. Other phenomena, such as effects of depletion over time of more easily settled particle sizes are considered to be adequately addressed by the above conservatisms and the significant residual conservatism mentioned in the original AEB-98-03 conclusions quoted above.
Elemental Iodine	Elemental iodine deposition velocities,	Elemental iodine deposition is analyzed using the well-mixed model of

Table 7: Limerick MSIV Leakage, AEB 98-03, and NEDC-31858P Parameters and Methods		
AEB 98-03 or NEDC-31858P Parameters and Methods	**Final Supplemented Value or Method Used in AST Analysis**	**Justification**
Removal Parameters	resuspension rates, and fixation rates are taken from RG 1.183, Appendix A, Reference A-9.	AEB-98-03, Appendix A, where settling velocities and deposition velocities are treated as analogous properties. The impacts of resuspension from deposited elemental iodine have been evaluated where all re-evolved iodine is treated as organic iodine and instantly released. The dose impacts are minimal.
Actual Inboard Piping Volumes and Surface Areas	For Aerosol Settling LINE Vol (ft^3) Area (ft^2) MS A 110 69.82 MS B 159 100.33 (Line B assumed broken and not credited) For Elemental Iodine Deposition LINE Vol (ft^3) Area (ft^2) MS A 258 512 MS B 306 608 (Line B assumed broken and not credited)	For Aerosols - only includes horizontal piping. Settling area is projected horizontal surface area of horizontal piping only. For Elemental iodine - includes total piping area and volume.
Outboard Piping Volumes and Surface Areas	For Aerosol Settling LINE Vol (ft^3) Area(ft^2) MS A 1133 716.82 MS B 1001 633.73 For Elemental Iodine Deposition LINE Vol (ft^3) Area(ft^2) MS A 1182 2350 MS B 1051 2089	For Aerosols - only includes horizontal piping. Settling area is projected horizontal surface area of horizontal piping only. For Elemental iodine - includes total piping area and volume.
Associated Containment Leak Rate	Leak Rate (cfh) = Leak Rate Acceptance Criterion (scfh) * $[14.7/(P_{MSIVtest}+14.7)]*(276+460)/(68+460)$ where $P_{MSIVtest}$ = 22 psig and 276 degrees F is peak drywell temperature at 2 minutes, per Calculation LM-0646 Rev 1,	LGS MSIVs are tested at the $P_{MSIVtest}$ of 22 psig. Leak rates are determined per standard 10CFR50 Appendix J practice for all PCIVs.

Table 7: Limerick MSIV Leakage, AEB 98-03, and NEDC-31858P Parameters and Methods		
AEB 98-03 or NEDC-31858P Parameters and Methods	**Final Supplemented Value or Method Used in AST Analysis**	**Justification**
	Attachment E. Leak Rates LINE (scfh) (cfh) (cfm) MS A 100 55.83 0.931 MS B 100 55.83 0.931	
Fluid Temperature for deposition velocity and flow rate assessments	550 °F from 0 to 24 hours 410 °F from 24 to 96 hours 200 °F from 96 to 720 hours	The deposition velocities, resuspension rates and fixation rates are temperature dependent. Outboard flow rates are also temperature dependent. Credit for temperature reductions in the current LGS AST LOCA analysis are very limited and conservative. The full normal operating pipe wall temperature is used for the first 24 hours. The value applicable at 24 hours is used from then until 96 hours, and the 96-hour value is used from then until 720 hours. The generic BWR cooldown curve has been found applicable to LGS, and the impact of decay heat from deposited radioactivity on steam piping has been found to be negligible.
Inboard Piping Node Flow Rate	LINES Flow Rate (cfm) MS A 0.931 MS B N/A (assumed broken)	Values are as determined from containment leak rate.
Outboard Piping Node Flow Rate	LINES Flow Rate (cfm) 0-24 hours MS A 3.188 MS B 3.188 24-96 hours MS A 1.513 MS B 1.513 96-720 hours MS A 1.148 MS B 1.148	Values are conservatively expanded based on outside pressure with fluid temperatures at conservative pipe wall temperatures for the accident duration, compared with standard conditions at 68 degrees F. Therefore, inboard flow rates are multiplied by: 100 scfh * 1 hr / 60 min * (550+460)/(68+460) = 3.188 cfm for 0-24 hours 100 scfh * 1 hr / 60 min * (410+460)/(68+460) * .551 = 1.513 cfm for 24 to 96 hours 100 scfh * 1 hr / 60 min * (200+460)/(68+460) * .551 = 1.148 cfm for 96 to 720 hours

Table 7: Limerick MSIV Leakage, AEB 98-03, and NEDC-31858P Parameters and Methods		
AEB 98-03 or NEDC-31858P Parameters and Methods	**Final Supplemented Value or Method Used in AST Analysis**	**Justification**
Leak Rate and flow rates after 24 hours	Leak rates and flow rates are assumed to be reduced to 55.1% of initial values after 24 hours. No further reductions are taken between 24 and 720 hours.	MSIV leakage is assumed to be reduced to a conservative 55.1% of the Technical Specification value at 24 hours, based on the following square root leakage rate correlation with containment pressure, MSIV leakage testing performed at the LGS $P_{MSIVtest}$ of 22 psig, and LGS calculation LM-0646 Revision 1, Attachment E indicating a containment pressure of approximately 21.4 psia (6.7 psig) is reached at 24 hours for minimum ECCS in operation: $(6.7 / 22)^{0.5} = 0.551$
Seismic Design of Credited Piping	Credited MS piping is that piping which has been evaluated to withstand a design basis SSE.	Piping and components from the reactor vessel to the outside of the outboard MSIVs are Seismic Category 1 and safety related due to their functions. Downstream piping through to the condenser is seismically rugged to assure secondary containment gas control boundary integrity. Therefore, all of this piping is available for aerosol gravitational settling or elemental iodine deposition.

Notes for Table 7:

The pre-AST, TID-14844 based analyses of the consequences of MSIV leakage are based on the methodology documented in NEDC-31858P, as applied to LGS in the license changes to eliminate the requirement for a MSIV-LCS. That analysis included the use of steam line drains as an alternative drain path to the condenser and the demonstration of seismic qualification or seismic ruggedness of credited equipment.

That historical analysis was based on the then standard 91 / 4 / 5 elemental/organic/particulate release percentages, but with the elemental and particulate iodine (96% total) treated identically. It is assumed that this was because particulate iodine is a small fraction that is more easily removed and, therefore, conservatively treated as elemental iodine.

AST LOCA analyses are based on 95 / 4.85 / 0.15 aerosol/elemental/organic release percentages. Because of the larger aerosol fraction, it is advantageous to treat it separately, with AEB-98-03 serving as guidance. AEB-98-03 also has the advantage of being a well-mixed model, meeting NRC preferences expressed in RG 1.183.

LGS uses the AEB-98-03 model for the inboard and outboard main steam piping, and for the condenser (treated as a pipe segment). One key difference is that a 50th percentile settling velocity is not used because of potential non-conservatisms. Rather, a 20 group settling velocity probability distribution is used based on a particle settling velocity distribution description in AEB-98-03. Another difference is that elemental iodine is also addressed with the same well-mixed treatment, with deposition velocities based on Cline.

Other differences between the LGS AEB-98-03 implementation for AST and the NEDC-31858P implementation include:

- The NEDC model treats main steam piping as a single node and credits deposition in drain lines to the condenser. The LGS AST model treats the main steam line as two nodes (inboard and outboard) and does not credit deposition in drain lines that form the Alternative Drain Pathway.
- The LGS AST model is a well-mixed nodal treatment; the NEDC-31858P pipe model is plug flow.

The LGS AST model condenser treatment has a number of conservatisms that credit less iodine removal than originally credited (99.7%) in the NEDC-31858P model.

Table 8: Limerick Tech Spec Proposed Changes Since 02/27/04 Submittal (Affected Pages)		
Tech Spec Proposed Changes (Affected Pages)	**Tech Spec Proposed Changes Submitted in February 27, 2004 AST Application**	**Final Supplemented Tech Spec Proposed Changes**
1-2		No change since 02/27/04 submittal.
1-6		No change since 02/27/04 submittal.
1-7		No change since 02/27/04 submittal.
3/4 1-19		No change since 02/27/04 submittal.
3/4 1-20		No change since 02/27/04 submittal.
3/4 3-16		No change since 02/27/04 submittal.
3/4 3-31		No change since 02/27/04 submittal.
3/4 3-64		No change since 02/27/04 submittal.
3/4 3-65		No change since 02/27/04 submittal.
3/4 3-66		No change since 02/27/04 submittal.
3/4 3-67		No change since 02/27/04 submittal.
3/4 4-23	MSIV closure time changed from 5 seconds to 10 seconds	**This change has been rescinded in a supplement dated 10/25/04. MSIV closure time remains at 5 seconds.**
3/4 6-3		No change since 02/27/04 submittal.
3/4 6-47		No change since 02/27/04 submittal.
3/4 6-50		No change since 02/27/04 submittal.
3/4 6-52		No change since 02/27/04 submittal.
3/4 6-53	The acceptance criteria for charcoal laboratory testing were changes from 0.5% to 1.25%.	**This change has been rescinded in this supplement.** Penetration acceptance remains at 0.5%.

Table 8: Limerick Tech Spec Proposed Changes Since 02/27/04 Submittal (Affected Pages)		
Tech Spec Proposed Changes (Affected Pages)	**Tech Spec Proposed Changes Submitted in February 27, 2004 AST Application**	**Final Supplemented Tech Spec Proposed Changes**
3/4 6-55	RERS minimum flow range changed from 60,000 +/- 10% to 30,000 to 66,000 cfm. Laboratory penetration test acceptance criteria changed from 2.5% to 15%.	**This change has been rescinded in this supplement.** RERS flow rate range remains 60,000 +/- 10%; penetration test acceptance criteria remains at 2.5%.
3/4 6-56	RERS minimum flow range changed from 60,000 +/- 10% to 30,000 to 66,000 cfm. Laboratory penetration test acceptance criteria changed from 2.5% to 15%.	**This change has been rescinded in this supplement.** RERS flow rate range remains 60,000 +/- 10%; penetration test acceptance criteria remains at 2.5%.
3/4 7-3	Operational Conditions did not require ESW during handling of recently irradiated fuel.	**This change has been rescinded in this supplement.**
3/4 7-4	Operational Conditions did not require ESW during handling of recently irradiated fuel.	**This change has been rescinded in this supplement.**
3/4 7-5	Operational Conditions did not require Ultimate Heat Sink during handling of recently irradiated fuel.	**This change has been rescinded in this supplement.**
3/4 7-6		No change since 02/27/04 submittal.
3/4 7-7	Laboratory penetration test acceptance criteria changed from 2.5% to 10%. Auto initiation of CREFAS changed to manual.	**This change has been rescinded in this supplement.** Penetration test acceptance criteria remains at 2.5%. Auto initiation of CREFAS retained.

Table 8: Limerick Tech Spec Proposed Changes Since 02/27/04 Submittal (Affected Pages)		
Tech Spec Proposed Changes (Affected Pages)	**Tech Spec Proposed Changes Submitted in February 27, 2004 AST Application**	**Final Supplemented Tech Spec Proposed Changes**
3/4 8-9	Operational Conditions did not require AC Sources during handling of recently irradiated fuel.	**This change has been rescinded in a supplement dated 10/25/04.**
3/4 8-14	Operational Conditions did not require DC Sources during handling of recently irradiated fuel.	**This change has been rescinded in a supplement dated 10/25/04.**
3/4 8-14A	Operational Conditions did not require DC Sources during handling of recently irradiated fuel.	**This change has been rescinded in a supplement dated 10/25/04.**
3/4 8-19 (Unit 2 only)	Operational Conditions did not require Electrical Distribution Sources during handling of recently irradiated fuel.	**This change has been rescinded in a supplement dated 10/25/04.**
3/4 8-20	Operational Conditions did not require Electrical Distribution Sources during handling of recently irradiated fuel.	**This change has been rescinded in a supplement dated 10/25/04.**

Table 9: Summary of Limerick LOCA Dose Analysis			
EAB (rem TEDE)	**LPZ (rem TEDE)**	**Control Room (DBA Case) (rem TEDE)**	**Dose Contributor**
0.866 (worst 2-hour period starts at 3.9 hours)	1.118	2.487	Filtered Primary Containment (PC) Leakage (unfiltered for 15.5 minutes, SGTS filtered thereafter) [100% of L_A], Control Room in Rad Mode
0.021 (worst 2-hour period starts at 10.4 hours)	0.145	0.611	MSIV Leakage with piping deposition credit, no condenser tube deposition credit, and resuspended iodine. *[200 scfh total all MS lines, 100 scfh max/line]*
0.000 (worst 2-hour period starts at 2.3 hours)	0.001	0.006	ECCS Leakage in Secondary Containment (SC) *[5 gpm]*
N/A	N/A	1.78	Gamma Shine to Control Room General Area
0.888	1.26	4.88	Total Calculated Value
25	25	5	Dose Limits

Table 10: Summary of Limerick MSLB Dose Analysis

EAB (rem TEDE)	LPZ (rem TEDE)	Control Room (DBA Case) (rem TEDE)	Analyzed Case
1.11E-01	4.40E-02	1.98E-01	Case 1: Normal Equilibrium of 0.2 μCi
2.22E+00	8.77E-01	3.97E+00	Case 2: Iodine Spike of 4.0 μCi
Case 1: 2.5 Case 2: 25	Case 1: 2.5 Case 2: 25	Case 1: 5 Case 2: 5	Dose Limits

Table 11: Summary of Limerick CRDA Dose Analysis

EAB (rem TEDE)	LPZ (rem TEDE)	Control Room (DBA Case) (rem TEDE)	Analyzed Case
0.0447	0.0312	1.52	RADTRAD Analysis Results (1% of the Condenser free volume leakage per day)
0.0226	0.00818	0.0221	SJAE
6.3	6.3	5	Dose Limits

Table 12: Summary of Limerick FHA Dose Analysis

EAB (rem TEDE)	LPZ (rem TEDE)	Control Room (DBA Case) (rem TEDE)	Analyzed Case
1.52	0.549	4.47	Drop over reactor for worst-case fuel damage, but with 23 feet Water Coverage
6.3	6.3	5	Dose Limits

ATTACHMENT 3

LIMERICK GENERATING STATION
UNITS 1 AND 2

Docket Nos. 50-352
50-353

License Nos. NPF-39
NPF-85

Supplement to License Amendment Request for
"LGS Alternative Source Term Implementation"

TS Bases Pages Markups

<u>Units 1 & 2</u>

B 3/4 6-5
B 3/4 7-1a

involving RECENTLY IRRADIATED FUEL

3/4.6.5 SECONDARY CONTAINMENT

Secondary containment is designed to minimize any ground level release of radioactive material which may result from an accident. The Reactor Enclosure and associated structures provide secondary containment during normal operation when the drywell is sealed and in service. At other times the drywell may be open and, when required, secondary containment integrity is specified.

Establishing and maintaining a vacuum in the reactor enclosure secondary containment with the standby gas treatment system once per 24 months, along with the surveillance of the doors, hatches, dampers and valves, is adequate to ensure that there are no violations of the integrity of the secondary containment.

The OPERABILITY of the reactor enclosure recirculation system and the standby gas treatment systems ensures that sufficient iodine removal capability will be available in the event of a LOCA or refueling accident (SGTS only). The reduction in containment iodine inventory reduces the resulting SITE BOUNDARY radiation doses associated with containment leakage. The operation of this system and resultant iodine removal capacity are consistent with the assumptions used in the LOCA and refueling accident analyses. Provisions have been made to continuously purge the filter plenums with instrument air when the filters are not in use to prevent buildup of moisture on the adsorbers and the HEPA filters.

Although the safety analyses assumes that the reactor enclosure secondary containment draw down time will take 930 seconds, these surveillance requirements specify a draw down time of 916 seconds. This 14 second difference is due to the diesel generator starting and sequence loading delays which is not part of this surveillance requirement.

The reactor enclosure secondary containment draw down time analyses assumes a starting point of 0.25 inch of vacuum water gauge and worst case SGTS dirty filter flow rate of 2800 cfm. The surveillance requirements satisfy this assumption by starting the drawdown from ambient conditions and connecting the adjacent reactor enclosure and refueling area to the SGTS to split the exhaust flow between the three zones and verifying a minimum flow rate of 2800 cfm from the test zone. This simulates the worst case flow alignment and verifies adequate flow is available to drawdown the test zone within the required time. The Technical Specification Surveillance Requirement 4.6.5.3.b.3 is intended to be a multi-zone air balance verification without isolating any test zone.

The SGTS fans are sized for three zones and therefore, when aligned to a single zone or two zones, will have excess capacity to more quickly drawdown the affected zones. There is no maximum flow limit to individual zones or pairs of zones and the air balance and drawdown time are verified when all three zones are connected to the SGTS.

The three zone air balance verification and drawdown test will be done after any major system alteration, which is any modification which will have an effect on the SGTS flowrate such that the ability of the SGTS to drawdown the reactor enclosure to greater than or equal to 0.25 inch of vacuum water gage in less than or equal to 916 seconds could be affected.

Total Effective Dose Equivalent

3/4.7.2 CONTROL ROOM EMERGENCY FRESH AIR SUPPLY SYSTEM - COMMON SYSTEM

The OPERABILITY of the control room emergency fresh air supply system ensures that the control room will remain habitable for operations personnel during and following all design basis accident conditions. Constant purge of the system at 1 cfm is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less ~~whole body, or its equivalent~~. This limitation is consistent with the requirements of ~~General Design Criterion 19 of Appendix A, 10 CFR Part 50~~ 10 CFR Part 50.67, Accident Source Terms.

3/4.7.3 REACTOR CORE ISOLATION COOLING SYSTEM

The reactor core isolation cooling (RCIC) system is provided to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel without requiring actuation of any of the emergency core cooling system equipment. The RCIC system is conservatively required to be OPERABLE whenever reactor pressure exceeds 150 psig. This pressure is substantially below that for which low pressure core cooling systems can provide adequate core cooling.

The RCIC system specifications are applicable during OPERATIONAL CONDITIONS 1, 2, and 3 when reactor vessel pressure exceeds 150 psig because RCIC is the primary non-ECCS source of emergency core cooling when the reactor is pressurized.

With the RCIC system inoperable, adequate core cooling is assured by the OPERABILITY of the HPCI system and justifies the specified 14 day out-of-service period. A Note prohibits the application of Specification 3.0.4.b to an inoperable RCIC system. There is an increased risk associated with entering an OPERATIONAL CONDITION or other specified condition in the Applicability with an inoperable RCIC subsystem and the provisions of Specification 3.0.4.b, which allow entry into an OPERATIONAL CONDITION or other specified condition in the Applicability with the Limiting Condition for Operation not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

The surveillance requirements provide adequate assurance that RCIC will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage and to start cooling at the earliest possible moment.

Since the Control Room Emergency Fresh Air Supply System is not credited for filtration in OPERATIONAL CONDITIONS 4 and 5, applicability to 4 and 5 is only required to support the Chlorine and Toxic Gas design basis isolation requirements.

involving RECENTLY IRRADIATED FUEL

3/4.6.5 SECONDARY CONTAINMENT

Secondary containment is designed to minimize any ground level release of radioactive material which may result from an accident. The Reactor Enclosure and associated structures provide secondary containment during normal operation when the drywell is sealed and in service. At other times the drywell may be open and, when required, secondary containment integrity is specified.

Establishing and maintaining a vacuum in the reactor enclosure secondary containment with the standby gas treatment system once per 24 months, along with the surveillance of the doors, hatches, dampers and valves, is adequate to ensure that there are no violations of the integrity of the secondary containment.

The OPERABILITY of the reactor enclosure recirculation system and the standby gas treatment systems ensures that sufficient iodine removal capability will be available in the event of a LOCA or refueling accident (SGTS only). The reduction in containment iodine inventory reduces the resulting SITE BOUNDARY radiation doses associated with containment leakage. The operation of this system and resultant iodine removal capacity are consistent with the assumptions used in the LOCA and refueling accident analyses. Provisions have been made to continuously purge the filter plenums with instrument air when the filters are not in use to prevent buildup of moisture on the adsorbers and the HEPA filters.

Although the safety analyses assumes that the reactor enclosure secondary containment draw down time will take 930 seconds, these surveillance requirements specify a draw down time of 916 seconds. This 14 second difference is due to the diesel generator starting and sequence loading delays which is not part of this surveillance requirement.

The reactor enclosure secondary containment draw down time analyses assumes a starting point of 0.25 inch of vacuum water gauge and worst case SGTS dirty filter flow rate of 2800 cfm. The surveillance requirements satisfy this assumption by starting the drawdown from ambient conditions and connecting the adjacent reactor enclosure and refueling area to the SGTS to split the exhaust flow between the three zones and verifying a minimum flow rate of 2800 cfm from the test zone. This simulates the worst case flow alignment and verifies adequate flow is available to drawdown the test zone within the required time. The Technical Specification Surveillance Requirement 4.6.5.3.b.3 is intended to be a multi-zone air balance verification without isolating any test zone.

The SGTS is common to Unit 1 and 2 and consists of two independent subsystems. The power supplies for the common portions of the subsystems are from Unit 1 safeguard busses, therefore the inoperability of these Unit 1 supplies are addressed in the SGTS ACTION statements in order to ensure adequate onsite power sources to SGTS for its Unit 2 function during a loss of offsite power event. The allowable out of service times are consistent with those in the Unit 1 Technical Specifications for SGTS and AC electrical power supply out of service condition combinations.

3/4.7 PLANT SYSTEMS

BASES

3/4.7.1 SERVICE WATER SYSTEMS - COMMON SYSTEMS

The OPERABILITY of the service water systems ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of these systems, assuming a single failure, is consistent with the assumptions used in the accident conditions within acceptable limits.

The RHRSW and ESW systems are common to Units 1 and 2 and consist of two independent subsystems each with two pumps. One pump per subsystem (loop) is powered from a Unit 1 safeguard bus and the other pump is powered from a Unit 2 safeguard bus. In order to ensure adequate onsite power sources to the systems during a loss of offsite power event, the inoperability of these supplies are restricted in system ACTION statements.

RHRSW is a manually operated system used for core and containment heat removal. Each of two RHRSW subsystems has one heat exchanger per unit. Each RHRSW pump provides adequate cooling for one RHR heat exchanger. By limiting operation with less than three OPERABLE RHRSW pumps with OPERABLE Diesel Generators, each unit is ensured adequate heat removal capability for the design scenario of LOCA/LOOP on one unit and simultaneous safe shutdown of the other unit.

Each ESW pump provides adequate flow to the cooling loads in its associated loop. With only two divisions of power required for LOCA mitigation of one unit and one division of power required for safe shutdown of the other unit, one ESW pump provides sufficient capacity to fulfill design requirements. ESW pumps are automatically started upon start of the associated Diesel Generators. Therefore, the allowable out of service times for OPERABLE ESW pumps and their associated Diesel Generators is limited to ensure adequate cooling during a loss of offsite power event.

3/4.7.2 CONTROL ROOM EMERGENCY FRESH AIR SUPPLY SYSTEM - COMMON SYSTEM

Total Effective Dose Equivalent

The OPERABILITY of the control room emergency fresh air supply system ensures that the control room will remain habitable for operations personnel during and following all design basis accident conditions. Constant purge of the system at 1 cfm is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less ~~whole body, or its equivalent~~. This limitation is consistent with the requirements of ~~General Design Criterion 19 of Appendix A, 10 CFR Part 50~~ 10 CFR Part 50.67, Accident Source Term

The CREFAS is common to Units 1 and 2 and consists of two independent subsystems. The power supplies for the system are from Unit 1 Safeguard busses, therefore, the inoperability of these Unit 1 supplies are addressed in the CREFAS ACTION statements in order to ensure adequate onsite power sources to CREFAS during a loss of offsite power event. The allowable out of service

Since the Control Room Emergency Fresh Air Supply System is not credited for filtration in OPERATIONAL CONDITIONS 4 and 5, applicability to 4 and 5 is only required to support the Chlorine and Toxic Gas design basis isolation requirements.

ATTACHMENT 4

LIMERICK GENERATING STATION
UNITS 1 AND 2

Docket Nos. 50-352
50-353

License Nos. NPF-39
NPF-85

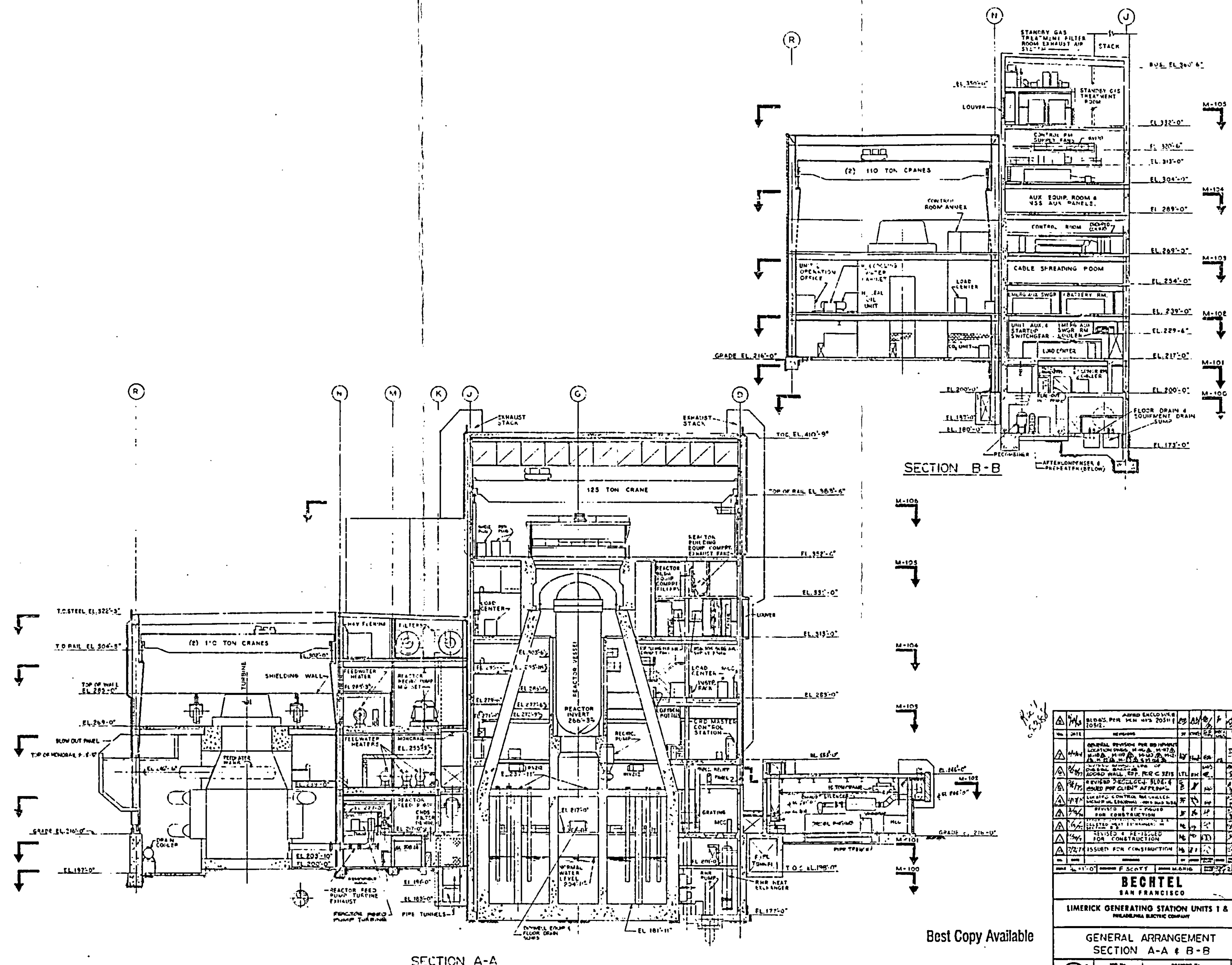
Supplement to License Amendment Request for
"LGS Alternative Source Term Implementation"

Requested Plant Drawings

M-0102, General Arrangement Plan at Elev. 217'- 0"

M-0107, General Arrangement Section A-A & B-B

Post LOCA Reactor Flowpath



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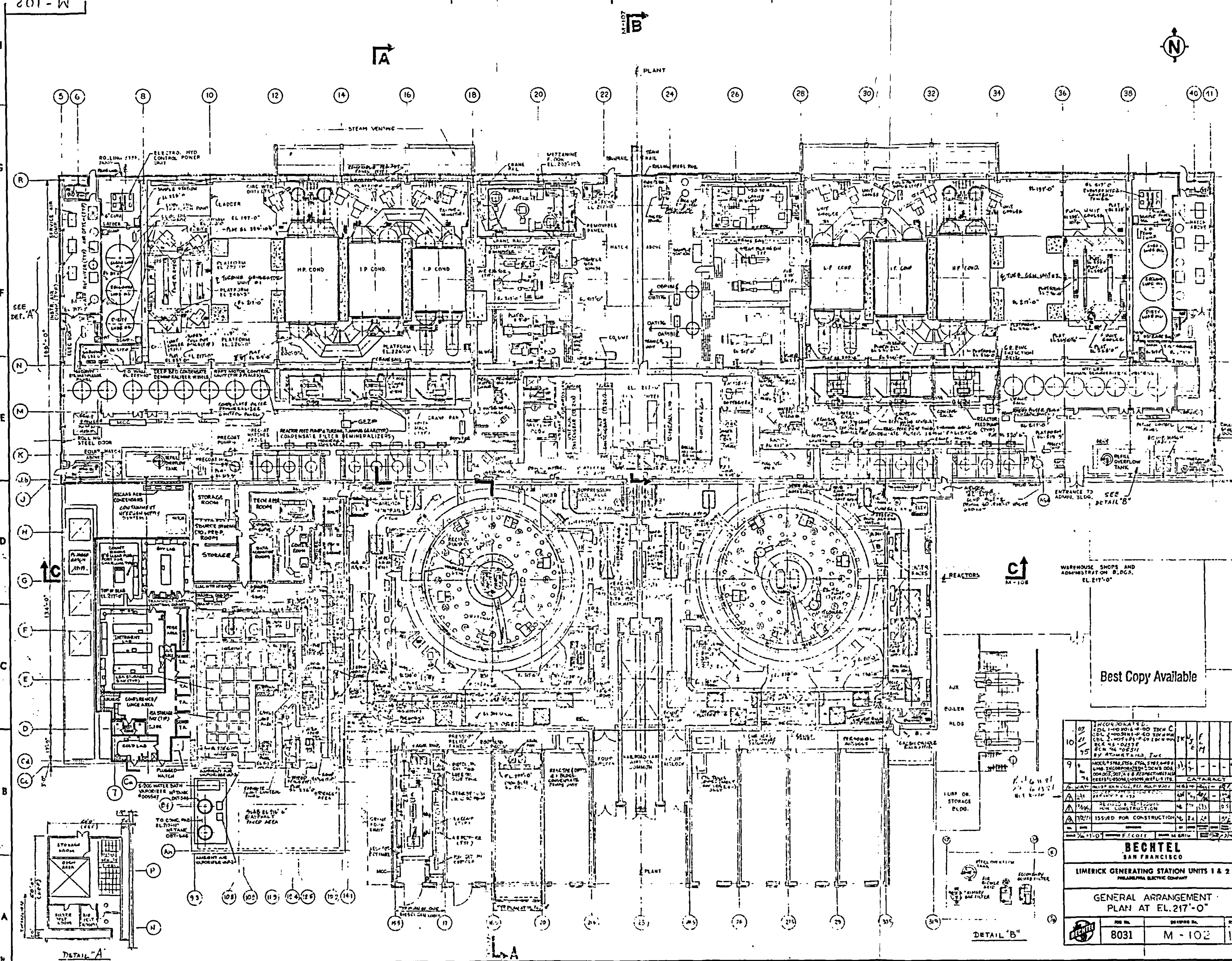
LIMERICK GENERATING STATION UNITS 1 & 2
PHILADELPHIA ELECTRIC COMPANY

GENERAL ARRANGEMENT
SECTION A-A & B-B

JOB No.	DRAWING No.	REV.
8031	M-107	8



No.	Date	Revision						
10	07/21/95	INCORPORATED: CDL 1-0 0016-P-00 TDCN C; CDL 2-00391-P-60; CDL 2-0054-P-67; ECR 95-02578; ECN 96-06511 BY ATOMETRICS, INC	IK	J E	F M			
9		MODS 5782, 5756, 5756, 5783, 6486 & 6148. INCORPORATED DCN'S 005, 004, 005, 007, A & B RESPECTIVELY ALSO ECR'S L-03094, L-03095 ...			-	-	-	
2		REVISED & RE-ISSUED						
1		REVISED & RE-ISSUED FOR CONSTRUCTION						
0	7/2/71	ISSUED FOR CONSTRUCTION						

CATARACT

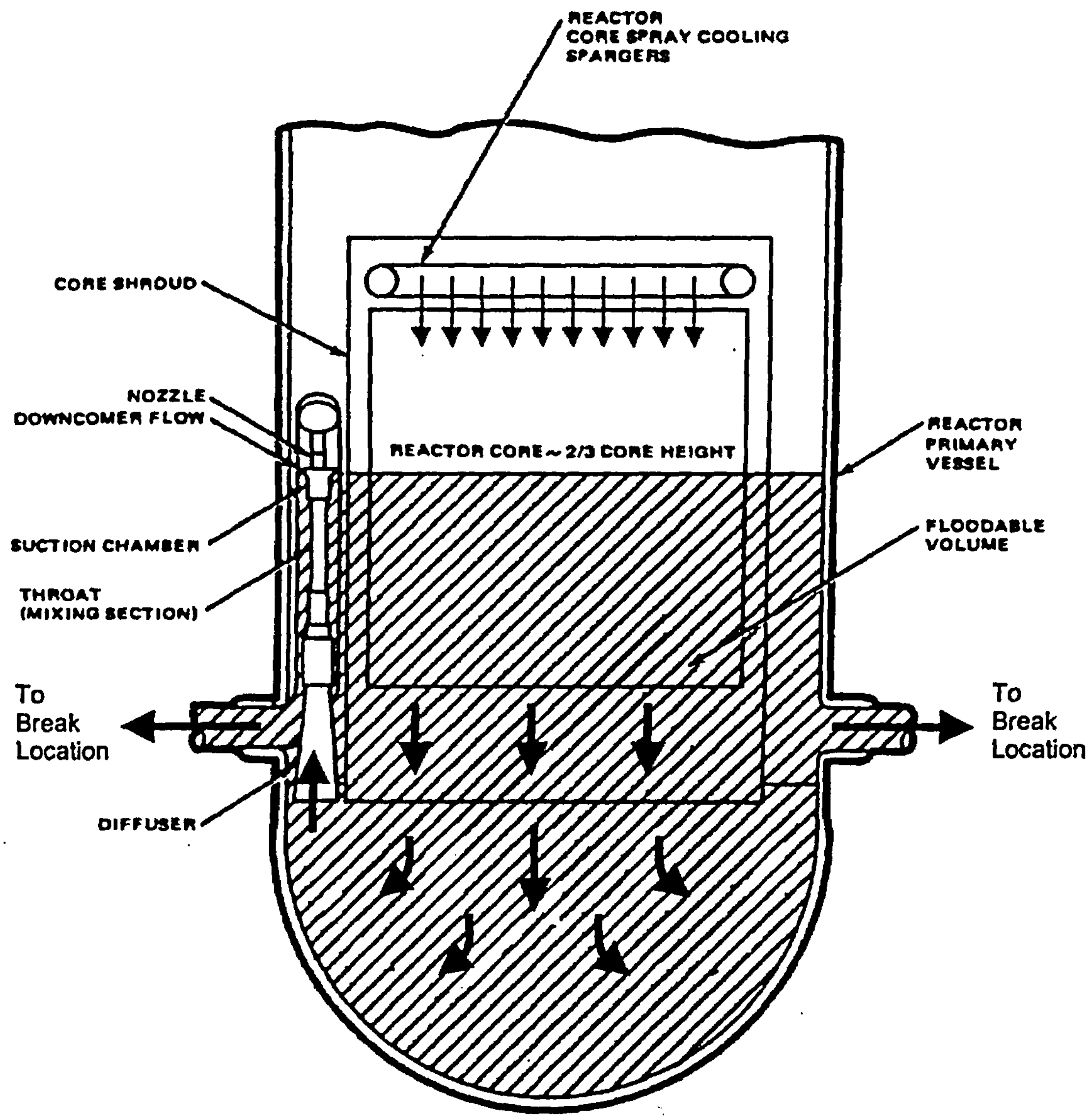
3/16" = 1'-0" F. SCOTT M. GRIG

BECHTEL
SAN FRANCISCO

LIMERICK GENERATING STATION UNITS 1 & 2
PHILADELPHIA ELECTRIC COMPANY

GENERAL ARRANGEMENT
PLAN AT EL. 217'-0"

JOB No.	DRAWING No.	REV
8031	M - 102	10



LIMERICK GENERATING STATION
UNITS 1 AND 2

**Post-LOCA
Reactor
Flowpath**

ATTACHMENT 5

LIMERICK GENERATING STATION
UNITS 1 AND 2

Docket Nos. 50-352
50-353

License Nos. NPF-39
NPF-85

Supplement to License Amendment Request for
"LGS Alternative Source Term Implementation"

Additional Commitments

SUMMARY OF ADDITIONAL EXELON COMMITMENTS
LIMERICK GENERATING STATION, UNITS 1 & 2

The following table identifies commitments made in this document by Exelon. (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
During periods when the Refueling Area Secondary Containment integrity is not required, but movement of irradiated fuel or Core Alterations need to be performed, procedures will be in place to assure that the appropriate refueling area HVAC system is in operation, which includes the exhaust radiation monitors, in order to assure that proper protection is provided in the event of a FHA.	Upon Implementation of AST at Limerick
A program will be in place to ensure prompt action(s) will be taken, if necessary, to isolate the refueling area and control/monitor any potential releases as a result of a FHA during periods when the Refueling Area Secondary Containment integrity is not required and movement of irradiated fuel or Core Alterations need to be performed.	Upon Implementation of AST at Limerick
Some SLC components including the squib valve operator and the pump motor were not previously environmentally qualified for the post-LOCA environment. The component records will be revised to address the Post LOCA EQ requirements prior to AST implementation.	Upon Implementation of AST at Limerick
Limerick Special Event procedure SE-10, "LOCA", which provides direction for operator action after a design basis loss of coolant accident initiation signal has been received, will be revised to include direction to inject sodium pentaborate in accordance with system operating procedure S48.1.B, "Standby Liquid Control System Manual Initiation".	Upon Implementation of AST at Limerick
Training will be provided to operators and appropriate Emergency Response Organization personnel for procedures that will be revised prior to implementation of the AST amendment to specifically direct boron injection for pH control following a LOCA.	Upon Implementation of AST at Limerick