FENOC
FirstEnergy Nuclear Operating Company

Beaver Valley Power Station
P.O. Box 4
Shippingport, PA 15077-0004

L. William Pearce
Vice President

724-682-5234
Fax: 724-643-8069

August 26, 2005
L-05-137

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

Subject: **Beaver Valley Power Station, Unit No. 1**
BV-1 Docket No. 50-334, License No. DPR-66
Response to a Request for Additional Information (RAI dated July 28, 2005) in Support of License Amendment Request No. 320

By letter dated July 28, 2005, the U.S. Nuclear Regulatory Commission (NRC) issued a request for additional information (RAI) pertaining to FirstEnergy Nuclear Operating Company (FENOC) License Amendment Request (LAR) No. 320 "Replacement Steam Generators" for Beaver Valley Power Station (BVPS) Unit No. 1. The Replacement Steam Generator LAR 320 proposed Technical Specification changes that will support operation of BVPS Unit No. 1 with replacement steam generators and credit the safety analyses at 2900 MWt submitted per Reference 1. Attachment A contains the FENOC responses to the July 28, 2005 RAI questions.

On October 4, 2004, FENOC submitted LAR 302 and 173 by letter L-04-125. This submittal requested an Extended Power Uprate (EPU) for BVPS Unit Nos. 1 and 2 and is known as the EPU LAR (Reference 2). FENOC has identified that the subject questions pertaining to the RSG LAR are also applicable to the EPU LAR.

No new regulatory commitments are contained in this submittal. If you have questions or require additional information, please contact Mr. Henry L. Hegrat, Supervisor - Licensing, at 330-315-6944.

A001

I declare under penalty of perjury that the foregoing is true and correct. Executed on August 26, 2005.

Sincerely,

L. W. Pearce

L. William Pearce

Attachment:
A. Response to RAI dated July 28, 2005

References:
1. FENOC Letter L-05-069, License Amendment Request 320, dated April 13, 2005.
2. FENOC Letter L-04-125, License Amendment Request 302 and 173, dated October 4, 2004.

c: Mr. T. G. Colburn, NRR Senior Project Manager
Mr. P. C. Cataldo, NRC Senior Resident Inspector
Mr. S. J. Collins, NRC Region I Administrator
Mr. D. A. Allard, Director BRP/DEP
Mr. L. E. Ryan (BRP/DEP)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION (RAI)

RELATED TO FIRSTENERGY NUCLEAR OPERATING COMPANY (FENOC)

BEAVER VALLEY POWER STATION, UNIT NO. 1 (BVPS-1)

STEAM GENERATOR REPLACEMENT

DOCKET NO. 50-334

By letter dated April 13, 2005, Agencywide Documents Access and Management System (ADAMS) Accession No. ML051080573, FENOC (licensee) proposed changes to the BVPS-1 operating license to allow operation with replacement steam generators (LAR 1A-320). The Nuclear Regulatory Commission (NRC) staff has reviewed the licensee's application and determined that it will need the additional information identified below to complete its review.

Part A - Table of Affected Technical Specifications

A.1 Question

The Nuclear Regulatory Commission (NRC) staff presumes that there needs to be a new definition of dose equivalent I-131 since the doses will now be calculated as TEDE (total effective dose equivalent). Please review your definition to incorporate TEDE.

Response:

Dose equivalent (DE) I-131 is currently defined in the BVPS-1 Technical Specifications by the equation noted below. It reflects the committed thyroid dose conversion factors derived from International Commission on Radiological Protection (ICRP) 30. This definition is utilized to control the primary and secondary coolant iodine concentrations to within the limits of the Technical Specification.

$$C_{I-131D.E.} = C_{I-131} + \frac{C_{I-132}}{170} + \frac{C_{I-133}}{6} + \frac{C_{I-134}}{1000} + \frac{C_{I-135}}{34}$$

"C" is the concentration, in microcuries/gram, of the iodine isotopes.

The above definition was not changed for the BVPS-1 Replacement Steam Generator (RSG) application, which includes an expanded selective implementation of Alternative Source Terms (AST), since the dose consequences estimated using coolant releases based on the above definition bound the estimated dose consequences obtained using a DE I-131 based on the committed effective dose equivalent (CEDE). The discussion below summarizes the basis for this conclusion.

The DE I-131 based on ICRP-30 CEDE dose conversion factors would be calculated by:

$$C_{I-131D.E.} = C_{I-131} + \frac{C_{I-132}}{86} + \frac{C_{I-133}}{5.6} + \frac{C_{I-134}}{250} + \frac{C_{I-135}}{27}$$

Based on the above definitions, it can be deduced that for a given mixture of iodine isotopes in the reactor coolant, the calculated DE I-131 based on CEDE dose conversion factors will be slightly greater than the corresponding value based on the thyroid dose conversion factors. This would result in the primary and secondary coolant concentrations, estimated based on coolant DE I-131 Technical Specification limits and CEDE dose conversion factors, being slightly less than the corresponding concentrations estimated based on coolant DE I-131 Technical Specification limits and thyroid dose conversion factors.

It is therefore concluded that use of the thyroid dose conversion factors in defining the DE I-131 predict a slightly higher Technical Specification primary and secondary coolant iodine concentration, which, when used in the accident analyses to estimate the releases, results in slightly higher dose consequences.

The degree of conservatism depends on the magnitude of other isotopes relative to that of I-131. For BVPS, the estimated Technical Specification primary and secondary coolant iodine concentrations based on thyroid dose conversion factors are approximately 2% higher than coolant concentrations estimated based on CEDE dose conversion factors. Note that the estimated post accident TEDE dose at the site boundary and control room, from the primary and secondary coolant iodine source (based on the current Technical Specification thyroid based dose equivalent I-131 definition), is also higher by approximately 2%.

It is acknowledged that defining the dose equivalent I-131 based on the committed effective dose has the advantage of being consistent with the post accident dose consequences that are calculated for BVPS using TEDE. However, the approach used at BVPS is conservative, and the accident analyses utilize coolant source terms that are consistent with the unchanged Technical Specification that defines the coolant concentrations.

Note that the difference between the two definitions is minimal. As explained above, the current Technical Specification primary and secondary coolant iodine concentrations based on thyroid dose conversion factors are approximately 2% higher than Technical Specification primary and secondary coolant iodine concentration based on CEDE. The estimated post accident TEDE dose from the coolant iodine source, based on the current Technical Specification DE I-131 definition, is also higher by approximately 2%.

Part B - Section 5.4 Steam Generator Tube Rupture (SGTR)

B.1 Question

With regard to Enclosure 1 of your July 8, 2005, extended power uprate (EPU) RAI response:

Item X.1, Section 5.4 of Enclosure 1 to your July 8, 2005, RAI response related to your extended power uprate LAR indicated that it would take 51 minutes to terminate the radioactive release from the ruptured tube's SG. Please provide the radiological analyses for this scenario which demonstrates that the 30-minute isolation time of your April 13, 2005, LAR is more limiting.

Response:

The licensing basis thermal hydraulic analysis model used to determine the post accident releases and associated dose consequences at the site boundary and control room for the BVPS-1 SGTR with RSGs and at EPU conditions is a simplified model, which was the common industry standard prior to 1980. It is based on a hand calculation that predicts conservative break flows and steam releases, and utilizes a termination time of 30 minutes for the break flow and releases from the ruptured SG. The dose consequences, based on break flows and steam releases associated with the above licensing basis thermal hydraulic analysis, were submitted to NRC in Enclosure 2, Section 5.11.9.8 of the BVPS-1 RSG LAR as it was determined to be bounding.

BVPS-1 has also performed an analysis that documents the dose consequences based on values for break flows and steam releases developed by Westinghouse with LOFTTR2 computer modeling. An "operational response case" was considered that reflects a more realistic EPU SGTR transient analysis with RSGs, and takes into consideration a simulator based operator action time. The operational response case utilized a calculated break flow termination time of 61 minutes as opposed to the 30-minute licensing basis model. The operational response model addresses single failure considerations, and includes margin for steam generator overfill. The operational response case was intended to be more consistent with current assumptions required of a SGTR transient analysis, as opposed to the BVPS-1 licensing basis model. Note that the 51 minute isolation time referred to in the NRC question regarding response to RAI Item X.1 pertained to an evaluation performed to support operation at the current power level.

The table below (Table B.1, Analysis Assumptions and Key Parameter Values, BVPS-1 Steam Generator Tube Rupture – Operational Response Case) lists the key input parameters from the thermal hydraulic analysis associated with the operational response case that were utilized to develop the radiological consequences following a SGTR at BVPS-1. The description of the dose calculation assumptions and methodology presented in Enclosure 2, Section 5.11.9.8 of the RSG LAR is also applicable to the operational response case. The analysis determined that the site boundary and control room dose estimates using the licensing basis thermal hydraulic analysis model are conservative and bound the dose estimates developed utilizing the thermal-hydraulic input data based on the operational response case.

Table B.1

Analysis Assumptions and Key Parameter Values
BVPS-1 Steam Generator Tube Rupture[1] – Operational Response Case

Core Power Level	2918 MWt
Reactor Coolant Mass	373,100 lbm
Break Flow to Faulted Steam Generator (SG)	0-120 sec (9,500 lbm) 120-3652 sec (176,200 lbm)
Time of Reactor Trip	120 sec
Amount of Break Flow that Flashes	0-120 sec (1810 lbm) 120-1902 sec (7128.3 lbm)
Leakage Rate to Intact SG's	150 gpd @ 63°F and 1 ATM for each SG
Failed/Melted Fuel Percentage	0%
RCS Tech Spec Iodine & NG Concentration	Table 5.11.4-1 (0.35 µCi/gm DE-I131)
RCS Equilibrium Iodine Appearance Rates	Table 5.11.4-2 (0.35 µCi/gm DE-I131)
Pre-Accident Iodine Spike Activity	Table 5.11.4-2 (21 µCi/gm DE-I131)
Accident Initiated Spike Appearance Rate	335 times equilibrium
Duration of Accident Initiated Spike	4 hours
Secondary System Release Parameters	
Intact SG Liquid Mass (min)	91,953 lbm
Faulted SG Liquid Mass (min)	91,953 lbm
Initial SG Liquid Mass per SG's	91,953 lbm
Tech Spec Activity in SG liquid	Table 5.11.4-1 (0.1 µCi/gm DE-I131)
Form of All Iodine Released to the Environment via SG's	97% elemental; 3% organic
Iodine Partition Coefficient (unflashed portion)	100 (all tubes submerged)
Fraction of Iodine Released (flashed portion)	1.0 (Released without holdup)
Fraction of Noble Gas Released from any SG	1.0 (Released without holdup)
Partition Factor in Condenser	100 elemental iodine 1 organic iodine/Noble Gases
Steam Flowrate to Condenser	0-120 sec (1202 lbm/sec from faulted SG) 0-120 sec (1188 lbm/sec per intact SG)
Faulted SG Steam Releases via MSSV/ADVs	120 sec – 3652 sec (74,900 lbm) 2 hr – 8 hr (41,800 lbm[2])
Intact SG Steam Releases via MSSV/ADVs	120 sec – 7200 sec (448,800 lbm) 2 hr – 8 hr (766,500 lbm)
Termination of Release from SGs	8 hours
Environmental Release Points	0-120 sec (Condenser Air Ejector) 120 sec –8 hr (MSSVs/ADVs)
CR Emergency Ventilation: Initiation Signal/Timing	
Control Room (CR) is maintained in normal ventilation mode	8 hours after DBA
CR Purge Initiation (Manual) Time and Rate	@16,200 cfm (min) for 30 min

Notes:

(1) Steam generator parameter values reflect the Replacement Steam Generators and Operations Assessment

(2) Brief depressurization release in preparation of shutdown cooling

B.2 Question

In your July 8, 2005, EPU RAI response to Item X.3, Section 5.4 of Enclosure 1, why wasn't failure of the atmospheric dump valve (ADV) considered at accident initiation? Would it be more limiting than assuming failure of the ADV at accident initiation when the SG with the tube rupture is isolated?

Response:

The subject RAI response stated that the limiting single failure in the supplemental BVPS-1 SGTR operational response analysis case for radiological dose analysis is a failure of the ADV to the open position on the ruptured steam generator at the time that the ruptured steam generator is isolated.

Although a failure of an ADV on the ruptured steam generator at accident initiation may produce greater mass release, it would not produce more conservative radiological dose consequences. The ADV is failed at the time of steam generator isolation in order to maximize the radiological dose consequences. The SGTR dose analysis methodology provides for a scenario that maximizes the radiological dose consequences of the SGTR event. As outlined in Section 5.11.9.8 of Enclosure 2 of the RSG LAR, the dose model assumes that the noble gases in the break flow and the iodine in the portion of the break flow that flashes is released instantaneously without holdup. The iodine in the non-flashed portion of the break flow mixes uniformly with the steam generator liquid mass and is released in proportion to the steaming rate and partition factor. Delaying the failure of the ADV on the ruptured steam generator until the time of ruptured steam generator isolation allows for an increase in the activity in the steam generator liquid due to the accumulation of the non-flashed iodine activity in the ruptured steam generator. Likewise, the delay allows for a buildup of activity in the RCS due to the accident initiated iodine spike. In summary, the break flow that does not flash will continue to build up the activity in the steam generator liquid and consequently the mass released at this later time in the transient contains a higher concentration of activity than a comparable release early on in the transient, providing for a conservative radiological release.

The supplemental BVPS-1 operational response analysis case for radiological dose analysis was performed to confirm that the BVPS-1 licensing basis analysis (mass and energy balance calculation) is conservative with respect to radiological dose consequences.

B.3 Question

With regard to enclosure 3 of the April 13, 2005, LAR:

Is it assumed or has it been verified that plant cooldown and steam releases from the intact SG cease at 8 hours following the SGTR initiating event?

Response:

As described in Section 5.4.1 of Enclosure 2 of the April 13, 2005, RSG LAR, the BVPS-1 licensing basis analysis for the SGTR event is a mass and energy balance calculation. This calculation includes assumptions that: 1) following the termination of primary-to-secondary break flow and steam release from the ruptured steam generator, the plant is stabilized at no-load temperature with steam release from the intact steam generators until 2 hours after initiation of the SGTR, and 2) the plant is then cooled down with steam release from the intact steam generators to RHR entry conditions within 8 hours after initiation of the SGTR, at which time the steam release from the intact steam generators is terminated.

The supplemental BVPS-1 SGTR operational response analysis uses the LOFTTR2 computer code to model the plant response to the SGTR event including the simulation of the operator actions for recovery from a SGTR based on the BVPS-1 Emergency Operating Procedures (EOPs), which are based on the Westinghouse Owners Group Emergency Response Guidelines. The LOFTTR2 analysis is performed for the time period from initiation of the SGTR until the primary and secondary pressures are equalized, at which time primary-to-secondary break flow is terminated. The BVPS-1 SGTR operational response analysis does not model plant response during the stabilization period that is assumed to last up to 2 hours after initiation of the SGTR or the cooldown to RHR entry conditions period that is assumed to last up to 8 hours after initiation of the SGTR.

Following break flow termination, the plant operators will stabilize and then cool down the plant to RHR entry conditions consistent with the BVPS-1 EOPs. The EOPs direct the operators to use the condenser steam dump valves for steam release to stabilize and cool down the plant. If the condenser steam dump valves are not available, the EOPs direct the operators to use the atmospheric steam dump (ASD) valves. Both the condenser steam dump valves and the ASD valves have adequate capacity to support the cooldown to RHR entry conditions within 8 hours after the SGTR.

The BVPS-1 plant configuration includes three ASD valves (one for each steam generator) and one residual heat release control valve, which is common for all steam generators. These valves have a total atmospheric steam dump capacity that is sufficient to cool down the plant from no-load temperature of 547°F to RHR entry temperature of 350°F in 4 hours (i.e., 50°F/hour cooldown). The ASD valves on the intact steam generators, which are sized to support a normal cooldown at 50°F/hour, have sufficient capacity to cool down the plant to the RHR entry temperature within 8 hours after the SGTR event, at which time RHR operation can be initiated and steam release from the intact steam generators can be terminated. Thus, it has been confirmed that the ASD valves on the intact steam generators have sufficient capacity to cool down the plant to RHR entry conditions within 8 hours following the SGTR.

Part C - Section 5.11 Radiological Analysis

C.1 Question

What is the basis for assuming a 30-minute purge of the control room envelope (CRE) following the completion of the accident sequence and are there procedures directing the operators to take such actions (Pg 5-225)?

Response:

The 30 minute purge of the CRE following completion of the accident sequence is utilized in the dose consequence analysis for the Main Steam Line Break and the Steam Generator Tube Rupture to support compliance with the regulatory dose criteria of 10 CFR 50.67 and Regulatory Guide 1.183 relative to maximum allowable post-accident exposure of the control room operator.

The actions for the purge of the CRE, as defined in the accident analysis, have been included in revised emergency operating procedures for the MSLB. The Engineering Change Process and the LAR implementation process will be followed to make the necessary revisions to the plant procedures for SGTR prior to implementation.

C.2 Question

It is stated that the control room shielding design is based upon the loss-of-coolant accident (LOCA) because the LOCA represents the worst case design basis accident (DBA) relative to radioactivity releases. Does the LOCA represent the worst shine dose to the control room operators of any DBA (Pg 5-231)?

Response:

The LOCA does represent the worst shine dose to the control room operators of any DBA. The BVPS control room shielding design is based on the LOCA because it not only represents the worst case radioactivity release, but it also represents the worst case DBA source term in contained sources (the airborne source in containment / ECCS components, control room filters, airborne sources in areas adjacent to the control room, etc.), as well as the overhead cloud, both of which contribute to the direct shine dose inside the BVPS control room.

C.3 Question

There appears to be no value assumed for inleakage into the CRE during the period that the CRE is operating in its normal mode of operation. In addition, the tracer gas tests performed did not include the inleakage characteristics while the normal control room ventilation system was operating. Also, the ventilation filter testing program (VFTP) contains no testing criteria for the normal ventilation system. There needs to be testing criteria since the normal ventilation system is now part of the response to an accident. Is there emergency power to the normal ventilation system and to the recirculation system? If not, can you assume that they are operating in the event of a loss-of-offsite power (LOOP) (Table 5.11.9-3)?

Response:

Table 5.11.9-3 shows 500 cfm as Normal Operation Unfiltered Intake. The normal outside air flowrate was measured in 1990 and the associated throttling dampers were set in place to achieve the appropriate flow rates. The damper settings are confirmed by a monthly walkdown by the System Engineer. The normal operation ventilation alignment outside air flowrate measurement will be added to surveillance procedure, 3BVT-1.44.01, Control Room Emergency Supply Fan Pressurization Test, and measured each operating cycle. The procedure change will be made as part of the LAR implementation process, and this action has been incorporated into the BVPS Corrective Action Program.

The tracer gas test performed in May 2001 using BVPS procedure 3BVT 1.44.05, Control Room Envelope Air In-Leakage Test, did not measure unfiltered inleakage in the normal operation ventilation alignment. The normal operation ventilation alignment utilizes the same Control Room Envelope (CRE) boundary (walls, floors, ceiling and ducting) as does the emergency pressurization and recirculation mode ventilation lineup. The boundary was shown to have no unfiltered inleakage across the boundary during the pressurization mode and 267 cfm unfiltered inleakage in the recirculation mode. Normal operation ventilation lineup differential pressure measurements indicate that the CRE areas were positive to adjacent areas during the normal operation mode. Adjacent areas that were positive with respect to the CRE during the recirculation mode of the tracer gas test were all negative when measured in the normal operation ventilation alignment mode. Consequently, the lack of unfiltered inleakage across the boundary during the pressurization mode is also applicable for the normal operation mode. Based upon this information, the tracer gas test performed per 3BVT 1.44.05 was acceptable and confirmed the boundary for unfiltered inleakage.

A CRE integrity assessment was performed in 2004 using the guidance provided in Regulatory Guide 1.196, Control Room Habitability at Light-Water Nuclear Power Plants and NEI 99-03, Rev. 1, Control Room Habitability Guidance with no findings that would call into question control room habitability.

The normal operation ventilation alignment provides no filtration by HEPA or charcoal filters. The Control Room Emergency Ventilation System provides for outside makeup air filtration and is part of the VFTP.

For BVPS, emergency power is provided to the normal control room ventilation system including all ventilation system components that are required to support control room operation in the recirculation mode.

Note that even if the normal operation control room ventilation system was not powered by emergency power, the BVPS dose consequence analyses would have to assume its continued availability post-accident since it presents the worst case ventilation configuration for purposes of developing dose consequences. For BVPS, the post-accident dose consequences based on an unfiltered air intake equivalent to that provided by the normal operation system (500 cfm) prior to initiation of the emergency ventilation system, or for accidents that do not credit initiation of the control room ventilation system, will bound the dose consequences based on a smaller unfiltered inleakage applicable to a control room with no ventilation intake flow such as that applicable to the BVPS control room when in the recirculation mode (267 cfm).

C.4 Question

What Is the basis for assuming that the iodine spike will occur for only 4 hours as stated in Tables 5.11.9-4A, 5.11.9-5A and 5.11.9-9?

Response:

During normal operation, the iodine gas in the gap is released to the reactor coolant via defective fuel pins. The release rate of the iodine gas from the fuel gap into the reactor coolant is identified as the equilibrium iodine release rate. Per regulatory guidance, this equilibrium iodine release rate (also called appearance rate in the primary coolant) is postulated to increase by a specified amount following several design basis accidents. This increase in the iodine release or appearance rate is called the Concurrent Iodine Spike.

The duration of the Concurrent Iodine Spike (CIS) is determined by the following:

a) post-accident iodine activity release rate from the gap of fuel pins that have defects to the reactor coolant, and

b) the amount of iodine activity in the gap of fuel pins that have defects.

Per Regulatory Guide 1.183, for accident analyses, the initial primary coolant iodine activity concentration is assumed to be at the Technical Specification iodine activity limit of 0.35 µCi/gm of Dose Equivalent (DE) I-131.

Per Regulatory Guide 1.183, the equilibrium iodine activity release rate at the Technical Specification concentration provided in Enclosure 2, Table 5.11.4-2, is assumed to be increased by 335 (for the Steam Generator Tube Rupture) and 500 (for the Main Steam Line Break). Per NUREG 0800, SRP 15.6.2, the equilibrium I-131 appearance rate at the Technical Specification concentration is assumed to be increased by 500 for the Small Line Break outside Containment.

The amount of iodine activity available for release is based on that available in the gap of fuel pins that have defects. The defective fuel percentage is estimated to be that corresponding to the Technical Specification iodine concentration of 0.35 µCi/gm DE I-131, and is calculated to be 0.095%. This is derived based on the calculated iodine concentration of 3.69 µCi/gm DE I-131 in the reactor coolant, assuming 1% fuel defects. The iodine isotopic inventory in the core is based on the end of an equilibrium fuel cycle, and is provided in Enclosure 2, Table 5.11.4-3. The iodine gap fractions are based on Regulatory Guide 1.183.

Based on the above, for the BVPS Steam Generator Tube Rupture, the estimated time prior to depletion of the available Iodine activity is less than 2.5 hrs.

For the BVPS MSLB or the Small Line Break outside containment, the CIS appearance rate is 500 times the equilibrium appearance rate; consequently, the time for depleting the available iodine activity in the gap of the defective fuel rods will be shorter.

Based on the above, a 4-hr. duration for the concurrent iodine spike was conservatively assumed for all BVPS design basis accidents that postulate a concurrent iodine spike.

C.5 Question

What is the basis for assuming that the faulted SG will be isolated in 30 minutes as stated in Table 5.11.9-5?

Response:

As described in Section 5.4.1 of Enclosure 2 of the April 13, 2005 RSG LAR and discussed in the July 8, 2005 EPU RAI response to Item X.1, Section 5.4 of Enclosure 1, the BVPS-1 licensing basis analysis for the SGTR event is a mass and energy balance calculation that assumes that the steam release from the ruptured steam generator is terminated at 30 minutes after initiation of the SGTR.

A supplemental BVPS-1 SGTR operational response analysis has been performed to show that the ruptured steam generator will not overfill and to develop thermal and hydraulic SGTR input data for radiological dose analysis. The supplemental SGTR operational response analysis includes consideration of single active failures, and the timing of operator actions in accordance with plant Emergency Operating Procedures (EOPs) and demonstrated performance during simulator exercises. The supplemental analysis case for SGTR input data for radiological dose analysis showed that primary and secondary pressures are equilibrated and that primary-to-secondary break flow and steam releases from the ruptured steam generator are terminated at 61 minutes after initiation of the SGTR. The supplemental analysis SGTR input data was used in supplemental radiological dose analysis that confirmed the conservatism in the licensing basis dose calculations based on the assumed 30 minute termination of break flow and steam releases from the ruptured steam generator.

This supplemental BVPS-1 SGTR operational response and radiological dose analysis confirmed that the 30 minute termination licensing basis analysis is conservative from a radiological dose standpoint even though the break flow termination time is greater than 30 minutes. This is because the 30 minute termination licensing basis analysis includes other conservative assumptions that result in higher break flow that flashes from the ruptured steam generator (which is the dominant contributor to dose consequences) than for the SGTR operational response analysis case where break flow and steam releases are terminated in 61 minutes.

C.6 Question

What is the basis for assumed particulate carry-over fraction in the SG as stated in Table 5.11.9-6?

Response:

As noted in Section 5.5.4 of Appendix E of Regulatory Guide 1.183, the retention of particulate radionuclides in the steam generator is limited by the moisture carryover from the steam generators.

For BVPS, the particulate carryover fraction in the steam generator used in the Locked Rotor analysis is conservatively based on the design (maximum) moisture carryover in the BVPS-1 original Steam Generators (0.25%). This value bounds the design moisture carryover fraction for the RSGs, which is listed as 0.1%.

C.7 Question

What letdown demineralizer flow rate and removal efficiency were assumed in the determination of the release rate for the iodine spiking for the main steam line break (MSLB) and SGTR accidents?

Response:

The calculation that determines the equilibrium iodine release rate from the fuel to RCS utilizes the BVPS-2 maximum design letdown demineralizer flow rate of 135 gpm (bounding for Unit 1), and a removal efficiency of 100%. In addition, the maximum Technical Specification primary coolant leakage of 11 gpm (10 gpm identified plus 1 gpm unidentified) was also included in determining the equilibrium iodine release rate.

Part D - Meteorology

D.1 Question

With regard to Enclosure 2 of the April 13, 2005, LAR;

Were the following DBAs those for which new atmospheric dispersion factors (χ/Q values) were provided for use in the dose assessments: i) MSLB outside of containment, ii) SGTR, iii) locked-rotor accident (LRA), and iv) small line break (SLB) outside of containment?

Response:

As noted in Section 5.11.9.2 of Enclosure 2 of the RSG LAR, the Exclusion Area Boundary (EAB) and the Low Population Zone (LPZ) atmospheric dispersion factors for BVPS-1 and BVPS-2 remain unchanged by this application and are consistent with current licensing basis.

As noted in Section 5.11.9.2 of Enclosure 2, new control room atmospheric dispersion factors based on ARCON96 methodology were utilized for the i) MSLB outside of Containment, ii) SGTR, iii) Locked Rotor Accident (LRA) – bounds the Loss of AC Power accident (LACP), and iv) Small Line Break (SLB) outside of Containment. These new atmospheric dispersion factors were summarized in Enclosure 2, Tables 5.11.9-2A and B. Ten separate release points are addressed in the referenced Tables. This is a complete list of the limiting control room χ/Q values utilized to support the new dose assessments developed for this application.

D.2 Question

Is this a complete list representing the limiting χ/Q values used in all of the new dose assessments for this LAR?

Response:

Yes. See responses to RAIs D.1 and D.3 for details.

D.3 Question

If so, is the following a correct pairing of the DBAs associated with each of the release/receptor locations?

Release	Receptor	Accident
Unit 1 Ventilation Vent	Unit 1 CR	SLB
Unit 1 MS Relief Valves	Unit 1 CR	MSLB
Unit 1 MSL (break)/AEJ*	Unit 1 CR	MSLB, SGTR, LRA
Unit 1 Ventilation Vent	Unit 2 CR	SLB
Unit 1 MS Relief Valves	Unit 2 CR	MSLB
Unit 1 MSL (break)/AEJ	Unit 2 CR	MSLB, SGTR, LRA
Unit 2 Ventilation Vent	Unit 1 CR	SLB
Unit 2 MS Relief Valves	Unit 1 CR	MSLB
Unit 2 Ventilation Vent	Unit 2 CR	SLB
Unit 2 MS Relief Valves	Unit 2 CR	MSLB

***AEJ = air ejector**

Response:

The correct pairing of the DBAs analyzed for the BVPS-1 RSG LAR associated with each of the release / receptor locations is provided in Table D.1 with some clarifications.

Table D.1

Release / Receptor Locations for BVPS-1 DBA Analyzed Accidents

Item #	Release	Receptor	Applicable Accident	Comment
1	Unit 1 Ventilation Vent	Unit 1 CR	U1 SLB	Used in: • Bounding U1/U2 SLB analysis
2	Unit 1 MS Relief Valves	Unit 1 CR	U1 MSLB, U1 SGTR, U1 LR/LACP	Used in: • U1 MSLB, • U1 SGTR • Bounding U1/U2 LR / LACP analysis
3	Unit 1 MSL (break)/AEJ	Unit 1 CR	U1 MSLB, U1 SGTR	Used in: • U1 MSLB (break location releases), • U1 SGTR (AEJ releases)
4	Unit 1 Ventilation Vent	Unit 2 CR	U1 SLB	Not used. Bounded by item 1
5	Unit 1 MS Relief Valves	Unit 2 CR	U1 MSLB U1 SGTR, U1 LR/LACP	Not used. Bounded by item 2
6	Unit 1 MSL (break)/AEJ	Unit 2 CR	U1 MSLB, U1 SGTR	Not used. Bounded by item 3
7	Unit 2 Ventilation Vent	Unit 1 CR	U2 SLB	Not used. Bounded by item 1
8	Unit 2 MS Relief Valves	Unit 1 CR	U2 LR/LACP	Not used. Bounded by item 2
9	Unit 2 Ventilation Vent	Unit 2 CR	U2 SLB	Not used. Bounded by item 1
10	Unit 2 MS Relief Valves	Unit 2 CR	U2 LR/LACP	Not used. Bounded by item 2

As noted in Section 5.11.9.4 of Enclosure 2 of the RSG LAR, BVPS is equipped with a joint control room with two ventilation intakes, one assigned to Unit 1 (U1) and one assigned to Unit 2 (U2). The radioactivity releases from release points associated with accidents at either unit will therefore enter the control room via a) both the Unit 1 and Unit 2 control room ventilation intakes and b) control room unfiltered inleakage. As noted in the Section 5.11.9.4 of Enclosure 2, based on tracer gas testing, the control room air intake atmospheric dispersion factors (χ/Qs) are determined to be representative of the worst case χ/Q values for control room unfiltered inleakage since the distances and directions from the accident release points to these receptors (intakes vs identified inleakage locations) are very similar.

Atmospheric dispersion factors were therefore developed for all release points associated with a unit specific accident, to the Unit 1 and 2 control room air intakes. The air intake with the higher atmospheric dispersion factor was selected for use in the unit specific dose consequence analysis.

As noted in Section 5.11 of Enclosure 2, the intent of the BVPS-1 RSG LAR is to only address those accidents impacted by the RSGs at BVPS-1. However, several of the dose consequence analyses developed to support BVPS-1 RSGs at EPU conditions utilize bounding parameter values to encompass an event at either BVPS unit. Consequently, these bounding dose consequence analyses utilize atmospheric dispersion factors associated with the most limiting release point (accident at either unit) / receptor (Unit 1 or Unit 2 control room air intake) combination.

Provided below is the list of a) accidents evaluated in support of this application, b) whether the analysis is bounding and applicable to both units, and c) the release point(s) applicable to the accident.

1. BVPS-1 & 2 Control Rod Ejection Accident (CREA) – *Release point: MS relief valves*
2. BVPS-1 Main Steam Line Break (MSLB) outside Containment – *Release points: MS relief valves and MS line break location*
3. BVPS-1 Steam Generator Tube Rupture (SGTR) – *Release points: MS relief valves and Air ejector (AEJ)*
4. BVPS-1 & 2 Locked Rotor Accident (LRA) – *Release point: MS relief valves*
5. BVPS-1 & 2 Loss of AC Power (LACP) – *Release point: MS relief valves*
6. BVPS-1 & 2 Small Line Break (SLB) Outside Containment – *Release point: Ventilation Vent*

Note that:

- The CREA was addressed in the application by reference only, since this accident analysis was done taking into consideration BVPS-1 RSGs and EPU conditions, and was approved by License Amendment No. 257.
- The LACP is bounded by the LRA.
- The MSLB and SGTR are unit specific analysis, therefore, only Unit 1 input parameter values / atmospheric dispersion factors are included in Section 5.11.9 of Enclosure 2.
- The LR/LACP and the SLB are bounding analyses; therefore, both Unit 1 and Unit 2 input parameter values / atmospheric dispersion factors are included in Section 5.11.9 of Enclosure 2.

D.4 Question

Why were χ/Q values associated with the ventilation vents and MS relief valves provided for postulated releases from BVPS-2 given that this LAR is for RSG at BVPS-1? Was the intent to use the BVPS-1 specific χ/Q values for the MSL (break)/AEJ dose assessment and the site limiting χ/Q values for the dose assessments related to postulated releases from the ventilation vent and MS relief valve, but not for the MSL (break)/AEJ? For the BVPS-1 MSL (break)/AEJ, is the break assumed to occur at the AEJ or were the more limiting χ/Q values assumed for a release from either the MSL (break) or AEJ?

Response:

See RAI response to Question – Meteorology, D.3.

Regarding the BVPS-1 MSL (break) / AEJ atmospheric dispersion factors, the worst case break location for the BVPS-1 MSLB, and the air ejector releases following a BVPS-1 SGTR, both occur in the BVPS-1 Turbine building. The single set of χ/Qs presented in Enclosure 2, Tables 5.11.9-2A as applicable to the BVPS-1 MSL (break) / AEJ conservatively represents a release point at the closest corner of the Turbine building relative to the Unit 1 and Unit 2 control room intakes.

D.5 Question

What license proceeding approved the 1996-generated exclusion area boundary and low-population zone χ/Q values referenced in a prior LAR dated June 5, 2002, presented in the BVPS-1 and 2 Updated Final Safety Analysis Reports, and used in the dose assessments for this LAR?

Response:

BVPS submittal dated March 10, 1997 "Proposed Operating License Change Request No. 240" provided the current licensing basis χ/Qs calculations, ERS-SFL-96-021, Rev 0, "RG1.145 Short Term Accident χ/Q Values for EAB and LPZ, Unit 1 and Unit 2, Based on 1986-1995 Observations." These χ/Q values for BVPS-1 and BVPS-2 were approved by License Amendment No. 205 dated September 10, 1997. It is noted that the χ/Q values that were used in the confirmatory calculation performed for Amendment 205 are different and more conservative than those χ/Q values used in the BVPS calculation and currently approved for BVPS-1 and BVPS-2. FENOC has no information regarding the source of the χ/Q values used in the confirmatory calculation but believes that they were conservative values used for purposes of validating the BVPS calculations.

Additional historical licensing basis documents were researched on this subject topic and the following items were identified:

- The following statement appears in the SER for Amendment 237 (BVPS-1) and Amendment 119 (BVPS-2) regarding Design Basis Accident Dose Consequence Calculation Revisions:

  "Revised atmospheric dispersion factors were reviewed and accepted by the NRC staff for BVPS-1 in License Amendment No. 205 dated September 16,1997 (sic) .... "

- The following statement appears in the SER for Amendment 241 (BVPS-1) and Amendment 121 (BVPS-2) regarding changes to the Fuel Handling Accident:

  "The NRC staff reviewed previously docketed material on calculation of the licensee's current licensing basis offsite χ/Qs, Beaver Valley Calculation ERS-SFL-96-021, Rev 0, "RG1.145 Short Term Accident χ/Q Values for EAB and LPZ, Unit 1 and Unit 2, Based on 1986-1995 Observations." These χ/Qs were approved by License Amendment No. 205 ..... "

Therefore, the current licensing basis χ/Q values used in the dose assessments for this LAR were approved by License Amendment No. 205 dated September 10, 1997, based on the calculation ERS-SFL-96-021 provided in a submittal dated March 10, 1997.

D.6 Question

Please provide reference to or the actual plant drawing of the site postulated release and receptor points applicable to this LAR of sufficient size and scale to facilitate a check that the inputs appear to be reasonable.

Response:

The plant drawing that includes the site postulated release and receptor points applicable to this LAR is 8700-RY-1C Rev 2. A full size copy is being provided, as well.