September 28, 2005

Mr. L. William Pearce
Vice President
FirstEnergy Nuclear Operating Company
Beaver Valley Power Station
P. O. Box 4
Shippingport, PA 15077

SUBJECT: BEAVER VALLEY POWER STATION, UNIT NO. 1 (BVPS-1), REPLACEMENT STEAM GENERATORS (RSGs) - REQUEST FOR ADDITIONAL INFORMATION (RAI) (TAC NO. MC6725)

Dear Mr. Pearce:

By letter dated April 13, 2005, you submitted a license amendment request (LAR) for BVPS-1 to allow operation with RSGs. The Nuclear Regulatory Commission (NRC) staff has determined that it will need the additional information identified in the enclosure to this letter to complete its review. As discussed with your staff, we request that you provide your response within 30 days of receipt of this request in order for the NRC staff to meet your requested LAR review schedule.

If you have any questions, please contact me at 301-415-1402.

Sincerely,

/RA/

Timothy G. Colburn, Senior Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-334

Enclosure: RAI

cc w/encl: See next page

September 28, 2005

Mr. L. William Pearce
Vice President
FirstEnergy Nuclear Operating Company
Beaver Valley Power Station
P. O. Box 4
Shippingport, PA 15077

SUBJECT: BEAVER VALLEY POWER STATION, UNIT NO. 1 (BVPS-1), REPLACEMENT STEAM GENERATORS (RSGs) - REQUEST FOR ADDITIONAL INFORMATION (RAI) (TAC NO. MC6725)

Dear Mr. Pearce:

By letter dated April 13, 2005, you submitted a license amendment request (LAR) for BVPS-1 to allow operation with RSGs. The Nuclear Regulatory Commission (NRC) staff has determined that it will need the additional information identified in the enclosure to this letter to complete its review. As discussed with your staff, we request that you provide your response within 30 days of receipt of this request in order for the NRC staff to meet your requested LAR review schedule.

If you have any questions, please contact me at 301-415-1402.

Sincerely,

/RA/

Timothy G. Colburn, Senior Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-334

Enclosure: RAI

cc w/encl: See next page

DISTRIBUTION:
PUBLIC MO'Brien ACRS TColburn PDI-1 R/F MBarillas
RBellamy, RGN-I JNakoski PClifford RLaufer OGC DLPM DPR
KWood

ACCESSION NO.: ML052430737

OFFICE	PDI-1/LA	PDI-1/PM	SRXB/SC	PDI-1/SC
NAME	MO'Brien	TColburn	JNakoski	RLaufer
DATE	09/27/05	09/26/05	09/26/05	09/27/05

OFFICIAL RECORD COPY

REQUEST FOR ADDITIONAL INFORMATION (RAI)

BEAVER VALLEY POWER STATION, UNIT NOS. 1 AND 2 (BVPS-1 AND 2)

REPLACEMENT STEAM GENERATORS (RSGs) LICENSE AMENDMENT REQUEST (LAR)

AND EXTENDED POWER UPRATE (EPU) LAR

1. The licensee indicated in its April 13, 2005, RSG LAR application that much of the supporting analyses for the request were contained in the previously submitted extended power uprate LAR dated October 4, 2004. Based upon the licensee's July 8, 2005, response to the BVPS-1 and 2 EPU LAR RAI dated May 5, 2005, it appears that the feedwater line break (FWLB) accident analysis methodology has been significantly altered from the Updated Final Safety Analysis Report (UFSAR) Analysis-of-Record (AOR). The AOR models the break flow as follows: "a saturated liquid discharge is assumed until all the water inventory is discharged from the affected steam generator [SG]." The analysis assumptions and initial conditions are built around this conservative modeling technique. For example, a full double-ended break area is assumed in order to rapidly deplete the affected SG inventory. The EPU analysis attempts to model the dynamic SG liquid level within the affected SG and credits an early transition from liquid to steam discharge. This methodology change results in a cooldown of the reactor coolant system (RCS) similar to that during a main steamline break (MSLB) accident analysis.

   The methodology change brings into question the analysis assumptions and initial conditions. For example, crediting steam discharge with a large-break size promotes a more rapid depressurization of the affected SG, which results in an earlier reactor trip and auxiliary feedwater system initiation as well as a more substantial cooldown of the RCS. Furthermore, since the transition from liquid to steam discharge has a first order effect, the ability of the Westinghouse (W) analytical tools to accurately predict dynamic conditions with both the affected and unaffected SGs takes on more importance. The pedigree of these models needs further assessment. The licensee is requested to discuss the apparent changes in methodology and either demonstrate that they conform to previously approved methodologies for FWLB and MSLB accident analyses, or provide justification for the change in methodology.

   Additionally, since the FWLB analysis is intended to evaluate more than one acceptance criteria which may be mutually exclusive to each other, the limiting case for one criteria may not be limiting with respect to the other criteria, e.g., bulk boiling versus peak pressure. Please verify that the FWLB analyses are limiting with respect to each of the acceptance criteria considered.

2. Due to the Nuclear Regulatory Commission (NRC) staff's concerns and the need to complete its safety evaluation (SE) in a timely manner, the NRC staff is requesting that it perform an audit of the supporting W engineering calculation. Dates of November 7-10, 2005, have been agreed to by your staff to allow the NRC staff to conduct this audit. The issues identified below will need to be resolved in order for the NRC staff to

Enclosure

complete its SEs related to the BVPS-1 RSG LAR and BVPS-1 and 2 EPU LAR.

a. The FWLB analysis methodology presented in the RSG and EPU LARs is significantly different than the methodology described in the BVPS-1 and 2 UFSARs. Specifically, the UFSARs state that the faulted SG break flow characteristics are conservatively modeled as liquid discharge; whereas, the new analysis employs a best-estimate technique which predicts steam discharge prior to reactor trip.

    1. Identify when and how the methodology changed.

    2. Provide a break spectrum analysis to identify the limiting break size.

    3. Quantify the degree of uncertainty associated with the "indicated" SG downcomer liquid level and its effects on the reactor protection system (RPS) and the engineered safety features actuation system (ESFAS) response.

    4. Quantify the degree of uncertainty associated with "actual" SG downcomer liquid level and its effects on break discharge characteristics.

    5. Investigate the modeling of the break flow through the downcomer feed ring and out through the break.

    6. With regard to the use of the "more detailed SG design codes," describe their use and interaction with LOFTRAN and RETRAN codes and discuss the licensing history and approval of these codes.

b. UFSAR Section 15.2.8.2.1 provides a pressurizer safety valve (PSV) operability assessment for the FWLB event. In its July 8, 2005, response to RAI U.3.c, the licensee stated that this section of the UFSAR will be deleted because it no longer was applicable to BVPS-1 and 2.

    1. Discuss the licensing basis and justification for deleting the UFSAR section identified above.

    2. From the BVPS-1 RSG/EPU transient response, it appears that the action of the power-operated relief valves (PORVs) (coupled with operators increasing AFW flow) terminates the RCS pressure rebound. Please evaluate the same case without PORV action to determine if RCS pressure and pressurizer liquid level remain within acceptable limits.

c. If the low SG pressure safety injection (SI) signal is being credited for actuating the AFW pumps, please provide the AFW system response time to this signal (this may be different from response time of the AFW system to a low SG level signal).

d. In reviewing the AFW flow characteristics of Figure U.7-1, from the July 8, 2005, RAI response, the NRC staff requests that you provide the AFW flow density as a function of time. The NRC staff notes that the AFW flow ramps up prior to crediting operator action at 924 seconds. Please explain the reason for the flow ramp upward?

3. The NRC staff's SE dated December 1, 1998, states that a utility's use of the WRB-2M correlation with a departure from nucleate boiling (DNB) ratio limit of 1.14 for plant safety analyses, as described in approved topical report, WCAP-15025-P-A, may be approved and used provided the specified four conditions are met. Condition 4 states the correlation should not be used outside its range of applicability defined by the range of the test data from which it was developed. The NRC staff requests that the licensee provide a table with the data that demonstrates that the use of this correlation at BVPS-1 is within the range of parameters stated in WCAP-15025-P-A, Table 4-1, and that condition 4 of the December 1, 1998, SE for use of the WRB-2M correlation has been met.

4. Technical specification change no. 8 listed in the licensee's April 13, 2005, RSG LAR consists of modifying the overpower delta temperature (OPΔT) and the overtemperature delta temperature (OTΔT) equations in TS 3.3.1.1, "Reactor Trip System Instrumentation." The NRC staff requests that the licensee provide the technical justification for why it is acceptable to add the lag compensators, yet leave out the lead/lag compensation function on both sides of the equation. Was the omission of the function and physical installation on the original design of the plant compensated for originally, and is there an effect on the plant that should be accounted for as an effect of this amendment?

5. Section 5.4 of the April 13, 2005, RSG LAR states that an operational response analysis of steam generator tube rupture (SGTR) was performed for BVPS-1. The NRC staff requests that the licensee provide a table listing the sequence of events and times from break initiation to event termination that shows operators can terminate the break flow from the ruptured SG within 51 minutes of accident initiation for the SG replacement and extended power uprate (EPU) conditions. Additionally, the staff requests that the licensee provide the results of the SGTR thermal-hydraulic analysis over time showing the pressurizer pressure, intact and ruptured SGs pressures, and ruptured SG water volume for the analysis to demonstrate that no overfilling of the SG occurs.

6. The NRC staff requests that the licensee provide the technical justification as to why a transient break flow analysis was not performed for the SGTR and why the 30-minute release assumption is conservative for the analysis when termination of the event exceeds 30 minutes.

7. Table 5.3.20-1A states that a generic (W) evaluation addresses peak pressures for the rod withdrawal-at-power accident analysis. In the licensee's response to RAI H.3 of Section 5.3.3 of L-05-112, the licensee references this generic W report again. Please state the name of the W report being referenced, and confirm that it is applicable to BVPS-1.

8. In Section 5.6.2.2.13, "Core Decay Heat," of the April 13, 2005, RSG LAR, the licensee states that the core decay heat generation assumed in calculating the MSLB mass and energy (M&E) release is based on the 1979 American Nuclear Society's (ANS's) Decay

Heat + 2σ model. The licensee states that this is a deviation from the current licensing-basis analysis of MSLB M&E releases outside containment for BVPS-1 which uses the 1971 standard (+20% uncertainty). For the NRC staff to review your use of the 1979 ANS Decay Heat model, please provide the following additional information:

a. Tabulated Decay Heat Curve used in the analysis

b. Assumptions used in tabulating your decay heat curve, and a basis for each assumption. These should include:
(1) U-238 Fission Fraction
(2) U-235 Fission Fraction
(3) Pu-239 Fission Fraction
(4) Other heavy element decay heat sources and their contributions to the total decay heat power
(5) Total recoverable energy associated with one fission and the standard deviation of this value
(6) Length of time fuel is assumed to be at full power
(7) Inputs into the calculation of the factor which accounts for neutron capture, or state if Table 10 of the 1979 ANS standard was used

The following questions refer to the licensee's RAI response dated July 8, 2005, related to the EPU LAR.

Steam System Piping Failure at Full Power

1. RAI W.7 requested the licensee to demonstrate that inclusion of a single failure and/or loss of offsite power (LOOP) would not create a more severe scenario. In response to RAI W.7, the licensee stated that a LOOP was considered in the post-trip hot zero power (HZP) MSLB event but was less limiting than the case with full RCS flow.

   a. The timing of a LOOP is different for the locked-rotor event and the feedwater line break event in the BVPS-1 and 2 UFSARs. Define the licensing basis for the timing of the LOOP relative to the reactor trip signal.

   b. Demonstrate that a LOOP would not create a more limiting scenario for the hot full power (HFP) MSLB event.

2. In the EPU LAR, DNB is used as the sole criteria for clad failure. With the rapid power excursion associated with the pre-trip MSLB event (beyond 120%), the licensee must demonstrate that clad failure does not occur as a result of excessive clad strain due to fuel thermal expansion or incipient fuel melt. Demonstrate that local power peaking does not produce clad failure as a result of clad strain or quantify the amount of clad failure and provide an estimate of the radiological consequences.

Steam System Piping Failure at HZP - Post-Trip Return-to-Power

1. RAI Q.16 requested the licensee to demonstrate that clad failure does not occur as a result of high local power density. In response, the licensee stated that DNB has been taken to be the necessary and sufficient demonstration that fuel clad failure does not

occur. The NRC staff does not accept this position. With the high local power peaking factors beneath the stuck-out control rod and a predicted post-trip return-to-power, the licensee must demonstrate that clad failure does not occur as a result of excessive clad strain due to fuel thermal expansion or incipient fuel melt. Demonstrate that local power peaking does not produce clad failure as a result of clad strain or quantify the amount of clad failure and provide an estimate of the radiological consequences.

2. In response to RAI Q.17, the licensee stated that a DNB ratio (DNBR) multiplier of 0.88 was used with the W-3 correlation. Please provide further discussion on the basis of the 0.88 multiplier.

Boron Dilution

1. In the response to RAI J.2 the licensee states, "The Mode 3 initial and critical concentrations are typically the most difficult to meet. Thus, in order to facilitate future reload evaluations, the Mode 3 initial and critical concentrations were adjusted so that the analytical limit of 15 minutes is met. Using overly conservative boron concentrations in the Mode 3 boron dilution calculation gives the core designers more flexibility during subsequent reload evaluations." Please provide a description of the adjustments that are made to the Mode 3 initial and critical concentrations and how they result in conservative boron concentrations in Mode 3.

2. In the response to RAI A.1 the licensee provided Table A.1-9, “Uncontrolled Boron Dilution” (EPU Licensing Report Section 5.3.5), and in response to RAI A.5 the licensee provided Table A.5-1, “Licensing Basis Safety Analyses.” In these tables several input parameters are described as nominal values. Explain how the use of these nominal values meets the requirement to use conservative values as indicated by the NRC review standard, RS-001, and NUREG-0800, “Standard Review Plan,” Section 15.4.6.

Beaver Valley Power Station, Unit Nos. 1 and 2

cc:

Mary O'Reilly, Attorney
FirstEnergy Nuclear Operating Company
FirstEnergy Corporation
76 South Main Street
Akron, OH 44308

FirstEnergy Nuclear Operating Company
Regulatory Affairs/Performance
Improvement
Larry R. Freeland, Manager
Beaver Valley Power Station
Post Office Box 4, BV-A
Shippingport, PA 15077

Commissioner James R. Lewis
West Virginia Division of Labor
749-B, Building No. 6
Capitol Complex
Charleston, WV 25305

Director, Utilities Department
Public Utilities Commission
180 East Broad Street
Columbus, OH 43266-0573

Director, Pennsylvania Emergency
Management Agency
2605 Interstate Dr.
Harrisburg, PA 17110-9364

Ohio EPA-DERR
ATTN: Zack A. Clayton
Post Office Box 1049
Columbus, OH 43266-0149

Dr. Judith Johnsrud
National Energy Committee
Sierra Club
433 Orlando Avenue
State College, PA 16803

J. H. Lash, Plant Manager (BV-IPAB)
FirstEnergy Nuclear Operating Company
Beaver Valley Power Station
Post Office Box 4
Shippingport, PA 15077

Director
Bureau of Radiation Protection
Pennsylvania Department of
Environmental Protection
Rachel Carson State Office Building
P.O. Box 8469
Harrisburg, PA 17105-8469

Mayor of the Borough of Shippingport
P O Box 3
Shippingport, PA 15077

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Resident Inspector
U.S. Nuclear Regulatory Commission
Post Office Box 298
Shippingport, PA 15077

FirstEnergy Nuclear Operating Company
Beaver Valley Power Station
ATTN: R. G. Mende, Director
Work Management (BV-IPAB)
Post Office Box 4
Shippingport, PA 15077

FirstEnergy Nuclear Operating Company
Beaver Valley Power Station
Mr. B. F. Sepelak
Post Office Box 4, BV-A
Shippingport, PA 15077