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August 23, 2001
L-01-111

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

Subject: **Beaver Valley Power Station, Unit No. 1 and No. 2**
BV-1 Docket No. 50-334, License No. DPR-66
BV-2 Docket No. 50-412, License No. NPF-73
Response to a Request for Additional Information
In Support of LAR Nos. 219 and 73

This letter provides the FirstEnergy Nuclear Operating Company (FENOC) response to a NRC Request for Additional Information (RAI), dated August 15, 2001, pertaining to FENOC letter L-01-038, dated March 19, 2001, as supplemented by letter L-01-094, dated July 6, 2001. FENOC letter L-01-038 submitted License Amendment Requests (LAR) 219 and 73 that proposed changes to the Beaver Valley Power Station (BVPS) fuel handling accident analyses and containment closure requirements for NRC review and approval. Letter L-01-094 submitted a partial withdrawal of LARs 219 and 73 in response to a NRC issue regarding spent fuel pool performance. The information provided by this letter consists of the following:

- clarification of the material properties used in the Holtec analysis simulations,
- elaboration on the structural modeling used in the Holtec analysis and the sensitivity of the results, and
- further explanation of the results for Case 1 and Case 2 presented in the Holtec analysis report.

The FENOC responses to the RAI are provided in Attachment A of this letter.

FENOC requests NRC approval of License Amendment Requests 219 and 73 prior to the start of the BVPS, Unit No. 1, fourteenth refueling outage (1R14) scheduled to commence on September 1, 2001.

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This information does not change the evaluations or conclusions presented in FENOC letter L-01-038 or L-01-094. If there are any questions concerning this matter, please contact Mr. Thomas S. Cosgrove, Manager Regulatory Affairs, at 724-682-5203.

Sincerely,

L. William Pearce for LWM

Lew W. Myers

Attachment

c: Mr. L. J. Burkhart, Project Manager
Mr. D. M. Kern, Sr. Resident Inspector
Mr. H. J. Miller, NRC Region I Administrator
Mr. D. A. Allard, Director BRP/DEP
Mr. L. E. Ryan (BRP/DEP)

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I, L. William Pearce, being duly sworn, state that I am Plant General Manager FirstEnergy Nuclear Operating Company (FENOC), that I am authorized to sign and file this submittal with the Nuclear Regulatory Commission on behalf of FENOC, and that the statements made and the matters set forth herein pertaining to FENOC are true and correct to the best of my knowledge and belief.

FirstEnergy Nuclear Operating Company

L. William Pearce

L. William Pearce
Plant General Manager - FENOC

COMMONWEALTH OF PENNSYLVANIA

COUNTY OF BEAVER

Subscribed and sworn to me, a Notary Public, in and for the County and State above named, this 23 th day of August, 2001.

Tracey A Baczek

My Commission Expires:

Notarial Seal
Tracey A. Baczek, Notary Public
Shippingport Boro, Beaver County
My Commission Expires Aug. 16, 2005

Member, Pennsylvania Association of Notaries

Response To Request For Additional Information (RAI)
Changes To Fuel Handing Accident Analyses And Containment Closure Requirements For Beaver Valley Power Station, Unit Nos. 1 and 2
Dated August 15, 2001
(License Amendment Request Nos. 219 and 73)

NRC RAI Question 1:

Material properties:

In section 2.0 of the report, it is recognized that the material properties could deviate from the nominal value significantly when the deformation is applied in an impulsive manner or the material is irradiated for a long period of time. Both of these situations are present in the currently postulated drop events. However, there is no discussion that the effects of these two factors have been considered in arriving at the material properties used in the analysis. Please provide additional information regarding this issue.

FirstEnergy Nuclear Operating Company (FENOC) Response

The material properties used in the simulations were the best estimates produced from the test data reported in Ref. 9.5 in the report. The tests performed in that references were on irradiated fuel and were performed at strain rates characteristic of what might be expected to be present in the reactor. Prior to using the results from Ref. 9.5 in the analysis, an independent confirmation of the acceptability of the data was obtained from Westinghouse. Therefore, in the specific analyses performed in the report, the effects of material irradiation and strain rate have been included to the extent possible based on accepted test data.

NRC RAI Question 2:

Structural Modeling:

In section 6.2 of the report, each of the fuel rods and guide tubes is modeled by 18 beam elements. The weight of the fuel pallet in each fuel rod is represented by a mass element at the lower end of the fuel rod. The lateral inertia force on the fuel rod due to the weight of the pallet [sic] will be neglected if the mass of the fuel rod is modeled at the lower end. There is no discussion why this modeling is bounding for determining the integrity of the fuel rod. Please provide this information.

FENOC Response

The configurations analyzed in the report concentrate the pellet mass at the bottom end fitting so as to ensure that the vertical inertia of the pellets does not induce compressive loads in the cladding column. This is in accordance with the assumptions used in the original work on the response of fuel rods reported in UCID-21246, "Dynamic Impact Effects on Spent Fuel Assemblies," LLNL, 1987.

The effect of the assumption in the analyses reported in the current report using the DYNA3D non-linear simulation is to induce more deformation in the lower end fitting than would result if the vertical inertia were distributed along the cladding length (i.e., the entire pellet mass would not act on the end fitting at a single instant in time, but would be spread out over time because of the wave propagation effects). The larger end fitting deformation induces lateral deformation to the cladding adjacent to the end fitting and the cladding ruptures predicted are largely concentrated near the end fitting. Therefore, it is concluded that the results of the analyses are conservative.

During discussions with the review staff, an additional verbal question concerning the effect of the input cladding failure stress level on the model response was raised. The results provided in the report are based on a numerical impact analysis, which were conservatively modeled to obtain an upper bound of damage estimates for the postulated drop accidents. However, to evaluate the sensitivity of the model responses to changes in the level of the cladding failure stress (design input), additional simulations have been performed assuming that the input failure stress is arbitrarily reduced to 95% and 90% of the value used (based on test data) in the simulations documented in the report. This reduces the slope of the cladding stress strain curve subsequent to reaching the yield strength of the cladding. Each fuel rod is defined by eighteen finite elements along its length. Once any one element in a rod reaches the failure stress, the rod is considered to have failed. Multiple failures within a rod do not increase the failed rod total. The additional cases simulated herein are for the dropped fuel assembly impacting onto a target fuel assembly. The results from the simulations with reduced cladding failure stress indicate that a reduction in failure stress of up to 10% may result in up to a 26% increase (36 rods) in the number of reported rod failures.

In the Case 2 analyses, the majority of rod failures occur in the target (fixed) fuel assembly; of the 137 rod failures reported in the submittal for Case 2, 18 occur in the dropped assembly with the remaining 119 in the target assembly.

Note that the dose calculation (ERS-JTL-99-009, Rev. 2) submitted with LAR 1A-219 and 2A-73 provided doses calculated for the rupture of 137 fuel rods at the Control Room, Exclusion Area Boundary (EAB), and Low Population Zone (LPZ) that were well within the applicable regulatory limits of 10 CFR 50.67. Specifically, the most restrictive dose reported was 2.2 rem (TEDE) for the Control Room, which compares to the 10 CFR 50.67 limit of 5 rem TEDE. This dose is directly proportional to the number of ruptured fuel rods assumed. Therefore, a maximum of 311 fuel rods could be ruptured before exceeding the 5 rem (TEDE) Control Room dose limit in 10 CFR 50.67.

NRC RAI Question 3:

Results of fuel drop event Cases 1 and 2:

In case 1, the fuel rods assembly drops from a height of 30' on a rigid floor and it results in a rupture of 74 fuel rods (about 28% of the total fuel rods). In case 2, the fuel rod assembly drops from a height of 16.7' (almost half the height of case 1) on another fuel rod assembly (much more flexible than the rigid floor in case 1) and results in the rupture of about 69 fuel rods in

each of the fuel rods assembly (26% of the total fuel rods, almost the same percentage as in case 1). Considering the differences in the rigidity of the impacting surface (rigid floor or fuel rod assembly), and the difference in the fuel rod assembly drop height, the results do not appear consistent. Please provide the basis for and explanation of the difference in the resultant number of ruptured fuel rods in case 1 and 2.

FENOC Response

It is true that the initial kinetic energy to be absorbed, associated with Case 1, is larger than the corresponding energy associated with Case 2. This does not necessarily translate into a conclusion that more rods should break in Case 1 (or conversely, that less rods should fail in Case 2). A detailed evaluation of the location of the rod failures indicates that in Case 2, the majority of failures occur in the lower (target) assembly. The rods in this assembly are subjected to a substantial increase in direct compression loads as they are "supporting" the entire dropped assembly for some time interval prior to failure. It is apparent that direct compression load increase in the lower assembly fuel rods leads to increased failure in the lower assembly. The difference in time of peak interface force between Case 1 and Case 2 are clearly evident in Figures 3 and 8 of Holtec International Report HI-992343. Figure 8 (Case 2) clearly identifies that there is a compression wave traveling down the target rod and reflecting back to the interface. Therefore, even though the input energy is less, the increased direct compression in the target assembly, plus induced bending near the fitting, leads to an increased number of failures in the lower (target) assembly. This is in contrast with Case 1, where the rod failures are primarily due to the induced bending of the rods near the lower fitting.