# NRC NEWS

**U.S. NUCLEAR REGULATORY COMMISSION**

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## NRC RETURNS FIRSTENERGY APPLICATION TO RENEW LICENSES OF BEAVER VALLEY NUCLEAR POWER PLANT

The Nuclear Regulatory Commission has determined that FirstEnergy Nuclear Operating Company's application to renew the operating licenses of Units 1 and 2 at the Beaver Valley nuclear power plant for an additional 20 years is unacceptable for docketing. The NRC requires that applications such as this meet very high quality standards, and the Beaver Valley application has not yet done so.

FirstEnergy Nuclear submitted the license renewal application Feb. 9, and NRC staff have been reviewing the application since that time. The staff had previously discussed, both with FirstEnergy and other licensees, quality issues with recent license renewal applications. Beaver Valley continues to operate safely.

"The NRC's primary mission is ensuring protection of public health and safety, and we can't do that for an additional 20 years of Beaver Valley operation unless we have complete, accurate and up-to-date information on the plant," said David Matthews, Director of the Division of Regulatory Improvement Programs in the NRC's Office of Nuclear Reactor Regulation. "Given the gaps in the current application, we simply could not properly review FirstEnergy's request."

In a March 24 letter to FirstEnergy (attached), the NRC gives the company 30 days to decide what course of action it will take concerning Beaver Valley's license renewal. If FirstEnergy submits a revised application that passes the staff's initial review and audit activities, the NRC will establish a review schedule for Beaver Valley.

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Mr. L. William Pearce
Vice President
Beaver Valley Power Station, Units 1 and 2
FirstEnergy Nuclear Operating Company
Post Office Box 4, Route 168
Shippingport, PA 15077

SUBJECT: BEAVER VALLEY POWER STATION, UNITS 1 AND 2 - RESULTS OF ACCEPTANCE REVIEW FOR LICENSE RENEWAL (TAC NOS. MC5913 AND MC5914)

Dear Mr. Pearce:

By letter dated February 9, 2005, FirstEnergy Nuclear Operating Company (FENOC) submitted an application for renewal of Operating License Nos. DPR-66 and NPF-73 for the Beaver Valley Power Station (BVPS), Unit 1 and Unit 2, respectively. Notice of receipt of this application was published in the Federal Register on March 4, 2005, (70 FR 10694). The purpose of this letter is to provide the results of the NRC staff's acceptance review of the license renewal application for BVPS. The acceptance review determines whether or not the application is sufficiently complete to allow the NRC staff to proceed with its detailed technical review.

The NRC staff has reviewed your request following the guidance in NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," and determined that the application is not complete and is not acceptable for docketing. A description of the deficiencies found in the application is included in the enclosure. FENOC has the opportunity to modify its application to provide the missing and incomplete information. The staff will review a revised application, if submitted, to determine whether it is acceptable for docketing.

We request that you notify us in writing within 30 days of the issuance of this letter of your plans with respect to your renewal application. If you have any questions on this matter, please contact the NRC Project Manager, Kimberley Corp, at 301-415-1091 or e-mail kar1@nrc.gov.

Sincerely,

Pao-Tsin Kuo, Program Director
License Renewal and Environmental Impacts Program
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Docket Nos. 50-334 and 50-412

Enclosure: As stated

DESCRIPTION OF DEFICIENCIES FOUND IN

BEAVER VALLEY POWER STATION, UNIT NOS. 1 AND 2,

LICENSE RENEWAL APPLICATION

By letter dated February 9, 2005, FirstEnergy Nuclear Operating Company (FENOC) submitted a license renewal application (LRA) for Beaver Valley Power Station (BVPS), Units 1 and 2. In accordance with 10 CFR 2.101(a), the NRC staff performed a review of the LRA following the Guidance in NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," to determine whether the LRA is complete and acceptable for docketing. The staff has found deficiencies in the LRA and has grouped examples of these deficiencies into eight general areas. Each portion of the LRA that the staff examined contains similar defects. Accordingly, should the applicant determine to revise the application, the applicant should closely examine all the information in the application to correct all similar deficiencies. The examples are provided under each of the following categories, however, they are not exhaustive:

**(I)** Information that is too general

Some examples are:

C The LRA provides information that is too general and inappropriate for review. For example, description of material type as "ANY" is listed in the aging management review (AMR). The staff cannot perform a specific review on "ANY" materials.

C The LRA cites component names that are too general:

- BVPS repeatedly lists material/environment instead of the component name. Material and environment are factors to be considered in aging management and, without more information, do not identify and list a component. For example, "structural steel in air," "concrete in air," and "polymer in soil above the ground water table" are inappropriately listed as components. If this type of listing is an attempt to identify commodity groups, it fails to identify the types or classes of components or structures falling within those groups.

- BVPS repeatedly lists "non-safety related (NSR) fluid-retaining components in safety-related buildings and areas" as components. 10 CFR 54.21(a)(1) requires the identification and listing of structures and components subject to an AMR. However, BVPS has not identified the structures and components within the scope of 10 CFR 54.4(a)(2).

- BVPS identified components that are too general for an AMR. For example, the LRA lists "Vessel (pressurizers)" as components, which are to be managed by the nickel-alloy nozzles and penetrations aging management program (AMP). In the LRA, however, this AMP only discusses reactor vessel control rod drive mechanisms (CRDM) and does not refer to pressurizers. The pressurizer nozzles should have been identified separately in the AMR and matched to an AMP, as appropriate.

C The LRA cites intended functions that are too general. For example, BVPS lists "NSR Functional Support" as an intended function. NSR Functional Support can represent different component intended functions that would affect the associated AMR.

C The LRA does not identify which piping and pipe fittings are buried. Also, it is not clear whether jockey pumps, fire hydrants, and hose stations are included within the scope of license renewal and subject to an AMR.

C The time-limited aging analysis (TLAA) for the metal fatigue program in the LRA did not identify plant-specific critical fatigue locations with high cumulative usage factors. The AMP description for the metal fatigue of the reactor coolant pressure boundary only includes generic locations described in NUREG/CR-6260.

C The definitions of the Notes shown in the LRA Tables 3.X.2-Y are not specific or valid (these tables appear in Chapter 3 of the LRA for multiple [X] groups of systems and multiple [Y] individual systems/components and are denoted as Tables 3.X.2-Y). The Nuclear Energy Institute (NEI) has established an agreed upon standard list of definitions, "Consistency Notes for Aging Management Review Results A-H," as set forth in NEI 95-10, Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule, Rev. 5, Nuclear Energy Institute, January 2005. The LRA deviates from this standard list by adding ambiguous definitions that are not consistent with NEI 95-10.

- Application Tables 3.X.2-Y, Note U states, "Specific material/environment combinations were not determined. Comparison to NUREG-1801 is not possible." The staff cannot evaluate any items to which Note U applies because an AMR is not possible in the absence of the identification of specific materials and environments.

- Application Tables 3.X.2-Y, Notes R, S, T, and V mention aging mechanisms. Aging mechanisms are not identified in the corresponding LRA tables. Accordingly, the significance of these notes is not apparent. Otherwise, these notes are the same as A, E, C, and B, in NEI 95-10, respectively. The staff has no objection to these notes in so far as they conform to notes in NEI 95-10.

**(II)** Insufficient Basis for Conclusions

Some examples are:

- The LRA does not provide bases for the conclusions stated under the "Discussion" column in Tables 3.X.1. In Section 3.5 of the LRA, all items included under the "Further Evaluation Recommended" column were dispositioned by the applicant under the "Discussion" column as either "not applicable" or "consistent with NUREG-1801." During the review, the NRC staff searched LRA Section 3.5 for additional information for items listed in this manner under the "Further Evaluation Recommended" and "Discussion" columns, and there is no sub-section in

Section 3.5 that specifically describes what further evaluation the applicant proposes for such items.

- The LRA states (on page 3-22) that the NUREG-1801 program for managing the loss of material due to pitting and crevice corrosion in the steam generator shell assembly is not applicable because the flow accelerated corrosion (FAC) will dominate the listed aging effects. The LRA indicates that further evaluation is documented in Appendix B of the LRA for the Steam Generator Tube Integrity (SGTI) program. However, there is no basis on why FAC will dominate other aging effects, and there is no information in Appendix B of the LRA to justify how a program to manage tube degradation can also manage loss of material from the steam generator shell.

- For Item 3.5.1-02 (on page 3-398), the LRA states that crack initiation and growth due to stress corrosion cracking (SCC)/cyclic loading is not applicable because BVPS components with this aging effect/mechanism are considered under row 3.5.1-01 (fatigue TLAA) on the same page. Row 3.5.1-01, however, indicates it is not applicable because there is no fatigue analysis. Thus, there is no information in the LRA to indicate how crack initiation and growth due to SCC/cyclic loading will be managed for license renewal.

**(III)** <u>Inconsistencies and Inadequate QA of Renewal Application</u>

Some examples are:

- The LRA does not clearly mark the system boundaries in the LRA drawings to identify which systems, structures, and components (SSCs) are within scope and the use of markings is inconsistent. For example, LRA Section 2.1.2.1, License Renewal (LR) Boundaries, page 2-9, states that LR boundaries are defined by identifying the boundaries associated with the intended functions of the system. License renewal drawing conventions are not defined. There is no explanation of what the symbols in the "boundary flags" mean. An example of this is "LR-1-09" found on LR-8700-RM-409-1, which is not defined in the application. Also, LRA Section 2.1.2.1, LR Boundaries, page 2-9, describes how the applicant defined the physical/functional boundaries of a system. It states that "component designations" were used from an Asset Equipment List (AEL) but does not identify what these component designations are.

- The LRA also does not clearly identify the boundaries of complex assemblies in the drawings. LRA Section 2.1.2.3, Paragraph 6, Page 2-11 states : "Some structures and components, when combined, are considered a complex assembly. For purposes of performing an AMR, boundaries of review were clearly established. BVPS results consider certain emergency diesel generator system components to be part of a complex assembly, based on their association with the active diesel engine component." However, complex assemblies are neither identified in the LRA nor are their boundaries shown on the LRA drawings. For example, LRA Section 2.3.3.17, Emergency Response Facility (ERF) Substation System, states that the ERF Substation receives standby power from a 2500 kw diesel generator, and LRA drawing 8700-RM-458E-2 shows the RG-EG-1 complex at location E2, but does not identify the boundaries of the ERF substation diesel generator. Similarly, LRA Drawing 8700-RM-458E-1 shows the engine mounted valve block at location B7, but does not identify the boundaries for the ERF substation diesel generator.

- The LRA tables do not correctly identify notes in the “Notes” column.

  - In Table 3.3.2-26 (on page 3-297), stainless steel piping components in treated water or steam environments are identified as subject to crack initiation and growth. The application assigns Note “Q” to such components, i.e., the component is identified as “not in NUREG-1801 for this material, environment, and aging effect.” The staff does not understand this, since this is a very common combination for component, material, environment, and aging effect in NUREG-1801. For example, in NUREG-1801, Item A.1.1 lists stainless steel piping and fittings in treated water for crack initiation and growth.

  - In LRA Table 3.1.2-1 (on page 3-37), the last line item on the page states that the external surfaces-pressurizer, which are made from materials of carbon steel, low alloy steel, and ductile malleable cast iron, are exposed to environments in air with potential for leaking borated water. This component item will experience the aging effect of loss of material and will be managed by the Boric Acid Corrosion (BAC) program. The application assigns Note “A” to this component, i.e., “consistent with NUREG-1801 item for component, material, environment, and aging effect.” The application states that this program is consistent with the NUREG-1801 AMP in Item IV.C2.5-b. However, this reference in NUREG-1801 only lists low-alloy steel as the applicable material.

  - In Table 3.3.2-1 (on page 3-172), the line item states that the “collar” component which is made of polymers and is used for pressure boundary is exposed to an air environment. The LRA states that this line item will experience crack initiation and growth, hardening and shrinkage, and loss of strength as its aging effect and will be managed by the System Monitoring program. The LRA assigns Note “A” to this component, i.e., that the AMP is consistent with NUREG-1801. However, in Appendix B of the LRA, the System Monitoring program is listed as plant specific.

**(IV)** Insufficient Explanation of Information

Some examples are:

- The scoping methodology description is incomprehensible. The LRA discussion of the methodology for identifying systems, structures, and components within scope under 10 CFR 54.4(a)(1) does not explain why some components classified as safety related were not included in scope. The LRA simply states, “This equipment was not considered within the scope of the LR Rule simply because of its classification.” The staff does not understand this statement.

- BVPS uses system realignment for scoping. However, the LRA does not explain the method used to realign components from one system to another system for scoping purposes. LRA Section 2.1.1.5, General Scoping Discussion, page 2-8, describes that some systems may contain only a few components that support intended functions. Further, it explains that for these types of systems, those few components may be realigned to an interfacing system, and as a result, the system that originally contained in scope components would be identified as not within the scope of license renewal. LRA Table 2.2-1, Plant Level Scoping Results, does not identify how specific components are realigned. In LRA Section 2.2, Plant Level Scoping Results, it is necessary to understand the effects of realignment.

- The SGTI is stated to be an existing program that with enhancements will be consistent with NUREG-1801. The LRA includes a vague statement regarding enhancement of the SGTI program, with no detailed description of the enhancement to credit this program to manage aging of the steam generator shell. Enhancement of the existing SGTI program is beyond the scope of the corresponding AMP in NUREG-1801. In fact, NUREG-1801 identifies other programs to manage the steam generator shell.

**(V)** <u>Lack of Non-Safety Related Structures and Components Scoping</u>

Some examples are:

- The LRA discussion of methodology and results for the scoping of 10 CFR 54.4(a)(2) components is vague and lacks description. The LRA does not provide an inclusive list of such components nor does the LRA include exclusion criteria. Components in scope under 10 CFR 54.4(a)(2) are simply listed as "NSR fluid-retaining components in safety-related buildings and areas."

- In Section 2.1.1.3, the methodology for scoping 10 CFR 54.4(a)(2) components is incomplete. The components identified as within the scope of license renewal resulting from 10 CFR 54.4(a)(2) scoping are not identified in the boundary drawings nor listed in the LRA tables, and other such components that are subject to an aging management review are not listed in the LRA tables. Some examples are LRA Tables 2.3.3.8-1, 2.3.3.24-1, and 2.3.4.7-1.

- There is no discussion of anchorage or the concept of an equivalent anchor in the LRA.

- The LRA also does not include how sources of design basis information for NSR components were used. Also plant specific and industry operating experience with respect to such components were not discussed.

**(VI)** <u>Insufficient Information for Severe Accident Mitigation Alternatives (SAMA) Analysis</u>

- The Environmental Report (ER) did not include a description of any unresolved probabilistic risk assessment (PRA) peer review findings or the potential impact of unresolved findings on the results of the SAMA analysis.

- The description of the chronology and interim results of the various updates to the Level 1 and Level 2 PRA since the Individual Plant Examination (IPE) was not adequate. Section C.1.1 appears to address only those changes made since the last revision in 1998. Core damage frequency (CDF) and large early release frequency (LERF) values are not provided. The ER also did not describe any changes to the binning of Level 1 sequences into release categories, nor did it describe changes to source terms used to represent each release category.

- The ER did not include a discussion of the major differences between the Unit 1 and Unit 2 plant designs and PRAs and how these are related to differences in the risk profiles for the two units.

- The ER did not provide a listing of the accident sequences contributing to each release category and the respective frequencies of each sequence; the source terms used in the MACCS2 analysis for each of the release categories, including release fractions, release time and duration, warning time, release height, and release energy; and the basis for the source terms.

- The SAMA identification process did not consider sequences important to population dose rather than LERF to identify additional SAMAS. Accordingly, no further assessment of any such SAMAs could be provided.

- The ER provided an inadequate description of the SAMA screening process and the screening criteria used. The ER did not identify the number of SAMAs eliminated by the application of each criterion.

- The ER failed to provide a description of the implementation status of each of the potential improvements identified in the IPE and individual plant examination of external events (IPEEE). No justification as to why these potential improvements were not implemented or addressed by a SAMA were provided.

- The ER did not include adequate discussion on how the reduction in population dose for each SAMA was estimated. Section C.3 provides estimates only for the reduction in CDF.

**(VII)** <u>Lack of Discussion of Recent Renewal Review Experiences</u>

Some examples are:

- Leak-before-break (LBB) analysis cannot be considered as an acceptable cast austenitic stainless steel thermal aging management program. The LRA does not include a flaw tolerance evaluation or enhanced volumetric inspection. Rather than using one of these inspection methods, the LRA substitutes LBB analysis.

- The LRA AMP for buried piping and tanks is solely opportunistic for inspections. The LRA does not identify at least one inspection during the first ten years of extended operations in the AMP.

- The TLAA for pressurized thermal shock (PTS) and Upper Shelf Energy does not include actual materials data, calculations and 10 CFR 50.61 analysis for 54 effective full power years (EFPY). This number is more representative of the EFPY at the end of the period of extended operation because of the high capacity factor based on current operating experience. BVPS only performed 10 CFR 50.61 analysis for 48 EFPY.

**(VIII)** <u>Technically Incorrect Information</u>

Some examples are:

- The LRA states that cast iron is used in the reactor coolant system (RCS). The staff has no knowledge of cast iron in the RCS for any plant.

- The LRA credits the Flow Accelerated Corrosion (FAC) program to manage cracking. However, FAC manages wall thinning, but cannot detect cracking.

- The LRA credits the Boric Acid Corrosion (BAC) program to manage cracking initiation and growth for RCS pressure boundary bolting and for managing loss of material for stainless steel components. However, BAC manages loss of material due to boric acid corrosion, but cannot detect stress corrosion cracking (SCC). In addition, stainless steel components are not susceptible to boric acid corrosion.

- The LRA credits the Metal Fatigue of Reactor Coolant Pressure Boundary program to manage crack initiation and growth for all mechanical systems. However, this program addresses the time-limited aging analysis (TLAA) for fatigue, but cannot detect SCC.

<u>Other Staff's Observations</u>:

In addition to the above concerns, the staff has identified matters that should be addressed should the applicant decide to revise the LRA. In particular, because of realignment, some unrelated structures are being put in common groups or identical items are being treated inconsistently, and this has caused longer review and required more staff resources. One such example is:

The LRA Section 2.4.2.2 is titled "Auxiliary Building," but includes the Service Buildings and Diesel Generator Buildings of both units and Unit 1's Fuel Building, Primary Water Storage Building, Safeguards Building, and Solid Waste Building. LRA Section 2.4.2.10 is titled "Safeguards Building," but does not include Unit 1's Safeguards Building and includes Units 2's Fuel and Decontamination Building, Main Steam Valve and Cable Vault area, Primary Demineralized Water Storage Tank Enclosure, Refueling Water Storage Tank Foundation Mat and Shield Wall, and Service Water Valve Pits. The presentation is inconsistent in that Unit 2's Safeguards Building is under the title of "Safeguard Building" while the Unit 1's Safeguard Building is under the title of "Auxiliary Building", and that Unit 1's Fuel Building is under the title of "Auxiliary Building" while Unit 2's Fuel Building is under the title of "Safeguards Building".

Another example is that LRA Table 3.X.2-Y, Notes K and P seem to state the same thing. This is confusing to the staff.