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December 3, 2012

L-12-349

10 CFR 50.59(d)(2)

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

SUBJECT:
Beaver Valley Power Station, Unit No. 2
Docket No. 50-412, License No. NPF-73
Report of Facility Changes, Tests and Experiments

In accordance with 10 CFR 50.59(d)(2), the FirstEnergy Nuclear Operating Company hereby submits the attached Report of Facility Changes, Tests and Experiments for Beaver Valley Power Station, Unit No. 2. The report covers the period of October 29, 2010 through September 24, 2012.

There are no regulatory commitments contained in this letter. If there are any questions or if additional information is required, please contact Mr. Thomas A. Lentz, Manager – Fleet Licensing, at (330) 315-6810.

Sincerely,

Paul A. Harden

Attachment:
Beaver Valley Power Station, Unit No. 2, Report of Facility Changes, Tests and Experiments, October 29, 2010 through September 24, 2012

cc: NRC Region I Administrator
NRC Resident Inspector
NRC Project Manager
Director BRP/DEP
Site BRP/DEP Representative

Beaver Valley Power Station, Unit No. 2,
Report of Facility Changes, Tests and Experiments
October 29, 2010 through September 24, 2012

Evaluation No. : 10-04767, Revision 0

Title: Revise BVPS-2 UFSAR Section 9.1.2.2, "Spent Fuel Pool Storage"

Activity Description:
Beaver Valley Power Station, Unit No. 2 (BVPS-2), Updated Final Safety Analysis Report (UFSAR), Section 9.1.2.2, stated that "the Spent Fuel Pool is sized to accommodate the storage of a minimum of one full core in the event the reactor must be emptied of fuel at any time during BVPS-2 life." This statement implies that full core offload capacity, termed "full core reserve storage capability," is a requirement to ensure that the spent fuel pool (SFP) must have sufficient space to accommodate the offload of all 157 assemblies from the BVPS-2 reactor core at any time.

Full core storage capability is not an NRC licensing or safety requirement, as determined by a review of both the industry and the specific BVPS-2 licensing bases.

The current UFSAR 9.1.2.2 statement is correct, only if some or all of the spent fuel discharged over the operating life of BVPS was stored in some type of long term depository external to the site. Construction of such a facility was an unstated assumption at the time the nuclear fleet of the United States was first constructed and was thus implicit in the BVPS-2 initial licensing process in 1987. However, such a facility has never been constructed, rendering the background information incorrect. Increasing the storage capacity of individual plants must be employed instead. The proposed clarification acknowledges that maintaining full core offload storage capacity is simply a prudent management practice, rather than a licensing bases requirement. This full offload capacity may be temporarily lost due to the need for additional regulatory review and approval of on-site storage facilities, when the existing SFP capacity is challenged.

The statement in Section 9.1.2.2 indicating the SFP can accept a full core offload at any time in BVPS-2 plant life was deleted, with an additional paragraph inserted explaining the position discussed above.

Summary of Evaluation:
There are no regulatory requirements either in the Code of Federal Regulations or the NUREGs (including NUREG 0800, Standard Review Plan) to have the ability to perform full core offloads. The statement that BVPS-2 would maintain that capability over its entire initial licensed operating life of 40 years was incorrect, since the same discussion in UFSAR Section 9.1.2.2 notes that the spent fuel pool storage capability is 1,088 assemblies, which is insufficient to support approximate 26 total 1.5 year-long operating cycles (40 years/1.5 years per operating cycle) averaging 57 assemblies per refueling (average based on a review of refueling data through the fall 2009 refueling outage [2R14]). Furthermore, there is no design analysis crediting the ability to perform full core offload to mitigate any identified design basis accident, and no discussion of such a mitigation strategy in the BVPS-2 emergency operating procedures. Additionally, the beyond design bases mitigation

strategies of the severe accident management guidelines do not discuss use of full core offload.

A review of the safety evaluation report for BVPS-2, as well as the standard review plan, leads to the conclusion that the discussion of core offload was meant to demonstrate that, should full core offloads be performed, the existing SFP cooling capability was sufficient to accommodate the heat load. BVPS-2 determined the existing SFP cooling was adequate by analysis, which the NRC accepted.

Evaluation No. : 11-00626, Revision 1

Title: Assessment of Beaver Valley Unit 2 Containment Response for Design Basis Accidents for Containment Atmospheric Conversion Project

Activity Description:
Calculation titled "Assessment of Beaver Valley Unit 2 Containment Response for Design Basis Accidents for Containment Atmospheric Conversion Project," Revision 2, is the approved calculation for the containment integrity analysis and recirculation spray pump net positive suction head for large break loss-of-coolant accident (LBLOCA) and main steam line break (MSLB). The mass and energy (M&E) release data for both LBLOCA and MSLB were developed and supplied by Westinghouse. Some inputs to the Westinghouse M&E calculations vary on a cycle to cycle basis and were confirmed to be bounding by the reload process. One input parameter affecting the MSLB M&E data is the fuel moderator density coefficient (MDC). For BVPS-2 Cycle 16 (April 2011 through October 2012), it was determined that the MDC assumed in the MSLB M&E analysis would not be bounding for the hot zero (HZP) power case. Therefore, a decision was made to increase the assumed MDC and calculate new M&E data for this case to be used in the containment analysis. The bounding 0 percent power case was chosen and assumed single failure of a main steam isolation valve to close. This case was designated as case 16M in the containment calculation. This addendum re-analyzed case 16M to determine the resulting changes in containment pressure, temperature and liner temperature. Associated UFSAR updates were also covered by this evaluation. The changes included the Section 6.2.1.1.3.7 discussion of MSLB results and Tables 6.2-10 and 6.2-11, as well as Figures 6.2-12 through 6.2-15. A change to the Technical Specification Bases, Sections 3.6.5, 3.6.6, and 3.6.7 was also required and covered by this evaluation.

Summary of Evaluation:
The hot zero power MSLB M&E release analysis incorporated input changes, which had the effect of increasing the return to power following the event. This resulted in an increase in the steam release to containment, which resulted in higher containment pressure and temperature. The peak containment pressure and containment temperature and liner temperature remained within acceptance limits. There was no impact on the reactor departure from nucleate boiling ratio (DNBR) analysis and the steam release for offsite and control room dose results were not changed. The changes did not introduce the possibility of a new accident or malfunction. There was no change in the method of evaluation, and no fission product barriers were challenged. All questions were answered no; therefore, a license amendment was not required.

Evaluation No. : 11-05225, Revision 0

Title: Pressurizer Surge Line LBB/SWOL Analysis (BVPS-2)

Activity Description:
In 1987, a leak-before-break (LBB) evaluation was performed for the Beaver Valley Power Station, Unit 2 (BVPS-2) pressurizer surge line (PSL) and approved by the Nuclear Regulatory Commission (NRC). In 1988, the original LBB analysis was updated to evaluate the effects of thermal stratification, and subsequently approved by the NRC. An updated PSL LBB evaluation was also performed for the 9.4 percent power uprate program.

To mitigate primary water stress corrosion cracking (PWSCC), BVPS-2 applied a structural weld overlay (SWOL) at the Alloy 82/182 weld location of the PSL to the pressurizer nozzle. The SWOL analysis is documented in WCAP-16612-P, Revision 0, "Beaver Valley Unit 2 Pressurizer Safety/Relief, Spray, and Surge Nozzles Structural Weld Overlay Qualification," September 2006. The subsequent LBB evaluation results are documented in WCAP-17394-P, Revision 0, "Leak-Before-Break Analysis, Update for the Beaver Valley Unit 2 Pressurizer Surge Line," December, 2011.

The NRC, in RIS 2010-07, "Regulatory Requirements for Application of Weld Overlays and Other Mitigation Techniques in Piping Systems Approved for Leak-Before-Break," states that any planned changes to an approved LBB analysis must be evaluated under 10 CFR 50.59.

Summary of Evaluation:
The original LBB evaluation methodology did not include PWSCC effects for leak rate calculations of the PSL because the PWSCC issue for Alloy 82/182 welds had not been identified at that time. The updated methodology, including SWOL, was approved by the NRC for the Waterford Steam Electric Station, Unit No. 3, via Amendment 232. Its application to BVPS-2 was consistent with the NRC approval. Therefore, the methodology change is acceptable under 10 CFR 50.59, and a License Amendment was not required.

Evaluation No. : 12-03459, Revision 0

Title: BVPS-2 Yard Excavation Between Pipe Trench and Electrical Manhole

Activity Description:
A through-wall leak was identified on an 8-inch service water line. The leak was located in a Beaver Valley Power Station, Unit No. 2 (BVPS-2) pipe trench between the service building and safeguards building. To support repair of the pipe leak, a yard excavation was made between the west wall of the pipe trench and the east wall of an adjacent electrical manhole. During the yard excavation activities, missile protection for safety-related electrical components necessary for safe shutdown was degraded. This equipment was maintained operable with the reduced missile protection. The excavation was performed per an approved work order.

The excavation permitted the implementation of a temporary modification that cut an opening in the pipe trench wall, allowing access for the pipe repair. The excavation uncovered the west side of the pipe trench and the east side of an adjacent electrical manhole, and was approximately 13 feet deep. It extended to the bottom of the pipe trench. About 1'-2" below the bottom of the pipe trench was an electrical duct. The duct runs east from the electrical manhole and passes under the pipe trench.

Excavation also required temporary removal of a portion of a yard storm water drainage pipe. This pipe drains storm water from the trough adjacent to the service building roll-up door and empties into a yard catch basin.

The work order provided measures to limit the amount of storm water entering the excavation and the roll-up door trough. A sump pump in the excavation was available to remove any storm water entering the excavation.

The electrical manhole, the electrical duct and the pipe trench are safety-related structures. The yard storm water drainage pipe is non-safety related, non-seismic.

Summary of Evaluation:
The requirement of the BVPS-2 UFSAR, Section 3.5.1.4, and safety evaluation report section 3.5.2 to maintain protection of structures, systems and components important to safety from tornado-generated missiles could not be maintained during the excavation activities. During excavation, one safety-related duct bank and one safety-related electrical manhole was uncovered and the 5 feet of soil missile protection was not maintained as defined in UFSAR Section 2.2.3.2. Soil cover depth is defined in a BVPS-2 document titled "Structural Design Criteria." The contingency plan included staging fill material in the immediate vicinity of the excavation with equipment available to refill the excavation within one hour of any tornado watch in accordance with a procedure titled "Acts of Nature - Tornado or High Wind Condition." Refilling the excavated hole restored the UFSAR described tornado-generated missile protection.

There have been several previous probabilistic risk assessment (PRA) evaluations for similar activities at BVPS-2. A PRA evaluation was completed for the risk analysis of the BVPS 1 and BVPS 2 equipment hatch missile shield removal. This evaluation used the latest PRA model for Unit 2. This PRA model also utilized missile probability in the evaluation per UFSAR Section 3.3.2. It concluded that the probability of a tornado-generated missile hitting the exposed equipment hatch and causing damage is 4.20E-9. The area of the unprotected equipment hatch used is 170 square feet.

The trench created by the work order had a smaller area – about 100 square feet. As such, the PRA evaluation bounded the proposed excavation activity and by comparison, plant operation proceeded since it was highly unlikely that core damage would occur as a result of the excavation.

FirstEnergy Nuclear Operating Company concluded the proposed activity could proceed without obtaining a license amendment.

Evaluation No. : 12-03475, Revision 0

Title: Reactor Coolant Pump Shutdown Seal

Activity Description:
The proposed activity was to replace the reactor coolant pump (RCP) number 1 seal inserts in the three Beaver Valley Power Station, Unit 2 (BVPS-2) RCPs with a modified design called the SHIELD® Shutdown Seal (SDS). This included replacement of the existing number 1 runner retainer sleeve and retainer sleeve adapter with a shutdown seal sleeve and a shutdown seal sleeve adapter.

With one exception, all potentially affected UFSAR-described system, structure, or component design functions were screened-out. The RCP design function, to provide core cooling flow during normal operating conditions, was screened-in for further evaluation given it could be adversely impacted if the SDS were to inadvertently actuate.

The BVPS-2 updated safety analysis report was reviewed; changes were suggested. The BVPS-2 Technical Specifications and Bases were also reviewed; no changes were required. FirstEnergy Nuclear Operating Company (FENOC) concluded the proposed activity could proceed without obtaining a license amendment.

Summary of Evaluation:
The SDS was analyzed, evaluated and tested to the extent that installation and use of the SDS was acceptable.

NOTE: For the period of this report, only one of three RCPs were modified with the SDS. The remaining two RCPs will be modified during future outages.