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10 CFR 50.59(d)(2)

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

SUBJECT:
Davis-Besse Nuclear Power Station, Unit No. 1
Docket No. 50-346, License No. NPF-3
Report of Facility Changes, Tests, and Experiments

In accordance with 10 CFR 50.59(d)(2), the FirstEnergy Nuclear Operating Company hereby submits the Report of Facility Changes, Tests, and Experiments for the Davis-Besse Nuclear Power Station, Unit No. 1. The attached report covers the period of February 15, 2008 through June 16, 2010.

There are no regulatory commitments contained in this submittal. If there are any questions or if additional information is required, please contact Mr. Thomas A. Lentz, Manager – Fleet Licensing, at (330) 761-6071.

Sincerely,

Barry S. Allen

Barry S. Allen

Attachment:
Davis-Besse Nuclear Power Station, Unit No. 1 Report of Facility Changes, Tests, and Experiments

cc: Nuclear Regulator Commission (NRC) Region III Administrator
Nuclear Reactor Regulation Project Manager
NRC Resident Inspector
Utility Radiological Safety Board

IE47
NRR

Title

Thirty-six Inch Main Steam Line Break in Room 601 and 602

Activity Description

The proposed activity is the release of engineering calculation, "36-inch Main Steam Line Break in Rooms 601 and 602." This calculation uses the PCFLUD (Thermofluid Dynamics for a System of Interconnected Compartments) computer software to predict pressures and temperatures during a main steam line break (MSLB) in Auxiliary Building rooms 601 and 602. The proposed calculation replaces the previous analysis documented in "Evaluation of Environmental Conditions From Main Steam, Main Feedwater and Steam Generator Blowdown System Line Ruptures Outside Containment," Report No. 02-1040-1339, Impell Corporation, March 1986.

The proposed activity changes the computer code used to determine the mass and energy release during a main steam line break from RELAP-3 to RELAP5/MOD2-B&W. As determined by the 10 CFR 50.59 screen process, this is the only change associated with the proposed activity that may require an evaluation with respect to the potential for a license amendment.

Summary of Evaluation

The proposed activity incorporates one change that must be evaluated under 10 CFR 50.59: the computer code used to determine the mass and energy release during main steam line break in the Auxiliary Building was changed from RELAP-3 to RELAP5/MOD2-B&W.

The mass and energy release data utilized by the computer code was determined with MSLB-specific methods previously approved by the Nuclear Regulatory Commission. Therefore, the change in RELAP versions does not constitute a departure from a method of evaluation described in the Updated Final Safety Analysis Report.

Based on the discussion above, FirstEnergy Nuclear Operating Company (FENOC) concludes that the proposed change associated with the Main Steam Line Break accident analysis in the Auxiliary Building evaluated herein does not satisfy any of the criteria in paragraph (c)(2) of 10 CFR 50.59. Therefore, a license amendment is not required to implement the proposed activity.

Title

Allowable Service Water Flow Diversion During Cold Weather

Activity Description

The proposed activity is the release of a revision to engineering calculation, "Allowable Service Water Flow Diversion During Cold Weather," which evaluates the effect of removing orifice plates FE11210 and FE11211 with respect to service water system pressure control operation. The analysis shows that the position of the spare component cooling water (CCW) heat exchanger's (HX) isolation valve must be restricted when pressure control is implemented to ensure that service water pump runout conditions will not develop during a safety features actuation system (SFAS) actuation. The valve restriction must be implemented by operator action. This new operator action represents an adverse change with respect to how the design functions of the service water system are controlled. This new operator action is evaluated herein.

Addendum 1 to Revision 2 of the engineering calculation implements the requirement that a safety-related (that is, seismic) service water return line (for example, via SW2929 or SW2930) must be utilized during service water system pressure control operation. This new operator action also represents an adverse change with respect to how the design functions of the service water system are controlled and is evaluated herein.

Summary of Evaluation

FENOC removed the orifice plates located on the discharge of the CCW HXs during the 2006 refueling outage (14RFO). Because of this, when service water system pressure control is implemented, the position of the isolation valve of the spare CCW HX must be restricted. Also, to maintain the pressure control design basis analysis conditions, a safety-related (that is, seismic) service water return line (for example, via SW2929 or SW2930) must be utilized. These requirements are controlled by operator actions as directed by plant procedures. Although these operator actions are not part of any accident mitigation actions, they constitute an adverse effect with respect to how design functions are controlled. Based on the evaluation documented herein, the operator actions are not time critical and do not result in a more than minimal increase in the likelihood of a malfunction of systems, structures, components (SSC) important to safety. Thus, FENOC concludes that the proposed new operator actions do not meet the criteria of paragraph (c)(2) of 10 CFR 50.59. Therefore, a license amendment is not required.

Title

RV Head CRDM Nozzle Repair

Activity Description

Engineering Change Package (ECP) 10-0141 provides for the mitigation of Alloy 600/182/82 component/dissimilar metal welds (DMWs) within the reactor coolant system from cracking due to primary water stress corrosion cracking. As part of the mitigation process the control rod drive mechanisms (CRDM) nozzles DMWs will be repaired. The repair removes the flawed portion of the CRDM nozzle penetration to create a new welding surface. The weld location will extend into the CRDM penetration above the existing J-groove weld and will leave the reactor vessel closure head (RV/CH) low alloy steel base metal in the annular region around the nozzles exposed to the effects of borated water in the reactor coolant. Exposure of unclad low alloy steel reactor vessel base metal to borated water creates an adverse affect on the reactor coolant system (RCS) pressure boundary design function as described in the Updated Final Safety Analysis Report design. Because there are outstanding inspections and the potential for an increase in scope for ECP 10-0141, this evaluation was performed assuming all 69 RV/CH penetrations would be repaired.

Summary of Evaluation

FENOC proposes repairs under ECP 10-0141 to mitigate primary water stress corrosion cracking concerns in the Inconel Alloy 600 material and Inconel Alloy 82/182 weld filler material used in the CRDM nozzles. These repairs involve methods that will leave the current stainless steel clad carbon steel RV/CH base metal exposed to RCS borated water in the annular region of the nozzles. This will increase the general corrosion rate of the RV/CH base metal in these regions from zero inches per year to 0.0039 inches per year based on Areva Document No. 51-9134656, "Corrosion Evaluation of Davis Besse RV Head Penetration IDTB Weld Repair." This will result in a loss of 0.016 inches of the RV/CH carbon steel shell in the CRDM nozzle annular regions over the four-year life of the current RV/CH, which is scheduled for replacement during the 2014 refueling outage (18RFO). FENOC concludes that this adverse affect is acceptable based on Areva Document No. 51-9134656. These corrosion rates also assure that the remnants of the J-groove welds do not become detached and enter the RCS as loose parts, nor affect the operation of the attached components. This 10 CFR 50.59 evaluation concludes that a NRC approved license amendment is not required for implementation of the described repairs.