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June 27, 2012
L-12-194

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
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. The attached report covers the period of June 17, 2010 through June 14, 2012.

There are no regulatory commitments contained in this letter. If there are any questions or if additional information is required, please contact Mr. Phil H. Lashley, Supervisor – Fleet Licensing, at 330-315-6808.

Sincerely,

David Imlay for BS Allen

Barry S. Allen

Attachment:
Davis-Besse Nuclear Power Station Report of Facility Changes, Tests, and Experiments

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

Title

Replacement of the Unit Load Demand (ULD) Function of the Integrated Control System (ICS)

Activity Description

Engineering change package (ECP) 02-0540 replaced the non-safety related analog ULD subsystem of the ICS with a digital system. The existing load control panel human-machine interface was replaced with new touchscreens mounted in the control room. To facilitate installation of the new digital system, new electrical raceways and cables were installed in the control room, cabinet room, and computer room.

The ULD system is implemented using Invensys developed software hosted on Invensys hardware, mounted in a new cabinet, installed in the cabinet room. Inputs for the new system are obtained from non-safety related cabinets in the cabinet room via the new cables.

The replacement ULD system performs the same functions as the existing system. The ULD allows the operator to manually establish the power output of the station, and the ULD initiates load limiting and runback functions to restrict operation to within prescribed limits.

The replacement ULD system is equipped with a new algorithm that includes a core thermal power (CTP) calculator to determine the power level of the plant. The ULD CTP calculation result is compared to the operator commanded power level within the new Invensys software. A load demand signal is generated and provided to the ICS based on the comparison between calculated CTP and the operator commanded power level.

In addition to the ULD modification, ECP 02-0540 improved the physical installation of the data acquisition and analysis system (DAAS). This ECP moved the existing DAAS processing equipment located on a cart into new cabinet C5761G in the cabinet room. ECP 02-0540 did not make any functional changes to the DAAS system.

ECP 02-0540 was implemented because the performance of the original Bailey 820 ICS was beginning to degrade and the components of the original analog system are no longer supported by the equipment manufacturer. The system is scheduled for replacement in a two phase process. ECP 02-0540 provided the design for the first phase, which was limited to the replacement of the ULD subsystem of the ICS. The

replacement of the remainder of the ICS system will be performed in Phase II under a different ECP, which is not included in the scope of this evaluation.

As the new system is of digital design, the system was evaluated for cyber security in accordance with industry standard practices documented in Nuclear Energy Institute (NEI) 04-04, *Cyber Security Program for Power Reactors*, Revision 1. The system was evaluated, and installed, in compliance with Level 4 security requirements.

Summary of Evaluation

The replacement ULD subsystem of the ICS performs the same design functions as the previous analog system. The replacement digital system has been designed to be reliable, fault tolerant, and suitable for the environments in which it will be installed. The software utilized by the digital ULD system has been developed under the Invensys Software Quality Assurance Plan, and extensively tested to ensure it meets design requirements.

The replacement ULD is not an initiator of an accident. The replacement ULD system does not directly control systems, structures, and components (SSCs) that perform safety functions. The ULD interface with SSCs that perform safety functions has been designed to ensure that no ULD failure modes can adversely affect the interfaced SSC. Therefore, the replacement ULD does not increase the frequency of occurrence of a previously evaluated accident and does not increase the likelihood of occurrence of previously evaluated malfunctions of an SSC important to safety.

The ULD does not directly control any SSC that mitigates the consequences of an accident. In the event of a failure of the ULD, the SSCs that act to mitigate an accident are unaffected by the ULD failure. The failure modes of the revised ULD system have been analyzed in the ULD failure modes and effects analysis, and there are no new failure effects that would initiate an accident or create a new malfunction of equipment.

The modification to the ULD does not require a departure from a method of evaluation described in the Updated Safety Analysis Report (USAR), and the ULD does not interact, directly or indirectly, with a fission product barrier.

In conclusion, the proposed activity does not meet the criteria in paragraph (c)(2) of 10 CFR 50.59 and, therefore, a license amendment was not required.

Title

Containment Vessel Opening

Activity Description

ECP 10-0459 addressed engineering activities related to cutting and restoration of the containment vessel wall during the mid-cycle outage. The opening was needed to provide a pathway for reactor vessel head replacement. The ECP activities included welding attachments to the containment vessel wall.

Summary of Evaluation

The methodology used for the evaluation of the restored containment vessel with welded attachments involves use of the ANSYS computer code. Other methods were described in the USAR for the evaluation of the containment vessel in its original configuration; thus, the activity involves a change in methods for the design bases and the activity was screened as a methodology change. However, the use of the ANSYS computer code for this purpose does not involve a departure from the method of evaluation described in the USAR, because the planned use of the ANSYS computer code is considered approved by the Nuclear Regulatory Commission (NRC) for the intended application [NUREG-1793, Volume 1, Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design, September 2004 (Accession No. ML043450344) and NUREG-1793, Supplement 1, Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design, December 2005 (Accession No. ML0603305570)]. The NRC has approved the use of the ANSYS computer code for the type of analysis planned for the evaluation of the containment vessel, and FirstEnergy Nuclear Operating Company (FENOC) satisfies the applicable terms, conditions, and limitations for its use. The proposed activity does not meet the criteria in paragraph (c)(2) of 10 CFR 50.59 and, therefore, a license amendment was not required.

Title

Control Room Dose Due to the Fuel Handling Accident Inside Containment

Activity Description

The proposed activity was the release of Revision 2 to engineering calculation, *Control Room Dose Due to the Fuel Handling Accident Inside Containment.* This calculation determines the dose to the operators in the control room during a fuel handling accident in the containment vessel. Revision 2 reduces the control room net free volume. As a result of the control room volume reduction, the unfiltered inleakage into the control room has been reduced by a small amount. These changes produce an increase in the predicted dose received by the operators during the accident. Revision 2 was issued as a result of the change to the control room net free volume determined by engineering calculation, *Free Volume of the Control Room*, Revision 1, Addendum 1.

Summary of Evaluation

The proposed activity changes two inputs to the control room dose analysis during a fuel handling accident inside of the containment vessel: (1) control room net free volume, and (2) control room unfiltered inleakage. These changes cause the dose received by the control room operators to increase. This increase is less than 10 percent of the differences between the regulatory limits and the current values. Therefore, as defined by NEI 96-07, *Guidelines for 10 CFR 50.59 Implementation*, Revision 1, the proposed change is less than a minimal increase in the radiological consequences of an accident previously evaluated in the USAR.

Based on the evaluation, the changes associated with the revised fuel handling accident (inside of the containment vessel) analysis do not meet the criteria of paragraph (c)(2) of 10 CFR 50.59. Therefore, a license amendment was not required.

Title

Davis-Besse Spent Fuel Pool Criticality Analysis Revised Code Bias Uncertainty

Activity Description

NRC Information Notice (IN) 2011-03, *Non-Conservative Criticality Safety Analyses for Fuel Storage*, identified a non-conservatism in the methodology previously approved by the NRC to determine the Monte Carlo code bias uncertainty used in spent fuel pool criticality analyses. The Davis-Besse spent fuel pool criticality analysis utilizes the method the NRC identifies as being non-conservative. Engineering calculation C-NF-062.02-027, *Davis-Besse Spent Fuel Pool Criticality Analysis Revised Code Bias Uncertainty*, supplements the spent fuel pool criticality analysis of record using an alternate method to determine the bias uncertainty. This engineering calculation also provides the basis for revising the Davis-Besse Technical Specification Bases Section 3.7.15 and USAR Section 9.1.2.1 to reference the regulatory-required acceptance criteria for the criticality analyses, rather than a more restrictive value based on analyses that used an incorrect bias uncertainty. Therefore, this evaluation addresses the Technical Specification Bases and USAR changes, as well as the engineering calculation.

Summary of Evaluation

NRC IN 2011-03 identified a non-conservatism in the methodology previously approved by the NRC to determine the Monte Carlo code benchmarking bias uncertainty used in spent fuel pool criticality analyses. The Davis-Besse spent fuel pool criticality analysis utilizes the method the NRC identifies as being non-conservative. Engineering calculation C-NF-062.02-027 supplements the spent fuel pool criticality analysis of record using an alternate method to determine the bias uncertainty. Use of the alternate method resulted in k-effective values that are closer to, but remain bounded by, the 10 CFR 50.68(b)(4) limit of a k-effective of 0.95.

The use of the alternate method to determine the bias uncertainty was determined to be a change to an element of a methodology. The result of implementing the alternate method is conservative and remains bounded by the regulatory limit of 0.95 set forth in 10 CFR 50.68; therefore, it is not a departure from a method described in the USAR, and NRC approval prior to use is not required. Therefore, a license amendment was not required.

Title

Control Room Radiation Doses Following a Maximum Hypothetical Accident

Activity Description

The proposed activity was the release of Revision 3 to engineering calculation, *Control Room Radiation Doses Following a Maximum Hypothetical Accident*. This engineering calculation determines the dose to the operators in the control room during a maximum hypothetical accident. Revision 3 changes the control room net free volume that is an input to the analysis. As a result of the control room volume change, the dose received by the operators in the control room increases by a small amount. Revision 3 was issued as a result of the change to the control room net free volume determined by engineering calculation, *Free Volume of the Control Room*, Revision 1, Addendum 1.

Summary of Evaluation

The proposed activity changes one input to the design analysis of the maximum hypothetical accident; that is, control room net free volume. This change causes a small increase to the dose received by the control room operators. This increase is less than 10 percent of the differences between the regulatory limits and the current values. Therefore, as defined by NEI 96-07, Revision 1, the proposed change is less than a minimal increase in the radiological consequences of an accident previously evaluated in the USAR.

Based on the evaluation, the change associated with the revised maximum hypothetical accident analysis does not meet the criteria of paragraph (c)(2) of 10 CFR 50.59. Therefore, a license amendment was not required.

Title

Control Room, Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) Doses Due to Waste Gas Decay Tank Rupture

Activity Description

The proposed activity was the release of Revision 0 to engineering calculation, *Control Room, EAB and LPZ Doses Due to Waste Gas Decay Tank Rupture*. This engineering calculation performs an analysis of the dose received during a waste gas decay tank rupture: (1) in the control room, (2) at the EAB, and (3) in the LPZ. For dose analysis purposes, the control room analysis was split into three parts: (1) a general update, (2) an update to the control room volume, and (3) the analysis performed with the control room emergency ventilation system (CREVS) operating in the pressurization mode. Since the previous version of the analysis was performed in 1973 (with CREVS operating in the recirculation mode), an update was necessary to revise the analysis to meet current standards. After the update was completed, an update to the control room dose was implemented by changing the control room net free volume. This change was necessitated by a change to the control room volume determined by engineering calculation, *Free Volume of the Control Room*, Revision 1, Addendum 1. An analysis with CREVS operating in the pressurization mode was then performed to provide the flexibility of allowing CREVS to be operated in any mode.

A general update to the dose analysis for the EAB and LPZ during a waste gas decay tank rupture was performed to revise the analysis to meet current standards.

Summary of Evaluation

The general update to the waste gas decay tank rupture accident resulted in dose changes (thyroid, beta-skin and whole-body) for the control room, EAB, and LPZ that are less than 10 percent of the difference between the current USAR reported doses and the dose limits.

Once the general update was completed, a change to the control room net free volume and operation of CREVS in the pressurization mode were evaluated. Each of these changes resulted in an increase to the control room doses. The combination of the changes resulted in dose increases that are less than 10 percent of the difference between the doses determined by the general update and the dose limits.

The proposed activity changed two analysis inputs that are considered to be changes to elements of methodology as follows: (1) waste gas decay tank source term, and (2) dose conversion factors. The change to the source term caused an increase to the dose. The changes to the dose conversion factors are based on regulatory guides.

Based on the evaluation, the changes associated with the revised waste gas decay tank rupture accident analysis do not meet the criteria of paragraph (c)(2) of 10 CFR 50.59. Therefore, a license amendment was not required.

Title

New Reactor Vessel (RV) Service Structure

Activity Description

ECP 10-0470 installed a new integrated head assembly (IHA) on the replacement reactor vessel closure head (RRVCH) to replace the existing combination control rod drive mechanism (CRDM) service structure and service structure support skirt (together referred to as SS). The IHA incorporates design enhancements that will eliminate, or reduce the time required for some of the disassembly and reassembly activities that are performed during outages for the existing SS. This improvement in outage activity performance will reduce both the time and the personnel radiation exposures associated with those activities. The design enhancements will also improve worker safety during several outage activities.

The scope of ECP 10-0470 included changing the USAR, changing engineering calculations, and the following activities:

- Installation on the RRVCH (implemented by ECP 10-0469) of an IHA that includes a new CRDM seismic clamp design, support for the reactor vessel closure head (RVCH) fixed lifting pendants, integral radiation shielding, a monorail for support of stud tensioners, a support rod for installation of temporary shielding, and new retractable support platforms (batwings) for CRDM power and position indication (PI) and temperature monitoring cables

- Replacement and re-routing of a portion of the existing RVCH continuous vent line (CVL) and replacement of the CVL metal reflective insulation (MRI)

- Replacement of the two existing, remote CRDM cooling fans and associated ductwork with six new fans that are integral to the IHA shroud

- Addition of an integrated service air manifold

- Replacement of the existing RVCH insulation with new MRI

- Replacement of the component cooling water (CCW) return and supply manifolds with redesigned manifolds on the IHA

- Replacement and re-routing of a portion of the plant-side CCW piping to interface with the new IHA CCW piping

- Replacement of the existing cables and connection hardware for CRDM power, PI, and temperature monitoring, and IHA air and CVL temperature monitoring (the replacement of the CRDMs was implemented by ECP 10-0469)

- Replacement of the existing plant-side disconnect panel for IHA air temperature monitoring and stud hoist power with two new panels at new locations to accommodate the CCW piping modification

- Replacement of the upper clevis pins and lower clevis shoulder bolts in the existing lifting fixture with new upper and lower clevis shoulder bolts

As a result of some of the changes implemented by ECP 10-0470, new structural analyses had to be performed for the CRDMs, IHA, RV, and RV internals. The analytical methodology used for the new analyses resulted in a conclusion that the proposed activity does involve revising or replacing a USAR-described evaluation methodology used in establishing the design bases or in the safety analyses.

Summary of Evaluation

The STALUM computer code was used to perform loading analyses that are presented in the Babcock & Wilcox Owners Group Topical Report BAW-1621, *Effects of Asymmetric LOCA [Loss of Coolant Accident] Loadings*, and that are currently described in the USAR. The BWSPAN computer code was used to perform new loading analyses, required by installation of the IHA, of some of the same components that were analyzed in BAW-1621. The change associated with using BWSPAN for these new analyses involved updating the mathematical models of the components so that the models would execute within the newer BWSPAN computer code. The way in which the loading calculations are performed by BWSPAN is unchanged from how the loading calculations were performed by STALUM.

BWSPAN is the computer code currently used by AREVA NP, Inc. (AREVA) for performing loading analyses, and it is the evolutionary replacement for the previously used STALUM computer code, which is now archived and no longer in use. BWSPAN incorporates enhancements in usability and features, but its method of performing loading calculations remains the same as was used in STALUM. BWSPAN has been verified and benchmarked against hand calculations, predecessor codes (including STALUM), NRC test cases, and other commercial codes in accordance with AREVA procedures. The results of this benchmarking show that BWSPAN, using the updated mathematical models, produces results that are essentially the same as the results from STALUM. Therefore, the use of updated mathematical models in the BWSPAN computer code, in place of using the original mathematical models in the STALUM computer code, does not constitute a departure from a method of evaluation described in the USAR used in establishing the design bases or in any safety analyses. Therefore, a license amendment was not required.