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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 Title 10 of the *Code of Federal Regulations*, Part 50, Section 50.59(d)(2), FirstEnergy Nuclear Operating Company (FENOC) hereby submits the Report of Facility Changes, Tests, and Experiments for the Davis-Besse Nuclear Power Station, Unit No. 1 for the period ending May 26, 2014.

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

Sincerely,

Raymond A. Lieb

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

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

Auxiliary Building HELB [High Energy Line Break] Pressure Analysis Using GOTHIC 7.0	
Evaluation Number: 11-02793	Revision 00
Reference Document: Calculation	C-NSA-000.02-012 Rev.01, A04

Activity Description/Change:

This calculation determines the pressure and vapor (air and steam) temperature response of the auxiliary building during a main feedwater line break accident. The objective of the analysis is only to predict the pressures and temperatures.

This change is the release of Addendum 4 to engineering calculation, C-NSA-000.02-012, R01, "Auxiliary Building HELB Pressure Analysis Using GOTHIC 7.0." This calculation uses the GOTHIC computer software to predict pressures and temperatures during various pipe breaks in the auxiliary building. Addendum 4 updates the main feedwater break analysis. This update changes the computer program used to determine the mass and energy release during a main feedwater 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 requires an evaluation with respect to the potential for a license amendment.

Summary of Evaluation:

This calculation incorporates one change that requires evaluation with respect to 10 CFR 50.59 criteria, which is the computer code used to determine the mass and energy release during a main feedwater line break in the auxiliary building that was changed from RELAP-3 to RELAP5/MOD2-B&W.

The mass and energy release data utilized by the change was determined with methods in previously approved Nuclear Regulatory Commission (NRC) topical reports and are incorporated by reference in Section 1.5 of the Davis-Besse Nuclear Power Station, Unit No. 1 (DBNPS) Updated Safety Analysis Report (USAR); therefore the change in RELAP versions does not constitute a departure from a method of evaluation described in the USAR.

Auxiliary Building HELB [High Energy Line Break] Pressure Analysis Using GOTHIC 7.0 (continued)
Based on the evaluation associated with the main feedwater line break accident analysis in the auxiliary building as compared to 10 CFR 50.59(c)(2) criteria, a license amendment is not required to implement this activity.

Main Feedwater Line Breaks and Cracks in the Auxiliary Building	
Evaluation Number: 11-04214	Revision 00
Reference Document: Calculation	C-NSA-000.02-005 Rev.02, A05

Activity Description/Change:

This calculation determines the pressure and vapor (air and steam) temperature response of auxiliary building during a main feedwater line break accident. The objective of the break analysis is only to predict the room temperatures.

This change is the release of Addendum 05 to engineering calculation, C-NSA-000.02-005, Revision 02, "Main Feedwater Line Breaks and Cracks in the Auxiliary Building." This calculation uses the PCFLUD computer software to predict temperatures due to various main feedwater pipe breaks in the auxiliary building. Addendum 05 updates the main feedwater line break analysis. This update changes the computer program used to determine the mass and energy release during a main feedwater 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 requires an evaluation with respect to the potential for a license amendment.

Summary of Evaluation:

This calculation incorporates one change that requires evaluation with respect to 10 CFR 50.59 criteria, which is the computer code used to determine the mass and energy release during a main feedwater line break in the auxiliary building that was changed from RELAP-3 to RELAP5/MOD2-B&W.

The mass and energy release data utilized by the change was determined with methods in previously approved NRC topical reports and are incorporated by reference in Section 1.5 of the DBNPS USAR; therefore, the change in RELAP versions does not constitute a departure from a method of evaluation described in the USAR.

Based on the evaluation associated with the main feedwater line break accident analysis in the auxiliary building as compared to 10 CFR 50.59(c)(2) criteria, a license amendment is not required to implement this activity.

Replacement of Electro-Hydraulic Control (EHC) and Turbine Supervisory Instrumentation (TSI) Systems	
Evaluation: 13-00037	Revision: 00
Reference Document: Engineering Change Package	05-0140, Revision 3

Activity Description/Change:

This change involves the main turbine electro-hydraulic control (EHC) system that was replaced with a GE Mark VIe Digital EHC (DEHC) system. The EHC system has been showing signs of degradation and the equipment technology has become obsolete. Replacement parts and technical support are no longer available.

The obsolete turbine supervisory instrumentation (TSI) system was replaced with a Bently Nevada BN 3500 TSI system.

The new DEHC system is functionally similar to the original system. The control and stop valve instrumentation and control devices associated with the turbine speed control and trip functions was replaced with new components to increase redundancy. The original EHC controls were not redundant. The new system incorporates dual or triple modular redundancy (TMR) for nearly all control and trip functions and for instruments used for control. This additional redundancy will reduce the failure to function potential and reduce the potential for false protective function actuation. The existing normal dual electronic speed channels, primary mechanical overspeed trip device, and backup electronic overspeed trip signal was replaced with TMR control for normal speed control, and primary and emergency TMR electronic controls for overspeed protection. As a result, the DBNPS USAR Section 7.7.1.4.2 was revised.

Summary of Evaluation:

The 10 CFR 50.59 screening process determined an evaluation was required for the following reasons:

1. The Mark VIe system removes the mechanical overspeed trip device. The new system uses two electronic overspeed trip systems. When compared to the existing system the change has both positive and negative impacts on the overspeed protection design function.

2. The change from analog to digital controls for the EHC system fundamentally alters the method that the design functions are implemented for protective trip, overspeed trip, and turbine stop valve closure.

Replacement of Electro-Hydraulic Control (EHC) and Turbine Supervisory Instrumentation (TSI) Systems (continued)

The 10 CFR 50.59 screen thoroughly documented the calculations, studies, and analysis of environmental issues such as electromagnetic interface, radio-frequency interference, seismic loading, control room heating, ventilation, and air condition (HVAC), and power loads and found that these issues do not require a 10 CFR 50.59 evaluation. The 10 CFR 50.59 screen found that the TSI system provides monitoring functions only, and the replacement system does not require a 10 CFR 50.59 evaluation. The screen also found that there were no method changes requiring evaluation under 10 CFR 50.59.

No increase in any accident frequency or malfunction likelihood was found based on several General Electric prepared reliability related studies and failure analyses, the TMR design, and the overspeed protection design.

Reliability reports included a reliability study, turbine missile generation analyses, a failure modes and effects analysis, and a single point failure analysis.

Common mode failure risks were considered from several perspectives for the DBNPS DEHC retrofit. General Electric reports summarized in this evaluation reveal extensive and systematic analysis of potential common mode failures and the risk with the DEHC hardware and software. The DEHC limits project specific software common mode failure risk by only using standard Mark VIe logic blocks and by minimizing variations from the GE standard block configuration. No custom software and no custom firmware was needed for the DBNPS DEHC.

The software common mode failure risk with the DEHC is very small because the most credible failure mode is limited to failures that would cause a spurious turbine trip. The likelihood of such a common mode failure is very low due to the quality assurance program that includes extensive design controls and testing.

Like the existing Mark I system, the DEHC system has a normal speed control function, and two overspeed detection and trip systems. The DEHC reduces the dependence on mechanical components to achieve overspeed protection. The overspeed protection design uses the GE Mark VIe system that is very similar to the system reviewed and approved by the NRC for the economic simplified boiling water reactor (ESBWR) in 2010. A key consideration was the adequacy of the Mark VIe approach to overspeed protection relative to diversity. The NRC Safety Evaluation Report "Economic Simplified Boiling Water Reactor Standard Design Final Design Approval", (Agencywide Documents Access and Management System Accession No. ML110540310) concluded that the diversity was adequate.

Replacement of Electro-Hydraulic Control (EHC) and Turbine Supervisory Instrumentation (TSI) Systems (continued)

Accident and malfunction consequences will remain unchanged by this change. This conclusion results from maintaining the turbine stop valve closure time at 1.0 seconds total time from steam and feedwater rupture control system (SFRCS) signal receipt at the EHC to valve closure. Maintaining this closure time also avoids altering a design basis limit for a fission product barrier.

Potential different accidents and malfunctions with different results are bounded by the existing USAR-described accidents and malfunction results. The common mode failure analyses and various reports and studies described in the analyses supported this conclusion.

Based on the evaluation as summarized above that included answering the NEI 01-01, "Guideline on Licensing Digital Upgrades, EPRI TR-102348," Revision 1, Attachment A supplemental questions for addressing 10 CFR 50.59 evaluations, FENOC concluded that ECP 05-0140 for the replacement of the EHC and TSI systems may be implemented without amending the DBNPS license.

Shield Building Design Calculation	
Evaluation: 13-00918	Revision: 01
Reference Document: Calculation	C-CSS-099.20-063, Revision 0

Activity Description/Change:

Calculation C-CSS-099.20-063, "Shield Building Design Calculation" provides the new design evaluation of the shield building (SB), including the effects of the laminar cracking, for the cylindrical shell wall, dome, and spring line areas of the building. The calculation includes the results of lab testing performed to determine the effect of laminar cracking on the structural behavior and strength of the structure. The calculation includes a change in the computer code. The new analysis of the SB was performed using a three-dimensional finite element model in ANSYS, which is a quality controlled computer program verified and validated for use in nuclear safety-related applications in accordance with the project procedures.

Revision 1
The scope of Revision 1 of the 50.59 Evaluation consists of correcting the record date and number for Bechtel Report 25593-000-G83-GEG-00016-000, "Effect of Laminar Cracks on Splice Capacity of No. 11 Bars Based on Testing Conducted at Purdue University and University of Kansas for Davis-Besse Shield Building," July 30, 2012.

Summary of Evaluation:

The SB with the extent of laminar cracking documented in a FENOC report behaves within the elastic range and is capable of withstanding the design basis loads and load combinations described in USAR Subsection 3.8.2.2. The compliance of the SB design to the applicable provisions of American Concrete Institute (ACI) 307-69 "Specification for the Design and Construction of Reinforced Concrete Chimneys" and ACI 318-63, "ACI Standard Building Code Requirements for Reinforced Concrete" is documented in the new design evaluation. The design evaluation demonstrates that the SB maintains its structural integrity, meets the requirements of the design codes, and can perform its intended design functions.

The methodology used for the new evaluation of the shield building involves use of the ANSYS computer code. Existing methods are described in USAR subsection 3.8.2.2. The use of ANSYS for this application was concluded a methodology change for the 10 CFR 50.59 screen. It does not involve a departure from the method of evaluation described in the USAR because the planned use of ANSYS is considered approved by the NRC for the intended application. The NRC has approved the use of ANSYS for the type of analysis planned for the evaluation of the shield building, and FENOC satisfies the applicable terms, conditions, and limitations for its use. Based on the evaluation of the shield building design calculation compared to 10 CFR 50.59(c)(2) criteria, a license amendment is not required to implement the activity.

Containment Vessel Analysis	
Evaluation: 13-04372	Revision: 00
Reference Document: Calculation	C-NSA-060.05-010, R07

Activity Description/Change:

This change involved a reanalysis of the containment vessel's response during a main steam line break (MSLB) accident. The analysis is documented in engineering calculation number, C-NSA-060.05-010, Revision 7, "Containment Vessel Analysis." The reanalysis was performed to address a condition report that identified deficiencies in the mass and energy release predictions that are used as input to the containment vessel response analysis during a MSLB. The corrected mass and energy release data (that is, steam flow rates and enthalpies) are higher than the previous data. No other changes are being implemented.

Summary of Evaluation:

The containment vessel's response during a main steam line break is performed by analyzing the double-ended guillotine rupture of a 36-inch diameter main steam line inside of the containment vessel. Mass and energy release rates from the break are determined by the fuel vendor with the RELAP5/MOD2-B&W (RELAP) code. As a result of incorporating error corrections pertaining to specific DBNPS plant inputs into the RELAP model, the predictions show that reactor shutdown will occur due to a reactor protection system (RPS) high flux trip rather than an RPS low reactor coolant system (RCS) pressure trip. This results in a longer time period between accident initiation and reactor trip. This is conservative with respect to mass and energy release calculations because the longer time to reactor trip increases the amount of energy transferred from the RCS to the secondary side. The DBNPS USAR sections 3.11.2.1, 6.2.1.3.2, Table 6.2-10, and Table 6.2-11 were revised accordingly.

The response of the containment vessel using the revised mass and energy release data was predicted with the COPATTA code. The revised mass and energy released data produce an increase in the containment's pressures and vapor temperatures. The containment vessel's metal temperature is also predicted to increase and revised equipment temperature profiles were predicted.

The predicted maximum containment pressure and metal temperature remain less than the respective acceptance criterion. The equipment temperature profiles have been verified to be acceptable with respect to environmental qualification criteria.

Containment Vessel Analysis (continued)
Evaluation: 13-04372

Due to an increase to the maximum containment pressure during the transient, actuation of the containment spray pumps are predicted. Previous MSLB analyses show that containment spray would not be actuated. There is at least a 10 second margin between the high pressure injections pumps reaching full electrical load and the startup of the containment spray pumps. The assumptions pertaining to pump startup sequence; therefore, remain valid and the undervoltage relays and offsite power sources will perform their design functions.

Based on the evaluation of the changes associated with the containment vessel's response during a main steam line break accident analysis as compared to the 10 CFR 50.59(c)(2) criteria, a license amendment is not required to implement this activity.

Containment Vessel Analysis	
Evaluation: 13-04680	Revision: 00
Reference Document: Calculation	C-NSA-060.05-010, R08

Activity Description/Change:

This change involved a reanalysis of the containment vessel's response during a main steam line break (MSLB) accident. The analysis is documented in engineering calculation number, C-NSA-060.05-010, Revision 8, "Containment Vessel Analysis." The reanalysis was performed to incorporate revised mass and energy release data for the replacement once-through steam generators (ROTSG).

Summary of Evaluation:

The containment vessel's response during a main steam line break is performed by analyzing the double-ended guillotine rupture of a 36-inch diameter main steam line inside of the containment vessel. Mass and energy release rates from the break are determined by the fuel vendor with the RELAP5/MOD2-B&W (RELAP) code as described in the USAR. As a result of the ROTSGs, revised mass and energy release data for a MSLB were generated. The predictions show that reactor shutdown will occur due to a reactor protection system low reactor coolant system pressure trip rather than a high flux trip. There is no specific acceptance criterion related to the type of reactor trip during a MSLB.

The response of the containment vessel using the revised mass and energy release data was predicted with the COPATTA code. The revised mass and energy released data produce an increase in the containment's pressures and vapor temperatures. The containment vessel's metal temperature is also predicted to increase and revised environmental qualification temperature profiles were predicted. The DBNPS USAR sections 3.11.2.1, 6.2.1.3.2, Table 6.2-10, and Table 6.2-11 were revised accordingly.

The predicted maximum containment pressure and metal temperature remain less than the respective acceptance criterion. The equipment temperature profiles have been verified to be acceptable with respect to environmental qualification criteria. There is no specific acceptance criterion directly related to the vapor temperature during a MSLB.

Based on the evaluation of the changes associated with the containment vessel's response during a main steam line break accident analysis as compared to the 10 CFR 50.59(c)(2) criteria, a license amendment is not required to implement the activity.

Thirty-six Inch Main Steam Line Break in Rooms 601 and 602	
Evaluation: 13-04729	Revision: 00
Reference Document: Calculation	C-NSA-000.02-016, R00

Activity Description/Change:

This change is the release of Addendum 1 to engineering calculation, C-NSA-000.02-016, R00, "36 Inch Main Steam Line Break in Rooms 601 and 602." Addendum 1 incorporates revised mass and energy release data for the main steam line break (MSLB) in the 600 series rooms of the auxiliary building. The mass and energy release data was revised because of errors that were discovered during replacement steam generator work as documented by condition report 2013-16627. The DBNPS UFSAR Section 3.6.2.7.1.3 was revise accordingly.

As a result of the revised mass and energy release data, two changes prompted further evaluation as indicated by the 10 CFR 50.59 screen process as follows: (1) the revised mass and energy release prediction indicate that the reactor shutdown will be due to a reactor protection system (RPS) high flux trip rather than an RPS low reactor coolant system (RCS) pressure trip and, (2) the revised mass and energy release data are based on predictions that credit actuation of the steam and feedwater rupture control system (SFRCS) with the pressure switches located on the main steam piping of the affected generator. The analysis-of-record credits actuation of the SFRCS pressure switches that are located on the main steam piping of the unaffected generator.

Summary of Evaluation:

The response during a main steam line break is performed by analyzing the double-ended guillotine rupture of a 36-inch diameter main steam line inside of the auxiliary building's 600 series rooms. Mass and energy release rates from the break are determined by the fuel vendor with the RELAP5/MOD2-B&W (RELAP) code. As a result of correcting several errors, revised mass and energy release data for a MSLB were generated. The predictions show that reactor shutdown will occur due to a RPS high flux trip rather than a RPS low pressure trip. Using the revised mass and energy release data, the temperature response of the auxiliary building's 600 series rooms remains acceptable with respect to environmental qualification of equipment, and the auxiliary building walls will perform its design function because room peak pressures are within limits. Thus, all results associated with a MSLB in the auxiliary building remain acceptable when using an RPS high flux trip rather than an RPS low RCS pressure trip.

Thirty-six Inch Main Steam Line Break in Rooms 601 and 602 (continued)

The revised mass and energy release data are based on predictions that credit actuation of the SFRCS with the pressure switches located on the main steam piping of the affected generator. There are no technical concerns (that is, environmental qualification) with crediting the SFRCS pressure switches located on the main steam piping of the affected steam generator. Review of related licensing documents indicates that crediting the SFRCS pressure switches on the affected steam generator does not involve any previously specified inputs by the NRC for the purpose of performing a conservative analysis.

Based on the evaluation of the main steam line break accident analysis as compared to the 10 CFR 50.59(c)(2) criteria, a license amendment is not required to implement the activity.