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Docket Number 50-346 10CFR50.59(d)(2)

License Number NPF-3

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United States Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-0001

Subject: 10 CFR 50.59 Summary Report of Facility Changes, Tests and Experiments

Ladies and Gentlemen:

The FirstEnergy Nuclear Operating Company (FENOC) hereby submits, pursuant to 10 CFR 50.59 (d)(2), the attached 10 CFR 50.59 Summary Report of Facility Changes, Tests and Experiments for the Davis-Besse Nuclear Power Station, Unit No. 1 (DBNPS). This report covers the period from March 27, 2004 through May 18, 2006. The last report was submitted on June 23, 2004.

Attachment 1 defines the acronyms and format description. Attachment 2 provides summaries of the 10 CFR 50.59 evaluations. Attachment 3, Commitment List, identifies that there are no commitments contained in this letter. If there are any questions or if additional information is required, please contact Mr. Gregory A. Dunn, Manager – FENOC Fleet Licensing, at (330) 315-7243.

Very truly yours,

Mark B. Bezilla

MSH

IE47

Attachments: 1. Summary
2. 10 CFR 50.59 Evaluation Summaries
3. Commitment List

cc: Regional Administrator, NRC Region III
DB-1 NRC/NRR Project Manager
DB-1 NRC Senior Resident Inspector
Utility Radiological Safety Board

Summary

This report is submitted in accordance with 10 CFR 50.59 (d) (2), and provides a brief description of the changes, tests, and experiments performed at the DBNPS and evaluated pursuant to 10 CFR 50.59. This report covers the period of March 27, 2004 through May 18, 2006.

This report provides a listing of abbreviations used in the evaluation summaries, a summary listing of the evaluations contained in the report, and summaries of the evaluations performed pursuant to 10 CFR 50.59. Each of these evaluations concluded that the change did not require a license amendment.

This report includes several references to Nuclear Energy Institute (NEI) 96-07, *Guidelines for 10 CFR 50.59 Evaluations.* Revision 1 of NEI 96-07 is endorsed by Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments."

Acronyms:

CAC	=	Containment Air Cooler
CCW	=	Component Cooling Water
CFR	=	Code of Federal Regulations
DBNPS	=	Davis-Besse Nuclear Power Station
DNB	=	Departure from Nucleate Boiling
EPROM	=	Erasable, Programmable, Read-Only Memory
EQ	=	Environmental Qualification
EVS	=	Emergency Ventilation System
FENOC	=	FirstEnergy Nuclear Operating Company
HELB	=	High Energy Line Break
LOCA	=	Loss of Coolant Accident
LOFW	=	Loss of Feedwater
MSLB	=	Main Steam Line Break
NEI	=	Nuclear Energy Institute
NRC	=	Nuclear Regulatory Commission
RCS	=	Reactor Coolant System
SER	=	Safety Evaluation Report
USAR	=	Updated (Final) Safety Analysis Report

Docket Number 50-346
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Attachment 2

10 CFR 50.59 Summary Report
of
Facility Changes, Tests, and Experiments
for
Davis-Besse Nuclear Power Station, Unit No. 1

March 27, 2004 – May 18, 2006

(10 pages follow)

Tornado Sub-compartment Pressurization Methodology Change
Permitting COMPARE Methodology
USAR Change Notice 03-058
10 CFR 50.59 Evaluation 04-02210, Revision 2

ACTIVITY DESCRIPTION

This activity changes USAR Section 3.3.2.2.1, "Differential Pressure Drop in Auxiliary Building," to provide the COMPARE-computer code as an alternate method for determining tornado induced differential pressures between Auxiliary building sub-compartments.

SUMMARY OF EVALUATION

This activity provides an alternative method for determining tornado induced differential pressures between building sub-compartments, and was identified as a change to a method of evaluation.

The change impact was evaluated for new methodology, regulatory requirements, and licensee qualification for performing safety analyses. The evaluation demonstrates that use of the COMPARE methodology is not a departure from a method of evaluation described in the USAR.

This activity complies with NEI 96-07, Section 4.3.8, as it was demonstrated that the COMPARE methodology is based on sound engineering practice, providing sound engineering formulae and technical application, and compliance with relevant industry standards, and is appropriate for the intended application, as the COMPARE computer code is a methodology developed by Los Alamos Laboratory under contract for the NRC to perform sub-compartment analysis.

The evaluation also demonstrates that Comanche Peak's application of COMPARE is the same as the intended application for the DBNPS. The difference between Comanche Peak and the DBNPS is the initiation humidity, and this difference does not affect the methodology. Analyses are performed in accordance with the guidelines contained in Generic Letter 83-11, Supplement 1, Licensee Qualification for Performing Safety Analyses.

It was concluded that the COMPARE methodology is within the limitations of the reference NRC Safety Evaluation Report for Comanche Peak, and therefore acceptable for use at the DBNPS.

Changes to Containment Vessel Analysis Inputs
Calculation C-NSA-060.05-010
10 CFR 50.59 Evaluation 04-03928

ACTIVITY DESCRIPTION

A reanalysis of the Containment Vessel's pressure and temperature response during a large break LOCA and Main Steam Line Break was performed. In general, the reanalysis was performed to address several Engineering Change Requests, and items entered into the FENOC Corrective Action Program, and to update input parameters based on revised input calculations. The most significant change was to reduce the Service Water flowrate to the Component Cooling Water (CCW) heat exchanger from 8000 to 7500 gallons per minute. This was done to accommodate increased measurement uncertainties associated with Service Water system flow balancing. Other changes included

- Changed CCW flowrate to the Emergency Diesel Generator jacket cooling water heat exchangers from 1100 to 900 gallons per minute,
- Changed the Low Pressure Injection flowrate during recirculation from 2700 to 3200 gallons per minute,
- Incorporated the heat duty table for the new Containment Air Cooler coils, and
- Analyzed the Ultimate Heat Sink response with the VPLUG computer code.

SUMMARY OF EVALUATION

LOCA Analysis Method

No changes to analysis techniques were implemented for the latest Containment Vessel accident analysis. As was done for the previous revision, the COPATTA computer code was utilized to compute the post-accident Containment Vessel response. Mass and energy release rates from the break were determined with the RELAP5/MOD2-B&W computer code. Also, the response of the Ultimate Heat Sink was determined with the VPLUG computer code.

LOCA Analysis Results

The changes implemented for the latest revision of the Containment Vessel analysis only affected long-term cooling. The short-term response was not affected. Consequently, the 14.14 foot-squared double-ended guillotine break at the Steam Generator inlet remains the bounding LOCA, with a peak pressure of 37.9 psig and peak temperature of 256.4°F.

The individual changes implemented by the latest revision of the Containment Vessel analysis resulted in minor changes to the long-term cooldown of the Containment Vessel. Also, the combination of the changes tended to cancel each resulting in only very small variations in all parameters. The long-term cooldown of the Containment Vessel was actually shortened since the increased CAC efficiency had a slightly greater impact on the analysis compared to the decrease in the Service Water flowrate to the Component Cooling Water heat exchanger. This was demonstrated by a reduction of 36 minutes in the time required to reduce the Containment Vessel's peak pressure by half.

Twenty-four acceptance criteria were reviewed and remain acceptable. These included criteria associated with peak Containment Vessel pressure and temperature, Containment Vessel metal temperature, electrical equipment qualification, and maximum Emergency Sump temperature.

Main Steam Line Break (MSLB) Analysis

No changes to the analysis techniques were implemented for the latest accident analysis of the MSLB in the Containment Vessel. As was done for the previous revision, the COPATTA computer code was utilized to compute the post-accident Containment Vessel response. Mass and energy release rates from the break were determined with the RELAP5/MOD2-B&W computer code. The analyzed break was the same as the Revision 3 analysis: a double-ended guillotine break located upstream of the Main Steam Isolation Valve.

Since the Containment Spray system, Emergency Sump and the Decay Heat/Component Cooling Water loops are not modeled in COPATTA for a MSLB, only one change was implemented for the MSLB which was to incorporate the heat duty table for the new Containment Air Cooler coils.

Due to the small consequences of the changes associated with the three addendums to the previous revision, a MSLB analysis (i.e., COPATTA run) of the combined changes was not performed. Therefore, the latest results were compared to the previous analysis. The results show that the Containment Vessel's peak pressure increased 0.2 psi, from 21.4 psig to 21.6 psig. The peak air/steam temperature increased 0.5°F, from 265.3°F to 265.8°F. Revision 4 of the containment analysis showed that all of the design requirements associated with the MSLB remain satisfied. These include peak Containment Vessel pressure and temperature, Containment Vessel metal temperature and electrical equipment qualification criteria. The Containment Vessel's metal temperature remains below 264°F since the metal temperature during the transient is equal to the saturation temperature, as discussed in USAR Section 15.4.4.2.3.3. The saturation temperature corresponding to the peak pressure is 238°F.

It was concluded that the changes associated with the re-analysis of the LOCA and MSLB do not meet any of the criteria in paragraph (c)(2) of 10 CFR 50.59.

Reanalysis of Loss of Feedwater Event
Calculation C-NSA-060.00-014 and USAR Change Notice 05-008U
10 CFR 50.59 Evaluation 05-00908

ACTIVITY DESCRIPTION

A new Loss of Feedwater (LOFW) analysis was prepared. This reanalysis was done to close a condition report that documented that some inputs were overly conservative and some inputs were non-conservative. The acceptance criteria that were used for the original LOFW analysis, which are presented in USAR section 15.2.8.2.1, included the requirement to maintain the Reactor Coolant System (RCS) pressure below code pressure limits, but they did not include the design goal of preventing the pressurizer from becoming water-solid. Because of this, the original analysis did not include a verification of the worst-case Pressurizer level. The original analysis used inputs that maximized the RCS pressure response, but it has been determined that some of those inputs were non-conservative with respect to calculating the maximum Pressurizer level. It was also identified that the Steam Generator (SG) tube-to-shell temperature difference (tubes hotter than shell) should be maintained less than 65°F to limit tube loading.

To address the impact of these issues, a new LOFW analysis was performed. The new analysis included two cases; one case used inputs and modeling that maximized the RCS pressure response, and the other case used inputs and modeling that maximized the Pressurizer level response. Both cases verified that the fuel response (i.e., no Departure from Nucleate Boiling (DNB)) and the tube-to-shell temperatures differences were acceptable. This approach ensured that all acceptance criteria and design goals were met in all cases. The new analysis was performed by Framatome using the RELAP5/MOD2-B&W computer code, Version 25. The NRC approved this program for general use in evaluating non-LOCA transients for Babcock and Wilcox plants, but it is a different program than was used for performing the previous LOFW event analysis that is currently presented in the USAR. Therefore, use of this program to analyze a LOFW event for the DBNPS constitutes replacing a USAR-described evaluation methodology that used in the safety analysis.

SUMMARY OF EVALUATION

This activity did not include or require any changes to plant Structures, Systems, Components, or procedures. Revising the LOFW analysis had no impact on the frequency or likelihood of any accident or malfunction and did not affect the consequences or results of any other (i.e., non-LOFW) accident or malfunction. The results of the analysis showed that all acceptance criteria and design goals are met. This demonstrated that the fuel will not experience DNB (i.e., there will be no fuel failure), which ensures that there will be no increase in the consequences of a LOFW event. The reanalysis did not create the possibility of a different type of accident or a malfunction with a different result. The results indicate an acceptable plant response that does not exceed the design basis limit for any fission product barrier, and no fission product barrier design basis limit is altered.

The LOFW re-analyses were performed using a different computer code than was used for the previous USAR analyses. However, the evaluation concluded that the methodology used for the new analyses was appropriate for the intended application, the terms and conditions for its use were satisfied, and the method was approved by the NRC. Since these conditions were met, the this activity does not meet any of the criteria in paragraph (c)(2) of 10 CFR 50.59.

Replacement of Obsolete Radiation Monitors
Engineering Work Request 01-0214-02
10 CFR 50.59 Evaluation 05-01462

ACTIVITY DESCRIPTION

This activity replaces several analog process and area radiation monitors with digital monitors, as well as replacing or upgrading their associated filter drives and iso-kinetic tips. Each of the affected radiation monitors is non-safety-related, non-seismic, and non-"EQ" (environmentally qualified). The existing Victoreen radiation monitors, installed in various areas throughout the plant, became obsolete and unreliable due to discontinued support from the vendor and the lack of available spare parts.

This activity upgrades the anchorage for the control room cabinet C5765, which holds the station's non-safety-related radiation rate meters. Cabinet C5765 is adjacent to several safety related control room cabinets, and must meet the applicable anti-fall-down seismic criteria.

Replacement of the rate meters for the radiation monitors introduces three failure modes. The remaining portion of the changes meets the 10 CFR 50.59 screening criteria. Therefore, the scope of the evaluation is to consider the new failure modes and their effects, with regard to potential adverse impacts on the process radiation monitor functions.

SUMMARY OF EVALUATION

The activity evaluated was limited to replacement of analog radiation monitors in the Containment Purge and Exhaust System, Fuel Handling System, and Radwaste Area Exhaust System with digital monitors. The affected instruments are not identified as accident initiators for any USAR-analyzed accident, and the activity does not change the design function or operation of the monitors, nor does it result in any new interfaces or interactions with other plant components. Thus it was concluded that these changes do not affect any structure, system, or component in a manner that would result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the USAR.

The Containment Purge and Exhaust monitors provide an alarm and indication function, and will, upon detection of a high level of radioactivity in the exhaust path of the containment purge and exhaust system, shut down the containment purge system supply and exhaust fans, and close their inlet and outlet dampers. A high radiation trip does not automatically isolate containment or start the Emergency Ventilation System (EVS). Operator action is relied upon to establish containment isolation and start the EVS. This operator response to the radiation monitor alarm is not changed as a result of the installation and use of the new radiation monitors. The new devices have the same level of qualification (non-safety, non-seismic) as the existing devices for this application, and will be installed, calibrated and tested in accordance with approved procedures. The change has no effect on seismic, separation, environmental qualification or single failure criteria. The Radiation Monitor software/firmware is developed and maintained under the manufacturer's 10 CFR 50 Appendix B Quality Assurance Program to ensure high reliability. The operating software is read-only (burned on an EPROM) and can not be altered by the user; therefore software failure can not be initiated by any human errors attributed to field personnel who will use and maintain the new devices. Each EPROM is serial-numbered and controlled to ensure the proper vendor supplied software is installed. Failure mode and effects analysis results indicate the the new devices are less susceptible to electromagnetic interference and radio frequency interference than the existing devices, and the effects of new failure modes for the digital devices are bounded by failure modes and effects for the existing analog devices. Based on an excellent service history of more than twenty years, the new digital devices for the Containment Purge Exhaust Radiation Monitors, Fuel Handling Exhaust System Radiation Monitors, and Radwaste Area Exhaust System Radiation Monitors are expected to be more reliable than the existing analog devices. Therefore, the activity will not result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system or component import to safety previously evaluated in the USAR.

As described in USAR Section 6.2.3, the Containment Purge System radiation monitors monitor the containment purge exhaust for particulate activity and isotopes I-131 and Xe-133, and initiate a control room alarm and automatically shut down the purge system supply and exhaust fans, and close the associated dampers. This USAR section further states that containment purge and exhaust penetrations must be isolated by operator actions. Since the containment purge and exhaust system is operated in the containment purge mode only during Modes 5 and 6, the only analyzed accident that would result in high radiation activity detected by these radiation monitors is a fuel handling accident inside containment. However, the accident analysis does not assume that containment isolation occurs. Therefore, a failure of the instrumentation to function will have no effect on the analyzed accident since the consequences are not dependent on isolation of the containment purge and exhaust system. The new devices will provide the same function as the existing devices in detecting high radiation levels, should a fuel handling accident occur inside containment. In addition, there are no analyzed accidents that rely on the Radwaste Area Exhaust System radiation monitors or the Fuel Handling Exhaust System radiation monitors to perform any mitigation functions. Therefore, the proposed activity does not result in more than a minimal increase in the consequences of an accident, or in the consequences of a malfunction of a structure, system, or component important to safety, previously evaluated in the USAR.

The devices affected by this change are not identified as accident initiators for any USAR analyzed accidents. The new devices have the same level of qualification (non-safety, non-seismic) as the existing devices, and will be installed, calibrated, and tested in accordance with approved procedures. The change has no effect on seismic, separation, environmental qualification, or single failure criteria. The radiation monitor software/firmware is developed and maintained under the manufacturer's 10 CFR 50 Appendix B Quality Assurance Program to ensure high reliability. The operating software is read-only (burned on an EPROM) and can not be altered by the user; therefore, software failure can not be initiated by any human errors attributed to field personnel. Each EPROM is serial numbered and controlled to ensure the proper vendor-supplied software is installed. Failure modes and effects analysis results indicate that the new devices are less susceptible to electromagnetic interference and radio interference than the existing devices, and the effects of new failure modes for the digital devices are bounded by failure modes and effects for the existing analog devices. Based on an excellent service history of more than twenty years, the new digital devices are expected to be more reliable than the existing analog devices. The instrumentation changes being made do not affect overall system performance or reliability, do not adversely affect instrument accuracy or response, do not cause systems to be operated outside their design limits, do not affect any system interface in a way that could lead to an accident, do not change safety system operation, and do not increase the possibility of operator error due to added complexity or other human factors conditions. Therefore, the activity does not create a possibility for an accident of a different type than any previously evaluated in the USAR.

The new digital devices for the Containment Purge Exhaust Radiation Monitors, Fuel Handling Exhaust System Radiation Monitors, and Radwaste Area Exhaust System Radiation Monitors have the same level of qualification as the existing devices, and will be installed, calibrated, and tested in accordance with approved procedures. Failure modes and effects analyses indicate that the new devices are less susceptible to electromagnetic interference and radio frequency interference than the existing devices. The change has no effect on seismic, separation, environmental qualification, or single failure criteria. Therefore the activity will not create a possibility for a malfunction of a structure, system, or component important to safety, with a different result than any previously evaluated in the USAR.

The proposed activity has no direct or indirect effects on the design basis limits for any fission product barrier. The proposed activity is limited to replacement of obsolete analog radiation monitors in the Containment Purge and Exhaust System, Fuel Handling Exhaust System, and Radwaste Area Exhaust System with a newer digital model. The new devices will provide the same functions as the existing devices. These non-safety devices are not relied upon to protect any fission product barrier. A failure of the instrumentation to function will have no effect on the design basis limits for any fission product barrier (fuel cladding, reactor coolant system, or containment), or on the results of any analyzed accident. Therefore, the proposed activity does not result in a design basis limit for a a fission product barrier, as described in the USAR, being exceeded or altered.

The proposed activity is limited to replacement of obsolete analog radiation monitors with a new digital model. The change is not associated with any methodology used to establish design bases or in performing safety analyses. Therefore, the proposed activity has no effect on any method of evaluation used in establishing the design bases or in the the safety analyses described in the USAR.

HELB Compartment Pressurization Analysis Outside Containment
Methodology Change to GOTHIC
USAR Change Notice 05-019U
10 CFR 50.59 Evaluation 05-03247

ACTIVITY DESCRIPTION

USAR Section 3.11.2.2.1, "HELB Computer Analysis," was revised to include use of the GOTHIC version 7.0 computer code as an alternative method for analyzing compartment pressure response to High Energy Line Breaks (HELBs) outside containment. GOTHIC version 7.0 was approved in an NRC Safety Evaluation Report (SER) for the River Bend Station. The USAR was also revised to add the NRC SER-identified limitations and restrictions for this application. HELB analyses have previously been analyzed for the DBNPS using the PCFLUD computer code. Because of the age and limitations of the PCFLUD code, it was desired to implement a newer, state-of-the-art HELB analysis code.

GOTHIC is widely used in the nuclear industry as a state-of-the-art, general purpose, thermal-hydraulics computer program for analyzing HELB compartment pressurization, and it has been approved by the NRC for that application. The NRC has also approved other applications of GOTHIC 7.0, such as for compartment pressurization analyses within containment and for establishing environmental qualification temperature profiles. However, the DBNPS use of GOTHIC will be limited to HELB sub-compartment pressure response analyses outside of containment.

GOTHIC solves the conservation equations for mass, energy, and momentum for multi-component, multi-phase flow for HELB compartment response analyses. Use of this thermal-hydraulics computer program for analyzing HELB compartment pressurization responses at the DBNPS constitutes a change in a USAR-described evaluation methodology that is used in the safety analyses.

SUMMARY OF EVALUATION

The proposed activity provided for the use of the GOTHIC code as an alternative method for determining compartment pressurization response to postulated HELBs in areas outside of containment. This evaluation demonstrates that the DBNPS proposed implementation of the GOTHIC methodology is in compliance with NEI 96-07, revision 1, *Guidelines for 10 CFR 50.59 Implementation*, Section 4.3.8, which allows the change to be classified as "not a departure from a method of evaluation described in the USAR.

The NRC staff's approval of GOTHIC 7.0 for River Bend Station was based on the NRC's confirmation that the code provides appropriate analytical results. The NRC's approval imposed certain restrictions on the use of GOTHIC to ensure it produces acceptable performance. The proposed USAR change identified and documented the requirement to comply with all imposed restrictions on the use of GOTHIC.

The only difference between the GOTHIC application at the River Bend Station and at the DBNPS is that this evaluation imposes an additional restriction on its use at the DBNPS. Specifically, the additional restriction is that GOTHIC will be used at the DBNPS only for HELB compartment pressure analyses outside containment. Although the NRC SER approved GOTHIC for additional uses, the USAR change does not allow use of GOTHIC for HELB sub-compartment analyses inside containment, or for establishing environmental qualification temperature profiles. These potential future applications will require further evaluation and USAR changes. This limitation does not affect the GOTHIC 7.0 methodology as approved by the NRC SER for River Bend.

The evaluation concluded that FENOC's proposed application of the GOTHIC methodology at the DBNPS, including the NRC-imposed restrictions, satisfies requirements for treating the change as "not a departure from a method of evaluation described in the USAR."

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Attachment 3

COMMITMENT LIST

THE FOLLOWING LIST IDENTIFIES THOSE ACTIONS COMMITTED TO BY THE DAVIS-BESSE NUCLEAR POWER STATION (DBNPS) IN THIS DOCUMENT. ANY OTHER ACTIONS DISCUSSED IN THE SUBMITTAL REPRESENT INTENDED OR PLANNED ACTIONS BY THE DBNPS. THEY ARE DESCRIBED ONLY FOR INFORMATION AND ARE NOT REGULATORY COMMITMENTS. IF THERE ARE ANY QUESTIONS OR IF ADDITIONAL INFORMATION IS REQUIRED, PLEASE CONTACT MR. GREGORY A. DUNN, MANAGER – FENOC FLEET LICENSING AT (330) 315-7243.

COMMITMENTS	**DUE DATE**
None	N/A