FENOC
FirstEnergy Nuclear Operating Company

Barry S. Allen
Site Vice President

419-321-7676
FAX: 419-321-7582

June 23, 2008

L-08-139

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 Report of Facility Changes, Tests, and Experiments for the Davis-Besse Nuclear Power Station is provided as Attachment 1. The report covers the period of May 19, 2006, through February 14, 2008.

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 – FENOC Fleet Licensing, at (330) 761-6071.

Sincerely,

Barry S. Allen

Barry S. Allen

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

cc: NRC Region III Administrator
NRC Resident Inspector
NRR Project Manager
Utility Radiological Safety Board

IE47
NRR

Title

Modifications to enhance Main Fuel Handling Bridge (MFHB) performance

Activity Description

This activity will install several enhancements to the Main Fuel Handling Bridge (MFHB). The MFHB performs the fuel and control component assembly handling activities within containment between the reactor and the fuel transfer area. The enhancements provided by this activity include both hardware changes and software (i.e., programmable logic controller (PLC) program) changes:

a. Modifying guide blocks inside the fuel mast to allow smoother fuel assembly entry into the fuel mast tube.

b. Activation of a bridge/trolley jog function to allow the slow reposition of the bridge and trolley without the fuel assembly being fully withdrawn into the fuel mast.

c. Control logic change to automatically utilize jog speeds when using the main fuel hoist controls.

d. Adding “Open Water” permissive controls (PLC logic, indicating light, enable switch, wiring). This feature is being installed for future use and is disabled through non-activation of the permissive switch.

e. Adding an “Intermediate Slow Zone” to increase the unloaded fuel hoist speed above the transfer basket.

f. Adding additional and separate jog speeds.

g. Maximize fuel hoist speed when the grapple is not engaged.

h. Improving the MFHB control console grounding to prevent erratic encoder indication.

i. Adding a blinking feature to the “Slow Zone” indicating lights.

All activities above, except item b) Activation of a bridge/trolley jog function to allow the slow reposition of the bridge and trolley without the fuel assembly being fully withdrawn into the fuel mast, were screened under 10 CFR50.59 and did not require a 10 CFR 50.59 evaluation.

Therefore, only activity b) "Activation of a Bridge/Trolley Jog function to allow the slow reposition of the bridge and trolley without the fuel assembly being fully withdrawn into the fuel mast" is addressed in the evaluation.

Summary of Evaluation

The current MFHB control logic contains a feature that includes a jog function where the bridge and trolley can be slowly repositioned without the fuel assembly being fully within the fuel mast tube. This feature has not been used and is currently deactivated through selection of a height permissive that ensures the permissive can not be met. This activity will activate the jog function whenever the fuel assembly is greater than approximately twelve inches from full down. This feature will be used either when fuel is raised or lowered "off-index" or for minor repositioning adjustments "on-index." For example, when a fuel assembly is lowered "off-index" this bridge/trolley jog feature provides the means whereby the fuel assembly can be moved to its proper core position. This change does not involve any PLC program change since the logic currently exists.

This setpoint change activates bridge and trolley control logic that will allow jog speed without the fuel assembly being fully withdrawn into the protective tube. For bridge and trolley speeds beyond the jog speed, the bridge and trolley are interlocked to prevent movement until the fuel assembly has been completely withdrawn into the protective mast tube.

The intent of this interlock is to prevent the potential for fuel assembly damage as a result of contact with other objects. The bridge and trolley speeds that will be permitted are sufficiently low that the bridge and trolleys motors will not be able to impart enough motion to inflict damage on the fuel assembly. This design change verifies the jog speeds are set such that the maximum lateral force on an exposed fuel assembly is less than the limit imposed by the fuel assembly manufacturer.

The changes identified do not modify the process or functions whereby fuel assembly or control rods are grappled, including lift height and weight limits. Bridge, trolley, and hoists speeds are maintained within the design specified limits for the equipment. Bridge and trolley structure integrity is not adversely affected. MFHB structural integrity and seismic response has not been adversely affected. The changes do not affect compliance with the codes and standards identified in UFSAR Section 9.1.4.2.2.

Since the scenario for fuel damage is essentially identical pre and post modification, there is no increase in the likelihood of occurrence of an accident or malfunction previously evaluated in the UFSAR to occur as a result of this modification.

The fuel handling accident described in UFSAR section 15.4.7 is the only accident that is related to fuel handling equipment. There is no discussion in the UFSAR on the failure mechanism which causes the fuel to fail. The changes in the modification do not modify the process or functions whereby a fuel assembly are grappled or handled, including lift height and weight limits. The proposed change does not affect the results of the fuel handling accident or any malfunctions as described the UFSAR, since any fuel failures caused by malfunctions of the proposed changes would be bounded by the UFSAR analysis.

The allowance of fuel movement without fuel being completely withdrawn within the mast tube can not cause accidents of a different type than previously evaluated in the UFSAR nor a malfunction with a different result since the only accident potentially connected with the MFHB is a fuel handling accident inside containment.

This change does not involve a change to any of the design basis of the three fission product barriers nor does it involve an evaluation methodology.

Therefore after implementation of the modification, the fuel handling system will continue to provide the necessary controls and interlocks to protect the fuel from any damage.

Based on the evaluation summarized above, it can be concluded that this change does not meet the criteria of Paragraph (c)(2) of 10 CFR 50.59. Therefore, a license amendment is not required.

Title

Changes to Dose Analysis for the Maximum Hypothetical Accident (MHA)

Activity Description

The activity being proposed is an increase to the control room dose during the MHA. This dose change is a result of a revision to the design analysis that incorporated the following:

- an increase to the unfiltered in-leakage into the control room when the control room is not pressurized (i.e., first ten minutes of the MHA) from 55.4 cubic feet per minute (cfm) to 63 cfm. This increased in-leakage is required as a result of control room envelope tracer gas tests.
- a change to the Xe-138 source term is required to correct the number used in the analysis of record.

Summary of Evaluation

The proposed activity primarily impacts the thyroid dose. The proposed activity increases the thyroid dose received by personnel in the control room during a MHA by 1.1 rem. Increases to the whole body dose and beta-skin dose are less than or equal to 0.1 rem.

The proposed increase in control room dose during the MHA is less than 10 percent of the difference between the regulatory limit and the current value. Therefore, as defined by NEI 96-07, Revision 1, the proposed activity does not result in a more than minimal increase in the radiological consequences of an accident previously evaluated in the UFSAR.

Based on the evaluation summarized above, it can be concluded that the changes associated with the revised MHA analysis do not meet the criteria of Paragraph (c)(2) of 10 CFR 50.59. Therefore, a license amendment is not required.

Title

Alloy 600 Mitigation

Activity Description

This change provides for the mitigation of sixteen Alloy 600/182/82 component / Dissimilar Metal Welds (DMWs) within the Reactor Coolant System (RCS) from cracking due to primary water stress corrosion cracking. As part of the mitigation process the six pressurizer level sensing line nozzles, the pressurizer vent nozzle, and the pressurizer sample nozzle DMWs will be mitigated with a half nozzle repair, which removes the outer half of the existing Inconel Alloy 600 nozzle and installs a new Inconel Alloy 690 nozzle to an Inconel Alloy 52/52M filler metal welded pad on the outside of the pressurizer. The pressurizer Inconel Alloy 600 thermowell will also be replaced with a new Inconel Alloy 690 thermowell. Both the half nozzle repairs and the thermowell replacement will leave the pressurizer carbon steel base metal in the annular region around the nozzles exposed to the effects of borated water in the reactor coolant. Exposure of borated water to an unclad carbon steel pressurizer base metal creates an adverse affect on the RCS pressure boundary UFSAR design function.

Summary of Evaluation

Repairs have been proposed under this change 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 pressurizer level instrument, vent, sample and thermowell nozzles. These repairs involve methods that will leave the current stainless steel clad carbon steel pressurizer shell base metal exposed to RCS borated water in the annular region of the nozzles. This will increase the general corrosion rate of the pressurizer shell base metal in these regions from zero inches per year to 1.42 mils per year. Over a forty year life of the plant, this will result in a loss of 57 mils of the pressurizer's carbon steel shell in the nozzle annular regions. This adverse effect has been determined to be technically acceptable. The allowable radial corrosion limit to meet ASME Section III, Class 1 Code design is 293 mils for the level instrument nozzles, 493 mils for the sample nozzle and 495 mils for the vent and thermowell nozzles. These corrosion limits will also assure that the original nozzle remnants and J-groove welds (located on the inside of the vessel shell) do not become detached and enter the RCS as loose parts, nor affect the operation of the attached components.

This 10CFR50.59 evaluation concludes that a NRC approved license amendment is not required for implementation of the described repairs.

Title

Containment Vessel Analysis

Activity Description

A revised analysis of the containment vessel's response during a large-break Loss of Coolant Accident (LOCA) and Main Steam Line Break (MSLB) accident has been performed. Specifically, the proposed changes are: the number of plugged Containment Air Cooler (CAC) tubes modeled for the containment vessel response analyses (i.e., both the LOCA and MSLB accidents) has been increased from 0 to 8 percent; the enthalpy of the steam released during the MSLB accident is being increased and; a thermal lag model, as defined by NUREG 0588, Revision 1, has been added to the COPATTA model for a MSLB in order to predict electrical equipment temperatures that will be used by environmental qualification analyses.

Summary of Evaluation

LOCA

The only change implemented for the analysis of the containment vessel's response during a large-break LOCA was an increase to the number of CAC plugged tubes. Based on refueling outage inspections, a sensitivity study was performed that increased the number of CAC plugged tubes from 0 to 8 percent.

Results of the sensitivity study indicate that there is only a very small impact with respect to containment vessel and Emergency Core Cooling System (ECCS) parameters when the CAC heat transfer capability by the equivalent of 8 percent plugged tubes. The changes are sufficiently small, such that, they are considered negligible. Consequently, all LOCA cases for the previous analysis remain applicable. A limitation has been added to the LOCA analysis documentation to ensure that all future LOCA analyses include a CAC that has 8 percent tube pluggage.

MSLB

The following changes were implemented as part of the revised containment vessel response analysis during a MSLB.

- The number of plugged CAC tubes was increased from 0 to 8 percent,
- The break release enthalpies were adjusted to conservatively account for liquid carryover and,
- A thermal-lag model was added to predict electrical equipment temperatures.

The increase to the break enthalpies caused a significant increase in the containment vessel's vapor temperature, such that, the vapor temperature was not utilized for input to equipment qualification analyses as was done for the previous analysis. Instead, a thermal-lag model, as specified by Nuclear Regulatory Guide (NUREG) 0588 was utilized to predict electrical equipment temperatures that are subsequently input to the environmental qualification analyses.

There are two specific acceptance criteria that are applicable to the MSLB: the peak containment vessel pressure shall be less than 38 pounds per inch gauge (psig) and the peak containment vessel structural metal temperature shall be less than 264°F. The proposed calculation documents that both of these acceptance criteria are satisfied.

Based on the discussion above, it can be concluded that the changes associated with the LOCA and MSLB accident analyses do not satisfy any of the criteria in Paragraph (c)(2) of 10 CFR 50.59.

Title

Dose Increase Caused by ECCS Leakage into the Auxillary Building and Borated Water Storage Tank (BWST)

Activity Description

The proposed activity is the release of a revised engineering calculation. This calculation analyzes the dose consequences of ECCS fluid leakage during the Maximum Hypothetical Accident (MHA). Fluid leakage into both the auxiliary building and BWST are considered. Increases to the control room, Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) doses are determined.

Summary of Evaluation

Previous dose analyses of the MHA considered the dose consequence of ECCS fluid leakage based on a value of 1.5 gallons per hour (gph). The proposed activity analyzes a total ECCS leakage to the auxiliary building of 40 gph (summation of both trains). A maximum value of 40 gph is verified by plant procedures. Although test procedures ensure that 40 gph of ECCS leakage is not exceeded for the auxiliary building, actual ECCS leakage to the BWST is not quantified. For this reason, the proposed calculation conservatively evaluates an additional leakage of 40 gph to the BWST. Credit for the Emergency Ventilation System (EVS) filters is considered for the ECCS leakage into auxiliary building since the leakage evaporates and becomes an airborne source. All of the activity released to the auxiliary building was modeled to be released instantaneously to the environment (i.e., no decay). Since the only motive force for a release from the BWST is the leaking fluid, the airborne release rate from the BWST was set to be equal to the ECCS fluid leak rate. No credit for emergency ventilation system filtration is taken for leakage to the BWST since that leakage is outside the EVS boundary. All other design inputs are based on approved engineering calculations.

The proposed activity primarily impacts the thyroid dose. The proposed activity increases the thyroid dose received as follows: 0.8 rem for the control room, 2.0 rem at the EAB and, 0.8 rem for the LPZ. Increases to the whole body dose and beta-skin dose are negligible for all locations.

The proposed increase in thyroid doses during the MHA are less than 10 percent of the difference between the regulatory limit and the currently reported UFSAR values. Therefore, as defined by, NEI 96-07, Rev. 1, the proposed activity will not result in more than minimal increase in the radiological consequences of an accident previously evaluated in the UFSAR.

Based on the evaluation summarized above and documented herein, it can be concluded that the changes associated with the revised MHA analysis do not meet the criteria of Paragraph (c)(2) of 10 CFR 50.59. Therefore, a license amendment is not required.

Title

Control Room Dose Increase Due to Emergency Ventilation System (EVS) and Control Room Emergency Ventilation System (CREVS) Charcoal Adsorbers

Activity Description

The proposed activity is the release of an engineering calculation. This calculation determines the control room dose following a MHA caused by the radioactivity from the charcoal Adsorbers that are part of the EVS and the CREVS.

Summary of Evaluation

Following the MHA (i.e., LOCA), the EVS and the CREVS are placed into service. Both of these systems employ charcoal Adsorbers that become radioactive sources during operation due to the absorption of radioactive iodine.

This analysis was performed with the MICROSHIELD computer program. The total cumulative control room whole body dose due to the EVS and CREVS charcoal Adsorbers was determined to be 0.34 rem.

The total cumulative control room dose caused by the fixed sources (EVS and CREVS charcoal Adsorbers, and containment vessel) is equal to 0.57 rem (0.34 rem + 0.23 rem). The previous UFSAR value was 0.23 rem and only included the direct gamma dose from the containment vessel.

The proposed increase in control room dose during the MHA is less than 10 percent of the difference between the regulatory limit of 5 rem and the currently reported UFSAR value. Therefore, as defined by, NEI 96-07, Rev. 1, the proposed activity will not result in more than minimal increase in the radiological consequences of an accident previously evaluated in the UFSAR.

Based on the evaluation summarized above and documented herein, it can be concluded that the proposed activity does not meet the criteria of Paragraph (c)(2) of 10 CFR 50.59. Therefore, a license amendment is not required.

Title

Control Room Radiation Dose Due to Fuel Handling Accident (FHA) inside Containment

Activity Description

The proposed activity is the release of a revision to an engineering calculation. This revision incorporates two changes to the analysis that determines the control room dose during a FHA inside of the containment vessel. These changes are the unfiltered in-leakage into the control room are being increased from 55.4 cfm to 61 cfm and the thyroid Dose Conversion Factor (DCF) is being changed.

Summary of Evaluation

The change to the DCF causes a decrease in the dose received by the control room operators during an FHA inside of the containment. The change to the thyroid DCF has been previously approved by the Nuclear Regulatory Commission.

The proposed activity increases the control room's unfiltered in-leakage during a FHA inside of the containment vessel from 55.4 cfm to 61 cfm. This increase is based on measured data. The increased in-leakage causes the dose received by the control room operators to increase. This increase is less than 10 percent of the difference between the regulatory limit and the current value. Therefore, as defined by, NEI 96-07, Revision 1, the proposed change is not more than minimal increase in the radiological consequences of an accident previously evaluated in the UFSAR.

Based on the evaluation summarized above and documented herein, it can be concluded that the changes associated with the revised FHA analysis do not meet any of the criteria of Paragraph (c)(2) of 10 CFR 50.59. Therefore, a license amendment is not required.

Title

Reanalysis of Startup Event

Activity Description

The startup accident analysis presented in UFSAR Section 15.2.1 was revised to use the maximum Moderator Temperature Coefficient permitted by Technical Specifications. This analysis was also revised so the RCS flow rate conformed to the Davis-Besse design flow rate and used a High RCS Pressure trip setpoint that bounds the existing value permitted by Technical Specifications.

The computer code used in the original analysis (KAPP4) is no longer available for use. However, an acceptable alternative computer code (RELAP5/Mod2 - B&W) has been approved for use in the analysis. The use of the new computer code is considered a change to a methodology described in the UFSAR.

UFSAR Section 15.2.1, which reports the results of the startup accident analysis, is being revised to reflect the results of the new analysis and to report the change in methodology used to determine the accident results. The computer codes used in the analysis are to be added to the list of references in the UFSAR. Additionally, UFSAR Appendix 4B, Table 7-1, which references part of the material being revised in UFSAR Section 15.2.1 is being revised to be consistent with the change in the analysis.

The Technical Specification Bases presently list the High RCS Pressure trip as 2355 psig. This does not include any instrument error, although that has been accounted for by the analysis. The Technical Specification Bases are to be revised to reflect the analytical value used in the startup event analysis. Actual field setpoints based on the listed analytical value are then developed in plant design documents using accepted practices.

Summary of Evaluation

The startup accident analysis was re-analyzed in its entirety due to identified issues with the existing analysis. The previously used computer codes (KAPP4 and CADDS) are no longer available. RELAP5/Mod 2 - B&W version computer code is the current code approved by the NRC for non-Loss of Coolant Accident analyses in Topical Report BAW-10193P-A-00.

The results obtained using the RELAP5 code meet the same acceptance criteria used in the original UFSAR analysis. The current UFSAR Section 15.2.1.3 analysis reports a peak power level of less than 50 percent of Rated Thermal Power (RTP), whereas the RELAP5 analysis finds a peak power of 67.4% of RTP. The maximum allowable power level is 112% of RTP. The peak pressure reported in the existing UFSAR analysis is 2700 psig which is 2714.4 pounds per square inch actual (psia), whereas the new analysis reports a peak pressure of 2750.2 psia. The maximum allowable peak pressure is 2750 psig (2764.4 psia).

Due to the changes in input values, it was necessary to run the safety analysis again. The results are not bounded by the existing analysis, and therefore the changes are considered to have an adverse effect. The new analysis determined that all acceptance criteria are met with all the input changes, so the changes made, while adverse, are minimally adverse. Therefore, NRC approval of the changes is not required.

The change in methodology does not alter how the plant is operated, nor alters the function or reliability of any equipment. Therefore all accident initiator and equipment malfunction frequencies are unaffected. Because all acceptance criteria are met, there is no change in the consequences of the accident. The analysis included the appropriate single-failure considerations so that no change in consequences occurs due to malfunction of equipment.

The change in methodology does not result in any change in how the plant is operated so that no new or different accidents or malfunctions are identified. No new results due to malfunctions of equipment occur since the plant design and operation are unchanged.

While the use of the new computer code is a change in method from that described in the UFSAR, the change has been approved by the NRC in Topical Report BAW-10193P-A-00. No fission product barrier limits are exceeded or altered. Therefore, the use of the code does not constitute a departure from a method, as discussed in NEI 96-07, Revision 1.

Based on the information discussed above, it is concluded that a license amendment is not required prior to implementing the proposed activity.

Title

Revision to Reactor Protection System (RPS), Safety Features Actuated System (SFAS), Steam and Feedwater Rupture Control System (SFRCS) Trip or Steam Generator (SG) Tube Rupture Procedure

Activity Description

The RPS, SFAS, SFRCS Trip or SG Tube Rupture procedure is being revised to require plant operators to manually align the Service Water (SW) return header to a safety related return flow path. This will also entail providing power to the discharge valve motor operator, if open, since the open discharge valve motor operator is kept de-energized to prevent inadvertent closure during a fire. Should the motor operator fail to close the valve, the procedure directs manual closure.

In order to accomplish these actions, two steps will be added to each of the following sections: 10, "Large LOCA Cooldown"; 11, "RCS Saturated with SGs Removing Heat Cooldown"; 12, "MU/HPI PORV Cooldown;" and 13, "RCS Subcooled with SG Removing Heat Cooldown." These steps will direct the plant operators to ensure that a safety related discharge path for service water exists, and if needed, to re-energize the motor operator to any open non-safety related SW discharge isolation valve, and to close the associated valve.

Summary of Evaluation

The calculations performed to support the changes to the RPS, SFAS, SFRCS Trip or SG Tube Rupture procedure determined that adequate service water cooling will exist to all required loads with the exception of the CREVS. That system is started manually. The procedural direction to start the CREVS will now specifically direct the plant operators to start the system with the air cooled mode available to ensure that the CREVS safety function can be accomplished even if there is less than adequate service water flow to the water cooled condensers. This is not replacing an automatic action with a manual action; rather, it is clarifying and enhancing a previously licensed manual action. If the diverse air cooled mode is not available, plant procedures direct the operators to provide a safety related flow path to service water to improve CREVS cooling in the water cooled mode to the design requirement.

Because the remaining loads on the Service Water System (SWS) are adequately cooled, the plant response to the LOCA is not significantly altered. All acceptance criteria of the calculation and associated output documents are met. All design basis functions are accomplished. However, credit for operator action is necessary to demonstrate adequate plant response across the entire spectrum of flow blockages within the non-safety related portions of the SWS discharge flow path. The required actions also prevent excessive diversion of volume from the ultimate heat sink. The manual actions are acceptable when compared to the expectations of NEI 96-07.

All design basis functions are met. No accident initiators are altered. The likelihood of equipment malfunction is not increased. The consequences of accidents or malfunction of systems, structures, or components are unchanged. No new or different accidents or malfunctions of systems, structures or components are created. No fission product barrier limits are changed or challenged. No new methods of analysis have been used in the safety analyses performed to assess the change. Consequently, the proposed revision of the procedure does not meet the criteria of Paragraph (c)(2) of 10CFR50.59 and NRC approval prior to implementation is not required.

Title

Containment Vessel Analysis

Activity Description

The proposed activity is the release of Revision 6 to engineering calculation "Containment Vessel Analysis." This revision incorporates the analysis of a partially-blocked nonsafety-related service water discharge line for both a LOCA and MSLB, and evaluates containment temperatures (vapor and sump) and system fluid temperatures based on an initial containment vapor temperature of 120°F rather than 90°F.

Summary of Evaluation

This evaluation considered the simultaneous occurrence of an accident and blockage of a nonsafety-related SW discharge line. This blockage will cause lower SW component flowrates with respect to design values. The components affected during accidents associated with the containment vessel response analysis (i.e., large-break LOCA and MSLB) are the containment air cooler and the component cooling water heat exchanger. The reduced flowrates only occur for a short duration (i.e., less than 2.8 hours) until the operators swap the blocked nonsafety-related SW discharge line to a safety-related discharge line. Results of the containment response analysis indicate that the effect of the component flowrate reductions on temperatures and pressure responses of the containment and associated systems is minimal. All acceptance criteria associated with the containment analysis were evaluated to remain satisfied.

Basing post-LOCA containment and system temperatures on an initial containment air temperature of 120°F rather than 90°F causes only minimal increases in the post-accident containment temperature response. This is because the energy added to the containment as a result of the break is significantly greater than the initial energy content of the containment. Thus, if the initial temperature is increased from 90°F to 120°F, the corresponding increase in peak temperature is 3°F (pressure reduction of about 0.1 psi). All acceptance criteria associated with the containment analysis were evaluated to remain satisfied with an initial temperature of 120°F.

Based on the discussion above, it can be concluded that the changes associated with the LOCA and MSLB accident analyses evaluated herein do not satisfy any of the criteria in Paragraph (c)(2) of 10 CFR 50.59.