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License Number NPF-3

Docket Number 50-346

Serial Number 2727

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United States Nuclear Regulatory Commission
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Subject: Supplemental Information Regarding License Amendment Application to Revise Technical Specification 3/4.5.2, Emergency Core Cooling Systems - ECCS Subsystems - $T_{avg} \geq 280°F$
(License Amendment Request No. 97-0007; TAC No. MB1689)

Ladies and Gentlemen:

On March 30, 2001, the FirstEnergy Nuclear Operating Company (FENOC) submitted an application for an amendment to the Davis-Besse Nuclear Power Station (DBNPS), Unit Number 1, Operating License Number NPF-3, Appendix A Technical Specifications, regarding Technical Specification 3/4.5.2, Emergency Core Cooling Systems - ECCS Subsystems - $T_{avg} \geq 280°F$. The proposed amendment (DBNPS Serial Number 2664) would allow a 24 month surveillance interval for the vacuum leakage rate test of the watertight enclosure for Decay Heat Removal System valves DH-11 and DH-12. The surveillance interval is presently 18 months.

The FENOC staff has informally received a request for additional information from the NRC staff regarding the license amendment application. Enclosure 1 provides the response to this request for additional information, as discussed with the NRC staff via a conference call on October 18, 2001. This information does not affect the conclusions of the Safety Assessment and Significant Hazards Consideration (SASHC) included in the license amendment application submitted on March 30, 2001. The SASHC concluded that the proposed change does not adversely affect nuclear safety and does not involve a significant hazards consideration.

With the submittal of this supplemental information, FENOC requests that the NRC staff complete its review and approval of the license amendment application by December 31, 2001.

A001

Should you have any questions or require additional information, please contact Mr. David H. Lockwood, Manager - Regulatory Affairs, at (419) 321-8450.

Very truly yours,

John J Messina
for Guy Campbell

MKL

Enclosures

cc: J. E. Dyer, Regional Administrator, NRC Region III
S. P. Sands, NRC/NRR Project Manager
D. J. Shipley, Executive Director, Ohio Emergency Management Agency, State of Ohio (NRC Liaison)
D. S. Simpkins, NRC Region III, DB-1 Resident Inspector
Utility Radiological Safety Board

Docket Number 50-346
License Number NPF-3
Serial Number 2727
Enclosure 1

SUPPLEMENTAL INFORMATION

IN SUPPORT OF THE

APPLICATION FOR AMENDMENT

TO

FACILITY OPERATING LICENSE NPF-3

DAVIS-BESSE NUCLEAR POWER STATION

UNIT NUMBER 1

Attached is supplemental information for Davis-Besse Nuclear Power Station (DBNPS), Unit Number 1 Facility Operating License Number NPF-3, License Amendment Request Number 97-0007 (DBNPS Serial Number 2664, dated March 30, 2001).

This information, submitted under cover letter Serial Number 2727, includes a response to an informal NRC Request for Additional Information.

I, Guy G. Campbell, state that (1) I am Vice President - Nuclear of the FirstEnergy Nuclear Operating Company, (2) I am duly authorized to execute and file this certification on behalf of the Toledo Edison Company and The Cleveland Electric Illuminating Company, and (3) the statements set forth herein are true and correct to the best of my knowledge, information and belief.

For: Guy G. Campbell, Vice President – Nuclear

By: John Messina

John Messina, Director – Work Management

Affirmed and subscribed before me this

Yvonne Hall

Notary Public, State of Ohio

YVONNE M. HALL
Notary
State of Ohio
My Commission Expires
July 25, 2004

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

REGARDING

LICENSE AMENDMENT REQUEST (LAR) 97-0007

FOR

DAVIS-BESSE NUCLEAR POWER STATION UNIT NUMBER 1

NRC Request for Information:

1. Over the time period reviewed in the submittal, please provide the dates of the tests which tested the valve housing structure leakage integrity. Indicate if it was an as-found or as-left test and its result. If the closure of the structure was broken by use of the small inspection port between the refueling outage tests, then provide the dates on which this occurred. If a leakage integrity test was performed afterwards, provide the result also. Furthermore, indicate the number of leaks detected, location on the structure, cause if known, type of fix for the leak, and if the leak location was repetitive.

DBNPS Response:

As stated in the license amendment application's "Effects on Safety" section, the time period included in the review spanned the Fifth Refueling Outage (5RFO) through 12RFO. As-left surveillance testing was performed on the watertight enclosure for Decay Heat Removal (DHR) System valves DH-11 and DH-12 ("decay heat valve pit") near the end of each of these refueling outages. In addition, surveillance testing was performed during the May 1997 Cycle 11 forced outage and the May 1999 Cycle 12 mid-cycle outage. The test completion dates, as-left test results, and acceptance criterion at the time of the test were as follows:

Outage	Test Date	Test Results (in-Hg)	Acceptance Criterion (in-Hg)
5RFO	11/15/88	8.6	≥ 5.5
6RFO	05/30/90	5.6	≥ 5.5
7RFO	10/26/91	6.0	≥ 5.5
8RFO	04/18/93	6.2	≥ 5.5
9RFO	11/04/94	6.0	≥ 5.5
10RFO	05/19/96	5.9	≥ 5.5
Cycle 11 Forced Outage	05/15/97	6.7	≥ 6.1
11RFO	05/11/98	6.5	≥ 6.1
Cycle 12 Mid-Cycle Outage	05/02/99	7.1	≥ 6.1
12RFO	05/07/00	6.4	≥ 6.1

As noted in the license amendment application, the valve pit cover is normally removed during shutdown outages for performing maintenance and test activities within the pit. Following these activities, and restoration of the pit cover, the surveillance test is performed near the end of each outage, prior to entry into Mode 4, i.e., as an as-left test. As an optional maintenance activity, the decay heat valve pit may be slightly pressurized and checked for leakage via a soap bubble test. In the event that the pressurization check or the initial surveillance test results indicate excessive leakage, additional maintenance activities are performed on the decay heat valve pit seals. Maintenance activities typically include the application of additional RTV or the replacement or repair of boot seals. A review of the maintenance records shows that details such as the number of leaks detected, location on the structure, cause, type of fix for the leak, and if the leak location was repetitive, has not typically been recorded.

As described in the license amendment application, the inspection port penetration consists of a 4-inch (nominal) pipe stub welded to the decay heat valve pit cover and sealed with a 4-inch (nominal) adapter and cap ("Kamlok" coupling) manufactured by Dover Corporation/OPW Division. This coupling is designed to be removed and reinstalled, and includes an integral viton gasket for sealing. Based on this design, and as reflected in SR 4.5.2.f and its associated Bases, surveillance testing is not required following opening of the inspection port. The opening of the inspection port is not tracked, however it is opened very infrequently at power.

As discussed in the license amendment application, Surveillance Requirement (SR) 4.5.2.f.1 is a non-standard, plant-specific requirement that was added to the DBNPS Technical Specifications at the time the Operating License was issued in 1977. This requirement was added due to the design of the valve pit cover, and was based upon performing testing at a refueling outage frequency rather than at a fixed absolute time-span. The purpose of this testing was to ensure the adequacy of the leak tightness of the valve pit watertight enclosure prior to commencing each operating cycle. The watertight enclosure is in a remote location during plant operation, and its cover is in a passive state once it is bolted and sealed in place. The SR also included performing testing after each opening of the enclosure and after any maintenance on or modification to the enclosure that could affect its integrity (e.g., SR 4.5.2.f.2 requires the test be performed "after" (not prior to) each opening of the enclosure).

As noted on page 3 of Enclosure 1 Attachment 1 of the license amendment application, the post-LOCA reactor vessel boric acid precipitation control (BPC) circulation flow path provided by valves DH-11 and DH-12 is no longer the primary BPC method that it was in 1977. The DBNPS was modified and licensed by the NRC in 2000 to use the pressurizer auxiliary spray flow path as the primary BPC method. Accordingly, although the BPC method provided by valves DH-11 and DH-12 is no longer the primary success path to mitigate a design basis accident, this SR will continue to provide reasonable assurance of the operability of the backup BPC method.

NRC Request for Information:

2. Describe how the submittal's stand-by failure rate for the valve housing structure was determined and how it was applied in the probabilistic safety assessment (PSA) model.

DBNPS Response:

The failure rate of the watertight enclosure was calculated using data from as-found leak (slight pressurization) checks. As described in the "Maintenance Records Review" section of the license amendment application, as-found leak checks were performed during four outages, and none of the checks revealed a significant leak. Therefore, the plant-specific data indicated 0 failures over approximately a 72-month interval (four 18-month periods). Generic data was reviewed but no appropriate data was determined to be available. Therefore, the failure rate was calculated by means of a Bayesian update, assuming a non-informative prior distribution, as described in NUREG/CR-2300, "PRA Procedures Guide," January 1983.

NRC Request for Information:

3. If the test is performed while the plant is at power: (1) how long does the test last and what percentage of this time could the valve housing structure be considered not leak tight (e.g., during test preparations), and (2) can the test increase the likelihood of a plant trip or affect the plant's mitigation capability?

DBNPS Response:

The surveillance test is not performed at power. It is performed prior to entry into Mode 4 during heatup coming out of a refueling outage. The surveillance test is not performed at power since the test draws a vacuum on the watertight enclosure via its pressure equalization vent. If a loss of coolant accident (LOCA) were to occur during a test, with the plant at power, the pressure external to the enclosure, due to the LOCA, combined with the vacuum inside the enclosure, would potentially cause the decay heat valve pit cover to fail due to loads, and flood the valve operators.

NRC Request for Information:

4. What is the test acceptance criteria for leakage?

DBNPS Response:

The decay heat valve pit is evacuated to 12 in-Hg at the start of the test, and the vacuum decay is then observed over 4-hour period. The test acceptance criterion requires a minimum vacuum of 6.1 in-Hg at the end of the 4-hour period. The 12 in-Hg vacuum creates a differential pressure for the enclosure equivalent to that created for the maximum flood in containment. The 6.1 in-Hg vacuum is a calculated value based on the following assumptions for the flooding scenario:

- A single leakage area at the lowest elevation of the decay heat valve pit
- Differential pressure on the leakage area as a result of maximum containment flood
- A water level in the decay heat valve pit of 6" prior to the containment flooding
- Water at the lowest elevation seam of the motor operator cover resulting in a disabling flood of the motor operator

For the test, constant temperature of the fluid in the decay heat valve pit during testing is assumed. If the 6.1 in-Hg vacuum is not achieved, the surveillance test allows Engineering to evaluate acceptance of the test based on containment and decay heat valve pit temperatures recorded during the test.

NRC Request for Information:

5. Evaluate, or justify a screening value, of the aging rate of the valve housing structure's leakage integrity during normal operation.

DBNPS Response:

As discussed on pages 8 and 9 of Enclosure 1 Attachment 1 of the license amendment application, the materials that comprise the sealed structure have been verified by testing not to degrade when subject to containment operational conditions. Therefore, no aging rate is assumed for the operating cycle.

NRC Request for Information:

6. Indicate the major changes to the updated PSA model since the individual plant examination had been completed. Does the updated PSA model include the new boron precipitation control (BPC) design? (This is necessary in order for the model to be used for this requested technical specification change application.) Discuss how the valve housing structure is included in the PSA model (i.e., impact of the structure failing). Please provide the updated PSA model's baseline core damage frequency (CDF) and the baseline large early release frequency (LERF) from internal, and if available, from external events. Furthermore, discuss the frequency of PSA model updates and quality assurance practices.

DBNPS Response:

Major Changes Since IPE Completion

The PSA model was changed since the individual plant examination (IPE) to reflect changes that address four major considerations:

- The plant has changed since the PSA was completed to support the IPE;
- Significant additional operating experience has been gained since the time of the IPE;
- Some areas of analysis have been expanded to make the PSA a more useful tool for various risk-based applications; and
- Some of the technical methods used have been improved since the previous PSA was prepared.

With regard to the first of these, it should be noted that since the freeze date for the IPE, the plant has continued to undergo modifications (although not nearly to the extent that was the case in the years prior to the IPE). Procedures and operating and maintenance practices have also undergone revision and refinement. Among the more important changes incorporated into the updated PSA are the following:

- The station blackout diesel generator has been fully implemented. At the time of the analysis for the IPE, this generator had been installed, but some procedures and other supporting details were not yet available.
- The startup feed pump had essentially been abandoned in place in the late 1980s. More recently, provisions have been made to use the pump as an additional line of defense in the event of failure of auxiliary feedwater and the motor-driven feed pump.

- New provisions have been installed to limit the effects of a flood that might originate in the turbine building due to a failure in the condenser circulating water system.

The update also involved an extensive review of more recent operating experience. Various types of records developed over the course of plant operations are examined to help in developing databases to support quantification of the PSA models. This is a very labor-intensive process, and for the IPE submittal, it was possible to evaluate experience at the DBNPS only through the sixth refueling outage (June 1990). The current update has accounted for experience since 1990. Changes in various practices in the late 1980s and into the 1990s have led to significantly increased attention to areas such as plant maintenance that have, in turn, yielded substantial improvements in plant performance. Among the areas in which these improvements have had a direct impact on the reliability data bases for the PSA are the following:

- The frequency of plant trips, and especially of trips due to or coincident with a loss of main feedwater, is much lower than it was for the period prior to 1990.
- Efforts to address problems with motor-operated valves have resulted in an improvement in their reliability by a factor of almost 5.
- Most other components for which operating experience was collected have experienced lower failure rates, although generally not as dramatic as that for motor-operated valves.

The modeling effort has been expanded in many areas to provide for more accurate reflection of actual operating practices, to allow for more effective applications to be made, and to correct errors uncovered since the IPE was submitted. Some of the modeling improvements include the following:

- The overall framework for the models for service water and component cooling water was revised to facilitate the consideration of restoring the systems by using the spare pumps.
- The event tree for sequences initiated by a steam generator tube rupture was modified both to reflect changes in the emergency procedure and to simplify the quantification process.
- The potential for flooding in the turbine-building basement due to failures in the condenser circulating water system was evaluated in detail. During the previous PSA, an assumption had been made that such floods could not pose a threat to core cooling, and no detailed analysis had been performed.
- The treatment of other internal floods was integrated more fully into the basic plant models than had been the case for the IPE.

Finally, the methods used for various parts of the PSA have improved since the IPE was prepared. The improvements employed during the PSA update included the following:

- The basic computer code package used to aid in the development and solution of the fault trees has been improved in many ways. The improved package supports easier development and modification of the fault trees, and has several tools to automate portions of the quantification process that previously involved tedious and time-consuming effort by the analysts.
- New tools have been used to perform analyses of the reliability data. These tools make it more efficient to document clearly the sources and treatment of the data, and to revise failure rates or other parameters as additional raw data is collected.
- The methods for human reliability analysis have been revised to address areas that were considered to be potential shortcomings in the PSA for the IPE. The tools for performing the analyses have also been improved somewhat, freeing the analysts to focus more on the potentially important elements of these evaluations.

Collectively, these changes have produced a PSA model for the DBNPS that is more representative of the plant as it currently operates; that can be exercised more effectively and reliably for a variety of intended applications; and that is much more thoroughly documented.

Boric Acid Precipitation Control Model

The DBNPS has developed a detailed model for the new boric acid precipitation control (BPC) design that provides a bounding analysis of the core damage frequency (CDF) and the large early release frequency (LERF) associated with the failure of post-LOCA methods of BPC. A summary of the analysis was previously provided to the NRC in support of a request for exemption from 10 CFR 50, Appendix K, for BPC methodology, DBNPS letter dated March 15, 2000 (DBNPS Serial Number 2633), as supplemented by letter dated April 3, 2000 (DBNPS Serial Number 2652). The results of the NRC review of the analysis are provided in the safety evaluation issued for the exemption, NRC letter dated May 5, 2000 (DBNPS Log Number 5659).

The BPC model has been used to analyze specific issues but is not normally included in the baseline PSA model. This is due to the negligible contribution to CDF from failure of BPC, and the uncertainty in the model. The largest source of uncertainty involves the capability of passive means of BPC. For this analysis, conservative assumptions were made, including no crediting of passive means for BPC. The following are the significant conservative assumptions that were made to provide a bounding analysis:

- It was assumed that if the active methods of BPC fail, the result will be core damage due to inadequate core cooling. No credit was taken for passive BPC methods mitigating precipitation in the core if the active methods fail after or during attempts to initiate them.

- It was assumed that active BPC is needed five hours after the event initiation. The primary dilution method must be initiated one hour prior to the time that dilution is needed, to allow time for the pressurizer to fill. Therefore, flow through the primary BPC dilution flow path was assumed to be required within 4 hours and flow through the alternate flow path was assumed to be required within 5 hours. These times are minimums based on the BPC analysis performed by Framatome Technologies Inc., which also include significant conservatisms. Therefore, this assumption represents a significant source of conservatism in the human reliability analysis performed for this calculation. A more precise method would have been to use a range of initiating event frequencies with corresponding response times. However, this approach would have added more complexity to the analysis and was judged to be not required for a bounding type calculation.

- It was assumed that a method of active BPC is required for all large or medium LOCAs. Not all of the medium LOCAs, as defined in the PSA, would be in the break size range that requires boric acid dilution flow. Therefore, including all medium LOCAs in the initiating event frequency contributed to a conservative result.

- Boric acid dilution is only required for breaks lower than the 573 foot elevation in the cold leg reactor coolant pump discharge piping. Therefore, it is assumed that 25% of the large and medium LOCA frequency involves breaks in locations that would require boric acid dilution.

The decay heat valve watertight enclosure is included in the PSA model in the logic as an event that represents leakage into the decay heat valve pit after a LOCA. Leakage into the decay heat valve watertight enclosure is assumed to fail the backup method of BPC.

PSA Baseline Results

The updated PSA baseline core damage frequency (CDF) is 1.6E-5 / year from internal events. The baseline large early release frequency (LERF) is 7.3E-8 / year from internal events.

PSA Maintenance and Quality Assurance Practices

In order to maintain the PSA as close as possible to the current plant configuration, a PSA review is included in the process for plant modifications. If changes occur that could affect the PSA model, the change is logged in a database and evaluated for impact on the PSA results. Non-

significant changes are collected until the next major update, which is performed at a minimum of once per operating cycle. If a change is determined to be risk significant, the PSA is revised as soon as possible. Non-incorporated changes are available in the database and are reviewed prior to performing an evaluation to ensure they will not impact the results. Plant-specific reliability data is collected and updated once per operating cycle. Since the IPE update was completed in October 1999, the PSA has been updated three times.

All PSA inputs and models, including system models, reliability data, accident sequence analysis and human reliability analysis, are required to be performed and checked by PSA engineers. Both the performer and checker are required to sign each revision of the documentation. Evaluations performed using the PSA are conducted using the calculation process.

PSA maintenance and update is performed in accordance with written guidelines that control the frequency of updates and the quality assurance practices. The PSA maintenance and documentation meets the appropriate attributes of a 10 CFR 50 Appendix B program with the exception of the requirement for audits. The process was assessed in depth during the Peer Review performed in November 1999, and all the observations made during that review have been resolved.

NRC Request for Information:

7. Discuss the sequences contributing to increase in CDF and LERF provided in the submittal.

DBNPS Response:

Results of the Level 1 Analysis

Examination of the cutsets derived from analysis of the BPC model reveals that ECCS room cooler failures contribute about 65% of the calculated CDF. Failure by the operators to align BPC comprises the next largest portion of the CDF. Single electrical component failures are a small contributor due to the lower probabilities of these events coincident with a large-break LOCA.

Sequences involving failure of the watertight enclosure involve loss of the backup method of BPC due to the enclosure failure. Failure of the watertight enclosure is a small contributor to the overall BPC core damage frequency contribution.

Results of the Level 2 Analysis

The LERF contribution from failure of BPC is very small, as would be expected, for the following reasons. Generally the LERF contribution from accidents that result in reactor coolant

system depressurization is small because induced tube ruptures and core debris dispersal beyond the cavity are both precluded at low RCS pressure. Additionally, for failure of boric acid dilution sequences, the success of low pressure injection and low pressure recirculation is implied. Therefore, at a minimum, the BWST would be injected and heat removal by low pressure recirculation would be available.

NRC Request for Information:

8. Provide the following risk measures using your updated PSA model:

   (a). The CDF assuming that the housing structure never fails (ie, has a probability of zero of failing).

   (b). The CDF assuming the housing structure always fails (ie, has a probability of one of failing).

The second CDF quantity should be calculated for two cases: (1) valves DH 11&12 are in the closed position as would be expected when the primary BPC method is being used, and (2) valves DH 11&12 are in the open position as would be expected when the backup method is being used.

DBNPS Response:

The base CDF including the failure of BPC sequence is 1.592E-5 / year. If the housing structure is assumed to never fail the CDF is reduced to 1.590E-5 / year. The F-V importance measure for the housing structure is 1.163E-3.

If the housing structure always fails the CDF increases to 1.608E-5 / year. This corresponds to a Risk Achievement Worth (RAW) of 1.01. The position of DH-11 and DH-12 are not considered in the evaluation. In all cases it is assumed that DH-11 and DH-12 are not open and the alternate BPC method will fail if leakage occurs.

Both the importance measures for the housing structure indicate a low risk significance for this component. Additionally, in both cases reported above, the results are based on a conservative bounding analysis due to the assumptions described in the response to Question 6. Therefore, the actual importance measures would be even less significant than the results calculated above.

NRC Request for Information:

9. Are the post-loss of coolant accident housing submergence conditions (i.e., level, etc.) discussed in the submittal bounding? If not, consider the significance of the different condition(s).

DBNPS Response:

See the above response to Question 4. Each of the assumptions used for determination of the test acceptance criteria is a bounding assumption.

NRC Request for Information:

10. How would the risk insights associated with extending the surveillance testing interval be used at the plant?

DBNPS Response:

Risk insights have been and will continue to be considered when performing maintenance that could cause either the primary or backup boron precipitation control (BPC) method to become unavailable. If one BPC method is planned to not be available for maintenance or testing, the alternate flow path is required to be available to ensure the BPC function is maintained.

As stated in the response to Question 11, BPC is listed as a maintenance rule function for the decay heat system. Therefore, failures of the housing would constitute a functional failure and the risk implications would be evaluated using the PSA.

NRC Request for Information:

11. Is the valve housing structure included in your maintenance rule program? If so, please provide the maintenance rule goal established for it.

DBNPS Response:

The valve housing structure was added into the Maintenance Rule Program, as part of Maintenance Rule Function 6, "Provide backup to the primary Post-LOCA Boron Precipitation Control flowpath for Train #2 of HPI."

The maintenance rule reliability goal is the same as the rest of the system: "Less than or equal to one functional failure every three cycles. No repetitive functional failures."

NRC Request for Information:

12. Discuss work controls which protect the valve housing structure leakage integrity.

DBNPS Response:

The decay heat valve pit is sealed and tested near the end of each refueling outage. It is in a remote area of the lower elevation of the containment and is not a "through traffic" area. The area has a gate that although not locked, is generally designated as a High Contamination Area. This deters much of the casual traffic in the area. At the end of the outage, since maintenance activities in the area are completed, any resultant traffic due to these activities is practically eliminated. Following sealing and surveillance testing of the decay heat valve pit there is only minimal passage consisting of Operations personnel performing valve lineups and verifications, and RP Technicians to remove signs that are normally posted in the back of the area. This minimal level of traffic following decay heat valve pit sealing and testing is not expected to affect the sealing surfaces of the decay heat valve pit.

NRC Request for Information:

13. Describe your configuration risk management program (e.g., procedures, risk assessment tools, etc.).

DBNPS Response:

The requirements for the configuration risk management program (CRMP) used at the DBNPS are described in DBNPS procedure NG-DB-00001, "On-Line Risk Management." The program includes the following elements:

- The risk for all plant configurations is evaluated using a risk meter that solves the DBNPS PSA for each configuration.
- The risk monitor is used to evaluate the risk for maintenance configurations prior to the start of each work week. Based on the assessed risk level, one of four risk levels is assigned to each maintenance configuration, and risk management actions are assigned per NG-DB-00001.
- Emergent activities are assessed by the on-shift operations staff in the event of an unplanned activity.
- Actions to be taken in the event that an emergent activity causes the unplanned entry into a higher risk level are included in NG-DB-00001.
- Guidance for considering other risk contributors that may not be directly quantifiable using the risk monitor and PSA is provided in NG-DB-00001.

Docket Number 50-346
License Number NPF-3
Serial Number 2727
Enclosure 2

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. PLEASE NOTIFY THE MANAGER – REGULATORY AFFAIRS (419-321-8450) AT THE DBNPS OF ANY QUESTIONS REGARDING THIS DOCUMENT OR ANY ASSOCIATED REGULATORY COMMITMENTS.

COMMITMENTS	DUE DATE
None	N/A