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Subject: Supplemental Information Regarding the Request for Exemption from 10 CFR 50, Appendix K, for Boric Acid Precipitation Control Methodology (TAC No. MA7831)

Ladies and Gentlemen:

On March 15, 2000, FirstEnergy (FE) submitted a request for an exemption from Title 10 of the Code of Federal Regulations (CFR), Section 50, Appendix K, "ECCS Evaluation Models," for the Davis-Besse Nuclear Power Station (DBNPS) boric acid precipitation control (BPC) methodology. This letter provides supplemental information relative to the March 15, 2000 submittal (DBNPS Serial Number 2633).

A summary of the risk evaluation supporting the exemption request was included as Attachment 2, "Summary of Boric Acid Precipitation Control Calculation C-NSA-099.16-26," of the March 15 submittal. In response to a conference call with the NRC staff, the calculation has been revised to address the probability of failure of the backup BPC method due to insufficient nozzle gap flow, and to address the effect on the core damage frequency (CDF) from improper BPC valve lineups. The enclosure summarizes the results of the calculation revision.

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As discussed with the NRC staff, the following specific wording of the exemption being requested by the March 15 submittal is proposed by the DBNPS:

> FirstEnergy, with respect to the Davis-Besse Nuclear Power Station, is exempt from the single failure criterion requirement of 10 CFR 50, Appendix K, Section I.D.1, with respect to:
>
> (1) Simultaneous failure of both the primary auxiliary spray method and the backup decay heat removal drop line method of controlling boron concentration due to failure of an emergency core cooling component that results in inability to initiate, or continue to operate, an active means of controlling core boron concentration, and
>
> (2) Not establishing that the backup decay heat removal drop line method of controlling boron concentration is otherwise in compliance with Appendix K and 10 CFR 50.46(b)(5) requirements. Specifically, when establishing that boron precipitation will not occur in the decay heat removal system cooler, the Davis-Besse Nuclear Power Station credited flow through hot leg nozzle gaps and did not include all of the specific conservatisms required by Appendix K.

Should you have any questions or require additional information, please contact Mr. James L. Freels, Manager - Regulatory Affairs, at (419) 321-8466.

Very truly yours,

MKL

Enclosures

cc: J. E. Dyer, Regional Administrator, NRC Region III
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K. S. Zellers, NRC Region III, DB-1 Senior Resident Inspector
Utility Radiological Safety Board

Summary of Revisions to Boric Acid Precipitation Control Calculation C-NSA-099.16-26

Failure of the Backup Method of BPC Due to Insufficient Nozzle Gap Flow

As a part of the supporting analysis of the backup boric acid precipitation control (BPC) methodology, Framatome Technologies, Inc. (FTI) determined that the boric acid concentration in the Decay Heat Removal (DHR) drop line coming from the core could exceed the solubility limits for the temperature in the DHR cooler if credit for the nozzle gaps is not taken. Analysis was performed to demonstrate the boric acid precipitation in the DHR cooler will not occur. For this analysis it was necessary to credit flow through the reactor vessel outlet nozzle gaps for the time period until the backup BPC method would be established.

To evaluate the risk associated with the crediting of gap flow the assumption was made for this calculation that the probability of failure of gap flow will be 0.1. Additionally it was assumed that in the event of boron precipitation in the cooler the backup BPC method will fail and core damage will occur.

The fault tree developed for BPC was modified to reflect the probability that the backup method could fail due to boron precipitation in the DHR coolers. A new event GAPSFAIL was created and a probability of 0.1 was assumed for this event.

The re-quantification of the model using the modified fault tree resulted in a core damage frequency of 1.3×10^{-7} per year.

The fault tree for BPC, modeled with the additional line from High Pressure Injection (HPI) pump 1-1 was also modified. Re-quantification using the modified fault tree resulted in a CDF of 2.1×10^{-8} / year.

Conclusions

The overall core damage frequency, assuming a 10% probability that the nozzle gaps will be insufficient to support the initiation of the backup BPC method, is 1.3×10^{-7} / year. This is approximately a 20% increase over the CDF calculated assuming no nozzle gap failure. Based on either of the guidelines listed in RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," dated July, 1998, this would not represent a risk significant contribution to the total core damage frequency.

Evaluation of Improper Lineup of Primary Boron Dilution Flow Path

The primary BPC method will involve the addition of two new valves to the HPI system that will pressurize the auxiliary spray system. During power operation the two closed isolation valves, DH200 and DH201, will isolate the DHR system from the higher pressures in the HPI system. If these valves are left open, the result could be a diversion of HPI flow and the failure of the Low Pressure Injection (LPI) train. Therefore, this analysis will calculate the probability of this latent human error and the effect on the overall CDF.

Evaluation of Pre-Initiator Human Interactions

The process used in the Davis-Besse PSA to evaluate pre-initiator (type A) human interaction is described in section 3.1 of the HRA notebook. The PSA uses a form of the technique for Human Error Rate Prediction (TERP) developed for the Accident Sequence Evaluation Program (ASEP) and published in NUREG/CR-4772, *Accidents Sequence Evaluation Program Human Reliability Analysis Procedure.*

The ASEP methodology does not provide explicit treatment for a case corresponding to the locked-valve procedure at Davis-Besse. Many of the valves at Davis-Besse that are required to remain in a particular position at all times other than for certain test and maintenance activities are locked to prevent inadvertent realignment. These valves are controlled by a separate procedure, in which the permission of the shift supervisor must first be obtained before their position can be changed, and in which locked valves that are out of their normal positions are tracked through use of a log. The level of control and verification goes beyond that represented by the independent verification as described in the ASEP methodology, but would seem to fall short of the level of verification that would be afforded by a positive test. The recovery factors suggested by the ASEP methodology for these two levels of verification are 0.1 and 0.01, respectively. For locked valves, an intermediate value of 0.03 was therefore applied as a recovery factor for restoration errors. This recovery value is multiplied by the basic human event probability of 0.03, given in the ASEP methodology to calculate the probability of the human error. For this case assuming a locked valve the probability is 9×10^{-4}.

As described in NUREG/CR-1278, *Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications*, the unavailability of a component due to a human interaction can be expressed as follows:

$$U = \frac{pd}{T}$$

where p = the probability of the unrecovered human error;
d = the average time the error could exist; and
T = the average time between opportunities to make the error.

The average time the error could exist, d, reflects the opportunities to discover the error by test or checking prior to the next time the component would be manipulated. For cases in which the opportunities to uncover the error are uniformly distributed with time (e.g., monthly or quarterly checks), the value of d can be calculated as follows:

$$d = \frac{h(1 - c^{T/h})}{1 - c}$$

where h = the average length of time between checks, and
c = the probability the error will not be detected at the check.

For this calculation the average time between opportunities to make the error (T) was assumed to be 12 months. This is based on the expected times between plant shutdown that will require the realignment of auxiliary spray from the DHR system.

The error could be detected during the quarterly locked valve verification test that is performed in accordance with plant procedure DB-OP-04004, "Locked Valve Verification." Failure to identify a particular mispositioned locked valve was assessed to have a probability of 0.1 in the ASEP methodology. The error could also be detected during the quarterly HPI pump quarterly test, plant procedure DB-SP-03219. For this test the failure to identify improper valve position was assessed to have a probability of 0.01, as suggested by ASEP. Additionally, the lineup error could be detected while using the HPI pumps for filling the core flood tanks. Core flood tank filling is performed approximately every other month using one of the HPI pumps. Although these opportunities to detect the misalignment would justify a shorter interval, the average length of time between checks will be assumed to be 3 months. The probability that the error will not be detected will be assumed to be 0.01 based on the HPI pump testing.

Using the probabilities the total unavailability due to the error can be calculated as follows:

$$\begin{aligned} U &= \frac{p \bullet h(1 - c^{T/h})}{T(1 - c)} \\ &= \frac{(0.0009)(3\,\text{mos.})(1 - 0.01^{12/3})}{(12\,\text{mos.})(1 - 0.01)} \\ &= 2.3x10^{-4} \end{aligned}$$

The probability calculated above applies only for one valve although there are four valves that would normally be manipulated in the line between HPI and LPI. Two valves, DH200 and DH201, would both have to be open in the auxiliary spray line from DHR to inadvertently pressurize the DHR system. However, due to the similarity of location and function these valves will be treated as totally dependent. HP210 and HP209 would also have to be open, and the failure to close DH200 and DH201 would imply some probability that HP210 and

HP209 would be left open. However, this probability will also be neglected and for this evaluation the total probability of the improper lineup will be based on a single valve.

Calculation of the Effect on Core Damage Frequency From Improper BPC Valve Lineups

The failure to close DH200 and DH201 would result in the pressurization of the LPI piping at the discharge of DHR cooler 1-2. Relief valves DH1509 and DH1550 protect the low pressure portion of the DHR system but the capacity of these valves is assumed to be insufficient to prevent the overpressurization of the LPI system. The most extreme consequence if the low pressure portions the decay heat train are pressurized would be a failure of the system integrity. For this evaluation a conservative approach will be taken and it will be assumed that in all cases overpressurization will lead to a failure of the LPI system that is significant enough to cause the loss of both the affected HPI and LPI train. An additional consequence of loss of system integrity could be flooding in the ECCS room from the BWST through the failed portion of the LPI system. However, this source of flooding could be isolated by closing DH7B, and the effect of flooding is effectively encompassed in the evaluation by the assumption that the entire train of HPI and LPI is lost.

The potential for the loss of integrity of the LPI system, initiated by the operation of the HPI pump, would not contribute to the increase in the initiating event frequency for interfacing system LOCAs. On the discharge side the system is protected from RCS pressure by two check valves, DH76 and CF30. The failure of these valves to protect the LPI system from primary pressure is already evaluated in the PSA interfacing system LOCA analysis.

To model the improper lineup of the BPC valves a new human event HHA0BPCL was created and assigned the value of 2.3×10^{-4}. This event was added to the master fault tree for revision 2 of the PSA so that it was an input for failure of both DHR train 2 and HPI train 2.

All supporting files for revision 2 of the Davis-Besse PSA were used to quantify the revised fault tree. The quantification was performed using the sequence specific PRAQUANT file with a screening value of 1E-2 for the probability of HHA0BPCL. After the cut sets were generated the new cut sets were isolated and the probability of HHA0BPCL was adjusted to the calculated value.

Results

The results of the quantification are shown in the following table for several values of HHA0BPCL.

Table 10. CDF Results with Human Event HHA0BPCL

Probability Operators Fail to Close DH201 and DH200	Total Core Damage Frequency	Change in Core Damage Frequency
0.0	1.827 E-5	0
2.3E-4	1.827E-5	5.68E-9
2.3E-3	1.832E-5	5.68E-8
1.0E-2	1.851E-5	2.47E-7

Effect on Large Early Release Frequency (LERF)

Due to the very small increase in core damage frequency the increase in LERF would be insignificant. The loss of integrity in the DHR system could lead to additional release, however, this failure would be expected to occur early in the accident sequence when the RCS pressure is relatively high. Therefore it would be expected that this failure would occur following automatic initiation of HPI, well before the DHR pumps are lined up with suction from the containment sump and BPC is initiated.

Conclusions

The increase in CDF due to the improper alignment of BPC valves was negligible when applying a conservative human failure probability. Even for a screening value of 1.0E-2 the increase in CDF was less than 1.5%.

The relative insensitivity of CDF to this latent human failure is not unexpected. This failure impacts the plant systems in a manner similar to a number of failures already included in the PSA. Since the affected equipment is all in one ECCS train, it is already subject to flooding in the ECCS room, loss of ECCS room cooling, loss of a vital 4160 bus, loss of a train of component cooling water, and loss of a train of service water. The probabilities of some of these failures are one to two orders of magnitude greater than the probability of the improper BPC lineup.

Docket Number 50-346
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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 (419-321-8466) AT THE DBNPS OF ANY QUESTIONS REGARDING THIS DOCUMENT OR ANY ASSOCIATED REGULATORY COMMITMENTS.

COMMITMENTS	**DUE DATE**
None	N/A