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Docket Number 50-346

License Number NPF-3

Serial Number 3317

April 20, 2007

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555-0001

Subject: Davis-Besse Nuclear Power Station, Supplemental Information Concerning Generic Letter (GL) 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions" (TAC No. M96803)

Ladies and Gentlemen:

Generic Letter (GL) 96-06, issued by the NRC on September 30, 1996, requested in part that licensees evaluate cooling water systems serving containment air coolers to assure that they are not vulnerable to waterhammer and two-phase flow. In response, the FirstEnergy Nuclear Operating Company (FENOC) provided an assessment of the GL 96-06 issues for the Davis-Besse Nuclear Power Station (DBNPS) in letters dated January 28, 1997; February 28, 1997; July 28, 1997; September 30, 1997; and July 25, 2006. FENOC's responses identified a large break Loss of Coolant Accident (LOCA) with a simultaneous Loss of Offsite Power (LOOP) as the bounding event for consideration.

By letter dated April 3, 2002, to the EPRI Waterhammer Project Utility Advisory Group, the NRC accepted EPRI Report TR-113594, "Resolution of Generic Letter 96-06 Waterhammer Issues," Volumes 1 and 2. The staff found this report acceptable for performing evaluations addressing GL 96-06 waterhammer concerns to the extent specified and within the limitations delineated in the EPRI report and in the associated NRC safety evaluation. The NRC safety evaluation stated that licensees who choose to use the methodology in EPRI TR-113594, Volumes 1 and 2, for addressing the GL 96-06 waterhammer issue, may do so by supplementing their response to include, among other items, *"Certification that the EPRI methodology, including clarifications, was properly applied, and that plant-specific risk considerations are consistent with the risk perspective that was provided in the EPRI letter dated February 1, 2002. If the uncushioned velocity and pressure are more than 40 percent greater than the cushioned values, also certify that the pipe failure probability remains bounding..."*

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A draft request for additional information regarding the above issue was provided by the NRC staff via facsimile transmission on February 8, 2007. Additional clarification of the desired response was obtained by teleconference between NRC and FENOC staff on March 21, 2007. Supplemental information in response to the above communications is provided in Attachment 1.

As identified in Attachment 2, there are no commitments contained in this letter or its attachments. If there are any questions or if additional information is required, please contact Mr. Henry L. Hegrat, Supervisor – FENOC Fleet Licensing, at (330) 374-3114.

The statements contained in this submittal, including its associated enclosures are true and correct to the best of my knowledge and belief. I am authorized by the FirstEnergy Nuclear Operating Company to make this submittal. I declare under penalty of perjury that the foregoing is true and correct.

Executed on: April 20, 2007

By: Mark B. Bezilla
Mark B. Bezilla, Vice President-Nuclear

TSC

Attachments:

1. Supplemental Information Concerning Generic Letter (GL) 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions"
2. Commitment List

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

Supplemental Information Concerning Generic Letter (GL) 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions"

By facsimile transmission on February 8, 2007, the Nuclear Regulatory Commission (NRC) staff provided the following draft request for additional information to the FirstEnergy Nuclear Operating Company (FENOC):

> By letter dated July 25, 2006 (Agencywide Documents Access and Management System Accession No. ML062090103), FirstEnergy Nuclear Operating Company (FENOC, the licensee) submitted a response to the Nuclear Regulatory Commission (NRC) staff's request for additional information concerning Generic Letter (GL) 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions".
>
> In order for the NRC staff to complete its review of the licensee's response to the generic letter, the NRC staff requests that the licensee provide a response to the following question:
>
> 1. Licensees who use the analytical methodology that was developed by the Electric Power Research Institute (EPRI), as documented in EPRI Technical Report (TR) 113594, Volumes 1 and 2 (ML003779585 and ML003781044), are requested to address the specific items that are detailed in Section 3.3, "Licensee Responses to GL 96-06," of the NRC Safety Evaluation that provided NRC approval for using the EPRI methodology and was issued on April 3, 2002 (ML020940132). In particular, a risk assessment similar to the one that was documented in the EPRI letter dated February 1, 2002 (ML020390063) is requested. If the uncushioned velocity and pressure are more than 40 percent greater than the cushioned values, certification that the pipe failure probability assumption remains bounding is also needed (as applicable). The EPRI letter and the NRC Safety Evaluation are included as Appendices to EPRI TR-113594.

Applicable portions of the above request are repeated below in bold-face type, followed by the FENOC response to the NRC question.

<u>NRC Question</u>: **In particular, a risk assessment similar to the one that was documented in the EPRI letter dated February 1, 2002 (ML020390063), is requested.**

<u>Response</u>: As discussed in the July 25, 2006 letter to the NRC, the FirstEnergy Nuclear Operating Company performed new waterhammer analyses using methodologies consistent with EPRI TR-1006456, a successor document to EPRI TR-113594. The plant-specific risk considerations are consistent with the EPRI risk considerations identified in EPRI TR-1006456.

Specific analyses are provided below to address the nine initiating event considerations identified in the EPRI letter dated February 1, 2002 (ML020390063).

Event Consideration 1. Occurrence of a LOCA or MSLB

> The probabilities of occurrence of LOCA and MSLB events are provided in NUREG/CR-5750. From that document, the mean frequency of occurrence of a large LOCA is $5 \cdot 10^{-6}$/year, a medium LOCA is $4 \cdot 10^{-5}$/year, and a MSLB is $1 \cdot 10^{-3}$/year. The LOCA probabilities are represented in NUREG/CR-5750 as "reasonable but conservative" estimates of the frequency of occurrence.

Davis-Besse uses the same values for LOCA frequencies as indicated in the EPRI response, that is, 5E-06/year for a large LOCA and 4E-05/year for a medium LOCA. Davis-Besse uses two events to represent steam line breaks in the Probabilistic Risk Assessment (PRA), one for each steam generator. Each of these events has a frequency of 5.7E-04/year, for a total of 1.14E-03/year. This is slightly higher than the number used in the EPRI response. However, the Davis-Besse events include both main feed and main steam line breaks in the derivation of the events. An inspection of the industry data used to derive the Davis-Besse frequency (from 1/75 through 12/95, 1760 critical years) shows that there are a total of two applicable failures, both of which are feed water breaks. Calculating the main steam line break frequency from this data, using only PWR years (1153 critical years) as in the NUREG/CR-5750 report, gives a total main steam line break frequency of 4.34E-04/year (or 2.17E-04/year per steam generator). Therefore, the EPRI risk information for this portion of the response is applicable to Davis-Besse.

Event Consideration 2. Occurrence of a LOOP following a LOCA or MSLB

> Studies provided in NUREG/CR-6538 and subsequent NRC work indicate that the dependent probability of a Loss of Offsite Power event following a LOCA event is approximately $1.4 \cdot 10^{-2}$/demand.

The Davis-Besse PRA does not specifically model a LOCA or MSLB induced LOOP. NUREG/CR-6538 calculated a LOOP following a LOCA by adding together frequencies for LOOP following a reactor trip and LOOP induced by ECCS actuation. The Davis-Besse PRA model does contain an event for a LOOP given a reactor trip. Based on industry and plant data from 1975-2005, the frequency for a loss of offsite power after a trip is 7.59E-04/demand. This is lower than the NUREG/CR-6538 results for the frequency for a loss of offsite power after a trip. Davis-Besse does not specifically model ECCS actuation induced LOOP events. However, an inspection of LOOP data for Davis-Besse indicates there have been no ECCS induced LOOP events at Davis-Besse. Therefore, if Davis-Besse were to model ECCS induced LOOP events, it would be necessary to utilize industry data in the same way as NUREG/CR-6538. Given that there have been zero such events at

Davis-Besse, performing a Bayesian update of the NUREG failure data with Davis-Besse plant specific data would result in a lower frequency of LOOP following an ECCS actuation. Therefore, the NUREG frequency remains bounding.

Event Consideration 3. Occurrence of a Simultaneous LOCA/LOOP Event

> The required design basis consideration is for the simultaneous occurrence of a LOCA or MSLB and a LOOP. The frequency of the combined event depends upon the probability of the LOCA and the MSLB and the dependent probability of the LOOP given that the LOCA has occurred. Using the values defined in each of the NUREGs referenced above gives a probability of the combined event on the order of $1.5 \cdot 10^{-5}$/year. For our purposes here, the value of probability of the design basis event (LOCA or MSLB occurring simultaneously with a LOOP) will be taken as $1 \cdot 10^{-5}$/year. With best estimate probabilities, this event likelihood of occurrence could be expected to be even lower.

Using the bounding NUREG/CR-6538 frequency for LOOP following a LOCA (1.4E-02/demand) and the Davis-Besse main steam line break frequency (4.34E-04/year), which is greater than, and therefore more conservative than using the Davis-Besse LOCA frequencies, gives a frequency of the combined event of approximately 6.1E-06/year. This is less than that determined by EPRI, so the EPRI report bounds the Davis-Besse data.

Event Consideration 4. Void Formation

> If we have a LOCA/LOOP event, a void will form in an open loop plant with certainty. In a closed loop plant, void formation will depend on the specific plant characteristics and a void may or may not form. If a void does not form, a waterhammer will not occur.

The Davis Besse plant design includes an open loop containment air cooling system with raised loops on both the cooler supply and discharge. This configuration is conducive to vapor void formation in the high points of the upper loops during a LOCA/LOOP event. However, based on the modifications made in the Davis Besse system, the FENOC analysis demonstrates that the system pressure at the high points remains above the fluid vapor pressure at all times precluding any vapor void formation at these critical locations.

Due to the elevated containment building temperature during the LOCA/LOOP event and the lack of cooling water flow, steam void formation in the cooling coils themselves is inevitable. The analysis shows that the steam void remains within the cooling coils and waterbox, however, since the time and the location of void collapse are not explicitly determined using the EPRI approach it could not be determined if the final void collapse would occur in the coils/waterbox or in the immediate downstream piping. Since the loads on the piping are dependent on the void collapse location, the FENOC analysis conservatively assumes both conditions are possible and evaluates the loads for each case, which is in agreement with the EPRI approach.

Event Consideration 5. Pump Restart

> The pumps will restart with certainty and the velocity of the fluid in the pipe, immediately prior to closing the void, will be defined by the pressure in the void, the piping geometry, and the pump characteristics. This uncushioned closure velocity can be reliably calculated. This velocity will not be higher than the rate at which the pumps, once restarted, can pump water. The calculation of the water velocity prior to closure is a plant specific analysis that can be conservatively performed.

For the current plant design subjected to the LOOP/LOCA conditions, the Service Water control valve logic is in place and has been analyzed to demonstrate that waterhammer is acceptably mitigated by the slow fill control valve logic. However, the FENOC analysis conservatively assumes that the Service Water control valve logic has been removed such that the valve would be full open and the maximum fluid velocity would be evaluated.

The pump start transient analysis was performed using detailed pump characteristics and the piping hydraulic resistance to calculate the transient flow in the pipe. Using this analysis, the water column velocity at the time of the column closure was determined. This velocity was used to calculate the peak waterhammer pressure. The column closure velocity was found to be less than the velocity upstream of the coolers due to increase in the pipe area and hydraulic resistance. The pressure rise when the two water columns meet is calculated using the Joukowski equation based on the differential velocity that has developed. This methodology is in agreement with the EPRI approach.

Event Consideration 6. Column Closure

> The water columns will refill the void and the velocity at closure cannot be larger than the largest calculated differential velocity for the upstream and downstream water columns.

The column closure velocity is the relative velocity between the velocities of the water columns upstream and downstream of the void. The FENOC analysis simulated the motions of the upstream and downstream columns to obtain the conditions just before the void collapse to obtain appropriate column closure velocity. This methodology is in agreement with the EPRI approach.

Event Consideration 7. Maximum Waterhammer Pressure

> An upper bound on the water hammer pressure can be calculated by the Joukowski relationship with the uncushioned closure velocity that corresponds to the pipe in which the closure will occur. The waterhammer pressure cannot be larger. With a probability of one, the waterhammer pressure will be equal to or less than the Joukowski pressure. The actual waterhammer pressure that will occur is stochastic and will have a wide variation. This variation is due to variations in the void distribution in the system immediately prior to final closure. This variation appears in all the integral system level experiments. The variation in the test data has been reviewed and, in the velocity range of interest, it varies from 50% to 100% of the maximum (for example, in the Configuration 2a tests, at a velocity of approximately 25 feet per second, the maximum pressure measured from the test was approximately 400 psig, the minimum pressure was approximately 200 psig, the Joukowski pressure for this velocity of closure is 775 psig -- see Figure 10-9 in the TBR). The variation in the test data that has been seen as part of the EPRI project is typical of many other waterhammer tests that have been previously performed and it indicates that it is unlikely that the Joukowski pressure will be attained given the scatter in the results of measured waterhammers compared to those predicted. It is assumed in the EPRI reports that the largest (Joukowski) pressure is attained for the calculated cushioned velocity, although it is very likely that the pressure less than the maximum seen in a test will be experienced.

The FENOC transient analysis did not take any credit for mitigation of the waterhammer pressure pulse due to presence of non-condensables and steam. Therefore, the peak pressure calculated by the analysis for each scenario is the same peak pressure as the one predicted by the Joukowski equation. Since EPRI has determined that the actual peak pressures are typically significantly less than the idealized peak pressure due to cushioning and void distribution, the results of the FENOC analysis are conservative.

Event Consideration 8. Cushioned Waterhammer

> With the cushioning that is predicted to occur due to gas and steam, the cushioned velocity will be on the order of approximately 30% to 40% lower than the maximum velocity (see User's Manual appendix - this depends on many parameters, including the amount of gas and steam). For closed loop plants this value may be only 10-15%. The waterhammer that is predicted, then, will be on the order of 30% to 40% less than the pressure calculated by Joukowski, as the relationship between pressure and velocity is linear. If the cushioning did not occur, the waterhammer pressure and the stresses in the piping would be equivalent to the uncushioned waterhammer that would not have the 30% to 40% adjustment. There are two ways to consider the impact of this potentially higher stress:

- The first is to consider actual plant performance. The occurrence of the waterhammer following a LOOP event - either simulated in a test or real - is known to have occurred many times in the industry. The waterhammer following a LOOP-only event is not cushioned by gas and steam in the void. The total number of occurrences of LOOP-only events are estimated to be on the order of at least several hundred, based on a review of the available plant data. These occurrences have all been in open loop plants and are more severe than a waterhammer that would occur following a LOOP/LOCA event. Without any cushioning, the LOOP waterhammer is more severe than that following a LOOP/LOCA. No piping failures have occurred in any of these events. This would indicate that the probability of failure for a more severe waterhammer (an uncushioned waterhammer) is of the order of 10^{-2} or lower.

- The other method is to take the ASME Code limits and to calculate the probability of failure if the code limits were to be exceeded by approximately 40%. For the purpose of this evaluation, it will be assumed that the piping system is designed so that all the ASME code stresses in the piping were at the faulted condition limit when the cushioned waterhammer occurred – that is, the EPRI methodology is used and that the pipe was designed up to the code acceptable limit for that load. To determine probability of failure, an assumed stress distribution is used around a stress that is 40% larger than the faulted allowable ($2.4S_h$) and compared to the actual tested material strengths for A106-Gr B piping. Based on the actual margins available in the ASME code (see NUREG/CR-2137), the probability of the stress exceeding the strength can be shown to be on the order of 10^{-4} or less.

For the purpose of continuing the "event progression", a probability of failure in the pipe if the cushioned waterhammer were exceeded will be taken to be on the order of 10^{-2}. It is probably much less likely.

The FENOC analysis ignores the effects of any air released during boiling. Since there will be no decrease in the sonic velocity due to the presence of a gas volume, the predicted pressure and loads on the piping are conservative.

Event Consideration 9. Likelihood of an Unacceptable Event

Given the low probability (10^{-5}/year) of the initiating events and the low probability (10^{-2}) of piping failure, the use of the methodology in the User's Manual and the Technical Basis Report will lead to a likelihood of an unacceptable event that is on the order of 10^{-7}. Again, for the purposes of this evaluation, the "unacceptable event" following a LOOP/LOCA event is taken as a breach of the service water system pressure boundary. The probability of 10^{-7} for this event is below the threshold for significant risk to the plant. Use of the methods in the User's Manual, therefore, will not compromise the safety of the plant for the systems within the bound provided in the User's Manual and Technical Basis Report. The methodology should be accepted as recommended in the report.

The methods proposed in the EPRI Technical Basis Report use the physics of gas compression to calculate a reduced closure velocity and waterhammer magnitude. The FENOC analysis ignores that void formations will have non-condensable gases and steam, and that some cushioning will occur. Instead, the FENOC analysis conservatively uses the Joukowski relationship with the uncushioned velocity. Based on the above discussion, the methodology used in the FENOC analysis does not lead to an unacceptable plant risk following a LOOP/LOCA event.

NRC Question: If the uncushioned velocity and pressure are more than 40 percent greater than the cushioned values, certification that the pipe failure probability assumption remains bounding is also needed (as applicable).

Response: Several modifications were developed and implemented to address the issue of void formation in the Containment Air Cooler (CAC) upper loops, focusing on the elimination of the potential for waterhammer. The FENOC analyses show that the modifications will prevent column separation or vapor voiding in high points of the Service Water (SW) CAC piping following a loss of SW pump flow to the CAC during a Loss of Offsite Power (LOOP). This was subsequently validated by confirmatory testing. Although the potential for steam void formation is limited by the modifications in a LOOP/LOCA event, the analysis assumed the steam void collapse could occur in the coolers or in the discharge piping immediately downstream. Air released during boiling is known to have a significant effect on the condensation rate and the speed of sound in the fluid; however, the analysis conservatively takes no credit for air released from the fluid.

The risk impact of a potential waterhammer is based on the uncushioned peak pressures and velocities, which is conservative with respect to the EPRI methodology.

COMMITMENT LIST

The following list identifies those actions committed to by the Davis-Besse Nuclear Power Station, Unit Number 1, (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. Henry L. Hegrat, Supervisor – FENOC Fleet Licensing, at (330) 374-3114.

COMMITMENTS	**DUE DATE**
None.	Not applicable.