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Docket Number 50-346

License Number NPF-3

Serial Number 2347

January 31, 1996

United States Nuclear Regulatory Commission
Document Control Desk
Washington, D. C. 20555

Subject: Ultimate Heat Sink/Service Water Temperature

Gentlemen:

In a letter dated October 12, 1995 (Toledo Edison Log Number 4628), the Nuclear Regulatory Commission (NRC) requested that Toledo Edison (TE) provide information regarding an apparent discrepancy between the Ultimate Heat Sink's maximum calculated water temperature and the Service Water System's maximum assumed inlet temperature used in the containment performance analysis for the Davis-Besse Nuclear Power Station (DBNPS) Unit No. 1. On January 16, 1996, Toledo Edison representatives notified Ms. L. L. Gundrum, NRC Project Manager, by telephone of its schedule for providing this information.

This letter provides the requested information which is summarized as follows:

- The DBNPS was designed and licensed during the late 1960's and early 1970's, at which time many of the regulatory requirements were still being developed. Accordingly, a detailed review was necessary to determine the specific requirements applicable to an older plant such as the DBNPS for the Ultimate Heat Sink (UHS) and containment performance analysis.

- The overall plant design and licensing basis for the DBNPS regarding seismic events and Loss-Of-Coolant Accidents (LOCA) is that a seismic event is not postulated to occur concurrently with a LOCA.

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- In order to determine that sufficient conservatism existed in the DBNPS UHS design for the dissipation of plant heat loads, the otherwise independent events of a LOCA, seismic event and extreme meteorological conditions were postulated, for this particular case, to occur concurrently. Severing of the UHS connection to Lake Erie, with minimum available Service Water System (SWS) intake volume and surface area available for heat dissipation, was combined with the maximum heat loads associated with a LOCA. This approach resulted in a variable SWS temperature, the maximum of which demonstrated an adequate degree of conservatism in the UHS design for dissipating heat loads.

- The probability for the combination of the independent events of a LOCA, seismic event, and extreme meteorological conditions is extremely low. As described in the enclosure, using the techniques of ANSI/ANS-2.12-1978, "Guidelines for Combining Natural and External Man-Made Hazards at Power Reactor Sites," the probability of the simultaneous occurrence is on the order of 1×10^{-12} per year.

- In the unlikely event of a seismic event inhibiting the use of Lake Erie as a water source, a DBNPS off normal procedure provides for the establishment of temporary pumping to the UHS from the lake.

- The licensing basis for the DBNPS containment performance analysis assumed a LOCA without loss of the UHS connection to Lake Erie. Furthermore, this analysis did not assume a seismic event concurrent with a LOCA. Accordingly, a constant SWS water temperature was appropriate for use in this analysis.

- A seismic evaluation performed recently for the DBNPS non-seismic Class I intake canal and intake conduit shows that these structures can be expected to withstand a Safe Shutdown Earthquake and remain functional. Accordingly, an increasing SWS temperature for the containment performance analysis would not occur, even in the event of an SSE, because the UHS is expected to remain connected to Lake Erie.

Further discussion on the UHS containment performance analysis and the appropriate SWS temperatures to use is provided below.

Ultimate Heat Sink

The UHS for the DBNPS during normal operations is Lake Erie, which is the source of cooling water in the intake canal forebay for the SWS. The lake water flows to the SWS intake forebay through a seismic Class II designed canal, which is connected to the lake by a single 96 inch diameter buried seismic Class II conduit. The buried conduit is connected to an intake crib located in the lake. The SWS intake forebay is of seismic Class I design and would serve as the UHS in the event of extreme low lake water or a loss of the connection to Lake Erie.

As described in more detail in the enclosure, in order to examine the degree of conservatism available in the UHS design for dissipating heat loads, the independent events of a seismic event, a LOCA and extreme meteorological conditions were postulated, for this particular case, to occur concurrently. Such a postulation resulted in a variable SWS temperature being the appropriate parameter to use for determining the amount of conservatism in the UHS design for dissipating heat loads.

The probability for the combination of the independent events of a LOCA, seismic event, and extreme meteorological conditions is extremely low. As described in the enclosure, using the techniques of ANSI/ANS-2.12-1978, "Guidelines for Combining Natural and External Man-Made Hazards at Power Reactor Sites," the probability of the simultaneous occurrence is on the order of 1×10^{-12} per year.

The DBNPS Updated Safety Analysis Report (USAP), Appendix 3D, summarizes the conformance of the DBNPS design with Safety Guide 27, "Ultimate Heat Sink," dated March 23, 1972, and the capability of the UHS to dissipate residual reactor heat even after a LOCA using only the UHS without the connection to Lake Erie. Appendix 3D also refers to an available procedure for maintaining long-term cooling by restoration of access to Lake Erie. This procedure is DBNPS Emergency Plan Off Normal Occurrence Procedure, RA-EP-028720, "Earthquake," which requires that, if a seismic event occurs and inhibits the use of Lake Erie as a water source, temporary pumping to the intake canal forebay will be established before the end of the 30-day stored water cooling period.

The seismic Class I UHS design was evaluated by both Toledo Edison and the NRC Staff as part of the Operating License application review and was determined to be sufficiently conservative in providing at least 30 days of cooling under these post-LOCA conditions, while assuring that the SWS equipment was capable of withstanding the elevated UHS temperatures. These evaluations also demonstrated that the UHS was capable of bringing the plant to a safe shutdown condition with low lake levels of 562 feet referenced to the International Great Lakes Datum (IGLD).

Containment Performance Analysis

As described in detail in the enclosure, the design and licensing basis for the DBNPS containment performance analysis assumes a LOCA without the loss of either the UHS intake canal or connection to Lake Erie. The DBNPS USAR describes the environmental design of mechanical and electrical equipment inside containment under LOCA conditions, which also does not assume a seismic event preceding, concurrent with, or following a LOCA.

This consideration of a single design basis event is consistent with the licensing basis documents applicable to the DBNPS. These include Final Safety Analysis Report (FSAR) Section 3.11, "Environmental Design of Mechanical and Electrical Equipment," and Section 6.2.1.3.2, "Containment Pressure Transient Analysis Break Spectrum," which established a LOCA as the Design Basis Accident (DBA) for the containment performance analysis, with initial conditions assuming a constant SWS inlet temperature of 85 degrees Fahrenheit. The 85 degrees Fahrenheit temperature was identified as lake water temperature and a seismic event was not assumed to occur in combination with the LOCA. The resulting NRC Operating License Safety Evaluation Report (NUREG-0136) likewise did not assume a seismic event in combination with a LOCA for the qualification of equipment within containment.

This design and licensing basis is reiterated in the present USAR containment performance analysis which reflects IE Bulletin 79-01B, "Environmental Qualification of Class 1E Equipment;" the October, 1980 NRC order regarding Environmental Qualification (EQ); NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Equipment;" and 10CRF50.49, "Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants."

Use of Appropriate SWS Temperatures

In order to examine the amount of conservatism available in the UHS design for dissipating heat loads, the independent events of a seismic event and LOCA were postulated under extreme meteorological conditions, for this particular case, to occur concurrently. The SWS temperature was assumed to initially be at 85 degrees Fahrenheit when a LOCA was postulated as occurring in combination with a seismic event. These evaluations concluded that there was sufficient conservatism in the design of the seismic Class I UHS.

Consistent with the regulatory requirements in effect and the NRC and Toledo Edison evaluations, the containment performance analysis did not postulate a LOCA occurring in combination with seismic event. This is consistent with NRC guidance such as the memorandum issued on July 7, 1985, by Mr. D. M. Crutchfield, NRC Assistant Director for Safety Assessment, regarding "Technical Specification Operability Requirements," which recognized that:

> "...all nuclear plants were not designed and built in conformance to the same regulatory requirements, the design basis events addressed in the FSAR's include differences depending on the requirements which existed at the time the construction permit and/or operating license was issued."

This memorandum went on to state:

> "Other than as specified by a regulatory requirement, each design basis event is taken as [an] individual case and not in combination with other design basis events."

and also state:

> "...as a design basis event, the SSE is not assumed to occur simultaneous with accidents."

Therefore, for the design and licensing of the DBNPS, the use of a variable SWS temperature for the UHS evaluation and the use of a constant SWS temperature for the containment performance analysis was not a discrepancy, but rather the use of the appropriate SWS temperature for the particular plant issue under analysis.

A more contemporary example of this approach is contained within Generic Letter 87-02, "Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue (USI) A-46," which explicitly states that for the USI A-46 issue, a LOCA is not postulated to occur simultaneously with or during a seismic event, and the seismic event is not postulated to cause a LOCA.

<u>Seismic Evaluation of Intake Canal and Intake Conduit</u>

After receiving the NRC's letter of October 12, 1995, Toledo Edison (TE), on its own initiative, had an evaluation performed of the ability of the intake conduit from Lake Erie, associated non-seismic Class I components, the intake canal dikes, and Lake Erie crib structure to withstand the effects of a Safe Shutdown Earthquake (SSE). This evaluation, performed for TE by EQE, International, considered both ground wave motion induced strains and displacements, inertial loads, and potential soils-related failures. A detailed discussion of this evaluation is contained in the enclosure to this letter.

The structures and components mentioned above were found to be capable of resisting the effects of a SSE with no loss of function, and no soil failures were predicted. Although the entire intake system was not designed as a seismic Class I system, it can be expected to withstand a design basis SSE and remain functional. Accordingly, an increasing SWS temperature for the containment performance analysis would not occur, even in the event of an SSE, because the UHS is expected to remain connected to Lake Erie.

Toledo Edison believes the enclosed information will support NRC closure of this item which was first raised in the NRC's Service Water System Operational Performance Inspection (SWSOPI) Report, 50-346/93016(DRS), dated December 29, 1993 (TE Log Number 1-2963) as Unresolved Item 93016-03. Toledo Edison initially provided information to the NRC to close this item in November, 1993.

Based on its review of the documents discussed in the enclosure to this letter and the present USAR descriptions, Toledo Edison plans to enhance the USAR description of the design and licensing basis for the UHS. These enhancements will be provided in the USAR revision issued following the upcoming Tenth Refueling Outage.

Should you have any questions or require additional information, please contact Mr. James L. Freels, Manager - Regulatory Affairs, at (419) 249-2366.

Very truly yours,

Dm — for JP Stetz

NKP/DRW/llh

Enclosure

cc: L. L. Gundrum, NRC Project Manager
H. J. Miller, Regional Administrator, NRC Region III
S. Stasek, DB-1 NRC Senior Resident Inspector
Utility Radiological Safety Board

DAVIS-BESSE NUCLEAR POWER STATION
ULTIMATE HEAT SINK/SERVICE WATER TEMPERATURE

REGULATORY REQUIREMENTS DURING INITIAL LICENSING OF THE DBNPS

At the time the Davis-Besse Nuclear Power Station (DBNPS) was being designed, constructed and licensed in the late 1960's and early 1970's, the existing General Design Criteria (GDC) and 10 CFR 100, Appendix A were not mandated in the Code of Federal Regulations. At that time, it was the practice of the Atomic Energy Commission (AEC) to require applicants for Construction Permits to specify the plant design, performance and siting criteria in Preliminary Safety Analysis Reports (PSARs) submitted by applicants in accordance with 10 CFR 50.34(a). This practice led to inconsistencies in the imposition of requirements for each plant being licensed during this time and, recognizing this, the AEC codified the GDCs in Appendix A to 10 CFR 50, which became effective on May 21, 1971 (Reference 1) and the Seismic and Geologic Siting Criteria in Appendix A to 10 CFR 100, which became effective on December 13, 1973 (Reference 2).

It should also be noted that at the time 10 CFR 50, Appendix A and 10 CFR 100, Appendix A were published, it was not the intent of the Commission to make them retroactive to plants that had received their Construction Permits (CPs) from the AEC. The regulations themselves, as well as the Statements of Consideration for these regulations, clearly state that they apply to proposed sites and those applying for a CP in accordance with 10 CFR 50.34(a) (References 1 and 2). This position has been supported even as recently as 1992 in correspondence between the Nuclear Regulatory Commission (NRC) Staff and the Commissioners (References 3 and 4) concerning resolution of deviations found during the NRC's Systematic Evaluation Program (SEP).

Since the CP for the DBNPS was issued on March 24, 1971 (prior to the codification of the GDCs), the design, performance, and siting criteria for the DBNPS were specified in the PSAR. Although the DBNPS criteria are similar to the GDCs, they are not identical and, in some instances, were applied in a manner differently than would be today.

Therefore, framework for the content and review of the DBNPS Final Safety Analysis Report (FSAR) was formed by the principle design performance and siting criteria contained in the PSAR, the AEC Safety Analysis Report (SAR) guide (Reference 5), and the NUREG-75/087, Standard Review Plan (Reference 6). It is important to note that the DBNPS FSAR, Appendix 3D contains information regarding the extent of Toledo Edison's committed compliance of the DBNPS design with the GDCs, the AEC Safety Guides, and the AEC Information Guides. However, while this appendix states the DBNPS design "meets the intent" of this guidance, the DBNPS design may not be entirely consistent with the regulations later issued (i.e., the GDCs), or later revisions of the safety and information guides referencing the GDCs.

THE DBNPS ULTIMATE HEAT SINK (UHS)

According to the DBNPS PSAR (circa 1970), Sections 2.4.1.2, "Lake Levels," 5.9.8, "Intake Water System," and 14.2.2.7, "Loss of Intake Canal," the UHS was designed such that in the event of a loss of communication with Lake Erie, the capacity of the seismic Class I portion of the UHS would be sufficient to allow for a safe and orderly plant shutdown. The two postulated occurrences that could cause the loss of communication were a seismic event or an extreme low lake level condition under severe meteorological conditions. There was no consideration of a coincident Loss-of-Coolant-Accident (LOCA) discussed in these sections, nor was there discussed a coincident loss of communication between the lake and the intake forebay in the PSAR LOCA analysis. Section 5.9.8 discussed an "emergency shutdown" and referred to Section 14.2.2.7 which stated that with the reactor tripped, sufficient cooling capacity would remain in the Seismic Class I portion for 60 days. Furthermore, as discussed in Section 14.2.2.7, the loss of the intake canal caused by a seismic event was an analyzed accident where the reactor was tripped, and a LOCA was not postulated to occur in combination with the seismic event.

Further discussion of the UHS design basis at this licensing stage is provided in the November 2, 1970 AEC Safety Evaluation Report (SER) supporting the issuance of the DBNPS CP (Reference 7), Section 8.4, "Intake Canal," which states:

> "The Class I portion of the intake canal provides 7.7 million gallons of cooling water between the water level of 560 feet MSL and the bottom of the service water pump suction inlet. This 7.7 million gallons of cooling water plus 250,000 gallons of condensate storage water is sufficient to cool the plant from power operation to cold shutdown and to remove the decay heat for more than 60 days for either normal or accident conditions...
>
> ...We have reviewed the intake canal design criteria, and we conclude that the design will provide an adequate cooling capability in the event Lake Erie cannot be used to supply cooling water due to either a seismic event or extremely low lake water level."

Taking the aforementioned PSAR basis for the 60-day cooling capacity into consideration, normal UHS conditions refer to conditions where communication between the lake and forebay exists, and UHS "accident conditions" actually refer to a condition where communication between the lake and forebay is lost by a seismic event in combination with an emergency shutdown by tripping the reactor (i.e., not by a LOCA).

Regulatory Guide 1.70, Revision 0, dated February, 1972, (Reference 5), contained guidance for the content of the DBNPS Final Safety Analysis Report (FSAR) regarding the UHS in Section 9.2, "Water Systems." This section stated that the capability of the UHS to dissipate waste heat "during normal and emergency shutdown conditions" should be discussed in the FSAR. A LOCA was not discussed and the "emergency shutdown" appeared to be the situation of the reactor being tripped as a result of a seismic event occurring. This section also noted that an AEC Safety Guide was under preparation to provide "additional guidance" on acceptable UHS features. Safety Guide 27, "Ultimate Heat Sink," (Reference 8) was issued by the NRC on March 23, 1972.

Safety Guide 27 implemented the requirements of GDC-44, "Cooling Water." It stated that the UHS "...performs two principal safety functions:

(1) Dissipation of residual heat after reactor shutdown, and

(2) Dissipation of residual heat after an accident."

It further stated that these "functions should be assured during and following the most severe natural phenomena (e.g., earthquake)."

The DBNPS FSAR described the design and capabilities of the UHS in a number of sections, most notably in Appendix 3D, where the degree of consistency with GDC 44 and Safety Guide 27 was described. In the discussion relating to GDC 44, Appendix 3D the DBNPS FSAR stated:

> "Heat from the component cooling water system is rejected to the service water system which, in turn, rejects heat to the cooling tower via the circulating water system during normal operation, or to the lake during a LOCA."

Similarly, in the discussion relating to Safety Guide 27, the DBNPS FSAR stated in Appendix 3D:

> "The ultimate heat sink for this station is Lake Erie, which is the source of cooling water for the service water system. While this is the single source for the sink, an analysis given in Section 9.2.5 demonstrates that the most severe natural phenomenon which can occur will not prevent a safe shutdown of the reactor. The seismic Class I portion of the intake structure forebay will provide adequate storage that will be capable of providing sufficient cooling for approximately 39 days. Procedures for ensuring a continued capability after this time will be available. A detailed comparison of the degree of compliance of this design with Safety Guide 27 is given below:

a. Sink Availability For At Least 30 Days

The heat sink for this station will provide adequate cooling for 39 days. Procedures for maintaining continued capability after this period will be available.

b. The Most Severe Phenomena Expected Taken Individually

An earthquake, which may result in loss of the source of lake water to the intake forebay, is the most severe event. This occurrence will not cause loss of the sink safety functions.

c. Site-Related Events

The occurrence of extremely low lake level, which reduces the quantity of available water in the forebay, in conjunction with loss of the canal, was considered. The lowest historical lake level was assumed for the analysis, and this condition does not preclude sink safety functions.

d. At Least Two Sources of Water For The Ultimate Heat Sink

This station has a single source of water for the heat sink. The collapse of the intake pipe or complete closure of the canal was postulated for the analysis. It is demonstrated that additional sources of water are not required since the stored water in the forebay is adequate for safe shutdown."

It is important to note that a LOCA is not discussed in the above Appendix 3D sections.

As part of the Operating License application, Final Safety Analysis Report Section 9.2.5.1, "Loss of Intake Canal," described a plant shutdown procedure to be used in the event of an earthquake that disabled the connection between the forebay and Lake Erie. In this FSAR discussion, a LOCA was not postulated in this scenario, rather after the earthquake occurred, the reactor was tripped and the UHS was shown to be capable of providing sufficient cooling for 39 days assuming an initial SWS temperature of 85 degrees Fahrenheit given these conditions. Similarly, Technical Specification 3.7.5, "Ultimate Heat Sink," requires a plant shutdown if the 85 degrees Fahrenheit UHS temperature limit is not met. In both of these cases, the action to shutdown the plant in the event of the postulated loss of the UHS function is analogous to situations where multiple trains of safety systems are inoperable and a safety function is lost. In these cases, the affected accident analyses are no longer bounding and an orderly shutdown removes the risk of having to cope with an accident in a degraded plant condition.

In November, 1973, the NRC posed a question (FSAR Question 9.2.8) to Toledo Edison (TE) with regard to the adequacy of the design of the UHS. The question coincided within several months prior to the issuance of Regulatory Guide 1.27, Revision 1, March, 1974 (Reference 9), which differed from its predecessor, Safety Guide 27, in that more detailed analyses were to be performed with regard to the long term heat removal capabilities of the UHS. According to the revised portions of Regulatory Guide 1.27:

> "Sufficient conservatism should be provided to assure that a 30-day supply of water is available and that the design basis temperatures of safety-related equipment are not exceeded. For heat sinks where the supply may be limited and/or the temperature of plant intake water from the sink may become critical (e.g., ponds, lakes, cooling towers, or other sinks where recirculation between plant cooling water discharge and intake can occur), transient analyses of supply and/or temperature should be performed."

Note that the "transient" referred to in this section of the regulatory guide concerns the time-dependent changes in the UHS quantity and/or temperature.

The FSAR Question 9.2.8, posed to TE was consistent with this revised portion of Regulatory Guide 1.27. The question stated:

> "This section of the FSAR contains design parameters and heat load utilized in the design of the "Ultimate Heat Sink". On what basis have the heat loads been calculated? Further, a staff review of available information does not support your conclusions that the service water system meets the suggested criteria of Regulatory Guide 1.27 "Ultimate Heat Sink". Your response should provide the following:
>
> a. The results of an analysis supporting your conclusions, in sufficient detail to permit an independent review;
>
> b. A discussion of how the Regulatory positions set forth in Safety Guide 1.27 were implemented. Identify each exception taken and provide the bases,
>
> c. A tabulation and plot spanning a thirty-day period of (1) the total heat rejected, (2) sensible heat rejected, (3) station auxiliary system heat rejected, and (4) decay heat from radioactive material..."

The key changes from the prior NRC positions (i.e., those given in the DBNPS FSAR and Safety Guide 27) were that the analysis was required to show that the UHS could supply cooling water for 30 days and that the postulated heat loads should be "sufficiently conservative".

To determine the amount of conservatism available in the UHS for dissipating heat loads, TE chose conservative conditions for heat load and initial plant conditions. This was done by assuming a LOCA and earthquake had occurred, and that the plant had been in an extreme low water condition from severe meteorological conditions. In its response to the NRC (FSAR Response 9.2.8), TE stated:

> "The ultimate heat sink has been evaluated using LOCA heat loads. The intake forebay is connected to Lake Erie by an intake canal and a seismic class II intake from the lake. In the analysis, a lake level of 562 feet, the minimum expected water level in the lake, was used. Concurrent with the LOCA, an earthquake was considered to have occurred which was assumed to collapse the intake from the lake as well as create an incredible collapse of the sides of the seismic class II portion of the intake canal. The intake canal collapse is assumed to leave one-third the water surface area and one-third of the water volume in the intake canal. In the analysis the water volume in the intake canal was used for make-up for evaporative losses from the forebay. However, the surface area may be used for cooling."

This analysis was performed assuming a combination of three independent events (i.e., LOCA, seismic event and extreme low water level). It ensured "sufficient conservatism" existed in the UHS design. The service water temperature at the start of the event was assumed to be 85.9 degrees Fahrenheit, and it was determined that "sufficient cooling was provided" for at least 30 days under post-accident (LOCA) conditions.

The NRC staff (Site Analysis Branch - Hydrologic Engineering Section) performed an independent analysis, assuming plant conditions similar to the TE analysis (i.e., LOCA with an earthquake and extreme low water conditions). This analysis, dated September 19, 1975 (Reference 10), reached the conclusion that the UHS could provide an adequate supply of water at a temperature 129 degrees Fahrenheit for at least 30 days and was, therefore, adequate.

The 129 degree Fahrenheit temperature was noteworthy because, as discussed in TE's response to NRC FSAR Question 2.4.11 (December 30, 1974), it was less than the 150 degrees Fahrenheit design water temperature limit for the Service Water System.

In December 1976, the NRC issued the Operating License Safety Evaluation Report (SER) for the DBNPS (Reference 11). In Section 2.4.3, "Water Supply," the intake canal forebay was identified as of seismic Class I design and used as the UHS in the event of low water or an accident. The SER discussed TE's analysis which demonstrated that the UHS would "provide emergency cooling for 39 days at a temperature no greater than 130 degrees Fahrenheit (maximum allowable plant return water temperature)." The SER also discussed the NRC Staff's independent analysis of the UHS which demonstrated a UHS maximum temperature of 129 degrees Fahrenheit with sufficient water for at least 30 days, and recognized that additional water was readily available from Lake Erie. Therefore, the UHS was found acceptable and in compliance with the criteria of Regulatory Guide 1.27.

In Section 9.3.3, "Ultimate Heat Sink," of the SER, it recognized that the UHS for the DBNPS is Lake Erie under normal conditions. Normal conditions, in the context of Section 9.3.3 of the SER, meant that the UHS was intact and communication between the lake and forebay existed. The SER discussed that in the event of a seismic event causing the collapse of the non-seismic Class I portion of the canal, the reactor would be tripped and placed in cold shutdown. The SER stated the UHS would be adequate to provide continuous cooling (after reactor cooldown) for 39 days. The SER stated that, based on the NRC Staff's independent evaluation of the UHS, it was concluded that the UHS design was adequate to safely shut down the plant and maintain it in a shutdown condition for a period of 30 days.

Following NRC issuance of the Operating License SER, the DBNPS Operating License was approved by the NRC. Appendix A to the Operating License contained the Technical Specifications which included Technical Specification 3/4.7.5, "Ultimate Heat Sink." The Technical Specification limited plant operation to a UHS minimum water level of 562 feet IGLD and an average water temperature to less than or equal to 85 degrees Fahrenheit. The associated Technical Specification Bases stated that these limitations were based on providing a 30-day cooling water supply to safety-related equipment without exceeding their design bases temperatures and were consistent with the recommendations of Regulatory Guide 1.27, Revision 1.

The current DBNPS Updated Safety Analysis Report (USAR), Appendix 3D, Section 3D.2.27, "Safety Guide 27 - 'Ultimate Heat Sink' (March 1972)" addresses consistency with the recommendations of Safety Guide 27. However, it also states the UHS design is consistent with the recommendations of Regulatory Guide 1.27, Revision 1, which included ensuring "sufficient conservatism" is in the UHS design.

The USAR discusses the evaluation of the UHS using LOCA heat loads in Section 9.2.5, "Ultimate Heat Sink." In this evaluation, the USAR states an earthquake was considered to have occurred concurrently with a LOCA, leaving only the intake forebay area to provide cooling water to the DBNPS. This USAR section discusses consistency with the recommendations of Regulatory Guide 1.27, Revision 1.

The evaluation in USAR Section 9.2.5.1, "Loss of Intake Canal," assumes the loss of the intake canal due to an earthquake, however, a LOCA is not specified. This evaluation shows the total time for which cooling is available after the loss of the intake canal is 39 days. Consistency with Safety Guide 27 or Regulatory Guide 1.27 is not discussed.

Another point of interest is the changes which occurred between the April, 1975 NUREG-75/087, "Standard Review Plan," and the current July, 1981 NUREG-0800, "Standard Review Plan," regarding Section 9.2.5, "Ultimate Heat Sink." NUREG-75/087, Section 9.2.5, begins by stating:

> "The Ultimate Heat Sink (UHS) is the source of cooling water provided to dissipate reactor decay heat and essential cooling system heat loads after a normal reactor shutdown or a shutdown following an accident."

NUREG-0800, Section 9.2.5, begins with the same statement, however, it added the words at the end "..., including LOCA." NUREG-75/087, Section 9.2.5, does not specify a "LOCA" anywhere in its text, while NUREG-0800, which does not apply to the DBNPS, specifies a LOCA be included.

To ensure access to Lake Erie, DBNPS Emergency Plan Off Normal Occurrence Procedure, RA-EP-02820, "Earthquake," requires that if a seismic event occurs and inhibits the use of Lake Erie as a water source, temporary pumping to the intake canal forebay will be established before the end of a 30-day stored water cooling period.

In summary, the DBNPS's early design and licensing basis for the UHS did not assume the combined occurrence of a design basis accident with a seismic event. In order to examine the amount of conservatism available in the UHS for handling heat loads, the independent events of a seismic event and a LOCA under extreme meteorological conditions were later postulated, for this particular case, to occur concurrently. Such a postulation resulted in a variable SWS temperature being the appropriate parameter to use for determining the amount of conservatism in the UHS design for handling heat loads.

CONTAINMENT PERFORMANCE ANALYSIS

The PSAR and CP SER did not discuss to any extent the containment performance analysis or the environmental qualification of mechanical or electrical equipment inside containment. The first discussion appears in the FSAR in Section 3.11, "Environmental Design of Mechanical and Electrical Equipment," which lists the equipment "...required to be operable during and subsequent to a Design Basis Accident." Neither a seismic event or a second Design Basis Accident (DBA) was postulated to occur concurrently.

FSAR Section 6.2.1.3.2, "Containment Pressure Transient Analysis Break Spectrum," delineated the initial conditions as a containment vessel temperature of 120 degrees Fahrenheit and a service water inlet temperature of 85 degrees Fahrenheit. Section 6.2.1.3.2 stated that "...the 14.14 ft^2 [LOCA] break is established as the Design Basis Accident (DBA)..." A seismic event was not postulated to occur concurrently with the LOCA for this analysis.

In FSAR Question 6.2.12, the NRC identified lake water as being used in the Component Air Coolers (CACs) for both normal and accident conditions and requested the Service Water System temperature. Toledo Edison's response to the NRC stated 85 degrees Fahrenheit was used as the "lake water" temperature to the CACs. This response shows that the NRC Staff was made aware during the FSAR-stage that a constant Service Water System temperature of 85 degrees Fahrenheit was being utilized in the accident analysis and thus, a coincident seismic event was not postulated to occur.

Supporting justification for not postulating a LOCA simultaneously with an additional external event is provided in the AEC SAR guide (Reference 5) and Chapter 15, "Accident Analyses," of the DBNPS FSAR, where external events (e.g., earthquakes, extreme weather conditions) are listed as design basis events. As such, it was the practice that these events be considered independently; i.e., one design basis event is not postulated to occur coincident with another design basis event, unless it can be shown that one causes the other. It is important to note that these external events were not specifically analyzed in Chapter 15 of the DBNPS FSAR because the effect of these phenomena were accounted for in the plant design (e.g., a seismic event would not cause a LOCA).

External events were considered to be independent and were not postulated to occur coincident with a design basis accident. In addition, no cause-effect relationships between these events were shown to exist, since critical piping and equipment was designed to withstand the effects of an external event or other accident without catastrophic failure.

Accordingly, the containment performance analysis described in Chapter 6 of the DBNPS FSAR was performed. The containment performance analysis assumed no seismic event occurred combined with the postulated LOCA, nor did a seismic event cause the postulated LOCA. Hence, a constant 85 degrees Fahrenheit service water inlet temperature was assumed for the analysis of containment heat removal and decay heat removal systems. This was reasonable, since, without a seismic event, the UHS would be intact. These assumptions are consistent with those provided in the NUREG-75/087 SRP (Reference 6) sections concerning the containment functional design. In addition, these SRP sections do not suggest that the two events be postulated simultaneously.

The resultant Operating License SER stated in Section 7.7, "Environmental Qualification," that "...all instrumentation, control and electrical equipment important to safety has been environmentally qualified in accordance with the requirements of IEEE 323-1971."
IEEE 323-1971 was a draft standard at this time titled "General Guide for Qualifying Electric Equipment for Nuclear Power Generating Stations," and did not provide requirements for a LOCA design basis event coincident with an external event (e.g., a seismic event).

USAR Section 3.11, "Environmental Design of Mechanical and Electrical Equipment," discusses environmental qualification. Subsection 3.11.1.1, "Qualification Evaluation," discusses that checklists were developed from the guidelines contained within NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Equipment," (January, 1980) and IE Bulletin 79-01B, "Environmental Qualification of Class 1E Equipment," (January 16, 1980). NUREG-0588 and IE Bulletin 79-01B defined the postulated accident condition in terms of a LOCA and/or High Energy Line Break (HELB) and did not postulate a coincident seismic event. Of note is the introduction to NUREG-0588 stated that seismic qualification was outside its scope.

USAR Section 3.11 also states that "Safety related electrical equipment, located in a harsh environment, is qualified to the requirements of 10CFR50.49 ["Electrical Qualification of Electric Equipment Important to Safety for Nuclear Power Plants"] using the qualification program developed for the station." 10CFR50.49, Section (e)(1) states "The time-dependent temperature and pressure at the location of the electric equipment important to safety must be established for the most severe design basis accident..." 10CFR50.49 does not state an external event, such as a seismic event, is required to be coincident with the most severe design basis accident (i.e., the LOCA).

USAR Section 3.11.1.2, "Environmental Conditions," states that the "Accident environmental conditions result from design basis accidents described in Sections 3.6, 6.2, and Chapter 15." None of these USAR sections postulate a LOCA coincident with a seismic event. Section 6.2.1.3.2, "Containment Pressure Transient Analysis Break Spectrum," specifies that the assumed initial SWS temperature was 85 degrees Fahrenheit and that the post-LOCA operation temperature was also 85 degrees Fahrenheit.

The SER issued by the NRC for the DBNPS on the "Environmental Qualification of Safety-Related Electrical Equipment," dated June 5, 1981, specifically stated under Section 2.2 that the scope was "limited to an evaluation of equipment which must function in order to mitigate the consequences of a loss-of-coolant-accident (LOCA) or a high-energy-line-break (HELB) accident..." A coincident event was not identified to be within the scope of this review.

In summary, the DBNPS design and licensing basis for the containment performance analysis did not assume the coincident occurrence of a design basis accident with a seismic event. This is consistent with the discussion in the next section of this enclosure.

RECENT NRC AND INDUSTRY POSITIONS

The Systematic Evaluation Program (SEP) was initiated by the NRC in 1977 to review the designs of older operating nuclear reactor plants in order to reconfirm and document their safety. The review provided an assessment of the significance of differences between current technical positions on safety issues and those that existed when a particular plant was licensed. The scope of plants to be reviewed under this program was determined by the date of issuance of the plant's CP; those plants with a CP issued before May 21, 1971 (the effective date of the GDCs) were generally considered to be "older operating plants" by virtue of the methods and requirements used to design and license them. Since the CP for the DBNPS was issued on March 24, 1971, it is considered an "older operating plant".

Among the issues reviewed as part of the SEP were the adequacy of UHSs, plant seismic design, and severe weather phenomena. Of particular note are the topics and safety objectives contained in many of plant specific SEP reports for the following three items:

- SEP Topic: II-3.C Safety-Related Water Supply (Ultimate Heat Sink (UHS))

  Safety Objective: To assure an appropriate supply of cooling water during normal and emergency shutdown procedures.

- SEP Topic: II-4 Geology and Seismology

  Safety Objective: To assure that accidents (for example, loss-of-coolant accident) do not occur and that plants can safely shut down in the event of geologic and seismologic phenomena which may occur at the site.

- SEP Topic: II-4.A Tectonic Province

  Safety Objective: To assure that plants can be safely shut down in the event of geologic and seismologic phenomena which may occur at the site.

As can be seen from these SEP topics, the NRC recognized that plants of the vintage of the DBNPS were not consistently designed and licensed in the areas of UHS and seismic design. The specific safety objectives for the three topics listed above provide regulatory guidance on the coupling of accidents and seismic events as they relate to UHS designs. The objectives do not postulate these events in combination. These safety objectives are likewise consistent with the DBNPS design and licensing bases described earlier in this enclosure.

Shortly after the SEP, the issue of seismic qualification of mechanical and electrical equipment emerged as Unresolved Safety Issue (USI) A-46 in December 1980. The safety concern was that equipment in nuclear plants with applications docketed before about 1972 (which includes the DBNPS) was not reviewed according to the current licensing criteria for seismic qualification of equipment. This equipment may not have been adequately qualified to ensure its survival and functionality in the event of a safe shutdown earthquake (SSE). The NRC staff determined that it was not feasible to require older operating plants to meet current licensing requirements and allowed alternate means to demonstrate adequate qualification.

As is described in References 12, 13 and 14, the NRC staff concluded that it was unnecessary to verify the seismic adequacy of all plant equipment defined as seismic Class I. This meant that only those systems, subsystems, and components required to bring the plant to a safe, hot shutdown condition and to maintain it in that condition were important to assure safety during and after a SSE event. The scope of the seismic verification, therefore, was limited to the minimum equipment necessary to perform the functions related to plant safe shutdown. This approach was consistent with seismic reviews conducted by the SEP and with regulatory guidance at the time on the simultaneous occurrence of a LOCA with a seismic event. Accordingly, the NRC staff developed assumptions related to defining the equipment scope and required plant functions for resolution of this issue. The assumptions listed in the enclosure to Generic Letter 87-02, which dictate the systems and equipment that are required to mitigate a seismic event, included:

(1) The seismic event does not cause a LOCA, a Steam-Line-Break Accident (SLBA), or a High-Energy-Line-Break (HELB).

(2) The LOCA, SLBA or HELB will not be postulated to occur simultaneously with or during a seismic event. However, the effects of transients that may result from ground shaking should be considered.

As was the case with the SEP review mentioned earlier, the DBNPS design and licensing basis is consistent with the assumptions developed for the resolution of USI A-46. These assumptions do not postulate a simultaneous LOCA and seismic event, unless a cause-effect relationship is shown.

As previously discussed, a similar position was taken by the NRC staff in its 1985 memorandum on Technical Specification operability requirements (Reference 15). In this memorandum, the NR^ Assistant Director for Safety Assessment stated:

> "Design basis events are analyzed to demonstrate that a plant can be operated without undue risk to public health and safety. Other than as specified by a regulatory requirement, each design basis event is taken as an individual case and not in combination with other design bases events. For those systems for which the design criteria specify that the safety function shall be provided with either onsite or offsite power system operation assuming the other is unavailable, a situation exists where the event of a loss of offsite power is considered simultaneously with an event in which the safety function of such systems is required.
>
> However, the fact that safety related structures, systems, and components are designed to remain functional during a Safe Shutdown Earthquake (SSE) and assure the integrity of the reactor coolant pressure boundary, the capability to shutdown the reactor and maintain it in a safe condition or the capability to mitigate the consequences of accidents, as a design basis event the SSE is not assumed to occur simultaneous with accidents."

Clearly, the NRC staff has provided regulatory guidance on several occasions that external events and accidents need not be considered simultaneously. However, the design criteria require that "appropriate combinations" of external events need be considered. The criteria for "appropriate combinations" today is generally determined using probabilities. These techniques were not widely used at the time of the licensing of the DBNPS, however, the conclusions made using probabilistic techniques support the conclusions made in the DBNPS licensing process.

As was mentioned earlier in this document, an analysis was performed by TE in response to NRC questions during the licensing process that assumed the simultaneous occurrence of a LOCA, an earthquake, and extreme meteorological conditions. This analysis was performed in order to produce a "sufficiently conservative" heat load for the UHS such that a judgment of adequacy of its design could be made and it was not intended to form the overall licensing basis of the DBNPS with regard to simultaneous occurrence of design basis accidents with external events. The DBNPS general licensing basis assumes that each design basis accident and each external event are treated independently. Guidance for consideration of the simultaneous occurrence of two or more external events can be found in ANSI/ANS-2.12-1978, "Guidelines for Combining Natural and External Man-Made Hazards at Power Reactor Sites," (Reference 16).

ANSI/ANS-2.12-1978 implies that seismic events and weather related events are independent events. The probability of the simultaneous occurrence of a combination of these events can be calculated based upon the probability of occurrence of each of the external events and the expected duration of these events. Criteria for exclusion from consideration of combinations of events is also given. If the probability of occurrence of a particular combination of external events is less than $1x10^{-6}$ per year, that combination need not be considered in the plant design.

The most conservative probability of occurrence of a safe shutdown earthquake is given in this standard as $1x10^{-3}$ per year (although the site specific probability is much less), and the event duration is given as 60 seconds. According to the DBNPS FSAR, a conservative probability of occurrence of the extreme meteorological conditions postulated is estimated to be $1x10^{-3}$ per year with a duration of 30 days.

Using these values and the techniques given in ANSI/ANS-2.12-1978 for calculating the probability of the simultaneous occurrence of these two independent events, the probability of simultaneous occurrence is calculated to be approximately $8x10^{-8}$ per year. This is well below the exclusion criteria of $1x10^{-6}$ per year provided above, and, as such, this combination of events need not be considered in the plant design.

Therefore, from a probability standpoint, postulating these events simultaneously is not credible and they can be treated as independent events. In fact, assuming a LOCA concurrent with this incredible combination of external events, the probability of the simultaneous occurrence is on the order of $1x10^{-12}$ per year.

SEISMIC EVALUATION OF INTAKE CANAL AND INTAKE CONDUIT

After receiving the NRC's letter of October 12, 1995, TE, on its own initiative, had an evaluation performed of the ability of the intake conduit from Lake Erie, associated components, the intake canal dikes and the Lake Erie crib structure to withstand the Safe Shutdown Earthquake (SSE). This evaluation, performed for TE by EQE, International, considered both ground wave motion induced strains and displacements, inertia loads and potential soils related failures. The following discusses this evaluation in more detail and provides the conclusions.

Background

The intake conduit is prestressed concrete pipe designed and fabricated to AWWA Specification C301-72. The conduit is fabricated in 20 foot lengths and is connected by bell and spigot joints with steel lugs and AISI C-1020 draw bolts. The conduit extends approximately 3200 feet into Lake Erie to a timber intake crib. The conduit is embedded in backfill in the glaciolacustrine layer with the bottom of the conduit typically located just above the dolomite bedrock. With the exception of two 45° wyes with bulkhead plugs and the inlet bell, the conduit consists of a straight run with no bends or elbows.

The intake crib is fabricated from oak timbers of various sizes in the form of a platform with open walls and inlet grates which encloses the conduit inlet bell. It is weighted to achieve negative buoyancy with limestone riprap and surrounded around its periphery with additional riprap. The crib is founded at elevation 554.92 feet with the top of the structure at 561.85 feet. The lake bottom at the crib location is at approximately 557.50 feet.

Soil Failure

The potential for liquefaction and slope instability for the non-seismic Class I portion of the intake water canal dikes and the glaciolacustrine and till deposits in which the 96-inch intake conduit is founded was investigated. With respect to liquefaction potential, both the glaciolacustrine and till deposits are classified as clays. The glaciolacustrine deposits meet the criteria for no liquefaction as defined in EPRI NP-6041, "Nuclear Power Plant Seismic Margin." The till deposits are judged to meet all but one of the criteria for excluding liquefaction as a credible concern. Evaluation of the till based on the very conservative assumptions that it is a fine-grained sandy soil and an earthquake of up to an approximate magnitude of 7.5, at a distance producing a peak acceleration well above 0.15g, demonstrates that this deposit will not liquefy. This finding is consistent with other experience with glacial till deposits.

From the review of construction drawings and backfill specifications, it is concluded that the dikes for the seismic Class I and non-seismic Class I portions of the intake canal were constructed to the same specification. The non-seismic Class I portions of the intake canal dike were evaluated by applying the same assessment methodology used for the seismic Class I portion of the intake canal dike. The factor of safety against seismic slope failure of the non-seismic Class I intake canal dikes is at least equal to that of the seismic Class I portion (2.5) and is likely more than twice as great.

Based upon the above findings, the entire intake canal is adequate for the SSE and the intake water conduit need only be evaluated for response to earthquake-generated ground strain.

It is noteworthy that USAR Section 9.2.5.1, "Loss of Intake Canal," although postulating a complete loss of the canal, recognizes that even if both of the dikes collapse into the canal, there will still be one-third of the total area available in the canal for water flow. Regulatory Guide 1.27, Revision 1, similarly states that the consequences of a postulated slide of earthen canal walls should be assumed; however, it is not necessarily required that one assume water flow ceases completely.

Concrete Conduit

The concrete conduit was analyzed for wave action induced strains based on an apparent wave propagation speed of 11,400 fps which corresponds to the compression wave velocity in the dolomite bedrock at the SSE peak ground acceleration. The SSE is defined as a peak free field surface acceleration of 0.15g. In view of the shallow thickness of the overburden in the vicinity of the structures evaluated here, the peak acceleration of the bedrock was also assumed to be 0.15g, resulting in a slight conservatism. Maximum axial strains of approximately 3.7×10^{-5} in/in result. Bending strains are negligible in the straight conduit run. Conduit segments are supplied in 20 foot lengths with 5-1/8 inch bell and spigot joints. One inch gaps between the joint shoulders is recommended for the outside joint space and 0.88 inch for the inside space. These values far exceed the maximum 0.009 inch end movement based on the above strain. Thus, the maximum compressive strain in the conduit results in a concrete stress of about 120 psi which is much less than the 3200 psi design strength for Class A pipe.

The conduit tensile strains are resisted by the connecting ASTM 570 Grade C steel cylinder. The cylinder has a nominal diameter of 100-1/4 inches and is 0.0538 inches thick. The minimum specified yield strength is 33,000 psi. At the 3.7×10^{-5} in/in axial strain, the stress in the cylinder is about 1060 psi which is well below the yield stress.

Two 1-1/2 inch diameter by 32 inch long AISI C-1020 draw bolts are specified for each joint. Based on the 0.009 inch conduit segment end motion and an assumed effective bolt length of 28 inches, nominal bolt stresses of about 9300 psi are expected compared to a yield strength of about 49,000 psi.

It is concluded that the concrete intake conduit can survive strains induced by a 0.15g probable ground acceleration seismic event at stresses well below corresponding yield strengths of the individual components and thus will suffer no loss of function.

Crib

The oak intake crib is fabricated from timbers of various sizes bolted together with 3/4 inch diameter threaded rods. Diagonal braces from the top of the crib to the corners are 1-1/2 inch diameter. The crib is approximately 59'-0" by 59'-2" in plan with a total height from the lower timber to the top of the cover timbers of 6'-10". The four corners (18'-2" by 18'-0" in plan) consist of two layers of 12 by 12 timbers loaded with limestone riprap to achieve a negative buoyancy. An additional 18'-6" of riprap is placed around the periphery of the crib. The crib is embedded approximately 2'-7" below the bottom of the lake.

The crib was checked for sliding and vertical uplift under the combined loads from dead weight, buoyancy, and earthquake. The crib was found to be stable for both vertical and horizontal excitation and no rigid body motion will occur for the 0.15g SSE.

Stresses in the oak timbers were computed assuming no friction between the riprap and the timber platform. This very conservative assumption results in bending stresses in the timber of about 470 psi compared to a Uniform Building Code allowable stress of 575 psi for #2 or better red oak. Shear stress in the threaded rods of 6200 psi were calculated for the same conservative assumptions with 1440 psi in the 1-1/2 inch diagonal braces.

Since no rigid body motion occurs and all crib stresses are well below the corresponding allowable strengths, it is concluded that no earthquake-induced loads are developed in the crib which could result in a loss of function in the concrete intake conduit.

Conclusion

The intake canal dikes, intake conduit, and Lake Erie crib structure were found to be capable of resisting the SSE with no loss of function, and no soil failures were predicted. Therefore, although the intake system was not designed as a seismic Class I system, it can be expected to withstand the design basis SSE and the UHS can be expected to provide adequate post-LOCA cooling with or without the coincident occurrence of a seismic event.

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3. SECY-92-223, September 18, 1992, "Resolution of Deviations Identified During the Systematic Evaluation Program".

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5. Regulatory Guide 1.70, Revision 0, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants", February, 1972.

6. NUREG-75/087, December 1975, "Standard Review Plan".

7. "Safety Evaluation by the Division of Reactor Licensing, U. S. Atomic Energy Commission in the matter of the Toledo Edison Company, the Cleveland Electric Illuminating Company, Davis-Besse Nuclear Power Station, Docket No. 50-346", November 2, 1970.

8. AEC Safety Guide 27, "Ultimate Heat Sink", March 23, 1972.

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10. Letter from H. R. Denton (NRR) to V. A. Moore (RL), September 19, 1975, "Revised Hydraulic Engineering Summary".

11. NUREG-0136, "Safety Evaluation Report by the Office of Nuclear Reactor Regulation, United States Nuclear Regulatory Commission, in the matter of Toledo Edison Company, Cleveland Electric Illuminating Company, Davis-Besse Nuclear Power Station, Unit 1, Docket No. 50-346", December 1976.

12. NUREG-1030, "Seismic Qualification of Equipment In Operating Nuclear Power Plants", February 1987.

13. NUREG-1211, "Regulatory Analysis for Resolution of Unresolved Safety Issue A-46, Seismic Qualification of Equipment in Operating Plants", February 1987.

14. NRC Generic Letter 87-02, "Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue (USI) A-46", February 19, 1987.

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