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Docket Number 50-346

License Number NPF-3

Serial Number 2439

January 28, 1997

United States Nuclear Regulatory Commission
Document Control Desk
Washington, D. C. 20555-0001

Subject: Interim Response to NRC Generic Letter 96-06: "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions"

Ladies and Gentlemen:

On September 30, 1996, the Nuclear Regulatory Commission (NRC) issued GL 96-06 (Toledo Edison letter Log Numb[illegible]919). That letter requested the Davis-Besse Nuclear Power Station ([illegible]S) Unit Number 1, to respond within 120 days and to address the following issues:

(1) Cooling water systems serving the containment air coolers (CACs) may be exposed to the hydrodynamic effects of water hammer during either a loss-of-coolant accident (LOCA) or a main steamline break (MSLB). These cooling water systems were not designed to withstand the hydrodynamic effects of water hammer and corrective actions may be needed to satisfy system design and operability requirements. Licensees are to determine if their plant's CACs cooling water systems are susceptible to waterhammer during postulated accident conditions.

(2) Cooling water systems serving the containment air coolers may experience two-phase flow conditions during postulated LOCA and MSLB scenarios. The heat removal assumptions for design-basis accident scenarios were based on single-phase flow conditions. Corrective actions may be needed to satisfy system design and operability requirements. Licensees are to determine if their plant's CACs cooling water systems are susceptible to two-phase flow conditions during postulated accident conditions.

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(3) Thermally induced overpressurization of isolated water-filled piping sections in containment could: 1) jeopardize the ability of accident-mitigating systems to perform their safety functions and, 2) could also lead to a breach of containment integrity via bypass leakage. Corrective actions may be needed to satisfy system operability requirements. Licensees are to determine if the piping system which penetrate their plant's containment are susceptible to thermal expansion of fluid so that overpressurization of piping could occur.

Generic Letter 96-06 also required that a written summary report be submitted to the NRC by Ja: ;' 28, 1997, stating: (1) the actions taken in response to the requested actions noted above, (2) the conclusions that were reached relative to susceptibility for waterhammer and two-phase flow in the CAC cooling water systems and overpressurization of piping that penetrates containment, (3) the basis for continued operability of affected systems and components, and (4) the corrective actions that were implemented or are planned to be implemented.

In a telephone call with the NRC Office of Nuclear Reactor Regulation Project Manager for the DBNPS on January 13, 1997, the DBNPS staff provided a brief progress report on the detailed evaluation requested by Generic Letter 96-06 and stated that the complete summary report would be delayed. On January 23, 1997 the DBNPS staff provided further details of the progress of the evaluation and requested an extension of 30 days from the due date of January 28, 1997. The NRC staff evaluated the DBNPS progress to date and concurred with an extended due date of February 28, 1997. The requested evaluations and summary report are under preparation, and the current progress to date is summarized below.

<u>CAC Water Hammer and Two-Phase Flow</u>

The DBNPS has contracted with Fauske and Associates Inc. to apply small-scale modeling experiments to study CAC water-hammer effects for geometries similar to those that exist at the DBNPS. Preliminary results indicate that waterhammer transients would not develop significant loads, and the CACs wi emain operable during bounding design basis events. In addition, review of the CACs' design and operation during the bounding design basis LOCA conditions indicate that the CACs are not susceptible to detrimental two-phase flow. Even if two-phase flow conditions should develop during a loss of power scenario, single phase flow would be restored at approximately the time the CACs are credited in the containment analysis. Therefore, the CACs remain operable.

<u>Thermal Overpressurization of Piping</u>

As a result of preliminary evaluations, thirteen containment piping penetrations were identified as potentially susceptible to post-LOCA thermal overpressurization. Four of these thirteen penetrations have been determined to meet the American Society of Mechanical Engineers (ASME) Boiler & Pressure Vessel Code allowable stress values and no further action would be required.

Three of the thirteen penetrations (Penetration Numbers 13 - Containment Normal Sump, 47A - Core Flood Sample, and 74C - Pressurizer Auxiliary Spray) meet the ASME Code interim allowable stress values. These penetrations will be modified, as necessary, to meet the ASME Code values during the next refueling outage (11RFO). Penetration Number 74C is potentially used as a non-preferred long term boron dilution flowpath. While piping stresses would remain acceptab e, pressurization could interfere with subsequent valve operation. Therefore, this penetration has been partially drained to support isolation valve operability.

The preliminary evaluations have determined that the remaining six containment penetrations could exceed ASME Code interim allowable values. None of these pipe penetrations support post accident mitigating functions and each of the penetrations remain operable as discussed below. One of these six penetrations (Penetration Number 4 - Component Cooling Water Outlet from Containment) has pre-existing, acceptable, measured leakage as determined from local leak rate tests performed in accordance with 10 CFR 50, Appendix J. This leakage is expected to prevent overpressurization of this penetration. Two of the six penetrations (Penetration Number 21 - Demineralized Water and Penetration Number 48 - Pressurizer Quench Tank Outlet) have air-operated globe valves which are oriented such that pressurization of the penetration piping is projected to be relieved by causing the disk to be lifted against the actuator spring force without causing valve or piping damage. One of the six penetrations (Penetration Number 32 - Reactor Coolant Drain to Reactor Coolant Drain Tank) has a diaphragm/air actuator which would be slightly displaced by the pressure increase, allowing adequate leakage to prevent overpressurization.

The remaining two penetrations' valves (Penetration Number 12 - Component Cooling Water to Control Rod Drive Mechanisms and Penetration Number 49 - Refueling Canal Fill) would likely develop packing leaks to limit the increase in pressurization. However, without crediting this relief mechanism, the thermal expansion of liquid in these penetrations is projected to not result in more than approximately 2% plastic elongation under worst case conditions and, therefore, the affected penetrations will remain operable. In addition, the piping in Penetration Number 49 will be drained slightly to reduce the volume of liquid subject to heatup and pressurization.

The conservatism of the preliminary analyses supports the reasonable conclusion that actual pressures experienced during a design bases event would not produce stresses beyond yield strength. The conservatisms include the use of worst case temperatures for containment and affected systems. For example, due to the cold environmental temperatures at this time of year the existing initial Containment, Service Water, and Borated Water Storage Tank temperatures are significantly lower than that used in the bounding analysis. This would lead to a more rapid reduction in containment temperature following a large break Loss of Coolant Accident (LOCA) and smaller penetration piping pressures. In addition, no leakage was assumed from the packing of valves in isolated containment penetration piping, while valve seat and stem leakage would be reasonably expected in most cases at the pressures involved.

As stated above, the DBNPS plans to respond in more detail to Generic Letter 96-06 by February 28, 1997. In the meantime, 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,

John K. Wood

FWR/laj

attachment

cc: A. B. Beach, Regional Administrator, NRC Region III
A. G. Hansen, DB-1 NRC/NRR Project Manager
S. Stasek, DB-1 NRC Senior Resident Inspector
Utility Radiological Safety Board

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Attachment

INTERIM

RESPONSE

TO

NRC GENERIC LETTER 96-06

FOR THE

DAVIS-BESSE NUCLEAR POWER STATION

UNIT NUMBER 1

This letter is submitted pursuant to 10 CFR 50.54(f). Attached is information pursuant to NRC Generic Letter 96-06, "Assurance of Equipment Operability and Containment Integrity During Design - Basis Accident Conditions" for the Davis-Besse Nuclear Power Station, Unit Number 1.

By: John K. Wood

John. K. Wood, Vice President - Nuclear

Sworn to and subscribed before me this 28th day of January, 1997.

Laura A. Jennison

Notary Public, State of Ohio

LAURA A. JENNISON
Notary Public, State of Ohio
My Commission Expires 8-15-2001