Douglas R. Gipson
Senior Vice President
Nuclear Generation

Detroit
Edison

Fermi 2
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(313) 586-5249

January 28, 1997
NRC-97-0003

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D. C. 20555

References: 1) Fermi 2
NRC Docket No. 50-341
NRC License No. NPF-43

2) NRC Generic Letter 96-06: Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions, dated September 30, 1996

3) Detroit Edison Letter to NRC, "Detroit Edison 30-Day Response to NRC Generic Letter 96-06", NRC-96-0118, dated October 30, 1996

Subject: Detroit Edison 120-Day Response to NRC Generic Letter 96-06

Reference 2 requested a 120 day response describing actions taken and conclusions reached relative to the issues communicated by Generic Letter 96-06. Reference 3 committed Detroit Edison to the 120 day response. Detroit Edison was requested to determine: 1) if containment air cooling water systems are susceptible to either waterhammer or two-phase flow conditions during postulated accident conditions; 2) if piping systems that penetrate the containment are susceptible to thermal expansion of fluid so that overpressurization of piping could occur.

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Based on engineering reviews that were performed it was determined that the containment air cooling water systems continue to be operable. Detroit Edison has determined containment penetrations are operable based on engineering judgment, taking credit for expected leakage of boundary valves to prevent over pressurization of piping. Please refer to the Attachment for Detroit Edison's complete response to the Generic Letter.

The following commitment is being made in this letter:

> Detroit Edison will monitor the ongoing discussion of these generic issues between the Nuclear Energy Institute and NRC and design and install modifications as necessary, to provide adequate overpressure protection of affected drywell piping penetrations. If necessary to maintain the Fermi 2 containment integrity licensing basis, these modifications will be installed by the completion of the next refueling outage.

If you have any questions, please contact Joseph M. Pendergast, Licensing Engineer at (313) 586-1682.

Sincerely,

Attachment

cc: A. B. Beach
M. J. Jordan
A. J. Kugler
A. Vegel

I, DOUGLAS R. GIPSON, do hereby affirm that the foregoing statements are based on facts and circumstances which are true and accurate to the best of my knowledge and belief.

DOUGLAS R. GIPSON
Senior Vice President

On this 28th day of January, 1997 before me personally appeared Douglas R. Gipson, being first duly sworn and says that he executed the foregoing as his free act and deed.

Rosalie A. Armetta
Notary Public

ROSALIE A. ARMETTA
NOTARY PUBLIC - MONROE COUNTY, MI
MY COMMISSION EXPIRES 10/11/99

RESPONSE to GENERIC LETTER 96-06

Requested Action Number 1: Determine if containment air cooler cooling water systems at Fermi 2 are susceptible to either waterhammer or two-phase flow conditions during postulated accident conditions.

Background

The design basis Loss of Coolant Accident (LOCA) concurrent with a Loss of Offsite Power (LOP) is the limiting event with regard to the concerns described in Generic Letter 96-06. At Fermi 2 the Reactor Building Closed Cooling Water (RBCCW) system supplies coolant for the drywell coolers. The RBCCW System and drywell coolers are not required for either safe shutdown or to mitigate postulated accidents. The Emergency Equipment Cooling Water (EECW) system provides emergency cooling to the essential equipment necessary for safe shutdown of the plant normally cooled by the RBCCW system.

Actions Taken and Basis for Continued Operability

As requested by the Generic Letter, Detroit Edison has performed an evaluation to determine if containment air cooler cooling water systems at Fermi 2 are susceptible to either waterhammer or two-phase flow conditions during postulated or Design Basis Accidents (DBA) conditions. The results of this evaluation are discussed below.

In the postulated accident scenario, the RBCCW supply to the drywell coolers is isolated automatically on high drywell pressure. The RBCCW isolation valves would be closing during the EECW pump start and acceleration interval. Although some steam bubble formation in the coolers during the early stages of the accident is possible, there would be virtually no flow through the cooling coils at this time and the EECW piping would not be affected.

Steam bubble collapse is not expected to occur because there is no flow of subcooled water into the system. This flow is precluded initially by automatic closure of the drywell closed cooling water supply line as noted above. Procedural controls prevent reestablishing cooling water flow to the drywell coolers if predetermined limits on drywell pressure and torus temperature have been exceeded thus minimizing the potential for waterhammer during the accident recovery process. Also, the effects of any possible steam formation in the RBCCW/EECW system would be mitigated by relief valves on EECW piping inside the drywell and / or RBCCW/EECW head tanks.

In summary, Fermi 2 does not rely on drywell coolers for post accident containment heat removal or safe shutdown. Waterhammer or two phase flow events in the RBCCW/EECW systems are not a concern based on review of the system design including the time it takes for the flow of RBCCW cooling water to stop during the design basis accident, the time to establish cooling water flow in the EECW system, and operating restrictions on system restart to reestablish flow to the drywell coolers.

Based on this evaluation, it was concluded that the EECW, RBCCW and drywell cooling systems continue to be operable without the need for any modifications to the design or procedures. Therefore, no further corrective active actions are planned.

Requested Action Number 2: Determine if piping systems that penetrate the containment at Fermi 2 are susceptible to thermal expansion of fluid so that overpressurization of piping could occur.

Actions Taken

Prior to restart from the fifth refuel outage an evaluation of containment piping penetrations was performed. This evaluation determined that some piping penetrations are susceptible to thermal overpressurization. These penetrations are:

1. Main Steam Drains [3" Nominal Pipe Size (NPS)]
2. Drywell Floor Drain Sump Discharge (3" NPS)
3. Drywell Equipment Drain Sump Discharge (3" NPS)
4. Reactor Water Sample (1" NPS)
5. Reactor Recirculation System (RRS) Pump A Seal Purge (1" NPS)
6. Reactor Recirculation System (RRS) Pump B Seal Purge (1" NPS)

The review of drywell piping systems inside of the inboard isolation valve also identified that the two drywell drain sump discharge lines between the pump discharge check valves and the inboard isolation valves are susceptible to thermal overpressurization.

As noted in Generic Letter 96-06 the potential for system failure as a result of thermally induced overpressurization is dependent on many factors. The Generic Letter further notes that these factors include leak tightness of valve seats, bonnets, packing glands, etc. Taking such expected leakage into account the determination was made that for these piping penetrations isolation valve leakage will limit the pressure rise to less than the failure pressure.

The Basis for Continued Operability

The review concluded that operability of containment penetrations listed above was not adversely affected by this phenomenon. The review was based on engineering judgment and included considerations such as those described in the following paragraphs.

For Reactor Water Sample, and the RRS Pump Seal Purge (2 penetrations), the containment isolation valves are globe valves and the expanding volume of water is small. The valves would be expected to leak to limit thermal overpressurization before the 3/4" and 1" schedule (sch.) 80 or 160 piping would fail.

The main steam drains drywell penetration is isolated hot after drain flow has reduced so the penetration piping is not likely to be solid. Also, the piping should be above ambient temperature due to interfacing high temperature systems on each side of the isolation valves. Lastly, although the isolation valves are gate valves, valve leakage to limit thermal overpressurization would be expected before the 3" sch. 160 piping would fail.

The piping between the drywell sump pump discharge check valves and the inboard isolation valves cannot significantly pressurize since the two check valves are hard seated and would leak to limit thermal overpressure. This piping is not safety related.

Also, since the inboard isolation valves are normally open, check valve leakage would drain the penetration piping at the high point of the piping system so the penetration piping is not likely to be solid. Although the isolation valves are gate valves, the valve will leak to limit thermal overpressurization before the 3" sch. 40 piping would fail.

The evaluation was based on the assumption that leakage would exist as discussed above. Local Leak Rate Test data was reviewed to confirm that all of the penetrations exhibited at least some leakage when last tested in the fifth refuel outage. The evaluation was based primarily on engineering judgment.

Corrective Actions Planned

Detroit Edison will monitor the ongoing discussion of these generic issues between the Nuclear Energy Institute and NRC and design and install modifications as necessary, to provide adequate overpressure protection of affected drywell piping penetrations. If necessary to maintain the Fermi 2 containment integrity licensing basis, these modifications will be installed by the completion of the next refueling outage.