FROM:
Northern States Power Company
Minneapolis, Minn. 55401
L.O. Mayer

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TO:
Dr. Peter A. Morris

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Ltr reporting a condition on 11-6-71 during an inspection of the torus internals, structural damage was observed...

ENCLOSURES:

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NORTHERN STATES POWER COMPANY

MINNEAPOLIS, MINNESOTA 55401

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December 15, 1971

Dr. Peter A Morris, Director
Division of Reactor Licensing
United States Atomic Energy Commission
Washington, D C 20545

Dear Dr. Morris:

Monticello Nuclear Generating Plant

Docket No. 50-263 License No. DPR-22

Damage to Torus Baffles

A condition was found at the Monticello Nuclear Generating Plant on November 6, 1971, that requires reporting to your office in accordance with the provisions of Appendix A, Technical Specifications, of the Provisional Operating License DPR-22. Reporting is required in accordance with Section 6.6.C.1 of the Specifications. The Region III Compliance Office has been notified in accordance with the requirements of Section 6.6.A.1 of the Technical Specifications.

Summary Description

On November 12, 1971, the reactor was shut down for a number of scheduled maintenance items. Primary containment was de-inerted to permit work activities in the drywell. On November 16, 1971, during an inspection of the torus internals, structural damage was observed. The torus was designed with 72 baffle sections included in the design to prevent a short term overpressure as observed in a series of quarter scale tests performed at Moss Landing. Eleven baffle sections were found displaced which had apparently resulted in a broken air line to the actuator of number 2382-H vacuum breaker, damage to 6 of the 8 low point 1" drain lines on the drywell vent distribution header, slight damage to the torus catwalk support braces, and a number of scratches in the torus wall paint with no significant damage to the base metal. In addition, some support bolts were sheared on baffle sections which remained in place. Also, the U bolts which anchored the relief valve discharge lines were found stretched and the paint on the torus wall was blistered and removed from the surface directly below the discharge lines. General Electric design engineering analysis attributed the damage to the initial transient dynamic loading caused by the discharge into the torus water. They recommended that the baffles be removed and the relief valve discharge lines

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be extended and terminated with a ram's head type of tee so as to direct the steam away from the torus wall. To date, the baffles have been removed from the torus and have been shipped to an approved radioactive waste disposal site. Work is in progress to complete the recommended modifications, repair the damage to small lines and catwalk bracing, and to repaint the areas affected.

Discussion

On August 9, 1971, it was found that vacuum breaker 2382-H did not respond to test operation. (It was subsequently determined that the air supply line to the test operator was broken.) The torus was entered on November 16, 1971 to determine the cause of the problem; the following damage was reported:

1. Baffle Sections - The baffle section design consists of two I-beams which span from one torus wall to the other just below the normal water level. A channel iron is welded along the length of each I-beam; the wide surface of the channel iron sits in a vertical plane. The two channel irons make contact and are welded to one another. Brackets on the torus wall support the weight of the baffle. On each end of the baffle are four 7/16" bolts connecting the baffle to the bracket. The bolts are sized such that they will fail before the torus wall is overstressed. Of the 72 baffle sections in the torus, 11 were dislodged from the support brackets at either one or both torus walls. A total of 114 bolts were found out of place; this includes bolts from baffles which apparently moved enough to shear the bolts but not enough to be dislodged from the support brackets. The damage was concentrated around the relief valve discharge lines and the HPCI turbine discharge line.

2. Air Supply to Vacuum Breaker 2382-H Actuator - This line was broken off inside the torus near the penetration. It was apparently the result of the impact caused when a baffle section struck the catwalk structural support to which the air supply line was strapped.

3. Drywell Vent Distribution Header Low Point Drain Lines - A 1" line drops from each of 8 low points in the distribution header to below the normal torus water level. One of these lines was found cracked and 5 others were found to be bent significantly. This damage appears to have been caused by the movement of the baffles.

4. Catwalk Bracing - Platforms extending from the torus catwalk to the vacuum breakers are braced by angle iron supports from the lower torus wall area. Four such supports were sheared off and four more were in place but bent out of shape. Again, the displaced torus baffles were assumed responsible for the damage.

5. Scratches on the Torus Wall - As the baffles were moved off the support brackets and slid along the torus floor during steam discharge to the torus, the sharp baffle ends scratched the wall surface. The most apparent result of the scratching was the loss of paint; no significant damage to the metal has been observed to date. In addition, where the relief valve discharge lines directed steam against the torus wall, the paint was damaged. In an elliptical area approximately 10" by 18", the paint was completely removed; over a larger area the paint was blistered. (The HPCI, and the much smaller RCIC, turbine discharge lines direct steam approximately tangential to the torus wall. The paint was not damaged in these areas.)

6. Relief Valve Discharge Pipe Supports - 3/4" U bolts were used to anchor the ends of the 10" relief valve discharge lines to the torus wall. On all occasions the U bolts were stretched and the brackets bent, presumably as a result of piping movement.

On November 16, 1971, NSP management and the Safety Audit Committee were informed of the situation. As a result of these discussions, Dr. J A Thie was asked to be present on the site on November 20, 1971, representing the Safety Audit Committee, to join in the investigation and analysis of the situation.

After inspecting the damage, representatives of the General Electric Company, in a report entitled "Containment Torus Baffle Problem - December 2, 1971", presented the following chronological description of events resulting in damage to the baffles:

1. "When a relief valve is opened, steam quickly pressurizes the fluid in the discharge pipe. This creates a pressure disturbance which propagates through the water in the relief valve piping (i.e. the vent submerges) and then expands outward into the suppression pool. Although the magnitude of this disturbance diminishes rapidly with distance from the pipe exit, it is high enough locally that it could dislodge the baffles nearest the pipe exit if the full pressure difference was sustained across the baffles. However, the time duration of this disturbance is very short and the baffle displacement required to transmit the pressure disturbance to the water on the other side is small so it is likely that the major part of this load will be transmitted across the baffles with no adverse effects.

2. "Following this pressure disturbance and resulting from it is a net movement of the water in the pool outward from the pipe exit. Calculations have shown that the net forces resulting from this fluid motion are insufficient to cause damage to the baffles.

3. "As the steam pressurizes the fluid in the discharge pipe it forcefully expels the water initially in the pipe. When this water exits the pipe (at a considerable velocity) it generates turbulent expansion motion in the surrounding water. However, the resulting forces are well within the design limit of the baffles.

4. "Immediately following this slug of water is a slug of compressed air. This is air which was initially in the relief valve piping and which is compressed by the steam flow. As this mass of compressed air is suddenly injected into the suppression pool it expandes rapidly, displacing the water in the suppression pool to the sides and upwards. The pressure forces acting on the baffles as the displaced water is forced through the baffles are of a sufficient magnitude to displace the baffles. This is expecially true for the baffles which are very close to the pipe exit. Because there is a large free surface area between the baffle nearest the pipe exit and the next baffle, a considerable portion of the displaced water will move upward and the baffles farther away are not likely to see force sufficient to dislodge them.

5. "After the short term transients mentioned above attenuate a steady state steam flow will exist. The fluid motion in the suppression pool resulting from the momentum of the steady state steam jet was examined and found to be insufficient to cause any damage to the baffles."

The General Electric report proceeds to recommend that the baffles be removed.

"Since it has previously been shown that the baffles are not required (Cooper Station Docket No. 50-298, Amendment #1) the recommended action is to remove all the baffles, which is consistent with the present design specifications, i.e. baffles are not being installed on any current plants.

"Suppression chamber baffles were originally included in the design to prevent a short term overpressure (pressure exceeding the end point pressure) of some 6 psig as observed in a series of 1/4 scale tests performed at Moss Landing. The basis for their removal is three fold:

1. The suppression chamber design pressure is 56 psig (based on code allowances for a maximum internal pressure of 62 psig) rather than 35 psig (non-code corrected maximum internal pressure), as was true when baffles were first proposed for Dresden Unit 2 (AEC Docket #50-237). Therefore, even if the observed overpressure were to occur, the design pressure of the suppression chamber would not be exceeded.

2. Convincing evidence exists that the overpressure would not occur in a full scale geometry.

3. The installation of baffles is not required to prevent azimuthal sloshing, uniform distribution, or other fluid perturbations."

As a means of removing the baffles, General Electric recommended cutting the baffles in sections small enough for convenient handling. They further advised, with the support of The Bechtel Corporation, that a hatch be constructed in the 935' elevation reactor building floor directly over the torus hatch.

The Monticello Operations Committee and NSP management reviewed the recommendations for baffle removal, found no unresolved safety questions, and approved the removal procedure on November 26, 1971. Dr. J A Thie was again brought to the site on December 1, 1971, to observe the removal procedure and to inform the Safety Audit Committee of the work progress. The baffle material contained smearable contamination of the order of 1000 dpm; it was therefore wrapped as it was removed from the torus before being shipped from the site. Airborne levels in the torus reached 10^{-9} uc/ml; torus ventilation air was routed through the Standby Gas Treatment System. The damage and the removal of the baffles was under close surveillance of Quality Assurance personnel.

The initial relief valve discharge line design released steam straight down near the inner torus wall, leaving the discharge line four feet below the normal torus water level, about 12 inches from the torus wall. General Electric has recommended a design modification to extend each discharge line to the deepest region of the torus where steam will be injected under about 9 feet of water. At that point a ram's head tee will direct flow so that it does not impinge on the torus walls. The General Electric report provides the following safety evaluation for the modification:

> "The addition of discharge piping would increase the discharge fl/d and could reduce the flow capacity of the relief valve. This has been examined and it has been determined that the additional piping and elbow needed to relocate the point of discharge into the pool will not affect the relief valve flow rate. Presently the relief valve flow rate is limited by the choked flow at the valve itself. This is insured by proper sizing and layout of the discharge piping. For Monticello the fl/d of the discharge piping would have to exceed 6.0 before the flow rate would start being affected. The fl/d of the "as built" piping is conservatively calculated to be 4.50. The additional pipe needed to relocate the discharge increases the fl/d to 5.0, still well below the point where the discharge pipe will start to influence the relief valve flow. Therefore, it is concluded that the proposed fix will not reduce the existing capacity of the relief valve system.

The pipe extension and elbow will increase the pressure in the discharge piping due to the additional fl/d. In the present system the discharge pressure is 150 psi upstream of the exit. Pressure increases to 525 psi at the relief valve discharge and upstream of the relief valve the pressure could be as high as 1200 psi. The new piping will have a discharge pressure of 75 psi (tee has twice the discharge area of previous discharge), 150 psi at the inlet to the tee, 570 psi at the relief valve discharge and 1200 psia upstream. Maximum pressure in the discharge line has increased by only +5 psi and is still within the discharge piping capability of 1200 psi. The short term pressure rise in the discharge piping while the water is being expelled from the discharge line has also been examined and found to be less than the 570 psi that exists during steady flow discharge. Therefore, it is concluded that the new discharge piping does not significantly reduce previous safety margin."

The Monticello Operations Committee and NSP management reviewed the recommendations for discharge line extensions, found no unresolved safety questions, and approved the modification on December 8, 1971. The Safety Audit Committee was informed that day of the proposed modification and that it involved no irretrievable work. The entire subject of torus work is on the agenda for the December 15 and 16, 1971, Safety Audit Committee meeting at the site.

All baffles have been removed and work is currently in progress on the relief valve discharge line modifications. Quality Assurance personnel are placing close surveillance on the torus work activities. A supplementary report of torus work will be issued on completion of the project.

Yours very truly,

L. O. Mayer

L O Mayer, P.E.
Director of Nuclear Support Services

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