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November 7, 1977

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Mr. C. R. Domeck
Nuclear Project Engineer
Toledo Edison Company
Power Engineering & Construction
300 Madison Avenue
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Subject: Toledo Edison Company
REPORT ON DEPRESSURIZATION
Davis-Besse Unit 1
B&W Reference NSS-14

Dear Mr. Domeck:

Our letter BWT-1589 dated November 1, 1977 forwarded input for a TECO report to NRC regarding the depressurization event on September 24. To supplement that letter, you will find attached another writeup which evaluates the reactor coolant components. It may be inserted into the previous report as Attachment B.

Very truly yours,

A. H. Lazar
Senior Project Manager

J. A. Lauer

J. A. Lauer
Project Manager

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Attachment

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ATTACHMENT B

REACTOR COOLANT COMPONENTS

B&W has completed their evaluation of the September 24 incident at Davis-Besse and found no harmful effects were incurred in the reactor vessel steam generator pressurizer and primary piping pressure boundary.

During this rapid depressurization event the reactor coolant pressure dropped from 2300 psig to 930 psig in 7-1/2 minutes and gradually recovered to 1800 psig in 2 hours. During the first 7-1/2 minutes the reactor outlet temperature dropped at varying rates from 580°F to 533°F. For this evaluation it is assumed that the total temperature drop occurred at the initial rate. This results in a 49° temperature drop over a 6 minute period. Approximately 30 minutes after the initial temperature drop a second slower and smaller temperature drop from 540°F to 505°F occurred over a 21 minute period. Following this second temperature ramp, the temperature gradually increased over a 2 hour period to 528°F. The reactor inlet temperature ramps and durations were the same as for the reactor outlet temperature.

The secondary side pressure in steam generator No. 1 reached a maximum of 1050 psig and decreased to 860 psig within 15 minutes and remained at that level. The secondary side pressure in steam generator No. 2 reached a maximum pressure of 980 psig decreased to a minimum of 610 psig in 14 minutes, and returned to 860 psig in 2 minutes. Twenty minutes later the pressure again dropped to 610 psig and recovered gradually over a 2 hour period.

The Design Specification for the Davis-Besse I plant required evaluation of 40 cycles of a rapid depressurization event which included a drop in the reactor coolant pressure from 2200 psi to 800 psi, a drop in the reactor coolant system average temperature from 563°F to 500°F in 15 minutes, and a drop of secondary pressure from 1050 psi to 640 psi.

The major difference between the actual transient and the design transient is the rate of the temperature drop in the reactor coolant system. The actual rate of temperature drop was twice the rate of the design transient but the total temperature change was only 78% of that of the design transient. The net result is that the fatigue usage of this one rapid depressurization is about the same as that predicted for one cycle of the design transient.

As a more direct comparison, the transient event identified was analyzed for the reactor vessel shell and compared to the design transient. The results were that the range in thermal radial gradient stress for the actual transient was 5400 psi and the range of thermal radial gradient stress for the design transient was 6600 psi. This comparison would be representative of other thicknesses throughout the pressure boundary.

The conclusions of the analysis are:

1. Stresses in the pressureboundary did not exceed those already calculated on a design basis. This is verified by the actual pressure not exceeding 2500 psi and the thermal transient being less severe than a combination of design transients for a rapid depressurization and a reactor trip.

2. Fatigue life of the reactor coolant components is not affected if one cycle of the reactor trip design transient and two cycles of a rapid depressurization design transient are considered to be used for this transient. Two cycles of the rapid depressurization transient are necessary because the HPI system was actuated twice during the event and two cycles are necessary to reflect the thermal transient in the high pressure injection nozzle.

   The effect of the entire event on the fatigue life of the steam generators can be accounted for by using one cycle of the design transient for rapid depressurization and one cycle of the design transient for loss of feedwater to one generator.

3. The effect of the change in water level on the pressurizer has a very minor effect on the pressurizer shell stresses. The pressurizer has been previously analyzed for the thermal effect of water-steam interface and the change of level does not affect that analysis.

4. No significant thermal shock should occur to the heaters since the heaters were deactivated due to a low water level sensor and not reactivated till the level recovered.

5. No dynamic effects were caused by the rapid pressure decrease. No specific analysis was done but a dynamic response of the shells would require a large pressure drop in the order of milliseconds and the actual change was on the scale of minutes.

The reduced feedwater flow to steam generator No. 2 was not sufficient to maintain a water level during the first five minutes of the event and this steam generator boiled dry. The primary concern with a dry generator is the tube to shell temperature difference. In this event a water level was established before the system cooldown was started and acceptable tube to shell temperature differences were maintained. This condition is similar to the loss of feedwater design transient followed by restart of a dry pressurized generator using the auxiliary feedwater system.

The burst rupture disc of the pressurizer quench tank resulted in a stream of steam and water impinging on steam generator No. 2. This stream removed a section of insulation 10' high and 20' wide from the lower shell of the generator and impinged directly on the generator shell. The temperature of the impinging water was assumed to be 212°F. A conservative evaluation of the rapid temperature change in this local region of the vessel shell was performed. The results of this evaluation indicate that this one event used less than 1% of the total fatigue life of the vessel. The predicted fatigue usage factor for the 40 year design life of the vessel in this area was less than 0.10. This jet impingement did not significantly reduce the fatigue life of the generator.



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