Babcock & Wilcox

Power Generation Group

P.O. Box 1260, Lynchburg, Va. 24505
Telephone: (804) 384-5111

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BWT-1579
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Mr. C. R. Domeck
Nuclear Project Engineer
Toledo Edison Company
Power Engineering & Construction
300 Madison Avenue
Toledo, Ohio 43652

cc: J. D. Lenardson
J. C. Lewis
D. J. DeLaCroix
E. C. Novak/2c

Subject: Toledo Edison Company
DEPRESSURIZATION EVENT OF SEPTEMBER 24, 1977
Davis-Besse Unit 1
NSS-14

POOR ORIGINAL

Dear Mr. Domeck:

Our letter BWT-1578 dated October 5, 1977 advised that B&W is reviewing the available data regarding the depressurization event of September 24 and we consider that there has been no degradation of safety in the plant. By telecon of October 6, you have advised that NRC would like more information regarding the basis for our conclusions.

The components are designed for forty cycles of a depressurization transient in which the pressure drops 1400 psi and the temperature drops 62°F in fifteen minutes. In this actual transient the pressure dropped 1250 psi and the temperature dropped 45°F in 7.5 minutes. The stresses due to pressure are not sensitive to time in these loading ranges. Since the pressure change of the actual transient is less than the pressure change of the generalized transient, the stress effect due to pressure would be less than calculated for the generalized transient. Although the rate of temperature change is higher in the actual transient, the overall temperature change is 17° less. These two differences tend to offset each other such that the resulting stresses for the actual transient were no worse than the calculated stresses for the design transient.

One steam generator apparently boiled dry during the depressurization event because the auxiliary feedwater turbine failed to come up to speed. The design transients include twenty cycles in which feedwater flow is lost to one generator and the generator is evaporated to a dry pressurized condition. The introduction of feedwater into a dry steam generator is a design condition and will have no harmful effects. The major concern while the steam generator is dry is variation in the tube-to-shell temperature differences. In the actual transient the steam generator was dry for a short period of time and the generator remained pressurized. During this time of approximately 13 minutes, the reactor coolant temperature dropped a total of 50°. This temperature drop was not sufficient to cause excessive stress or permit deformation of the tubes, and is within the established design limits. 8001170836 5

The actual stresses were no worse than the calculated stresses from the design transient and consequently the fatigue usage resulting from the actual transient is no worse than that for the design transients. The predicted fatigue usage for this transient is the same as that of one design cycle of rapid depressurization and one design cycle of startup of a dry steam generator. There is no change in the calculated fatigue life of the components. Since the stresses and deformations resulting from design transient analysis are acceptable, there is no reason to expect overstressing or material deformation in the RC system due to the actual transient. We do not consider it necessary to

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conduct a detailed inspection of hangers and restraints for evidence of damage or deformation.

The reactor coolant pumps were all operated at or near saturation pressure (A2 and B1 for about one minute and A1 and B2 for about 45 minutes). There is some risk of cavitation damage to the impellers and also a risk that saturated steam would cause dry bearings and resulting damage. In addition, radial offsets due to cavitation may damage the seals. All four pumps suffered either loss of or erratic seal injection flow for about 1-3/4 minutes after containment isolation valves were closed. We have reviewed these conditions with the pump manufacturer and concluded that the risk of damage is small. Disassembly and inspection of the seals, bearings, and impellers would not provide 100% assurance that they will operate properly. Therefore, we have recommended that the pumps be instrumented to measure shaft vibration, seal cavity pressures, RC pressure, standpipe leakage, and seal injection flow and temperature. Each pump has been run for two minutes with this instrumentation in Mode 5 and the observed parameters show no indication of damage. We expect to have similar test runs in Mode 3 when the RC pressure is above 1300 psi. If these runs also show no indication of damage, B&W would then recommend that the pumps may be safely operated as designed.

B&W has evaluated the 9/24 incident with regard to its effects upon fuel performance and has concluded that there are no safety concerns with respect to the reactor fuel. This conclusion is based upon the following considerations:

- Prior to the subject transient the reactor had been operating for approximately one week at a maximum of 15% of rated power; immediately prior to trip the power level was approximately 10% of rated power, therefore, the heat generation in the core (decay heat) during the depressurization transient was extremely low and significantly less than that produced by the reactor coolant pumps.

- The core burnup on 9/24 was approximately 1 EFPD.

- During the transient the maximum fuel rod internal pressure has been conservatively estimated to have been no more than 300 psi greater than the minimum RC system pressure; the maximum fuel rod cladding temperature was 550F. The tensile stresses imposed on the cladding as a result of the 300 psi pressure differential existed for less than one hour. For cladding with low irradiation exposure exposed to this temperature/pressure combination no deformation or failure would be predicted.

- Reactor coolant temperature, pressure, and flow rate data obtained during the course of this transient indicated that there was no significant heat generation in the reactor core; this data further indicates that no significant boiling occurred in the core.

Very truly yours,

A. H. Lazar, Senior Project Manager

J. A. Lauer, Project Manager

JAL/hj

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