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OCT 23 1981

RECEIVED OCT 26 1981 U.S. NUCLEAR REGULATORY COMMISSION

Ms. Robin Berger
125 Forestway Drive
Deerfield, Illinois 60015

Dear Ms. Berger:

Thank you for your letter to Thomas Murley regarding your concerns for the safety at the Zion Station. Your letter was forwarded to me since I am the Nuclear Regulatory Commission's (NRC) project manager for licensing matters at Zion.

I must assume that the safety concern you have mentioned is the postulated thermal shock to the reactor vessel following an overcooling transient; we refer to this simply as "thermal shock." For your information, I have enclosed a short synops.s on this issue. The Zion Station vessels have not received the radiation exposure that would make them a safety concern at this time, however, our program is scheduled to resolve the matter before the vessels are susceptable to damage from any overcooling transient. We hope this information will be of benefit to you.

If you have any further questions on the thermal shock issue or any other matter that you feel presents an undue hazard, please let us know. Also for your information, the NRC maintains a resident inspector at the Zion Site; Joel Kohler can be reached on telephone number 312-746-2313.

Sincerely,

Original Signed By:

David Wigginton, Project Manager
Operating Reactors Branch No. 1
Division of Licensing

Enclosure:
As stated

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SURNAME	DWigginton/rs	SVarga				
DATE	10/21/81	10/ /81				

Enclosure

NUCLEAR REACTOR PRESSURE VESSEL INTEGRITY WHEN SUBJECTED TO THERMAL SHOCK AND SUBSEQUENT REPRESSURIZATION DURING AN OVERCOOLING TRANSIENT

(PRESSURIZED THERMAL SHOCK)

Pressure vessel thermal shock has been considered for many years in the context of assuring integrity of the vessel when subjected to cold emergency core cooling water during a large loss of coolant accident (LOCA). Based on a series of thermal shock experiments (unpressurized) conducted at Oak Ridge National Laboratory (ORNL) beginning in 1976 and based on fracture mechanics analyses verified by the experiments, it was concluded that a postulated flaw would not propagate through the vessel wall during a large LOCA. Therefore, the vessel integrity would be maintained during subsequent reflooding which would occur at relatively low pressure due to presence of the large break.

As the result of operating experience, it was subsequently recognized that there could be transients in pressurized water reactors (PWRs) in which the vessel could be subjected to severe overcooling (thermal shock) followed by repressurization. In these pressurized thermal shock transients, vessels would be subjected to pressure stresses superimposed upon the thermal stresses resulting from the temperature difference across the vessel wall. The Rancho Seco event of March 20, 1978 is believed to represent the most severe (and prolonged) overcooling transient experienced to date. In that event, a lightbulb being replaced in the non-nuclear instrumentation/integrated control system (NNI/ICS) panel was dropped and caused a short to occur while the plant was at approximately 70% power. About 2/3 of the pressure, temperature and level indication was lost. The reactor tripped, feedwater was lost and the once through steam generators (OTSGs) dried out. Subsequent refilling by the main feedwater (MFW) system caused a primary system overcooling and an actuation of high pressure injection (HPI) and emergency feedwater

(EFW). Actuation of HPI and EFW caused severe overcooling rates (approximately 300°F/hr) until the pumps were partly secured by plant operators. Actuation of HPI also caused represscrization of the primary system. Operators did not recognize until appr..imately one hour later that primary system temperature had been reduced to about 285°F (because of preoccupation with restoration o. NNI/ICS equ:pment).

If an overcooling event such as that at Rancho Seco in 1978 were to occur even for the vessel with the worst material properties in the current population of reactor vessels, the staff would not expect a failure. The staff conclusion is supported by an analysis of the Rancho Seco event performed by the Oak Ridge National Laboratory which indicated that it would be se. al years before any B&W-designed facility reached the threshold irradiation level for crack initiation (that is, small cracks growing to larger ones assuming conservative initial material properties for pressurized overcooling events equal in severity to the Rancho Seco event). Some reactor vessels in Combustion Engineering (CE) and Westinghouse (W) facilities have somewhat higher irradiation histories; however, other mitigating factors provide a ignificant margin to failure should a pressurized overcooling event similar to that at Rancho Seco occur.

In order to define what transient conditions more severe than the Rancho Seco event would be necessary to propagate a flaw through the entire vessel thickness, a number of investigations were initiated by the staff beginning in early 1980. These investigations included defining the cooldown transients and accidents of interest and their respective probability, development of a computer code to perform the thermal transient and fracture mechanics analyses, and planning for pressurized thermal shock tests in the Heavy-Section Steel Technology Program at ORNL.

The staff evaluations of this analytical work indicated that there could be a problem if pressure vessels having initial material properties (fracture toughness) less favorable than those fabricated more recently were subjected to severe pressurized cooldown transients after many years of neutron irradiation. In order to assess the need for any immediate action, the PWR industry Regulatory Response Groups (RRGs) and PWR reactor manufacturers were briefed on this issue by the staff on March 31, 1981. In a progress briefing on April 29, 1981, the PWR Owners' Group asserted that there was no need for immediate corrective action. On May 15, 1981, the Westinghouse, Combustion Engineering and Babcock & Wilcox Owners' Groups filed written responses supporting and reiterating their conclusion that no immediate action was required on any operating reactor.

The staff has determined that no immediate licensing actions are required for plants under construction, plants under review for operating licenses, or operating facilities; however, the staff has taken the following actions:

1. Meetings have been held on many occasions with industry representatives for detailed discussions and exchanges of information.

2. Evaluations are continuing for refinement of the staff's understanding of this safety concern and better definition of what actions the industry and staff must take to resolve this issue.

A number of efforts are now underway by the NRC staff to develop a better technical basis for a final resolution for this problem. These programs may show the need for more extensive corrective action before vessels approach their end of design life state. A new project has been initiated at Oak Ridge National Laboratory (ORNL) to bring together a comprehensive evaluation of the many aspects of this problem in order to define the best course of regulatory action toward its understanding and resolution. The Heavy-Section Steel Technology Program at ORNL is continuing, and first tests using a new pressurized thermal shock test facility are scheduled for FY1982. The development of a computer code for probabilistic analysis of reactor pressure vessel failure utilizing fracture mechanics and Monte Carlo simulation techniques is continuing.

Several potential corrective actions are possible, and will be considered. These include:

1. Reducing the neutron irradiation of the pressure vessel by replacing some or all of the outer row of fuel elements in the core with partially loaded or reflector elements;

2. Annealing the reactor pressure vessel in-situ to restore a major fraction of the fracture toughness which was lost due to neutron irradiation. Annealing is feasible from a metallurgical standpoint, but practical application is difficult and potentially expensive;

3. Reducing the thermal shock during some transients by raising the temperature of the emergency core cooling system (ECCS) injection water; and

4. Reducing the probability of the event by control system designs that would prevent repressurization, and/or by operator actions to prevent repressurization.

The NRC staff and its contractors have been, and will continue to be, extensively involved in the development of the technology of this issue.

Robin Berger
125 Forestway Drive
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WASHINGTON
PM
27 AUG
1981

Thomas Murley
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~~20014~~