UNITED STATES
NUCLEAR REGULATORY COMMISSION
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MEMORANDUM FOR: Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Edson G. Case, Deputy Director
Office of Nuclear Reactor Regulation

FROM: Darrell G. Eisenhut, Director
Division of Licensing

SUBJECT: THERMAL SHOCK TO PWR REACTOR

In response to E. Case's note to me dated April 16, 1981, we have coordinated an interdivisional technical review of the reactor vessel fracture issue to determine if immediate licensing actions should be required.

Our preliminary review has concluded that although no immediate action is required for operating reactors, the staff should continue to evaluate this issue in the near future with the actions identified herein.

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cc: NRR DIV DIRs

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PRELIMINARY ASSESSMENT OF THERMAL SHOCK TO PWR REACTOR PRESSURE VESSELS

INTRODUCTION

During the past few months the subject of reactor pressure vessel thermal shock has received increased attention by the NRC staff. Most recently, on March 31, 1981, NRC representatives met with the Pressurized Water Reactor (PWR) industry Regulatory Response Groups and the PWR reactor manufacturers. In addition, concerns have been raised regarding the safety of operating reactors.

In order to determine whether any immediate licensing action is necessary relative to the potential for thermal shocks in pressurized water reactor (PWR) pressure vessels, the staff has evaluated (1) the types of transients or accidents that could lead to overcooling of the reactor system; (2) experience to date with transients that have occurred in U.S. PWRs; (3) the probability that such overcooling events will occur; and (4) the capability of reactor vessels to withstand these transients. Item 4 focused on the likelihood of a flaw existing in a reactor vessel (RV), the copper content of RV welds, and the extent of RV irradiation (fluence).

BACKGROUND

Severe reactor-system overcooling events which could be followed by repressurization of the RV can result from a variety of causes. These include instrumentation and control system malfunctions and postulated accidents such as small-break loss-of-coolant accidents (LOCAs), main steamline breaks, or feedwater pipe breaks. Rapid cooling of the RV internal surface causes a temperature distribution across the RV wall. This temperature distribution results in thermal stress, with a maximum tensile stress at the inside surface of the vessel and a compressive stress at the outside surface. These stresses combine with the hoop stress caused by the internal pressure in the vessel. The magnitude of the thermal stress depends on the temperature differences across the RV wall.

As long as the fracture resistance of the RV material remains high, such transients will not cause failure. After the fracture toughness of the vessel is reduced by neutron irradiation, severe thermal transients could cause fairly small flaws near the inner surface to initiate -- and result in -- significant cracking. The vessels of concern are those with a history of high radiation exposure, which are made of material that has a high sensitivity to radiation damage (such as those made with welds of high copper content).

For failure to occur, a number of contributing factors must be present. These factors are: (1) a reactor vessel flaw of sufficient size to propagate, (2) high copper content, (3) a relatively high level of irradiation, (4) a severe overcooling transient with repressurization, and (5) a resulting crack of such size and location that the ability of the RV to maintain core cooling is affected.

EVALUATION

The staff review of overcooling events and their probabilities included a review of the Office of Reactor Research (RES) study on overcooling events at Babcock & Wilcox (B&W) plants (Ref. 1 attached); a survey of operating experience on Westinghouse (W) and Combustion Engineering (CE) plants (Ref. 2); a review of available accident analysis in Final Safety Analysis Reports (FSARs) and in vendor topical reports; and a preliminary probabilistic analysis performed by the Division of Safety Technology (DST) (Ref. 3 attached). The preliminary results of these evaluations indicate that there is a probability of about 10^{-3} per reactor year that a B&W-designed plant will experience a severe overcooling transient similar or greater in magnitude to that experienced at Rancho Seco on March 20, 1978. This transient is the most severe overcooling transient experienced by any PWR in the U.S. This probability of 10^{-3} per reactor year includes contributions from steam generator control system malfunctions (the dominant contributor); small-break LOCAs; main steamline or feedwater line breaks; and complete loss of feedwater flow. The staff estimates that the probability of such an overcooling event in CE or W-designed reactors is lower, perhaps by an order magnitude, than for B&W-designed reactors. This difference is based on design differences and on operating experience.

In the 1978 Rancho Seco transient, reactor pressure was maintained at a fairly high level (1500 psig to 2100 psig) throughout the cooldown. The minimum temperature of the reactor coolant (280°F) during the transient was high enough to maintain the material toughness of the reactor vessel. Moreover, this evaluation leads the staff to believe that if this transient were to be repeated at Rancho Seco or any other B&W-designed facility within the next few years, the RV failure would still be unlikely. Nonetheless, the possibility of vessel failure as a result of an overcooling event cannot be completely ruled out. If an overcooling event such as that at Rancho Seco were to occur, based on the many factors pertinent to an analysis of vessel failure, the staff would expect much less than one failure in the current population of reactor vessels. Even for the vessel with the worst material properties, the staff would not expect a failure.

The staff conclusion is supported by the analyses of the Rancho Seco event performed by the Oak Ridge National Laboratory (ORNL) (Ref. 4 attached). The ORNL analyses indicate that the threshold irradiation level for crack initiation (that is, small cracks growing to larger ones assuming conservative initial material properties such as $RT_{NDT} = 40°F$ and copper content of 0.35%) would be in the range of 0.75×10^{19} to 1.7×10^{19} neutron/cm^2. The highest fluence to date in a B&W-designed facility is less than half the minimum value listed above. It would, therefore, be several years before any B&W-designed facility reached its threshold irradiation level.

Some reactor vessels in CE and W facilities have somewhat higher fluences; however, other mitigating factors -- such as lower values of initial RT_{NDT} -- provide a significant margin to failure should an overcooling event similar to that at Rancho Seco occur.

CONCLUSIONS AND RECOMMENDATIONS

As a result of its evaluations to date, the staff has concluded that the probability of a severe overcooling transient (similar in magnitude to the Rancho Seco event) is relatively low. For B&W-designed reactors this probability is estimated to be about 10^{-3} per reactor per year, and for W- and CE- designed reactors, it is lower, perhaps by an order of magnitude. In addition, the staff has concluded that, based on present irradiation levels at operating reactors, RV failure from such an event is unlikely. Accordingly, the staff believes that no immediate licensing actions are required on operating reactors; however, the staff recommends that the following actions be taken:

1. Request industry representatives meet with the NRC staff in the near future to discuss:

   a. industry progress since the March 31, 1981 meeting
   b. bases for continued safe operation
   c. the letter of April 10, 1981 from D.L. Basdekas to Chairman Udall (Ref. 5 attached)

   The Division of Licensing has requested such a meeting with the PWR vendors and Owners Groups. The meeting is scheduled for April 29, 1981.

2. The staff should continue to refine its understanding of this safety concern. This continuing assessment, taken together with information being provided by industry Owners Groups (including the Owners Group Action Plan due May 15, 1981) should permit the staff to define what

actions the industry and the NRC must take to resolve this safety concern. The staff's efforts during this short term should include, but not necessarily be limited to, the following areas:

a. Development of a better understanding of overcooling transients and accidents. Factors to be examined or addressed in this continuing evaluation would include:

   (1) human factors considerations

   (2) refinements in the analysis of the probability of such events occurring, including considerations of overcooling events more severe than at Rancho Seco

   (3) an understanding of improvements in instrumentation and control systems implemented since the event at Rancho Seco and other overcooling events and the effects of these improvements on the probability of overcooling events.

b. Development of a better understanding of the potential for and effects of RV thermal shock including:

   (1) a categorization of the susceptibility of operating RVs to cracking as a result of rapid cooling, considering the combination of irradiation levels, vessel impurity content, and existing flaw sizes

   (2) a sensitivity study of the effects of fluid mixing and the development of realistic models and assumptions.

c. An assessment of further requirements and of the overall contribution to safety of potential improvements.

d. An overall integrated assessment and report of conclusions and recommendations developed in connection with the above items.

REFERENCES

1. "Insights on Overcooling Transients in Plants with the B&W NSSS," M. Taylor to S. Fabic, dated October 29, 1980.

2. Nuclear Power Experience 1980, Bernard J. Verna, Publisher; Nuclear Power Experience, Inc., Encino, CA.

3. Frequency of Excessive Cooldown Events Challenging Vessel Integrity. A. Thadani to G. Lainas, dated April 21, 1981.

4. Parametric Analysis of Rancho Seco Overcooling Accidents, ORNL letter, R.D. Cheverton to M. Baginis (NRC, RFS), 3/3/81.

5. Letter from D. Basdekas to The Honorable Morris K. Udall, dated April 10, 1981.