ACRSR-1951

July 23, 2001

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: CIRCUMFERENTIAL CRACKING OF PWR VESSEL HEAD PENETRATIONS

Dear Chairman Meserve:

During the 484th meeting of the Advisory Committee on Reactor Safeguards, July 11-13, 2001, we heard presentations by and held discussions with representatives of the NRC staff and the Electric Power Research Institute (EPRI) Materials Reliability Program regarding industry and staff actions relative to cracking and leaking observed in pressurized water reactor (PWR) Alloy 600 reactor vessel head penetrations, including control rod drive mechanism (CRDM) nozzles. This matter was also discussed during a July 10, 2001, meeting of the Materials and Metallurgy and the Plant Operations Subcommittees. During our reviews, we had the benefit of the documents referenced.

Conclusions and Recommendations

1. The decision to issue a bulletin addressing the recent incidents of circumferential cracking of CRDM nozzles in U.S. PWRs is timely and appropriate.

2. The staff should urgently address technical issues associated with risk assessment, the effectiveness of inspection techniques, and the completeness of damage accumulation prediction.

Discussion

Cracks were recently detected during inspections of CRDM nozzles at Oconee Units 1, 2, and 3 and Arkansas Nuclear One (ANO) Unit 1. Preliminary risk assessment indicates that the issuance of a bulletin is appropriate to request operational information from the licensees as soon as possible.

The staff's in-depth analysis has raised a number of technical concerns. Although plans are in place to resolve them, the following concerns are of particular importance:

- Risk Assessment

  The risk assessment activities should be expanded to include rod ejection with coincident small-break loss of coolant accident and potential damage to adjacent control rods.

- Prioritization of Inspection Schedules

  Inspection schedule prioritization during the upcoming refueling outages will be based on an analysis of the susceptibility of cracking of CRDM nozzles in different plants. This approach relies on the assumption that susceptibility is determined by time of service and vessel head temperature. This has led to the grouping of each PWR into one of four “bins.” The 14 reactors in the two highest susceptibility bins should receive highest priority in inspections of all CRDM nozzles in 2001. Although this approach is reasonable from a technical standpoint at present, its accuracy will become apparent as inspections proceed. It is prudent to consider potential modifications to this methodology including the following:

  (a) The cracking susceptibility will depend on other conjoint plant-specific factors that can affect cracking and that are not considered explicitly in the current susceptibility algorithm, which addresses only vessel head temperature and operating time. These further factors include residual stress, material composition, heat treatment, welding practices, and local chemical environment.

  (b) As more information on the cracking of CRDM nozzles accumulates from the upcoming U.S. inspections and from past observations overseas, the basis for a risk-informed methodology may be formulated.

  The staff should be prepared to modify any proposed inspection program and timing depending on the results of inspections of the first group of plants (i.e., Fall 2001). These early inspection results may show that it is imperative to inspect the vessel heads of the remaining pressurized water reactors promptly. On the other hand, they may show that it is appropriate to delay the inspections of the remaining plants to allow improvements in diagnostic capabilities.

- Inspection Methods

  The current visual inspection process, which relies on detecting boron crystals at the top of the annulus, indicates the possible presence of circumferential cracks at the base of the annulus, but gives no information on the size and/or orientation of these cracks in the Alloy 600 material. In addition, the absence of visible boron crystals does not give complete assurance that a concentrated chemical environment at the annulus does not exist, resulting in the rapid growth of a circumferential crack. This concern could be addressed during the fall outage by a full volumetric inspection of all CRDM nozzles (i.e., including those with no boron crystals) at Oconee Units 1, 2, and 3, and ANO Unit 1. Volumetric inspections by a qualified process in such cases makes abundant sense. Assessment of the inspection methods used to detect and size cracks in CRDM nozzles and nozzle welds is necessary, especially for the circumferential cracks initiating at the base of the annulus between the CRDM nozzles and the pressure vessel head.

- Inspection Periodicity

  The inspection intervals once cracks are detected depend on knowledge of crack propagation rates as a function of the local material, environmental, and stress conditions. There are data for Alloy 600 cracking as a function of stress intensity and the temperature of the PWR primary coolant. Also, there are limited data relevant to the axial cracking in the Inconel 182 J-weld connecting the CRDM nozzle to the vessel head. The quality of these data is being evaluated by separate expert committees convened by industry and the staff. There is no similar data set relevant to the circumferential cracks that initiate in and adjacent to the J-weld and that present the greatest potential structural integrity concern. The reason for this lack of cracking data is that the local environment in the annulus between the pressure vessel and the CRDM nozzle is not known with sufficient certainty. This problem is also being addressed by the staff.

Consideration of the above issues in conjunction with the issuance of the bulletin should ensure that this matter is satisfactorily addressed for the short term. The Committee wishes to be updated once the licensee responses to the bulletin are evaluated.

A crucial issue confronted in the proposed bulletin is the urgency of inspections of vessel head penetrations, especially for plants thought to be less susceptible to CRDM stress corrosion cracking. Risk would be the metric best suited for determining the urgency. Unfortunately, neither the NRC's phenomenological capabilities, such as the ability to predict time-dependent stress corrosion cracking, nor the NRC's risk assessment capabilities are sufficiently developed at this time to provide defensible bases for decisions on the urgency of vessel head inspections. Sustained research to better the agency's integrated capabilities in probabilistic fracture mechanics and risk assessment will be needed to assist NRC in confronting future issues of reactor coolant system degradation.

Dr. William J. Shack did not participate in the Committee's deliberations regarding this matter.

Sincerely,

/RA/

George E. Apostolakis
Chairman

References:
1. Letter dated June 29, 2001, from A. Marion, Nuclear Energy Institute, to Brian W. Sheron, Office of Nuclear Reactor Regulation, NRC, Subject: Response to June 22, 2001, letter from Dr. Brian Sheron (NRC) to Mr. Alex Marion (NEI) transmitting NRC staff questions on EPRI Interim Report TP-1001491, Part 2 (Proprietary).
2. U. S. Nuclear Regulatory Commission Proposed Bulletin 2001-XX, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," dated June 25, 2001.
3. Memorandum dated June 21, 2001, from C. E. Carpenter, Office of Nuclear Reactor Regulation, NRC, to W. Bateman, Office of Nuclear Reactor Regulation, NRC, Subject: Summary of June 7, 2001, Meeting with the EPRI Materials Reliability Program on Generic Activities Related to CRDM Cracking.
4. U. S. Nuclear Regulatory Commission Information Notice 2001-05: "Through-Wall Circumferential Cracking of Reactor Pressure Vessel Head Control Rod Drive Mechanism Penetration Nozzles at Oconee Nuclear Station, Unit 3," dated April 30, 2001.
5. Electric Power Research Institute, TP-1001491, Part 2, "PWR Materials Reliability Program, Interim Alloy 600 Safety Assessments for US PWR Plants (MRP-44)," Interim Report, May 2001.
6. Letter dated April 17, 2001, from Brian W. Sheron, Office of Nuclear Reactor Regulation, NRC, to Alex Marion, Nuclear Energy Institute, Subject: Issues to be Addressed in a Generic Justification for Continued Operation of PWRs.
7. U. S. Nuclear Regulatory Commission, NUREG/CR-6245, "Assessment of Pressurized Water Reactor Control Rod Drive Mechanism Nozzle Cracking," October 1994.
8. Letter dated June 1, 2001, from Alex Marion, Nuclear Energy Institute, to Brian Sheron, Office of Nuclear Reactor Regulation, NRC, regarding NRC's Assessment of Topical Report MRP-44 - Summary of NRC/NEI Telecon of May 30, 2001.
9. Letter dated May 18, 2001, from Alexander Marion, Nuclear Energy Institute, to Brian Sheron, Office of Nuclear Reactor Regulation, NRC, Subject: PWR Reactor Pressure Vessel Head Penetrations, dated, May 18, 2001.
10. Letter dated December 11, 1998, from David Modeen, Nuclear Energy Institute, Subject: Responses to NRC Requests for Additional Information on Generic Letter 97-01.
11. U. S. Nuclear Regulatory Commission Generic Letter 97-01, Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations, dated April 1, 1997.
12. Letter dated November 19, 1993, from William Russell, Office of Nuclear Reactor Regulation, NRC, to W. Rasin, Nuclear Utility Management and Resources Council (now NEI), transmitting the Safety Evaluation for Potential Reactor Vessel Head Adaptor Tube Cracking.
13. Generic Letter 88-05, Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants, March 1998.