DCS MS-016

Docket Nos. 50-282
and 50-306

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Mr. D. M. Musolf
Nuclear Support Services Department
Northern States Power Company
414 Nicollet Mall
Midland Square - 4th Floor
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Dear Mr. Musolf:

Subject: NUREG-0737 Item, II.K.2.13, "Thermal-Mechanical Report"
Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2

We have completed the review of licensee submittals concerning NUREG 0737 Item II.K.2.13, "Thermal-Mechanical Report."

We have concluded that the information submitted adequately demonstrates reasonable assurance that vessel integrity is maintained for a II.K.2.13 event and have found that the requirements set forth in NUREG-0737 Item II.K.2.13 have been satisfied; therefore, this item is considered complete. Our Safety Evaluation is enclosed.

The issues related to Item II.K.2.13 were studied as a sub-set of Unresolved Safety Issue (USI) A-49, "Pressurized Thermal Shock," and our conclusions are based on findings related to USI A-49. The staff is currently completing work on USI A-49 and is also studying Decay Heat Removal as USI A-45. Should the resolution of either of these USIs result in any change to the conclusions provided in the enclosed Safety Evaluation or require any additional actions related to Item II.K.2.13, we will notify you.

Sincerely,

Original Signed by J. R. Miller

James R. Miller, Chief
Operating Reactors Branch #3
Division of Licensing

Enclosure: Safety Evaluation
Concerning NUREG 0737 Item, II.K.2.13,
"Thermal-Mechanical Report"

cc w/enclosure:
See next page

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION CONCERNING NUREG-0737 ITEM II.K.2.13, THERMAL-MECHANICAL REPORT -- EFFECT OF HIGH PRESSURE INJECTION ON VESSEL INTEGRITY FOR SMALL-BREAK LOSS-OF-COOLANT ACCIDENT WITH NO AUXILIARY FEEDWATER FOR ALL OPERATING PRESSURIZED WATER REACTOR PLANTS

BACKGROUND

The accident at Three Mile Island Unit 2 (TMI-2) on March 28, 1979, involved a main feedwater transient coupled with a stuck-open pressurizer power-operated relief valve and a temporary failure of the auxiliary feedwater system. The resulting severity of the ensuing events and the potential generic aspects of the accident on other operating reactors led the NRC to initiate prompt actions to: (a) assure that other reactor licensees, particularly those with plants similar in design to TMI-2, took the necessary action to substantially reduce the likelihood for TMI-2 type events, and (b) investigate the potential generic implications of this accident on other operating reactors.

TMI Action Plan (references 1 and 2) Item II.K.2.13, titled "Thermal-Mechanical Report," was one of the generic issucs which resulted from the NRC review of, and subsequent actions taken following, the accident.

IE Bulletins 79-05 and 79-06 were issued to Babcock and Wilcox (B&W) licensees and to the other PWR licensees, respectively, in April 1979. These bulletins were supplemented in order to either provide new information, to clarify the original bulletins, or to request other actions or information. These supplements were 79-05A, 79-05B, 79-05C, 79-06A, 79-06B, and 79-06C. The text of these bulletins may be found in reference 3.

The key issues, relevant to II.K.2.13, identified in these bulletins were to maintain high pressure safety injection (HPI) for at least 20 minutes (bulletin series A and B), and to trip all reactor coolant pumps (RCPs) upon HPI initiation on low reactor coolant system pressure (bulletin series C). The requirement to maintain HPI for 20 minutes was withdrawn in bulletins 79-05C and 79-06C, in July 1979.

Consideration of the TMI-2 accident as a small break LOCA with extended loss of all feedwater, coupled with the injection of cold HPI into a potentially stagnant reactor coolant system, gave rise to the concern identified as the Thermal-Mechanical Report, II.K.2.13.

The NRC position taken was that:

"A detailed analysis shall be performed of the thermal-mechanical conditions in the reactor vessel during recovery from small breaks with an extended loss of all feedwater." (reference 1)

This position was later clarified as:

"The position deals with the potential for thermal shock of reactor vessels resulting from cold safety injection flow. One aspect that bears heavily on the effects of safety injection flow is the mixing of safety injection water with reactor coolant in the reactor vessel. . . . . PWR vendors are also required to address this issue with regard to recovery from small breaks with an extended loss of all feedwater. In particular, demonstration shall be provided that sufficient mixing of the cold high-pressure injection (HPI) water with the reactor coolant would occur so that significant thermal shock effects to the vessel are precluded." (reference 2)

The potential for thermal shock of reactor vessels was later broadened in scope to include all over-cooling events and has been identified, and studied, as Unresolved Safety Issue A-49, "Pressurized Thermal Shock." The specifics of II.K.2.13 have been included in these studies.

DISCUSSION

The PWR Owners Groups responses to II.K.2.13 were provided in references 4, 5 and 6. The licensees covered by these responses are listed in Tables 1, 2, and 3.

The Babcock and Wilcox Owners Group (BWOG) and Combustion Engineering Owners Group (CEOG) reports dealt specifically with the Thermal-Mechanical Report issue. The Westinghouse Owners Group (WOG) report was broader in scope and was the first attempt at addressing the general Pressurized Thermal Shock (PTS) issue.

The analyses provided by the Owners Groups were based on conservative thermal-hydraulic models. Input options and assumptions were selected to enhance the overcooling of the reactor vessel. Thermal mixing of the cold safety injection water was considered by employing some simplified mixing models, again selecting conservative parameters. Deterministic fracture mechanics models were used, based on end-of-life fluence and material properties, to evaluate the vessel integrity. The analyses concluded was that vessel failure (e.g. a through-wall crack) would not occur for the II.K.2.13 event. Two predominant issues surfaced concerning these analyses.

The first issue was related to the thermal mixing concern, the fundamental concern which led to the development of II.K.2.13. Since the thermal-hydraulic models did not consider multi-dimensional effects in the reactor vessel, nor did these models consider flow stratification or stagnation of the fluid in the cold leg piping, how good were the mixing models being used? No experimental data was available for the expected flow conditions and for the PWR geometries to verify these mixing models.

The second issue was related to the conservative nature of the analyses. By selectively enhancing the overcooling and causing a rapid transient event, and considering the importance of the time dependent pressure and temperature histories on the deterministic fracture mechanics analyses, how good was the conclusion of no vessel failure (e.g. a through-wall crack)? Would changes in the pressure and temperature histories result in a different conclusion? A deterministic fracture mechanics calculation, based on a given pressure and temperature history, may result in either a crack or a no-crack conclusion.

The thermal mixing concern was investigated by the industry through the Electric Power Research Institute (EPRI). EPRI investigated, using 1/5-scale experimental models, the thermal mixing of the cold HPI water with the warm water in both the cold leg piping and the reactor vessel downcomer for each of the three PWR vendor geometries. A wide range of HPI flow rates, injection locations, and loop flow rates (including zero loop flow) were studied. For the B&W design, flow from the vent valves into the downcomer was included. The experiments were performed by Creare Incorporated and have been commonly referred to as the Creare/EPRI thermal mixing data (references 7 through 12).

These data were used by the staff to develop an empirical mixing model which could be used to describe the thermal mixing of the cold HPI fluid with the reactor coolant system fluid (references 13 and 14). This model calculates the time dependent temperature history at any point in the reactor vessel downcomer (e.g. at the inner vessel surface where a critical weld occurs). Additional investigators have independently verified, and further enhanced, this model for use in the PTS program (reference 15).

Deterministic fracture mechanics analysis techniques (references 16 and 17), were modified by the staff to treat the fracture mechanics as a probabalistic assessment of through-wall cracking. A Monte Carlo simulation, which samples the vessel material property and fluences, was used to obtain the conditional probability of through-wall cracking for a stylized thermal-hydraulic transient. The methodology, refered to as the VISA model, is described in Appendix H to SECY-82-465 (reference 18).

The improvements in the understanding of the thermal mixing issue, as a result of EPRI test data, and the advancements in the area of fracture mechanics, as a result of the staff efforts with the VISA model and with the PTS program, have provided the information needed to complete the review of II.K.2.13, the Thermal-Mechanical Report issue.

SUMMARY

The following points summarize the finding of the investigations into the thermal mixing issue:

(1) The cold HPI fluid, even under the condition of no loop flow, does not behave as a perfectly stratified fluid sliding along the bottom of the cold leg and falling along the length of the downcomer exposing the vessel wall or critical weld to severe cooling and thermal stress. It was this perception that led to the development of the II.K.2.13 issue.

(2) Loop flow rates of only a few times that of the HPI flow rate are adequate to significantly reduce the cooling effects. A regional, mean-mixed thermal mixing model can be used to describe the temperature history.

(3) Under very low or zero loop flow rate conditions, stratification does control the temperature response. However, as a result of stratification, large thermal circulation paths are established and the HPI mixes with the reactor coolant system fluid in the loop seal, cold leg, vessel downcomer and vessel lower plenum. As a result of the system thermal inertia, due to the large fluid volume, the global cooldown is rather slow. While the stratified fluid layer temperature may be about 50°F lower than the mixed fluid temperature near the downcomer entrance, the vessel wall temperature in the areas of interest (one or two pipe diameter lengths from the entrance) are representative of the mixed fluid temperature.

(4) The B&W vent valves provide a source of heated water flowing directly to the upper downcomer for mixing with the cold leg fluid. As a result the cooldown is of longer duration and reduces the potential for loss of vessel integrity for a II.K.2.13 event.

(5) Application of these mixing models resulted in a better, more realistic estimate of the temperature history at the critical weld location.

The following points summarize the findings of the investigations into the fracture mechanics area:

(1) The transient cooldown characteristics for the II.K.2.13 event can be described by a stylized thermal model (exponential cooldown) used in the probabalistic fracture mechanics studies. (See Appendix H of reference 18.)

(2) The deterministic fracture mechanics analyses provided by the licensees show no loss of reactor vessel integrity as a result of a II.K.2.13 event for plant-specific end-of-life vessel material properties. This was shown for both the conservative analyses and for revised analyses based on the new mixing models.

(3) The staff has developed a proposed screening criteria for the Pressurized Thermal Shock issue, which was supported in part by the probabalistic fracture mechanics studies reported in U. S. Nuclear Regulatory Commission Policy Issue Paper on Pressurized Thermal Shock, SECY-82-465, dated November 23, 1982. The II.K.2.13 event, based on the thermal mixing models described, was included in the studies. A separate evaluation was performed for B&W (reference 19) using the same methodology. No change to the proposed screening criteria resulted. The proposed screening criteria are stated in terms of the vessel properties. The nil-ductility transition reference temperature is used. The values proposed are 270°F for longitudinal welds and 300°F for circumferential welds.

(4) The conditional probability of a through-wall crack, for a vessel at the screening criteria, as a result of a II.K.2.13 event was found to be less than one in one hundred (given the occurrence of the event). If the operator were to intervene and either limit repressurization or throttle HPI, this probability would be lowered. The staff estimates the probability of a II.K.2.13 event to be on the order of one in ten-thousand per reactor year for Westinghouse or Combustion Engineering plants, and one in one-hundred thousand per reactor year for Babcock and Wilcox plants.

CONCLUSIONS

TMI Action Item II.K.2.13, the Thermal Mechanical Report, resulted from the staff review of the TMI-2 accident and the staff investigations of the potential generic implications of this accident (references 1, 2, and 3).

The combined concerns related to (1) auxiliary feedwater system availability and reliability, (2) loss of forced coolant flow due to tripping all RCPs, and (3) extended HPI injection into a stagnant reactor coolant system (because of the loss of the heat sink and the loss of the RCPs), during a small-break LOCA, suggested that a potentially unanalyzed safety issue existed which could result in the loss of reactor vessel integrity. The vessel integrity issue was later broadened in scope and identified as Unresolved Safety Issue A-49, Pressurized Thermal Shock (PTS).

The staff review of the initial industry responses to II.K.2.13 (references 4, 5 and 6) resulted in a significant research effort, on the part of the industry, to understand the thermal mixing issue (references 7 through 15). In addition, a probabalistic fracture mechanics model (references 16 through 19) was developed, by the staff, to supplement the deterministic fracture mechanics models and to study the impact of uncertainties in both the thermal-hydraulic data and the reactor vessel material data.

The industry responses to II.K.2.13, coupled with the experience gained through the PTS program and with changes in requirements concerning HPI operation, are judged by the staff to be adequate in demonstrating vessel integrity. Deterministic fracture mechanics analyses have demonstrated no loss of vessel integrity at end-of-life condition for a II.K.2.13 event. A probabilistic assessment indicated that the conditional probability of through-wall cracking, given a II.K.2.13 event, is less than one in one hundred occurrences. This probability is sufficiently low within the context of the proposed PTS rule. That is, the probability of a through-wall crack due to a II.K.2.13 event is on the order of one in one-million reactor years. A through wall crack does not necessarily lead to loss of vessel integrity (for example, the crack size may be small enough to allow the safety injection systems to maintain core cooling).

On the basis of the above, the staff concludes that the information provided by the licensees is adequate in demonstrating reasonable assurance that vessel integrity is maintained for a II.K.2.13 event. The staff finds that all PWR licensees have satisfied the requirements set forth in TMI Action Plan Item II.K.2.13.

Dated: JUN 11 1984

Principal Contributor: E. Throm

Table 1

Babcock and Wilcox (BWOG)

Plant	Docket
Arkansas 1	50-313
Crystal River 3	50-302
Davis Besse	50-346
Oconee 1	50-269
Oconee 2	50-270
Oconee 3	50-287
Rancho Seco	50-312
TMI-1	50-289

Table 2

Combustion Engineering (CEOG)

Plant	Docket
Arkansas 2	50-368
Calvert Cliffs 1	50-317
Calvert Cliffs 2	50-318
Fort Calhoun	50-285
Maine Yankee	50-309
Millstone 2	50-336
Palisades	50-255
San Onofre 2	50-361
San Onofre 3	50-362
St. Lucie 1	50-335
St. Lucie 2	50-389

Table 3

Westinghouse (WOG)

Plant	Docket
Beaver Valley 1	50-334
Cook 1	50-315
Cook 2	50-316
Diablo Canyon 1	50-275
Farley 1	50-348
Farley 2	50-364
Ginna	50-244
Haddam Neck	50-213
Indian Pt. 2	50-247
Indian Pt. 3	50-286
Kewanee	50-305
McGuire 1	50-369
North Anna 1	50-338
North Anna 2	50-339
Point Beach 1	50-266
Point Beach 2	50-301
Prairie Island 1	50-282
Prairie Island 2	50-306
Robinson 2	50-261
Salem 1	50-272
Salem 2	50-311
San Onofre 1	50-206
Sequoyah 1	50-327
Summer 1	50-395
Surry 1	50-280
Surry 2	50-281
Trojan	50-344
Turkey Pt. 3	50-250
Turkey Pt. 4	50-251
Yankee Rowe	50-029
Zion 1	50-295
Zion 2	50-304
McGuire 2	50-370
Sequoyah 2	50-328

References

1. "NRC Action Plan Developed As A Result of the TMI-2 Accident," U.S. Nuclear Regulatory Commission, NUREG-0660, August 1980.

2. "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, NUREG-0737, November 1980.

3. "Report of the Bulletins and Orders Task Force," U.S. Nuclear Regulatory Commission, NUREG-0645, January 1980 (Vol. II, Appendix A)

4. "Thermal-Mechanic Report - Effect of HPI on Vessel Integrity for Small Break LOCA Event with Extended Loss of Feedwater," Babcock and Wilcox, BAW-1648, November 1980.

5. "Evaluation of Pressurized Thermal Shock Due to Small Break LOCAS with Loss of Feedwater for the Combustion Engineering NSSS," Combustion Engineering, CEN-189, December 1981.

6. "Summary Report on Reactor Vessel Integrity for Westinghouse Operating Plants," Westinghouse, WCAP-10019, December 1981.

7. "Fluid and Thermal Mixing in a Model Cold Leg and Downcomer With Vent Valve Flow," EPRI, NP-2227, March 1982.

8. "Thermal Mixing in a Model Cold Leg and Downcomer With Loop Flow," EPRI, NP-2312, April 1982.

9. "Evaluation of Thermal Mixing Data From a Model Cold Leg and Downcomer," EPRI, NP-2773, December 1982.

10. "Thermal Mixing in a Model Cold Leg and Downcomer at Low Flow Rate," EPRI, NP-2935, March 1983.

11. "Thermal Mixing in a Rectangular Geometry Model of a Cold Leg with High Pressure Injection and a Downcomer, EPRI, NP-2924, March 1983.

12. "Transient Cooldown in a Model Cold Leg and Downcomer," EPRI, NP-3118, May 1983.

13. "An Approximate Prediction of Heat Transfer During Pressurized Thermal Shock," Levy, S., S. Levy Incorporated, Report SLK-8213, June 1982.

14. "An Approximate Prediction of Heat Transfer During Pressurized Thermal Shock with No Loop Flow and Metal Heat Addition," Levy, S., J.M. Healzer, S. Levy Incorporated, Report SCI-8220 (Rev. 1), August 1982.

15. "Decay of Buoyancy Driven Stratified Layers with Application to PTS," Theofanous, T.G., et. al, Purdue University, U.S. Nuclear Regulatory Commission, Eleventh Water Reactor Safety Information Meeting, NUREG/CP-0048, October 1983.

16. "OCA-I, A Code for Calculating the Behavior of Flaws on the Inner Surface of a Pressure Vessel Subjected to Temperature and Pressure Transients," Iskander, S.K., et. al., Oak Ridge National Laboratory, NUREG/CR-2113 (ORNL/NUREG-84), August 1981.

17. "Modification of OCA-I for Application to a Reactor Pressure Vessel with Cladding on the Inner surface," Sauter, A., et. al., Oak Ridge National Laboratory, NUREG/CR-3155 (ORNL/TM-8649), May 1983.

18. "Pressurized Thermal Shock," U.S. Nuclear Regulatory Commission, Policy Issue, SECY-82-465, November 23, 1982.

19. "PTS Screening Criteria for B&W Plants," U.S. Nuclear Regulatory Commission, Memorandum from H.R. Denton to V. Stello, Jr., June 30, 1983.