OCT 27 1983

MEMORANDUM FOR: C. J. Heltemes, Jr., Director
Office for Analysis and Evaluation
of Operational Data

FROM: Harold R. Denton, Director
Office of Nuclear Reactor Regulation

SUBJECT: ABNORMAL OCCURRENCE REPORT TO CONGRESS
FOR THIRD QUARTER CY 1983

As requested in your memorandum of October 12, 1983, we have reviewed the proposed items for inclusion to the Abnormal Occurrence Report to Congress for Third Quarter CY 1983. Our comments are included in the enclosure.

Original Signed by
H. R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosure:
As stated

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REPORT
TO CONGRESS

ENCLOSURE

POSSIBLE ABNORMAL OCCURRENCES

1. Extensive Pipe Cracking in BWRs

We agree that a new abnormal occurrence (AO) report on BWR pipe cracks should be proposed. The magnitude and severity of pipe cracks that we have been observing since Nine Mile Point Unit 1 are quite different from those originally reported in AO 75-7. Therefore a new AO is justified. We propose the following:

Observations beginning in March 1982 at Nine Mile Point Unit 1 (See NUREG-0090, Vol. 5, No. 2) were the first examples of major cracking in large diameter piping in the United States (cracking in large diameter piping has been reported in some foreign reactors). At Nine Mile Point Unit 1, extensive intergranular stress corrosion cracking (IGSCC) was found in heat affected zones near weld areas of the large diameter reactor coolant recirculation system. The licensee decided to replace the recirculation piping in all five recirculation loops, all ten safe ends, and branch piping, as warranted.

As discussed in succeeding updated reports to NUREG-0090, Vol. 5, No. 2, the NRC issued Inspection and Enforcement Bulletin No. 82-03, Revision 1 (Ref. B-1) in October 1982 for action by nine BWR plants scheduled for refueling outages in late 1982 and early 1983. Inspections pursuant to this bulletin showed cracking in five of the first seven plants examined, prompting issuance of Inspection and Enforcement Bulletin No. 83-02 in March 1983 (Ref. B-2). This bulletin required augmented inspection of welds in the recirculation system piping, using ultrasonic testing (UT) inspection procedures of demonstrated effectiveness, for all plants beyond those identified in Bulletin No. 82-03, Revision 1, at their next refueling or extended outage but no later than January 1984. No indications of pipe cracking were found at Quad Cities 1, Millstone 1, Oyster Creek, Big Rock and Duane Arnold. At Fitzpatrick one defect was characterized as probably due to IGSCC; however, after multiple inspections the defect was determined to be well within NRC acceptance criteria for continued operation without repair.

In conjunction with these bulletins, joint efforts by the NRC and industry have been underway to train and qualify inspection personnel, using improved UT procedures on well-characterized pipe cracks in pipe segments removed from Nine Mile Point Unit 1, to assure higher reliability in the inspection process. Although this has considerably upgraded the reliability of UT in crack detection field situations, there still remains concern about the ability of current UT procedures, in field situations, to adequately characterize the depth of identified cracks although it is believed that the discovery of cracking, where it exists, is probable.

Inspections conducted in response to these bulletins, and other inspections, have revealed extensive cracking both in large diameter recirculation and residual heat removal (RHR) system piping welds. In Orders issued to certain plants on August 26, 1983, as discussed below, inspections were mandated for susceptible systems for 4" diameter and larger pipes.

Table B-1 is a summary of the cracking observations from BWRs where piping has been examined and defects found. The summary is as of mid-August 1983 and indicates the extent of cracking in large diameter recirculation and RHR system piping. The total number of welds range from 100 to 135 per plant.

Although IGSCC in the sensitized material of the heat-affected zone in BWR piping is influenced by the environmental conditions existing in the BWR reactor coolant system and stresses in the piping, including residual stresses induced by welding, there is no clear correlation between extent of cracking and operating time. Some plants with a relatively brief operating history, e.g., Hatch Unit 2, show extensive cracking. The licensee for Hatch 2, Georgia Power Company, will replace the affected piping in 1984.

The NRC staff has been reviewing the inspection results of each plant on a case-by-case basis. In general, for the plants where such cracking has been observed, repairs, analysis and/or additional surveillance conditions were required. Where repair was proposed, consideration was given to the strength (relative to ASME Code margin) of the repair, its effect on the piping system, and further inspectability. Where repair was not proposed, consideration was given to uncertainties in the measurements of cracking depth and to projected growth of cracks during subsequent operation. NRC staff evaluation criteria require maintaining the inherent factor of safety prescribed by Section III of the ASME Boiler and Pressure Vessel Code for normal and faulted conditions with consideration of the uncertainties in crack size and growth rate.

Further, the NRC staff accelerated the inspection of five operating BWRs (Browns Ferry Unit 3, Brunswick Unit 2, Dresden Unit 3, Pilgrim Unit 1, and Quad Cities Unit 2) by Orders dated August 26, 1983. As of early July 1983, these plants were scheduled for inspection at various times from August 1983 through Janaury 1984. However, the NRC concluded that these uninspected facilities may have similar IGSCC, which may be unacceptable for continued safe operation without inspections and repair or replacement of the affected pipes and additional surveillance requirements.

On August 4, 1983, EPRI presented to the NRC staff the results of a "round robin" UT program to compare crack depth measurements made by UT versus actual destructive examination. Also, on August 4, 1983, the NRC staff met with representatives from General Electric. On August 8 and 9, 1983, the NRC staff met with licensee representatives from the five BWR

plants yet to be inspected to discuss their responses to the 50.54(f) letters. As a result of the meeting with the five licensees, accelerated schedules for inspections and interim additional compensatory measures (improved leak detection capability, ECCS availability, and operator training) were committed to by the licensee. The staff evaluated the information and commitments received from the licensees. On August 24, 1983, the NRC staff met with the Commission and advised them of its intent to issue Orders for each of the five plants that would confirm these accelerated inspection schedules and impose new interim compensatory measures, or confirm compensatory measures proposed by the licensee. On August 25, 1983, Orders were issued to each of the five plants. Further reports will be made as appropriate. Discussions between the NRC staff and industry are continuing to establish the scope and schedules for future augmented inspections.

2. Improper Control Rod Insertions (Quad Cities Unit 2/Hatch Unit 2)

Both of these events involved procedural errors in withdrawing control rods. These events were primarily reviewed by the regional offices and did not receive a detailed NRR review. We consider these events as isolated incidents and do not have generic implications. Although these procedural errors were undesirable, it is not obvious that they represent a serious deficiency in management of procedural controls in major areas. We will defer our judgment on whether these events represent an abnormal occurrence until we review the proposed abnormal occurrence writeup.

3. Plant Systems Interaction Event (Hatch Unit 2)

The Hatch Unit 2 event of August 25, 1982 included a number of related events which resulted in high temperature reactor coolant being released through the drain system to open areas of the reactor building. Although the event is very interesting from a systems interaction viewpoint, it does not represent a major degradation of safety related equipment. As stated in the AEOD case study, "At no time during the event was there any danger of inadequate core cooling or inadequate core cooling capability...". Therefore, we do not agree that this event meets the abnormal occurrence criteria. This may be an appropriate Appendix C item (Other Events of Interest).

POSSIBLE OTHER EVENTS OF INTEREST (APPENDIX C TO THE AO REPORT)

1. Fuel Failures (Millstone Unit 2)

The Westinghouse fuel failures and the resulting investigation by the licensee represent an extended plant outage. The licensee's investigation is continuing and the results are not scheduled to be submitted

for staff review until mid-November. We agree that this item should be included in Appendix C to the AO report. However, it may be advisable to review the licensee's results and report our findings in the Fourth Quarter CY 1983 Report to Congress.

OTHER INFORMATION REQUESTED FROM NRR

1. License Suspensions

There were no licenses suspended during the third quarter CY 1983.

2. Orders Covering License Modifications for Safety Reasons

On August 26, 1983, Orders were issued to five BWRs to shut down and inspect their primary system piping for Intergranular Stress Corrosion Cracking. These five plants were Quad Cities Unit 2, Dresden Unit 3, Pilgrim Unit 1, Brunswick Unit 2 and Browns Ferry Unit 3.

On October 13, 1983, an Order was issued to Turkey Point Unit 4 to remain in hot shutdown condition rather than going to cold shutdown condition as specified in the plant's Technical Specifications. This action was taken because it was believed that remaining at hot shutdown was more desirable than going to cold shutdown in view of the fact that only one of the two RHR trains was operable.

3. Identification of Those Generic Safety Concerns Approved by the Director, NRR

The following generic issues have been prioritized by the Division of Safety Technology and have completed NRR's peer review process during the third quarter CY 1983. Schedules for their completion will be prepared and resources will be allocated.

Generic Issue No.	Subject	Prioritization
45	Inoperability of Instrumentation Due to Extreme Cold Weather	*
66	Steam Generator Requirements	*
75	Generic Implications of ATWS Events at Salem Nuclear Plant	*
77	Flooding of Safety Equipment Compartments by Backflow Through Floor Drains	High
79	Unanalyzed Reactor Vessel Thermal Stress During Natural Convection Cooling	Medium

*Although these issues recently completed the NRR peer review process, resources have previously been allocated to them.

The following generic issues have been found to be either covered in other issues or determined to have either a low or drop priority. Therefore, resources will not be allocated to the following generic issues:

Generic Issue No.	Subject
31	Natural Circulation Cooldown
41	BWR Scram Discharge Volume Systems

TABLE B-1

Summary of Piping Weld Crack Observations

	12" through 28" Pipe Welds	
	No. Examined	No. Defective
Browns Ferry Unit 1	123	47
Browns Ferry Unit 2	34	2
Brunswick Unit 1	32	3
Cooper	135	22
Dresden Unit 2	51	10
Hatch Unit 1	58	7
Hatch Unit 2	108	39
Monticello	135	6
Peach Bottom Unit 2*	45	16
Peach Bottom Unit 3	111	15
Vermont Yankee	60	34

*Preliminary results - still being evaluated