OCT 1 8 1982

MEMORANDUM FOR: William J. Dircks
Executive Director for Operations

FROM: Victor Stello, Jr., Chairman
Committee to Review Generic Requirements

SUBJECT: MINUTES OF CRGR MEETING NUMBER 22

The Committee to Review Generic Requirements met on Friday, October 8, 1982. A list of attendees is enclosed.

1. Mr. A. Dromerick (IE) presented for Committee review the proposed bulletin, titled "Stress Corrosion Cracking in Thick-Wall Large Diameter Stainless Steel Recirculation System Piping at BWR Plants." The subject of BWR pipe cracking was also discussed at CRGR Meeting #20.

The bulletin is to notify all power reactor licensees and CP holders of a matter, which may have a high degree of safety significance, and to require specific actions. Specifically, this matter involves intergranular stress corrosion cracking in the recirculation system piping in the reactor coolant pressure boundary (RCPB) that was found at the Nine Mile Point, Unit 1, Nuclear Generating Station. This information was also described in considerable detail in IE Information Notice 82-39 dated September 21, 1982. As stated in the bulletin, action (by the nine affected licensees) is required to (1) provide a reasonable level of assurance that inspections which are currently being performed or scheduled are sufficient to detect cracking in BWR thick wall recirculation piping welds and (2) to assist the NRC in determining the generic significance of the piping degradation found at Nine Mile Point. The affected licensees are those owners whose plants are currently in or scheduled to be in a refueling mode or extended outage through January 31, 1983. The bulletin is to be provided to all other licensees and holders of construction permits for information only.

Although complete cost-benefit information was not available, the Committee judged that the bulletin was reasonable and recommends that it be issued. In addition to suggesting changes to the bulletin's organization, the Committee offered the following comments:

(a) The bulletin should indicate how licensees other than the nine identified are to be addressed.

(b) Concerning the demonstration of the effectiveness of the crack detection capability of the ultrasonic testing (UT) methdology used or planned to be used to examine recirculation system piping welds, the demonstration should employ the same procedures

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and standards, the same type of equipment and representative UT personnel from the inservice inspection (ISI) organization utilized or to be utilized in the examination at the plant site. In this regard, the bulletin should note that arrangements have been made to have samples from the Nine Point plant available for industry demonstrations of UT methodology. The samples have been taken to Battelle Memorial Institute in Columbus, Ohio for characterization and subsequent use.

(c) The bulletin should indicate what actions are to be taken if cracks are detected.

(d) The bulletin should indicate that the information requested in the bulletin is needed to assist NRC's further evaluation of the issue.

2. The CRGR continued its review of the issue of reactor coolant pump trip criteria for PWRs from meeting No. 20. It was noted that there has been a large number of analyses performed over the past 3 years to study this question, by licensees, vendors and NRC contractors. The Committee did not attempt to review these analyses in detail, and in fact most of the licensee and vendor studies are not available at this time. In addition to these analyses, there have been experiments in Semiscale and LOFT, as well as operating events where the pumps have been tripped according to the guidelines in Bulletins 79-05C and 79-06C.

The CRGR agreed that there is a wide range of transients and LOCAs (such as steam generator tube ruptures and pipe breaks 2" dia.) where it is beneficial for the operators to maintain forced circulation cooling and mixing through operation of the RCPs. In addition, continued operation of the pumps enables the operators to have better control of system pressure and thereby enhances recovery from these events and reduces the potential for additional complications resulting from operator errors.

On the other hand, some of the available calculations show that there can be accidents (small LOCAs with less than full capacity high pressure safety injection) where continued operation of the pumps or continued operation followed by delayed pump trip could lead to core damage. The available information on this point is contradictory; some calculations (believed to be conservative) show core damage for delayed trip, while others show little difference between early trip and delayed trip. The results appear to depend strongly on the physical models embedded in the computer calculations, as well as the physical characteristics of the specific plant design. It was not possible to resolve these calculational differences at this time -perhaps this should be a longer-range goal for the research program.

In a case like this, where the analyses are contradictory, one should place substantial weight on the views of the reactor designers. These views were uniform in asserting that some small LOCAs with less than full capacity HPSI and delayed RCP trip could lead to core damage.

While acknowledging the potential for core damage for this accident sequence, the CRGR noted that its estimated frequency of occurrence is relatively low, perhaps on the order of 10^{-5}/reactor year. On the other hand, there are other accident sequences of much higher frequency where the absence of forced circulation makes the operator's job more difficult and can increase the likelihood of operator errors. Examples of these accidents are steam generator tube ruptures and small LOCAs (less than 2-inch diameter breaks). The higher frequency of these accidents, when combined with an increased operator error probability, can lead to a frequency of core damage comparable to that described above. Therefore, a balance should be struck between the competing risks associated with tripping the pumps early and leaving them running until appropriate pump trip criteria are reached.

In an attempt to assess the merits of continued pump operation versus tripping the pumps early, the Committee reviewed a number of safety considerations, shown in Table I. Although one cannot draw quantitative conclusions from Table I, it is clear that there are important advantages to have the pumps running for many transients and accidents.

Based on extensive discussions with licensees and reactor manufacturers, the NRR staff has proposed that licensees develop pump trip set points designed to keep the pumps running for those accidents where forced circulation and system pressure control is a major aid to the operators. The set points would be designed to trip the pumps in those LOCAs where continued operation or delayed trip and less than full capacity HPI delivery could lead to core damage. This approach is clearly a reasonable one, and the proposed approach for setting pump trip set points appears plausible to accomplish the objectives above.

However, there is not sufficient information available to show conclusively that the pump trip set points suggested by the reactor designers will strike the right balance between early pump trip and continued pump operation. NRC contractor analyses show that the plant-specific modeling of loop seals, ECC injection location and upper plenum to downcomer bypass flow is very important in calculating the degree of core uncovery in small LOCAs.

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B Based on this review of the information available, the Committee makes the following recommendations:

(a) NRC should not require licensees to install automatic pump trip systems. However, licensees may wish to propose this option if they feel it is needed to meet operator action time criteria.

(b) NRC should send a letter to PWR licensees informing the licensees that we intend to rescind the pump trip instructions in Bulletins 79-05C and 79-06C. The letter should outline the concerns associated with delayed pump trip during small LOCAs with less than full capacity HPSI, and the opposite concerns associated with having no forced circulation during steam generator tube breaks, small LOCAs (less than 2" dia.) and other similar accidents. Each licensee should be requested to send NRC a schedule for implementing revised pump trip set points, if any revisions are proposed, in their operating procedures. Licensees should be advised that NRC will audit their revised operating procedures and supporting analyses. The licensees should be informed that separate letters are being sent to the vendors.

(c) NRC should send a letter to PWR vendors and Yankee Atomic requesting information on how their evaluation models meet the 10 CFR 50 Appendix K criteria for proposed pump trip set points.

(d) The staff should make it clear to licensees that their proposals for pump trip criteria can take advantage of all available instrumentation and equipment in which the operators have confidence, regardless of its safety grade classification.

Original Signed by
V. Stello

Victor Stello, Jr. Chairman
Committee to Review Generic Requirements

Enclosure: List of Attendees

cc: Commission (5)
Office Directors
Regional Administrators
CRGR Members
G. Cunningham

Table fixed / meeting
W.S.

Distribution:
VStello
TEMurley
DEDROGR cf
DEDROGR staff
Central File
PDR (NRG/CRGR)
SStern
FCameron
BBrach
RErickson
FHebdon
WLIttle (R-III)
JGagliardo (R-IV)
JZwetzig (R-V)

OFFICE	DEDROGR/D	DEDROGR/S	DEDROGR				
SURNAME	TEMurley:bg	WSchwink	VStello				
DATE	10/15/82	10/15/82	10/ /82				

Table I

Safety Considerations for Evaluating Pump Operation

vs. Early Pump Trip

1. Augmented Control of RCS Pressure
2. Augmented Control Over Subcooling
3. Forced Circulation Cooling of Core
4. RCS Inventory Control
5. Thermal Mixing of Cold HPI Coolant
6. Augmented Core Heat Removal
7. Augmented Heat Removal by SGs
8. Improved Rate of Coolant Delivery to Core
9. Improved Diversion of ECCS
10. Improved Access to Low Pressure ECCS
11. Improved Rate of Boration to Core
12. Improved Recovery Time when HPI Degraded
13. Improved Cooldown Control to SGs
14. Extra Pump Heat to RCS
15. Net Inventory Loss of RCP Operation
16. Enhances Readout of ICC Instrumentation
17. Improved Control and Dissolution of RCS Voids
18. Enhanced Cooling of Degraded Core in Reflood
19. Degraded Detection of Reflood by ICC Instruments
20. Degraded Restart of RCPs Once Tripped
21. Enhanced Readout Accuracy of PZR Level Indications

CRGR Meeting #22

List of Attendees

October 6, 1982

CRGR MEMBERS

Vic Stello
Bob Bernero
Darrell Eisenhut
Jack Heltemes
Joe Scinto
Ed Jordan

OTHERS

Tom Murley
Walt Schwink
Bob Erickson
Ed Abbott
Tom Cox
Mat Taylor
C. Y. Cheng
J. J. Blake
Wayne Lanning
Gary Zech
Steve Stern
Bob Baer
Jim Lieberman
Jim Sniezek
Frank Miraglia
Bill Johnston
Wayne Houston