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Files (Docket No. 50-155)
THRU: D. L. Ziemann, Chief, ORB #2, DRL

BIG ROCK POINT - LEAK DETECTION - INSERVICE INSPECTION OF NUCLEAR REACTOR COOLANT SYSTEMS - ECCS INTERIM CRITERIA (CONSUMERS POWER COMPANY)

Consumers Power Company (CPCo) has reported the following information(1 and 8 App 2) with respect to leak detection capabilities at Big Rock Point:

1. A dew cell is installed in an exhaust duct from the steam drum cavity. A moderate valve packing leak will raise the dew point temperature noticeably.

2. A dirty waste collection system typically collects 15 gallons of radioactive wastes per hour. Doubling of this rate for unknown reasons will be reported by the operator. A grab sample for air particulate activity will be taken to confirm or deny the presence of a leak.

3. Very small leaks in the control rod drive room can be heard on inspection rounds because the background noise level is very low.

4. Air particulate samples are routinely taken on a weekly basis from the steam drum enclosure exhaust line. The sensitivity of this leak detection method is 5.2×10^{-4} gpm (1.97 cc/min). This method allows detection of very small valve packing leaks.

CPCo later reported(2) that an additional leak detection system had been installed recently and another had been modified in an effort to provide a more quantitative measure of leakage from the primary coolant pressure boundary. Running time meters have been installed on the containment dirty and clean sump pumps. The dew cell in the ventilation recirculation duct from the steam drum cavity has been relocated to the ventilation exhaust duct from the recirculating pump room. CPCo also has ordered a continuous air radiation monitor to be installed in 1972 to sample the air discharged from the containment.

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The existing Technical Specifications have no restrictions regarding primary coolant leakage rates and CPCo does not propose such specifications because of insufficient experience with leak detectors that have been installed and the large free volume of the Big Rock Point containment. However, CPCo has incorporated requirements into the Big Rock Point operating procedures that suspected primary system pressure boundary leaks be investigated and the plant be shut down immediately if the leak is due to cracking of the primary coolant pressure boundary.

The containment free volume of 940,000 cubic feet will cause appreciable dilution of radioactivity that may be released from a primary coolant system leak, thus affecting the sensitivity of air radiation monitor measurements located in the air discharge from the containment. The time required to detect a sudden primary coolant leak is dependent on the air mixing within containment, the fresh air intake and exhaust rate, and the location of the leak with respect to the monitoring instrumentation. The radioactivity in the liquid and steam that leaks into the containment also affects the sensitivity of the continuous air radiation monitor measurements of air discharges from the containment vessel. Steam leaks have distinctive characteristics, in contrast to water leaks, that can significantly influence leak detection sensitivity and therefore the ability to detect primary coolant system boundary cracks. It may be possible to obtain useful analytical information by intentionally releasing small amounts of primary coolant and steam to the containment and observing the behavior of the radiation and other leak detection monitors. From the brief description of the leak detection sensors provided by CPCo, it is not evident that leak detection has been optimized to detect cracks in the reactor vessel, the largest pipes, two 20-inch coolant water pipes at the bottom of the reactor vessel that allow coolant recirculation to enter the reactor vessel, or the circulation pump suction headers which are 24 inches in diameter. A request for additional information to complete our evaluation of the Big Rock Point leak detection capability and limits to be specified in the Technical Specifications has been included in a DRL letter to CPCo (dated March 28, 1972).

CPCo reported(2) in response to a DRL request(3) that plans have been made to reduce the inspection interval defined by paragraph IS-241 of Section XI of the ASME Boiler and Pressure Vessel Code from ten years to three years for items 1.1 through 4.6 of Table IS-261 of Section XI covered in the Big Rock Point program. A "copy" of the Big Rock

Point Inservice Program was included for DRL information. We note that 60% of the items listed in Table IS-261 of Section XI of ASME Boiler and Pressure Code dated January 1, 1970, have been excluded because of inaccessibility or high radiation levels. For similar reasons, there are exceptions to 60% of the Heat Exchanger and Steam Generator items and 75% of Piping Pressure Boundary items. There are no exceptions listed for Pump & Valve Pressure Boundary items. Design and construction of the plant was completed before the Code was adopted and accounts for the large number of exceptions to the Inservice Inspection of Nuclear Reactor Coolant System items that are specified in Section XI of the ASME Boiler and Pressure Code, i.e., the plant was not designed to meet the Code requirements. Because of the large number of exceptions, intensive efforts must be expended to detect primary system leaks due to cracks in the primary system pressure boundary in time to prevent catastrophic failure. Consistent with the requirements for other licensed nuclear power facilities, CPCo has been requested by our letter of March 28, 1972, to identify for inclusion in the Big Rock Point Technical Specifications, the list of items and the frequency of Inservice Inspections of the Primary Coolant System.

We authorized plant modifications and technical specification changes(4) to increase the reliability of emergency core cooling. Our evaluation(5) of the new backup core spray system recognized the improved reliability of emergency core spray cooling, but dependence on offsite power for continued feedwater pump operation and the calculated peak clad fuel temperatures (above 2300°F) were identified as subjects requiring further evaluation. CPCo was advised by GE(5) that if the peak clad temperature calculations were repeated using FLECHT data and channel wetting effects, that the calculated peak temperature would be 200 - 300°F lower than reported. The most recent calculations(6), using calculational methods as specified by the AEC Interim Acceptance Criteria for the largest pipe breaks, show that the calculated peak clad temperatures actually increase from the 2800°F temperature originally reported to 3000°F instead of decreasing as expected. CPCo submitted results(6) from a reanalysis of peak clad temperatures that was less conservative than specified by DRL(3) showing peak clad temperatures below 2300°F for the entire spectrum of breaks except for the 0.03 to 0.2 ft^2 breaks where clad melting will occur (3350°F). Additional protection will be provided according to CPCo(6) to prevent clad melting for breaks between 0.03 to 0.2 ft^2. In our last evaluation of the Big Rock Point ECCS(5), we concluded that the core could be depressurized in time to prevent uncovering the core without the continuous injection of feedwater for breaks smaller than 0.0027 ft^2 (a single end 3/4-inch pipe break). According to GE calculations at that time, continuous injection of primary coolant(4) through the high pressure feedwater system (Figure 5, ref. 8) was necessary to prevent clad melting for break sizes between 0.0027 and 0.05 ft^2. These calculated values differ noticeably from the most recent calculations(6). For example, the older calculations(8) showed that clad melting would not occur for coolant system breaks greater than 0.026 ft^2 assuming no feedwater injection during the post-accident period. The recent calculational results(6) show clad melting for the range of breaks between 0.03 ft^2 and 0.2 ft^2 assuming continuous feedwater injection at 965 gpm initiated 60 seconds after the accident. Conservative assumptions, such as a reduced primary system water inventory, account for the calculated severity of the accident in this range of break sizes. Table 1 presents the results to illustrate calculational changes that have occurred over the last two years. It can be expected that revised calculational methods in accordance with a DRL directive(9) will cause the results to be even more unfavorable. CPCo has stated(6) their intention to provide either a high pressure core spray system or an automatic depressurization system if analyses in progress confirm the necessity of such modifications to prevent excessive temperatures following coolant system breaks in the intermediate range of about 0.03 to 0.20 ft^2. No further action on the ECCS system will be taken by DRL until the results from a reanalysis of the ECCS are presented by CPCo in accordance with DRL's directive(9) and CPCo commitments for further analysis(6).

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Enclosures:
1. Table 1
2. References

cc w/enclosures:
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TABLE 1

CHANGES IN CALCULATED PRIMARY SYSTEM BREAK SIZES VS PEAK CLAD TEMPERATURES THAT HAVE OCCURRED IN TWO YEARS WITH CONTINUOUS FEEDWATER INJECTION

Break Size - ft^2	
Up to .004 or single-ended break of one inch pipe (Ref. 8 App. 2 - 2/2/71)	The plant can be cooled down at the normal rate (100°/min) and core spray initiated about 3 hours later when pressure has decayed to the spray injection pressure and water level has fallen to the mid-core plane as long as feedwater injection continues at the normal rate of 2000 gpm.
Up to .011 or a double-ended break of a one-inch pipe (Ref. 8 App. 2 - 2/2/71)	The plant must be cooled down at 300°F/hr.
Up to .05 (7 in^2) or a double-ended break of a two-inch pipe (Ref. 10 p. 2 and Fig. V - 2/9/70)	According to Figure V (Ref. 10) continuous feedwater injection will depressurize the primary system to allow emergency core spray before clad melting occurs for breaks up to .05 ft^2. For all breaks greater than .05 ft^2 but less than the largest double-ended break of the 20" recirculation line, low pressure emergency core spray is sufficient to prevent clad melting.

WITHOUT CONTINUOUS FEEDWATER INJECTION

	No offsite power available
Up to .0027 or single-ended break of 3/4 inch pipe (Ref. 8 App. 2 - 2/2/71)	Using control rod drive cooling water and the emergency condenser, the plant can be depressurized in about 1-1/2 hours to permit low pressure core spray before water level falls below core mid plane.

TABLE 1 (Cont'd)

*Above .0027 but less than 0.026 (Ref. 10, Fig. VI - 2/9/70)	Peak clad temperature reach melting, 3350°F, and for breaks up to 0.032 ft^2 all rods perforate.
*Above 0.026 (Ref. 10, Fig. VI - 2/9/70)	Peak clad temperatures decrease from 3350°F as break increases towards 0.2 ft^2 when peak temperatures are 2000°F and increase again to 2800°F for 3.6 ft^2 breaks.

* According to Fig. V of Ref. 10 (2/9/70) Clad melting occurs, in contradiction to Figure VI, for all breaks up to 0.05 ft^2

RESPONSE TO ECCS INTERIM DESIGN CRITERIA

Up to 0.028 (Ref. 6, Fig. 2 - 12/30/71)	Assuming feedwater is provided at 965 gpm 60 seconds after the break, peak clad temperatures remain below 600°F.
Between 0.029 and 0.2 (Ref. 6, Fig. 2 - 12/30/71)	With same assumptions peak clad temperature reaches melting at 3350°F. Break area limits for clad melt will change slightly if feed flow is not restarted.
Greater than 0.2 (Ref. 6, Fig. 2 - 12/30/71)	Peak clad temperature decreases reaching a minimum of 2150°F when break size increases to 0.25. As break size increases above 0.25, peak clad temperature increases to 2300° for the maximum break of 4.3 ft^2.

REFERENCES

(1) CPCo letter dated September 11, 1970, answer to question 7 in DRL letter dated August 6, 1970.

(2) CPCo letter dated September 29, 1971, responsive to DRL letter of July 20, 1971, requesting interim improvements related to emergency core cooling requirements.

(3) DRL letter dated July 20, 1971 - Interim acceptance criteria for the performance of emergency core cooling systems.

(4) Change No. 26 dated July 27. 1971, concluded that a new backup core spray and associated modifications should be added to the emergency core cooling system as soon as possible to increase reliability, and authorized Technical Specification changes related to these ECCS modifications. CPCo was directed to continue ECCS evaluations to improve core cooling reliability, especially in the range of small breaks, but in accordance with the interim criteria described in Ref. 3 above.

(5) Memo to Files dated July 27, 1971 - Evaluation of Big Rock Point Emergency Core Cooling System.

(6) CPCo letter dated December 30, 1971 - Preliminary results of reanalysis of emergency core cooling system performance in response to Ref. 3 above.

(7) Memo to Files dated December 9, 1971 - Calculations in accordance with Ref. 3 above cause peak clad temperatures to increase above 2800°F rather than decrease as expected and discussed in Ref. 5 above.

(8) CPCo letter dated February 2, 1971 - Proposed Change 27 - Redundant Core Spray System.

(9) DRL letter dated January 17, 1972, directs CPCo to revise the calculated results of Ref. 6 above to reflect the DRL criteria and design methods of Ref. 3 above.

(10) CPCo letter dated February 9, 1970, in response to DRL letter dated December 30, 1966, requesting review of the Big Rock Point emergency core cooling provisions.