APR 08 1976

Docket No.: 50-155

LICENSEE: Consumers Power Company

FACILITY: Big Rock Point

SUMMARY OF MEETING HELD ON MARCH 18-19, 1976, REGARDING EMERGENCY CORE COOLING SYSTEM

On March 18-19, 1976, representatives of Consumers Power Company (CPCo) and their consultant, NUS Corporation, met with a task force of the Nuclear Regulatory Commission (NRC) staff in Charlevoix, Michigan. A list of attendees is attached. NRC requested the meeting to review the Emergency Core Cooling System (ECCS). Information required to complete our ECCS review in the electrical, instrumentation and control areas are shown in Enclosure 1. Other concerns relating to the existing Big Rock Point ECCS are discussed below.

Following is a status of the discussion held and the CPCo responses (note the first 13 responses are numbered to coincide with the questions in Enclosure 1):

1. Paragraph 3.2 of the February 27, 1976 CPCo report lists ten short-term single failure recommendations. Items 5, 6, 9 and 10 fail to meet the criteria of IEEE Std 279-1971. CPCo agreed to respond but stated the requirement to meet the standard appears unnecessary at Big Rock Point.

2. Item 7, paragraph 3.2 of the February 27, 1976 CPCo report discusses elimination of the emergencyddiesel generator's dependency on the station battery. CPCo will bring this into compliance with IEEE Std 279-1971.

3. LOCA environment, seismic information and potential high energy lines breaks were discussed. Although these items were previously addressed by CPCo they agreed to provide additional information on these subjects.

4. Submerged LOCA equipment was previously reviewed and documented in a May 15, 1975 CPCo report. Other submerged equipment which might affect ECCS power busses will be reviewed by CPCo.

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5. Previously identified valves MO-7069 (spray crossconnect) and MO-7072
& (long-term cooling heat exchanger bypass) do not conform to Branch
6. Technical Position EICSB 18. CPCo agreed to provide an evaluation of this subject. Mechanical blocks for each valve will be considered.

The station transformer fire system deluge valve failure probability is considered extremely low by CPCo. The NRC staff task force advised that an ECCS actuation override should be considered. The task force made an onsite inspection of the entire plant fire protection system.

7. Previously identified need for an electrical interlock for enclosure spray valves MO-7064 and MO-7068 was discussed. The NRC staff noted that corrective action would be relatively easy and should be considered. CPCo agreed to consider this matter and advise NRC of its conclusions.

8. During a tour of the facility it was observed that the physical separation criteria may not be met for some cable and wiring. CPCo noted that the separation criteria for ECCS is as was stated in their 1971 submittal for the redundant core spray system. All wiring for the backup core spray system is routed in metal conduit, physically and electrically separate from the ring core spray system. CPCo agreed to review this matter and provide additional information.

9. CPCo advised that these questions were answered in the matrix of
& Table 4.1 and Table 5.1 respectively of Attachment 2 to the February 27,
10. 1976 CPCo report. NRC agreed.

11. This question was modified by NRC to include not only the diesel generator but also the diesel fire pump. CPCo agreed to respond but stated they considered noncompliance with NRC Branch Position EICSB 17 to be one of the current NRC guidelines not applicable to Big Rock Point.

12. It was mutually agreed that Big Rock Point does not comply with NRC Branch Position EICSB 27. NRC explained currently acceptable alternatives for the five valves in question. CPCo agreed to evaluate and provide a response to this concern.

13. CPCo's risk analysis, Attachment 4 to the February 27, 1976 CPCo report was discussed in considerable detail. CPCo was advised that overall reliability of the system must include considerations of the test and maintenance unavailabilities, common mode failure considerations and human error considerations. Secondly, the defense in depth concept including possible use of the feedwater system for emergency core cooling should be addressed. CPCo agreed to update their previous submittal by letter.

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14. Recent General Electric information relati g to effectiveness of the nozzle core spray was discussed. CPCo was advised of the current status and requested to study the effect of this information on their nozzle core spray effectiveness CPCo agreed to evaluate this subject and provide the results of their review to the NRC.

15. An inconsistency was noted by the NRC task force between the current Big Rock Point technical specification relating to the LOCA analysis pressure of 1350 psig and the specification which would allow operation to 1485 psig. CPCo agreed to evaluate this inconsistency and advise us of their conclusions.

16. CPCo's November 26, 1975 submittal relating to the core spray coefficients correlation between the 7 x 7 and 11 x 11 assemblies was discussed. CPCo advised that the information was included in an EXXON report submitted to NRC. CPCo agreed to advise us of the date of the report.

17. NRC requested information relating to the fire protection system. The above ground and below ground piping system may be subject to passive failures. CPCo agreed to provide piping specifications, pipe schedules and test requirements.

18. NRC requested additional information regarding the ring core spray system which was installed in 1961. Specifically, the pipe schedules and the detailed valve specification or design class for MO-7051 and the adjacent check valve was requested. CPCo agreed to provide the information.

During telephone conversations following the meeting, NRC asked and CPCo agreed to respond to the following:

1. Since the deluge valve is near the station transformer, would a break in this valve cause a loss of all offsite power?

2. Clarify that the break sizes analyzed included the correct calculated break size for the 24 inch reactor recirculation pump suction line.

3. Consider diesel generator single failure in relation to ECCS acceptability. Although this matter has been addressed by NRC previously, NRC considers it appropriate to reconsider the matter at this time.

CPCo agreed to respond to the outstanding NRC concerns described above by March 26, 1976, if at all possible.

Original signed by
Edward A. Reeves

Edward A. Reeves, Project Manager
Operating Reactors Branch #2
Division of Operating Reactors

Enclosures:
1. List of Attendees
2. Generic Infromation Request for Review of ECCS in the Electrical, Instrumentation and Control Areas

cc: R. B. Sewell
Nuclear Licensing Administrator
Consumprs Power Company
212 West Michigan Avenue
Jackson, Michigan 49201

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DATE	4/8/76	4/8/76				

LIST OF ATTENDEES

CONSUMERS POWER COMPANY

MEETING OF MARCH 18-19, 1976

CPCo

R. Sewell
D. De Moor

NRC Staff

E. Reeves
N. Anderson
F. Rosa
P. Shemanski

NUS

G. Rudy

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GENERIC INFORMATION REQUEST FOR REVIEW OF ECCS IN THE ELECTRICAL, INSTRUMENTATION AND CONTROLS AREAS

The Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors, 10 CFR Part 50.46, requires that an analysis of possible failure modes of ECCS equipment and of their effects on ECCS performance be performed. This analysis should demonstrate that your ECCS and supporting subsystems meet the single failure criterion. We require that documentation of this analysis be provided in sufficient detail to enable the staff to (1) verify that the analysis demonstrates that the ECCS and supporting subsystems meet the single failure criterion as defined in IEEE Std 279-1971, and (2) determine the acceptability and verify the implementation of any proposed design modification required as a result of your analysis. Therefore, we require that the following information be submitted to support the single failure analysis of the ECCS and supporting subsystems:

1. Identify any nonconformance of the design of the ECCS actuation system with the single failure requirements of IEEE Std 279-1971. Describe any changes proposed for meeting these requirements.
2. Identify any nonconformance of the design of the onsite emergency power system, a-c and d-c, with the single failure requirements of IEEE Std 279-1971. Describe any changes proposed for meeting these requirements.
3. Identify all the electrical equipment required for the ECCS and supporting subsystems to enable performance of the ECCS safety function.

Define the qualification status (ability to withstand the design basis seismic and environmental conditions) of this equipment, and the basis for such qualification, to provide reasonable assurance that the equipment will be capable of performing its safety function. Describe any proposed design modifications, analyses, or test programs for meeting the environmental and seismic qualification requirements.

4. Identify all electrical equipment, both safety and non-safety, that may become submerged as a result of a LOCA. For all such equipment that is not qualified for service in such an environment, provide an analysis to determine the following: (1) the safety significance of the failure of the equipment (e.g., spurious operation, loss of function, loss of accident/post-accident monitoring, etc.) as a result of flooding, (2) the effects of Class IE electrical power sources serving this equipment as a result of such failures, and (3) the proposed design changes resulting from your analysis. Your response to item (2) should specifically address breaker and fuse coordination and the isolation capabilities of this aspect of your design.

5. Identify any single electrically operated fluid system component, including manually-controlled electrically-operated valves, whose failure could result in loss of capability of the ECCS to perform its safety function. Failure in both the "fail to function" sense and in the "undesirable function" sense should be considered, and this should

apply even though the component may not be required to function in a given safety operational sequence.

6. With regard to the equipment identified in item (5), provide a detailed description of any proposed design changes deemed necessary by your analysis for meeting the single failure criterion. Your response should specifically address but should not be limited to changes made to meet the single failure criterion by conformance to Branch Technical Position EICSB 18, "Application of the Single Failure Criterion to Manually-Controlled Electrically-Operated Valves", of Appendix 7A of the Regulatory Standard Review Plan. This position establishes the acceptability of disconnecting power to the electrical components of a fluid system as one means of meeting the single failure criterion.

7. Identify any electrical interlocks between redundant portions of the ECCS and supporting subsystems. Define the consequence of failure of any interlock on the capability of the ECCS to perform its safety function. Describe any proposed design modifications resulting from this review.

8. Provide the electrical and physical separation criteria for your design of redundant safety equipment and functions. Include the features in your design that minimize the vulnerability of the ECCS and supporting subsystems to common failure modes.

9. In reference to the NUS Corporation report (5069-P-01, Rev. C) entitled "Big Rock Point Nuclear Power Plant ECCS and Associated Systems Evaluation" dated February 23, 1976, the Failure Mode and Effects Analysis (FMEA) identified 37 electrical single failures which result in unacceptable ECCS or associated systems operating conditions. The impact of these failures on the ECCS and its associated systems focused only on 10 explicit system features which require design modification due to unacceptable single failures. Three of these 10 NUS design modification recommendations are being implemented during the present plant shutdown while the remaining 7 recommended modifications may or may not be implemented depending on Plant Life Exemption Action. In summary, 8 of 37 identified electrical single failures which result in unacceptable ECCS or associated systems operating conditions are being implemented. Provide justification why the remaining 29 electrical single failures identified in the FMEA should not be corrected and the resulting impact on the risk analysis as a result of not implementing corrective action.

10. As shown in Figure 5 (Attachment 9), of your February 27, 1976 Report on Evaluation of Adequacy of ECCS, the station battery system loads include, but are not limited to, the plant annunciators, ECCS valves MO-7051/MO-7061, ESS valve MO-7064, and the emergency diesel generator

and its associated breaker (1A-2B and 2A-2B) control circuits. Identify any nonconformance of the station battery system design and any changes proposed for meeting the requirements of GDC 17 and IEEE Std 308-1971 for the ECCS and its associated systems with regard to sufficient capacity, capability and reliability to supply the required distribution system loads in the event of a LOCA. Discuss whether the station battery system has been sized to accommodate the added non-safety-related loads during emergency conditions and if not, whether automatic disconnection of those non-safety-related loads upon detection of the emergency condition is provided.

11. Provide a description of the diesel generator protection system and any nonconformance of this design with regard to providing protection against spurious trips as the position described in Appendix 7A of the Regulatory Standard Review Plan (EICSB 17).

12. Provide a description of your design criteria for thermal overload protection for motors of motor operated valves in the ECCS and compare this to the Position (EICSB 27) found in Appendix 7A of the Regulatory Standard Review Plan.

13. The risk analysis presented in Attachment 4 of the February 27, 1976 ECCS report failed to consider common mode failures. When considering

the risk of failure to achieve ECCS performance within FAC limits due to single failures, common mode and dependency considerations must be considered. Modify your risk analysis accordingly to include the effects of common mode failures.

MEETING SUMMARY DISTRIBUTION:

Docket
NRC PDR
Local PDR
ORB #2 Reading
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