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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

Docket Nos: 50-373
and 50-374

JUL 23 1979

Mr. Byron Lee, Jr.
Vice President
Commonwealth Edison Company
P. O. Box 767
Chicago, Illinois 60690

Dear Mr. Lee:

SUBJECT: ADDITIONAL INFORMATION REQUIRED FOR NRC STAFF GENERIC REPORT ON BOILING WATER REACTORS

On June 28, 1979, the NRC staff met with representatives from each of the licensees of boiling water reactors (BWR's) as well as the applicants for near-term operating licenses for BWR's. At that meeting we discussed our short-term program for reviewing the implications of the Three Mile Island Unit 2 accident on operating BWR's and near-term Operating License applications for BWR's. At the meeting we held a discussion of our general information needs and noted that our review will concentrate on two basic areas, i.e., systems and analysis. We stated that we would provide you with our formal requests for information at a later date.

Enclosure 1, which consists of three attachments, contains our requests for additional information in the systems area. Enclosure 2 contains our requests for additional information in the analysis area. In order for us to maintain our schedule we request that you provide clear and complete responses to the enclosed requests by August 27, 1979. If you cannot meet this schedule or if you require any clarification of these matters please contact William F. Kane, (301) 492-7745 immediately.

Sincerely,

Olan D. Parr

Olan D. Parr, Chief
Light Water Reactors Branch No. 3
Division of Project Management

Enclosures:
1. Request for Additional Information
(Systems Area)
2. Request for Additional Information
(Analysis Area)

cc w/enclosures:
See next page

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7908160309

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ENCLOSURE 1

REQUESTS FOR ADDITIONAL INFORMATION

BULLETINS & ORDERS SYSTEMS GROUP

Attachment 1

Information on Systems Capable of Providing Post-Accident and Transient Core Cooling

Instructions

Table I is intended to be an all inclusive list of the systems that are capable of providing post-accident and transient core cooling for all types of BWRs. However, if your plant has additional or alternate systems that provide core cooling, that have not been specifically identified, they should be included in your submittal.

Table II contains a list of information that should be provided as applicable, for the systems identified in Table I. The information that only requires a yes/no answer has been identified. As noted on the table some of the information may be provided by utilizing drawings, however, the drawings must be large enough to be clearly legible, the systems and components marked (particularly if plant P&ID drawings are used), and drawing legends provided where needed.

If questions arise pertaining to the interpretation of the type of information requested contact Byron Siegel (301-492-7341) or Wayne Hodges (301-492-7588).

NOTE: We are aware that much of the information we are requesting may have already been submitted on your docket. However, in order to expedite our review, we are requesting that you compile and resubmit the information in this attachment.

Table I

Systems for which information is requested

1. Reactor Core Isolation Cooling System (RCIC)
2. Isolation Condenser
3. High Pressure Core Spray System (HPCS)
4. High Pressure Coolant Injection System (HPCI)
5. Low Pressure Core Spray System (LPCS)
6. Low Pressure Coolant Injection System (LPCI)
7. Automatic Depressurization System (ADS)
8. Safety Relief Valves
9. Residual Heat Removal System (RHR) including Shutdown Cooling, Steam Condensing, Suppression Pool Cooling and Containment Spray Modes
10. Standby Coolant Supply System
11. Reactor Closed Cooling Water System
12. Control Rod Drive System
13. Condensate Storage Tank
14. Main Feedwater System
15. Recirculation Pump/Motor Cooling Systems

Table II

Information on Systems Capable of Providing Post-Accident and Transient Core Cooling

General System Design Information

- Safety Classification & Seismic Category
- Plant Steam By-Pass Capacity
- Potential of Systems & Component Flooding (i.e., injection of water from CST in excess of Technical Specification min.) and Separation of Trains
- Normal Position of Valves, Indication Location Direct or Indirect Indication[1]
- Failed State of Each Valve[1]
- Normal Power Sources for System Operation[1]
- Normal Power Sources for Support System Operation[1], e.g., lube oil, lube oil cooling, ventilation
- Systems and Components Shared Between Units
- Air Sources for Pneumatic Valves, Cycling Capacity & Alternate Sources
- Number of Safety & Relief Valves & Relieving Capacity
- Relief & Safety Valve Setpoints
- System Trips
- Methods of Cooling System Components (i.e., pumps, valves)

System Activation

- Automatic Startup Logic (initiation signals) & Power Source
- Automatic Sequencing Back onto Diesel Following Reset (Yes/No)
- Auto Initiation Overriding Capability
- Auto Initiation Built in Time Delay
- Manual Initiation Capability, Procedure, Time Req'd, Locations, Manpower Req'd
- Potential Commonalities with Control Systems
- System Interlocks & Diversion
- Operator Actions Required for System Operation & Control

Water Sources

- Safety Classification & Seismic Classification
- Primary Water Source, Total & Dedicated Capacity, Time Available
- Secondary and Backup Water Sources, Automatic/Manual, Procedure, Time, Req'd
- Strainers in System and Location

Power Source

- Number of Trains
- Pumps Connected to Diesel Generators
- AC & DC Bus Arrangement for Trains
- Loss of Offsite Power - System Response, Operator Action, Time Req'd
- Loss of On-site AC Power - System Response Operator Action, Time Req'd
- Loss of All AC Power - System Response, Operator Action, Time Req'd

Instrumentation & Control

- Safety Classification & Seismic Category
- Automatic and Manual Control from Control Room (Yes/No)
- Alarms Located in Control Room
- System Indications Located in Control Room (pump, valves, level etc.)
- Remote Control Panels
- Methods of Detecting Leaking Safety/Relief Valves (i.e., leaking bellows, unseated valve)

Testing/Technical Specifications

- Limiting Conditions for Operation
- Frequency of System & Component Tests
- System Testing Lineups[1]
- System Bypass and/or Test Loops[1]
- Method of Verification of Correct Test Lineup and Restoration to Normal Condition

- Allowable System Outage Times
- System & Componentional Testing Following Maintenance
- Components Not Periodically Tested
- Auto Override During Tests
- Other Components or System Affected by Tests

1/ May be provided by a drawing

Attachment 2

Information Needed for Containment Isolation System

I. For each fluid line and fluid instrument lines penetrating the containment, provide a table of design information regarding the containment isolation provisions which include the following information:

a. Containment Penetration number;

b. System name;

c. Fluid contained;

d. Engineered safety feature system (yes or no);

e. Figure showing arrangement of containment isolation barriers;

f. Isolation valve number;

g. Location of valve (inside or outside containment);

h. Valve type and operation;

i. Primary mode of valve actuation;

j. Secondary mode of valve actuation;

k. Normal valve position;

l. Shutdown valve position;

m. Postaccident valve position;

n. Power failure valve position;

o. Containment isolation signals, including parameters sensed and their set point;

p. Valve closure time;

q. Power source;

r. Valve position indication (direct or indirect)

II. Discuss the design requirements for the containment isolation barriers regarding:

a. The extent to which the quality standards and seismic design classification of the containment isolation provisions follow the recommendations of Regulatory Guides 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Water-Containing Components of Nuclear Power Plants," and 1.29, "Seismic Design Classification";

b. Assurance of the operability of valves and valve operators in the containment atmosphere under normal plant operating conditions and postulated accident conditions.

c. Qualification of closed systems inside and outside the containment as isolation barriers;

d. Qualification of a valve as an isolation barrier;

e. Required isolation valve closure times;

f. Mechanical and electrical redundancy to preclude common mode failures;

g. Primary and secondary modes of valve actuation

III. Discuss the provisions for detecting leakage from a remote manually controlled system (such as an engineered safety feature system or essential line) for the purpose of determining wher to isolate the affected system or system train. Specify the parameters sensed, their set point, and procedure for initiation of containment isolation.

IV. Discuss the design provisions for testing the operability of the isolation valves.

V. Identify the codes, standards, and guides applied in the design of the containment isolation system and system components.

VI. Discuss the normal operating modes and containment isolation provision and procedures for lines that transfer potentially radioactive fluids out of the containment.

Attachment 3

Additional Systems and Operational Information Required

I. Provide copies of the procedures for loss of feedwater and small break LOCA.

II. Discuss the reactor water level measurement system. In particular:

1. Provide a diagram showing location of pressure taps used in measuring level. The diagram should be detailed enough to show whether the measurement is inside or outside the core shroud.
2. Describe the instrument piping arrangements and types of transducers used.
3. Which levels are monitored in the control room and how are they indicated (i.e., recorders, meters)?
4. Which measurements provide signals for safety systems, which for control systems, which for other systems?
5. Describe the dynamic response of each of the level measurement and indicating instruments for conditions typical of a small break LOCA.
6. What are the level measurement uncertainties?
7. What level difference is expected between core and measurement location for:
   a. normal operations,
   b. reactor shutdown with decay heat and with recirculation pumps running,
   c. reactor shutdown with decay heat and recirculation pumps not running, and
   d. moderate level transient as for a small break LOCA or stuck open SRV.

699130

ENCLOSURE 2

REQUESTS FOR ADDITIONAL INFORMATION

BULLETINS & ORDERS ANALYSIS GROUP

699131

Enclosure 2

REQUEST FOR ADDITIONAL INFORMATION

REGARDING SMALL BREAK LOCA ANALYSIS

I. The response of the reactor system of a given plant to a small break LOCA will differ greatly depending upon the break size, the location of the break, mode of operation of the recirculation pumps, number of ECCS systems functioning, and the availability of isolation condensers or RCIC. In addition, this response may differ for different plants designed by the same NSSS vendor because of differences in the recirculation loop configuration or different ECCS designs. In order for the staff to complete its evaluation of the response of currently operating BWR designs to postulated small break LOCA's, the following information is needed:

(1) Provide a qualitative description of expected system behavior for (a) a range of postulated small break LOCA's, including the zero break case, and (b) feedwater-related limiting transients combined with a stuck-open safety/relief valve. These cases should include situations where HPCI and RCIC (or isolation condenser) are assumed available and not available. The cases considered should also include breaks large enough to (a) depressurize the reactor coolant system, (b) maintain the reactor coolant system at some intermediate pressure and (c) repressurize the primary system to the safety/relief valve setpoint pressure. Various break locations in the reactor coolant system should be considered.

(2) Provide a qualitative description of the various natural circulation modes of expected system behavior following a small break LOCA. Discus any ways in which natural circulation can be degraded, such as fluid stratification in the lower plenum caused by inoperation of the cleanup system. Assess the possible effects of non-condensible gases.

II. The following questions pertain to your small break LOCA analysis methods:

(3) Demonstrate that your current small break LOCA analysis methods are appropriate for application to each of the cases identified in items (7) through (10) below. This demonstration should include an assessment of the adequacy of system noding potential counter current flow limitations, and water accumulation above the core.

If, as a result of the above assessment, you modify your analysis methods (e.g., system noding), provide justification for any such modification.

(4) Verify the break flow model used for each break flow location analyzed in the response to Item (7) below.

(5) Verify the analytical calculation of fluid level in the reactor vessel for small break LOCA's and feedwater transients.

(6) Provide integral verification of your small break loss-of-accident method through comparison with experimental data. TLTA and LOFT small break tests are possible examples.

III. For each of the analyses requested in Items (7) through (10) below.

(i) Provide plots of the output parameters specified in Table 1 of this enclosure.

(ii) Indicate when the System safety/relief valve would open.

(iii) Include appropriate information about the role of control systems in the course of the transient. Describe how the system response would be affected by control systems.

(iv) If the scenario is different for different classes of plants (jet pump, non-jet pump, BWR 4, BWR 5), provide an example of each kind.

(7) Provide the results of a sample analysis of each type of small break behavior discussed in the response to item (1) (e.g., depressurization, pressure hangup, repressurization).

(8) Provide the results of an analysis of the worst small break size and location in terms of core uncovering assuming a failure in the ECCS and the RCIC (or isolation condenser). This may be a break which does not result in HPCI initiation. This may require more than one calculation.

(9) Provide the results of an analysis for a single stuck open safety/relief valve, and the maximum number of valves that could open following the worst single failure.

(10) Provide the results of a small break LOCA analysis assuming loss of feedwater. The case with the worst break location which affords the least amount of time for operator action should be analyzed. A single failure in the ECCS and failure of the RCIC (or isolation condenser) should be considered.

(11) Provide a list of transients expected to lift the SRVs; identify the assumed steam and two-phase flow rates through the valves for these transients. Provide justification for your assumptions, including the time at which two-phase discharge, if it is calculated to occur, would be experienced.

(12) Provide revised emergency procedures or guidelines for the preparation of operational procedures for the recovery of plants following small LOCA's. This should include both short-term and long-term situations and follow through to a stable condition. The guidelines should include recognition of the event, precautions, actions, and prohibited actions.

If recirculation pump operation is assumed under two-phase conditions, a justification of pump operability should be provided. Discuss instrumentation available to the operator and any instrumentation that might not be relied upon during these events. What would be the effect of this instrumentation on automatic protection actions?

IV. In addition to the short term requirement identified above, it is requested that the following information be provided by November 1, 1979.

(13) Provide an analysis of the symptoms of inadequate core cooling and required operator actions to restore core cooling. These analyses should include cases assuming the recirculation pumps are both operating and not operating. The calculation should include the period of time during which inadequate core cooling is approached as well as the period of time during which inadequate core cooling exists. The calculations should be carried out far enough so that all important phenomena and instrument indications are included. Each case should then be repeated taking credit for correct operator action.

(14) Provide emergency procedures or guidelines for the preparation of emergency procedures for plant recovery from inadequate core cooling.

(15) Provide revised emergency procedures or guidelines for the updating of emergency procedures for accidents and transients considered in Section 15 of plant SAR's.

(16) The NRC is planning to perform audit calculations of the BWR small break LOCA. The necessary computer program input information and comparative calculations should be provided to facilitate this study. To assist in the review of these cases, we will require computer output information in excess of that specified in Table 1.

TABLE 1

Plotted Output Parameters

Core: $\underline{L}$, $X_{AVG.}$, W, T_{clad}

Reactor Vessel:

Lower Plenum: L, X - or T_{SUB}, P

Downcomer: L, X or T_{SUB}

Leak:

SRV, W, X

or

Break, $\underline{W}$, $\underline{X}$, $\int \underline{Wdt}$

Nomenclative: P - Pressure
L - Mixture Level
X - Quality
T - Temperature
W - Mass Flow Rate
H - Enthalpy