Omaha Public Power District

1623 HARNEY ■ OMAHA, NEBRASKA 68102 ■ TELEPHONE 536-4000 AREA CODE 402

May 17, 1982
LIC-82-198

Mr. Robert A. Clark, Chief
U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Division of Licensing
Operating Reactors Branch No. 3
Washington, D.C. 20555

Reference: Docket No. 50-285

Dear Mr. Clark:

Subject: Pressurized Thermal Shock

The Commission's letter dated April 22, 1982 and recent discussions with our NRC Project Manager, Mr. Ed Tourigny, have confirmed that the Commission intends to visit the Fort Calhoun Station the week of June 7, 1982 to conduct an audit of procedures and training related to the pressurized thermal shocking (PTS) issue. Mr. Tourigny requested that the District provide the Commission with information regarding the Fort Calhoun Station's operator training and the operating and emergency procedures relating to PTS prior to the plant visit. Accordingly, the requested information is attached. This information includes an outline and lesson plan for the RO/SRO training program and PTS related Fort Calhoun Station operating and emergency procedures.

Sincerely,

W. C. Jones
Division Manager
Production Operations

Attachments

cc: LeBoeuf, Lamb, Leiby & MacRae
1333 New Hampshire Avenue, N.W.
Washington, D.C. 20036

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FORT CALHOUN STATION RO/SRO TRAINING PROGRAM
REGARDING PRESSURIZED THERMAL SHOCKING (PTS)

A. Training-Lectures

1. Content

a) Elements of PTS

1) Fort Calhoun vessel material

2) Pre-existing flaw

3) Thermal shock due to cooldown

4) Pressure

b) Probable Event Initiators

c) Sources of Pressure

d) Emergency Procedure Actions

1) Control RCS pressure

2) Observe Technical Specifications cooldown rate limits

e) Fort Calhoun System Response

1) Large SLB

2) Stuck open SG safety valve

3) Small break LOCA

f) Review of PTS Experience

1) Fort Calhoun excess feedwater event

2) ANO-2 excess heat extraction

3) Rancho Seco

g) Competing Requirements of Core Cooling and PTS

h) Effect of RCS Head Void on Pressure Control

2. Schedule

a) Training to be Completed by June 1, 1982

B. Quizzes

1. Quizzes are normally given after each lecture to determine the operator's knowledge of the subject and the effectiveness of the training.

FORT CALHOUN STATION
PRESSURIZED THERMAL SHOCK LESSON PLAN

1. Elements of Pressurized Thermal Shock

a. Fort Calhoun Station Reactor Vessel Material:

The reactor vessel welds, as compared to the reactor vessel plate material, have a higher RT_{NDT} shift due to a higher copper content and higher fluence. The reactor vessel plate material is of less concern because of its lower copper content.

b. Pre-Existing Flaw:

A pressurized thermal shock event can only damage the vessel if a flaw exists in the reactor vessel material. These flaws may be on the lower limits of inservice inspection detectability limits and, thus, may not be detected.

c. Thermal Shock Due to Cooldown:

An overcooling event can only be a potential pressurized thermal shock event if the cooldown is $\geq 100^{o}F$ at a rate $\geq 100^{o}F/hr$ and achieves a low temperature $\leq RT_{NDT} + 100^{o}F$. The transient must last at least ten minutes.

d. Pressure:

In order for a pressurized thermal shock event to occur, the thermal shock must occur when the system is at a significantly high pressure (i.e., $\geq$ 1000 psia as an arbitrary example).

2. Probable Event Initiators

The most probable events to cause pressurized thermal shock are excess heat extraction from the secondary system, a small break LOCA, or a steam generator tube rupture. Excessive secondary system heat extraction could result from a steam line break or excessive feedwater addition. The main feedwater isolation system decreases the probability that an excess feedwater event could cause a pressurized thermal shock event. However, the Fort Calhoun Station experienced one case of overcooling during an excess feedwater event in 1974.

3. Sources of Pressure

The primary sources of pressure are the high pressure safety injection and charging pumps. Secondary and long term sources are the decay heat removal system and pressurizer heaters.

4. Emergency Procedure Actions

a. Controlling Reactor Coolant System Pressure:

Observe the high pressure safety injection and charging pump termination criteria in the emergency procedures. This is the primary action to be taken to reduce the severity of a pressurized thermal shock event. The operation of the charging pumps should be terminated. One high pressure safety injection pump may be stopped, but the other two should be maintained in the recirculation mode to be immediately available if the subcooling criteria are not being maintained.

Warning: Because of the relatively large capacity of the high pressure safety injection pumps compared to reactor coolant system volume, the Fort Calhoun Station reactor coolant system can be rapidly refilled and overpressurized.

b. Observe Technical Specification Cooldown Rate Limits:

The 100°F/hr cooldown rate curve provides the best guidance to minimize the potential for pressurized thermal shock induced damage. This cooldown curve should be carefully maintained at all times. If a transient takes the reactor coolant system to the left of the 100°F/hr curve (operation not allowed), the reactor coolant system shall be depressurized. Depressurization can be accomplished by shutting off the charging pumps and throttling the high pressure safety injection flow, utilizing the auxiliary pressurizer sprays, simultaneously depressurizing the steam generators, or finally, utilizing the power operated relief valves.

5. Fort Calhoun Station System Response

A review of the calculated primary and secondary system temperatures and pressures and pressurizer and steam generator levels for the large steam line break, stuck open steam safety valves and small break LOCA.

6. Review of Pressurized Thermal Shock Experience

a. 1974 Fort Calhoun Station Excess Feedwater Event:

Review the circumstances of the Fort Calhoun Station excess feedwater event and available strip charts.

b. ANO-2 Excess Heat Extraction:

A stuck open steam safety valve did occur at ANO-2 and rediscuss the Fort Calhoun Station response to such an incident.

c. Rancho Seco:

Discuss the Rancho Seco transients and how the Fort Calhoun Station design makes this event relatively unlikely.

7. Competing Requirements of Core Cooling and Pressurized Thermal Shock

The reactor coolant system pressure and temperature must be maintained between two limits. One is defined by the 50°F subcooling line and the second is defined by the 100°F/hr cooldown rate curve. The first priority is always to maintain 50°F subcooling and the second is to not violate the 100°F/hr cooldown rate curve.

8. Effect of Reactor Coolant System Head Void on Pressure Control

During a rapid depressurization, a void can form in the reactor coolant system head and, with high pressure safety injection flow and subsequent subcooling, this bubble will collapse. After about one hour of natural circulation, the bubble can reform. Indication of a bubble is a rapid increase in pressurizer level with little decrease in pressure when the auxiliary pressurizer sprays are activated. If a bubble is formed, pressure control with the auxiliary sprays is not effective. The bubble should be eliminated by starting a reactor coolant pump. If a reactor coolant pump cannot be continuously operated, "bumping" a reactor coolant pump will probably force enough cool water into the head region to collapse the bubble and allow utilization of auxiliary sprays to subsequently control the reactor coolant system pressure.