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POWER BUILDING
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WILLIAM O. PARKER, JR.
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TELEPHONE: AREA 704
373-4083

October 9, 1981

Mr. J. P. O'Reilly, Director
U.S. Nuclear Regulatory Commission
Region II
101 Marietta Street, Suite 3100
Atlanta, Georgia 30303

Re: McGuire Nulcear Station Unit 1
Docket No. 50-369

Dear Mr. O'Reilly:

Please find attached Reportable Occurrence Report RO-369/81-151. This report concerns T.S.3.4.10.1, "The Reactor Coolant System (except the pressurizer) Temperature and Pressure Shall be Limited in Accordance with the Limit Lines Shown on Figures 3.4-2 and 3.4-3 During Heatup, Cooldown, Criticality, and In-service Leak and Hydrostatic Testing with: . . . b. A Maximum Cooldown of 100°F in any one Hour Period . . . ". This incident was considered to be of no significance with respect to the health and safety of the public.

This report is also being submitted pursuant to T.S.6.9.2 in fulfillment of the special report provision of T.S.3.5.2 (Action b).

Very truly yours,

William O. Parker, Jr.

PBN/smh

Attachment

cc: Director
Office of Management and Program Analysis
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Ms. M. J. Graham
Resident Inspector-NRC
McGuire Nuclear Station

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McGUIRE NUCLEAR STATION

REPORTABLE OCCURRENCE

REPORT NUMBER: 81-151

REPORT DATE: October 9, 1981

OCCURRENCE DATE: September 14, 1981

FACILITY: McGuire Unit 1; Cornelius, N. C.

IDENTIFICATION OF OCCURRENCE: During the performance of the Loss of Control Room Test, operators had trouble controlling the Auxiliary Feedwater (CA) System. An excessive amount of feedwater was pumped into the steam generators (S/G) causing an abnormally fast cooldown rate and a safety injection (SI) actuation.

CONDITIONS PRIOR TO OCCURRENCE: Mode 1, Power Operation (20% power)

DESCRIPTION OF OCCURRENCE: The following is a chronological listing of the events that occurred during the incident.

01:12 About this time, the test started; operators left the control room to man the reactor trip circuit breakers (located in a room adjacent to the control room), the auxiliary shutdown panel, and the CA pump control panels (located in the CA pump room several floors below the control room).

01:13:19 Operators manually tripped the reactor from the reactor trip breakers.

Almost immediately after the reactor was tripped, the main feedwater (CF) isolation valves closed on coincident reactor trip and T_{ave} less than 564°F and S/G levels dropped to the Lo-Lo level setpoint which resulted in automatic start signals of the three CA pumps. All three CA pumps started and pumped water into the S/G's at maximum flow-rate since the CA throttle valves had been fully opened by the automatic start signal. Steam generator levels began increasing. During the next several minutes operators tried unsuccessfully to gain control of the CA throttle valves. As a result, the S/G's filled rapidly with the excessive CA flow. Pressurizer level and NC pressure fell at a corresponding rate.

01:20:26 Operators started the second centrifugal charging pump to recover pressurizer level and NC pressure.

As NC pressure dropped below the P-11 setpoint (1955 psig), operators blocked SI to prevent further complicating the transient. Both charging pumps were running at this time.

01:21:06 Pressurizer level indication decreased to zero. Operators tripped the 4 NC pumps.

01:30:14 As the two charging pumps refilled the pressurizer, NC pressure was also restored. The minimum NC pressure reached was approximately 1900 psig. When the NC pressure exceeded the P-11 setpoint, the SI block was cleared and SI was initiated by S/G low pressure. The main steam isolation valves closed due to the SI which helped stem the heat loss from the NC system.

01:33:40 SI was reset on both trains. The two D/G sequencers were reset and the recovery proceeded smoothly from this point on except that a second SI was initiated on train A when the reactor trip circuit breakers on that train were reset without sufficient S/G pressure. The reactor trip breakers were reset in an effort to satisfy the CF valve logic and open the CF isolation valves.

APPARENT CAUSE OF OCCURRENCE: The operators were unable to gain control of the CA flow to the S/G's due to a combination of the following: Design Deficiency--the lack of CA flow control reset switches on the auxiliary down panels; Procedural Deficiency--the procedure specifying when the reactor would be tripped relative to the manning of the auxiliary shutdown panels and no mention of the CA flow control reset switches in the control room; and Personnel Error--possibly due to inadequate training or experience in controlling CA flow from the auxiliary shutdown panels.

ANALYSIS OF OCCURRENCE: CF isolation on low T_{ave} coincident with reactor trip will occur on virtually every reactor trip. This response and the rapid S/G level drop which resulted in the automatic start of all three CA pumps are characteristics of Westinghouse units similar to McGuire Unit #1. S/G levels dropped from the normal operating values to the Lo-Lo setpoint in less than five seconds. During this event, the lack of significant decay heat addition to the primary system exacerbated the cooldown effect.

On an automatic start signal to the CA system, solenoid valves interrupt the air supply to each of the CA control valves and the valves go to the fully open position. Recirculation valves on each pump go to the fully closed position. The solenoid valves must be reset to allow normal operation of the CA throttle and recirculation valves. Resetting all of the CA control valves following an automatic start (other than SI) is done with two mechanically latching switch modules located in the control room. The CA control valves can be operated, following an automatic start, if reset switched are depressed and latched and the proper control station is selected on the auxiliary shutdown panels. The sequence of these actions is irrelevant.

Since the reactor was tripped as soon as the first group of operators reached the reactor trip breakers, the transient was well underway when the second group of operators reached the auxiliary shutdown panels (the auxiliary shutdown panels are much further from the control room than the reactor trip breakers). The operators were unaware of the rapid level drop characteristics of the S/G's following reactor trip and CF isolation. Upon reaching the auxiliary shutdown panels the operators tried unsuccessfully to reset the CA flow controls. Operators that were stationed in the control room depressed the reset pushbuttons located there with no apparent results. The CA flow logic has since been tested completely including the equipment, wiring and functional tests of the circuitry. Logic involved with the two CA flow control reset switch modules is completely separate and the possibility of a double failure is judged to be remote. Some confusion was present during the test resulting from the rapidity and severity of the transient and the lack of experience in controlling the plant from the auxiliary shutdown panels.

During the resulting transient, NC temperatures exceeded the 100°F/hour cooldown rate addressed in Technical Specification 3.4.10.1. Maximum temperature change during the hour following the reactor trip was 116.8°F. Excessive cooling by the S/G's, the large amount of water injected into the NC system during the initial level recovery and the tripping of the NC pumps contributed to the excessive cooldown.

The operators tripped the NC pumps out of a perceived need to protect the pumps from damage due to loss of level in the pressurizer. This action, although specified in the Emergency Procedure, was premature in that the conditions specified for tripping the pumps were not reached. In this particular event, tripping the pumps did not pose a safety concern, due to the lack of significant decay heat generation. However, this action was not appropriate.

SAFETY ANALYSIS: During the test all of the safety systems worked as designed in their emergency configurations. There was never any danger of uncovering the core or damaging the fuel. Sufficient subcooling margins were maintained at all times. The test was designed to uncover any problems the operators would have in controlling the plant without the use of the control room.

Westinghouse reviewed the data from the cooldown transient and concluded that although the technical specification limit for 100°F/hour cooldown of the reactor vessel was exceeded, no fracture concern exists because the absolute temperatures remained high and the ASME Code Section III Appendix G stress limits were not exceeded. In addition, the fatigue usage factor contribution of this transient is insignificant. Therefore, this transient did not create a safety concern for continued safe operation of the plant.

CORRECTIVE ACTION: Both the Loss of Control Room test procedure and operating procedure were revised to include depressing the CA flow control reset switches before leaving the control room and establishing communications between the reactor trip breakers and the auxiliary shutdown panels before the reactor is tripped. CA flow control reset switches will be installed on the auxiliary shutdown panels. Duke Power Company is developing plans to modify the auxiliary shutdown panels. The changes should clarify the operation of equipment and monitoring of system parameters from these panels. Additional guidance for maintaining forced flow in the NC system will be developed and provided to the operators.

A second Loss of Control Room Test was conducted on September 17, 1981 using the revised procedure. In this test the operators were able to control the CA flow to the S/G's and the NC system parameters responded as expected. All of the acceptance criteria were met.