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DUKE POWER

April 24, 1990

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: Catawba Nuclear Station
Docket No. 50-413
LER 413/90-22

Gentlemen:

Attached is Licensee Event Report 413/90-22 concerning TECHNICAL SPECIFICATION PRESSURIZER TEMPERATURE LIMITS VIOLATED DUE TO MANAGEMENT DEFICIENCY.

This event was considered to be of no significance with respect to the health and safety of the public.

Very truly yours,

Tony B. Owen

Tony B. Owen
Station Manager

keb\LER-NRC.TBO

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NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES 8/31/88

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Catawba Nuclear Station, Unit 1	0 5 0 0 0 4 1 3	1 OF 0 9

TITLE (4) Technical Specification Pressurizer Temperature Limits Violated Due To Management Deficiency

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
0 3	2 5	9 0	9 0	0 2 2	0 0	0 4	2 5	9 0	N/A	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 5

POWER LEVEL (10): 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

[ ] 20.402(b)	[ ] 20.405(c)	[ ] 50.73(a)(2)(iv)	[ ] 73.71(b)
[ ] 20.405(a)(1)(i)	[ ] 50.36(c)(1)	[ ] 50.73(a)(2)(v)	[ ] 73.71(c)
[ ] 20.405(a)(1)(ii)	[ ] 50.36(c)(2)	[ ] 50.73(a)(2)(vii)	[ ] OTHER (Specify in Abstract below and in Text, NRC Form 366A)
[ ] 20.405(a)(1)(iii)	[X] 50.73(a)(2)(i)	[ ] 50.73(a)(2)(viii)(A)	
[ ] 20.405(a)(1)(iv)	[ ] 50.73(a)(2)(ii)	[ ] 50.73(a)(2)(viii)(B)	
[ ] 20.405(a)(1)(v)	[ ] 50.73(a)(2)(iii)	[ ] 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER	
	AREA CODE	
R.M. Glover, Compliance Manager	8 0 3	8 3 1 - 3 2 3 6

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

[ ] YES (If yes, complete EXPECTED SUBMISSION DATE) [X] NO

EXPECTED SUBMISSION DATE (15): MONTH | DAY | YEAR

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

On March 25, 1990, Catawba Unit 1 was operating in Mode 5, Cold Shutdown, with preparations underway to perform the Engineered Safety Features Actuation Periodic Test on Train A. Test initiation began at 2300 hours and resulted in injection of approximately 5000 gallons of water into the Reactor Coolant System. The resultant insurge carried relatively colder water into the pressurizer and resulted in the 200 degree F per hour cooldown limit being exceeded. Conclusion of the test and action to reduce pressurizer level and pressure resulted in an outsurge and subsequent heatup in excess of the 100 degrees F per hour heatup limit. A subsequent portion of the test was performed on March 26 with similar results. This incident is attributed to a Management Deficiency; a revision of the test procedure was not prepared to address the potential impact of the injection on pressurizer temperature. A second injection occurred as a result of an Inappropriate Action in not performing an appropriate engineering evaluation after the first injection. Appropriate procedure revisions have been made. An engineering evaluation of the effects on the pressurizer has been performed with acceptable results. Feedback will be provided on pressurizer temperature limitations and the need for proper evaluation of transients.

NRC Form 366 (9-83)

BACKGROUND

The Engineered Safety Features (ESF) Actuation Periodic Test, PT/1(2)/A/4200/09, is performed in accordance with Section 7.3 of the Catawba FSAR and applicable Technical specifications as listed in Section 2.0 of the procedure. Sections 12.1 and 12.4 of this test verify that ESF components actuate correctly to their safety position within their required time limits in response to a Safety Injection, Phase A Isolation, Phase B Isolation, and a Blackout. In addition, proper operation of the Diesel Generator [EIIS:GEN] (D/G) is verified. Sections 12.2 and 12.5 are essentially a repeat of 12.1 and 12.4 except there is not a blackout of the essential switchgear and the signals are generated using the manual pushbuttons in the Control Room. Sections 12.3 and 12.6 verify proper D/G starting, load shedding, and load sequencing in response to a blackout. Correct valve [EIIS:V] movements, response times and system responses are verified. Sections 12.7 and 12.8 verify that each cold leg accumulator valve opens when a simulated Reactor Coolant [EIIS:AB] (NC) System pressure signal exceeds the P-11 setpoint and upon receipt of a Safety Injection Signal. Sections 12.9 and 12.10 verify the proper operation and response time of the automatic swapover to Containment recirculation upon a Safety Injection coincident with Lo-Lo Refueling Water Storage Tank (FWST) level. Section 12.11 verifies that the Main Steam Isolation valves, Main Steam Isolation Bypass valves and the Steam Generator [EIIS:HX] (S/G) Power Operated Relief Valves (PORVs) close upon a manual Main Steam isolation.

The pressurizer is a vertical, cylindrical vessel with hemispherical top and bottom heads constructed of carbon steel, with austenitic stainless steel cladding on all internal surfaces exposed to the Reactor Coolant. A stainless steel liner or tube may be used in lieu of cladding in some nozzles. The pressurizer surge line connects the pressurizer to one Reactor hot leg and enables continuous coolant volume pressure adjustments between the NC System and the pressurizer.

The surge line nozzle and removable electric heaters [EIIS:EHTR] are located in the bottom of the pressurizer. A thermal sleeve is provided to minimize thermal stresses in the surge line nozzle. A retaining screen at the nozzle prevents any foreign matter from entering the NC System and baffles in the lower section of the pressurizer prevent an insurge of cold water from flowing directly to the steam/water interface and assist in mixing.

Equipment which serves as part of the pressure boundary in the NC loop include the S/Gs, the NC pumps, the pressurizer, and the Reactor vessel [EIIS:VSL]. This equipment is ANS Safety Class 1 and the pressure boundary meets the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB. This equipment is evaluated for the loading combinations associated with Design, Normal, Upset, and Faulted Condition Classifications.

The equipment is analyzed for 1) the normal loads of dead-weight, pressure and thermal, 2) mechanical transients of operating base earthquake, safe shutdown earthquake, and pipe [EIIS:PSP] ruptures, including the effects of asymmetric pressurization, and 3) pressure and temperature transients outlined in Section 3.9.1.1 of the Catawba FSAR.

To provide the necessary high degree of integrity for the equipment in the NC System, the transient conditions selected for equipment fatigue evaluation are based upon a conservative estimate of the magnitude and frequency of the temperature and pressure transients resulting from various operating conditions in the plant. To a large extent, the specific transient operating conditions to be considered for equipment fatigue analyses are based upon engineering judgement and experience. The transients selected are representative of operating conditions which prudently should be considered to occur during plant operation and are sufficiently severe or frequent to be of possible significance to component cyclic behavior. The transients selected may be regarded as a conservative representation of transients which, used as a basis for component fatigue evaluation, provide confidence that the component is appropriate for its application over the design life of the plant.

The heatup and cooldown transients analyzed for the pressurizer are based upon temperature transients of 100 degrees F per hour and 200 degrees F per hour, respectively. Technical Specification 3.4.9.2 states that pressurizer temperature shall be limited to these rates of change. The associated Action Statement stipulates that, with the temperature limits in excess of the above limits, temperature is to be restored within the limits within 30 minutes; an engineering evaluation is to be performed to determine the effects of the out-of-limit condition on the structural integrity of the pressurizer; and a determination made that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 500 psig within the following 30 hours.

The Chemical and Volume Control [EIIS:CB] (NV) System provides Reactor Coolant System charging, letdown, and Reactor Coolant pump seal water injection. The charging and letdown functions are employed to maintain a programmed water level in the pressurizer. This is achieved by means of a continuous feed and bleed process during which the feed rate is automatically controlled by pressurizer water level (during normal operation). During shutdown conditions, without Reactor Coolant pumps running, letdown is accomplished via the Residual Heat Removal [EIIS:BP] (ND) System suction lines off NC Loops B and C. The discharge of the ND Pumps can be aligned to the NV pump suction. NC charging flow from the Centrifugal Charging Pumps is injected into NC Loops A cold leg. A portion of charging flow is directed to the Reactor Coolant pumps via the seal water injection filters [EIIS:FLT], entering the pumps at a point between the labyrinth seals and the number 1 face seal. Two parallel cold leg injection line isolation valves, NI-9A and NI-10B, are provided to open on a Safety Injection signal. When these normally closed valves open, a path is provided from the Centrifugal Charging Pumps to the NC cold legs.

EVENT DESCRIPTION

Catawba Unit 1 was in Mode 5, Cold Shutdown, on March 25, 1990, with preparations underway to perform the Engineered Safety Features Actuation Periodic Test, PT/1(2)/A/4200/09, for Train A. NV Pump 1A was aligned in the recirculation mode per the procedure and NV Pump 1B was aligned to provide NC charging and NC pump seal injection flow. None of the NC pumps were running, thus pressurizer spray was not available; pressurizer heaters were not in service. Continued NC pump seal injection was desirable to prevent reverse flow through the seals due to higher NC pressure.

Previous performance of this test had occurred during the early stages of refueling outages; a decision had been made to perform the test this time in the latter stages of the Unit 1 end of cycle 4 outage. Conditions differed during the current test in that the NV pump in the train not under test (i.e. Pump 1B) was in operation; during previous tests it was not in operation.

The Operations Shift Supervisor and Performance Test Coordinator discussed the effects of the operating NV pump. Their concern focused on the affect on pressurizer level and pressure of water injection through NI-9A, the Train A Safety Injection valve. Although the test procedure states that the ESF actuation signal is not to be reset until the last D/G Sequencer load group has been connected, about 12 minutes after test initiation, discussions were held and provisions were made for Control Room personnel (CROs) to reset the actuating signal and close the safety injection valves sooner if conditions warranted. These discussions did not consider the effects of cold water injection on pressurizer temperature limits.

At 2234 hours, on March 25, 1990, the ESF Train A Blackout/LOCA test (Section 12.1) was initiated. System and equipment responses were verified; 1NI-9A opened as expected. Approximately 480 gpm were injected from the FWST for a 10 minute period. At 2244 hours, CROs reset the ECCS signal and closed 1NI-9A. During the injection period, pressurizer level increased from approximately 40% to 70%; NC pressure increased from approximately 190 to 235 psig. The resultant pressurizer insurge carried relatively cold NC Hot Leg water (135 degrees F) into the hotter pressurizer. Indicated surge line temperature decreased from approximately 290 to 135 degrees F. Indicated pressurizer water temperature decreased from approximately 380 to 140 degrees F. The Operator At the Controls (OATC), who was logging NC heatup data pursuant to PT/1/A/4600/16, Surveillance Requirements for Unit 1 Startup, Enclosure 13.5 reported a cooldown rate in excess of the Technical Specification limit (200 degrees F per hour).

CROs initiated action to reduce pressurizer level and pressure. NC pressure decreased from 235 to 180 psig; pressurizer level decreased toward its original value. The resultant pressurizer outsurge carried relatively hot pressurizer water into the NC Loop B Hot Leg. Indicated hot leg temperature increased to approximately 160 degrees F; indicated pressurizer surge line temperature increased from 135 to 215 degrees F; pressurizer water temperature increased from 140 to 340 degrees F and subsequently increased further to 380 degrees F.

CROs, the Operations Shift Supervisor, and the Unit Supervisor evaluated the available data and indications. It was concluded that a significant temperature transient affecting the pressurizer/surge line metal had not occurred. This conclusion was based upon observation that the pressurizer water temperature had returned to its original, pre-test value; similarly the pressurizer vapor temperature remained nearly constant at 375 degrees F. These results combined with the absence of heat input to the NC System, and to the pressurizer in particular (neither NC System pumps nor pressurizer heaters were operating), led to a conclusion that the observed temperature response was most indicative of water temperature and not of component temperature.

Based on the above evaluation, it was concluded that continuation of the ESF test was acceptable. Preparations were made and the Train A LOCA test (Section 12.2) was initiated at 0303 hours on March 26. Similar system response was seen as before due to the resultant injection through 1NI-9A. Subsequent portions of the Train A test were performed; however these did not result in the opening of either 1NI-9A or 1NI-10B. Hence no additional transient in pressurizer temperatures occurred.

The Operations Shift Supervisor on duty during this testing included the pressurizer temperature transient in his morning turnover. As a result, Problem Investigation Report 1-C90-0099 was initiated at 0700 hours on March 26, 1990, to continue the assessment of the event. The NSSS vendor, Westinghouse Electric Corporation, was requested by Design Engineering to perform an evaluation of the data associated with the event.

The evaluation performed by Westinghouse, and concurred to by Design Engineering, concluded that Technical Specification 3.4.9.2 conditions "a" (heatup) and "b" (cooldown) were exceeded during each of the two temperature transients in the pressurizer. The evaluation was based on the data provided by Duke Power Company and evaluated in comparison to several similar heatup and cooldown events experienced by other plants, that were previously evaluated/analyzed by Westinghouse. The evaluation provided by Westinghouse stated that neither the "design life" nor the pressurizer structural integrity have been compromised by the (2) temperature transients experienced at Catawba Unit 1 on March 25 and 26, 1990. A more detailed engineering evaluation will be provided (as additional documentation) by Westinghouse at a later date.

Based on Westinghouse's evaluation, Design Engineering concurred that the structural integrity of the Unit 1 pressurizer is acceptable for continued power escalation and thus the plant is considered "operable".

CONCLUSION

During this incident, the pressurizer cool down rate limit in Technical Specification Section 3.4.9.2 was exceeded; however the associated Action Statement was met. Similarly, the heatup rate limit was exceeded; however the Action Statement was not met.

This incident, violation of Technical Specification Section 3.4.9.2, is attributed to a Management Deficiency, deficient procedure preparation and issuance. The decision to perform this test under conditions different than previously encountered did not result in a procedure change to address the potential impact of the injection on pressurizer temperature. The Operations Shift Supervisor and Performance Personnel accurately assessed the effects of the operating NV pump on NC pressure and pressurizer level prior to the test. They established provisions for terminating the test if these parameters reached undesirable conditions. Their assessment did not, however, recognize the potential affects on pressurizer temperature limits. Criteria and actions for terminating the test were not incorporated into the test procedure.

The assessment by Operations personnel of the first test results considered the available data. The conclusion to proceed with subsequent portions of the test without action to prevent injection through 1NI-9A was reached based upon a misinterpretation of the significance of pressurizer temperature data. The Performance Test Coordinator was not involved in this evaluation; neither were Design Engineering personnel. The decision to proceed with the second portion of the test, based upon a less than adequate evaluation of the results of the first, is attributed to an Inappropriate Action, action taken was not the best alternative.

In response to the determination that the pressurizer heatup and cooldown limits had been violated, the ESF test procedure has been revised to remove power from valves NI-9A and NI-10B to preclude their opening during testing of their respective train. Subsequent performance of the Train B ESF Actuation Periodic Test was successfully accomplished.

This event will be reviewed with Operations shift personnel with emphasis on the need to request appropriate support for evaluation of plant transients, the need to keep Performance Test Coordinators informed of abnormalities, and limitations on the cool down of pressurized vessels. This event will be reviewed with appropriate Performance and Operations personnel with emphasis on the need to incorporate into test procedures the special measures/actions needed to control plant conditions, including test termination criteria and actions.

Additional procedures for tests involving the potential for water injection will be reviewed to ensure that adequate precautions and guidance are given to control plant conditions and modes.

A review of the past 24 months experience in the Operating Experience Program database identified one previous event involving testing with a root cause of procedure deficiency leading to a Technical Specification violation. LER 413/89-023, Revision 1 involved inoperability of both trains of the Control Room Area Ventilation [EIIS:UC] (VC) System due to defective pre-operational testing procedures. Also, LER 414/90-001 involved an auxiliary feedwater [EIIS:BA] (CA) autostart due to performing a test with both main feedwater pumps tripped and the CA autostart enabled; the procedure did not require compensating measures to prevent the autostart. LER 414/89-013 involved a feedwater isolation on hi-hi steam generator level during valve stroke time testing due to inadequate procedure precautions concerning Unit status. The dissimilar aspect of the current event is that the effect of the operating NV pump was recognized, in terms of its affect on pressurizer level and pressure; the effect on pressurizer temperature limits was not recognized. Due to this dissimilarity between the previous events cited and the current event, it is concluded that previous corrective actions could not have have prevented this event. This is not considered a recurring problem.

CORRECTIVE ACTION

SUBSEQUENT

1) CROs closed 1NI-9A to terminate injection flow.

2) PIR 1-C90-0099 was initiated to request Design Engineering evaluation of the recorded pressurizer temperature data.

3) Westinghouse initiated an evaluation of the data and concluded that the design life and the pressurizer structural integrity were not compromised. Design Engineering concurred that continued operation was acceptable.

4) The ESF test procedure was revised to remove power from NI-9A or NI-10B to preclude their opening during testing of their respective train.

PLANNED

1) Westinghouse will complete and send to Duke Power a more detailed engineering evaluation including fatigue and fracture analyses to determine the specific effect of this type of pressurizer cool down on the design life of the plant. This LER will be revised if significantly different results are reached.

2) Operations shift personnel will review this event with emphasis on:

-The need to request appropriate support for evaluation of plant transients;

-The need to keep Performance Test Coordinators informed of abnormalities;

-Limitations on the cool down of pressurized vessels with respect to thermal shock and stress minimization.

3) Appropriate Performance and Operations personnel will review this event with emphasis on the need to incorporate into test procedures the special measures/actions needed to control plant conditions, including test termination criteria and actions.

4) Other procedures for tests involving the potential for water injection will be reviewed to ensure adequate precautions and guidance are given to control plant conditions and modes.

SAFETY ANALYSIS

The engineering evaluation performed by Westinghouse provides the basis for the following conclusion:

During the evening of March 25, and early morning of March 26, 1990, two pressurizer cooldown transients occurred at Catawba Unit 1. The transient histories and relevant parameter data describing the transients were provided to Westinghouse. The data showed the plant experienced cooldown rates in excess of 200 degrees F per hour.

Over the past 3 years, Westinghouse has performed evaluations of off-normal heatup and cooldown transients at over ten plants. The scenarios are similar, with a large insurge causing rapid cooldown in the lower shell regions of the pressurizer, with a possible subsequent heatup to recover. The temperatures involved, as well as plant mode and operating status, are also similar.

The temperature and pressure data for the Catawba Unit 1 transients were reviewed and compared with similar events at other plants as described above, as well as with evaluations of historical operating records performed as part of transients and fatigue cycle monitoring programs performed by Westinghouse for several plants. Based on this review, it is our judgement that the allowable pressurizer fatigue life has not been approached to date and that pressurizer structural integrity has not been compromised.

A detailed engineering evaluation, including fatigue and fracture analysis, to determine the specific effect of this type of pressurizer cooldown on the design life of the plant (if any), will be performed. A letter report containing the results will be provided.

Thus the health and safety of the public were not affected by this event.