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AUTH.NAME AUTHOR AFFILIATION
LOWERY,H.R. Duke Power Co.
BARRON,H.B. Duke Power Co.
RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 91-SPECIAL-00:on 910311,pressurizer safety relief valve actuation setpoint drift occurred.Caused by procedural deficiency.Spare pressurizer safety relief valves actuation setpoint reset.W/910710 ltr.

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Duke Power Company
Oconee Nuclear Station
P.O. Box 1439
Seneca, SC 29679

(803)885-3000

DUKE POWER

July 10, 1991

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

Subject: Oconee Nuclear Station
Docket Nos. 50-269, -270, -287
Special Report Concerning Pressurizer
Relief Valve Setpoint Inaccuracies

Gentlemen:

This report is provided for information regarding pressurizer relief valve setpoint inaccuracies.

If you have any questions, please contact Rick Matheson at (803) 885-3119.

Very truly yours,

Joe M Davis
for H. B. Barron
Station Manager

RSM/ftr

Attachment

xc: Mr. S. D. Ebneter
Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
101 Marietta St., NW, Suite 2900
Atlanta, Georgia 30323

Mr. L. A. Wiens
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

INFO Records Center
Suite 1500
1100 Circle 75 Parkway
Atlanta, Georgia 30339

NRC Resident Inspector
Oconee Nuclear Station

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NRC FORM 366 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Oconee Nuclear Station, Unit 3	0 5 0 0 0 2 8 7	1 OF 0 9

TITLE (4) Pressurizer Safety Relief Valve Actuation Setpoint Drift Occurs Due to Unknown Causes, Possible Test Procedure 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	1 1	9 1	9 1	SPECIAL	0 0	0 7	1 0	9 1		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): N

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)	[X] OTHER *(Specify in Abstract below and in Text, NRC Form 366A)*
[ ] 20.405(a)(1)(iii)	[ ] 50.73(a)(2)(i)	[ ] 50.73(a)(2)(viii)(A)	SPECIAL REPORT
[ ] 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	
Henry R. Lowery, Chairman, Oconee Safety Review Group	8 0 3	8 8 5 - 3 0 3 4

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

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	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 4, 1991, with Unit 3 at cold shutdown and the Reactor Coolant System (RCS) depressurized for a refueling outage, notification was received from Wyle Laboratories that the actuation setpoints of the pressurizer code safety relief valves had been found outside the range specified in the basis of Technical Specifications. These valves had been removed and sent to Wyle earlier in the outage. Spare pressurizer relief valves, whose actuation setpoints had previously been set within the required band, were installed prior to RCS heatup and pressurization. An investigation was initiated using expertise from the valve manufacturer, Wyle Laboratories, and Duke Power. Testing and maintenance activities associated with seat leakage, seat repair, and other variables were investigated. The root cause is unknown, possible procedural deficiency. Procedure improvements have been implemented and further research has been initiated.

BACKGROUND

The Reactor Coolant System (RCS) [EIIS:AB] serves as a barrier to prevent radionuclides in the reactor coolant from reaching the atmosphere. System pressure limits have been established to assure the integrity of the RCS. The design pressure of the RCS is 2500 psig. The maximum transient pressure, as specified by American Society of Mechanical Engineers (ASME) code, Section III, Summer 1967, is 110 percent of design pressure. Thus, the safety limit for RCS pressure is 2750 psig. Technical Specification 3.1.1.c requires that both pressurizer code safety relief valves, RC-67 and RC-68, be operable whenever the reactor is critical and be set at 2500 psig. The basis of this Technical Specification indicates that there should be a plus or minus 1 percent allowance for error. One pressurizer relief valve must be operable whenever all RCS openings are closed. Pressurizer code safety relief valves are required by Technical Specifications to be tested every refueling outage, whereas ASME code, Section XI requires testing only once every five years.

The code pressurizer safety relief valves were manufactured by Dresser Industries. Since equipment to test and establish setpoints for these valves is not present on site, the valves are removed during scheduled refueling outages, shipped to a vendor (Wyle Laboratories), tested and adjusted as necessary, and returned to Oconee. Meanwhile, spare relief valves are installed to replace those being tested. When the tested valves are returned to Oconee, they become the spares. In this manner, each pair of relief valves is rotated between the three Oconee units.

A drawing of the safety relief valve internals is presented in Attachment 1. The valve disc is forced against its seat by a spring whose compression can be adjusted using an adjustable compression screw. The valve disc area exposed to RCS pressure increases when the valve lifts. Consequently, the valve will reseat at a lower pressure than its actuation setpoint. Middle and lower adjusting rings can also be set to affect the opening and closing setpoints of the valve.

EVENT DESCRIPTION

On February 19, 1991, during the last Unit 3 refueling outage, with the unit at less than 200 degrees F and at atmospheric pressure, the pressurizer code safety relief valves, 3RC-67 and 3RC-68, were removed from the pressurizer, packaged, and sent to Wyle Laboratories for testing and maintenance. On March 4, 1991, Nuclear Maintenance Engineer A and Design Engineer A received a notification by telephone from Wyle Laboratories personnel that the "as received" actuation setpoints of these two valves (tag numbers BT4976 and BT8894) had been found to be outside the allowed tolerance of one percent above the desired setpoint of 2500 psig. A Notice of Anomaly was sent by Wyle Laboratories to the same Duke personnel on March 11, 1991. It was found that valve BT4976 actuated at 2686 psig (plus 7.4 percent error) and valve BT8894 actuated at 2567 psig (plus 2.6 percent error) on the first actuation test.

Technical Specification 3.1.1.c.1 requires that the valves be operable when the reactor is critical. The basis states that they shall actuate within one percent of 2500 psig. Design Engineering performed an operability evaluation for the past Oconee Unit 3 fuel cycle. An analysis of the effects of the observed amount of relief valve setpoint drift was performed using a computer model of plant systems. This analysis used conservatively bounding core physics and operating assumptions based on the fuel cycle during which the relief valves were installed. It showed that the maximum safety limit of 2750 psig would not be exceeded during the most limiting accident with relief valves set at 7.4 and 2.6 percent above 2500 psig. The relief valve settings were established within one percent of design pressure by the vendor when the relief valves were installed. Spare relief valves, whose setpoints had been adjusted to within the required band, were installed on February 24, 1991 prior to startup. Therefore, the valves were considered to have been operable.

The previous history of the Oconee relief valves, similar relief valves at Catawba Nuclear Station, and generic safety relief valve problems pointed to a need to extend the investigation into a search for the root cause of the problem. Duke, Wyle Laboratories, and Dresser Industries representatives met between May 20 and May 22, 1991 to discuss and investigate the problem. Duke representatives toured the Wyle Laboratory facility and reviewed Wyle procedures.

The results of this review showed that the Wyle test procedure sequence is as follows: After a preliminary receipt inspection and radioactive contamination survey, the valve is mounted on a test stand. Pressurized saturated steam is supplied to the valve. Valve temperature is controlled by monitoring three thermocouples strapped to the body and bonnet flanges. Electric heaters are strapped to the outside of the valve to maintain its temperature while steam supplied to the valve inlet heats the valve internals. The valve is first leak tested with steam using a mirrored surface probe placed in the discharge stream of the valve. Condensation observed on the mirror (fogging) is evidence of leakage. Actuation setpoint tests are performed by raising the steam pressure and observing the point at which the valve relieves. If the initial actuation setpoint is not within the desired one percent tolerance band, a Notice of Anomaly is generated and sent to the valve owner. Valves that fail the actuation test are adjusted until their actuation setpoints are within the desired tolerance. Three successive actuations within the required setpoint band are necessary to accept the test as successful. A final steam leakage test is then performed.

After the successful setting of the relief valve actuation setpoint, whether or not leakage is indicated by the steam leakage test, the valve may undergo a "jack and lap" procedure. The valve is partially disassembled while maintaining spring compression and the seat surfaces are polished. Dresser Industries engineers have stated that a single performance of this procedure does not change the setpoint by more than one percent. The valve is then tested using a nitrogen pressure leak test. This test is more sensitive than the steam leak test. Duke Power requested that Wyle Laboratories begin using the nitrogen leak test in the early 1980s because the valves had developed a history of leakage. Since the nitrogen test was very difficult to pass without first performing the "jack and lap" procedure, Wyle Laboratories began using the "jack and lap" procedure routinely prior to performing the nitrogen leakage tests. If the nitrogen leak test is not successful, the valve may either be disassembled and refurbished or have another "jack and lap" procedure performed. Refurbishment or the performance of a second "jack and lap" procedure requires that the valve relief setpoint be reverified on steam, "jack and lapped", and successfully pass the nitrogen leak test. After correct setpoint adjustment and after successfully passing both steam and nitrogen leak tests, the valve is packaged and transported to Oconee Nuclear Station where it is stored until the next refueling outage.

In addition to the inspection of the Wyle procedures and the discussion between Wyle, Dresser, and Duke technical personnel, the history of the Oconee and Catawba pressurizer relief valves was investigated. Records of setpoint and leakage results were reviewed.

CONCLUSIONS

The investigations into pressurizer relief valve setpoint actuation drift did not determine a single cause for the anomalies. However, several potential causes were identified and are outlined below:

1) Valve leakage during setpoint test. A valve which is leaking during the setpoint test can have more seat surface area pressurized and will therefore relieve earlier. The compression screw would be adjusted to further compress the spring resulting in an artificially high setpoint if the leakage is subsequently repaired.

2) The "jack and lap" process. The "jack and lap" procedure introduces some error into the as-tested setpoint since it changes the condition of the valve seat after the setpoint has already been established.

3) Setpoint trending control. During the setpoint testing, control over the trending of setpoints may not have been strict enough to ensure stable valve performance.

4) Ring adjustments after a setpoint test. Changes to ring adjustments could change the valve performance if seat leakage exists.

5) Temperature effects. Valve heatup times and the number of actuations can change internal temperatures and thus valve performance. This problem is discussed in detail in NRC Information Notice 89-90, Supplement 1.

6) Spring performance. Several factors may effect spring performance: temperature, aging, complete relaxation during rebuild, or variable friction between the spring and the spring washer and spindle.

7) Seat adhesion effects. A perfectly lapped seat or seat corrosion could contribute to a high as-found setpoint.

8) Transportation/handling. Excessive valve shock or acceleration could impact valve performance.

9) Test process parameters. Steam quality, pressurization rates, valve installation effects, equipment calibration errors and personnel errors in the test process could impact the results.

The history of the pressurizer relief valves at Oconee shows that some of the highest setpoint anomalies occurred when the valve was also found to be leaking. However, this was not true in all cases of leaking valves and therefore cannot completely explain the setpoint drift phenomenon. The 7.4 percent drift in setpoint of this report is the worst case of pressurizer relief valve setpoint drift to occur at Oconee to date. No definite trend in setpoint drift over the history of Oconee can be determined. The phenomenon appears to be random. Individual valves do not show a particular tendency to drift more than other valves. It has been determined that a pressurizer relief valve was found to be outside its plus or minus one percent tolerance band following removal from the RCS 28 out of the last 40 valve tests. Therefore, this is a recurring problem.

In view of the above evidence, it appears that there are multiple factors involved in the setpoint drift problem. The root cause of this event is unknown, possible procedure deficiency in the Wyle test procedures. The bulk of the evidence suggests continued inadequacies in the testing procedures, particularly those associated with seat leakage and the "jack and lap" procedure. It is possible that the number of different variables involved prevent the valves from maintaining a repeatable actuation setpoint within the desired tolerance. Dresser Industries representatives maintain that a tolerance of plus or minus 3 percent would be a more appropriate target when comparing "as left" and "as found" setpoints. The results of the planned corrective actions, most of which are test procedure improvements, should determine the role of test procedures in setpoint error and indicate which variables contribute the greatest variance to actuation setpoint.

Oconee has never experienced an event in which RCS pressure reached the code safety relief valve setpoints.

The health and safety of the public were not jeopardized nor was personnel injury involved. The valves are Dresser model 31739A and are NPRDS reportable.

CORRECTIVE ACTIONS

Immediate

1. Spare pressurizer safety relief valves whose actuation setpoints were set within one percent of 2500 psig were installed on Unit 3 prior to startup of the unit.

Subsequent

1. Investigations into the history and the root causes of the setpoint drift were initiated.

Planned

1. The following enhancements of the Wyle Laboratories test procedures will be performed:

   a. Leaking valves will be repaired and then retested on steam. The criteria for leakage will be described more precisely.

   b. After the "jack and lap" procedure to polish the valve seat, a setpoint verification test using steam will be performed.

   c. In addition to the requirement of three successful setpoint actuations, acceptance requirements will be increased to require that actuations setpoints are not trending substantially in one direction.

   d. No ring adjustments will be allowed after the final setpoint verification test.

   e. Enhancement of temperature monitoring to assure even heating will be implemented.

   f. Torquing guidelines will be established for mounting the valve on the test stand.

   g. The test header pressurization rate will be standardized.

   h. The minimum hold time between actuations will be expanded to 10 minutes.

   i. Steam leakage criteria will be changed to allow no fogging at 93 percent of setpoint.

2. Dresser will conduct a testing program on a spring removed from Catawba valve BS-2871 to determine the effect of spring variables on setpoint drift.

3. Dresser will investigate the effect of varying friction factors at the spring/spring washer surfaces and the need to upgrade lubricants on the bearing surfaces.

4. Dresser will investigate the need to maintain the test header pressurization rate within a certain range.

5. Duke Power will obtain temperature data during power operation on the spring and body of a pressurizer code relief valve to be installed on Unit 1. This information will be provided to Wyle so that adjustments of the Wyle test temperature requirements can be made.

6. Valves will be retested after several months in storage at Wyle to determine the magnitude of setpoint repeatability without introducing the variables of transportation and handling.

SAFETY ANALYSIS

The primary safety function of the pressurizer code safety relief valves is to limit Reactor Coolant System (RCS) pressure to less than the safety limit of 2750 psig. During normal operation and most plant transients, the pressurizer code relief setpoint is not challenged because of the normal pressure control mechanisms. The pressurizer spray valve opens at approximately 2205 psig, introducing a cooler RCS water spray which quenches pressurizer steam and reduces RCS pressure. If pressure reaches approximately 2355 psig, the Reactor Protection System (RPS) [EIIS:JC] receives a trip signal which shuts down the reactor and reduces heat input to the RCS. If pressure increases to 2450 psig, the pilot operated relief valve (PORV), RC-66, will open and relieve RCS pressure. It is only during plant transients in which the normal pressure control functions are insufficient that the pressurizer code safety relief valves become necessary.

An evaluation was performed, using a computer model of plant response, which demonstrated the operability of the pressurizer code safety relief valves during the previous fuel cycle on Unit 3 (Unit 3 Cycle 12). Using conservative assumptions and the "as found" code safety relief valve setpoints, it was shown that the peak RCS pressure would be 2700 psig during the limiting Rod Bank Withdrawal Accident at Hot Zero Power.

Duke Design Engineering has performed another evaluation which analyzes the effect of both safety relief valves having a plus six percent or minus four percent setpoint drift on peak RCS pressure during certain plant transients for which normal pressure control is inadequate. This analysis was performed using the most conservative core physics assumptions for all Oconee units as of May 24, 1991. Therefore, it applies only to the currently operating fuel cycles at Oconee. Design Engineering determined that two relief valves with a setpoint drift of six percent conservatively bound an analysis of one relief valve with 7.4 percent drift and another with 2.6 percent drift. It was found that RCS pressure would not exceed 2750 psig during either a loss of feedwater or a rod withdrawal accident with the current Oconee fuel loadings and using conservative assumptions of relief valve and plant conditions. Also, a negative drift in setpoint of 4.0 percent would not result in code safety relief valves actuating prior to an RPS high pressure trip.

No radioactive releases or personnel radiation overexposures occurred as a result of this event.

In view of these analyses and the fact that the code safety relief valves have never been challenged at Oconee, it is concluded that the health and safety of the public has not been jeopardized.

Attachment-1

Pressurizer Safety Relief Valve Internals
(from Dresser Drawing 3CP-10800)