TVA

Tennessee Valley Authority, Post Office Box 2000, Soddy-Daisy, Tennessee 37379

J. L. Wilson
Vice President, Sequoyah Nuclear Plant

January 13, 1992

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

Gentlemen:

TENNESSEE VALLEY AUTHORITY - SEQUOYAH NUCLEAR PLANT UNIT 1 - DOCKET NO. 50-327 - FACILITY OPERATING LICENSE DPR-77 - LICENSEE EVENT REPORT (LER) 50-327/91027

The enclosed LER provides details concerning the manual closure of the main steam isolation valves as a result of a reactor coolant system cooldown event. This event is being reported in accordance with 10 CFR 50.73(a)(2)(iv) as a manual actuation of an engineered safety feature.

Sincerely,

JLWilson

J. L. Wilson

Enclosure
cc: See page 2

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cc (Enclosure):

INPO Records Center
Institute of Nuclear Power Operations
1100 Circle 75 Parkway, Suite 1500
Atlanta, Georgia 30339

Mr. D. E. LaBarge, Project Manager
U.S. Nuclear Regulatory Commission
One White Flint, North
11555 Rockville Pike
Rockville, Maryland 20852

NRC Resident Inspector
Sequoyah Nuclear Plant
2600 Igou Ferry Road
Soddy-Daisy, Tennessee 37379

Mr. B. A. Wilson, Project Chief
U.S. Nuclear Regulatory Commission
Region II
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323

NRC Form 366 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

Approved OMB No. 3150-0104
Expires 4/30/92

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Sequoyah Nuclear Plant, Unit 1	0 5 0 0 0 3 2 7	1 OF 0 5

TITLE (4) Manual closure of the main steam isolation valves as a result of a malfunction of a steam dump valve controller resulting in a cooldown of the reactor coolant system

EVENT DAY (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)
1 2	1 3	9 1	9 1	0 2 7	0 0	0 1	1 3	9 2		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 3

POWER LEVEL (10): 0 0 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)	XX 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)	__ 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
J. W. Proffitt	AREA CODE 6 1 5 8 4 3 - 6 6 5 1

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
X	S B	P C	F 1 8 0	Y					
B	S B	F C V	C 6 3 5	Y					

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
__ YES (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

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

On December 13, 1991, at 1459 Eastern standard time (EST), Unit 1 reactor coolant system (RCS) experienced an inadvertent cooldown event. Upon observing pressurizer level decreasing, Operations increased charging flow and manually isolated letdown in an attempt to stabilize the pressurizer level. Review of plant parameters determined that the event was due to a problem on the secondary side of the plant. Operations proceeded to manually close the main steam line isolation valves and manually closed the level control valves to limit the cooldown effect of auxiliary feedwater. In order to complete steam generator (S/G) isolation, the operators isolated S/G blowdown. After isolation of all four S/Gs, plant parameters stabilized. The root cause of this event was the intermittent failure of a steam dump valve controller resulting in the opening of two steam dump valves. The severity of the cooldown was caused by a steam dump valve failing to close after the controller malfunctioned. A work order was performed that replaced the pressure controller and controlling potentiometer.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
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TEXT (If more space is required, use additional NRC Form 366A's) (17)

I. PLANT CONDITIONS

Unit 1 was operating in Mode 3, hot standby, with a reactor coolant system (RCS) temperature of 548 degrees Fahrenheit (F), RCS pressure of approximately 1900 pounds per square inch gauge (psig), and steam generator pressure of approximately 1005 psig.

II. DESCRIPTION OF EVENT

A. Event

On December 13, 1991, at 1459 Eastern standard time, Unit 1 RCS (EIIS Code AB) experienced an inadvertent cooldown event.

Unit operators observed pressurizer (EIIS Code AB) level decreasing and increased charging flow and manually isolated letdown in an attempt to stabilize the pressurizer level. Plant parameters were reviewed in order to diagnose the event. Since containment pressure, temperature, and humidity did not indicate leakage from the RCS, and since auxiliary building radiation was not increasing, it was determined that the event was due to a problem on the secondary side of the plant. Operations proceeded to manually close the main steam line isolation valves (MSIVs) (EIIS Code SB) to limit the cooldown effect of auxiliary feedwater (AFW) (EIIS Code BA), and manually closed the level control valves (LCVs) (EIIS Code LCV). In order to complete steam generator (S/G) (EIIS Code AB) isolation, the operators isolated S/G blowdown (EIIS Code WI).

After isolation of all four S/G, plant parameters stabilized with RCS average temperature (T_{avg}) at 535 degrees F, pressurizer level at 25 percent and increasing, and S/G pressure at approximately 900 psig. The operators stabilized RCS temperature at approximately 539 degrees F by use of the S/G atmospheric relief valves.

Personnel were dispatched to the east and west valve rooms and turbine building to determine the source of steam leakage that could cause the cooldown. A bonnet leak was found on a main steam check valve in the west valve room. This leak was initially thought to be the source of the RCS cooldown. Further evaluation of plant parameters identified that this leak could not be the source because the steam header downstream of the MSIVs remained approximately 900 psig following closure of the MSIVs. Additionally, the operators identified minute leakage through a S/G safety valve. This leakage was determined to be minor because the safety valve exhaust piping was cool.

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TEXT (If more space is required, use additional NRC Form 366A's) (17)

Operations personnel had not noticed the opening of any steam dump valves (EIIS Code SB) or S/G atmospheric relief valves during the event. The records from the Technical Support Center (TSC) computer were reviewed and identified that one steam dump valve indicated opened at 1458 for three seconds, and that a second valve indicated opened at 1501 for one second. These times coincide directly with the time of the event. It was later determined that a steam dump valve did not close after opening, therefore increasing the severity of the cooldown.

B. Inoperable Structures, Components, or Systems that Contributed to the Event

None.

C. Dates and Approximate Times of Major Occurrences

1. December 13, 1991, at 1459 EST — Unit 1 experienced an inadvertent RCS cooldown event, and a notification of unusual event was declared.

2. December 13, 1991 — Operations increased charging flow and manually isolated letdown in an attempt to stabilize pressurizer level.

3. December 13, 1991, at 1502 EST — Operations manually closed the MSIVs.

4. December 13, 1991, at 1504 EST — Operations manually closed the S/G generator level control valves and isolated S/G blowdown. At this point, plant parameters were stable.

D. Other Systems or Secondary Functions Affected

None.

E. Method of Discovery

The event was discovered during routine observations of plant parameters by the main control room operations personnel.

F. Operator Actions

As a result of the event, Operations increased charging flow and manually isolated letdown in an attempt to stabilize pressurizer level. Operations proceeded to manually close the MSIVs, manually close the S/G level control valves, isolate S/G blowdown, and stabilize the plant.

G. Safety System Responses

No automatic safety system actuation conditions were reached. Manual closure of the MSIVs terminated the cooldown.

III. CAUSE OF THE EVENT

The root cause of this event was the intermittent failure of a steam dump valve controller resulting in the opening of two steam dump valves. The severity of the cooldown was caused by one of the steam dump valves failing to fully close after the controller malfunction.

The cause of the steam dump valve failure was separation of the pilot valve to the main stem resulting from poor manufacturer thread tolerances.

IV. ANALYSIS OF THE EVENT

This event is considered to be bounded by the analysis of an accidental depressurization of the main steam system as described in the Final Safety Analysis Report (FSAR). The FSAR analysis assumes the worst combination of plant parameters to demonstrate that the departure from nucleate boiling design basis is met following the inadvertent opening of a steam dump. atmospheric relief, or safety valve.

The FSAR analysis assumes end-of-life reactivity effects to provide the greatest positive reactivity insertion from the cooldown of the RCS and the greatest reduction in shutdown margin. As Unit 1 has not operated following the completion of the Cycle 5 reload, sufficient margin exists to ensure the FSAR analysis to be bounding. At the time of the event, the Unit 1 RCS was borated to approximately 2049 ppm, which was in excess of the 1363 ppm requirement for maintaining adequate shutdown margin at 200 degrees F, and the boron concentration was above the Mode 6 required 1947 ppm. RCS Tavg did not fall below 535 degrees F. Shutdown margin was maintained during the entire event.

Estimation of steam flows during the estimated two minutes prior to MSIV closure was determined to be approximately 200,000 lbm/hr. This value is less than the accident analysis assumption of one valve fully open for five minutes. Full valve capacity is approximately 800,000 lbm/hr.

Other FSAR analysis assumptions concerning AFW flowrates to the S/Gs bounded those experienced during the event.

The applicable rate-lag circuit demonstrated that the steam line pressure drop experienced (105 psig/2 min) did not exceed the drop rate allowed (2 psig/sec.) such that auto steam isolation would have actuated.

Because the actual event is bounded by the FSAR analysis, it is concluded that the transient did not pose a nuclear safety threat to plant personnel or the public.

V. CORRECTIVE ACTIONS

A work order was performed that replaced the pressure controller and controlling potentiometer. The steam dump valve that failed to closure was manually isolated. The other Unit 1 steam dump valves were externally inspected and appeared to be fully intact. SQN is continuing to monitor and evaluate the performance of the steam dump valves.

VI. ADDITIONAL INFORMATION

A. Failed Components

Intermittent failures of the steam dump controller were identified by monitoring the pressure transmitter output and the controller pressure outputs. This monitoring showed that the controller failed out the bottom, and then attempted to return to program setpoint. When the controller returned, its reset circuit drove the output high, which caused the "A" bank of steam dump valves to open.

B. Previous Similar Events

A review of previous reportable events was conducted to identify potential similar events. Three LERs were identified which were associated with anomalous steam dump system operation. In two events, mechanical failures of the valve or valve operator were noted. In the third event, the controller program for steam dump system was determined to allow excessive cooling following plant trips. Causes and corrective actions for these events are not associated with the failure of the controller.

VII. COMMITMENTS

None.

PL090204/1607