TVA

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

Robert A. Fenech
Vice President, Sequoyah Nuclear Plant

March 31, 1993

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

Gentlemen:

TENNESSEE VALLEY AUTHORITY - SEQUOYAH NUCLEAR PLANT UNIT 2 - DOCKET NO. 50-328 - FACILITY OPERATING LICENSE DPR-79 - LICENSEE EVENT REPORT (LER) 50-328/93001

The enclosed LER provides details concerning a manual reactor trip as the result of high voltage conditions on the main generator and shutdown boards. This event is being reported in accordance with 10 CFR 50.73(a)(2)(iv) as a manual actuation of the reactor protection system.

Sincerely,

Robert A. Fenech

Robert A. Fenech

Enclosure
cc: See page 2

9304060077 930331
PDR ADOCK 05000328
S PDR

IE22

cc (Enclosure):

INPO Records Center
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U.S. Nuclear Regulatory Commission
Region II
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323-0199

LER Number
455: 93-002

Title of Event: Unit 1 Service Water Pump Availability to Unit 2

Occurred: 03-11-93/ 1120
Date Time

Acceptance by Station Review:

R. Wegner , 3/29/93 J. ________ , 3/26/93
OE Date TSS Date

P. B________ , 3/25/93 ________ , ________
RAS Date OTHER Date

Approved by: G K Schwartz , 3/29/93
Station Manager Date

(L994R/VS-1)

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) Sequoyah Nuclear Plant, Unit 2

DOCKET NUMBER (2) 0|5|0|0|0|3|2|8 PAGE (3) 1 OF 0|6

TITLE (4) Extraction Steam Line Rupture Causes High Generator Output Voltage and Manual Reactor Trip

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)
										0|5|0|0|0|
0|3	0|1	9|3	9|3	0|0|1	0|0	0|3	3|1	9|3		0|5|0|0|0|

OPERATING MODE (9): 1

POWER LEVEL (10): 1|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
K. L. Meade, Compliance Licensing	AREA CODE 6|1|5 8|4|3 - 7|7|6|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 1, 1993, Unit 2 experienced a steam leak on the non-nuclear side of the plant. The No. 2 extraction line to the B2 feedwater heater ruptured, causing a 3- by 6-inch hole in the line. Steam escaping from the ruptured pipe flowed into the nearby main generator voltage regulator cubicle, resulting in regulator malfunction and voltage increasing to approximately 27 kilovolts (kV) (normal operating voltage is approximately 24 kV). As a result of the high voltage on the system and the inability to reduce the voltage, the reactor was manually tripped. The plant response during and after the manual reactor trip was consistent with responses described in the Final Safety Analysis Report. The root cause of this event was a programmatic failure of the erosion/corrosion (E/C) program resulting from insufficient management oversight and review of the program. Corrective actions include repair of affected piping and components; E/C program review and Engineering evaluation of other piping susceptible to erosion, including inspections and repairs/replacements; evaluation of impact of over-voltage condition on plant equipment; and review of other site technical programs.

I. PLANT CONDITIONS

Unit 2 was operating at approximately 100 percent rated thermal power.

II. DESCRIPTION OF EVENTS

A. Event

On March 1, 1993, at 1419 Eastern standard time (EST), Unit 2 experienced a steam leak on the non-nuclear side of the plant. The No. 2 extraction line (EIIS Code SE) to the B2 feedwater heater (EIIS Code SD) ruptured, causing a 3- by 6-inch hole in the line. Steam escaping from the ruptured pipe flowed into the nearby main generator voltage (EIIS Code TB) regulator cubicle, causing output voltage to increase to approximately 27 kilovolts (kV) (normal operating voltage is approximately 24 kV). As a result of the high voltage on the system and the inability to reduce the voltage, the reactor was manually tripped.

The afternoon of March 1, 1993, the turbine building assistant unit operator (AUO) heard a loud noise that sounded like a steam leak. The AUO determined that the leak was in the area of the No. 2 feedwater heaters on Elevation 685 of the turbine building. The steam from the leak was blowing up through the grating on Elevation 706 and around the No. 3 feedwater heaters. The steam was also very close to the voltage regulator cubicle. The AUO immediately informed the Unit 2 main control room personnel of this condition. The Unit 2 main control room personnel were involved with a main feedwater perturbation at the time of the AUO notification. The Loop 3 steam generator (S/G) level control valve (EIIS Code JB) was in manual, controlling S/G level. Main generator alarms began to annunciate in the control room almost immediately. The Operations crew started to investigate the alarms and determined that main generator voltage was high (27 kV), shutdown voltage was high (8,200 volts), and the voltage regulator had tripped. The Operations crew attempted to reduce voltage, using the base adjust controller. This attempt was unsuccessful. Based on plant conditions, the assistant shift operations supervisor (ASOS) and shift operations supervisor (SOS) decided to trip the reactor. Unit 2 was manually tripped and the emergency instructions were entered. The main generator output breakers opened 30 seconds after the reactor trip as designed, thus, ending the over-voltage condition. A review of plant data indicates that the over-voltage condition lasted approximately three and one-half minutes. The Operations crew properly transitioned through the emergency instructions and entered the plant general operating instructions. Personnel were dispatched to investigate the source of the steam leak. An ASOS informed the main control room personnel that the leak was from the No. 2 extraction steam line to the B2 feedwater heater. As a precautionary measure, the area that contained the steam leak was roped off by Security personnel to avoid personnel injury.

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

None.

C. Dates and Approximate Times of Major Occurrences

March 1, 1993 1419 EST	Through a review of plant data, the steam leak is estimated to have occurred at approximately this time. The AUO notified the Unit 2 Operations crew of the steam leak.
March 1, 1993 1420 EST	Several main generator alarms annunciated in the main control room, including exciter rectifier power loss or failure, Unit 2 exciter insulation resistance low, and generator exciter field overcurrent. The Operations crew immediately started investigating these alarms.
March 1, 1993 1421 EST	Additional main generator alarms annunciated, including generator volts per cycle high, generator voltage regulator trip, 6.9 kV shutdown 2B-B, and 2A-A board overvoltage.
March 1, 1993 1422-1423 EST	The Operations crew noted that the generator output voltage was approximately 27 kV (normal is 24 kV) and generator reactive increased to approximately 600 MVAR (normal is ± 100 MVAR). The shutdown voltage had reached approximately 8,200 V (normal is 7,000 V) as observed by an ASOS. The voltage regulator tripped to manual. Attempts to reduce main generator voltage manually were unsuccessful.
March 1, 1993 1424 EST	After evaluating plant conditions, the SOS and ASOS decided to manually trip the reactor. The Operations crew entered the emergency instructions. The main generator output breakers tripped 30 seconds after the manual reactor trip as designed, ending the over-voltage condition. The Operations crew properly transitioned through the emergency procedures and entered the general operating instructions. The unit stabilized in Mode 3. The steam leak was verified to be isolated. The area was roped off to prevent any potential personnel injury.

D. Other Systems or Secondary Functions Affected

None.

E. Method of Discovery

The steam leak on the No. 2 extraction steam line was discovered by the turbine building AUO upon investigation of a loud noise within the building. The over-voltage condition was identified by receipt of various main control room alarms and annunciators.

F. Operator Action

Operations personnel evaluated the over-voltage conditions that existed in the plant and manually tripped the reactor. The operators responded to the event in accordance with appropriate plant procedures. The plant was stabilized in hot standby.

G. Safety System Response

All safety systems responded as designed upon receipt of the manual reactor trip signal.

III. CAUSE OF EVENT

A. Immediate Cause

The immediate cause of this event was the failure of the No. 2 extraction steam line as a result of erosion-induced wall-thinning. The subsequent steam plume engulfed the voltage regulator cubicle and caused the over-voltage condition on the main generator.

B. Root Cause

The cause of this event was a programmatic failure of the erosion/corrosion program. This resulted in the failure to properly assess prior inspection data and perform requisite sample expansion. Thinning in the area of the subject steam leak had been previously identified through inspection for an adjacent maintenance activity but was incorrectly evaluated and was not documented in the corrective action program for long-term tracking and resolution. A computer program used to predict piping areas most susceptible to erosion had deficiencies as the result of modelling input errors. The cause of the program failure was insufficient management oversight and review of the program, primarily due to inadequate change management.

IV. ANALYSIS OF EVENT

The plant conditions resulting from a rupture of the secondary side extraction steam piping are bounded by the existing Final Safety Analysis Report (FSAR) analysis. The FSAR bounding conditions include accidental depressurization of the main steam system, minor secondary system pipe breaks, and major secondary system pipe ruptures. The present analyses bound a spectrum of size and number of extraction piping breaks by indicating that nuclear safety is maintained through a wide range of steam release rates.

The plant response during and after the manual reactor trip was consistent with responses described in the FSAR and accordingly, the event did not affect the health and safety of the public.

V. CORRECTIVE ACTION

A. Immediate Action

Once the reactor/turbine was tripped, the Operations crew verified that the steam source was isolated. A quick evaluation concluded that there were no injuries or visible damage to adjacent plant equipment. As a precautionary measure, the area that contained the steam leak was roped off to prevent any potential personnel injury.

The effects of the over-voltage condition are being evaluated by site Engineering personnel.

B. Action to Prevent Recurrence

An independent review of the erosion/corrosion program for adequacy and completeness is being performed, using the Electric Power Research Institute (EPRI) developed strategic plan for guidance. Based on this review, Sequoyah Nuclear Plant (SQN) will ensure that the required program attributes are defined, documented, and implemented. SQN is presently reevaluating appropriate piping systems on both Units 1 and 2. Inspections, as well as repair and replacements, will be performed based on the results of this evaluation. SQN is also utilizing other industry expertise to ensure program adequacy.

A review of other site technical programs is being performed to determine if weaknesses similar to those noted in the erosion/corrosion program exist.

SQN recognizes the need to effectively manage change as the result of past ineffective management oversight of organizational change at the site. SQN is presently applying the lessons learned to effectively manage the ongoing changes at the site.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Sequoyah Nuclear Plant, Unit 2	0 5 0 0 0 3 2 8	9 3	0 0 1	0 0	0 6 OF 0 6

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

VI ADDITIONAL INFORMATION

A. Failed Components

The piping in the No. 2 extraction line to the B2 feedwater heater ruptured as the result of erosion of the pipe wall. The main generator voltage regulator tripped as the result of steam engulfing the voltage regulator cubicle. The voltage regulator, as well as other components affected by the over-voltage condition, will be addressed by the over-voltage evaluation.

B. Previous Similar Events

A review of previous similar events identified two previous LERs involving program weaknesses and inadequate change management. LERs 50-327/91009 and 50-327/92026 described problems in the fire protection and American Society of Mechanical Engineers Section XI programs. The above described broad corrective actions are intended to prevent recurrence of these types of problems.

VII. COMMITMENTS

There are no additional commitments identified beyond those specified in TVA's letter from O. D. Kingsley to S. D. Ebneter dated March 4, 1993.