TVA

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J. L. Wilson
Vice President, Sequoyah Nuclear Plant

September 30, 1991

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) 50-327/91023

The enclosed LER provides details concerning a potential accident scenario resulting in a loss of containment sump inventory to outside containment during a small break loss of coolant accident. This event is being reported in accordance with 10 CFR 50.73(a)(2)(ii)(B) as a condition that is potentially outside the design basis of the plant.

Sincerely,

JLWilson

J. L. Wilson

Enclosure
cc: See page 2

IE22
1/1

cc (Enclosure):

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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 1

DOCKET NUMBER (2): 0|5|0|0|0|3|2|7

PAGE (3): 1 OF 0|5

TITLE (4) Potential for loss of containment sump inventory to outside containment during a small break loss of coolant accident.

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 8	3 0	9 1	9 1	0 2 3	0 0	0 9	3 0	9 1	Sequoyah, Unit 2	0 5 0 0 0 3 2 8
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 9 9

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)	__ 50.73(a)(2)(i)	__ 50.73(a)(2)(viii)(A)	
__ 20.405(a)(1)(iv)	XX 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)
Greg S. Kniedler	6 1 5 8 4 3 - 7 4 6 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

SUPPLEMENTAL REPORT EXPECTED (14)

XX YES (If yes, complete EXPECTED SUBMISSION DATE) __ NO

EXPECTED SUBMISSION DATE (15): MONTH 1 2 DAY 3 1 YEAR 9 1

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

On August 30, 1991, at 1500 Eastern daylight time (EDT) with Units 1 and 2 operating in Mode 1, it was determined that a potential accident scenario identified by a D. C. Cook Nuclear Plant Nuclear Experience Review existed at SQN for a loss of postaccident containment sump inventory outside containment. The scenario involves a high suction pressure for the centrifugal charging pumps resulting in the seal water heat exchanger relief valve diverting containment sump inventory to the volume control tank during a small break loss of coolant accident (LOCA). SQN's personnel reviewed the identified scenario, performed a similar accident scenario on the SQN simulator, and concluded that a similar condition at SQN potentially existed. The root cause of the event has not yet been determined; the investigation into the event by Westinghouse Electric Corporation and TVA is ongoing. Operations' crews were briefed on the potential event and the appropriate mitigating actions. The emergency procedures were reviewed and determined to be adequate to mitigate small break LOCA scenarios without compromising core and nuclear safety. An emergency preparedness implementing procedure was revised to include this potential small break LOCA concern with various actions to address the scenario.

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

Description of Event

On August 30, 1991, at 1500 Eastern daylight time (EDT) with Unit 1 operating in Mode 1 (99 percent power, 2,235 pounds per square inch gauge [psig], 577 degrees Fahrenheit [F]), and Unit 2 operating in Mode 1 (100 percent power, 2,235 psig, and 578 degrees F), it was determined that a potential accident scenario (identified by D. C. Cook Nuclear Plant while conducting a simulator exercise) existed at SQN for a loss of postaccident containment sump inventory outside containment. SQN became aware of this problem through the Nuclear Experience Review network. The scenario involves a high suction pressure for the centrifugal charging pumps (CCPs) (EIIS Code CB) following swapover to the sump resulting in the seal water heat exchanger relief valve (EIIS Code CB) diverting containment sump inventory to the volume control tank (VCT) (EIIS Code CB) during a small break loss of coolant accident (LOCA). SQN's personnel reviewed the D. C. Cook scenario, SQN design documents, and emergency procedures, then set-up and performed a similar accident scenario on the SQN simulator and concluded that a similar condition potentially existed at SQN.

The condition for this event is a small break LOCA that holds the reactor coolant system (RCS) (EIIS Code AB) pressure above the shutoff head of the residual heat removal (RHR) (EIIS Code BP) pumps and emergency core cooling system (ECCS) recirculation (swapover to the containment sump) is required to mitigate the event. The initial action to recover from the LOCA is to provide cold leg injection to the RCS from the refueling water storage tank (RWST); then, upon reaching the RWST (EIIS Code BQ) low level setpoint, core cooling is switched to long-term recirculation of the containment sump inventory to the RCS cold legs then hot legs. See the attached sketch for the potential flow path.

The RCS pressure (for a two-inch or smaller break size) during the containment sump recirculation phase may still be greater than RHR pump shutoff head. The RHR pumps take suction from the sump and provide suction pressure and flow directly to the CCP and safety injection pumps. The high suction pressure could cause the seal water and CCP miniflow return line check valve (62-697) to close, isolating the miniflow return line thus increasing the miniflow pressure beyond the seal water heat exchanger relief valve (62-649) setpoint. The relief valve discharges sump inventory to the VCT, which is in turn relieved to the holdup tank. The VCT and hold-up tanks are located in the auxiliary building.

NRC was notified on August 30, 1991, at 1552 EDT of this potential condition.

Cause of Event

The root cause of the event has not been determined; the investigation into the event by Westinghouse and TVA is ongoing.

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

The most likely cause is that the permanent solution chosen by TVA for the Inspection and Enforcement Bulletin 80-18 issue, "Maintenance of Adequate Minimum Flow Thru Centrifugal Charging Pumps Following Secondary Side High Energy Line Rupture," created a potential interaction that was not analyzed by TVA or Westinghouse, i.e., removal of the safety injection isolation signal from the miniflow valves. Additionally, the miniflow valves were deenergized and locked open to address a 10 CFR 50, Appendix R, interaction. It appears TVA and Westinghouse both believed that the configuration was bounded by the existing modeled injection cases.

The preliminary review by Westinghouse concluded that the findings reported by this LER may be a generic issue for Westinghouse-designed plants because of the Emergency Response Guidelines. The condition may be of potentially greater concern for ice condenser plants because of the greater likelihood the plant will be switched to the recirculation sump following a small break LOCA. This is because the lower containment spray initiation setpoint for an ice condensor containment makes it more likely containment spray will be activated resulting in more rapid depletion of the refueling water storage tank.

Analysis of Event

This event is being reported in accordance with 10 CFR 50.73(a)(2)(ii)(B) as a condition that is potentially outside the design basis of the plant pending the results of ongoing investigation efforts.

In the event of the subject scenario, the operators would have followed the existing emergency procedures and monitored plant operating conditions. Operation's licensed personnel are trained to identify intersystem LOCAs and mitigate their consequences. Operations would follow procedure guidance to mitigate the small break LOCA by initiating cooldown and depressurizing the RCS to reduce RCS pressure below that of the RHR pumps shutoff head, which eliminates the condition. In addition, manual actions could be taken to provide makeup to the RWST.

The CCP's ECCS flow capacity to provide adequate core cooling flow is expected to be unchanged. Loss of the CCP miniflow water does not adversely impact the ECCS performance or response because SQN is currently analyzed for reduced CCP injection due to miniflow loss. As described above, actions can be taken to terminate the scenario and provide additional inventory make-up.

There are no anticipated significant changes in the doses to the auxiliary building or offsite, because for a small break LOCA, there is no additional failed fuel over what is assumed in the Updated Final Safety Analysis Report (UFSAR). The small break LOCA for the condition is two inches or less. Chapter 15 of the UFSAR indicates that the peak clad temperatures for a two-inch small break LOCA is less than what is allowable in the UFSAR. This does not compromise the clad temperature to the fuel and thus does not result in any fuel damage.

Corrective Actions

Operations' crews were briefed on the potential event and appropriate mitigating actions. The emergency procedures were reviewed and determined to be adequate to mitigate small break LOCA scenarios without compromising core and nuclear safety. However, Emergency Preparedness Implementing Procedure 6, "Activation and Operation of the Technical Support Center," was revised to identify this potential small break LOCA concern and actions to address the scenario.

The investigation into this event is ongoing by Westinghouse and TVA. The ongoing investigation will determine the cause of the event and appropriate long-term corrective actions.

Additional Information

No similar reportable events were identified.

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NRC Form 366A
(6-89)

U.S. NUCLEAR REGULATORY COMMISSION

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LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

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