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November 10, 2000
RC-00-0352

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

Gentlemen:

Subject: VIRGIL C. SUMMER NUCLEAR STATION
DOCKET NO. 50-395
OPERATING LICENSE NO. NPF-12
LICENSEE EVENT REPORT (LER 2000-008-00)
REACTOR COOLANT SYSTEM PRESSURE BOUNDARY DEGRADATION

Stephen A. Byrne
Vice President
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Attached is Licensee Event Report (LER) No. 2000-008-00, for the Virgil C. Summer Nuclear Station (VCSNS). The report describes conditions that resulted in VCSNS being in an unanalyzed plant condition and outside the requirements of the facility Technical Specifications. Initial notification was made in accordance with 10 CFR 50.72(b)(2)(i).

Should you have any questions, please call Mr. Mel Browne at (803) 345-4141.

Very truly yours,

BCW Williams for

Stephen A. Byrne

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NRC FORM 366 (6-1998) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 06/30/2001
LICENSEE EVENT REPORT (LER)	Estimated burden per response to comply with this mandatory information collection request: 50.0 hrs. Reported lessons learned are incorporated into the licensing process and fed back to industry. Forward comments regarding burden estimate to the Records Management Branch (T-6 F33), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and to the Paperwork Reduction Project (3150-0104), Office of Management and Budget, Washington, DC 20503. If an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

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TITLE

Reactor Coolant System Pressure Boundary Degradation

EVENT DATE			LER NUMBER			REPORT DATE			OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
10	12	2000	2000	-- 008 --	00	11	10	00	FACILITY NAME	DOCKET NUMBER

OPERATING MODE	6	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more)							
			20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(viii)
POWER LEVEL	0		20.2203(a)(1)		20.2203(a)(3)(i)	X	50.73(a)(2)(ii)		50.73(a)(2)(x)
			20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
			20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)		OTHER
			20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC FORM 366A
			20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
M. N. Browne Manager, Nuclear Licensing & Operating Experience	(803) 345-4141

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
B	RC			Y					

SUPPLEMENTAL REPORT EXPECTED			EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
X	YES (If yes, complete EXPECTED SUBMISSION DATE).	NO		02	15	2001

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)

On 10/7/00 plant personnel identified an accumulation of boric acid near the "A" loop of the reactor vessel on the 412 feet elevation of the reactor building.

On 10/12/00, at 0630 hours, visual inspection revealed small amounts of boron buildup on the weld between the vessel nozzle and the hot leg pipe. Within hours, the suspect area was cleaned and a dye penetrant (PT) examination of the pipe identified a 4 inch indication at the weld between the hot leg piping and the reactor vessel nozzle. The weld is located approximately 3 feet from the vessel in pipe near the nozzle. The indication was located about 17" from the top of the pipe. This pipe has a nominal inside diameter of 29 inches and is approximately 2.5 inches thick.

Subsequent ultrasonic examination from the inside diameter identified an axial flaw less than 3 inches long. The same examination determined that the original indication was not the source of the leak.

The issue resolution strategy consists of a root cause evaluation, repair of the weld, and restart justification. A group of industry experts in the areas of root cause, failure analysis, materials, welding, inspection, and fracture mechanics has been assembled to assist in the recovery effort. The NRC has an investigative team on site to monitor activities associated with the resolution of this event.

A supplemental report will be issued within 30 days of restart from this event providing detailed information on the cause and corrective actions. Restart is tentatively scheduled for early January, 2001.

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TEXT *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

PLANT IDENTIFICATION

Westinghouse - Pressurized Water Reactor

EQUIPMENT IDENTIFICATION

Reactor Coolant System

EIIS Code AB

IDENTIFICATION OF EVENT

On 10/7/00 plant personnel identified an accumulation of boric acid near the "A" loop of the reactor vessel on the 412 feet elevation of the reactor building.

On 10/12/00, at 0630 hours, visual inspection revealed small amounts of boron buildup on the weld between the vessel nozzle and the hot leg pipe. Within hours, the suspect area was cleaned and a dye penetrant (PT) examination of the pipe identified a 4 inch indication at the weld between the hot leg piping and the reactor vessel nozzle. The weld is located approximately 3 feet from the vessel in pipe near the nozzle. The indication was located about 17" from the top of the pipe. This pipe has a nominal inside diameter of 29 inches and is approximately 2.5 inches thick.

EVENT DATE

October 12, 2000

REPORT DATE

November 10, 2000

The event is documented in the VCSNS Corrective Action Program under Condition Evaluation Reports CER 00-1392, CER 00-1324, and CER 00-1396.

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CONDITIONS PRIOR TO EVENT

Mode 6 – Refueling (0% - RCS Depressurized)

DESCRIPTION OF EVENT

An accumulation of boric acid was identified around the bottom half of the "A" reactor coolant loop (RCL) hot leg air boot during a routine outage inspection. This inspection was performed following a plant shutdown for refueling outage 12 (RF-12) which began on October 7, 2000. Further inspection estimated a total accumulation of approximately 100 to 200 pounds of boric acid near the "A" hot leg area of the reactor vessel and RCL piping. Some insulation and boric acid were removed from the area of the suspected leak path to allow for further inspection.

On October 12, 2000, plant personnel visually identified a potential leak area on the nozzle to pipe connection of the "A" loop hot leg. A visual inspection revealed boron on the weld between the vessel nozzle and the hot leg piping. Based on this preliminary information, it was suspected that some leakage had occurred through the pressure boundary at this weld.

On October 12, 2000, a cleanup and partial/preliminary dye penetrant test of the weld on the "A" RCL Hot Leg was completed. The dye penetrant test was not implemented by a qualified procedure, but identified a 4" circumferential indication in the weld between the hot leg piping and the reactor vessel nozzle. This weld is located approximately 3 feet from the reactor vessel inside diameter, and is accessible from the inspection port at the reactor vessel flange area. The reported indication is located at approximately 270° to 285° when viewed toward the reactor vessel.

On November 8, 2000, a preliminary report of the inside diameter ultrasonic, eddy current, and remote visual examinations identified the flaw as axially oriented and less than 3 inches in length, with evidence of through-wall extension. Results from all other ultrasonic examinations in the remaining nozzles showed no recordable indications

CAUSE OF EVENT

SCE&G has not determined a cause for this event. A detailed equipment failure analysis, and root cause evaluation are in progress. Issue resolution strategy consists of an equipment failure analysis, repair of the weld, restart safety evaluation and final root cause evaluation. The root cause will consist of:

1. A metallurgical failure analysis utilizing hot cell laboratory examinations,
2. Examination of potential failure modes and extent of condition,
3. An evidence matrix to support/refute findings obtained from the metallurgical failure analysis and other failure analysis aspects of the root cause assessment, and
4. The final root cause determination.

The final root cause determination will be presented in the supplemental report.

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ANALYSIS OF EVENT

A detailed fracture mechanics evaluation is being performed as part of the safety assessment. Two specific calculations are being performed for the fracture mechanics evaluation: 1) critical flaw size and 2) leak rate. The flaw size calculation is being performed for both stainless steel, and Alloy 182. The leak rate calculation is being performed using a calculation procedure typically used for leak-before-break calculations.

Preliminary calculations were performed assuming two cases; a circumferential or an axial flaw. Conclusions that may be made from the fracture mechanics evaluation are:

1. A very large through-wall crack would be required to cause a failure of the piping.
2. The plant was operating in a safe condition, even after the leakage occurred.

CORRECTIVE ACTIONS

The issue resolution strategy contains three parts:

1) A root cause evaluation,
2) Repair of the weld, and
3) Restart justification.

The root cause evaluation is utilizing a rigorous and multidiscipline approach, and will consist of:

1. A metallurgical failure analysis utilizing hot cell laboratory examinations,
2. Examination of potential failure modes and extent of condition,
3. An evidence matrix to support/refute findings obtained from the metallurgical failure analysis and other failure analysis aspects of the root cause assessment, and
4. The final root cause determination.

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CORRECTIVE ACTIONS (Cont'd)

The restart justification will consist of:

1. A licensing basis review and acceptance of the repair,
2. Design basis acceptance of the repair,
3. Continued monitoring of effectiveness of the repair and failure analysis,
4. An operating experience review, and
5. A safety assessment.

The safety assessment will include a design stress report demonstrating that the pipe repair is in compliance with ASME Section XI requirements, completion of the root cause evaluation, and a safety evaluation in accordance with 10 CFR 50.59.

Regulatory Interface is on-going during this issue and has consisted of:

Conference calls were held with the NRC on October 13, 2000 and October 20, 2000.

A presentation to the NRC was made at the Region II office in Atlanta, GA on October 25, 2000.

The NRC has provided two sets of questions on October 23, 2000 and November 3, 2000 which SCE&G is currently addressing.

An on-site NRC Special Inspection Team (SIT)

Daily updates to site resident inspectors.

Responses to the NRC questions will be developed at the completion of the root cause assessment. Some of the questions are of a generic nature and may require industry responses.

A group of industry experts in the areas of root cause, failure analysis, materials, welding, inspection, and fracture mechanics from EPRI, Florida Power Corporation, Westinghouse, WesDyne, DESI, Framatome Technologies, and several others has been assembled to assist in the recovery effort.

A third party review will be performed to ensure that the issue resolution strategy is comprehensive, technically adequate, and in compliance with the regulations.

Upon completion of the above actions, a supplemental to this report shall be submitted. Projected completion of all actions is early January, 2001.

PRIOR OCCURENCES

None