Stephen A. Byrne
Vice President, Nuclear Operations
803.345.4622

SCE&G®
A SCANA COMPANY

March 15, 2001
RC-01-0061

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 2000-008-01)
REACTOR COOLANT SYSTEM PRESSURE BOUNDARY
DEGRADATION

Attached is supplemental Licensee Event Report (LER) No. 2000-008-01, 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. This issue is reported in accordance with 10 CFR 50.73(a)(2)(ii)(A).

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

Very truly yours,

Stephen A. Byrne

JWT/SAB
Attachment

c: N. O. Lorick
N. S. Carns
T. G. Eppink (w/o attachment)
R. J. White
L. A. Reyes
K. R. Cotton
NRC Resident Inspector
H. C. Fields, Jr.
D. M. Deardorff
Paulette Ledbetter
D. L. Abstance
EPIX Coordinator
K. W. Sutton
INPO Records Center
J&H Marsh & McLennan
NSRC
RTS (O-C-00-1396)
RTS (O-L-00-0224)
File (818.07)
DMS (RC-01-0061)

IE22

NRC FORM 366 (1-2001)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 6-30-2001
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)		Estimated burden per response to comply with this mandatory information collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to bjs1@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose 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.

FACILITY NAME	DOCKET NUMBER	PAGE
Virgil C. Summer Nuclear Station	05000395	1 OF 6

TITLE
Reactor Coolant System Pressure Boundary Degradation

EVENT DATE			LER NUMBER			REPORT DATE			OTHER FACILITIES INVOLVED	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000395
10	12	2000	2000	- 008 -	01	03	15	01	FACILITY NAME	DOCKET NUMBER

OPERATING MODE	6
POWER LEVEL	0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)

	20.2201(b)		20.2203(a)(3)(ii)		50.73(a)(2)(ii)(B)		50.73(a)(2)(ix)(A)
	20.2201(d)		20.2203(a)(4)		50.73(a)(2)(iii)		50.73(a)(2)(x)
	20.2203(a)(1)		50.36(c)(1)(i)(A)		50.73(a)(2)(iv)(A)		73.71(a)(4)
	20.2203(a)(2)(i)		50.36(c)(1)(ii)(A)		50.73(a)(2)(v)(A)		73.71(a)(5)
	20.2203(a)(2)(ii)		50.36(c)(2)		50.73(a)(2)(v)(B)		OTHER Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iii)		50.46(a)(3)(ii)		50.73(a)(2)(v)(C)		
	20.2203(a)(2)(iv)		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(D)		
	20.2203(a)(2)(v)		50.73(a)(2)(i)(B)		50.73(a)(2)(vii)		
	20.2203(a)(2)(vi)		50.73(a)(2)(i)(C)		50.73(a)(2)(viii)(A)		
	20.2203(a)(3)(i)	X	50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(B)		

LICENSEE CONTACT FOR THIS LER

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

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
B	RC			Y					

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

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) **(16)**

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

Subsequent inspections 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 approximately 3 feet from the vessel between the hot leg piping and the reactor vessel nozzle. The indication was located about 17 inches 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 PT indications were later determined to be steam cutting/boric acid corrosion at the nozzle butter to nozzle interface.

The axial flaw was determined to have resulted from the extensive repairs performed during initial installation which created high welding residual stresses in the material, combined with a material susceptible to stress corrosion cracking, and an environment known to cause primary water stress corrosion cracking. The welding technology of the codes, standards and processes in use during initial installation did not account for the extent of repairs required on the weld.

Weld repairs were completed on February 9, 2001. All safety issues are satisfied. Several long term corrective actions are being tracked in the plant Corrective Action Program. Any additional actions will be addressed through industry response to EPRI and NRC developed initiatives.

FACILITY NAME	DOCKET	LER NUMBER			PAGE
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
V.C.Summer Nuclear Station	05000395	2000	-- 008 --	01	2 OF 6

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

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 foot 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 **approximately 3 feet from the vessel** between the hot leg piping and the reactor vessel nozzle. The indication was located about 17 **inches** from the top of the pipe. This pipe has a nominal inside diameter **(ID)** of 29 inches and is approximately 2.5 inches thick.

EVENT DATE

October 12, 2000

REPORT DATE

November 10, 2000
Supplemental: March 15, 2001

The event is documented in the VCSNS Corrective Action Program under Condition Evaluation Reports CER 00-1392, CER 00-1324, CER 00-1396, **CER 00-1676, and CER 00-1821.**

CONDITIONS PRIOR TO EVENT

Mode 6 (0% -RCS Depressurized)

FACILITY NAME	DOCKET	LER NUMBER			PAGE
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
V.C.Summer Nuclear Station	05000395	2000	-- 008 --	01	3 OF 6

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

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 **an** 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 **informational** dye penetrant test of the weld on the "A" RCL Hot Leg was completed. The dye penetrant test identified a 4 inch 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 **was** located at approximately 270 degrees to 285 degrees when viewed toward the reactor vessel. **This indication was later determined to be steam cutting/boric acid corrosion at the nozzle butter to nozzle interface.**

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 addressed the cause of this event in Root Cause Report C-00-1392. A team of industry experts was assembled and, based on a thorough review of available records and documents and on the results of the metographic examinations, the following root causes and contributing factors were identified:

1. **Extensive repairs on the VCSNS reactor vessel "A" hot leg nozzle to pipe weld created high welding residual stresses in a material (Alloy 182/Alloy 82) exposed to an environment known to cause primary water stress corrosion cracking (PWSCC).**
2. **Neither the codes, standards nor the welding process recognized or required consideration of the cumulative effect of multiple repair welding and weld grinding in the creation of high residual stresses.**

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

CAUSE OF EVENT (Cont'd)

Contributing Factors

1. **The possibility of hot cracking may have exacerbated the flaw growth.**
2. **Nondestructive Examination (NDE) detection of the flaw was not conclusive.**

The effect of surface contour, surface roughness, and detector physical parameters were not adequately considered in the qualification of the NDE process for this type of flaw.

The pre-service and 1987 in-service inspections both included UT performed with multiple angles. The 1993 in-service inspection was performed using only a 70 degree angle transducer that investigated only the inner 1/3 volume of the weld.

3. **The dissimilar metal weld was fabricated in the field rather than the manufacturer's shop.**
4. **Automatic welding was initially qualified at VCS for use on these welds using stainless steel materials instead of Alloy 82.**
5. **There is no code guidance on the extent of repairs made on welds using these materials.**

ANALYSIS OF EVENT

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

Preliminary calculations were performed assuming two cases: a circumferential **flaw and** 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.

During initial construction, the V. C. Summer reactor vessel nozzles which are made from low alloy steel, were "buttered" using a nickel-based material, Alloy 182, at the manufacturer. The buttering was stress relieved with the rest of the vessel and then the nozzle weld preparation (a "J" grove to be welded entirely from the outside surface) was machined.

At the V. C. Summer site during plant construction, the stainless steel piping was welded to the nozzle weld using nickel-based material, Alloy 82 or Alloy 182. During field welding for the "A" hot leg, numerous indications were discovered in the weld that required significant repair. The weld in the "A" hot leg nozzle essentially became a "double-V" design because welding and grinding were performed from both the inside and outside surfaces. The weld was inspected (radiographic, ultrasonic, liquid penetrant, and visual) during preservice inspection (PSI) and was ultrasonically inspected in 1987 and 1993 for inservice inspection (ISI). No surface connected flaws were discovered in these inspections.

FACILITY NAME	DOCKET	LER NUMBER			PAGE
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
V.C.Summer Nuclear Station	05000395	2000	-- 008 --	01	5 OF 6

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

ANALYSIS OF EVENT (Cont'd)

On October 7, 2000, after completing cycle 12, V. C. Summer discovered boron deposits in the vicinity of the reactor vessel "A" hot leg nozzle. Unidentified RCS leakage during the cycle had been in the range of approximately 0.3 gallons per minute (GPM) with identified and unidentified leakage well within the Technical Specifications. It was subsequently confirmed that the boron deposits originated from a through-wall crack in the reactor vessel nozzle to piping weld. Isotropic and tritium analysis estimates the start of the through-wall leaking in August 1999, several months after the start of cycle 12.

Because of the nature of this event, significant inspection, evaluation, root cause, and destructive examination of the weld crack was required. Outside diameter (OD) dye penetrant examinations, outside and inside visual examinations, and inside diameter ultrasonic (UT) and eddy current (ECT) inspections were performed. A twelve-inch segment (spool piece) from the reactor coolant system piping containing the nozzle to pipe weld joint was removed and shipped to Westinghouse for non-destructive and destructive examination.

The metallurgical examination confirmed the presence of an axial crack located 7 degrees clockwise from the top of the pipe (as viewed from the centerline of the reactor vessel). Metallographic examinations suggest multiple crack initiation sites on the ID. The crack extends approximately 2.5 inches along the inside surface and is intersected by a shorter (about 2.0 inch long) circumferential crack. The axial crack was contained on the pipe side by the heat affected zone of the stainless steel pipe and by the low alloy steel reactor vessel nozzle on the opposite end. The axial crack, therefore, was entirely within the Alloy 82 weld metal and the flaw reached the pipe OD surface as a single small weep hole.

The circumferential crack intersecting the axial crack at the 7 degrees location was contained within the Alloy 182 cladding/buttering under the carbon steel nozzle. Crack depth was about 0.2 inches and resulted in some minor corrosion pitting at the interface of the low alloy steel. One end of the circumferential crack was turned toward the axial direction. Results of metallurgical analysis indicate that both the axial and circumferential cracks followed micro-fissuring and interdendritic morphology.

Several indications were identified in the "A" hot leg as well as the other legs using eddy current techniques. These indications were not initially identified as such by ultrasonic inspections. Several of the eddy current indications present on the spool piece were investigated using non-destructive and destructive techniques. The largest of the indications was confirmed to be approximately 0.750 inch long on the ID surface by 0.615 inch deep. A post mortem review of the UT results revealed that there might have been an indication present at that location. It was postulated in the final In-service Inspection Report that an inside surface pipe counter bore at that location caused the UT transducer to be unable to see the indication initially. All examined cracks were similar, described as having an interdendritic morphology.

CORRECTIVE ACTIONS

The root cause evaluation utilized a rigorous and multidiscipline approach, and **consisted** of:

1. A metallurgical failure analysis utilizing hot cell laboratory examinations,
2. Examination of potential failure modes and extent of condition,

FACILITY NAME	DOCKET	LER NUMBER			PAGE
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
V.C.Summer Nuclear Station	05000395	2000	-- 008 --	01	6 OF 6

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

CORRECTIVE ACTIONS (Cont'd)

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 weld repair sequence consisted of:

1. **Removal of a spool piece containing entire weld,**
2. **Preparation and welding of a new stainless steel spool piece utilizing Inconel 52/152, a material with superior corrosion resistance as compared to Inconel 82/182,**
3. **Use of a narrow groove welding technique to minimize weld shrinkage and high residual tensile stresses**

The restart justification **consists** 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 **includes** a design stress review 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.

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

Key long term actions for VCSNS resulting from this issue:

1. **Develop enhancements to the VCSNS Boric Acid Inspection Program.**
2. **Determine enhancements to the leak detection capability.**
3. **Perform NDE on Hot Legs of Loop B and C at the corresponding weld location during Refueling Outage 13.**
4. **Perform inspection of all reactor vessel nozzles during Refueling Outage 14.**

PRIOR OCCURRENCES

None