SCE&G®
A SCANA COMPANY

November 7, 2000
RC-00-0349

Stephen A. Byrne
*Vice President*
*Nuclear Operations*
803.345.4622

South Carolina Electric & Gas Co.
Virgil C. Summer Nuclear Station
P. O. Box 88
Jenkinsville, South Carolina
29065

803.345.5209
803.635.1461

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 2000-007-00)
MAIN STEAM SYSTEM SUPPORT FOUND MISSING

Attached is Licensee Event Report (LER) No. 2000-007-00, for the Virgil C. Summer Nuclear Station (VCSNS). The report describes conditions that resulted in VCSNS being in a condition outside the design basis of the facility.

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

Very truly yours,

B C Williams for

Stephen A. Byrne

AJC/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
J. B. Knotts, Jr.
INPO Records Center
J&H Marsh & McLennan
NSRC
RTS (O-C-00-1359)
File (818.07)
DMS (RC-00-0349)

 IE22

NRC FORM 366 (6-1998) U.S. NUCLEAR REGULATORY COMMISSION

# LICENSEE EVENT REPORT (LER)

**APPROVED BY OMB NO. 3150-0104 EXPIRES 06/30/2001**



| FACILITY NAME | DOCKET NUMBER | PAGE |
|---|---|---|
| Virgil C. Summer Nuclear Station | 05000395 | 1 of 4 |

**TITLE**

Main Steam System Support Found Missing

| 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 | 9 | 2000 | 2000 -- | 007 -- | 00 | 11 | 8 | 00 | FACILITY NAME | DOCKET NUMBER |

| OPERATING MODE | 5 |
|---|---|
| POWER LEVEL | 0 |

**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) |
| | 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<br>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 | MS | SPT | | YES | | | | | |
| | | | | | | | | | |

| 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)

On 8/15/00, during a reanalysis of the Reactor Coolant System and Main Steam System inside the Reactor Building, plant engineering personnel identified a concern that a discrepancy existed between the pipe analysis and the documentation of a tie-back support for the 3/4" vent to XVT12803/12804. The analysis performed for snubber reduction under Modification Request Form 22494B takes credit for the existence of the tie-back support on the "B" Main Steam header. Condition Evaluation Report (CER-O-C-00-1019) was written to document this apparent discrepancy. Because the area was inaccessible, a visual inspection of the main steam line was planned to be performed during the upcoming refueling outage (RF) 12 to determine whether or not the tie-back support existed.

On 10/9/00, at 2045 hours, plant personnel visually verified that the tie-back support did not exist on the "B" main steam line in the Reactor Building. Condition Evaluation Report (CER-O-C-00-1359) was written to document the reportability of this condition. Evaluation of the condition revealed that the plant was outside the design basis. Immediate notification was not required per 10 CFR 50.72 since the condition was found while the plant was shutdown, but a 30 day written report was required per 10 CFR 50.73(a)(2)(ii)(B).

The cause of this event was personnel error.

To correct this condition, a tie-back support for the 3/4" vent to XVT12803/12804 will be installed prior to restart from RF 12.

| FACILITY NAME | DOCKET NUMBER | LER NUMBER | | | PAGE |
|---|---|---|---|---|---|
| | | YEAR | SEQUENTIAL NUMBER | REVISION NUMBER | |
| V.C. Summer Nuclear Station | 05000395 | 2000 | 007 | 00 | 2 of 4 |

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

## PLANT IDENTIFICATION

Westinghouse - Pressurized Water Reactor

## EQUIPMENT IDENTIFICATION

MAIN STEAM SYSTEM

EIIS Code SB

## IDENTIFICATION OF EVENT

On 8/15/00, during a reanalysis of the Reactor Coolant System and Main Steam System inside the Reactor Building, plant engineering personnel identified a concern that there was no evidence that a tie-back support existed for the 3/4" vent to XVT12803/12804. Condition Evaluation Report (CER-O-C-00-1019) was written to document this concern. Because the area was inaccessible, a visual inspection of the Main Steam line was planned to be performed during the upcoming refueling outage (RF) 12 to determine whether or not the tie-back support existed.

On 10/9/00, at 2045 hours, while in Mode 5, plant personnel visually verified that the tie-back support did not exist on the main steam line vent in the Reactor Building. Condition Evaluation Report (CER-O-C-00-1359) was written to document the reportability of this condition. Evaluation of the condition revealed that the plant was outside the design basis. Immediate notification was not required per 10 CFR 50.72 since the condition was found while the plant was shutdown, but a 30 day written report was required per 10 CFR 50.73(a)(2)(ii)(B).

CER 00-1019 and NCN 00-1019

## EVENT DATE

October 9, 2000

## REPORT DATE

November 8, 2000

The event is documented in the VCSNS Corrective Action Program under Condition Evaluation Reports CER 00-1019 and CER 00-1359.

## CONDITIONS PRIOR TO EVENT

Mode 5 – Refueling (0%)

| FACILITY NAME | DOCKET NUMBER | LER NUMBER | | | PAGE |
|---|---|---|---|---|---|
| | | YEAR | SEQUENTIAL NUMBER | REVISION NUMBER | |
| V.C. Summer Nuclear Station | 05000395 | 2000 | 007 | 00 | 3 of 4 |

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

## DESCRIPTION OF EVENT

On 8/15/00, during a reanalysis of the Reactor Coolant System and Main Steam System inside the Reactor Building, plant engineering personnel identified a concern that a discrepancy existed between the pipe analysis and the documentation of a tie-back support for the 3/4" vent to XVT12803/12804. Condition Evaluation Report (CER-O-C-00-1019) was written to document this apparent discrepancy. Because the area was inaccessible, corrective action was generated to perform a visual inspection of the main steam line during the upcoming refueling outage (RF) 12 to determine whether or not the tie-back support existed.

On 10/9/00, at 2045 hours, while in Mode 5, plant personnel visually verified that the tie-back support did not exist on the main steam line in the Reactor Building. Condition Evaluation Report (CER-O-C-00-1359) was written to document the reportability of this condition. Evaluation of the condition revealed that the plant was outside the design basis. Immediate notification was not required per 10 CFR 50.72 since the plant was shutdown, but a 30 day written report was required per 10 CFR 50.73(a)(2)(ii)(B).

## CAUSE OF EVENT

The cause of the tie-back support not being installed on "B" main steam line vent was personnel error. The supports on main steam lines "A" & "C" were added at a time when there was significant construction activity being performed in the reactor building. These supports were commonly installed from a typical design and as-built through the Field Construction Request (FCR) program. Physical walkdowns of the Main Steam lines verified the supports were installed on the "A" and "C" main steam line vents.

## ANALYSIS OF EVENT

The original plant analysis qualified the vent on the "B" main steam header without a tie-back support. The reanalysis of this line for snubber reduction and Steam Generator (S/G) replacement assumed that a tie-back support was present. The second reanalysis of this line for S/G hydraulic snubber removal also assumed that a tie-back support was present. It was during the recent third reanalysis of this line for S/G center of gravity discrepancies that this was questioned. A drawing search was unproductive in ascertaining the existence of the tie-back support.

An analysis was performed to determine if the unsupported vent could be qualified for an upset (seismic) event. The vent could not be qualified for seismic conditions. The calculated stresses were ten times greater than the ASME Code limits for the upset condition and 3.5 times greater than GL 91-18 operability limits for the upset condition. Assuming that a tie-back support did not exist, an evaluation was performed of the consequences to plant operation for a loss of the 3/4" vent line from the main steam headers during an upset (seismic) event. The evaluation concluded that this event is bounded by the following: the analysis for core damage as described in FSAR Chapter 15 for a stuck open secondary side safety relief valve (SRV) or power operated relief valve (PORV), and the analysis for reactor building pressure and environmental effects during a loss of coolant accident (LOCA) as described in FSAR Chapter 6. A probabilistic risk assessment was performed assuming that either the SRV or PORV stuck open. This resulted in no change to the core damage frequency, since the plant is designed for a double ended guillotine break of the main steam line. However, not having the tie-back support installed placed the plant outside the design basis.

## CORRECTIVE ACTIONS

The tie-back support for the 3/4" vent to XVT12803/12804 will be installed prior to plant restart from RF 12.

Condition Evaluation Report CER-00-1359 was initiated on discovery and evaluated this condition.

**LICENSEE EVENT REPORT (LER)**
TEXT CONTINUATION

| FACILITY NAME | DOCKET NUMBER | LER NUMBER | | | PAGE |
|---|---|---|---|---|---|
| | | YEAR | SEQUENTIAL NUMBER | REVISION NUMBER | |
| V.C. Summer Nuclear Station | 05000395 | 2000 | 007 | 00 | 4 of 4 |

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

## PRIOR OCCURRENCES

None