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March 31, 1998
RC-98-0065

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 97-004)
REVISION 1

Attached is Revision 1 to Licensee Event Report (LER) No. 97-004 for the Virgil C. Summer Nuclear Station. This report is submitted pursuant to the requirements of 10 CFR 50.73(a)(2)(i)(B) and 10 CFR 50.73(a)(2)(ii)(B).

Should you have any questions, please call Mr. Charles J. McKinney at (803) 345-4723.

Very truly yours,

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NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Virgil C. Summer Nuclear Station	05000395	1 OF 7

TITLE (4)

Technical Specification Noncompliance and Piping Analysis Exceeds Code Allowables Due to Snubber Failures

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
										05000
10	12	97	97	-- 004 --	1	11	11	97	FACILITY NAME	DOCKET NUMBER
										05000

OPERATING MODE (9)	6	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)					
			20.2201(b)	20.2203(a)(2)(v)	X	50.73(a)(2)(i)	50.73(a)(2)(viii)
POWER LEVEL (10)	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
			20.2203(a)(2)(iv)	50.36(c)(2)		50.73(a)(2)(vii)	or in NRC FORM 366A

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
A. R. Rice, Manager, Nuclear Licensing & Operating Experience	(803) 345-4232

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
X	AB	SNB	P029	Y	X	BJ	SNB	P029	Y
X	WI	SNB	P029	N					

SUPPLEMENTAL REPORT EXPECTED (14)				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 October 12 and 15, 1997, plant personnel found two (2) snubbers which were in the locked position. These components are mechanical shock arrestors attached to the Reactor Coolant pressurizer spray line and Nuclear Blowdown piping respectively. The inoperable snubbers were found and replaced during refueling outage (RF) surveillance testing, performed in accordance with Technical Specification (TS) 3/4.7.7, Snubbers.

Preliminary engineering analysis of potential impacts to the associated piping concluded that the stress on the piping had exceeded ASME Code allowable limits and was therefore, outside the design basis. System walkdowns did not find any transient induced damage and ultrasonic inspection of welds on the pressurizer spray line failed to find any indications of stress/service induced flaws. Due to the incompatibility of the Nuclear Blowdown line with standard NDE techniques the line was replaced to return the pipe to its design basis condition. Westinghouse completed an ASME code fatigue qualification on the pressurizer spray line on March 17, 1998 with the conclusion that the calculated usage factors for 40 years remains less than 1 and no restrictions to plant transients are required. Both of these systems are considered to be operable at this time. Exceeding the design basis for these piping systems is reportable per the requirements of 10 CFR 50.73(a)(2)(ii)(B).

On October 18, engineering review of three (3) snubber failures on a feedwater line determined that a transient on December 5, 1994 had caused the snubber damage. System walkdowns during this outage (RF 10) and at the time of the transient failed to find any apparent damage; however, due to personnel oversight, freedom of motion tests of suspect snubbers were not verified during RF 9 (spring of 1996) per the requirements of TS Surveillance 4.7.7.c. This TS noncompliance is being reported per the requirements of 10 CFR 50.73(a)(2)(i)(B). Station Administrative Procedure 1122 will be revised by April 1, 1998 to ensure that engineering initiates component evaluations required by TS 4.7.7.

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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:

MK-RCH-0328, EIIS--AB
MK-BDH-4007, EIIS--WI
MK-FWH-0126, EIIS--BJ
MK-FWH-0128, EIIS--BJ
MK-FWH-0353, EIIS--BJ

IDENTIFICATION OF EVENT:

Engineering evaluation of potential impacts to associated plant piping from inoperable snubbers, found during surveillance testing, identified two (2) sections of piping which exceeded ASME Code allowable limits and a failure to perform a required Technical Specification surveillance following a plant transient to a section of feedwater piping on December 5, 1994.

EVENT DATE: October 12, 1997

REPORT DATE: November 11, 1997

This report was initiated by Non-Conformance Notices 97-0938, 97-1016, & 97-1052.

CONDITIONS PRIOR TO THE EVENT: Mode 6, 0% Power

DESCRIPTION OF EVENT:

The Virgil C. Summer Nuclear Station (VCSNS) has been conducting two (2) snubber related programs concurrently. The programs include a long term snubber reduction program, which either removed snubbers entirely from plant

DESCRIPTION OF EVENT (cont.):

systems or replaced them with rigid struts, and a Technical Specification (TS) 3/4.7.7 18-month snubber surveillance testing program. All snubbers are mechanical snubbers that provide protective restraint to a piping system during a dynamic event. When performing properly they present no significant load to the system and allow essentially free motion during thermal movement of the piping during normal plant operation. Locked in place conditions impact a snubbers ability to allow the expected thermal movement of the piping and induces stresses in the piping and supports that may impact structural reliability and service life.

Individual snubber failures found during the Refueling Outage No. 10 snubber testing program and addressed in this report are as follows:

1) On October 5 plant personnel found snubber MK-FWH-0353 to have excessive drag force in both the tension and compression mode of operation. This snubber is a Pacific-Scientific Model PSA-10 mechanical shock arrestor, attached to the "B" loop feedwater piping, which was being replaced with a rigid strut under the snubber reduction program. Subsequent testing of other snubbers attached to this system found two (2) additional PSA-10 snubbers (MK-FWH-0126 and MK-FWH-0128) which were inoperable. MK-FWH-0126 was found to be locked in position.

   During the engineering review on October 18 it was determined that the snubber failures were related to a plant transient which had occurred on December 5, 1994. Following this event additional surveillance of snubbers, attached to the line, should have been performed during the next refueling outage in the spring of 1996; however, due to personnel oversight the TS Surveillance 4.7.7.c had not been performed.

2) On October 12 plant personnel found snubber MK-RCH-0328 to be inoperable (i.e., locked in position). This snubber is a Pacific-Scientific Model PSA-3 mechanical shock arrestor attached to a four (4) inch pressurizer spray line about twenty (20) linear feet from the reactor coolant loop pipe nozzle.

3) On October 15 plant personnel found snubber MK-BDH-4007 to be inoperable (i.e., locked in position). This snubber is a Pacific-Scientific Model PSA-1/4 mechanical shock arrestor attached to a section of one (1) inch line in the Steam Generator Blowdown System about six (6) linear feet from the steam generator nozzle.

DESCRIPTION OF EVENT (cont.):

All snubbers were subjected to testing to address functionality and, in the case of failures, an engineering evaluation was made for adverse impacts to the associated plant piping systems. These evaluations identified one (1) section of piping (pressurizer spray line) which potentially exceeded ASME Code allowable limits and one (1) section of piping (blowdown line) which exceeded ASME Code allowable limits. The engineering review of snubber failures on the feedwater line also identified a failure to perform a required Technical Specification refueling outage surveillance following a plant transient, which occurred on December 5, 1994.

CAUSE OF EVENT:

There are two causes for the events addressed in this report.

The failure to recognize the requirement to inspect snubbers attached to safety systems that have experienced unexpected, potentially damaging, transients in accordance with TS Surveillance Requirement 4.7.7.c is considered to have resulted from a lack of familiarity with the surveillance requirement. The water hammer occurred at the end of Refueling Outage 8 during startup testing. Engineering personnel performed a walkdown of the system to inspect for damage; however, following the determination that there was no apparent damage to components or piping, plant personnel failed to recognize the need to schedule and perform an inspection of potentially impacted snubbers during the next refueling outage in the spring of 1996.

The snubber failures associated with the design basis impacts to the blowdown and spray piping is believed to be caused by lubrication degradation. Stroke testing of snubbers, being removed under the snubber reduction program, identified an abnormally high failure rate. Engineering personnel disassembled and inspected snubbers that failed during testing and determined that lubrication degradation appeared to have been caused by fretting of the bearing/load transfer surfaces throughout the snubbers. Further investigation found that fretting was even evident in snubbers meeting normal industry test criteria (i.e., <5% drag force). Fretting is an application induced condition (high frequency, low amplitude vibration) which led to high drag forces in both the tension and compression mode of snubber operation or, in the worst case, a locked in position condition.

Analysis of Event:

Feedwater Snubbers (MK-FWH-0126, MK-FWH-0128, & MK-FWH-0353: Review of the "as installed" configuration of these snubbers determined that only MK-FWH-0126 would have potentially acted as a locked snubber. MK-FWH-0128 and MK-FWH-0353 would have functioned normally in their installed configuration.

Engineering evaluation of MK-FWH-0126, as a locked snubber, and those with high drag force concluded that the condition had no adverse affect on the design basis stress criteria for the piping and supports. To support this conclusion, several independent walkdowns were performed of the piping with no anomalies found. The review of the piping, insulation, and pipe supports found them in their design configuration with no indications of deformation or dynamic movement.

Reactor Coolant System Snubber MK-RCH-0328: Westinghouse evaluation of the "locked in position" snubber determined that the Reactor Coolant loop nozzle safe end weld and the first three (3) elbows downstream of the nozzle had exceeded the ASME equation 12 allowable limits. The pipe supports would have met the Code allowables in the as found condition.

The welds were inspected by ultrasonic examination and there were no observed indications of stress/service induced flaws. Utilizing data from known transient cycles to date, anticipated transient cycles till the next refueling outage, and the weld inspection has allowed Westinghouse to conclude that the ASME code requirements for fatigue life are shown to be met for operation of the spray line through the next refueling cycle.

Nuclear Blowdown Snubber MK-BDH-4007: The "locked in position" snubber was evaluated by VCSNS engineering with the conclusion that the associated piping exceeded ASME Code allowables. The pipe supports would have met the Code allowables in the as found condition. A field walkdown of the piping found no indication that a transient had occurred to contribute to the snubber failure and testing of other snubbers on the line found no other failures. Being a socket weld configuration piping section, VCSNS personnel were unable to perform NDE on the internal surfaces of pipe welds; therefore, a section of the pipe with calculated high stress was replaced to ensure that the piping was not impacted by the inoperable snubber.

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

IMMEDIATE CORRECTIVE ACTIONS:

VCSNS was in Refueling Outage 10 when the inoperable snubbers were discovered. Failed snubbers that were not part of the snubber reduction project were replaced with new snubbers. Engineering performed an investigation into the cause of failure and potential impacts on piping systems for all snubbers. The engineering review initiated the following additional actions:

- System walkdowns were performed to check for any damage to piping or associated components.

- Ultrasonic weld examinations were performed on the pressurizer spray line with no observed indications of damage.

- Additional snubbers were tested on piping systems where the failed snubbers were attached. This additional testing was used to determine the extent of possible damage.

- All snubbers attached to the feedwater system, that experienced the operational transient, were stroke tested as required by TS 4.7.7.c.

- The section of the Nuclear Blowdown line, which could not be NDE tested, was replaced in order to return the line to a known design basis condition.

All of these actions were completed prior to returning the systems to an operable status.

The failure to recognize and implement snubber inspections per the surveillance requirement of TS 4.7.7.c. is considered to be an isolated event. VCSNS implemented a new event reporting and review system in 1996 which will provide a better overview of corrective actions taken for plant events. The administrative program addressed in Station Administrative Procedure (SAP) 1122, "Condition Evaluation, Reporting, and Trending System," has established a multiple discipline (maintenance, licensing, operations, and engineering) screening committee that performs a preliminary and final review of actions taken for plant events to better insure their adequacy.

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ADDITIONAL CORRECTIVE ACTIONS:

1. An ASME code fatigue qualification, completed by Westinghouse on March 17, 1998, concluded that the calculated usage factors on the pressurizer spray line, where snubber MK-RCH-0328 is attached, for the 40 year life remains less than 1. The ASME code limitation on fatigue usage factor is 1.0. This fatigue qualification considered the pipe stress cycles with the condition of the locked snubber and pipe stress cycles from all the original anticipated design basis transients. Therefore, the current pressurizer spray piping configuration is acceptable and no restrictions in regards to plant transients (heatups/cooldowns) are required, other than those included in the original plant design basis.

2. SAP-1122 will be revised by April 1, 1998 to require that engineering address TS 4.7.7 inspection requirements whenever an unexpected, potentially damaging, transient to plant safety system piping has occurred. This program upgrade will ensure that initial visual inspections (engineering assessment) and subsequent refueling outage visual inspections and freedom of motion tests of mechanical snubbers (Quality Control surveillance) are performed.

PRIOR OCCURRENCES:

None.

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