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

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

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

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
HOPE CREEK GENERATING STATION	05000354	1 OF 14

TITLE (4)
Failure to Comply with Required Action Statement upon Removal of Failed Snubber on the RHR Shutdown Cooling Line

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
11	27	95	95	-- 038 --	02	03	14	96	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	5	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)			
POWER LEVEL (10)	0	20.2201(b)	20.2203(a)(2)(v)	X 50.73(a)(2)(i)(B)	50.73(a)(2)(viii)
		20.2203(a)(1)	20.2203(a)(3)(i)	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 (12)

NAME	TELEPHONE NUMBER (Include Area Code)
C. Manges, Hope Creek Station Licensing Engineer	(609) 339-3234

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)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE).	X NO		XX	XX	XX

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

On December 7, 1995, Snubber 1-P-BC-049-H042, which had been inoperable for ten days, was identified as supporting both the 'A' and 'B' loops of Residual Heat Removal (RHR) shutdown cooling. The applicable Technical Specification (TS) action statements had not been entered. Snubber 1-P-BC-049-H042 is located on the RHR shutdown cooling common suction piping, downstream of the drywell penetration. The snubber had been removed for testing on November 27, 1995 and found to be inoperable. A Level 1 action request was initiated to evaluate the root cause of this failure, as well as two previous failures, prior to reinstalling the snubber. The root cause of not entering the appropriate TS action statement was the incorrect assignment of the Limiting Condition for Operation (LCO) to snubbers supporting the RHR shutdown cooling suction line. The repeat snubber failure is attributed to ineffective corrective actions following previous failures. Corrective actions include ensuring assignment of the correct LCO to snubber work packages, procedure revisions to the RHR system operating procedure, significant enhancements to the Corrective Action Program to address ineffective corrective action implementation, and actions to evaluate the potential for other long-standing unresolved performance or reliability issues on safety significant systems at Hope Creek.

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PLANT AND SYSTEM IDENTIFICATION

General Electric - Boiling Water Reactor (BWR/4)
Residual Heat Removal, EIIS Identifier: BO

IDENTIFICATION OF OCCURRENCE

TITLE (4): Failure to comply with required action statement upon removal of failed snubber on the RHR Shutdown Cooling Line

Event Occurrence: 11/27/95
Event Time: N/A
Discovery Date: 12/08/95

CONDITIONS PRIOR TO OCCURRENCE

Plant in OPERATIONAL CONDITION 5 (Refueling)
Reactor Power 0% of rated

DESCRIPTION OF OCCURRENCE

Discussion of Historical Failures

Snubber 1-P-BC-049-H042 was tested for the first time during Refueling Outage (RFO) 4 and failed the test. No root cause determination has been located; however, it has been determined that the snubber was replaced and a stress evaluation was completed which demonstrated that the design requirements of the system were maintained in the "as found" condition.

During RFO 5, Snubber 1-P-BC-049-H042 was again tested as required by TSs and failed its functional test. A walkdown of the common suction piping was conducted and magnetic particle examination of one pipe elbow was performed which showed no indications. An evaluation was again performed which demonstrated that the design requirements of the system were maintained in the "as found" condition. Engineering also performed an evaluation to determine the probable cause and corrective actions to prevent future damage to the RHR shutdown cooling snubbers. Based on this evaluation, Engineering suspected that the damage was caused by water hammer and recommended changes to Operations Procedure HC.OP-SO.BC-001,

DESCRIPTION OF OCCURRENCE (CONT.)

'RHR System Operating Procedure.' The changes were related to initial system filling and venting and actions to prevent system transients following a system shutdown or isolation. Specifically, the recommendations included incorporating guidance to pressurize the suction piping with the condensate transfer system, use of the high point vents during filling operations, and correction of pressure readings to compensate for elevation head. Although the recommendations were discussed with Operations, they were not entered into the corrective action program and no procedure revision requests were initiated. The recommendations were consequently not implemented.

Discussion of Actions to Investigate and Evaluate the Latest Failure

Snubber 1-P-BC-049-H042 was again discovered to be failed during functional testing in RFO6. This failure was identified and evaluated under the enhanced Nuclear Business Unit (NBU) Corrective Action Program (CAP). A thorough root cause analysis was completed and the associated comprehensive corrective actions are being implemented.

Actions taken to investigate and evaluate the latest snubber failure are described as follows. A walkdown of the system was performed in December 1995 to assess the damage to the piping as a result of overload conditions on the piping supports. Welds on two pipe elbows located in close proximity on both sides of the snubber were examined by magnetic particle examination and no indications were found. An engineering evaluation was subsequently completed to determine the impact on the structural integrity of the piping which concluded that the piping and components supported by the failed snubber were not adversely affected by the failure and remained capable of meeting the design service in the "as found" condition. Additional walkdowns and magnetic particle examinations of the integral welded attachments that could have been impacted by the water hammer event were conducted in January 1996. The pressure boundary in the vicinity of the welded attachments was found to be free of any indications.

Discussion of Issues Related to the Technical Specification (TS) Non-Compliance

This latest failure resulted in a condition prohibited by the Hope Creek Generating Station TSs. A detailed discussion of the issues and circumstances associated with TS non-compliance is provided below.

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

DESCRIPTION OF OCCURRENCE (CONT.)

On November 25, 1995 at 0517 hours, following entry into Operational Condition (OPCON) 5, the 'B' channel outage window was opened, permitting all associated components to be removed from service.

On November 27, 1995, all Residual Heat Removal (RHR) snubber work packages associated with 'B' channel work were presented to work control for Nuclear Shift Supervisor (NSS) approval to work, including a work package for Snubber 1-P-BC-049-H042. The NSS reviewing the work packages acknowledged that the work packages had been assigned to the 'B' loop outage window. The assignment of work packages to pre-staged action statements had been previously reviewed by operations department Senior Reactor Operator (SRO) licensed personnel. The NSS, confident in the SRO pre-outage review of action statement assignments, approved the work packages.

Snubber 1-P-BC-049-H042 is located on the RHR shutdown cooling common system piping downstream of the drywell penetration. It was removed for testing on November 27, 1995 and found to be inoperable. A Level 1 action request was initiated to evaluate the root cause of this failure, as well as two previous failures during RFO4 and RFO5.

On December 7, 1995, Snubber 1-P-BC-049-H042 was recognized as supporting both the 'A' and 'B' loops of RHR shutdown cooling. The Senior Nuclear Shift Supervisor (SNSS) was informed of the effects on both RHR shutdown cooling loops. Technical Specifications 3.7.5 and 3.9.11.1 were then entered.

Technical Specification 3.7.5, 'Snubbers,' requires either replacing or restoring an inoperable snubber to operable status and performing an engineering evaluation for the attached component within 72 hours, or declaring the attached system inoperable and following the appropriate action statement for that system. Technical Specification 3.9.11.1, 'Residual Heat Removal and Coolant Circulation,' requires at least one shutdown cooling mode loop of the residual heat removal (RHR) system be operable.

Failure to take the required actions within the time specified by Technical Specifications 3.7.5 and 3.9.11.1 resulted in a condition prohibited by the Hope Creek Generating Technical Specifications and is reportable under 10CFR50.73(a)(2)(i)(B).

ANALYSIS OF OCCURRENCE

Technical Specification Non-compliance

Hope Creek Generating Station has utilized pre-staged action statements to support work control during refueling outages. The pre-staged action statements were created by operations department SROs based upon system or loop outage windows. Recurring work packages are assigned to these action statements to eliminate administrative burden on the NSS during an outage. The use of pre-staged action statements to support snubber inspections began during RFO3. An error was made by assigning all of the RHR shutdown cooling common suction piping snubbers to a pre-staged action statement associated with only the 'B' shutdown cooling loop.

During the preparation for RFO6, the pre-outage review of the work activities was inadequate. The review of recurring tasks for snubber inspections relied on previously established pre-staged action statements. The pre-outage review of planned snubber work did not include an evaluation of the work package and associated supporting documentation to re-verify assignment to the pre-staged action statement.

As a result, the snubber 1-P-BC-049-H042 continued to be incorrectly assigned to the 'B' loop pre-staged action statement. Work packages, for work other than snubber inspections, included a review and pre-approval of the entire work package.

During the investigation of this occurrence, the Operations department has identified that this same condition probably occurred during RFO4. During RFO4, snubber 1-P-BC-049-H042 failed the inservice inspection test. The snubber was removed from service for eight (8) days. During this period of time, RHR shutdown cooling was in service. The incorrect action statement assignment was not identified at that time because the Technical Specification non-compliance was not recognized.

The failure and rework of snubber 1-P-BC-049-H042 during RFO5 was accomplished within the 72 hour allowed out-of-service time permitted by Technical Specification 3.7.5.

Repeat Failure Analysis

The initial failure of Snubber 1-P-BC-049-H042 was discovered during RFO4. Upon discovery, the snubber was replaced but no analysis has been located to verify that a cause determination was performed or that corrective actions to preclude recurrence were identified. It is assumed that this

ANALYSIS OF OCCURRENCE (CONT.)

resulted in the repeat failure discovered during RFO5. The fact that a root cause analysis was not performed for the initial failure is attributed to the following:

1. Failure to recognize the significance of the problem and lack of sufficient questioning attitude on the part of the personnel involved.

2. A weak corrective action process that did not have a sufficiently low threshold for problem identification.

The cause of the third failure (second repeat failure), discovered during RFO6, was a lack of follow through with the Engineering recommendations for the second failure discovered during RFO5. The second failure had been documented using a discrepancy report (DR) that was dispositioned by Nuclear Engineering. The DR disposition included an analysis which identified water hammer as the cause and made recommendations to address this cause. The recommendations were conveyed to Operations but not incorporated into procedures. The failure to implement the Engineering recommendations is attributed to the following:

1. Inadequate interface between the two organizations in that no tracking of the item was performed after the face-to-face meeting between Engineering and Operations.

2. A weak corrective action program which did not contain sufficient controls to ensure tracking and implementation of corrective actions.

ROOT CAUSE OF OCCURRENCE

Technical Specification Non-compliance

The root cause of these failures to enter the Technical Specification action statements is the failure to properly assign the RHR shutdown cooling common suction line snubbers to the correct pre-staged action statement. Contributing factors to the occurrences are: 1) failure to verify the action statements assignment against controlled technical information (i.e., piping and instrumentation drawings (P&IDs)or system isometric drawings) during RFO4, RFO5, and RFO6, and 2) failure to properly verify the impact of a work package prior to approval to work.

ROOT CAUSE OF OCCURRENCE (CONT.)

Repeat Failure

The root cause of the first repeat failure was failure to perform a root cause analysis as a result of: 1) not recognizing the significance of the issue and a lack of questioning attitude on the part of personnel involved and 2) a weak corrective action process that did not contain a sufficiently low threshold for problem identification.

The root cause of the second repeat failure was lack of follow through with the Engineering recommendations which were developed to disposition the discrepancy report associated with the failure. The lack of follow through was the result of: 1) a weak corrective action process and 2) inadequate interface between Engineering and Operations.

SAFETY SIGNIFICANCE

The safety significance of this occurrence relative to RHR shutdown cooling and its ability to remove decay heat from the reactor core was minimal. The basis for this conclusion is provided below.

No Piping Damage

Based on system walkdowns and magnetic particle examinations, the piping system was found undamaged. Walkdowns of the system were conducted during RFO5 and RFO6. Magnetic particle examination of the elbows in close proximity to the failed snubber found no indications. Magnetic particle examination of integral welded attachments that could have been impacted by the postulated water hammer event demonstrated that the pressure boundary in the vicinity of the integral welded attachments was free of indications. Although the piping system has been exposed to at least three water hammer events in the past, based on the results of walkdowns and non-destructive examination, no detrimental effects on piping integrity have been identified.

Design Service Capability Was Maintained in the "As Found" Condition

A stress analysis was performed to determine whether the "as found" condition of the piping system was capable of meeting the applicable

SAFETY SIGNIFICANCE (CONT.)

design basis requirements including seismic loads. The results of this analysis concluded that all pipe stresses were below design stress allowable values, local stresses at welded connections were acceptable, valve accelerations were less than allowable limits, and remaining support loads were within the design allowable values. These results indicate that the piping and components supported by the failed snubber were not adversely affected and remained capable of meeting the design service

Over the life of the plant, snubbers other than 1-P-BC-049-H042 on the common shutdown cooling line that have been randomly inspected and tested in accordance with the TSs have passed their functional tests. Additionally, the shutdown cooling system has been routinely placed in service over the life of the plant and has experienced at least three water hammer events without any indication of pipe damage. This historical information provides reasonable assurance that pipe integrity would be maintained should another water hammer event have occurred in the RHR system.

A conservative manual calculation has been performed by FPI to demonstrate that the pipe integrity is preserved during and after the previously experienced water hammer events. All the probable events examined by the independent assessment were found not to generate forces that could result in pipe rupture.

PREVIOUS OCCURRENCES

A review of previously documented occurrences did not identify any other Technical Specification action statements that had not been entered due to pre-staged action statements being improperly assigned to an outage work package. As discussed previously, this condition probably existed in RFO4 when snubber 1-P-BC-049-H042 was removed for eight (8) days while RHR shutdown cooling was in service. This previous occurrence is being reported in this LER.

CORRECTIVE ACTIONS

Technical Specification Non-compliance

1. All snubbers on the RHR shutdown cooling suction piping have been restored to their original design condition.

2. All remaining snubber work packages have been reviewed by the ISI supervisor and the Operations department to ensure assignment of the correct pre-staged action statement. No additional errors were identified. This corrective action scope is based upon the fact that work packages other than those for snubber inspections included a review and pre-approval of the entire work package.

3. The method for conducting a pre-outage review of snubber work packages will be revised to ensure adequate communication between specific work groups and operations department personnel. (Prior to RFO7)

Repeat Failure

It was recognized well before RFO6 that the Corrective Action Program in place at the time was not effective in determining root cause or implementation of effective corrective actions. The corrective actions which address the root causes of the repeat failure concern are comprehensive and have been underway since July 1995 and are described below.

1. Implementation of An Enhanced Corrective Action Program (CAP)

A consolidated Corrective Action Program has been implemented to communicate NBU management expectations on timely problem identification and resolution and provides clear definition of roles and responsibilities. The CAP was designed using input from other utilities that have effectively managed program consolidations as measured by improved program and station performance. The consolidated program includes a low threshold for reporting problems, provides aggressive problem assessment/root cause determination expectations and places management in charge of root cause and corrective action completion times.

CORRECTIVE ACTIONS (CONT.)

The Director - Quality Assurance/Nuclear Safety Review has oversight responsibility for the CAP. He has dedicated resources, under the Manager - Corrective Action and Quality Services, to fulfill that responsibility. Measures have been established to monitor the performance of the corrective action process and Station management receives daily reports on any overdue actions.

Accountability for CAP implementation rests with station line management. As such, station managers review root cause evaluations for completeness and adequacy. A Corrective Action Review Board (CARB) has been established at Hope Creek and the General Manager - Hope Creek Operations is its chairman. Completed root cause assessments for significant issues are presented to the CARB for evaluation of the adequacy of the cause determination and corrective actions. A performance measure has been established which tracks the acceptance/rejection rate for CARB presentations. This indicator is included in the monthly report to senior management.

In summary, the NBU CAP has been significantly enhanced and provides comprehensive corrective actions to address the repetitive snubber failure. Aspects of the program that relate specifically to the subject failure include the following. The program contains a low threshold for reporting problems. Corrective actions are assigned to a responsible manager with a scheduled completion date. The corrective action tracking record cannot be closed until all actions are complete. Due date extensions are strictly controlled and all records receive a closure review by the responsible manager to verify that specified actions are tracked and that actions specified have been completed and are effective. Prior to final closure, the Corrective Action Group performs a review to verify all specified actions, including effectiveness reviews, have been properly completed.

CORRECTIVE ACTIONS (CONT.)

2. **Questioning Attitude Being Fostered/Training Being Provided**

The need for a questioning attitude has been repeatedly communicated to Hope Creek personnel and has created a heightened awareness of its importance at the station. A measure of the improvement in this area is the large number of Action Requests being written by Hope Creek personnel.

The FPI Human Error Reduction and the FPI Equipment Root Cause Analysis courses continue to be provided to Engineering personnel. The latter course includes a section on water hammer analysis. Training will be completed by April 30, 1996.

3. **Failure Being Addressed Under the New CAP**

The latest snubber failure was identified and evaluated and is being tracked under the new Corrective Action Program. The root cause analysis has been completed and corrective actions to preclude repeat failures have been identified and assigned and are being implemented. Corrective actions include the following:

The procedure changes expected to preclude recurrence of void-related water hammer events have been made.

Corrective actions to avoid unnecessary shutdown cooling isolations and the associated potential for water hammer events have been identified and assigned in the Corrective Action Program.

The RHR valve closing times have been reviewed to determine if changes can be made to eliminate potential depressurization scenarios. Engineering has determined that the valve closing times should not be changed; however, procedural changes have been made to enhance recovery from the potential depressurization scenarios.

The shutdown cooling system has been reviewed to identify unintended leak paths that could depressurize the shutdown cooling suction line. A sampling of these valves was tested for leakage with negative results. The list of potential leak paths has been provided to the system manager for future reference.

CORRECTIVE ACTIONS (CONT.)

A walkdown of the system was performed in December 1995 to assess the damage to the piping as a result of overload conditions on the piping supports. In addition, welds on two pipe elbows located in close proximity on both sides of the snubber were examined by magnetic particle examination and no indications were found.

An engineering evaluation was completed to determine the impact on the structural integrity of the piping which concluded that the piping and components supported by the failed snubber were not adversely affected by the failure and remained capable of meeting the design service in the "as found" condition.

Additional walkdowns and magnetic particle examinations of the integral welded attachments that could have been impacted by the postulated water hammer event were conducted in January 1996. The pressure boundary in the vicinity of the welded attachments was found to be free of any indications or defects.

The failed snubber will be visually examined and functionally tested during the next refueling outage. This testing will continue until we are confident that corrective actions are effective.

A visual examination and a functional test on Snubber 1-P-BC-049-H042 were satisfactorily completed following the removal of shutdown cooling from service.

A conservative manual calculation has been performed by FPI to demonstrate that the pipe integrity is preserved during and after the previously experienced water hammer events. All the probable events examined by the independent assessment were found not to generate forces that could result in pipe rupture.

In addition, FPI performed a computer (HYTRAN) water hammer hydraulic transient analysis to determine the pressure pulsations in the shutdown cooling line produced by the postulated water hammer scenarios. This analysis included consideration of the ASME Class 1 sections and appropriate ASME Class 2 sections of the shutdown cooling line. Using the output from the hydraulic transient analysis, a piping stress analysis was performed. Peak water hammer pressures and forces were determined and a bounding case for the historical events was derived using the snubber/hanger damage evidence and other data collected.

CORRECTIVE ACTIONS (CONT.)

The results of the computer analysis indicate that the ASME Class 1 piping sections and the appropriate ASME Class 2 sections satisfy the Code stress allowable values. The suspected highest stress piping section identified in the analysis was inspected for potential cracking and to determine actual component thickness. The results of this inspection showed no component damage and allowed the analysis to be updated based on the actual thickness.

A review of the cyclic stresses resulting from the water hammer events does not change any of the conclusions of the design calculation for this Class 1 section of piping, except for increasing the fatigue cumulative usage factor by not more than 2%. The previous cumulative usage factor was 3.7%, resulting in a total usage factor of less than 6%.

4. Actions to determine the extent of the problem

The following actions were taken to evaluate the potential for similar long-standing unresolved performance or reliability issues on safety significant systems at Hope Creek:

System readiness affirmations of twenty selected systems were completed. These twenty systems were selected for detailed review and affirmation based upon historical performance and risk significance of the system. Results of the assessment of these systems were presented to, and accepted by, the Outage Review Committee (ORC).

The Hope Creek Safety Review Group has performed an assessment of Emergency Diesel Generator (EDG) performance and technical issues. The assessment team performed a common cause analysis of the technical, equipment, procedural/processes, and human error issues involving the EDG events and failures over the past year and evaluated the adequacy of the root causes and the effectiveness of the corrective actions that have been initiated. The causal factor analyses were found to be adequate, and the corrective actions are being properly implemented.

Nuclear Engineering has reviewed a sample of 110 of the discrepancy reports that they provided dispositions to over the past five years. No other instances have been identified by this review in which recommended corrective actions were not implemented.

NRC FORM 366A (4-95)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

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CORRECTIVE ACTIONS (CONT.)

In summary, we conclude that the potential for similar long-standing unresolved performance or reliability issues on safety significant systems at Hope Creek is minimal.