UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO NRC BULLETIN 95-02

'UNEXPECTED CLOGGING OF A RESIDUAL HEAT REMOVAL (RHR) PUMP STRAINER

WHILE OPERATING IN SUPPRESSION POOL COOLING MODE"

WASHINGTON PUBLIC POWER SUPPLY SYSTEM

NUCLEAR PROJECT NO. 2 (WNP-2)

DOCKET NO. 50-397

1.0 INTRODUCTION

The U.S. Nuclear Regulatory Commission (NRC) issued Bulletin 95-02, "Unexpected Clogging of a Residual Heat Removal (RHR) Pump Strainer While Operating in Suppression Pool Cooling Mode," on October 17, 1995. It requested all holders of boiling water reactor (BWR) operating licenses or construction permits for nuclear power reactors to take five actions to ensure that unacceptable buildup of debris that could clog strainers does not occur during normal operation. By letters dated November 16, 1995, October 16, 1996, and June 25, 1998, Washington Public Power Supply System (Supply System) responded to the bulletin for WNP-2. In its response, the Supply System stated its intent to comply with the requested actions in the bulletin.

2.0 DISCUSSION

The following describes the requested actions in NRC Bulletin 95-02 and the Supply System's response to each requested action.

Action 1

Verify the operability of all pumps which draw suction from the suppression pool when performing their safety functions (e.g., ECCS [emergency core cooling system], containment spray, etc.), based on an evaluation of suppression pool and suction strainer cleanliness conditions. This evaluation should be based on the pool and strainer conditions during the last inspection or cleaning and an assessment of the potential for the introduction of debris or other materials that could clog the strainers since the pool was last cleaned.

Summary of Response

NRC Bulletin 95-02 is applicable to the following WNP-2 ECCS systems: high pressure core spray (HPCS), low pressure core spray (LPCS), and the residual heat removal (RHR) pumps. The reactor core isolation cooling (RCIC) system is not an ECCS, but is considered within the scope of this requested action because it is capable of taking suction from the suppression pool during performance of a safety function.

Operability of the ECCS pumps depends in part on the net positive suction head (NPSH) required to support operation in an accident. The available NPSH for the LPCS, HPCS, and three RHR pumps is approximately 40 feet, assuming a 50 percent suction strainer blockage at design flow rates, with suppression pool water conditions per Regulatory Guide 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps." The required NPSH for these pumps is approximately 14 feet, as depicted in the Final Safety Analysis Report (FSAR). The available NPSH for the RCIC pump is approximately 60 feet with a pool temperature of 140°F, and the required NPSH is approximately 20 feet. Consequently, the design provides adequate NPSH margin for the ECCS and RCIC pumps with 50 percent suction strainer blockage.

In a letter dated November 16, 1995, the licensee stated that the suppression pool was inspected by divers in 1986, 1989, 1993, 1994 and 1995. The suppression pool was cleaned of debris and sediment in 1994. The 1995 inspection found the strainers unobstructed and in good condition.

In addition, in a letter dated October 16, 1996, the licensee stated that the suppression pool had been inspected in 1996. In a letter dated June 25, 1998, the licensee stated the following: (1) that the suppression pool had been inspected in 1997, (2) the existing suction strainers had been replaced with new large capacity suction strainers in 1998, and (3) inspection and vacuuming of the suppression pool is scheduled for the 1999 refueling outage.

The inspections found a thin layer of loose sediment on the strainers, which is not a safety concern since the sediment is made up of small particles that could easily pass through the strainers and the ECCS system if they become suspended in the pool water. There was no evidence of fibrous material in the pool water or sediment. The licensee stated that no fibrous air filters or other temporary sources of fibrous material that could be dislodged by a loss-of-coolant accident (LOCA) were located inside the primary containment.

On August 3, 1993, the reactor scrammed from full power due to closure of the main steam isolation valves. The main steam safety relief valves (MSRVs) opened automatically, venting the reactor to the suppression pool. The MSRVs were used by the operators in the early stages of the event to control reactor pressure. The RHR system was started in the suppression pool cooling mode to reduce the suppression pool temperature. During the two days the RHR system operated in the suppression pool cooling mode, no indication of strainer blockage occurred. This event preceded the 1994 suppression pool cleaning and indicated that the sediment stirred up by operation of the MSRVs does not contain fibrous material which could block the suction strainers.

Suppression pool and suction strainer conditions are such that the RHR, LPCS, HPCS, and RCIC strainers are observably clear, and there is no indication of the presence of fibrous materials in the suppression pool. Water-leg pumps for the ECCS systems draw suction through the same suction strainers and therefore, are protected to the same degree as the larger ECCS pumps. Control of items brought into primary containment was exercised under the licensee's Foreign Materials Control (FMC) Program since the suppression pool inspection on May 29, 1995. Consequently suppression pool cleanliness conditions are such that NPSH margins are not impaired. Based on the above, the licensee has determined that the RCIC and all ECCS pumps are operable.

Action 2

Confirm the operability evaluation in requested action 1 above through appropriate test(s) and strainer inspection(s) within 120 days of the date of this bulletin.

Summary of Response

The licensee stated that WNP-2's fuel pool cooling (FPC) system design includes the capability to clean the suppression pool water by drawing water through a strainer that has the same size holes as the ECCS strainers and passing it through a filter demineralizer unit before returning it to the suppression pool. The filter and demineralizers are capable of removing particles substantially smaller than particles that could cause clogging of the suction strainers. As a scheduled maintenance item, the FPC system is operated in the suppression pool cleaning mode for 3 continuous days each month. The water volume of the suppression pool is filtered through the FPC system 2.5 times each month, resulting in removal of 90 percent of any suspended particulate present in the pool water at the beginning of the month. The procedure to operate the FPC system requires a specified flow rate that would be difficult to achieve if the suction strainer was blocked, thereby providing the operators with an indication of strainer cleanliness. Periodic surveillance tests of the RHR and LPCS systems are performed and suction, discharge and flow rate data are collected to calculate NPSH losses due to strainer blockage.

The licensee also proposed to perform the inspection of the suppression pool and suction strainers as part of the Foreign Materials Control Program close-out program for the Spring 1996 outage and on an annual basis. Samples of the pool water and sediment will be analyzed for the presence of fiber. If significant quantities of fiber are found then the sediment will be removed by underwater vacuuming.

Action 3

Schedule a suppression pool (torus) cleaning. The schedule for cleaning the suppression pool should be consistent with the operability evaluation in requested action 1 above. In addition, a program for periodic cleaning of the suppression pool should be established, including procedures for the cleaning of the pool, criteria for determining the appropriate cleaning frequency, and criteria for evaluating the adequacy of the pool cleanliness.

Summary of Response

The Supply System responded that based on a number of inspections, fibrous debris does not exist in any significant amounts in the suppression pool. Inspections will continue on a periodic basis to monitor the integrity of the underwater corrosion resistant coating and to identify and remove any debris that could result in blockage of more than one percent of any suction strainer. Pool cleaning will be conducted every four years. Sediment removal will be done in accordance with written procedures.

Action 4

Review foreign material exclusion (FME) procedures and their implementation to determine whether adequate control of materials in the drywell, suppression pool, and systems that interface with the suppression pool exists. This review should determine if comprehensive FME controls have been established to prevent materials that could potentially impact ECCS operation from being introduced into the suppression pool, and that workers are sufficiently aware of their responsibilities regarding FME. Any identified weaknesses should be corrected. In addition, the effectiveness of the FME controls since the last time the suppression pool was cleaned and the ECCS strainers inspected, and the impact that any weaknesses noted may have on the operability of the ECCS should be assessed.

Summary of Response

The licensee responded that a Foreign Materials Control Program has been developed based on industry experience. The program involves procedures governing control of (1) all items taken into the drywell which could fit through the suppression pool downcomer openings and thus fall into the suppression pool from the drywell, (2) maintaining control of all items taken into the wetwell which could clog ECCS stainers or damage ECCS pump internals.

The workers are trained in FMC procedures and practices including industry events. The training emphasizes the importance of continual attention to control of foreign material and involves specific hands-on training such as, valve disassembly to practice control of foreign material.

Action 5

Consider additional measures such as suppression pool water sampling and trending of pump suction pressure to detect clogging of ECCS suction strainers.

Summary of Response

The licensee performs quarterly surveillance tests of the three RHR subsystems and the LPCS systems. The surveillance tests evaluate system performance, including possible loss of NPSH due to suction strainer blockage. The data is evaluated so that a gradual decline in performance would be identified prior to reaching unacceptable levels. No degradation in NPSH has been identified to date, which is consistent with the results of the underwater suction strainer inspections.

Surveillance tests of the RCIC and HPCS systems are performed with the systems aligned to the condensate storage tank instead of the suppression pool. Therefore, the data collected is not indicative of suppression pool cleanliness.

The licensee developed an abnormal condition procedure, "ECCS Suction Strainer Plugging" to provide guidance to the operators on the actions to take if a suction pressure alarm is received, indicating a possible ECCS or RCIC suction strainer blockage.

3.0 EVALUATION

The purpose of the requested actions in the bulletin is to ensure that ECCS and other pumps drawing suction from the suppression pool do not experience an unacceptable buildup of debris that could clog strainers during normal operation which would prevent them from performing their safety function. Action 1 requested licensees to evaluate the operability of their pumps based on the cleanliness of the suppression pool and strainers. Action 2 then requested a verification of the licensee's assessment through a pump test and strainer inspection. These two actions serve to ensure that the pumps are currently operable and not experiencing an unacceptable debris buildup. Actions 3, 4 and 5 serve to ensure that appropriate measures, such as cleaning of suppression pools and strengthening of FMC practices, are taken in the long term to prevent debris accumulation in the pool.

The staff has concluded that the Supply System had a reasonable basis for determining that the pumps were operable in its response to NRC Bulletin 95-02 dated November 16, 1995. The Supply System cleaned the suppression pool and ECCS strainers in 1994 and inspected the strainers as part of the 1995 refueling outage.

In addition, the Supply System conducted inspections of the suppression pool in 1996 and 1997. The August 3, 1993, reactor scram and subsequent operation of the main steam relief valves provided a turbulent mixing of the water in the suppression pool. This turbulent mixing would have stirred up any sediment that could have clogged the suction strainers. The RHR pumps operated for two days in the suppression pool cooling mode without any indication of strainer blockage. Therefore, the sucessful operation of the RHR pumps following actuation of the MSRVs, together with the pool/strainer cleaning and inspection provided an appropriate basis for not conducting pump tests as requested in action 2 of the bulletin. The staff has concluded that these actions are sufficient to ensure operability of the ECCS pumps, and together meet the intent of requested actions 1 and 2 of Bulletin 95-02.

The staff has also concluded that the Supply System's evaluation of their FMC program and establishment of a suppression pool cleaning program meets the intent of requested actions 3 and 4, and are acceptable. The Supply System's programs for trending pump NPSH data, the low suction pressure annunciators, and periodically inspecting the strainers and torus provide additional opportunity for early identification of potential strainer fouling. The staff also notes that during the Spring 1998 refueling outage, the Supply System replaced their old strainers with new larger capacity strainers in response to related Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors." The new strainers will minimize the potential for clogging of the ECCS suction strainers.

The staff has concluded that these additional actions meet the intent of requested action 5 and are acceptable.

4.0 CONCLUSION

On the basis of the above evaluation, the NRC staff finds that the Supply System has complied with the actions requested by the bulletin, and therefore, the Supply System's response to NRC Bulletin 95-02 for WNP-2 is acceptable.

Principal Contributor: J. Cushing

Date: March 26, 1999

50-397
3/2/2000

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From: Stefanie Fountain

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IE22 - 50.73/50.9 Licensee Event Report (LER), Incident Rpt, etc.

Docket: 05000397

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March 2, 2000
GO2-00-042

Docket No. 50-397

U.S. Nuclear Regulatory Commission
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Gentlemen:

Subject: **WNP-2, OPERATING LICENSE NPF-21,**
LICENSEE EVENT REPORT NO. 2000-002-00

Transmitted herewith is Licensee Event Report No. 2000-002-00 for WNP-2. This report is submitted pursuant to 10 CFR 50.73. The enclosed report discusses items of reportability and corrective action taken.

Should you have any questions or desire additional information regarding this matter, please call me or Mr. PJ Inserra at (509) 377-4147.

Respectfully,

RL Webring
Vice President, Operations Support/PIO
Mail Drop PE08

Attachment

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INPO Records Center
NRC Sr. Resident Inspector - 927N (2)
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TE22

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
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TITLE (4)

Potential for a Water Hammer Condition in Reactor Core Isolation Cooling System

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV. NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
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POWER LEVEL	100%		20.405(a)(1)(i)		50.36(c)(1)		50.73(a)(2)(v)		73.71(c)
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LICENSEE CONTACT FOR THIS LER (12)

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SUPPLEMENTAL REPORT EXPECTED (14)				EXPECTED	MONTH	DAY	YEAR
X	YES (If yes, completed EXPECTED SUBMISSION DATE).				07	31	2000

ABSTRACT:

On February 2, 2000 at 1134 hours, with the plant in Mode 1 at 100% power, it was determined that the plant was in a condition that was outside of the design basis. During accident or transient scenarios, or Appendix R fire scenarios which may require the Reactor Core Isolation Cooling (RCIC) System to cycle on and off to maintain reactor vessel water level, restart of the RCIC System in conjunction with a failure of the water leg or "keep-fill" pump (RCIC-P-3) could create a water hammer in RCIC System piping. The potential water hammer might jeopardize the integrity of the reactor coolant pressure boundary or primary containment isolation barriers associated with the RCIC System, as well as result in the potential loss of safe shutdown equipment from pipe whip, jets, and flooding.

Immediate corrective actions included the performance of an assessment that concluded that the RCIC System, reactor coolant pressure boundary, safe shutdown systems, and primary containment integrity were operable but non-conforming to "single failure" design criteria. Furthermore, to eliminate the possibility of water hammer, plant operating procedures have been revised to inhibit RCIC System operation during accident or transient scenarios if a loss of RCIC water leg pump is identified, or if a RCIC water leg pump low discharge pressure condition exists.

The cause of this event is yet to be determined, but appears to be associated with the original design of the plant. This issue is being addressed by the General Electric (GE) Company and discussed with the BWR Owners Group (BWROG).

ABSTRACT (CONTINUED):

Pending further information from GE and the BWROG, it is presently concluded that there are minimal safety consequences associated with this event. As previously stated, the RCIC System, reactor coolant pressure boundary, safe shutdown systems, and primary containment integrity were assessed to be operable but non-conforming to "single failure" design criteria.

This event was previously reported on February 2, 2000 as a 1-hour non-emergency report in accordance with 10CFR50.72(b)(1)(ii)(B).

A supplement to this Licensee Event Report will be issued in order to update the staff on the cause of this event and on any necessary additional corrective actions determined to be necessary.

Event Description

On February 2, 2000 at 1134 hours, with the plant in Mode 1 at 100% power, it was determined that the plant was in a condition that was outside of the design basis. The Reactor Core Isolation Cooling (RCIC) System [BN] is equipped with a water leg or "keep-fill" pump which is designed to maintain the RCIC System full of water at all times to prevent water-hammer. However, the RCIC water leg pump is not single failure proof. During accident or transient scenarios, or Appendix R fire scenarios which may require the RCIC System to cycle on and off to maintain reactor vessel water level, restart of the RCIC System in conjunction with a failure of the water leg pump, or low water leg pump discharge pressure, could create a water hammer in the RCIC piping system. The potential water hammer might jeopardize the integrity of the reactor coolant pressure boundary or primary containment isolation barriers associated with the RCIC System, as well as result in the potential loss of safe shutdown equipment from pipe whip, jets, and flooding.

Immediate Corrective Action

Immediate corrective actions included the performance of an assessment that concluded the RCIC System, reactor coolant pressure boundary, safe shutdown systems, and primary containment integrity were operable but non-conforming to "single failure" design criteria. In addition, discussions were initiated with subject experts at GE on the generic implications of this issue, and the issue is being discussed with the BWROG. Furthermore, to eliminate the possibility of water hammer, plant operating procedures have been revised to inhibit RCIC System operation during accident or transient scenarios if a loss of RCIC water leg pump is identified, or if a RCIC water leg pump low discharge pressure condition exists. Fire tours have also been initiated to ensure a fire does not render the RCIC water leg pump inoperable should the RCIC System be used for plant safe shutdown.

Further Evaluation

The existing RCIC System is designed to automatically start if reactor water level decreases to the low level signal setpoint (Level 2) and automatically stop at the high water level signal setpoint (Level 8). If the reactor water level again decreases to low level signal setpoint, the RCIC System will restart. When RCIC first initiates, a lube oil cooling valve (RCIC-V-46) automatically opens but does not receive a signal to close when RCIC automatically stops. The open valve provides a drain

path that will depressurize the system. The restart of the RCIC System on reaching Level 2 in conjunction with a postulated failure of the water leg pump could create a water hammer in the RCIC piping system. The water hammer could jeopardize the integrity of the reactor coolant pressure boundary or primary containment, cause flooding, or result in the inability to safely shutdown during accident/transient scenarios or during an Appendix R fire.

The original RCIC System design did not include automatic restart after trip at high water level (Level 8). At the Level 8 trip, the RCIC System was tripped and required local reset of the turbine steam inlet control valve before restart could occur. This original design was revised in 1982 following BWROG/GE design change recommendations as a result of the Three Mile Island incident.

Root Cause

The cause of this event is yet to be determined, but appears to be associated with the original design of the plant. A supplement to this Licensee Event Report will be issued in order to update the staff on the cause of this event.

Further Corrective Action

Further corrective actions will be developed after obtaining the results of a review of this issue by GE and the BWROG. Specific information regarding further corrective actions will be provided to the staff in a supplement to this LER.

Assessment of Safety Consequences

Pending further information from GE and the BWROG, it is presently concluded that there are minimal safety consequences associated with this event. The RCIC System, reactor coolant pressure boundary, safe shutdown systems, and primary containment integrity are operable but non-conforming to "single failure" design criteria. Plant operating procedures have been revised to inhibit RCIC System operation during accident or transient scenarios if a loss of RCIC water leg pump is identified, or if a RCIC water leg pump low discharge pressure condition exists. Operators have been trained on the revised procedures. In addition, because the RCIC water leg pump is not protected from the effects of fire, compensatory actions consisting of hourly fire tours are now in place. Lastly, the water leg pump is considered reliable, as it is supplied with safety-related power, and receives a high level of monitoring and preventative maintenance.

Similar Events

There have been no previous similar events at WNP-2.

50-391
3/1/2000

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From: Stefanie Fountain

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Docket: 05000397

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March 1, 2000
GO2-00-041

Docket No. 50-397

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

Gentlemen:

Subject: **WNP-2, OPERATING LICENSE NPF-21,**
LICENSEE EVENT REPORT NO. 2000-001-00

Transmitted herewith is Licensee Event Report No. 2000-001-00 for WNP-2. This report is submitted pursuant to 10 CFR 50.73 and is the follow-up to Event Report Number 33651. The enclosed report discusses items of reportability, corrective action taken, and action to preclude recurrence.

Should you have any questions or desire additional information regarding this matter, please call Mr. PJ Inserra or me at (509) 377-4147.

Respectfully,

GO Smith
VP Generation/Plant General Manager
Mail Drop 927M

Attachment

cc: EW Merschoff - NRC-RIV
JS Cushing - NRC-NRR
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FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
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TITLE (4)

Condition that could have allowed fulfillment of a safety function beyond the Technical Specifications allowable limit

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV. NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
02	01	2000	2000	001	00	03	01	2000	FACILITY NAME	DOCKET NUMBER

OPERATING MODE	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)							
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POWER LEVEL	100		20.405(a)(1)(i)		50.36(c)(1)	X	50.73(a)(2)(v)(c)		73.71(c)
			20.405(a)(1)(ii)		50.36(c)(2)		50.73(a)(2)(vii)		OTHER
			20.405(a)(1)(iii)	X	50.73(a)(2)(i)(B)		50.73(a)(2)(viii)(A)		
			20.405(a)(1)(iv)		50.73(a)(2)(ii)		50.73(a)(2)(viii)(B)		
			20.405(a)(1)(v)		50.73(a)(2)(iii)		50.73(a)(2)(x)		

LICENSEE CONTACT FOR THIS LER (12)

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COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

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SUPPLEMENTAL REPORT EXPECTED (14)			EXPECTED	MONTH	DAY	YEAR
YES (If yes, completed EXPECTED SUBMISSION DATE).	X	NO				

ABSTRACT:

On February 1, 2000 with the plant in mode 1 at 100% power it was determined that the Main Steam Isolation Valves (MSIVs) [JM] would not have closed automatically in response to a condenser low vacuum condition within the value allowed by the Technical Specifications. This condition was discovered after performance of a channel functional test of main condenser vacuum-low instrumentation that functions to generate a signal to close the MSIVs. The setpoints for two pressure switches that monitor condenser vacuum were found to be out of tolerance in a non-conservative direction that exceeded the Technical Specifications allowable value of greater than or equal to 7.2 inches of mercury (Hg) vacuum. A review of the condition determined that the switches had been misadjusted during a previous calibration on November 10, 1999 and had remained out of tolerance since that time. Upon discovery of this condition, the pressure switches were adjusted to the nominal setpoint and the automatic MSIV closure on low condenser vacuum function was restored to actuate at the required value. This condition would have required the control room operators to manually close the MSIVs, according to procedure, when the automatic function did not occur at the expected condenser pressure. Considering the misadjusted setpoint, the automatic closure of the MSIVs would have occurred at 2.8 inches of Hg vacuum. Initiation of automatic MSIV closure at that pressure would have been adequate to prevent condenser over-pressurization. A positive condenser pressure could rupture the diaphragm that is installed to protect the turbine exhaust hood and prevent a potential radiation leakage path following an accident. Condenser vacuum has been maintained within the normal operating range throughout the current operating cycle.

Event Description

On February 1, 2000 with the plant in mode 1 at 100% power it was determined that the Main Steam Isolation Valves (MSIVs) [JM] would not have closed automatically within the value allowed by the Technical Specifications. This condition was discovered after performance of a channel functional test of the main condenser vacuum-low instrumentation to meet Surveillance Requirement (SR) 3.3.6.1.2 for Function 1.d of Table 3.3.6.1-1 of the WNP-2 Technical Specifications. The instrumentation functions to generate a close signal to the MSIVs upon sensing a low condenser vacuum condition. The setpoints for two pressure switches that perform the function were found to be 2.8 and 2.4 inches of mercury (Hg) vacuum. This setting exceeded the Technical Specifications allowable value of greater than or equal to 7.2 inches Hg vacuum. A subsequent review revealed that the switches had been misadjusted during the previous calibration on November 10, 1999, and the out of tolerance condition had existed since then. Had a loss of condenser vacuum occurred, the MSIVs would not have closed automatically in response to low condenser vacuum until the pressure reached 2.8 inches Hg vacuum. Initiation of the automatic function at the higher pressure would have been adequate to fulfill the purpose of this function. MSIV closure is initiated to prevent additional condenser pressurization that could possibly rupture the diaphragm that is installed to protect the turbine exhaust hood and prevent a potential radiation leakage path following an accident.

Immediate Corrective Action

Immediately after the pressure switches were found to be out of tolerance they were readjusted to the correct setpoint in accordance with the applicable surveillance procedures. This action restored the automatic initiation of the MSIV closure on condenser low vacuum function to the required value. Additionally, a check of the accuracy of the calibrated test gauge used to measure the setpoint of the switches was performed when both switches were found out of tolerance. The test gauge was found to be within its required calibration tolerance.

Further Evaluation

The isolation logic scheme for MSIV closure is a "one out of two taken twice" arrangement. This means that to close the MSIVs, a close signal must be generated from channel A or C and channel B or D. The pressure switches that were found out of tolerance were the B and D channels. Since channels B and D would not have actuated at the proper setpoint, the MSIV closure logic would not have been satisfied until condenser pressure increased to 2.8 inches of Hg vacuum. November test data for channels A and C did not indicate an abnormal setpoint. The A and C channels were tested subsequent to the discovery of this condition and found to be within the required setpoint range. A depiction of this logic arrangement for automatic MSIV closure is found on Figure 7.3-2 in the WNP-2 Final Safety Analysis Report.

Root Cause

The root cause for this condition has been determined to be a human error that resulted in inadequate performance of the channel functional test when the channel B and D pressure switches were tested on November 10, 1999. This is apparent when analyzing the as found data taken from the November and January tests. Because the November 1999 test was not performed correctly, these switches appeared to be set at 13.6 and 12.8 inches Hg vacuum and were then adjusted to what appeared to be a value within the normal setpoint range. The January test was performed correctly and the setpoints were found to actually be 2.4 and 2.8 inches of Hg vacuum. The as found data for the January test was out of tolerance by approximately the same magnitude from the nominal setpoint but in the opposite direction from the November test. This supports the conclusion that because the November 1999 test was performed inadequately, it resulted in false high readings that prompted the test performer to adjust the setpoint to a lower non-conservative setpoint. In the November test an electric vacuum pump and vent valve arrangement was used to establish and regulate a negative pressure to check the setpoint of the switch. The vacuum within the test boundary was measured with a calibrated test gauge connected to the pressure switch with approximately 30 feet of tubing. This arrangement is not conducive to establishing a static pressure within the test boundary. This configuration was recreated during the root cause investigation. It was demonstrated that if there was a restriction in the tubing and if the fittings connecting the tubing were not tight, it created an offset between the pressure at the switch and the pressure read on the calibrated test gauge used to measure the setpoint. In the January test, a closed system was used featuring a mechanical bellows for changing the volume of the test boundary and thereby controlling vacuum. Because this was a closed system any leaks would be manifest by an inability to maintain a constant vacuum. The closed test system also establishes a static pressure within the test boundary that is applied equally to the pressure switch and the calibrated test gauge and is a more accurate method of measuring the setpoint of the condenser vacuum switch.

Further Corrective Action

To prevent recurrence of a human error resulting in an inadequately performed test, surveillance procedures involving vacuum test applications will be revised. Additional precautions will be added instructing test performers to either use hand operated vacuum devices such as a mechanical bellows, or perform testing to verify the integrity of the test boundary when using an electric vacuum pump.

Assessment of Safety Consequences

MSIV closure in response to a condenser low vacuum condition is initiated to prevent the addition of steam that would lead to additional condenser pressurization. Elevated condenser pressure could possibly rupture the diaphragm that is installed to protect the turbine exhaust hood. Rupture of this diaphragm would also create a potential radiation leakage path following an accident. The condenser pressure that could have resulted from the automatic MSIV closure at 2.8 inches of Hg vacuum would not have been sufficient to reach the turbine exhaust diaphragm burst pressure of 5 pounds per square inch (psi). For this reason, it is not postulated that a leakage path for radioactive steam could have been created by this condition. Additionally, control room operators are trained and expected to manually initiate any automatic function that fails to occur at the required process value. This training is consistent with guidance provided in NUREG-1021. Other automatic functions that occur to mitigate a decrease in condenser vacuum are a turbine trip at 20 inches of Hg vacuum and closure of the turbine bypass valves at 7 inches Hg vacuum. The WNP-2 abnormal condition procedure for loss of condenser vacuum provides instructions for control room operators during this condition. The procedure instructs the operators to consider initiating a reactor scram if an automatic turbine trip is imminent and reduce reactor power as necessary to maintain turbine exhaust pressure and temperature within limits. The procedure also instructs the control room operators to verify all automatic functions have occurred. Implicit in this statement is the expectation to manually initiate those functions that have not automatically occurred as expected. For these reasons and considering that condenser vacuum has been maintained within the normal operating range throughout the current operating cycle, there have been no safety consequences as a result of this condition.

Similar Events

There have been no previous similar events at WNP-2 that have resulted in the possibility of an automatic function occurring outside of the process value required by Technical Specifications due to allowing a setpoint to remain out of tolerance.

50-397
1/17/2000

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Distribution Sheet

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Subject:
LER 99-003-00, "Failure to Comply with the Technical Specification Bas
es Description of a Channel Check."

Body:
ADAMS DISTRIBUTION NOTIFICATION.

Electronic Recipients can RIGHT CLICK and OPEN the first Attachment to

A04

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the Document in ADAMS. The Document may also be viewed by searching f
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IE22 - 50.73/50.9 Licensee Event Report (LER), Incident Rpt, etc.

Docket: 05000397

ENERGY NORTHWEST

P.O. Box 968 ■ Richland, Washington 99352-0968

January 17, 2000
GO2-00-012

Docket No. 50-397

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

Gentlemen:

Subject: **WNP-2, OPERATING LICENSE NPF-21,**
LICENSEE EVENT REPORT NO. 1999-003-00

Transmitted herewith is Licensee Event Report No. 1999-003-00 for WNP-2. This report is submitted voluntarily because it is a condition that might be of generic interest or concern. The report discusses items of reportability, corrective action taken, and action to preclude recurrence.

Should you have any questions or desire additional information regarding this matter, please call me or Mr. PJ Inserra at (509) 377-4147.

Respectfully,

D. W. Coleman for

RL Webring
Vice President, Operations Support/PIO
Mail Drop PE08

Attachment

cc: EW Merschoff - NRC-RIV
JS Cushing – NRC-NRR
INPO Records Center
NRC Sr. Resident Inspector - 927N (2)
DL Williams - BPA/1399
TC Poindexter - Winston & Strawn

JE22 1/1

ML003677728

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Washington Nuclear Plant - Unit 2	50-397	1 OF 3

TITLE (4)

Voluntary Licensee Event Report of failure to comply with the Technical Specification Bases description of a Channel Check

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV. NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
10	22	1999	1999	003	00	1	17	2000	FACILITY NAME	DOCKET NUMBER

OPERATING MODE	2
POWER LEVEL	0%

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

	20.402(b)		20.405(c)		50.73(a)(2)(iv)		73.71(b)
	20.405(a)(1)(i)		50.36(c)(1)		50.73(a)(2)(v)		73.71(c)
	20.405(a)(1)(ii)		50.36(c)(2)		50.73(a)(2)(vii)		OTHER
	20.405(a)(1)(iii)		50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		X
	20.405(a)(1)(iv)		50.73(a)(2)(ii)		50.73(a)(2)(viii)(B)		
	20.405(a)(1)(v)		50.73(a)(2)(iii)		50.73(a)(2)(x)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
F. A. Schill, Licensing Technical Specialist	(509) 377-2269

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX

SUPPLEMENTAL REPORT EXPECTED (14)			EXPECTED	MONTH	DAY	YEAR
YES (If yes, completed EXPECTED SUBMISSION DATE).	X	NO				

ABSTRACT:

On October 22, 1999 at 1309 hours, with the plant in mode 2 and power ascension in progress following the R-14 refueling outage, plant operators questioned the adequacy of the method in which a surveillance activity was being performed. Technical Specifications Surveillance Requirement (SR) 3.3.5.2.1 requires a 12-hour Channel Check for the Condensate Storage Tank (CST) [KA] level-low function. This function automatically transfers the Reactor Core Isolation Cooling (RCIC) [BN] system suction from the CST to the suppression pool when low CST level is sensed. The Channel Check for this function is performed by verifying the status of the CST level-low alarm is consistent with indications on independent CST level instruments. This method meets the Technical Specifications definition of a Channel Check; however, it is inconsistent with the Technical Specifications Bases description for SR 3.3.5.2.1 which states that performance of a Channel Check will ensure that a gross failure of instrumentation has not occurred. The Bases description does not apply to the non-indicating design of the instrumentation that performs the CST level-low function for the RCIC system.

Immediate corrective action was taken to declare the RCIC CST level instruments inoperable and configure the RCIC system to take suction from the suppression pool. Further corrective actions are to revise the Bases for SR 3.3.5.2.1.

The cause of this event is failure to include deletion of SR 3.3.5.2.1 for the CST level-low function in the deviations from NUREG 1434 when the Improved Technical Specifications (ITS) were implemented at WNP-2.

There are no safety consequences associated with this event. The Technical Specifications permit the current configuration of the RCIC system for an indefinite period of time and the CST is still available as a source of water if needed. This voluntary report is submitted due to the generic interest this condition may hold for the staff and other licensees.

Event Description

On October 22, 1999 at 1309 hours, with the plant in mode 2 and power ascension in progress following the R-14 refueling outage, plant operations personnel questioned the adequacy of the method in which a Channel Check surveillance activity was being performed. Technical Specifications Surveillance Requirement (SR) 3.3.5.2.1 requires a 12-hour Channel Check for the Condensate Storage Tank (CST) [KA] level-low function. This function automatically transfers the Reactor Core Isolation Cooling (RCIC) [BN] system suction from the CST to the suppression pool when low CST level is sensed. The Channel Check for this function is performed by verifying that the status of the CST level-low alarm is consistent with independent CST level indicating instruments. The Technical Specifications Bases description for SR 3.3.5.2.1 states that performance of a Channel Check will ensure that a gross failure of instrumentation has not occurred. Because a gross failure of the instrument channel, such as a stuck displacer or open circuit would not be detected by this Channel Check method, it is inconsistent with the Technical Specifications Bases description. Technical Specifications defines a Channel Check as the qualitative assessment by observation, of channel behavior that shall include, where possible, comparison of the channel indication or status to other indications or status from independent instrument channels measuring the same parameter. This definition is consistent with the method in which SR 3.3.5.2.1 for the RCIC function is performed.

Immediate Corrective Action

Immediate corrective action was taken to declare the RCIC CST level instruments inoperable and issue instructions to control room supervisory personnel to declare the RCIC system inoperable whenever its suction is lined up to the CSTs. Because the RCIC function of the level instruments was declared inoperable, the RCIC system was configured to take suction from the suppression pool.

Further Evaluation

This voluntary report is submitted with respect to the Statement of Considerations for 10 CFR 50.73 which encourages licensees to report any event or condition that might be of generic interest or concern. The Technical Specification Bases description of SR 3.3.5.2.1 originated from NUREG 1434, the BWR/6 Standard Technical Specifications. Although WNP-2 is a BWR/5 design, NUREG 1434 was adopted by WNP-2 during implementation of Improved Technical Specifications (ITS). Many Channel Check SRs were deleted when NUREG 1434 was adapted for the WNP-2 ITS. These deletions were justified because the current WNP-2 design does not provide indication in the deleted instrument channels. The Channel Check deletions were accepted in the staff's Safety Evaluation Report because verification of the status of an alarm every 12 hours does not provide any further information to the operators that is not constantly available through the absence of an alarm. This event may be of generic interest to the staff and to other licensees who are in the process of transitioning to ITS at their facilities.

Root Cause

The cause of this event is failure to include deletion of SR 3.3.5.2.1 for the CST level-low function in the deviations from NUREG 1434 when Channel Checks were deleted during implementation of the ITS at WNP-2.

Further Corrective Action

The following further corrective actions are planned:

1) Revise the Technical Specifications Bases for the identified Channel Check SRs to provide descriptions that are consistent with the design of the instrumentation.

2) Initiate a Technical Specifications amendment request to delete Channel Check SR 3.3.5.2.1.

Assessment of Safety Consequences

There are no safety consequences associated with this event because the Technical Specifications permit the current configuration of the RCIC system for an indefinite period of time and the CSTs are still available as a source of water if needed.

Similar Events

There have been no previous similar events at WNP-2.

50-397
11/29/99

Distribution Sheet

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Body:
PDR ADOCK 05000397 S

Docket: 05000397, Notes: N/A

A04

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P.O. Box 968 ■ Richland, Washington 99352-0968

November 29, 1999
GO2-99-205

Docket No. 50-397

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

Gentlemen:

Subject: **WNP-2, OPERATING LICENSE NPF-21,**
LICENSEE EVENT REPORT NO. 1999-002-00

Transmitted herewith is Licensee Event Report No. 1999-002-00 for WNP-2. This report is submitted pursuant to 10 CFR 50.73(a)(2)(i)(B) and discusses items of reportability, corrective action taken, and action to preclude recurrence.

Should you have any questions or desire additional information regarding this matter, please call me or Mr. PJ Inserra at (509) 377-4147.

Respectfully,

RL Webring
Vice President, Operations Support/PIO
Mail Drop PE08

Attachment

cc: EW Merschoff - NRC-RIV
JS Cushing - NRC-NRR
INPO Records Center
NRC Sr. Resident Inspector - 927N (2)
DL Williams - BPA/1399
TC Poindexter - Winston & Strawn

IE22

PDR ADOCK 05000397

9934101 25

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Washington Nuclear Plant - Unit 2	50-397	1 OF 3

TITLE (4)

Operation or condition prohibited by WNP-2 Technical Specifications due to a missed surveillance

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV. NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
10	27	1999	1999	002	00	11	29	1999	FACILITY NAME	DOCKET NUMBER

OPERATING MODE	1
POWER LEVEL	100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

	20.402(b)		20.405(c)		50.73(a)(2)(iv)		73.71(b)
	20.405(a)(1)(i)		50.36(c)(1)		50.73(a)(2)(v)		73.71(c)
	20.405(a)(1)(ii)		50.36(c)(2)		50.73(a)(2)(vii)		OTHER
	20.405(a)(1)(iii)	X	50.73(a)(2)(i)(B)		50.73(a)(2)(viii)(A)		
	20.405(a)(1)(iv)		50.73(a)(2)(ii)		50.73(a)(2)(viii)(B)		
	20.405(a)(1)(v)		50.73(a)(2)(iii)		50.73(a)(2)(x)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
F. A. Schill, Licensing Technical Specialist	(509) 377-2269

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
E	IJ	CNV	FUJI	N					

SUPPLEMENTAL REPORT EXPECTED (14)			EXPECTED	MONTH	DAY	YEAR
YES (If yes, completed EXPECTED SUBMISSION DATE).	X	NO				

ABSTRACT:

On October 27, 1999 at 1300 hours, with the plant in Mode 1 at 100 percent power, plant operators questioned a discrepancy between indicated drywell identified leakage rate and the value of drywell identified leakage rate measured by an alternate method. Upon further investigation plant staff determined that the identified leakage rate signal to the control room indicating instrument was not being received because its signal converter was de-energized. Further investigation determined this condition existed since the signal converter batteries discharged after power was removed during an electrical bus maintenance period in the R-14 refueling outage. Total drywell leakage is required to be verified within specified limits every 12 hours in modes 1, 2, or 3 pursuant to Technical Specification Surveillance Requirement (SR) 3.4.5.1. Control room operators at WNP-2 used the erroneous indication to calculate total drywell leakage from the time the plant entered the mode of applicability for SR 3.4.5.1 on 10/22/99, until the de-energized instrument was discovered on 10/27/99.

The cause of the signal converter failure is inadequate interface between project engineers and procedure writers during the design change process resulting in inadequate procedure revisions to be implemented.

Upon discovery of this condition, corrective action was taken to restore power to the signal converter at which time the correct drywell identified leakage indication was immediately observed on the control room indicator.

There were no adverse safety consequences associated with the lapse of identified leakage rate indication. Although the requirement of SR 3.4.5.1 to verify total leakage rate within the limit was not met during this period, the absence of the identified leakage sump fill rate monitor alarm and recorded unidentified leakage rate indicate that the limit for total leakage of 25 gpm in a 24 hour period was not exceeded.

Event Description

On October 27, 1999 at 1300 hours, with the plant in Mode 1 at 100 percent power, a discrepancy was observed between drywell identified leakage rate as indicated on a control room recorder and the value for drywell identified leakage rate measured by alternate means. Upon further investigation it was determined that the identified leakage rate signal to the control room recorder was not being provided because its signal converter was de-energized. Further evaluation indicated this condition existed since the signal converter's batteries had discharged during the time its supplying electrical bus was de-energized during the R-14 refueling outage.

On 10/22/99, WNP-2 entered the mode of applicability for Technical Specification Surveillance Requirement (SR) 3.4.5.1 during power ascension following refueling outage R-14. With the plant in modes 1, 2, or 3, Limiting Condition for Operation (LCO) 3.0.1 requires LCO 3.4.5 Reactor Coolant System (RCS) Operational Leakage to be met. SR 3.4.5.1 requires verification every 12 hours that total leakage rate into the drywell is less than or equal to 25 gallons per minute (gpm) averaged over the previous 24-hour period. Total leakage rate is determined by totaling identified leakage rate from the Drywell Equipment Drains (EDR) and unidentified leakage rate from the Drywell Floor Drains.

Records indicate the control room operators performed the procedure that implements SR 3.4.5.1 within the required frequency intervals during the modes of applicability. However, the erroneous data for identified leakage rate was used during these performances to determine total RCS operational leakage.

Immediate Corrective Action

Immediate corrective action was taken to re-energize the signal converter by turning its power switch on. When this was done, the correct drywell identified leakage indication was immediately observed on the control room recorder. The restored indication was then verified to be correct by correlation with the results of a measurement of drywell identified leakage flow rate by alternate means.

Further Evaluation

By design, the signal converter requires its power switch to be turned on to restore operation whenever the backup batteries are discharged following power interruption. The signal converter is powered by Nickel-Cadmium batteries that are continuously charged by an AC to DC adapter during normal operation. When the AC power to the adapter was secured on 10/2/99 during the electrical bus outage, the batteries continued to power the converter until battery voltage dropped below the minimum threshold at which time the converter automatically turned off. With the plant shutdown during R-14, the electrical bus was re-energized on 10/4/99. At this time, the converter batteries automatically recharged and the converter then required its power switch to be turned on in order to resume functioning.

Supplement 1 to Generic Letter (GL) 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping", dated February 4, 1992, indicates that it is not only necessary to verify leakage to be within limits, but that the intention of the SR is also to quantitatively measure leakage rate. The Generic Letter 88-01 supplement discusses the allowed outage time for leakage rate measurement instruments and indicates that it is necessary to ensure the capability to quantitatively measure leakage is not lost because this capability is essential for safe plant operation.

This condition was determined to be reportable because the time that the leakage instrument was not functioning but was relied upon to calculate total leakage exceeded the SR interval plus the SR 3.0.2 extension plus the LCO action completion time. This condition meets the reporting criteria of 10 CFR 50.73(a)(2)(i)(B) as an operation or condition prohibited by the plant's Technical Specifications. The condition was determined to meet reportability criteria of NUREG-1022 revision 1 because there is firm evidence the discrepancy existed prior to the time of discovery.

Root Cause

The cause of the signal converter failure is inadequate interface between project engineers and procedure writers during the design change process. The potential problems associated with power loss, battery failure, and subsequent restoration were recognized prior to installation. This design feature was not adequately communicated to procedure authors when they were tasked to prepare procedure revisions to implement the design change. This resulted in procedures that did not contain instructions to turn on the signal converter following power restoration.

Further Corrective Action

The restoration procedures for the electrical bus that provides power to the signal converter will be revised to require manual reset of the flow signal converter following power restoration.

The administrative procedure governing design changes will be revised to require formalized correspondence between project engineers and procedure authors containing specific details regarding design changes.

Assessment of Safety Consequences

The erroneously indicated EDR leakage rate for the 5-day period between 10/22/99 and 10/27/99 did not present any adverse safety consequences. Other alarms were operable during this period that would have alerted the operators if the identified leakage rate approached the limit for total leakage specified in LCO 3.4.5. The EDR sump fill rate monitor alarm actuates when the leakage rate exceeds 15.5 gpm. Records indicate that during the time the EDR indication was erroneous, FDR leakage was always less than 5 gpm. Therefore, total drywell leakage did not exceed the LCO 3.4.5 limit of 25 gpm. Proper function of the EDR sump fill rate monitor alarm was verified subsequent to restoration of the EDR leakage instrumentation.

Similar Events

There have been no events in the last two years in which inadequate communications between personnel implementing a design change resulted in failure to meet a Technical Specification surveillance requirement.

50-397
9/9/99

Distribution Sheet

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From: Geetha Raghavan

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Subject:
Event tracking sheet regarding 990503 event involving on 980623,Region IV sent memo to NRR on subject of potential inadequacies in testing requirements for nuclear air treat ment systems.WNP-2 license had not established routine test methodology.

Body:

Docket: 05000397, Notes: N/A

A04

OEAB EVENT TRACKING SHEET

No Sort Specified
QUERY> "HODGE" $ Assigned To & Assigned Date >= 05/03/99 & Assigned Date <= 05/03/99 & "WASHINGTON NUCLEAR" $ Plant Nam

Plant: WASHINGTON NUCLEAR **Unit:** 2 **Engineer:** HODGE V.

Event: 05/03/99 **Morning Report:** **Briefing:**

50.72#: 0 **LER#:** 050000009900000 **PN#:**

Other Notification: REPORTED BY REGION IV

System: **Component:**

OPERATING MODE

(1) - Operation
2 - Startup
3 - Hot Standby
4 - Hot Shutdown
5 - Cold Shutdown
6 - Refueling
7 - Other _____

SIGNIFICANCE

A - Reactor Protection System
B - Safety-Related Cooling System
C - Fuel Cladding
D - Reactor Coolant Pressure Boundary
~~E~~ - Containment
F - Plant Power
G - Unexpected Plant Performance
(H) - Other: ~~HVAC~~ Filter Bypass Leakage.

CAUSE

1 - Equipment Failure
2 - Design or Installation Error
3 - Operating Error
(4) - Maintenance Error
5 - External
6 - Other _____

EVENT TYPE

SIG - Significant Event
EOI - Event of Interest
TBD - To Be Determined
OTH - Other

POTENTIAL AO: **Criterion:** _____

Proposed By: HODGE V. Vern Hodge 90909
Engineer

Approved: _____
Section Leader

J.E. Lyons 9/20/99
Deputy Branch Chief

Kathy A. Gray 10/14/99

EVENTS ASSESSMENT PANEL **First Screening:** **Closure:**

Significance Description:

On 23 Jun 98, Region IV sent a memo to NRR on the subject of potential inadequacies in testing requirements for nuclear air treatment systems. The WNP2 license had not established a routine test methodology for evaluating system bypass leakage pathways (e.g., floor drains and bypass dampers). Due to the potential of off-site dose consequences from leakage via these pathways, the licensee agreed that routine testing would be appropriate and is in the process of establishing a methodology to

DFX2

9932500026

No Sort Specified
QUERY> "HODGE" $ Assigned To & Assigned Date >= 05/03/99 & Assigned Date <= 05/03/99 & "WASHINGTON NUCLEAR" $ Plant Nam

Significance Description (cont.)

implement a testing program. This situation may have generic implications.

G:\FORMS\ETS.W61

ASSIGNMENT SHEET- PART 21

PANEL -- _ YES x NO

ASSIGNED TO:	Hodge	ASSIGNMENT DATE	05/03/99
PLANT & UNIT/VENDOR	WNP2		
REPORTED BY:	Region IV	REPORT DATE	06/23/98
EN NO:		EVENT DATE	
MR NO:		EVENT TYPE	
PN NO:		SPECIALITY CODE	
ACCESSION NO:		PART 21 NO	
OTHER SOURCE REPORT:			
RELATED REPORTS:			

EVENT/CONDITION SUMMARY

On 23 Jun 98, Region IV sent a memo to NRR on the subject of potential inadequacies in testing requirements for nuclear air treatment systems. The WNP2 license had not established a routine test methodology for evaluating system bypass leakage pathways (e.g., floor drains and bypass dampers). Due to the potential of off-site dose consequences from leakage via these pathways, the licensee agreed that routine testing would be appropriate and is in the process of establishing a methodology to implement a testing program. This situation may have generic implications.

SPECIFIC FOLLOW-UP ASSIGNMENT

DETERMINE DETAILS, EVALUATE SAFETY SIGNIFICANCE AND GENERIC IMPLICATIONS. IN ADDITION, ADDRESS THE FOLLOWING SPECIFIC CONCERNS:

PREPARE TO BRIEF: _ YES _ NO

TARGET CLOSEOUT SCHEDULE:

INITIAL SCREENING BY PANEL

REGULATORY ASSESSMENT:

EVENT/CONDITION SAFETY SIGNIFICANCE: _ OTH _ EOI _ SIG _ AO

REMAINING OR ADDITIONAL FOLLOW-UP ITEMS:

CLOSEOUT TEXT

SUMMARY ASSESSMENT: (Abstract of Closeout/Findings)

Event Notification 27689 from Millstone was reviewed and closed. In Oct 94, the Millstone licensee reported 1/4 inch diameter holes in the ducting of each of the two trains of the standby gas treatment system that might jeopardize control room habitability following certain postulated accidents. These holes were part of a drain system used to remove water from a deluge system and were apparently left open when the drain system was removed. A check of Pilgrim and Vermont Yankee stations disclosed deluge systems there with small drain lines still intact. Therefore, open drain lines do not appear to be a generic problem.

The safety significance of this event depends on location of fans and filters with respect to the drain and entry holes (pressure inside ducting), separation of ducting, and whether the gas would be flowing through charcoal adsorbers. Because of the torturous path it is believed that only a small amount of radioactive material would get into the control room system. Even so, a radiation alarm for the control room ventilation system is located in the surrounding area outside of the ducting to alert operators.

The NRC addressed the broad question of ventilation system leakage in Information Notices 93-06 and 90-02. No further regulatory action is needed.

CLOSEOUT/FINDINGS:

06/23/98 Assigned to J. Tappert but not entered in the events tracking system (ETS)
05/17/99 Assigned to V. Hodge

Additional documents addressing this concern:

1. NRC Inspection Report Inspection Report 50-397/98-03

Relevant parts of this inspection report are copied below; this is the substance of the message in the 23 Jun 98 memo from Region IV to NRR:

Licensee technical specification (TS) 3.6.4.3 requires the standby gas treatment (SGT) system to be tested in accordance with the Ventilation Filter Testing Program as described in TS 5.5.7. TS 5.5.7 requires the high efficiency particulate air (HEPA) and charcoal filters of the SGT system to be tested in accordance with American Society of Mechanical Engineers (ASME) standard ASME-N510-1989, "Testing of Nuclear Air Treatment Systems," to demonstrate that penetration and system bypass is less than 0.05 percent.

The inspector reviewed the 1975, 1980, and 1989 versions of ASME-N510 and found that each of the standards describes the methodology for testing the penetration and bypass of the HEPA and charcoal filters. In the 1975 and 1980 versions of the standard an appendix was included regarding the importance of in-place leak tests. In these versions it was stated that "an installed system can be assumed to have an efficiency equivalent to that of the [factory tested charcoal] sample only if: . . .there are no leaks or bypasses in either the individual [charcoal] cells (factory tests) or the installed system (field tests)." The 1989 version went further and included a specific testing method and frequency for system bypass leakage. The purpose of the test was described as follows:

Systems using HEPA filters and adsorber banks may contain bypass dampers, ducts, conduits, floor drains, pipe penetrations, etc., which could potentially defeat the purpose of high efficiency

nuclear air treatment components. Therefore, it is necessary to perform tests which challenge all these potential bypass leakage paths...

ASME N510-1989 recommends a frequency of once per operating cycle for performing the system bypass leakage test.

The inspector noted that the SGT filter train contains a number of floor drains for directing the flow of fire protection deluge water to the reactor building floor drain system. These drains are normally closed by means of a passive check valve in each drain line. Several of the drains are located in the exhaust section of the filter train, and as such, present a potential bypass leakage path if the check valve failed or was not properly seated. In questioning the licensee on testing of these valves, it was determined that the licensee did not have a program for visually inspecting the valves or a procedure to measure system bypass leakage from the floor drain system.

In response to the inspector's concerns, the licensee attempted to measure the air flow into the common header of the SGT floor drain system to evaluate if any of the check valves may be leaking by their seats. Measurements made with a pitot tube velocimeter at the outlet of the common drain header found that the air velocity into the header on both trains was less than that detectable with the instrument (25 ft/min). The licensee then estimated that leakage flow into the system, via the check valves, was less than 0.5 scfm, well within the analysis assumption of 14 scfm.

Although the leakage rate past the check valves was found to be acceptable, the inspector questioned the licensee on the need to test this leakage path on a periodic basis as recommended by ASME N510-1989. The licensee reviewed the testing requirements of the SGT system and did not identify any licensing basis for testing the bypass flow path. Specifically, in implementing the testing requirements of ASME N510-1989, through TS 5.5.7, only testing of the in-place HEPA filters and charcoal adsorbers is required. However, due to the potential consequences of bypass leakage, the licensee agreed that testing would be prudent. As a result, the licensee will be implementing a periodic testing program for the SGT floor drains.

A review of previously closed ETS assignments showed one relevant item:

Site Name:	Event No.: 27689
	Event Date: 10/31/94
MILLSTONE	Closeout Date: 12/12/94

CLOSEOUT: During a licensee walkdown to document plugging of holes previously detected in the standby gas treatment system ducting, 2 additional small holes (1/4 inch dia) were found in the ducting of each of the 2 trains of SBGTS. These holes were reported because they were additional paths by which gas in the SBGTS could leak out into the surrounding room and possibly get pulled into the adjacent ducting of the control room ventilation system, also located in the same room. The concern is that the control room habitability might be jeopardized following certain postulated accidents.

The holes have been judged to be drain holes that were intended to be part of a drain system used to remove water from a deluge system. [Drains at two other facilities had closed block valves or loop seals.] It appears that when the drain collection system was removed at Millstone, the drain connections were left open. The followup of the prior holes disclosed these connections, but they were incorrectly determined to have already been plugged. A check of Pilgrim and Vermont

Yankee stations disclosed that they have a deluge system and small drain lines that are still intact; therefore, open drain lines do not appear to be a generic problem. The broad question of ventilation system leakage was previously discussed in IN 93-06 and therefore nothing further needs to be done.

Plant Systems has been asked to determine if this is a generic problem. The above discussion would indicate that the open drain holes are not generic. Whether this event has any significance is harder to establish because the significance the potential gas entry into the control room heating and ventilation system depends upon location of fans and filters with respect to the drain and entry holes (pressure inside ducting), separation of ducting, and whether the gas would be flowing thru charcoal adsorbers. Because of the torturous path it is believed that only a small amount of radioactive material would get into the control room system. Even so, a radiation alarm for the control room ventilation system is located in the surrounding area outside of the ducting and therefore is expected to alert the operators prior to having a major problem.

The NRC has issued the following generic communications pertaining to the standby gas treatment system at boiling water reactors:

IN 93-06, "POTENTIAL BYPASS LEAKAGE PATHS AROUND FILTERS INSTALLED IN VENTILATION SYSTEMS," January 22, 1993

Discusses potential problems resulting from missing or deteriorated seals around shafts that penetrate fan or filter housings and inadequately sealed ducting seams used in engineered safety feature (ESF) ventilation systems.

IN 90-02, "Potential Degradation of Secondary Containment," January 22, 1990

Discusses potential problems involving degradation of secondary containment as a result of unforeseen interactions with various normal plant ventilation systems and inadequate surveillance testing of secondary containment integrity.

The testing program deficiency discovered at WNP2 in June 1998 was corrected though estimated SGT inleakage, based on measured air flow into the common header of the SGT floor drain system, was well within the SGT analysis assumption for that leakage. Similar reports from other licensees were corrected or retracted upon further evaluation. The NRC issued information notices for similar problems in previous years. While generically applicable, this problem does not have great safety significance. No further regulatory action is needed.

The NRR Plant Systems Branch agrees with this assessment. On 09 Sep 99, the Region IV memo was closed by telephone.

FINAL PANEL ASSESSMENT:

The panel considered the significance of this to be..... It was judged to (not) be generic. Factors impacting priority determination were....

EVENT/CONDITION SAFETY SIGNIFICANCE: _ OTH _ EOI _ SIG _ AO

BASIS: _ RISK _ PROGRAMMATIC _ MARGIN _ N/A