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RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 91-013-02:on 910507,first several items of non-compliance w/facility tech spec identified as part of program surveillance procedure verification.Corrective actions include testing & procedure changes.W/910829 ltr.

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WASHINGTON PUBLIC POWER SUPPLY SYSTEM

P.O. Box 968 • 3000 George Washington Way • Richland, Washington 99352

Docket No. 50-397

August 29, 1991
G02-91-159

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

Subject: NUCLEAR PLANT NO. 2
LICENSEE EVENT REPORT NO. 91-013-02

Reference: Supply System Letter G02-91-142, Same Subject, dated July 30, 1991.

Gentleman:

The reference letter discussed the schedule for the submittal of LER 91-013-02 and proposed a submittal date of September 30, 1991. However, an additional item of reportability was discovered on August 1, 1991 causing an earlier date for submittal of the revision.

This revision of the LER also discusses the general root cause and corrective actions being taken in response to the LER as discussed in earlier versions of the document. We do not plan to make any additional revisions to this LER. If additional reportable items are discovered as a result of ongoing reviews we will report them in a separate LER.

One of the items reported in this LER (the fourth item entitled Effluent Monitors) has resulted in a number of the monitors becoming technically inoperable for a period in excess of 30 days. This in itself is reportable per the requirements of Technical Specification paragraphs 3.3.7.11.b and 3.3.7.12.b. This LER revision is intended to satisfy this reportability requirement.

Very truly yours,

LT Harold for
J. W. Baker
WNP-2 Plant Manager

Enclosure:
Licensee Event Report No. 91-013-02

cc: Mr. John B. Martin, NRC - Region V
Mr. C. Sorensen, NRC Resident Inspector (M/D 901A)
INPO Records Center - Atlanta, GA
Ms. Dottie Sherman, ANI
Mr. D. L. Williams, BPA (M/D 399)
NRC Resident Inspector - walk over copy

Ref No P085602788

IE22

NRC FORM 366 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92

ESTIMATED BURDEN PER RESPONSE TO COMPLY WTH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Washington Nuclear Plant - Unit 2	0 5 0 0 0 3 9 7	1 OF 2 7

TITLE (4) Technical Specification - Surveillance Procedure Verification Program Identification of Non-Conforming Conditions

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 NAMES	DOCKET NUMBER(S)
0 5	0 7	9 1	9 1	0 1 3	0 2	0 8	2 9	9 1		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 5

POWER LEVEL (10): 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check one or more of the following)* (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 *(Specify in Abstract below and in Text, NRC Form 366A)*
[ ] 20.405(a)(1)(iii)	[X] 50.73(a)(2)(i)	[ ] 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 (AREA CODE)
Carl L. Fies, Compliance Engineer	5 0 9 3 7 7 - 2 0 3 9

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

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

[ ] YES *(If yes, complete EXPECTED SUBMISSION DATE)* [X] NO

EXPECTED SUBMISSION DATE (15): MONTH | DAY | YEAR

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

On May 7, 1991 the first of several items of non-compliance with the WNP-2 Technical Specification was identified as part of a program of Surveillance Procedure verification. This Surveillance Procedure Verification Program was initiated by the Supply System as a result of problems found in recent months. A total of approximately 145 potential deficiencies were identified by contract Engineers and this effort is complete. A further evaluation by the Plant Staff is being performed on each potential problem item to identify the validity and necessary follow-up actions. A total of twelve items were identified as reportable problems by this process when the time cutoff date for this LER revision occurred. The twelve items cover a variety of subjects involving items of non-compliance. Some of the items are issues where compliance is not possible and a Technical Specification Change or other relief will be requested.

Immediate and further corrective actions include additional testing, Plant Procedure changes, requests for Technical Specification changes or other relief, and possible design changes. The general corrective action is to charter a Quality Action Team (QAT), the Supply System's formal problem solving program, to evaluate and make recommendations to Plant Management for long term improvements to the Technical Specification Surveillance Program.

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

Abstract (continued)

The root causes for these events include less than adequate barriers and controls for program changes and less than adequate test procedures, directives/requirements, and design. The general root cause has been determined to be less than adequate management control of the surveillance program.

A contributing factor for a number of these issues in the total population of 145 is a heightened expectation to resolve any discrepancy between the wording of the Technical Specification requirements and the actual means of accomplishing the requirement.

The safety significance of each item and the overall program was evaluated and it has been concluded the events posed no threat to the health and safety of either the public or plant personnel.

Plant Conditions

Power Level -0 %
Plant Mode - 5

General Event Description

On May 7, 1991 the first of several items of non-compliance with the WNP-2 Technical Specification was identified as part of a program of Surveillance Procedure verification. This Surveillance Procedure Verification Program was initiated by the Supply System as a result of problems found in recent months (see Similar Events below). The goal of the effort was to assure:

1. That each Technical Specification Surveillance Requirement is addressed in one or more plant procedures that implements the surveillance requirements.

2. That the plant procedures utilized to meet the surveillance requirements properly reflect the frequency requirements. Further, the review was to assure that conditional requirements for performance of surveillance activities were identified in an appropriate initiating procedure.

3. That any special equipment configurations or conditions which must be met during surveillance testing, and were specifically identified in the Technical Specifications, were properly included in the appropriate procedures. For example "both flow control valves are in the same position".

4. That each procedure contains a specific set of acceptance criteria which clearly indicate to the performer that the surveillance requirements have been met.

5. That the Scheduled Maintenance System (SMS) data base (Technical Specification Surveillance Listing) contains each surveillance requirement and denotes the applicable MODE and frequency (if applicable) as well as the implementing plant procedure.

   A total of 145 potential deficiencies were identified by contract Engineers and this effort is complete. These potential problems were then assigned to Plant Staff Engineers for further evaluation. Plant Staff Engineers evaluated each potential problem for validity and, where necessary, wrote a Problem Evaluation Request (PER). This made the problem visible to management and allowed resources to be assigned to fix the problem and for the formal evaluation of the item for reportability. Some issues require further evaluation. This part of the effort is in the final stages of review but has not yet been completed. A total of twelve items were identified as reportable problems by this process. Any additional reportable items will be included in a follow-up LER. In the event description below each reportable condition is described in a separately numbered paragraph.

   This LER is written with each item discussed as a separately numbered paragraph under the major headings of Specific Event Description, Immediate Corrective Action, Further Evaluation, Specific Further Corrective Action, and Specific Safety Significance. A general discussion of all items is found under General Event Description above and General Further Corrective Actions, General Safety Significance, and Similar Events below.

<u>Specific Event Description</u>

1. <u>Response Time Testing</u> - The first problem discovered by the program concerns the Response Time Test Requirements for Instruments associated with the Reactor Protection System (RPS) (Technical Specification Table 3.3.1-2), Isolation Actuation (Table 3.3.2-3), and Emergency Core Cooling Systems (ECCS) (Table 3.3.3-3). The Technical Specification requirement associated with these three tables each state, "Each test shall include at least one channel per trip system such that all channels are tested at least once every N times 18 months, where N is the total number of redundant channels in a specific isolation trip system" (Paragraphs 4.3.1.3, 4.3.2.3, and 4.3.3.3).

   For the majority of the trip functions there are two trip systems with 2 redundant channels in each trip system. For this condition all four channels are to be tested in 36 months [(N=2)X18]. A review of the present testing schedule shows the channels are tested on a 12 month frequency resulting in all four channels being tested in 48 months which does not meet the Technical Specification requirement.

   An additional problem was identified with the phrase, "Each test shall include at least one channel per trip system....". In some cases surveillance testing was performed on channels A and B or on Channels C and D which does not meet Technical Specification Requirements.

2. SLC Functional Test - The second item discovered was an error in procedure PPM 7.4.1.5, Standby Liquid Control (SLC) Injection Functional Test. This test is used to verify Reactor Water Cleanup (RWCU) Valve 4 (RWCU-V-4) isolation as part of SLC initiation. This is a required Technical Specification CHANNEL FUNCTIONAL TEST to be performed at least every 18 months per Table 4.3.2.1-1 (Item 3.f). There are two channels (corresponding to the two SLC trains) feeding the trip system to close RWCU-V-4 and only one was being tested each year (every 12 months) rather than both in 18 months as required by the Technical Specifications.

3. Sensor Channel Calibration - This item documented the failure to implement literal compliance with the Technical Specification definition of CHANNEL CALIBRATION for items such as thermocouples, resistance temperature detectors, orifices, pitot tubes, etc. The definition of CHANNEL CALIBRATION states it shall, "....encompass the entire channel including the sensor....". This cannot be done for items such as those specified on Technical Specification Table 4.3.2.1-1 Items 1.d, 1.e, 3.b, 3.c, and many more involving temperature sensing and other loops.

4. Effluent Monitors - This concerns the Technical Specification Surveillance requirements for radioactive gaseous and liquid effluent monitoring. For ease of description this item is separated into four parts as noted below:

   a. This part concerns the CHANNEL FUNCTIONAL TEST performed on the Low Range Noble Gas Activity Monitors (REA-RIS-19, TEA-RIS-13, and WEA-RIS-14). These monitoring systems measure the radioactivity in the reactor building, turbine building and radwaste building exhausts prior to discharge to the atmosphere. They provide no automatic mitigation functions. The quarterly surveillance requirements are listed in Technical Specification Table 4.3.7.12-1 and the attendant Table Notations (1) on p.3/4 3-95. Table Notation (1)b requires demonstrating that control room alarm annunciation occurs if a circuit failure is present. This action was removed in a previous revision to the surveillances in August 1988. In addition, Table Notation (1)c requires demonstrating that control room alarm annunciation occurs if the "instrument controls not set in operate mode". These monitors are KAMAN Science model 952279 Log Count Rate Meters (LCRM). These are not provided with an operate mode switch and therefore cannot successfully meet this requirement.

b. This part concerns the CHANNEL FUNCTIONAL TEST performed on the Liquid Radwaste Effluent Line Gross Radioactivity Monitor (FDR-RIS-606). This monitoring instrument measures the radioactivity in the radwaste liquid effluent discharge prior to its entering the cooling tower blowdown line. It provides automatic closure of the radwaste system discharge valve. The quarterly surveillance requirements are listed in Technical Specification Table 4.3.7.11-1, Item 1.a, and the attendant Table Notation (1) on p.3/4 3-88. The Technical Specification requires demonstrating that, "...automatic isolation of this pathway and control room alarm annunciation occurs...." for any of the listed conditions (measured levels above alarm/trip setpoints, high voltage abnormally low, when the instrument indicates downscale failure, and when the instrument controls are not set in the operate mode). The presently installed instrument provides automatic isolation of the discharge pathway only on measuring radiation levels above the trip setpoint. The requirement to provide isolation on high voltage abnormally low, when the instrument indicates downscale failure and when the instrument controls are not set in the operate mode cannot be met with our current design.

c. This part concerns the CHANNEL FUNCTIONAL TEST performed on the Turbine Building Sump Gross Radioactivity Monitors (FD-RIS-1, 2, and 3). This system monitors the three nonradioactive drain sumps in the turbine building. The normal effluent discharge path is via the storm drain system to an evaporative basin. In the event of any radioactive liquid system failure contaminating these sumps to a level above the instrument setpoint, the monitor would automatically divert the effluent to the radwaste system for processing. The trip setpoints are set such that diversion is initiated prior to exceeding technical specification limits for liquid effluent. The quarterly surveillance requirements are listed in Technical Specification Table 4.3.7.11-1, Item 1.b and the attendant Table Notation (5) on p.3/4 3-88. The Technical Specification requires demonstrating that, "...automatic isolation of this pathway and control room alarm annunciation occurs...." for any of the listed conditions (measured levels above alarm/trip setpoints; high voltage abnormally low; and when the instrument indicates downscale failure). The presently installed instrument provides automatic isolation of the discharge pathway only on measuring radiation levels above the trip setpoint. The requirement to provide isolation on high voltage abnormally low and when the instrument indicates downscale failure cannot be met with our current design.

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

d. This part concerns the CHANNEL FUNCTIONAL TEST performed on the Turbine Service Water System Effluent Line Gross Radioactivity Monitor (TSW-RIS-5). This monitor measures the radioactivity level of the service water returning flow to the circulating water system. This monitor provides no automatic mitigation function. The quarterly surveillance requirements are listed in Technical Specification Table 4.3.7.11-1, Item 2.a and the attendant Table Notation (2) on p.3/4 3-88. Table Notation (2) Item 3 requires demonstrating that control room alarm annunciation occurs if the "instrument controls not set in operate mode". This monitor is a KAMAN Science model 952279 Log Count Rate Meter (LCRM). This monitor is not provided with adjustable control switch, an operate mode switch and therefore cannot successfully meet this requirement.

5. Primary Containment Valve Position - Technical Specification Table 4.3.7.5-1, Item 27 requires "Primary Containment Valve Position" to have a CHANNEL CHECK monthly and a CHANNEL CALIBRATION at least once per 18 months. Two problems were discovered in the surveillance program relative to this requirement. First, Primary Containment valves associated with the Reactor Recirculation System Flow Control Valve Hydraulic Lines (HY-V-17A, -17B, -18A, -19A, -19B, -20A, - 20B, -33A, -33B, -34A, -34B, -35A, -35B, -36A, and -36B) were not having a monthly CHANNEL CHECK performed. Second, for the valves that were being tested the frequency of testing was per the requirements of the ASME Pump and Valve Test Program under Technical Specification 4.0.5 which requires a Valve Position Indication (VPI) verification or equivalent every two years. While this was being performed, it does not meet the 18 month frequency requirements of Technical Specification Table 4.3.7.5-1, Item 27, which requires a CHANNEL CALIBRATION at least once per 18 months.

6. Emergency Ventilation Damper Position - Technical Specification Table 4.3.7.5-1, item 25 requires the "Emergency Ventilation Damper Position" to have a CHANNEL CHECK monthly and a CHANNEL CALIBRATION at least one per 18 months. There were two problems associated with implementation of this requirement:

a. A CHANNEL CHECK was not being performed on the equipment required. FSAR Paragraph 7.5.1.16, Emergency Ventilation Damper Position Indication, states, "Damper position indication is provided in the control room for all dampers necessary to prevent release of radioactive gases to the environment or for the protection of operating personnel during accident conditions." PPM 7.4.3.7.5.1, Accident Monitoring Instrumentation-Channel Checks, listed 14 components where a monthly CHANNEL CHECK is done to this requirement. In addition, PPM 7.4.7.2.1, Control Room Emergency Filtration System "A" Operability and PPM 7.4.6.5.3.1, SGT System Operability Test performs a CHANNEL CHECK on additional components. A total of four dampers (SGT-V-4A1, 4A2, 4B1, and 4B2) were identified where a CHANNEL CHECK was not being performed per the requirements of the Technical Specifications. These dampers are open when SGT is running in the recirculation mode and is discharging into the Reactor Building.

b) A CHANNEL CALIBRATION per the requirements of the Technical Specifications was not being performed on any Emergency Ventilation Damper Position Indication items.

7. Balance of Plant Isolation - Technical Specification surveillance requirement 4.6.3.2 states that, "Each primary containment automatic isolation valve shown in Table 3.6.3-1 shall be demonstrated OPERABLE during COLD SHUTDOWN or REFUELING at least once per 18 months by verifying that on a containment isolation test signal each automatic isolation valve actuates to its isolation position". This requirement is implemented by Plant Procedure PPM 7.4.3.2.2.11, Balance of Plant Logic System Functional Test. A review of this procedure showed that three valves associated with containment exhaust and supply purging (CEP-V-1A, CEP-V-1B, and CSP-V-1) were inadvertently omitted from the procedure in 1987. Thus, the surveillance requirement was not demonstrated for these valves during refueling in 1988, 1989 and 1990.

8. Containment Purge Butterfly Valves - Technical Specification 4.6.1.8.1 states, "When being opened, the drywell and suppression chamber purge supply and exhaust butterfly isolation valves shall be verified to be blocked so as to open to less than or equal to 70 percent open, unless so verified within the previous 31 days." This verification was not being performed. This requirement applies to eight valves at WNP-2: Containment Supply Purge Valves CSP-V-1, - 2, -3, and -4 and Containment Exhaust Purge Valves CEP-V-1A, -2A, -3A, and -4A. The supply and exhaust to the drywell (CSP-V-1 and 2 and CEP-V-1A and 2A) are 30-inch diameter valves and the remaining valves that provide supply and exhaust to the wetwell are 24-inch diameter valves.

9. Offgas Hydrogen Monitor - Technical Specification surveillance requirement 4.3.7.12, in conjunction with Table 4.3.7.12-1, requires a CHANNEL CALIBRATION of the Main Condenser Offgas Treatment System Explosive Gas Hydrogen Monitors (OG-AY-12A and OG-AY-12B) at least one per 92 days. Note 3 on Table 4.3.7.12-1 states: "The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal

a. 0.0 volume percent hydrogen, balance nitrogen, and

b. 2.0 volume percent hydrogen, balance nitrogen."

The WNP-2 Hydrogen Monitors were not provided with a 0.0 volume percent hydrogen, balance nitrogen sample. The present surveillance procedure uses instrument air for this calibration and; therefore, did not meet the Technical Specification requirement.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

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

10. MSLC Valve Operability - Technical Specification surveillance requirement 4.0.5 requires inservice testing of ASME code Class 1, 2, and 3 valves including the valves associated with the Main Steam Isolation Valve (MSIV) Leakage Control (MSLC) System. At WNP-2 there are sixteen valves associated with this system (MSLC-V-1A, -3A, -2A, -1B, -3B, -2B, -1C, -3C, -2C, -1D, -3D, -2D, -5, -4, -9, and -10) that function to mitigate any leakage through the MSIVs under postulated accident conditions. These valves are to be tested in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50. The specific implementing requirements are contained in the WNP-2 Pump and Valve Test Program Plan and ASME Section XI, Paragraph IWV-3411, which requires quarterly exercising of the identified MSLC valves. The required Operability Surveillance performed by Plant Procedure PPM 7.4.6.1.4.2, Main Steam Isolation Valve (MSIV) Leakage Control (MSLC) System Surveillance, was being performed at cold shutdown rather than quarterly as required.

11. EOC Recirculation Pump Trip Response Time - Technical Specification surveillance paragraph 4.3.4.2.3 states the requirements for END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME testing. One of the requirements is given as follows: "The time allotted for breaker arc suppression, 83 ms, shall be verified by test at least once per 60 months. The design of the End-of-Cycle (EOC) Recirculation Pump Trip (RPT) logic is shown on FSAR Figure H.3.3-2 (attached). For high speed operation, each Reactor Recirculation Pump (RRC- P-1A and RRC-P-1B) is powered by 6900 volt power through two safety-related breakers. Breakers RPT-3A and RPT-4A provide power to RRC-P-1A and breakers RPT-3B and RPT-4B provide power to RRC-P-1B. The EOC-RPT function is composed of two trip systems (A and B), either of which is capable of tripping both pumps. If the "A" system is initiated breakers RPT-3A and RPT-4B will trip. If the "B" system is initiated breakers RPT-3B and RPT-4A will trip. The Technical Specification Surveillance Procedure, PPM 7.4.3.4.2.3.3, EOC Recirc Pump Trip System Time-Breaker Arc Suppression Time, measured the arc suppression time of breakers RPT-3A and RPT-3B. The arc suppression time for breakers RPT-4A and RPT-4B was not being measured by the surveillance program as required by the Technical Specifications.

12. Electrical Voltage Measurements - This item involves the Electrical Power Onsite Distribution Systems which have surveillance requirements as stated in paragraphs 4.8.3.1 and 4.8.3.2 of the Technical Specifications. This surveillance is concerned with both A.C. and D.C. power distribution during operating and shutdown conditions. The power distribution is defined in each of three divisions for both A.C. and D.C. power. The requirement states that, "Each of the above required power distribution system divisions shall be determined energized at least once per 7 days by verifying correct breaker alignment and voltage on the busses/MCCs/panels." This requirement is implemented in Plant Procedure PPM 7.4.8.3.2, Division 1, 2, and 3 A.C. and D.C. Weekly Breaker Alignment Checks. This procedure verified the "correct voltage" on some equipment such as the 4160 volt buses E-SM-4, E-SM-7, and E-SM-8. However, for the majority of the Busses, Motor Control Centers (MCCs), and Panels listed the correct voltage was not measured weekly as required by the Technical Specifications.

Immediate Corrective Action

Immediate corrective action was initiated for each of the items discovered by the surveillance procedure verification program. They are enumerated below in separately numbered paragraphs corresponding to the event description above:

1. Response Time Testing - Additional surveillance tests were identified which would bring the plant into compliance with Technical Specification Requirements 4.3.1.3, 4.3.2.3 and 4.3.3.3. A total of 26 additional tests will be performed.

2. SLC Functional Test - A procedure change was made to Plant Procedure PPM 7.4.1.5, Standby Liquid Control (SLC) Injection Functional Test, to add the additional contact into the logic chain for RWCU isolation.

3. Sensor Channel Calibration - No immediate corrective action was taken on this item. However, action was assigned to resolve this item with the NRC.

4. Effluent Monitors-The problems with the effluent monitors were reviewed by the Plant Operating Committee (POC) and a determination was made to declare the instruments inoperable. At 1300 hours on May 21, 1991 Technical Specification Action Statement (TSAS) 3.3.7.12.b (110) was entered for the associated Noble Gas Activity Monitors. At the same time TSAS 3.3.7.11.b (100) and (101) were entered for the associated Liquid Effluent Monitors. These action statements require periodic sampling and analysis of the effluent streams. The POC also directed that an evaluation be performed and presented to the NRC to justify exit from the TSAS noted above. As a result of this evaluation, and after consultation with the NRC, it was determined that the plant would remain in the action statement until the relocation of RETS was accomplished in accordance with generic letter 89-01 (See discussion under Further Corrective Action below).

5. Primary Containment Valve Position - A Technical Specification Interpretation (TSI 91-002) was prepared and approved to define the Primary Containment Isolation Valves that require a CHANNEL CHECK and CHANNEL CALIBRATION for position indication.

6. Emergency Ventilation Damper Position - An effort was undertaken to identify all Emergency Ventilation Dampers. A total of thirty dampers were identified that fall under the definition in the Technical Specifications and the FSAR. This includes twenty dampers associated with Secondary Containment and ten associated with Control Room Ventilation. Plant Procedure, PPM 7.4.3.7.5.1, Accident Monitoring Instrumentation-Channel Checks, was modified to perform additional channel checks on dampers (SGT-V-4A1, 4A2, 4B1, and 4B2). A new Plant Procedure PPM 7.4.3.7.5.1A, Emergency Ventilation Damper Channel Calibration Test was written to perform the CHANNEL CALIBRATION on the thirty valves.

7. Balance of Plant Isolation - PPM 7.4.3.2.2.11 was modified to add the three valves to the procedure.

8. Containment Purge Butterfly Valves - Plant Procedure PPM 7.4.6.1.1, Primary Containment Integrity Verification, was modified by adding a step to perform the required surveillance.

9. Offgas Hydrogen Monitor - An urgent design change was initiated to make provision for the use of 0.0 volume percent hydrogen, balance nitrogen as a calibration gas.

10. MSLC Valve Operability - There was no immediate corrective action since the plant was in cold shutdown and the Technical Specification Surveillance, PPM 7.4.6.1.4.2, Main Steam Isolation Valve Leakage Control System Surveillance, was performed as previously scheduled.

11. EOC Recirculation Pump Trip Response Time - Surveillance Procedure, PPM 7.4.3.4.2.3.3, EOC Recirc Pump Trip System Time-Breaker Arc Suppression Time, will be revised to require testing of breakers RPT-4A and RPT-4B.

12. Electrical Voltage Measurements - On August 1, 1991 at 1624 hours the Division 1, 2, and 3 Power Distributions Systems were declared technically inoperable and the Technical Specification Action Statements 3.8.3.2a and 3.8.3.2b were entered. Immediate corrective action was taken to prepare a general revision to PPM 7.4.8.3.2 to verify correct voltage. A part of this procedure revision included the acceptance criteria for the voltage measurements.

Further Evaluation and Corrective Action

A. Specific Further Evaluation

Further evaluations were performed on each of the items discovered by the surveillance procedure verification program. They are enumerated below in separately numbered paragraphs corresponding to the event description above:

1. Response Time Testing - Further evaluation indicates the requirements of this Technical Specification were never fully implemented. For example, the Nuclear Steam Supply Shutoff System (NSSSS) trip function of High Drywell Pressure was reviewed. A test was performed on all channels during the startup period in 1984. However, revision 0 to Plant Procedure PPM 7.4.3.2.3.3, Primary Containment Isolation On High Drywell Pressure (Channel A)-RTT, states under the purpose (paragraph 7.4.3.2.3.3.1), ".....this test for trip Channel A is to be performed during the first refueling cycle and at each 36 month interval thereafter." Revision 0 to PPM 7.4.3.2.3.32, Primary Containment Isolation on High Drywell Pressure (Channel B) - RTT, states, "....this test for trip Channel B is to be performed with Channel A and at each 36 month interval thereafter." These statements do no meet the Technical Specification requirements. This was verified by a review of testing records which showed that both Channels B and C for this trip function were not tested in 1985. Thus, the requirement that each test include one channel per trip system every 18 months was not met.

In addition, changes were made to the test program in 1987 that allowed 48 months between channel testing. This resulted in a violation of the N X 18 requirement in addition to the one channel per trip system per 18 month requirement.

2. <u>SLC Functional Test</u> - Further evaluation of Plant Procedure PPM 7.4.1.5, Standby Liquid Control (SLC) Injection Functional Test, showed this error was not the result of a change but was an oversight in initial procedure preparation.

3. <u>Sensor Channel Calibration</u> - This issue is recognized as common to the industry and is being documented because a literal interpretation of Technical Specifications would mandate it. As a generic issue the Supply System is attempting to determine a resolution consistent with recognized industry practices and implement it accordingly. Subsequent discussion with the NRC concluded that no action was required and the Technical Specifications could remain as written.

4. <u>Effluent Monitors</u> - Further evaluation of the problem with the effluent monitors showed the problems could be divided into three categories:

   a. The Noble Gas Monitors (REA-RIS-19, TEA-RIS-13, and WEA-RIS-14) and the Service Water System Effluent Monitor (TSW-RIS-5) cannot meet the Technical Specification requirement for control room annunciation if "instrument controls not set in operate mode". In this case the inherent features of the instrument do not allow for compliance with the Technical Specifications.

   b. The Noble Gas Monitors (REA-RIS-19, TEA-RIS-13, and WEA-RIS-14) also have the problem of not meeting the requirement to alarm if a circuit failure is present. The procedure that performed this surveillance was changed in August 1988 when the low counts alarm setpoint was being changed. This change was made to prevent frequent downscale alarm conditions.

   c. The Liquid Radwaste Effluent Line Monitor (FDR-RIS-606) and the Turbine Building Sump Gross Radioactivity Monitors (FD-RIS-1, 2, and 3) do not meet the Technical Specification Requirement to provide isolation on high voltage abnormally low and when the instrument indicates downscale failure. In this case the instruments have this feature but the output is fed to an alarm circuit rather than an isolation circuit. In addition, for FDR-RIS-606 the instrument has an operate mode switch. However, this switch provides an alarm only, and does not provide isolation as required by the Technical Specification.

5. Primary Containment Valve Position - Further evaluation showed problems with the surveillance procedures and the Surveillance Monitoring System (SMS). There are 19 PPMs listed in the Scheduled Maintenance System (SMS) that implement this Tech Spec requirement number, 4.3.7.5.27. A review of these procedures found that only 1 out of the 19 referenced the requirement. Thus there is a risk that the procedure could be changed, over time, and the requirement eliminated.

   Further evaluation showed there is no listing of components defined where this requirement applies. This is complicated by the fact that applicability of this requirement to specialty valves such as check valves and squib valve has not been formally documented.

6. Emergency Ventilation Damper Position - A review of surveillance procedures in the SMS data base found they did not reference Tech Spec requirement number 4.3.7.5.25 nor did they perform a CHANNEL CALIBRATION that meets requirements. There were two procedures listed in the SMS data base that implement this requirement. PPM 7.4.6.5.2.1, Reactor Building Ventilation Isolation Valve Operability, and PPM 7.4.6.5.3.4, Standby Gas Treatment System - Manual Initiation, Bypass Damper, and Heater Test. No other surveillance procedures were found that performed the surveillance. The Plant Procedures that performed the CHANNEL CHECKS, PPM 7.4.3.7.5.1, Accident Monitoring Instrumentation-Channel Checks, PPM 7.4.7.2.1, Control Room Emergency Filtration System "A" Operability and PPM 7.4.6.5.3.1, SGT System Operability Test were not listed in the SMS data base.

7. Balance of Plant Isolation - Further evaluation of the event showed revision 2 of PPM 7.4.3.2.2.11, in effect during the 1987 refueling outage, referenced CEP-V-1A, CEP-V-1B, and CSP-V-1. These valves were listed on Attachment A, page 3 of 3, of the procedure which is the data sheet for Relay Cabinet One (RC-1). In September 1987 procedure revision 3, which included the addition of an item on the data sheet, was approved by the Plant Operating Committee. This copy of the procedure continued to list the three CEP and CSP valves. However, when revision 3 of the procedure was signed on September 23, 1987 the valves were not listed on page 3 of 3 on Attachment A. It is assumed they were deleted somewhere in the clerical/administrative process used to prepare procedures for final signature.

8. Containment Purge Butterfly Valves - Further evaluation showed the valves were modified during plant startup in early 1984. At that time a design change (PED S215-M-6924) was implemented to prevent the valves from opening more than 70 degrees. This was accomplished by welding a two-inch stop tube extension on each valve operator to physically prevent the air operator from opening the valve more than the allowable value. Since the modification prevented the valve from opening beyond 70 degrees it was thought that the surveillance was not required.

A review of the Scheduled Maintenance System (SMS) showed this surveillance was identified in the data base. The entry under Technical Specification 4.6.1.8.1 references Plant Procedure PPM 7.2.3.1 with the notation "DW/SPSUPPLY & EXH PURGE VALVE LE 70 DEG OPEN." However, no completed surveillance procedures were found in the Plant files.

9. <u>Offgas Hydrogen Monitor</u> - The difference between the Technical Specification requirement and the Surveillance Procedure, PPM 7.4.3.7.12.23, Off Gas Hydrogen Analyzers A and B Channel Functional Test/Channel Calibration, was discussed with plant personnel familiar with the equipment and the requirement. In the past the use of air for the 0.0 percent calibration was known to be technically correct and representative of a 0.0 percent hydrogen condition. However, at the present time there is a heightened effort to resolve any discrepancy between the wording of the Technical Specification requirements and the actual means of accomplishing the requirement.

10. <u>MSLC Valve Operability</u> - Further evaluation shows a conflict between the Technical Specification 4.0.5 requirements described above and the requirements of paragraph 4.6.1.4.b. The latter requirement states that "Each MSIV leakage control subsystem shall be demonstrated OPERABLE......during each COLD SHUTDOWN, if not performed within the previous 92 days, by cycling each depressurizing valve and steam isolation valve through at least one complete cycle of full travel." Technical Specification 4.0.5 requires a quarterly test.

    Plant Procedure PPM 7.4.6.1.4.2, Main Steam Isolation Valve Leakage Control System Surveillance, had a prerequisite added in revision 4 (in 1987) to require the plant to be in cold shutdown to perform the test. This is in conflict with paragraph 4.1 in the precautions section of the procedure which states: "Individual component testing may be done at any time."

    The FSAR in paragraph 6.7.3.i states: "Components of the system downstream of the MSLC system isolation valves may be tested at any time during plant operation. The isolation valves may also be operated sequentially at any time during plant operation. Simultaneous operation of the isolation valves and leak testing can be performed only during reactor shutdown in order not to interfere with normal plant operation."

    A review of the SMS system showed the "Plant Condition" fields defined as Modes 1 (Power Operation), 2 (Startup), and 3 (Hot Shutdown). According to Plant Procedure PPM 10.1.5, Scheduled Maintenance System, "Plant Condition" signifies when the procedure can be performed. However, the word "shutdown" was present in the "Task ID" in the SMS data base and, in fact, the test was being performed only in cold shutdown.

11. <u>EOC Recirculation Pump Trip Response Time</u> - Surveillance Procedure, PPM 7.4.3.4.2.3.3, EOC Recirc Pump Trip System Time-Breaker Arc Suppression, was originally written and approved in May 1984 during plant startup. The initial procedure did not identify the RPT-4A and RPT-4B breakers.

NRC FORM 366A (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

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12. Electrical Voltage Measurements - Further evaluation found that Surveillance Procedure, PPM 7.4.8.3.2, had not been verifying correct voltage on all required equipment since plant startup. The combination of breaker alignment checks, alarms and indicating lights on various electrical equipment were believed to satisfy the requirement.

A root cause was evaluated for each item discovered by the review.

1. Response Time Testing - This testing was not being done correctly during the startup phase of WNP-2 and later changes were made to decrease the frequency of testing. The root cause was determined to be Less Than Adequate Procedures and Barriers and Controls for Program Changes.

2. SLC Functional Test - The Logic System Functional Test for SLC did not verify the logic for RWCU-V-4 Closure for both SLC trains each year. The root cause was Less Than Adequate Test Procedures.

3. Sensor Channel Calibration - The root cause for the Primary Sensor Calibration on Thermocouples, Orifices, etc. was Less Than Adequate Directives/Requirements since the hardware does not support the requirements.

4. Effluent Monitors - The Radioactive Effluent Monitoring Instrumentation Problems have multiple root causes. These include Less than Adequate Directives/Requirements, Inadequate Barriers and Controls for Program Changes and Less Than Adequate Design.

5. Primary Containment - For the Primary Containment Isolation Valve Position CHANNEL CHECK and CALIBRATION the root cause was Less Than Adequate Procedures.

6. Emergency Ventilation Valve Position - For the Emergency Ventilation Damper Position CHANNEL CHECK and CALIBRATION the root cause was less than adequate procedures.

7. Balance of Plant Isolation - The Three Primary Containment Valves were omitted from the procedure somewhere in the Administrative process. The root cause was Inadequate Barriers and Controls for Program Changes.

8. Containment Purge Butterfly Valves - The root cause for the missing surveillance was Less than Adequate Procedures. A contributing cause was Less Than Adequate Directives/Requirements since the Technical Specifications should have been changed when the design change was made.

9. <u>Offgas Hydrogen Monitor</u> - The root cause for the lack of 0.0 percent hydrogen calibration gas was less than adequate design.

10. <u>MSLC Valve Operability</u> - The root cause associated with this item was Less Than Adequate Directives/Requirements. A contributing cause was Less Than Adequate Procedures and Change Management.

11. <u>EOC Recirculation Pump Trip Response Time</u> - The root cause for the missing surveillance test for breakers RPT-4A and 4B was Less Than Adequate Procedures.

12. <u>Electrical Voltage Measurements</u> - The root cause for the failure to measure correct voltage was Less Than Adequate Procedures. This is another example of the present heightened awareness of the exact wording in the Technical Specifications.

B. <u>General Further Evaluation</u>

These events are being reported per the requirements of 10CFR50.73(50.73(a)(2)(i)(B) as "....Any operation or condition prohibited by the plant's Technical Specifications...."

An additional item of reportability involves the Effluent Monitors as discussed under item 4. After consultation with the NRC, it was decided that the plant would remain in the action statement until the relocation of RETS was accomplished in accordance with generic letter 89-01 (See discussion under Further Corrective Action below). Since the instruments are "technically" inoperable during this time period, which exceeds 30 days this is reportable, under Technical Specification requirement 3.3.7.11.b and 3.3.7.12.b. This LER is intended to satisfy that reportability requirement. It should be noted that the instruments involved are expected to function normally in the plant throughout this period of "technical" inoperability.

The general root cause analysis consisted of a review and categorization of the root causes of the twelve reportable events in this LER, the non-reportable contractor identified deficiencies, and all past technical specification violation LERs. The categorization of root causes result in five generic root causes being identified. The results of the categorization are shown in Table 1. The generic root causes are explained in the text that follows.

Table 1

Root Cause	Reportable Items	Contractor Issues	Previous LERs
Procedures LTA	7	69	26
Change Management	3	8	13
Requirements LTA	3	10	2
Design LTA	2	2	0
Programmatic	All	20	Not Evaluated
Non-Issue	0	36	66

The five general root causes identified by this review are described below:

1. Procedures Less Than Adequate (LTA) - The largest population of root causes was in the area of inadequate procedures that existed from the startup time frame. The surveillance procedures did not fully implement the requirements.

2. Change Management - This category covers items where procedure revisions, procedure deviations or plant changes introduced errors into the Technical Specification Surveillance Program.

3. Requirements LTA - Technical specifications were accepted at the time of startup that could not be complied with because of hardware restraints. These issues were recognized at the time and non-documented verbal agreements were made between WNP-2 and the NRC. Since that time literal compliance has become the standard and we have no basis for not complying.

4. Design LTA - There were two examples where plant design did not meet technical specification surveillance requirements.

5. Programmatic - This is a summation root cause. It cites the fact that Plant Procedure PPM 1.5.1, Technical Specification Surveillance Testing Program, and PPM 10.1.5, Scheduled Maintenance System (SMS), do not provide adequate control of the surveillance program.

In December 1990, the Plant initiated a long term verification and validation program to upgrade Plant procedures. One element of this program ensures the correct implementation of technical specification surveillance requirements.

C. Specific Further Corrective Actions

Further corrective action was initiated for each of the items discovered by the verification program. They are enumerated below in separately numbered paragraphs corresponding to the event description above:

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

1. <u>Response Time Testing</u> - Twenty-six additional surveillance tests have been performed prior to startup to bring the plant into compliance with Technical Specification Requirements 4.3.1.3, 4.3.2.3 and 4.3.3.3.

   A comprehensive Engineering review was performed of the requirements for instrumentation Response Time Testing (RTT). This included a review of the logic involved in each trip system and a definition of the terms used for RTT. This Engineering document was used to review RTT testing done in the past. This review showed that in 1985-86 40 out of 48 required test were performed. During the period from 1987-90 30 out of 48 required tests were performed. In addition, this document was used to establish requirements for RTT in the future.

2. <u>SLC Functional Test</u> - The channel functional test for RWCU-V-4 was performed using modified procedure PPM 7.4.1.5.

3. <u>Sensor Channel Calibration</u> - A Technical Specification change or other relief will be requested. For this item, the only means of resolution would appear to be a change in the wording of the Technical Specification.

4. <u>Effluent Monitors</u> - Further corrective actions for the effluent monitors are being taken consistent with the further evaluation discussed above:

   a. For the Noble Gas Monitors (REA-RIS-19, TEA-RIS-13, and WEA-RIS-14) and the Service Water System Effluent Monitor (TSW-RIS-5) where the inherent features of the instrument do not allow for compliance with the requirements a Technical Specification change or other relief will be requested.

   b. For the Noble Gas Monitors (REA-RIS-19, TEA-RIS-13, and WEA-RIS-14) the CHANNEL FUNCTIONAL TEST Procedures (PPMs 7.4.3.7.12.5, 7.4.3.7.12.11, and 7.4.3.7.12.17) were changed to require verification of the front board annunciator, 4.851.S1, Drop 6.5, BOARD-RAD 24 TROUBLE.

   c. For the Liquid Radwaste Effluent Line Monitor (FDR-RIS- 606) and the Turbine Building Sump Gross Radioactivity Monitors (FD-RIS-1, 2, and 3) a Technical Specification change or other relief will be requested from the requirement to isolate on high voltage abnormally low and when the instrument indicates downscale failure.

   d. For items a. and c. corrective action will be accomplished after RETS is removed from the Technical Specifications in accordance with Generic Letter 89-01. A part of this effort will include the requirements for the Effluent Monitors (items a. and c.) that are being moved to the Offsite Dose Calculation Manual (ODCM). When this is approved, the 10CFR50.59 process will be used to modify the requirements so they match the plant hardware. The 10CFR50.59 review is not expected to result in an Unreviewed Safety Question.

5. Primary Containment Valve Position - Primary Containment valve position indication CHANNEL CALIBRATIONS will be performed on all valves that require testing as defined by Technical Specification Interpretation 91-002 prior to startup from the refueling outage.

   Plant Procedure PPM 7.4.3.7.5.1, Accident Monitoring Instrumentation Channel Checks, will be modified to require a CHANNEL CHECK on the Primary Containment valves associated with the Reactor Recirculation System Flow Control Valve Hydraulic Lines (HY-V-17A, -17B, -18A, -19A, - 19B, -20A, -20B, -33A, -33B, -34A, -34B, -35A, -35B, -36A, and -36B).

   An FSAR change notice will be issued to permanently record the list of valves, and the bases for the list, which fall under this Technical Specification requirement.

6. Emergency Ventilation Damper Position - Additional Emergency Ventilation Damper position indication testing will be performed to bring the station into compliance with the Technical Specification prior to startup from the refueling outage.

7. Balance of Plant Isolation - The three valves CEP-V-1A, CEP-V- 1B, and CSP-V-1 were tested prior to startup from this refueling outage. There were no failures involving these valves during the test.

8. Containment Purge Butterfly Valves - The surveillance will be performed every 31 days in accordance with the requirements of Plant Procedure PPM 7.4.6.1.1, Primary Containment Integrity Verification.

9. Offgas Hydrogen Monitor - The design change (Plant Modification Request 91-220) was implemented to allow the use of 0.0 percent hydrogen gas, balance nitrogen. Plant Procedure, PPM 7.4.3.7.12.23, Off Gas Hydrogen Analyzers A and B Channel Functional Test/Channel Calibration, will be modified to perform the CHANNEL CALIBRATION the with 0.0 percent hydrogen gas, balance nitrogen gas.

10. MSLC Valve Operability - Corrective action will be taken to provide justification for testing valves MSLC-V-3A, -2A, -3B, -2B, -3C, -3D, -2D, -4, -5, -9, and -10 only during cold shutdown. The justification will be contained in the ASME, Section XI, Pump and Valve Test Program Plan. ASME Section XI, IWV-3412 states: "Valves shall be exercised to the position required to fulfill their function unless such operation is not practical during plant operation.....Valves that cannot be exercised during plant operation shall be specifically identified by the Owner and shall be full-stroke exercised during cold shutdowns."

    PPM 7.4.6.1.4.2 will be revised to require testing of MSLC- V-1A, 1B, 1C, and 1D on a quarterly basis. These valves are depressurization valves on the inboard Main Steam Leakage Control System. Since there are two valves in series between these four valves and primary system pressure during plant operation they can be tested at any time.

The FSAR will be changed to allow testing of valves MSLC-V-3A, -2A, -3B, -2B, -3C, -3D, -2D, -4, -5, -9, and -10 only during cold shutdown.

11. EOC Recirculation Pump Trip Response Time - Corrective action will be taken to measure the arc suppression time on breakers RPT-4A and RPT-4B.

12. Electrical Voltage Measurements - An evaluation will be performed to determine how to deal with this requirement over the long term. This will include consideration of design changes to simplify the taking of voltage measurements and consideration of Technical Specification or other relief from the requirement.

D. General Further Corrective Actions

The Supply System recognizes that the number of specific items of non-compliance is symptomatic of a broader programmatic issues. The five general root causes were reviewed to determine general corrective action.

For procedures and change management two actions will be taken:

1. The verification procedure will be revised to strengthen the technical specification surveillances verification process.

2. The Scheduled Maintenance System procedure will be revised to include specific signoffs for SMS changes to technical specification surveillance requirements.

3. For the programmatic root cause: The Plant Technical Specification Surveillance Program will be reviewed by a Quality Action Team, the Supply System formal problem solving process. The QAT will address the specific and general root causes identified in this LER.

No general corrective actions are planned for the requirements and design root causes. These problems occurred prior to Plant startup and are being corrected by the specific corrective actions specified in Section C, Specific Corrective Actons.

Specific Safety Significance

The Safety Significance of each item is discussed below. Individually these items have no safety significance.

1. Response Time Testing - Tests were being performed on all redundant channels and trip systems. There has never been a failure of an instrumentation response time test at WNP-2 which would imply there is no safety significance to this non-compliance item. Additional tests will be completed before plant startup.

2. <u>SLC Functional Test</u> - All logic for isolation of RWCU-V-4 on SLC initiation was being tested on a 24 month frequency versus the Technical Specification requirement of an 18 month frequency. No failures were detected during this testing.

3. <u>Sensor Channel Calibration</u> - The failure of sensors is covered by other regulatory requirements of redundancy and diversity. Therefore, failure of an individual sensor will have no safety significance. There have been no channel calibration failures.

4. <u>Effluent Monitors</u> - The Safety Significance of each part of the item associated with the radioactive gaseous and liquid effluent monitoring is discussed below.

   a. The Low Range Noble Gas Activity Monitors (REA-RIS-19, TEA-RIS-13, and WEA-RIS-14) measure the radioactivity in the reactor building, turbine building and radwaste building exhausts prior to discharge to the atmosphere. If these instruments were to become inoperable the intermediate range monitors would continue to monitor the release from each building at higher levels. These instruments are diverse and not subject to the same failure modes. In addition, a sampling program is in place in accordance with the requirements of Technical Specification 4.11.2.1.2. This program requires sampling on a periodic and event driven basis.

   b. The Liquid Radwaste Effluent Line Gross Radioactivity Monitor (FDR-RIS-606) measures the radioactivity in the radwaste liquid effluent discharge prior to its entering the cooling tower blowdown line. It provides automatic closure of the radwaste system discharge valve. This instrument has a downscale or inoperable alarm function that would notify plant operators in the event of a malfunction. Operator actions a called out in Plant Procedure PPM 4.602.A5, Window 6-6 and include the direction to secure the discharge in the event of an alarm. In addition, if this instrument were to be inoperable other methods are in place to monitor the Liquid Radwaste release. First, keylock valves are in place to assure a release only occurs under controlled conditions. Second, a batch sample is collected and analyzed prior to discharge in accordance with the requirements of Technical Specification 4.11.1.1.1. This is used to project the dose expected from the discharge in accordance with the requirements of 4.11.1.1.2. A discharge will not be allowed unless the samples are in conformance of these requirements.

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c. The Turbine Building Sump Gross Radioactivity Monitors (FD-RIS-1, 2, and 3) monitors the three nonradioactive drain sumps in the turbine building. The normal effluent discharge path is via the storm drain system to an evaporative basin. In the event of excess contamination in these sumps, the monitor would automatically divert the effluent to the radwaste system for processing. The Turbine Building does not normally contain significant sources of radioactive liquids. A composite sampler is installed in each of the monitored non-radioactive drain sumps. As an added precaution monthly samples are taken in the discharge area. In addition, an alarm is provided to alert plant operators in the event of instrument failure. Plant operator actions are given in Plant Procedure PPM 4.001.A18.

d. The Service Water System Effluent Line Gross Radioactivity Monitor (TSW-RIS-5) measures the radioactivity level of the service water returning flow to the circulating water system. This monitor provides no automatic mitigation function. If this instrument were inoperative, however, the monitor in the Circulating Water Blowdown line, CBD-RIS-608, would detect elevated levels of activity and isolate the blowdown. In addition, a monthly sludge sample is taken in the Cooling Tower Basins. These samples detect very low levels of radioactivity which have seasonal variations providing some assurance that significant releases would be detected.

5. <u>Primary Containment Valve Position</u> - A CHANNEL CALIBRATION was being performed on all Primary Containment valves (as defined in Technical Specification Interpretation 91-002) in accordance with the requirements of the ASME Pump and Valve Test Program under Technical Specification 4.0.5. Thus, the CHANNEL CALIBRATION was being performed on a two year frequency rather than the more frequent 18 month frequency. The number of failures detected by the ASME two year Valve Position Indication (VPI) Program is not large making the safety significance of this item small. For example, during this refueling outage there have been three failures discovered on 171 valve tests. There is also only a minor safety significance for the part of this item concerned with the CHANNEL CHECK not being performed on the position indication for the Primary Containment valves associated with the Reactor Recirculation System Flow Control Valve Hydraulic Lines. During LOCA conditions the Flow Control Valves lock in their as-is position and the hydraulic line isolation valves close. The need for position indication on the Hydraulic Line isolation valves during post-accident conditions is very small.

6. Emergency Ventilation Damper Position - For the Emergency Ventilation Damper Position issue, the intent of the CHANNEL CALIBRATION is achieved by performing other surveillance procedures. The twenty dampers associated with Secondary Containment have the following four tests performed that, taken together, verify damper position:

Procedure	Title	Frequency
7.4.6.5.1.2	SGT Functional Test	18 month
7.4.6.5.2.1	Reactor Building Ventilation Isolation Valve Operability	3 month
7.4.6.5.3.1	SGT Operability Test	1 month
7.4.6.5.3.2A,B	SGT Pressure Drop	18 month

The ten Emergency Ventilation Dampers associated with the Control Room also meet the intent of the CHANNEL CALIBRATION by performing other procedures as follows:

Procedure	Title	Frequency
8.3.147	Normal and Remote Isolation Valve Leakage	18 month
7.4.7.2.8	Control Room Vent. Flow Test	18 month
7.4.7.2.3	Control Room Filter Test	18 month

The successful completion of these tests is an indication of the operability of position indication on the Emergency Dampers.

7. Balance of Plant Isolation - For the three Containment Isolation Valves (CSP-V-1, CEP-V-1A, and CEP-V-1B) that were inadvertently dropped from the Logic System Function Test other valves in series (CSP-V-2, CEP-V-2A, and CEP-V-2B) were part of this test. Thus one of the containment valves on each penetration was always tested.

8. Containment Purge Butterfly Valves - The valves in question were modified in 1984. It is physically impossible for the valve operator to open the valves more than 70 degrees. This was accomplished by welding a two-inch stop tube extension on each valve operator to physically prevent the air operator from opening the valve more than the allowable value.

9. Offgas Hydrogen Monitor - Redundant monitors sample and analyze the off-gas process stream. The process sample contains small amounts of noble gases and trace of amounts of un-recombined hydrogen, but it is mostly air. Consequently, the actual process sample normally contains approximately 20 percent oxygen. The presence or absence of oxygen has little or no effect on the hydrogen detector. The instrument air which has been used for calibration contains essentially zero hydrogen and is more representative of the actual operating condition than is the hydrogen/nitrogen gas composition required by the Technical Specifications. For these reasons, we believe the intent of the Technical Specification has been met at all times.

10. MSLC Valve Operability - All valves except MSLC-V-1A, -1B, -1C, and -1D, are located with two valves in series subjected to primary system steam pressure during power operation. Even though the FSAR (section 6.7.3.i) states that these valves can be tested sequentially at any time during plant operation, exercising these valves at power can challenge valve integrity by subjecting the valves to full differential pressure (1000 psig). There are pressure interlocks to prevent system alignment when steam line pressure is 5 psig or reactor pressure is greater than 35 psig. However, in the case of a leak in one of the valves in series during the exercise of the other valve, the low pressure side of the system would be subjected to overpressure. This is considered an unnecessary risk and formed the basis for not testing these valves during power operation.

    Pages 4.4-52 through 4.4-54a of the Pump and Valve Test Program Plan describes the current exceptions for valve test frequency. Several components are justified exceptions for similar reasons to those given above for the MSLC valves.

11. EOC Recirculation Pump Trip Response Time - A review was performed of routine maintenance testing on high voltage circuit breakers RPT-4A and RPT-4B. This review showed that Plant Procedure PPM 10.25.13, Westinghouse High Voltage Circuit Breakers, was used along with the TR-2 Motion Analyzer to record the time required to open the breaker contacts. In 1987 tests were performed on breaker RPT-4A and RPT-4B which measured the elapsed time from when the breaker main contacts first open to the full open position. The elapsed time was approximately 56 ms for RPT-4A and 58 ms for RPT-4B. While this is a different method of testing than that referenced in PPM 7.4.3.4.2.3.3, EOC Recirc Pump Trip System Time-Breaker Arc Suppression, and does not meet the specific requirement of the Technical Specifications, it is indicative of an arc suppression time of less than 83 ms.

12. Electrical Voltage Measurements - The safety related Electrical Power Distribution busses, motor control centers, and panels are, in general, well instrumented. Indicating lights associated with the equipment powered from Electrical Power Distribution would provide an indication of loss of voltage. In addition, numerous Bypassed and Inoperable Status Indicators (BISI) annunciators monitor the status of electrical power. Plant operators would, therefor, be aware of abnormal conditions in the Electrical Power Distribution System.

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

For the A.C. Electrical Power Distribution System action was taken in 1989 to install additional voltage regulators on the supply side of nine 120 volt power panels. This new feature maintained voltage within the rated operability range of all components supplied down to the operating point of the degraded voltage transfer relays.

General Safety Significance

Since a number of non-compliance items were discovered as a result of the Surveillance Program verification an evaluation of the general safety significance is in order. Each individual item appears to be backed up by other equipment and/or procedures. In several cases the frequency of testing was not in compliance with the Technical Specifications but periodic tests were being performed.

We do, however, regard the programmatic aspects of this item as an important issue that has potential safety significance. General corrective actions are defined to prevent recurrence of this problem in the future (see General Further Corrective Actions above).

Similar Events

During the period from 1984 to 1989 there were instances of reported events concerning non-compliance with the Technical Specifications. However during 1990 and early 1991 these events appeared to increase in frequency, due in part, to an on-going Technical Specification Improvement Program at WNP-2. LER 90-007 reported that there was no Plant procedure requirement to check for water in the Diesel Generator Fuel Oil Day Tanks. LER 90-008, reported Plant procedures had not required one portion of the Triaxial Response Spectrum Recorder Transmitter to be calibrated or tested as required by the Technical Specifications. LER 90-027 reported the failure to sample diesel generator fuel oil for sulfur. LER 91-002 reported on the condition where the Reactor Recirculation System Jet Pump Operability Testing was not in Literal Compliance with the Technical Specifications due to inadequate procedures. LER 91-003 reported on inadequate Technical Specification Surveillance Testing of the Standby Gas Treatment HEPA filters. LER 91-003 will be revised to address.a similar deficiency in Charcoal filter testing when the root cause analysis is complete. LER 91-005 reported a condition where the oxygen concentration in the Wetwell was not being monitored once per seven (7) days as required by the Technical Specifications. LER 91-008 reported a condition where mode 5 (refueling) was entered with the reactor coolant temperature above the Technical Specification value of 140 degrees F. This requirement was not in a Plant Procedure. As a result of the recent trend in Technical Specification Non-Compliance events the Supply System hired an outside contractor to review the Technical Specification surveillance requirements against Plant procedures. This LER is a direct result of the contractor review.

NRC FORM 366A
(6-89)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

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

Text Reference	EIIS Reference System	Component
Reactor Protection System (RPS)	JC	--
Emergency Core Cooling Systems (ECCS)	BM,BO,BG	--
Standby Liquid Control (SLC)	BR	--
Reactor Water Cleanup (RWCU)	CE	--
RWCU Valve 4 (RWCU-V-4)	CE	V
Low Range Noble Gas Activity Monitors REA-RIS-19, TEA-RIS-13 & WEA-RIS-14	IL	RA
Liquid Radwaste Effluent Line Gross Radioactivity Monitor EDR-RIS-606	IL	RA
Turbine Building Sump Monitors (FD-RIS-1, 2, &3)	IL	RA
Turbine Service Water Effluent Line Monitor (TSW-RIS-5)	IL	RA
Primary Containment	NH	-
Containment Exhaust Purge Valves 1A & 1B (CEP-V-1A, 1B)	VB	V
Containment Supply Purge Valve 1 (CSP-V-1)	VB	V
Nuclear Steam Supply Shutoff System (NSSSS)	BD	-
Standby Gas Treatment (SGT)	BH	-
SGT-V-4A1, 4A2, 4B1, 4B2	BH	V
Containment Exhaust Purge Valves 2A & 2B (CEP-V-2A, 2B)	VB	V
Containment Supply Purge Valve 2 (CSP-V-2)	VB	V

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

Text Reference	EIIS Reference System	Component
Secondary Containment	BH	-
Control Room Ventilation	VH	-
Reactor Recirculation (RRC) System	AD	-
RRC Hydraulic (HY) Valves	AD	ISV
HY-V-17A, 17B, 18A, 19A, 19B	AD	ISV
HY-V-20A, 20B, 33A, 33B, 34A	AD	ISV
HY-V-34B, 35A, 35B, 36A, 36B	AD	ISV
Containment Supply Purge (CSP) System	VB	-
CSP-V-1, 2, 3, 4	VB	V
Containment Exhaust Purge (CEP) System	VB	-
CEP-V-1A, 2A, 3A, 4A	VB	-
Offgas (OG) Treatment System	WF	-
Hydrogen Monitor OG-AY-12A	WF	45
Hydrogen Monitor OG-AY-12B	WF	45
Main Steam Leakage Control (MSLC) System	SB	-
MSLC-V-1A, 3A, 2A, 1B	SB	ISV
MSLC-V-3B, 2B, 1C, 3C	SB	ISV
MSLC-V-2C, 1D, 3D, 2D	SB	ISV
MSLC-V-5, 4, 9, 10	SB	ISV
Pumps RRC-P-1A, 1B	AD	P
Breakers RRC-RPT-3A, 3B	AD	BKR
Breakers RRC-RPT-4A, 4B	AD	BKR

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

Text Reference	EIIS Reference System	EIIS Reference Component
Electrical Power Onsite Distribution System	EB, EI	-
4160 Volt bus, E-SM-4	EB	BV
4160 Volt bus, E-SM-7	EB	BV
4160 Volt bus, E-SM-8	EB	BV