NRC FORM 366 (4-95)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104

EXPIRES 04/30/98

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

(See reverse for required number of

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Millstone Nuclear Power Station Unit 2	05000336	1 OF 11

TITLE (4)

Technical Specification Violations

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
									FACILITY NAME	DOCKET NUMBER
06	02	97	97	-- 022 --	03	5	19	98		

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

OPERATING MODE (9)	N		
POWER LEVEL (10)	000		

	20.2201(b)		20.2203(a)(2)(v)	X	50.73(a)(2)(i)	50.73(a)(2)(viii)
	20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)	50.73(a)(2)(x)
	20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)	73.71
	20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)	OTHER
	20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)	Specify in Abstract below in NRC Form 366A
	20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
R. G. Joshi, MP2 Regulatory Compliance Manager	(860) 440-2080

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)			EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

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

This LER identifies those issues in which a violation of the unit's Technical Specifications (TS) occurred. A detailed review of the unit's TS is ongoing and is expected to identify additional issues. Each of the following TS violations has been characterized as a deficiency which is historical in nature and in which the safety significance of the deficiency is low.

The cause of these conditions is a failure to achieve compliance to Technical Specification Requirements.

A TS change will be initiated or appropriate Surveillance Procedures will be revised. Technical Specification surveillance procedures will be reviewed to ensure compliance with Technical Specifications surveillance requirements as part of Millstone Unit 2 Operational Readiness Plan (Reference NNECO Commitment No. B16076-2).

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Millstone Nuclear Power Station Unit 2	05000336	97	-- 022 --	03	2 OF 11

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

I. Description of Event

This LER identifies those issues in which a violation of the unit's Technical Specifications (TS) occurred. A detailed review of the unit's TS is ongoing and is expected to identify additional issues which may be reported periodically in LER supplements. Each of the following TS violations has been characterized as a deficiency which is historical in nature and in which the safety significance of the deficiency is low. At the time of discovery, the unit was defueled.

A. On June 2, 1997, during a review of the unit's TS, as part of the 10 CFR 50.54(f) project, it was discovered that the Containment Spray System (CS) [BE] TS 3.6.2.1 and the Containment Air Recirculation System (CAR) [BK] TS 3.6.2.2 do not address the condition when two CAR fans and coolers and a CS train are inoperable. This condition would occur when the associated Reactor Building Closed Cooling Water (RBCCW) [CC] loop, which provides cooling water to both of these containment systems, was inoperable. The inoperability of the RBCCW loop would occur when the associated Service Water (SW) [KE] train is declared inoperable for system repairs, or for the purpose of performing SW surveillance testing.

Research of the unit's records found that on May 22, 1984, at 1645 hours, with the unit in Mode 1, at 100% power, the SW train was removed from service to repair a pipe leak. The SW train was declared inoperable, resulting in the need to declare the RBCCW loop A, the A & B CAR fans and coolers, and the CS train A inoperable. Action statements for the SW system TS 3.7.4.1 and Diesel Generator TS 3.8.1.1 were entered. However, the CAR fans and coolers and CS train were not declared inoperable. Without the appropriate action statement in the unit's TS for this situation, TS 3.0.3 should have been entered and action should have been initiated within one hour to place the unit in Hot Standby within the next 6 hours, and Hot Shutdown in the following 6 hours. Since TS 3.0.3 was not entered and the required action was not initiated, TS 3.0.3 was violated. The SW repairs were completed and the train was returned to service the same day. On January 26, 1996 another event occurred when a Surveillance Procedure was performed on the SW system. Entry into the action statement for TS 3.7.4.1 was made, however, the cascading inoperability affect was not considered, and TS 3.0.3 was not entered. The SW loop was inoperable for 26 minutes.

B. On May 5, 1997, while assessing the E-bar surveillance requirements for the current Mid-Cycle 13 outage, it was questioned whether TS 3.4.8, Reactor Coolant System (RCS)[AB] - Specific Activity Surveillance requirements for E-bar had been met during Refuel Outage 12 (RFO12). A review of completed surveillance procedures found E-bar analysis had not been performed prior to entering Mode 5 on June 4, 1995, and greater than six months had elapsed since the previous E-bar analysis had been performed on September 26, 1994. An E-bar analysis was performed on September 25, 1995 following a return to full power operation.

The E-bar surveillance procedure performed on March 21, 1995, included a prerequisite, "...sample to be taken after a minimum of 24 days of 100% power operation...". The completed surveillance was annotated, "...use E-bar and 100/E-bar from 9/26/94 SP until representative data is obtained." No RCS sample was taken for measurement and analysis as required by TS prior to returning to Mode 5. TS 3.4.8 is not applicable in Mode 6. The prerequisite of "24 Days of 100% power operation" ensures radioactive isotopes are at equilibrium and representative of power operation conditions. The provisions of TS 4.0.4 were violated in RFO12 with entry into Mode 5 and power operation without meeting the applicable E-bar surveillance requirements. Additional review determined TS 4.0.4 had similarly been violated during RFO11. Therefore, this condition is a violation of TS 4.0.4.

A review of the E-bar surveillance procedure history identified the following conditions: Revisions 0 through 4 of the applicable surveillance procedure (effective 2/14/77 through 2/14/97) did not identify the modes in which TS 3.4.8 is applicable. The associated surveillance form incorrectly identified the TS applicable modes. Procedure prerequisites for revisions 2, 3, and 4 (effective 11/22/85 through 2/14/97) required the E-bar sample to be obtained during Mode 1 power operation.

C. On June 10, 1997, during an ongoing self assessment of gaseous radioactive effluents, it was discovered that the surveillance procedure for determining quarterly off-site air dose due to noble gas effluents did not include effluents from the containment purges [VA] and the waste gas decay tanks [WE] as required by TS Surveillance Requirement 4.11.2.2.1. TS 6.15 establishes the requirements for the Radiological Effluent Monitoring and Off-Site Dose Calculation Manual (REMODCM). TS Surveillance 4.11.2.2.1 requires the quarterly and yearly cumulative air dose contributions to be determined in accordance with Section II of the REMODCM once every 31 days. This requirement was violated.

D. On July 1, 1997, as part of the facility 10 CFR 50.54(f) project, it was discovered that a Fire Protection System (FPS) [KP] Outside Containment Header Isolation Valve (2-FIRE-108) [ISV], was not listed on the Surveillance Procedure which cycles FPS valves that are not testable during plant operation. Technical Requirements Manual (TRM) B.4.1.e.2 requires that each valve in the flow path that is not testable during plant operation be tested through at least one complete cycle of full travel at least once every 18 months. This surveillance requirement (SR) was moved to the facility TRM by Amendment 191 (November 3, 1995). Prior to Amendment 191, this requirement was part of TS 4.7.9.1.1.e.2. Thus, the failure to cycle this valve did not satisfy the facility TS SR.

E. On July 24, 1997 it was discovered that Iodine (I)-131 and I-133 releases from the Steam Generator (SG) blowdown vent pathway are not considered in the quarterly offsite dose calculation due to radioiodines. Section II of the REMODCM provides the method for determining quarterly dose due to iodines and particulates. Method 1, used to satisfy TS SR 4.11.2.3.1, requires the inclusion of gaseous effluents of I-131 and Iodine-133, but does not identify which effluents should be considered. Iodine releases due to SG blowdown venting have never been considered in the quarterly dose calculations. TS SR 4.11.2.3.1 requires the quarterly and yearly cumulative dose contributions be determined in accordance with Section II of the REMODCM once every 31 days. This requirement was not met.

F. On July 28, 1997 it was discovered that the requirements of TS SR 4.1.1.1.2 were not being met. This SR requires that predicted reactivity values for boron concentrations be normalized to corresponding actual reactor [RCT] core conditions prior to a fuel burn-up of 60 Effective Full Power Days (EFPDs) after each refueling. The associated surveillance procedure to meet this SR states that the normalization calculation can be performed as desired, and is not required. Historically, the actual values have not deviated significantly from the predicted values (~20 to 30 ppm) and therefore, this calculation typically has not been performed.

G. On July 29, 1997 with the plant defueled, the current scope of valves cycled as required by TS 4.5.2.a.6 to verify automatic valve operation for the Emergency Core Cooling System (ECCS) Charging subsystem [CB] flowpath was found to be incomplete. The current surveillance tests the process flow valves in-line with the flowpath on a monthly basis, but does not cycle six other valves in the system that also receive a Safety Injection Actuation Signal (SIAS) signal and act to isolate portions of the Chemical and Volume Control System (CVCS) [CB] from the flowpath. The ECCS Charging subsystems consist of the Boric Acid Storage Tanks (BASTs), through the gravity feed valves or Boric Acid Pumps [P] to the Charging Pump suction. These six valves do not control process flow in the flowpath, but act to create a boundary for the flowpath.

H. On August 2, 1997, during the Configuration Management Program review of the fire protection system, it was discovered that the ASME Class 2 portion of the non code Fire Protection system piping in a containment penetration (between the inboard and outboard containment isolation valves) was not Code System Pressure Tested as required by Technical Specification Surveillance Requirement 4.0.5. The penetration has been tested with the Appendix J test for the penetration using air at 54 psig. However, the ISI Code System Pressure Test using water at normal system operating pressure and temperature had not been performed.

I. On September 9, 1997, during a review of TS 3.1.3.4, Control Element Assembly (CEA) [AA] drop times, it was found that the LCO is applicable in MODE 3, but the Surveillance Requirement 4.1.3.4 can not be performed until after entering MODE 3. This is in conflict with Surveillance Requirement 4.0.4 which requires that Surveillance Requirements applicable to that MODE be performed before entering that MODE. Therefore, Surveillance Requirement 4.0.4 has been violated at the start of each operating cycle.

J. On August 13, 1997, during a review of TS Surveillance Requirements, a discrepancy was discovered between the actual pressurizer [AB] auxiliary spray temperature and the instrumentation used to indicate that temperature. TS 3.4.9.2c states, "The Pressurizer temperature shall be limited to: A maximum spray water temperature differential of 350°F "The temperature instrument used to determine the spray water temperature is located on the charging line downstream of the regenerative heat exchanger, prior to the branch connection to the auxiliary spray line. During operation the charging line has flow. However, the auxiliary spray line has approximately 187 feet of pipe, containing approximately 22 gallons of stagnant water. With no flow in the auxiliary spray line the actual water temperature would be the same as the containment ambient temperature. A review of past auxiliary spray actuations during the time period from 1981 through 1994 shows that on several occasions the TS 350 degree differential temperature limit was exceeded, assuming the spray water temperature was at the containment ambient temperature of 100 degrees.

K. On October 2, 1997, during a walkdown of the Emergency Diesel Generator (EDG) fuel system piping, the seismic qualification of the "A" and "B" EDG fuel day tank sight glass piping was questioned. Initially, the preliminary seismic analysis indicated the stresses were higher than Code allowable in some locations including the threaded connection to the sight glass. However, it was concluded the pipes remained operable since the calculated stresses were within the operability limits. Subsequent walkdowns identified geometric differences between the "A" and "B" EDG sight glass piping. Reanalysis concluded the "B" EDG piping was operable but the "A" piping was inoperable when calculated stresses were compared to the operability limits. Historically, the "B" EDG has been inoperable for maintenance or other reasons, and the "A" EDG was considered operable to meet the TS LCO 3.8.1.1.b or 3.8.1.2.b requirements. During these times, both EDGs should have been declared inoperable and the appropriate actions should have been taken.

L. On November 11, 1997, during a review for development of a TS change on a related issue, it was discovered that the existing surveillance procedure for verifying primary containment integrity does not include verification of the position of four containment purge isolation valves on a monthly basis, as required by SR 4.6.1.1.a. The four isolation valves are required to be locked closed and electrically deactivated in Modes 1 through 4 by TS 3.6.3.2, and are verified to be locked closed and electrically deactivated prior to reactor startup under SR 4.6.1.7.

M. On November 21, 1997, during the ongoing review of surveillance procedures to confirm compliance with surveillance requirements, it was discovered that a current surveillance procedure to verify ECCS subsystem electrical alignment was inadequate. This procedure is used to verify that ECCS subcomponents are operable

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

and. "aligned to receive electrical power from separate operable emergency busses", on a monthly basis as required by SR 4.5.2.a.9. Charging system Motor Operated Valve (MOV) 2-CH-501 is not full stroke testable during power operation. The control room panel position indication is powered by the same breaker as the MOV, thus indicating power is available to the valve. Control room personnel frequently check that power is available to the MOV by observing the position indicating lights, however, the checks were not documented in a surveillance procedure. This valve is open during normal operation. During an accident, the MOV receives a SIAS signal and goes closed.

N. On November 26, 1997, during the Generic Letter 96-01 review of Reactor Protection System (RPS) surveillance procedures, several deficiencies were discovered in the testing of the Thermal Margin/ Low Pressure (TM/LP) and Variable High Power Trip (VHPT) channels. The following types of deficiencies were identified: 1) a functional test was performed each refueling outage but not quarterly as required by SR Table 4.3-1, channel checks and calibrations were performed. 2) no procedure was found which performed a quarterly channel calibration required by SR Table 4.3-1 item 2, however daily and monthly calibrations were performed, 3) testing of permissives and automatic removals required by the SR Table 4.3-1 are performed in one test procedure, not during each channel calibration test. 4) TS 4.3.1.1.1 required channel checks were not documented in a surveillance procedure. These types of test deficiencies constitute missed surveillances.

O. On January 21, 1998, during simulator training, operators questioned whether the existing surveillance procedure for the Main Steam Isolation Valves (MSIV) partial stroke test adequately met the TS SR 4.7.1.5a. During a plant startup, by procedure, the MSIV partial stroke test is performed in Mode 2 or 3 with the RCS temperature greater than 300 degrees F. SR 4.7.1.5a, applicable in Modes 1, 2 & 3, requires that any open MSIV(s) be partial stroke tested every 92 days. After an extended shutdown (greater than 92 days plus the allowed 25 percent extension), the plant was heated above 300 degrees, and both MSIVs were opened and tested a short time later. While the MSIVs were open, and before they were tested, the MSIVs should have been declared inoperable. TS 3.7.1.5 does not address both MSIVs being inoperable at the same time, placing the unit outside TS 3.7.1.5 and into TS 3.0.3. Review of operating logs shows this TS violation had occurred on at least eight occasions between April 1978 and August 1995.

These events are being reported pursuant to 10 CFR 50.73 (a)(2)(i)(B) any operation or condition prohibited by the plants Technical Specifications.

II. Cause of Event

The cause of these conditions is a failure to achieve compliance to Technical Specification Requirements.

III. Analysis of Event

A. The CAR system and the CS system work in conjunction with each other in the event of a Loss of Coolant Accident (LOCA) or Main Steam Line Break (MSLB) to provide a means of cooling the containment atmosphere to reduce containment building pressure, and thus reduce the potential leakage of airborne and gaseous radioactivity. Each system, (CAR and CS) has two redundant, independent and separate sub systems. Two of the four CAR units in conjunction with one train of the CS system limits the containment pressure and structural temperature to less than containment design values following a LOCA.

FACILITY NAME (1)	DOCKET	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Millstone Nuclear Power Station Unit 2	05000336	97	-- 022 --	03	6 OF 11

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

The RBCCW system, which is cooled by the SW system, has two independent, redundant subsystems (trains) each having 100% heat removal capacity following a LOCA. A spare RBCCW pump and heat exchanger can supply either loop. The SW system also has two independent, redundant subsystems (trains), each having 100% heat removal capacity. A spare SW pump can supply either train. Two independent cross-connectable trains with isolation valves are provided to each heat exchanger. Following a LOCA, one SW pump and train are required to provide cooling to the RBCCW and diesel generator heat exchangers.

Considering the historical nature of this condition, redundancy of the affected systems, the existing installed spare pumps and heat exchangers, and the installed system cross-ties, the safety significance of this event is low.

B. E-bar is a quantitative measurement of the average beta and gamma energies per disintegration for isotopes with half-lives longer than 15 minutes, excluding iodine. Because the value of E-bar does not change rapidly, analysis is only required every 6 months. The limits on the specific activity of the reactor coolant ($\leq$ 100/E-bar uci/gm) ensure that following a steam generator tube rupture accident, the site boundary 2 hour dose does not exceed a small fraction of the 10 CFR 100 limits.

The Standard Technical Specifications (STS) for Combustion Engineering Units, NUREG-1432, Revision 1, changes the E-bar applicability to Mode 1, 2, & 3 with RCS average temperature $\geq$ 500 degrees F. The STS also provides direction that the E-bar determination is "Not required to be performed until 31 days after a minimum of 2 EFPD and 20 days of Mode 1 operation have elapsed since the reactor was last subcritical for $\geq$ 48 hours." This demonstrates the need to achieve equilibrium conditions at full power to obtain a meaningful value for E-bar.

Considering the historical nature of this condition, the corrective action (Surveillance Procedure and Form revision) that has been completed, and that a representative value was used ("use E-bar and 100/E-bar from 9/26/94 SP until representative data is obtained"), the safety significance of this event is low.

C. Technical Specification 6.15 requires that the station have an REMODCM with Section I, Radiological Effluents Monitoring Manual (REMM), and Section II, Offsite Dose Calculation Manual (ODCM). This manual describes the sampling and analysis programs to determine the concentration of radioactive materials released offsite, and the methodology and parameters used in the offsite dose calculations. ODCM Paragraph D.2, 10 CFR 50 Appendix I - Noble Gas Limits, requires the quarterly air dose determination using noble gas from all sources - ventilation, containment purges, and waste gas tanks. The surveillance procedure did not account for the noble gas released from the containment (CTMT) purges and Waste Gas Delay Tanks (WGDT), which are both specifically required by the ODCM. TS 3.11.2.2.a places a limit on the air off-site dose due to noble gas for each calendar quarter of $\leq$ 5 mrad for gamma and $\leq$ 10 mrad for beta. Surveillance Requirement 4.11.2.2.1 states, "Cumulative dose contributions for the current calendar quarter and current calendar year shall be determined in accordance with Section II of the REMODCM once every 31 days." Failure to include noble gas from containment purge and WGDT is a violation of the TS Surveillance Requirement.

The safety significance of this event is low. The doses resulting from releases from containment purges and waste gas tanks are calculated separately for the Annual Radiological Effluent Report. These doses have never exceeded 10 CFR 50 Appendix I design guidelines.

D. The purpose of this SR is to assure operability of the fire suppression system. Valve 2-Fire-108 is the first isolation valve outside containment on the main feed line for the containment fire suppression system. It is normally locked closed during operation and open when the plant is shutdown. As such it is not used to achieve safe shutdown.

Valve 2-FIRE-108 has been opened at the beginning of each outage to verify the operability of the fire system within containment. It was closed at the end of each outage to reestablish containment integrity. Thus, the valve was cycled each outage through one complete cycle, but it was not documented as a surveillance. Therefore, this condition is not safety significant.

E. Technical Specification 6.15 requires that the station have an REMODCM with Section I, REMM, and Section II, ODCM. This manual describes the sampling and analysis programs to determine the concentration of radioactive materials released offsite, and the methodology and parameters used in the offsite dose calculations. ODCM Paragraph D.3, 10 CFR 50 Appendix I - Iodine and Particulate Dose, requires the quarterly dose determination using the total curie of I-131 and I-133 in gaseous effluents. The surveillance procedure did not account for the SG blowdown vent, a potential iodine source during periods of primary to secondary leakage. Surveillance Requirement 4.11.2.3.1 states, "Cumulative dose contributions for the current calendar quarter and current calendar year shall be determined in accordance with Section II of the REMODCM once every 31 days." Failure to include SG blowdown vent iodine is a violation of the TS Surveillance Requirement. Releases from the SG blowdown vent have been addressed in annual off-site dose calculations, with the resulting off-site dose significantly less than Appendix I limits. Therefore, this condition is not safety significant.

F. The purpose of SR 4.1.1.1.2 is to identify anomalies between measured and predicted core reactivity. The fact that the RCS Boron Concentration Curve was not renormalized does not affect its ability to identify anomalies. The RCS Boron Concentration Curve is evaluated at the beginning of each cycle and frequently during operation such that any anomaly would have been identified. Therefore, the intent of this SR is met and this condition is not safety significant.

G. The Charging subsystem of the ECCS is designed to align a flow path and provide flow from the BASTs through the Boric Acid pumps or gravity feed valves to the Charging pump suction for injection into the RCS to provide core cooling.

The In-service test (IST) and Integrated Facility Tests, cycle and verify automatic actuation of the valves that act to isolate other portions of the CVCS system, providing assurance that the flowpath boundary can be created. Additionally, other manual isolation valves could be closed to create the boundary if an automatic valve failed to actuate to the correct position. These six valves have been stroke tested as part of the IST program on a quarterly basis and actuation tested as part of the Integrated Test of Facility Components on an 18 month basis. The current testing configuration has provided adequate assurance that these valves will perform their safety function. Therefore, this condition is not safety significant.

H. The portion of the Fire Protection piping which was not Code System Pressure Tested is the containment piping penetration, between the inboard and outboard containment isolation valves. The isolation valves are normally closed during operation and are opened during outages to provide fire system pressure inside containment. Historically, a system pressure test is performed and a containment walkdown is conducted that includes a visual exam of all of the penetrations, including the penetration of interest. However, the exam was not documented as a Code system Pressure Test. A satisfactory Code System Pressure Test was conducted on August 29, 1997. This condition is not safety significant.

I. The LCO 3.1.3.4 requires the CEA drop time be ≤ 2.75 seconds, with $T_{AVG} \geq 515$ degrees F, and all reactor coolant pumps running. This LCO is applicable in MODE 3 and it has no provision stating that 4.0.4 does not apply. The temperature and coolant pump conditions cannot be achieved before entering Mode 3. Surveillance Requirement 4.0.4 states, "Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the Surveillance Requirement (s) associated with the Limiting Condition for Operation have been performed within the stated surveillance interval or as otherwise specified." Therefore, the provision of 4.0.4 can not be met with respect to performing Surveillance Requirement 4.1.3.4 prior to entering Mode 3. The Surveillance Requirement 4.1.3.4 states, "The CEA drop time of full length CEAs shall be demonstrated through measurement prior to reactor criticality." Criticality is limited to MODES 1 and 2. Therefore, performing the CEA drop time surveillance before entering MODE 2 will satisfy this requirement. This condition is not safety significant.

J. The auxiliary spray is used to reduce pressurizer pressure during periods when the reactor coolant pumps are not running to provide normal spray flow. The TS differential temperature limit is to minimize the thermal cycle duty of the pressurizer spray nozzle safe end. The nozzle has been inspected in 1990, 1992 and 1994 with no unacceptable indications. The total number of auxiliary and main spray actuations is 226, which is much less than the design allowable thermal cycle limit of 600 auxiliary sprays at 400 degree F differential temperature. A preliminary evaluation concluded that the structural integrity of the nozzle has not been compromised. Based on this information, this is not a safety significant condition.

K. The EDGs are designed to provide electrical power to critical plant equipment in the event of a loss of normal electrical power. The EDGs must be capable of providing power during and following certain seismic events. The most probable failure mode of these piping components due to the overstress of a seismic event would be a pipe crack at the "A" EDG fuel day tank sight glass threaded connection. Such a crack could result in leakage of fuel from the day tank into the EDG room. Since a seismic event is not postulated during a design basis accident and the EDG are not required during a station blackout, this condition is not safety significant.

L. The requirement to verify containment integrity ensures that the release of radioactive material from the containment will be limited such that the site boundary radiation doses are within the limits of 10 CFR 100 during an accident.

Before 1980 the two containment purge supply and two containment purge exhaust isolation valves were normally closed and received a Containment Isolation Actuation Signal. These valves were full stroke tested for operability every 92 days, as required by SR 4.6.3.1.1.a. The four valves were listed on Table 3.6-2, power operated valves that are testable during plant operation. In October 1980, the four valves were removed from Table 3.6-2, and new TS 3.6.3.2 and SR 4.6.1.7, which are specific to these four containment purge isolation valves, were implemented. TS 3.6.3.2 requires the containment purge supply and exhaust isolation valves to be locked closed and electrically deactivated. SR 4.6.1.7 requires verification that they are locked closed and electrically deactivated prior to each reactor startup. This surveillance requirement has been met, however, an additional requirement for verification every 31 days under SR 4.6.3.1.1.a was overlooked. The valves are maintained in their closed and deenergized condition by removal of the power fuses and installation of a locking device which prevents inadvertent replacement of the fuses. Therefore, this condition is not safety significant.

M. The purpose of SR 4.5.2.a.9 for the ECCS subsystems is to ensure it is aligned electrically to provide sufficient ECCS capability in the event of an accident, assuming loss of one subsystem due to a single failure. Most ECCS automatic valves are stroke tested per SR 4.5.2.a.6 to verify operability and power available. However, certain automatic valves, like 2-CH-501, are not full stroke testable during normal operation, because stroke testing could disrupt operations. Although the power available checks for MOV 2-CH-501 had not been documented in a surveillance procedure, the frequent scanning of control boards by Operations personnel each day would identify a loss of power to the MOV. Additionally, the MOV is stroked as part of certain plant operations. Based on this information, this is not a safety significant condition.

N. The RPS monitors selected nuclear steam supply conditions to affect reliable and rapid reactor shutdown if any condition deviates from a preselected operating range. The system functions to protect the core and reactor coolant pressure boundary. The TM/LP trip is provided to prevent operation when the Departure from Nucleate Boiling Ratio is less than 1.17. The Low Pressurizer Pressure portion functions to trip the reactor in the event of a LOCA. The VHPT is provided to trip the reactor in the event of a reactivity excursion too rapid to result in a high pressure or TM/LP trip. Although certain SR were missed, other tests verified operability, and this condition is considered to be of low safety significance.

O. Operability of the MSIVs ensures that no more than one steam generator will blow down in the event of a MSLB. These restrictions minimize the positive reactivity effects of an RCS cool down resulting from a blowdown, and limits containment pressure rise for a MSLB inside containment. Although TS 3.0.3 requirements were violated during the time period between opening the MSIVs and performing the surveillance procedure, the valves were proven operable with the satisfactory performance of the partial stroke test. Therefore, this event was not safety significant.

IV. Corrective Action

As a result of these events, the following actions have been, or will be, performed.

A.1. A Technical Specification Change Request has been submitted to add appropriate action statements for the Containment Spray and Containment Air Recirculation Systems.

B.1. The E-bar surveillance procedure and surveillance form have been revised to conform to the applicable Technical Specification Surveillance Requirements.

C.1. The noble gas surveillance procedure has been revised to properly conform to the applicable Technical Specification referenced Offsite Dose Calculation Manual Requirements.

D.1. The applicable surveillance procedure has been revised to require testing of the Outside Containment Header Isolation Valve (2-FIRE-108) every refueling outage.

E.1. The applicable surveillance procedure has been revised to include Steam Generator Blowdown vent iodines as an input for quarterly dose calculations.

F.1. The applicable surveillance procedure has been revised to require normalization of the RCS Boron Concentration Curve prior to a fuel burnup of 60 EFPDs after each refueling outage.

G.1. The applicable surveillance procedure(s) will be revised prior to entry into Mode 4 from the current outage to include monthly testing of the six identified boundary valves.

H.1 A Code System Pressure Test was conducted on August 29, 1997 with satisfactory results, and the system declared operable.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Millstone Nuclear Power Station Unit 2	05000336	97	– 022 –	03	10 OF 11

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

H.2 A Code System Pressure Test for the penetration will be added to the ASME Section XI program before entry into Mode 4 from the current outage, to assure it is performed as required.

I.1 A Technical Specification Change Request to change the applicability of LCO 3.1.3.4 to Modes 1 and 2 has been submitted to the NRC.

J.1 Appropriate procedures will be revised to use containment ambient temperature to indicate auxiliary spray temperature when initiating auxiliary spray. This revision will be implemented before entry into Mode 5 from the current outage.

J.2 An evaluation will be completed to confirm that structural integrity of the nozzle has not been compromised. This evaluation will be completed before entry into Mode 4 from the current outage.

K.1 Seismic analysis determined that additional pipe supports were required for both the "A" & "B" Emergency Diesel Generator day tank sight glass piping. The additional pipe supports have been installed which corrected the seismic qualification deficiency.

L.1 The appropriate surveillance procedure will be revised to fulfill the surveillance requirements. This revision will be implemented prior to entry into Mode 4 from the current outage.

M.1 The appropriate surveillance procedure will be revised to ensure the surveillance requirement is fulfilled. This revision will be implemented prior to entry into Mode 3 from the current outage

N.1 The appropriate surveillance procedure will be revised to fulfill the surveillance requirements. This revision will be implemented prior to entry into Mode 4 from the current outage.

N.2 A new surveillance procedure will be written to perform the required quarterly channel calibration. This procedure will be implemented prior to entry into Mode 4 from the current outage.

O.1 A Technical Specification change has been submitted to the NRC to incorporate the Technical Specification 4.0.4 exemption, which will allow the test to be performed in Mode 3.

All. Technical Specification surveillance procedures will be reviewed to ensure compliance with Technical Specifications surveillance requirements as part of Millstone Unit 2 Operational Readiness Plan (Reference NOV 336/96-08-07, NNECO Commitment No. B16076-2).

V. Additional Information

Similar Events

Previous LERs that involve Technical Specification violations include:

LER 83-010: Pressurizer Spray Temperature Differential Exceeded Technical Specification Limit
LER 95-019: Failure to meet Technical Specification Requirements for Plant Systems - Snubbers
LER 95-030: Violation of Technical Specification 3.0.4 During Reactor Plant Heatup
LER 96-001: Reactor Coolant System Heatup Rate Exceeded Technical Specification Limit

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Millstone Nuclear Power Station Unit 2	05000336	97	-- 022 --	03	11 OF 11

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

LER 96-003: Failure to Enter Technical Specification Limiting Condition for Operation 3.0.3 After Discovery that the Service Water Strainers were Inoperable Due to Ice Blockage
LER 96-005: Failure to Enter Technical Specification Action Statement during Maintenance and Inservice Testing
LER 96-007: Reactor Coolant System Cooldown Rate Exceeded Technical Specification Limit
LER 96-011: Plant Shutdown Required by Technical Specifications Completed and Required Time to Enter Mode 5 was Exceeded
LER 96-015: Failure to Perform Action Requirement for the Technical Specifications Limiting Condition for Operation 3.3.1.1
LER 96-023: Discrepancies Found in Various Technical Specification Required Valve Lineups
LER 96-024: Inadequate Surveillances for Reactor Protection System and Engineered Safety Actuation System Response Time Testing
LER 96-025: Enclosure Building Filtration Actuation Signal/Auxiliary Exhaust Actuation Signal Interlock Not Tested Periodically
LER 96-026: Incomplete Technical Specification Required Surveillance - Valve Lineups Inside Containment
LER 96-035: Failure to Perform Periodic Surveillance Testing for Interlock Function Associated with the Main Steam Isolation System Function of the Engineered Safeguards Actuation System
LER 96-037: Inadequate Surveillance Procedure for Verifying Average Water Temperature at the Unit 2 Intake Structure
LER 96-038: Inadequate Surveillance Procedures Used to Verify Emergency Diesel Generator Operability
LER 96-039: Failure to Perform Periodic Surveillance Testing for Containment Purge System Containment Isolation Valves in Accordance with Technical Specification 4.9.10
LER 96-040: Inadequate Surveillance Procedure for Verifying Motor Circuit Breaker Position in Accordance with Technical Specification Requirements 4.1.2.3.2, 4.1.2.3.3, and 4.4.1.4
LER 97-003: Historical Technical Specification Noncompliance of Plant Surveillance Procedure used to Perform Periodic Inspection of Fire Protection System Smoke Detectors
LER 97-004: Violation of Technical Specification 3.1.2.3 Requirement for Number of High Pressure Safety Injection Pumps Capable of Injecting into the Reactor Coolant System
LER 97-005: Inservice Test Instrumentation Does Not Meet ANSI/ASME Chapter XI Requirements
LER 97-007: Inadequate Surveillance Procedure for Verifying Operability of Reactor Coolant System Vents
LER 97-008: Insufficient Testing of RPS Logic Circuitry (Generic Letter 96-01 Review)
LER 97-009: Insufficient ESFAS Surveillance Testing (Generic Letter 96-01 Review)
LER 97-010: Heavy Dummy Fuel Assembly and Handling Tool Weight Exceeds Technical Specification Limit
LER 97-012: Inadequate Diesel Generator Start Surveillance Test
LER 97-013: Surveillance Procedure Bypasses Wrong Radiation Monitor Annunciator
LER 97-016: Technical Specification 4.0.4 Incorrectly Applied to Surveillance Requirements for the TDAFW Pump
LER 97-017: Bypassed Refuel Machine Overload Cut-off
LER 97-019: Automatic Test Initiator Sends Repeated Trip Signals to the RSST Feeder Breaker Timing Circuit
LER 97-020: Insufficient Verification of B HPSI Pump Header as Boron Injection Flowpath

Energy Industry Identification System (EIIS) codes are identified in the text as [XX].