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

LICENSEE EVENT REPORT (LER)

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

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Millstone Nuclear Power Station Unit 1	05000245	1 of 4

TITLE (4)
RBCCW Containment Isolation System Single Failure Vulnerability

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
01	27	97	97	004	00	02	28	97	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)			
POWER LEVEL (10)	000	20.2201(b)	20.2203(a)(2)(v)	50.73(a)(2)(i)	50.73(a)(2)(viii)
		20.2203(a)(1)	20.2203(a)(3)(i)	X 50.73(a)(2)(ii)	50.73(a)(2)(x)
		20.2203(a)(2)(i)	20.2203(a)(3)(ii)	50.73(a)(2)(iii)	73.71
		20.2203(a)(2)(ii)	20.2203(a)(4)	50.73(a)(2)(iv)	OTHER
		20.2203(a)(2)(iii)	50.36(c)(1)	50.73(a)(2)(v)	Specify in Abstract below or in NRC Form 366A
		20.2203(a)(2)(iv)	50.36(c)(2)	50.73(a)(2)(vii)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Robert W. Walpole, MP1 Nuclear Licensing Manager	(860)440-2191

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	MONTH	DAY	YEAR
X YES (If yes, complete EXPECTED SUBMISSION DATE).	NO		04	30	97

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

On January 27, 1997, at 1200 hours, with the plant in COLD SHUTDOWN, a single failure vulnerability of Reactor Building Closed Cooling Water System (RBCCW) was identified. Containment Isolation Valve 1-RC-206 has an allowable closure time of 35 seconds per Technical Requirements Manual. The valve is powered from MCC-E3 which is connected to the Gas Turbine (GT) for its emergency power source. Following a Loss of Normal Power (LNP) event, AC power will not be restored to MCC-E3 for a maximum Technical Specification limit of 48 seconds. A time limit of 60 seconds is assumed in the offsite dose calculations to establish primary containment integrity. On a Loss of Coolant Accident (LOCA) with a LNP, a failure of DC operated valve 1-RC-207 will result in penetration X 24 taking longer than the assumed 60 seconds to completely isolate. Any post-LOCA gas escaping from primary containment through this penetration would be released into secondary containment. This leakage would be in addition to the 300.3 SCF per hour Appendix J leakage from the Drywell assumed in the off site dose calculations. Thus, this condition could exceed the existing calculated site boundary dose limits. The cause of this condition was the failure to adequately establish a design basis for RBCCW containment isolation. An engineering evaluation was performed that determined that the leakage from containment as a result of this delay in establishing primary containment integrity would be an additional leakage of approximately 25.1 SCF from the post-LOCA containment atmosphere. The radiological impact of this additional leakage is being reviewed and the results will be provided in a supplemental report. There were no safety consequences as a result of this event.

I. Description of Event

On January 27, 1997, at 1200 hours, with the plant in COLD SHUTDOWN and the reactor defueled, a condition was discovered that affects the calculated offsite dose. The offsite dose analysis assumes that containment integrity is established not greater than 60 seconds following a containment isolation signal. RBCCW Containment Isolation Valve 1-RC-206 is an AC motor operated valve and has an allowable closure stroke time of 35 seconds per Millstone Unit No. 1 Technical Requirements Manual, Table 3.7.D. A design basis event, a LOCA with a concurrent LNP will remove power from the bus supplying the valve until the emergency power source is available. The emergency power source for this bus is the GT. The GT is required to be able to accept emergency load within 48 seconds in accordance with Technical Specification 4.9. The maximum total closure time for 1-RC-206 is 48 seconds to restore power to the bus plus 35 seconds to stroke the valve closed. This yields a total closure time of 83 seconds. A single failure of DC powered valve 1-RC-207 to close, along with the LOCA/LNP design basis event, will result in the penetration X-24 failing to isolate within the 60 seconds assumed in the offsite dose analysis.

The requirement to isolate primary containment within 60 seconds is assumed by the Millstone Unit No. 1 radiological analysis following the initiation of a large-break LOCA. This is based on an interpretation of NUREG-0800, "Containment Isolation," Section III, Page 6.2.4-9. For this penetration, the release from primary containment is through the RBCCW surge tank vent located in secondary containment, assuming failure of the RBCCW piping within the Drywell. This event was immediately reported pursuant to 10CFR50.72(b)(2)(i) as being in an unanalyzed condition that significantly compromises plant safety.

The original design of the unit considered the RBCCW system within containment to be a closed loop. Two isolation valves were installed in the system as part of the original system design, a check valve on the supply piping and a remotely operated motor operated valve on the return piping. However, the effects of a high energy line break on the RBCCW system within containment was not considered. This deficiency and challenge to containment integrity was reported in LER 89-003-00. A commitment was made in response to LER 89-003-00 to upgrade the containment isolation capability of the RBCCW to meet 10CFR50, Appendix A, General Design Criteria 54 and 57 and Appendix J. These criteria were met but no consideration was given to the offsite dose calculations which assumed containment isolation within 60 seconds.

The upgrade of the containment isolation capability included the installation of two remote actuated valves in series on both the RBCCW inlet and outlet lines outside the containment penetrations. Plant procedures instructed the operator to close these valves when containment pressure reached 5 psig. The containment isolation design required a maximum valve stroke time of 35 seconds, and the valves were powered from redundant and diverse power supplies to assure isolation. The design did not factor the consequences of a LNP event and the delay in restoring power from the GT into the total closure time. The effect on the offsite dose calculations was not considered.

The containment isolation capability was later modified to add an automatic closure feature when containment pressure reached 5 psig. The modification automated the operator function to close the RBCCW isolation valves when containment pressure reached 5 psig. Again, the effect on the offsite dose calculations was not considered.

II. Cause of Event

The cause of this condition was the failure to adequately establish a design basis for RBCCW containment isolation. There was a failure to properly address the downgrading of the RBCCW piping in the drywell with respect to the radiological offsite dose assessment.

III. Analysis of Event

An engineering evaluation concluded that the total valve closure time of 83 seconds exceeded the allowable closure time of 60 seconds and the plant was in an unanalyzed condition. This condition could result in an increase in the calculated offsite dose in that containment isolation does not occur within the assumed time period. This report is pursuant to 10CFR50.73(a)(2)(ii)(A), an unanalyzed condition that significantly compromised plant safety.

The piping inside primary containment is not qualified and is assumed to fail at the start of the event. The release path from this penetration would be through the failed RBCCW piping to the RBCCW surge tank. At the start of the event, RBCCW system pressure would be above primary containment pressure and no leakage from the primary containment would be possible. As RBCCW system pressure decreases due to RBCCW pump coastdown and loss of inventory through the failed piping, primary containment pressure will be increasing due to the LOCA. A leakage path from primary containment to secondary containment will be established once the primary containment pressure is greater than the pressure in the RBCCW piping.

Engineering evaluations were performed to quantify the release path and flow rates from containment due to the failure of 1-RC-206 to isolate within 60 seconds. This evaluation concluded that an additional leakage of approximately 25.1 SFC of post-LOCA atmosphere would be released from primary containment due to the delay in isolating penetration X-24. At this time, no formal quantitative analysis exists to determine the radiological impact of the scenario. This analysis will be completed via radiological assessment and engineering calculations to determine if significant impact exist. The results of this analysis will be provided in a LER supplement. There were no actual safety consequences from this event because the unit has not had a LOCA with concurrent LNP and the unit is currently shutdown with the reactor defueled.

Twelve additional containment isolation valves were identified as being supplied emergency power from the GT. The function and arrangement of the valves were reviewed and it was determined that these valves were not adversely impacted by the delay required to have power from the GT available. In addition, the function and arrangement of remaining containment isolation valves powered from the Diesel Generator were reviewed and it was determined that these valves would not be adversely impacted by the delay required to have power from the Diesel Generator available.

IV. Corrective Action

No immediate corrective actions are required since the plant is in COLD SHUTDOWN with the reactor defueled.

A radiological assessment will be performed by March 21, 1997, to determine the impact on off-site dose calculations based on the additional leakage from the penetration X-24. Based on the results, an engineering evaluation will be performed on containment isolation valve 1-RC-206 to determine the corrective actions required to assure that the offsite dose remains within allowable limits. The results of the radiological assessment and any required corrective actions will be provided in a supplement to this report by April 30, 1997. The required corrective actions will be completed prior to primary containment being required for operating Cycle 16.

In LER 96-061-00, Commitment No. B16078-2 has committed to the review of the systems to address the containment isolation function. This review is required prior to startup for operating Cycle 16 as part of the ongoing 10CFR50.54(f) effort

V. Additional Information

Similar Events

LER 96-050-00, LOCA Concurrent with LNP, and Loss of DC Power Prevents Closure of LPCI Torus Test Return Valves.

LER 96-061-00, Failure of Containment Isolation Function in a Design Basis Accident Concurrent with a Loss of One Train of DC System.

Manufacturer Data

Not Applicable