Entergy

Entergy Nuclear Generation Co.
Pilgrim Nuclear Power Station
600 Rocky Hill Road
Plymouth, MA 02360

Mike Bellamy
Site Vice President

10 CFR 50.73

February 27, 2002
ENGC Ltr. 2.02.013

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

Docket No. 50-293
License No. DPR-35

Dear Sir:

The enclosed Licensee Event Report (LER) 1999-003-01, "Local Leak Rate Test Results Exceeding Allowable Technical Specification Leakage Rates," is submitted in accordance with 10 CFR 50.73.

This letter contains no commitments.

Please do not hesitate to contact me if there are any questions regarding this report.

Sincerely,

FOR

Mike Bellamy

JLR/
Enclosure: LER 1999-003-01

cc: Mr. Hubert J. Miller
Regional Administrator, Region 1
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Senior NRC Resident Inspector

Mr. Douglas Starkey, Project Manager
Office of Nuclear Reactor Regulation
Mail Stop: 0-8B-1
U. S. Nuclear Regulatory Commission
1 White Flint North
11555 Rockville Pike
Rockville, MD 20852

INPO Records

IE22

202013

NRC Form 366 U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104
LICENSEE EVENT REPORT (LER) (See reverse for number of digits/characters for each block)	Estimated burden per response to comply with this mandatory information collection request: 50 hrs. Reported lessons learned are incorporated into the licensing process and fed back to industry. Forward comments regarding burden estimate to the 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. If an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE(3)
PILGRIM NUCLEAR POWER STATION	05000-293	1 of 5

TITLE (4)
Local Leak Rate Test Results Exceeding Allowable Technical Specification Leakage Rates

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 N/A	DOCKET NUMBER 05000
05	16	99	1999	003	01	2	27	02	FACILITY NAME N/A	DOCKET NUMBER 05000

OPERATING MODE (9)	N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR: (Check one or more) (11)							
			20.2201(b)		22.2203(a)(3)(i)		50.73(a)(2)(i)(C)	x	50.73(a)(2)(vii)
POWER LEVEL (10)	0		22.2202(d)		20.2203(a)(3)(ii)	x	50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(A)
			20.2203(a)(1)		20.2203(a)(4)		50.73(a)(2)(ii)(B)		50.73(a)(2)(viii)(B)
			20.2203(a)(2)(i)		50.36(3)(1)(i)(A)		50.73(a)(2)(iii)		50.73(a)(2)(ix)(A)
			20.2203(a)(2)(ii)		50.36(3)(1)(ii)(A)		50.73(a)(2)(iv)(A)		50.73(a)(2)(x)
			20.2203(a)(2)(iii)		50.36(c)(2)		50.73(a)(2)(v)(A)		73.71(a)(4)
			20.2203(a)(2)(iv)		50.46(a)(3)(ii)		50.73(a)(2)(v)(B)		73.71(a)(5)
			20.2203(a)(2)(v)		50.73(a)(2)(i)(A)	x	50.73(a)(2)(v)(C)		OTHER
			20.2203(a)(2)(vi)	x	50.73(a)(2)(i)(B)		50.73(a)(2)(v)(D)		Specify in Abstract below or in NRC Form 366A

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Bryan Ford – Manager Licensing	508-830-8403

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
X	SB	ISV	A585	Y	X	BN	SHV	A391	Y
X	SB	ISV	A585	Y					

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

ABSTRACT (16)

On May 16, 1999, the Control Room was informed that the containment isolation valves for the 'C' main steam line exceeded the Technical Specifications (T.S.) allowable leakage rates during Appendix 'J' Local Leak Rate Testing that was being performed during the 1999 Refueling Outage (RFO 12). The in-series main steam line 'C' isolation valves were found to have leakage of 280 standard liters per minute (SLM) and 255 SLM. The test failure of the inboard valve (AO 203-1C) was caused by minor imperfections in the seating surface due to normal age related wear. The test failure of the outboard valve (AO 203-2C) was caused by flow induced vibration that caused wear on the main disk and pilot poppet seating surfaces. Both valves were refurbished and successfully leak rate tested before being returned to service in RFO 12.

On May 24, 1999, a globe stop-check valve in the Reactor Core Isolation Cooling System turbine exhaust piping did not meet the leakage acceptance criteria during the performance of Appendix J testing. Valve disassembly revealed that a portion of the resilient seat ring was missing. The seat ring was replaced and the valve was successfully leak rate tested before being returned to service in RFO 12.

The conditions were discovered during reactor refueling operations with the reactor mode selector switch in the REFUEL position. The reactor vessel pressure was zero psig and reactor coolant temperature was 84° Fahrenheit. The condition posed no threat to public health and safety.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
PILGRIM NUCLEAR POWER STATION	05000-293	1999	003	01	2 of 5

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

REASON FOR SUPPLEMENT

This supplementary report is submitted in accordance with our commitment made in the original report. This report describes the root cause(s) for the LLRT failures and the corrective actions taken to resolve these failures.

BACKGROUND INFORMATION

The safety objective of the Main Steam Isolation Valves (MSIVs) is to limit the release of radioactive materials by closing the primary containment barrier in case of a major leak from the nuclear system inside the primary containment. There are two, in-series MSIVs located in each of the four main steam lines. In each steam line, one MSIV is located inside primary containment (inboard valve) and one MSIV is located outside primary containment (outboard valve). The MSIVs are Atwood & Morrill Y - Pattern, 20 inch MSIV's, Model Number 20849-H.

Each valve is a 20-inch globe valve having a Y-pattern body with a cylindrical main disk moving in a centerline 45 degrees upward from the axis of the horizontal main steam line. The main disk, guided at the bottom by hard faced ribs cast integral with the valve body, has a hard faced seal surface at the bottom which mates with a hard faced seat welded into the valve body when the valve is closed. A pilot poppet is attached to the bottom end of the valve stem and seals against a mating hard faced seat in the middle of the main disk. An internal, helical spring assists the valve stem in lifting the pilot poppet off its seat when the main poppet is to be opened. This provides a means of balancing the pressure across the main disk just before it is lifted. The pilot poppet remains open whenever the main disk is off its seat.

The pilot poppet is attached to the stem by a nut and split retaining rings. Set screws are provided to prevent the pilot poppet/nut assembly from loosening. A welded pilot/poppet nut assembly that does not use set screws can also be used. An upper and lower spring plate retains the internal spring. The upper spring plate is secured to the stem by either two retaining pins or a single through-stem taper pin.

Check valve CK-1301-64 is located in the steam exhaust piping between the Reactor Core Isolation Cooling (RCIC) System turbine and its termination in the suppression pool. The safety function of the stop-check valve is to provide automatic containment isolation by closing if flow reversal in the RCIC turbine exhaust line occurs. If flow reversal occurs, the weight of the disk along with the force of the reverse pressure, seals the disk against the seat. The valve has a secondary seat that uses a resilient elastomer seat and a retaining ring designed to provide leak-free isolation. The 8 inch stop check valve was manufactured by Anchor-Darling Company, Model Number E9982.

EVENT DESCRIPTION

On May 16, 1999, at 0805 hours, the Control Room was notified that the inboard and outboard MSIVs for the 'C' main steam line (penetration X-7C) exceeded Technical Specifications (T.S.) 4.7.A.2.a.3 and 4.7.A.2.a.4 allowable leakage rates during Appendix 'J' Local Leak Rate Testing (LLRT) that was being performed during the 1999 refueling outage (RFO 12). The in-series main steam line 'C' isolation valves AO-203-1C and AO-203-2C were found to have leakage of 280 standard liters per minute (SLM) and 255 SLM, respectively. This condition was documented by Problem Report 99.9216. At 0835 hours on May 16, 1999, the NRC Operations Center was notified via the Emergency Notification System in accordance with 10 CFR 50.72(b)(2)(i).

Disassembly of AO-203-1C revealed wear on the main disk seating surface and minor low spots on the body seat. The axial movement of the pilot poppet on the stem was approximately one inch as compared to a normal movement of 1/64 inch.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
PILGRIM NUCLEAR POWER STATION	05000-293	1999	003	01	3 of 5

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

Disassembly of AO-203 2C revealed that the internals had severe wear. The pilot poppet and lower spring plate surface, where it contacts the pilot poppet, were worn. The upper spring plate was loose and the single, through-stem taper pin attaching the plate to the stem fell out during disassembly. There was wear on the main poppet seating surface where it contacted the guide ribs.

Also there was minor corrosion product buildup on the seating surfaces of both valves.

On May 24, 1999, at 0515 hours, PR 99.9266 was written to document that check valve CK-1301-64 failed its seat leakage test because the piping could not be pressurized between it and a blank flange (installed for local leak rate testing). On May 25, 1999 at 2129 hours, the NRC Operations Center was notified via the Emergency Notification System in accordance with 10 CFR 50.72(b)(2)(iii). Valve disassembly revealed that a small portion of the resilient seat was missing.

The conditions were discovered during reactor refueling operations with the reactor mode selector switch in the REFUEL position. The reactor vessel pressure was zero psig and the coolant temperature was approximately 84° Fahrenheit.

CAUSE

MSIV 203-1C failed the LLRT due to minor imperfections in the seating surfaces due to long term wear. This valve had been in service without disassembly since the 1986 Refueling Outage (RFO-7). Excessive axial movement of the pilot poppet on the stem was caused by long term wear of the split retaining rings.

MSIV 203-2C failed the LLRT due to wear of valve internals caused by steam-flow induced vibration. Excessive looseness or clearances resulting from the loose taper pin caused excessive wear rates due to vibration caused by steam flow turbulence. This valve had been overhauled in the 1997 Refueling Outage (RFO-11) and had been in service since RFO 7. Contributing to the failure was the location of the rigging points for installation and removal of the main poppet. The rigging points are offset from the valve centerline by a considerable amount. This can result in cocking of the main disk during valve assembly. There have been instances where damage to the main disk seat occurred because the main disk hung up on the top edge of the guide rib.

Both MSIVs in the 'C' steam line were placed in the closed position for approximately one month prior to the start of RFO 12. It is likely that corrosion product buildup on the seats while the valves were closed contributed to higher leakage during the MSIV LLRTs.

Valve CK-1301-64 failed the LLRT due to long term service related wear of the valve resilient seal. The valve was replaced in 1985 and no failures had occurred until RFO 12.

CORRECTIVE ACTION

Immediate corrective action was taken to verify the integrity of the secondary containment. Work documents were initiated to repair MSIVs AO-203-1C, AO-203-2C and CK-1301-64 prior to plant startup from RFO 12.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
PILGRIM NUCLEAR POWER STATION	05000-293	1999	003	01	4 of 5

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

The following actions were taken for valves AO-203-1C and 2C:
AO-203-1C was rebuilt using a new stem assembly with a welded pilot poppet/pilot poppet nut. Imperfections were blended out on the main disk seat and guide rib. Following repair, the valve was successfully retested to confirm that all design leakage rates were met. The pilot poppet stem assembly was inspected and no significant problems were identified.

AO-203-2C was rebuilt using a new stem assembly with a welded pilot poppet/pilot poppet nut and a through-stem taper pin. A new main disk assembly was installed due to the damage at the guide rib contact locations. Following repair, the valve was successfully retested to confirm that all design leakage rates were met.

Maintenance procedures were revised to include a check of taper pin installation for proper engagement and staking and to break the sharp edge at the top of the guide rib. Single piece stems with integral upper spring plate/stem assemblies to eliminate the taper pins and the potential for loosening were procured. This improved stem assembly is currently installed in AO-203-2C.

The following actions were taken for valve CK-1301-64:
A new valve disk, resilient seat and retaining ring were installed. The valve was forward flow tested and successfully local leak rate tested to confirm that all design leak rates were met.

SAFETY CONSEQUENCES AO-203-1C and 2C

A radiological assessment of the as-found test results for AO-203-1C and 2C indicated that 10 CFR 100 offsite dose limits and GDC19 control room dose limits would not be exceeded using the Design Basis Accident (DBA) LOCA source term as specified in the Pilgrim Station Updated Final Safety Analysis Report Appendix R.3. Thus even though the Technical Specification limits were exceeded, the consequence to public health and safety was acceptable.

SAFETY CONSEQUENCES CK-1301-64

The allowable total combined leakage rate for all Containment Isolation Valves (CIVs) that are considered to have the function of maintaining suppression pool water inventory (i.e., located in piping that terminates below the surface of the suppression pool) was 10.1 gpm in RFO 12 and was recently revised to 4.75 gpm. This criteria is based on the leakage rate that could result in a decrease in the suppression pool water volume to a level that would uncover piping terminating in the pool. It assumes that a sustained primary containment pressure of 1.1 times the calculated peak containment pressure following a DBA LOCA (P_a) exists over a thirty day period. The individual valve leakage criterion for CK-1301-64 is $\leq$ 4 gpm. This criterion reflects guidance adopted from ANSI/ASME Operations and Maintenance Standard, Part 10 (OM-10) of 0.5 gpm allowable leakage per nominal inch of valve diameter up to a maximum of 5 gpm.

The leakage rate through CK-1301-64 was not quantified but valve disassembly revealed this valve was closed with only a small portion of the resilient seat missing. The upstream swing check valve, CK-1301-41, was found to be closed even though it was not completely leak tight. Although the leakage criteria for this CIV (CK-1301-64) is based on a constant pressure of 1.1 P_a for 30 days, in actuality the containment pressure associated with a DBA LOCA decreases with time and the leakage rate through CK-1301-64 would also decrease with time. System differential pressure across the seats of both check valves would reduce leakage further. In addition, operators using Emergency Operating Procedures would control suppression pool volume/level such that it would not decrease to less than the level corresponding to the level of the exhaust lines that terminate in the suppression pool. Therefore, since suppression pool water level would be maintained, there was no consequence to the health and safety of the public.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
PILGRIM NUCLEAR POWER STATION	05000-293	1999	003	01	5 of 5

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

REPORTABILITY FOR AO-203-1C AND A0-203-2C:

This report was submitted in accordance with 10 CFR 50.73(a)(2)(i)(B) because the leakage through the in-series MSIVs AO-203-1C and AO-203-2C exceeded T.S. 4.7.A.2.a.4. A leakage rate of 21.7 SLM (46 scfh) as specified by T.S.4.7.A.2.a.4 is the maximum leakage allowed for all four main steam lines combined. The leakage is assumed to have occurred as a result of gradual degradation and therefore, not just at the time of the test. Therefore, although discovered while shut down, the 24 hour LCO specified by T.S. 3.7.A.6 is assumed to have been exceeded.

This report was also submitted in accordance with 10 CFR 50.73(a)(2)(ii). NUREG-1022, (Rev 1) Section 3.2.4 Discussion Criterion (1), subitems (f)(i) and (f)(iii) provide guidance that the problem is reportable via 10 CFR 50.73(a)(2)(ii), as significant degradation. For criterion (3): outside the design basis (a)(2)(ii)(B) applies because, assuming the problem had existed during power operation, the design function identified in the Updated Final Safety Analysis Report (UFSAR) Section 5.2.1 (maintain radioactive releases less than 10 CFR 100 limits) and T.S. Bases 3/4.7.A that relates Part 100 limits to one percent per day primary containment leakage could have been compromised if a design basis accident had occurred with the as-found leak rates of valves AO-203-1C and AO-203-2C.

This report was also submitted in accordance with 10 CFR 50.73 subparts (a)(2)(v)(C) and (a)(2)(vii)(C) because both trains (inboard and outboard) of the primary containment system penetration X-7C were inoperable.

REPORTABILITY FOR CK-1301-64:

This report was submitted in accordance with 10 CFR 50.73(a)(2)(ii)(B). The fact that check valve CK-1301-64 failed the leak rate test indicates the problem did not likely occur at the time of the surveillance and therefore, the problem likely existed for a period of time greater than the 24 hour LCO specified by Tech Spec 3.7.A.6. The design basis limit of 4 gpm (for CK-1301-64) was exceeded and therefore, is reportable in accordance with 10 CFR 50.73(a)(2)(ii)(B).

SIMILARITY TO PREVIOUS EVENTS

A review for similarity was conducted of Pilgrim Station LERs. The review focused on LERs documenting MSIVs or RCIC valves that failed their LLRT. The review identified LER 86-011-00, "Leakage Past MSIV's in Excess of LLRT Criteria."

ENERGY INDUSTRY IDENTIFICATION SYSTEM (EIIS) CODES

The EIIS codes for this report are as follows:

Components	Codes
Valve, Isolation	ISV
Valve, Check	SHV
Systems	
Containment Leakage Control System	BD
Main Steam System	SB
Reactor Core Isolation Cooling System	BN