AmerGen

An Exelon/British Energy Company

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December 14, 2001
5928-01-20353

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

Dear Sir or Madam:

SUBJECT: THREE MILE ISLAND NUCLEAR STATION, UNIT 1 (TMI-1)
OPERATING LICENSE NO. DPR-50
DOCKET NO. 50-289
LICENSEE EVENT REPORT (LER) NO. 2001-003-0
"DEGRADED OTSG TUBE"

This letter transmits LER No. 2001-002-0, regarding the discovery of a degraded condition that was initially reported to the NRC Operations Center as an 8 hour, non-emergency report on October 20, 2001. For a complete description of the evaluated condition, refer to the text of the report provided on Forms 366 and 366A.

This condition did not adversely affect the health and safety of the public. For additional information regarding this LER contact Mr. John Schork of TMI Unit 1 Regulatory Assurance at (717) 948-8832.

Sincerely,

George H. Gellrich
Plant Manager

GHG/jss

ATTACHMENT: List of Regulatory Commitments

cc: TMI Senior Resident Inspector
Administrator, Region I
TMI-1 Senior Project Manager
File No. 01085

IE22

SUMMARY OF AMERGEN ENERGY CO. L.L.C. COMMITMENTS

The following table identifies commitments made in this document by AmerGen Energy Co. L.L.C. (AmerGen). Any other actions discussed in the submittal represent intended or planned actions by AmerGen. They are described to the NRC for the NRC's information and are not regulatory commitments.)

COMMITMENT	COMMITED DATE OR "OUTAGE"
There are no NRC Commitments contained in this LER	N/A

NRC FORM 366 (6-1998)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 06/30/2001

LICENSEE EVENT REPORT (LER)

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

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Three Mile Island, Unit 1	05000289	1 OF 6

TITLE (4)

Degraded OTSG Tube

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
10	20	2001	2001	- 003 --	00	12	14	2001	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9): N

POWER LEVEL (10): 0

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

	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)
John S. Schork, TMI-1 Regulatory Assurance	(717) 948-8832

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
B	AB	SG	B015	No					

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

ABSTRACT (16)

On October 20, 2001, TMI-1 determined that consequential degradation from the sever of a plugged tube in the "B" once through steam generator (OTSG) [AB/SG] had resulted in degrading an adjacent in-service tube. The damage was such that the degraded tube may not have remained intact under accident conditions. Three (3) other adjacent in-service tubes were damage by the severed plugged tube but would have remained intact under accident conditions.

The degradation of tube B65-130 was caused by the fretting of the severed plugged tube B66-130 against B65-130. The 360 degree sever of the B66-130 was caused by the combined effects of leak by of water past the mechanical plug into the plugged tube followed by subsequent "trapping" of the water in the plugged tube (that resulted in over-pressurization of the plugged tube), low amplitude high cycle fatigue and the presence of an existing flaw on the outer diameter of the plugged tube.

The extent of condition in the "A" and "B" OTSGs was evaluated and found one circumferentially severed plugged tube in the "A" OTSG and found no other circumferentially severed plugged tubes in the "B" OTSG. The severed plugged tube in the "A" OTSG did not degrade any of the adjacent in-service tubes because the sever was captured in the tube support plate.

The corrective actions taken in response to the discovery of the condition were de-watering of 870 mechanically plugged tubes prior to plugging/re-plugging, the insertion of stabilizers into plugged tubes or the surrounding of plugged tubes with stabilized plugged tubes.

I. Plant Operating Conditions Before The Event:

TMI Unit 1 was in cold shutdown at the time the condition was discovered.

II. Status of Structures, Components, or Systems That Were Inoperable At The Start Of The Event And That Contributed To The Event:

Both the "A" Once Through Steam Generator and the "B" Once Through Steam Generator had been removed from service in order to perform the OTSG tube inspections.

III. Event Description:

In October 2001, Three Mile Island Nuclear Station, Unit 1 (TMI-1), was shut down for a refueling outage. At the time of the shut down and during the preceding operating cycle, there was no significant primary-to-secondary leakage. During the outage, steam generator tube inspections were performed. During these inspections, four tubes *[TBG] were identified that exhibited signs of tube wear near the secondary face of the upper tubesheet. There were no eddy current recorded signals of tube wear on these tubes during the prior steam generator tube inspections performed approximately 2 years earlier. As estimated by eddy current examination, the maximum depth of tube wear observed in the four active tubes ranged from 37% to 92% through-wall and the overall length of the wear scars ranged from approximately 2.8 inches to 8.3 inches.

As a result of the pattern and location of wear, it was determined that a plugged tube (B66-130) may have caused the wear on these four adjacent active tubes. The plug in the upper tubesheet was removed from this tube of the "B" steam generator, and a video inspection of the tube was performed. The video inspection discovered B66-130 severed near the secondary face of the upper tubesheet. This tube was originally plugged in 1986 with an alloy 600 mechanical plug as a result of inter-granular attack near the fifth tube support plate (i.e., there was no observable degradation at the location of the severance at the time the tube was originally plugged). The original plug in this tube was replaced in 1997 with an alloy 690 mechanical plug as part of a program to replace many of the alloy 600 plugs in the OTSGs upper tubesheets.

To investigate the severity of the wear indications, in-situ and laboratory pressure testing was performed on four of the degraded tubes. In-situ testing was performed on three of the tubes (B67-130, B66-131, B65-129) and laboratory testing was performed on the fourth tube (B65-130). As a result of the pressure tests performed on these tubes with wear indications, two tubes did not meet the design basis structural performance criterion for steam generator tubes. The burst pressure for one of these tubes, B65-130 was near the differential pressure that would be observed during a postulated main steam line break (MSLB). Another tube's (B66-131) burst pressure exhibited a margin against burst near the margin of 3 times against the normal operating pressure differential discussed in the Nuclear Energy Institute's (NEI's) guidelines, NEI 97-06, "Steam Generator Program Guidelines."

In addition to removing the degraded portion of two of the tubes with wear indications for destructive examination, the lower portion of the fractured surface of the severed tube was removed to assess the root cause. The preliminary laboratory investigation of the severed tube indicated the tube exhibited signs of high cycle fatigue, ductile failure, and outside diameter initiated intergranular attack (OD IGA). In addition, the tube diameter was expanded when compared to the nominal tube diameter indicating the tube had swelled. The video inspections performed in the steam generator confirmed the severed tube

was in physical contact with the drilled hole tube support plate, whereas the tube-to-drilled hole tube support plate crevices for the surrounding tubes were open (i.e., no visual evidence of tube expansion).

IV. Assessment of Safety Consequences & Implications of the Event:

The degradation of OTSG tube B65-130 was identified prior to any failure of the tube. Therefore there was no leakage during normal plant operations before the degradation of the tube was identified. Thus, there were no actual safety consequences from the degradation of OTSG tube B65-130.

The fretting of the severed plugged tube, B66-130 degraded B65-130 to the extent that the tube may have failed in the event of a Main Steam Line Break (MSLB) accident, an analyzed event in the TMI-1 licensing basis (Chapter 14, TMI-1 updated final safety analysis report (UFSAR)). The existing MSLB accident analysis assumes increased accident induced leakage but does not assume the amount of leakage that would occur from a totally failed tube. Thus, the potential implications of the dose from a MSLB accident with the severed single tube from a single OTSG was evaluated to determine the potential safety consequences of the degradation of OTSG tube B65-130.

The radiological consequences that would have resulted from a MSLB accident with an assumed double-ended sever of a single tube from a single OTSG at a conservative primary-to-secondary leak rate of 435 gallons per minute have been determined to be:

EAB (2-hour)

- Whole Body: 0.0742 Rem
- Thyroid: 18.9 Rem

LPZ (30-day)

- Whole Body: 0.0516 Rem
- Thyroid: 15.4 Rem

This dose is well within 10 CFR 100 guidelines and is bounded by the existing licensing basis dose calculated to result from the maximum hypothetical accident as documented in Chapter 14 of the TMI-1 UFSAR. The acceptance criteria for this combined accident (i.e., MSLB with SGTR) are not specifically cited in 10CFR100. However, since we are assuming a pre-incident spiking factor of 500, the dose acceptance criteria are the same as those for a MSLB and/or SGTR accident under these conditions (25 Rem whole body and 300 Rem thyroid). The doses cited above are well within these criteria.

The current licensing basis does not have a control room operator dose specifically calculated. However, since the resulting offsite doses are less than those resulting from a LOCA, it is implied that the doses to control room operators (via filtered intake) due to this event are well within the 10CFR20 limits (i.e., less than 5 Rem WB and 30 Rem Thyroid).

V. Previous Events & Extent of Condition:

To determine the extent of the condition, 870 non-stabilized and non-welded plugs in both the "A" and the "B" OTSG upper tubesheet were removed and the de-plugged tubes were inspected. One additional severed plugged tube was identified, A2-24, in the "A" OTSG. However, the in-service

tubes adjacent to A2-24 were inspected and were not degraded. The adjacent tubes were not impacted because the sever of A2-24 was captured within the top (15th) tube support plate (TSP). Including B66-130 and A2-24, twenty-nine plugged tubes (23 in the "A" OTSG and 6 in the "B" OTSG) were identified as being swollen.

There have been no recent events at TMI-1 where an in-service steam generator tube has been degraded to the extent that it might fail in the event of licensing basis accident. However, several years ago, in 1990, a sever of an in-service tube occurred during power escalation coming out of a refueling outage. That event, reported to the NRC in LER 90-005, occurred when a periphery tube located in the lane wedge area in the "A" OTSG, A77-1 experienced a 360 degree circumferentially oriented crack. The cause of the crack was reported to be environmentally assisted high cycle fatigue. The leaking tube, A77-1 and one other tube A78-28 in the lane wedge area were plugged and stabilized. In 1991 and 1993, the lane-wedge area tubes in the TMI-1 OTSGs were preventively sleeved and have successfully prevented similar tube failures.

VI. Identification of Root Cause

The degradation of four in-service tubes, B67-130, B66-131, B65-129 and B65-130, was caused by fretting from an adjacent plugged tube, B66-130, that had severed at the upper tubesheet face. The high velocity secondary steam flow caused the severed tube B66-130 to wear against the four neighbor tubes, thus causing tube damage. The extent of the damage to the in-service tubes caused B66-131 to fail its structural performance criterion and B65-130 to fail both the structural and accident leakage performance criteria. Additionally, during the course of investigating the B66-130 failure, it was determined that tube A2-24 was severed within the 15th tube support plate (TSP). However, this tube severance did not cause damage to adjacent in-service tubes, because the circumferential break in that tube was captured within the TSP.

The investigation concluded that the severance of plugged tube B66-130 was a result of the combined effects of 1) the swelling of the tube into the top TSP due to plug leak-by resulting in, 2) decreased damping of the tube at the 15th TSP causing high cycle fatigue due to a high secondary flow velocity region compounded by 3) existing OD IGA at the upper tubesheet face causing a local area of stress concentration. The discussion and conclusions that follow summarizes the findings from the preceding sections of this report and supports the stated root cause.

Tube B66-130 was removed from the steam generator for laboratory analysis. Initial visual, dimensional and SEM analyses have been completed. The results indicate that the tube hydraulically expanded 72 to 74 mils beyond nominal tube inner diameter (ID). The expanded tube diameter was larger than the tubesheet bore and larger than the tube support plate drill hole and broach hole diameters. Scanning electron microscopic examination of the fracture surface revealed the presence of OD-IGA at depths ranging from a few grains deep to 20% through wall. The area of OD-IGA transitioned to an area of Stage 1 high cycle fatigue, with an estimated depth of approximately 50% through wall. The remainder of the wall thickness, approximately 50%, failed by Stage 2 high cycle fatigue. There were isolated areas of ductile tearing near the ID surface in the Stage 2 fatigued area. The Stage 1 fatigue area was covered by a relatively thick deposit and required cleaning to reveal characteristic of Stage 1 fatigue. The thicker deposits in the Stage 1 area are indicative of slower long-term crack propagation. The Stage 2 fatigue area was relatively clean, with fatigue striations readily observed after minor cleaning of the fracture surface. The fatigue striations of the Stage 2 area are indicative of faster crack propagation.

Industry experience and laboratory testing have determined that a plugged tube may become over-pressurized resulting in the tube swelling if the mechanical plugs leak by and fill the tube with water. Changes to the plug sealing due to plug re-rolling or plug replacement and/or the leaking plug may trap

the water in the tube. The water in the tube thermally expands and causes pressure buildup during plant heat-up and operation if there are no throughwall defects in the tube or leakage back out through the plug.

Mechanical tube plugs are not designed to be leak tight and low levels of leakage can be expected. The TMI-1 de-plugged tube inspection results show that the tubes that were affected by over-pressurization and expansion due to plug leakage were of two (2) categories:

- tubes that contained plugs that were repaired by re-rolling in the 1980's (e.g., Westinghouse Alloy 600 rolled plugs) were found to be affected.
- tubes that had the upper tube end FTI Alloy 600 rolled plugs replaced with FTI Alloy 690 rolled plugs were found to be affected.

Tubes with installed FTI Alloy 690 rolled plugs in both the upper and lower tube ends were not affected by plug leak-by and over-pressurization.

The extent of swelling of tube B66-130 is consistent with pressures that can be expected with a plugged tube that contains water. Therefore, the tube was restrained at least to some degree by the tube support plates and also a significant expansion transition was created at the upper and lower tubesheet secondary faces. The expansion transition created a geometric stress riser in the tube at the upper tubesheet and 15th tube support plate interface, thus creating a condition that may be conducive to defect propagation or leading to flow induced vibration fatigue.

Sensitized alloy 600 steam generator tubes are susceptible to OD-IGA. Laboratory testing and examination of tubes removed from operating steam generators have demonstrated this. Eddy current examination of tubes deplugged during the current TMI 1 outage identified two (2) tubes that had OD-IGA present in locations that were not identified to have indications of OD-IGA present at the time of plugging. Scanning electron microscopic examination of tube B66-130 found OD-IGA present at various depths up to 20% through wall. The overpressurization and the resulting expansion of plugged tube B66-130 further aggravated this condition.

Flow induced vibration (FIV) and fatigue analyses were performed for an unrestrained tube and for a tube restrained at the top tube support plate at the radial location of the failed tubes. The secondary side steam flow at the upper tubesheet increases with radial distance from the center of the bundle. The lane/wedge tubes and peripheral tubes experience the highest flow velocity in the top tube support plate span. Under normal conditions when the tube is not restrained at the top tube support plate, the tube is fluid elastically stable. When a peripheral tube is restrained at the top support plate such that the tube's damping ratio is decreased, the tube is conservatively predicted to become marginally fluid elastically unstable, thus potentially reducing its fatigue life. The tube is analytically predicted to fail at the upper tubesheet or the top support plate location due to fatigue within a short time period, dependent on the degree of tube restraint.

The tube sample analysis results from the B66-130 tube pull supports the root cause that the tube sever was a result of a combined effect of tube restraint and stress from pressurized tube swelling, low amplitude high cycle fatigue and the presence of an existing OD flaw. The pressurized tube swelling with resulting increased 15th tube support plate tube restraint or decreased damping was a necessary precursor to this failure. Based upon the tube pull analysis and inspection results of other plugged tubes, it is concluded that an individual causal factors alone (increased tube diameter due to plug leak by pressurization, FIV or OD IGA) would not lead to a severance of a plugged tube. All three would need to be present.

Corrective Actions:

The following actions have been completed to prevent recurrence of damaging of in-service tubes by the circumferential severance of plugged tubes:

1. 870 mechanically (rolled) plugged tubes (tubes that did not have welded plugs) had their upper tubesheet plug removed, were de-watered, had an Alloy 690 rolled plug installed and were stabilized from the upper tubesheet through the 14th tube support plate.
2. Plugged tubes with a welded plug (or selected mechanical (rolled) plugs) but not stabilized were surrounded by tubes that were plugged with an Alloy 690 rolled plug and stabilized unless the tube was already surrounded by a welded stabilized plugged tube or sleeved tube.
3. 27 of 29 swollen tubes found in the inspections of the 870 tubes were stabilized with full-length stabilizers. The two (2) exception tubes, B150-14 and B66-130, were surrounded with plugged, stabilized tubes.

* The Energy Industry Identification System (EIIS), System Identification (SI) and Component Function Identification (CFI) Codes are included in brackets, [SI/CFI] where applicable, as required by 10 CFR 50.73 (b)(2)(ii)(F).