January 9, 2002

Mr. Mark E. Warner
Vice President - TMI Unit 1
AmerGen Energy Company, LLC
P.O. Box 480
Middletown, PA 17057

SUBJECT: THREE MILE ISLAND UNIT 1 - NRC INSPECTION REPORT 50-289/01-012

Dear Mr. Warner:

On November 29, 2001, the NRC completed a Special Inspection of your Three Mile Island facility. The enclosed report documents the inspection results which were discussed on November 29, 2001, with you and other members of your staff.

The Special Inspection Team examined activities related to the discovery of a previously plugged once-through steam generator tube that circumferentially severed during the previous operational cycle. The activities inspected by the Special Inspection Team were those conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, the team identified no licensee performance issues associated with the severed plugged steam generator tube which caused wear degradation to adjacent in-service tubes in the "B" steam generator. Your staff conducted a thorough extent-of-condition review. They appropriately identified other plugged tubes that exhibited some of the characteristics that were precursors to the severed tube. Your corrective actions to stabilized these plugged tubes, or to surround these tubes with other stabilized tubes, were also appropriate.

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Sincerely,

/RA/

Wayne D. Lanning, Director
Division of Reactor Safety

Docket No. 50-289
License No. DPR-50

Enclosure: Inspection Report 50-289/01-012

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No:	50-289
License No:	DPR-50
Report No:	50-289/01-012
Licensee:	AmerGen Energy Company, LLC (AmerGen)
Facility:	Three Mile Island Station, Unit 1
Location:	P.O. Box 480 Middletown, PA 17057
Dates:	October 29, 2001 to November 29, 2001
Inspectors:	M. Modes, Team Leader C. Khan, Senior Materials Engineer A. Smith, Materials Engineer C. Smith, Resident Inspector J. Trapp, Senior Reactor Analyst C. Dodd, NRC Contractor
Approved by:	David Lew, Chief Performance Evaluation Branch Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000289/2001-012, on 10/29/2001 through 11/29/2001; AmerGen Energy Company, LLC (AmerGen), Three Mile Island Station, Unit 1, Special Inspection Team.

The inspection was conducted by a region-based inspector, a resident inspector, two staff members of the Office of Nuclear Reactor Regulation (NRR), and was supported by a regional senior reactor analyst. The inspection was implemented in response to the degradation of tubes adjacent to a previously plugged tube which separated from the upper tube sheet of once-through steam generator “B”. This inspection provided facts to NRR for their assessment of AmerGen’s investigation and root cause evaluation of the tube failure, evaluation of the structural integrity of the tubes impacted by the separated tube, and corrective actions and extent of condition analysis. This inspection also provided facts to support NRR’s determination of possible generic implications of the tube failure.

No findings were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609 “Significance Determination Process” (SDP). Findings for which the SDP does not apply are indicated by “No Color” or by the severity level of the applicable violation.

<u>Inspector Identified Findings</u>

Cornerstone: Barrier Integrity

The team did not identify any licensee performance issues associated with the severed plugged tube that caused wear degradation to adjacent in-service tubes in the “B” steam generator. An independent review, by the NRC, of past eddy current test results, developed during previous refueling outages by the licensee, indicated there were no apparent wear indications present. The team did not identify any precursors which should have caused the licensee to take actions prior to the tube severing.

Based on the team’s review, the licensee’s extent-of-condition investigation was appropriate. The licensee identified other plugged tubes that exhibited some of the characteristics that were precursors to the severed tube in the “B” steam generator. The corrective actions taken by the licensee in response to these tubes were also appropriate. The licensee de-plugged and performed inspections of tubes previously plugged, and stabilized the upper portions of these tubes. For tubes that were plugged and not inspected, the licensee stabilized the surrounding tubes to prevent potential wear degradation if an uninspected tube were to sever. Additionally, the licensee stabilized the full length of all the tubes that exhibited hydraulic expansion along the full length of the tube with the exception of tubes B150-14, B66-130, and A133-11. All these tubes were surrounded by plugged and stabilized tubes.

Report Details

Background

Summary of Plant Event

On October 20, 2001, during performance of the fourteenth refueling outage (1R14) steam generator eddy current inspections, the licensee identified four tubes in the “B” once-through steam generator (OTSG), which exhibited signs of outside diameter (OD) wear. The previous eddy current inspection, performed during the thirteenth refueling outage (1R13), had not identified wear in these tubes. The maximum depth of tube wear observed in the four active tubes was estimated by eddy current examination to range from 37% to 92% through-wall. The overall length of the wear scars on the four tubes ranged from approximately 2.8 inches to 8.3 inches.

The pattern and location of the wear led the licensee to remove the hot-leg tube plug in tube B66-130, located in the center of the pattern of wear, in order to insert a remotely operated camera and perform a visual inspection of the tube. The visual inspection confirmed tube B66-130 was circumferentially severed at the secondary face of the upper tube sheet (UTS). As a consequence of being severed, the free end of tube B66-130 oscillated in response to steam flow past the tube which caused tube B66-130 to impinge on surrounding tubes. Based on eddy current data, the wear degradation of the four tubes was estimated as follows:

- Tube B67-130 - 41% maximum through-wall.
- Tube B66-131 - 62% maximum through-wall.
- Tube B65-129 - 37% maximum through-wall.
- Tube B65-130 - 92% maximum through-wall.

To investigate the severity of the wear indications, the licensee performed in-situ pressure testing of some tubes and removed the degraded portion of tubes B65-130 and B66-131, and the lower portion of severed tube B66-130 for destructive examination. As a result of the pressure tests performed on the tubes with wear indications, the licensee determined that tubes B67-130 and B65-129 met the structural performance criterion of three times normal operating differential pressure (3ΔP). 3ΔP is a steam generator tube integrity limit discussed in the Nuclear Energy Institute’s (NEI’s) guidelines, NEI 97-06, “Steam Generator Program Guidelines.” 3ΔP is a limit acceptable to the NRC and allows a licensee to conclude a steam generator tube has adequate structural integrity with sufficient safety margin. The basis for 3ΔP is that a steam generator tube must be able to hold, as a minimum, a hydrostatically induced pressure of three times the normal operational differential pressure across the tube wall.

Based on the hydrostatic tests, the licensee determined tube B66-131 did not meet the structural limit of 3ΔP. The licensee concluded however the tube would probably remain intact during a main steam line break (MSLB) or feed water line break (FWLB) accident. The hydrostatically induced burst pressure of tube B65-130, however, was near the differential pressure that would be observed during a MSLB or FWLB. As a consequence the licensee concluded tube B65-130 was worn to such a depth that it might not have remained intact during a MSLB or FWLB.

On October 29, 2001, the NRC initiated a special inspection in response to this event. The special inspection independently developed facts about the licensee's investigation, root cause evaluation, evaluation of the structural integrity of the tubes impacted by the separated tube, corrective actions, and extent of condition analysis. The special inspection also independently developed an understanding of the circumstances surrounding the separation of the plugged tube and provided facts to support NRR's determination of possible generic implications of the tube separation. The special inspection developed a characterization of the risk significance of the tube separation. The special inspection also focused on possible regulatory compliance or performance deficiencies that may have contributed to the separation of this tube.

TMI Steam Generator Description

Steam generators transfer heat from the reactor coolant system to the secondary system via conduction through the steam generator tubes. Three Mile Island Unit 1 has two vertical, straight tube and shell, once-through steam generators (OSTGs). Each steam generator contains 15,531 Inconel-600 tubes. Each tube is about 56 feet long and its ends are inserted into holes drilled into two 24-inch thick carbon steel tube sheets at the top and bottom of the steam generator. Additional support for the tubes are provided by 15 tube support plates that are spaced between 36 and 46 inches apart. The lowest tube support plate is the first tube support plate and the uppermost tube support plate is the fifteenth support plate.

Reactor coolant flow is from the upper steam generator head through the inside of the tubes to the lower steam generator head. Secondary feed water enters the center of steam generator and flows downward in the annulus region between the inside of the steam generator shell and the tube bundle wrapper. At the lower tube sheet, the flow turns radially inward and flows up the inside of the wrapper and around the tubes. Before it reaches the upper tube sheet, the feed water transforms into dry superheated steam. When it reaches the upper tube sheet, the super heated steam turns radially outward. The area of radial outward flow under the upper tube sheet represents the highest steam velocity that impinges on the steam generator tubes. The steam then turns downward in the annulus region between the steam generator shell and wrapper and flows out the steam nozzles to the turbine.

The outer two or three rows of the upper tube support plate have drilled holes. Drilled holes support the steam generator tubes concentrically with a gap of 0.637 - 0.642" between the tube and the support plate. The remainder of the steam generator support plates have broached holes. Broached holes support the steam generator tube at three locations equally spaced around the periphery of the tube with a gap of 0.640 - 0.646" between the tube and the support plate. Broached holes allow steam or liquid to flow along the tube through the support plate while drilled holes divert the flow. This was done because the designers were concerned that wet steam could travel up the periphery of the steam generator bundle and exit the generator. By introducing drilled holes at the periphery they turned the possibly wet steam into the primary flow for further mixing and heating before it could exit the generator. The drilled hole, because it has a tighter concentric gap, contributes higher damping to a hydrostatically expanded tube than does a broached hole. This leads to lower instability ratios in tubes clamped by drilled holes because they are less free to move and consequently have a higher probability of failure.

Tube B66-130 was plugged in 1986 with an alloy-600 mechanical rolled plug in the hot-leg and a ribbed alloy-600 plug in the cold-leg because the licensee identified a form of tube degradation called Inner Diameter Intergranular Attack (ID-IGA) above the fifth tube support plate (TSP). The degree of ID-IGA required that the tube be removed from service by inserting a plug into each end of the tube. At the time the tube was originally plugged, no degradation at the upper tube sheet (UTS) was identified. Because the industry experienced primary water stress corrosion cracking (PWSCC) of alloy-600 steam generator tube plugs, the licensee, during the twelfth refueling outage in 1997, replaced tube B66-130's alloy-600 hot-leg tube plug with a mechanical rolled plug made of alloy-690; an alloy more resistant to degradation.

4. OTHER ACTIVITIES [OA]

4OA3 Event Follow-up (93812)

.1 Once Through Steam Generator Tube Failure Investigation

a. Inspection Scope

The inspectors interviewed AmerGen personnel including the Eddy Current Level III, steam generator system engineer, Framatome steam generator design engineer, and other steam generator vendor personnel. The inspectors reviewed the results of the eddy current examination for tubes B67-130, B66-131, B65-129, and B65-130 and the remote camera video tapes for tube B66-130. The inspectors reviewed the metallurgical examination data and metallography for the harvested tube segments and the fracture face of B66-130. The inspectors reviewed historical records of the steam generators including engineering analysis for the previous sulfite intrusion, plug design information, stabilizer design and installation information, and independently plotted tube inspections on tube sheet maps for each generator.

The independently developed information provided the team with an understanding of the causes and conditions surrounding the separation of tube B66-130 and the affects it had or could have had on the surrounding tubes. The team also developed an understanding about the conditions preceding the tube separation, the steam generator and associated systems response, the overall equipment performance, precursors, human factors considerations, quality assurance considerations and radiological considerations. The inspectors also interviewed AmerGen's staff to ascertain what communications with other utilities, with similarly designed OTSGs, were occurring to identify potential generic safety concerns in a timely manner.

b. Findings

As a consequence of this inspection, the team did not identify any licensee performance issues. The licensee conducted an appropriate extent of condition review. The licensee removed and inspected 657 plugs in the A steam generator and 225 plugs in the B steam generator. Those tubes with plugs that could not be removed were caged in by other tubes that were stabilized. It was more efficient to surround them with tubes that had stabilizers in them, as a barrier to their possible movement, than to try and remove the plug and stabilize them individually. In some cases groups of tubes were caged in by stabilized tubes.

The tubes were inspected via eddy current and examined for water. The inspections identified an additional tube (tube A2-24), which had circumferentially separated. Tube A2-24's sever was captured within the top tube support plate and was not fretting against adjacent tubes. The adjacent tubes showed no evidence of wear.

In addition to circumferentially severing, tubes A2-24 and B66-130 showed evidence of tube expansion or swelling. In total, twenty three previously plugged tubes in OTSG “A” and six previously plugged tubes in OTSG “B” showed swelling. One of the additional tubes, B150-14, contained an axial through-wall separation, as well as swelling, indicating that the pressure inside the tube had been high enough to cause the tube wall to separate. This type of plugged tube failure does not allow the tube to oscillate and, as a consequence, the probability of fretting and wear on adjacent tubes is greatly reduced.

To investigate the severity of the wear indications, the licensee performed in-situ pressure testing of tubes B66-131, B65-129, and B67-130 and laboratory pressure testing of B65-130. The licensee also removed portions of tubes B65-130, B66-130, and B66-131 above the 15th tube support for destructive examination. As a result of the pressure tests performed on the tubes with wear indications, two tubes challenged the design basis structural performance criteria for steam generator tubes. The burst pressure for tube B65-130 was near the differential pressure that would be observed during a main steam line break (MSLB) or feed water line break (FWLB) accident. As a consequence the licensee concluded tube B65-130 was worn to such a depth that it might not have remained intact during a MSLB or FWLB. Tube B66-131 exhibited a burst pressure that the licensee determined did not meet the regulatory structural limit of $3\Delta P$, although the licensee concluded the tube would probably remain intact during a MSLB or FWLB.

.2 Root Cause Analysis

a. Inspection Scope

Based primarily on the laboratory evaluation of the harvested steam generator tubes, and industry experience with plugged tube failures, the licensee concluded the failure of tubes A2-24 and B66-130 was caused by a combination of effects starting with plug leak-by.

It is postulated that because the mechanical tube plugs are not designed to be leak-tight, the plugs allowed in-leakage at lower generator temperatures. As the generator rises in temperature, the plugs became more securely seated in the tube and prevent the egress of the captured primary side fluid. As the generator temperature continues to rise, the fluid, having no escape, expands inside the tube applying increasing hydraulic pressure on the tube wall. The hydraulic pressure causes the tube to expand along its length.

As a consequence of the hydraulic expansion the tube becomes restrained in the support plate and in the tube sheet. In areas of high velocity, turbulent flow, such as the periphery of the upper tube sheet area, the tubes are designed to resist the induced forces by flexing along their length. The movement of the tube in this manner allows the tube to resist flow-induced vibration and extends the fatigue life of the tube. When a peripheral tube is restrained in the top support plate and top tube sheet, the tube's damping is increased which decreases its instability ratio; a ratio used to indicate its susceptibility to flow-induced vibration. As a consequence of locking the tube in this manner, it is conservatively predicted to become marginally fluid elastically unstable; thus potentially reducing its fatigue life. The tube is analytically predicted to fail due to fatigue, dependent on the degree of tube restraint.

Tube B66-130 also showed evidence of outside diameter intergranular attack (OD-IGA) which may have contributed to shortening the resultant fatigue life of the locked tube. The OD-IGA acts as a precipitator of fracturing. Based on tube pull analysis and inspection results of other tubes, the licensee concluded an individual causal factor alone (restraining the tube due to increased tube diameter, flow induced vibration or OD IGA) would not lead to a plugged tube severing.

b. Findings

The licensee presented their root cause during a public meeting with the Office of Nuclear Reactor Regulation on November 9, 2001. Data provided at the meeting was consistent with the NRC's independent on-site verification. This information was an input to NRR for its review of the generic implications and other potential causal factors. At the time this inspection concluded, the NRC had not identified any significant issues with the licensee's root cause analysis.

.3 Prior Opportunities for Identification

a. Inspection Scope

The inspectors reviewed the licensee's root cause evaluation to determine whether the licensee identified how long the tube failure causal factors existed and whether there were opportunities for prior identification of the problem. The inspectors reviewed the type of metallographic examination the licensee was undertaking to determine that a sufficient amount of information was being developed from the samples taken from the generators.

The inspectors also utilized an expert eddy current contractor to independently assess the eddy current data for the current and prior outages to determine if there were prior opportunities to identify the tube wear. The contractor reported the results of his review to the inspectors as part of the inspection effort. The contractor independently analyzed eddy current data supplied by the licensee in order to determine when signals might have been present to such a degree that an analyst employed by the licensee would have noted them. The contractor reviewed differential and absolute data from 2001 and 1999 at 200, 400, and 600 KHz for tubes B65-129, B65-130, B67-130 and B66-131. The contractor also reviewed the calibration data for the eddy current techniques and the calibration data for the wear characterization.

The inspectors also reviewed previous eddy current data by reviewing screen images of indications contained for the tube and reviewed previous written eddy inspection reports to determine if prior opportunities existed to identify the separated tube. The inspectors reviewed steam generator primary-to-secondary side leakage rates for the previous outage to determine if there was an increase in secondary leakage that could have been attributed to the severed tubes. The team reviewed records of steam generator loose parts monitoring to determine if the fretting of tube B66-130 had been captured by the monitoring system.

b. Findings

As a consequence of this inspection, the inspectors did not identify any licensee performance issues associated with the severed plugged tube that caused wear degradation on adjacent in-service tubes in the “B” steam generator. An independent review, by the NRC, of past eddy current test results, developed during previous refueling outages by the licensee, indicated there were no apparent wear indications present. The team did not identify any precursors which should have caused the licensee to take actions prior to the tube severing.

.4 Corrective Actions

a. Inspection Scope

The inspectors reviewed the corrective actions identified in the licensee’s root cause evaluation report to determine whether they addressed the causal factors. Additionally, the inspectors reviewed whether the corrective actions had been prioritized with consideration of the risk significance. The inspectors reviewed procedures and quality records related to the installation of stabilizers in the steam generator tubes. The inspectors monitored the licensee’s inspection of the steam generators to assure the extent of condition was captured.

b. Findings

As a consequence of this inspection, the team did not identify any licensee performance issues. The corrective actions taken by the licensee in response to degraded, previously plugged tubes were appropriate.

The licensee removed plugs, at the hot leg, from tubes with a mechanical plug and without a stabilizer, in OTSG “A” and “B”, in order to evaluate the condition of each tube. A stabilizer is essentially a braided stainless steel cable with an outside diameter only slightly smaller than the inside diameter of the steam generator tube that it is slipped into. The stabilizers are long enough to reach approximately down into the 14^{th} tube support plate. The stabilizers are fixed in the tube by crimping to a plug at the upper end that is, in turn, mechanically rolled into the upper tube sheet plate. The end of the cable that is resting inside the 14^{th} tube support plate has a tapered cap on it. The cable is stiff enough to prevent the tube, if it were to come free, from moving around. The steam generators contained 609 stabilized tubes in OTSG “A” and 145 stabilized tubes in OTSG “B” before the 1R14 outage commenced.

The licensee determined there was a remote possibility that unstable liquid flow at the lower end of the steam generators could cause the swollen tubes to separate. As a conservative step the licensee inserted full length stabilizers into these tubes with the exception of tubes B150-14, B66-130, and A133-11. All these tubes were surrounded by plugged and stabilized tubes. B150-14 could not be stabilized because it contained a stuck eddy current probe from a previous inspection, B66-130 could not be stabilized because it was harvested.

.5 NRC Risk Assessment of the Steam Generator Tube Separation

a. Inspection Scope

The inspectors worked with the regional senior risk analyst (SRA) and the licensee's risk analyst to assess the risk significance of the separated tube.

b. Findings

The Revised Oversight Program (ROP) significance determination process (SDP) was not used to determine the safety significance of this issue because improper licensee performance did not contribute to steam generator tube degradation. As a consequence no color was assigned to this issue. A risk assessment was performed, however, to determine the increase in core damage frequency (CDF) and large early release frequency (LERF) caused by the degraded tubes.

There were no actual consequences caused by the degraded steam generator tubes because no tubes failed. If a main steam line or feedwater line failed (MSLB/FLWB) however, one of the steam generator tubes was degraded sufficiently that the pressure transient caused by an MSLB/FWLB could have induced a rupture. The licensee's risk assessment assumed the steam generator remained vulnerable to this type of tube failure for an entire year even though the licensee's best estimate for the tube separation was approximately 6 months before it was discovered. Based on this assumption the licensee calculated the change in CDF to be approximately 1E-5/year. The change in LERF was approximately equivalent to the change in CDF. The increase in CDF is approximately equivalent to the normal risk associated with operating the plant at full power for 3 months. The increase in LERF is approximately equivalent to the normal risk of operating the plant for 5 years. The NRC conducted an independent risk assessment which confirmed the licensee's results.

4OA6 Meetings

.1 Exit Meeting Summary

On November 29, 2001, the special inspection team presented their inspection results to Mr. M. Warner and other members of the licensee's staff. During the inspection, the team reviewed one proprietary vendor study which was returned to the licensee. The team verified that the inspection report does not contain proprietary information.

ATTACHMENT

Chronology of Events

1986 During a refueling outage steam generator eddy current inspection, an indication of Inner Diameter Intergranular Attack (ID-IGA) was identified above the fifth tube support plate in the tube span area. As a consequence, tube B66-130 was plugged with an alloy-600 mechanical rolled plug in the hot-leg and a ribbed alloy-600 plug in the cold-leg.

1997 Because the industry experienced primary water stress corrosion cracking (PWSCC) of alloy-600 steam generator tube plugs, the licensee replaced tube B66-130's alloy-600 hot-leg tube plug with a mechanical rolled plug made of alloy-690; an alloy more resistant to degradation.

1999 The in-service tubes surrounding plugged tube B66-130 was examined by eddy current during a refueling outage. No indications of wear were observed.

2001 On October 20th, during performance of refueling outage steam generator eddy current inspections, the licensee identified four tubes in the “B” once-through steam generator (OTSG), which exhibited signs of outside diameter (OD) wear. These tubes were B67-130, B66-131, B65-129 and B65-130. The maximum depth of tube wear observed in the four active tubes was estimated by eddy current examination to range from 37% to 92% through-wall. The overall length of the wear scars on the four tubes ranged from approximately 2.8 inches to 8.3 inches.

On October 29th, the NRC initiated a special inspection in response to this event. The special inspection independently developed facts about the licensee’s investigation, root cause evaluation, evaluation of the structural integrity of the tubes impacted by the separated tube, corrective actions, and extent of condition analysis. The special inspection also independently developed an understanding of the circumstances surrounding the separation of the plugged tube and provided facts to support NRR’s determination of possible generic implications of the tube separation.

In November 2001, the licensee identified an additional tube (tube A2-24), which had circumferentially separated. This tube was located within the 15th support plate in the A steam generator. Because, the location of the tube severance was captured within the top tube support plate, the severed tube was not fretting against adjacent tubes. In addition, tube B150-14 contained an axial through-wall separation. This type of plugged tube failure does not allow the tube to oscillate and, as a consequence, the probability of fretting and wear on adjacent tubes is greatly reduced.

On November 9th, the licensee presented a summary of their inspections and repair to NRR staff at a meeting in Rockville, MD. This meeting is documented in NRC’s letter “Summary of November 9, 2001, Meeting with AmerGen Regarding TMI-1 Steam Generator Severed Tube Root Cause (TAC MB3305),” dated November 21, 2001.

PERSONS CONTACTED

<u>Licensee</u>	
Mark Warner*	VP TMI Unit 1
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David Lew	Branch Chief Performance Evaluation
John Rogge	Branch Chief Projects Branch 7
Daniel Orr	Senior Resident Inspector TMI
Cheryl Khan*	Senior Materials Engineer
April Smith*	Materials Engineer
Caius Dodd*	NRC Contractor

*****attended exit meeting via telephone

DOCUMENTS REVIEWED

FSP-FP-002(83) “Rolled Plug Installation Hands On”
AD-TM-101, Attachment 3, “50.59 Applicability Form”
Framatome Letter of 11/16/01 “Clarifying Full Length Stabilizer Qualification”
General Maintenance Procedure 1401-4.8 “Install/Remove B&W Rolled Mechanical OTSG Tube Plugs and Stabilizers.”
Stabilizer Rolled Plug PN 1196122-002, Wire Rope Stabilizer Assembly PN 1227163-007 QA Data Package 23-1265630-00.
TMI-1 Inspection Degradation Assessment and Condition Monitoring Checklist for 1R14, dated October 8, 2001.
Licensee Event Report No. 2001-003-0 “Degraded OTSG Tube.”
Calculation C-1101-224-E220-074, “Risk Evaluation due to degraded OTSG Tube B65-130 Found During the T1R14 Refueling Outage.
5015346-00 “TMI 1R14 Root Cause Analysis of Severed Tube B66-130 and Condition Monitoring/Operational Assessment of Adjacent Tubes.”

LIST OF ACRONYMS USED

FWLB	Feedwater Line Break
ID-IGA	Inside Diameter Intergranular Attack
MSLB	Main Steam Line Break
NEI	Nuclear Energy Institute
OD-IGA	Outside Diameter Intergranular Attack
OTSG	Once Through Steam Generator
PWSCC	Primary Water Stress Corrosion Cracking
TSP	Tube Support Plate
UTS	Upper Tube Sheet