September 12, 2002

Mr. John L. Skolds, President
and Chief Nuclear Officer
Exelon Nuclear
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: THREE MILE ISLAND NUCLEAR STATION, UNIT 1 (TMI-1), RE: ONCE-TROUGH STEAM GENERATOR (OTSG) SEVERED PLUGGED TUBE EVENT FOLLOW-UP (TAC NO. MB3305)

Dear Mr. Skolds:

On November 9, 2001, AmerGen Energy Company, LLC (AmerGen or the licensee), representatives met with the Nuclear Regulatory Commission (NRC) staff to present its root-cause assessment of the plugged tube severance in the OTSGs at TMI-1 discovered during the 2001 refueling outage. On November 21, 2001, the NRC staff issued its meeting summary of that meeting (accession no. ML013240523). Enclosure 4 of the meeting summary contained a number of NRC follow-up questions to which you responded by letters dated November 26, 2001, and January 18, 2002. The enclosure to this letter contains the NRC staff's review of your responses. Based on its review, the NRC staff concludes that, in general, your responses support your root-cause determination presented at the November 9, 2001, meeting. However, with respect to the analytical models for flow-induced vibration, the NRC staff does not agree that the licensee has provided a sufficient basis to support its conclusion that, given the conservatism of the analysis, a fluid-elastic stability margin (FSM) of 1.0 is appropriate as a design requirement. To properly support this conclusion, the licensee should demonstrate that any additional alternating stress associated with the stable fluid-elastic activity (due to excitation which begins before the FSM is reduced to 1.0) remains acceptable, or alternatively, the licensee can show that when the calculated FSM equals 1.0, that the actual FSM is sufficiently higher such that no fluid elasticity is taking place. The NRC staff notes that further industry analysis of the FSM design value and generic implications of the TMI-1 severed plugged tube event is ongoing. This completes our effort under TAC number MB3305.

If you have any questions, please contact me at 301-415-1402.

Sincerely,

/RA/

Timothy G. Colburn, Senior Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-289

Enclosure: Review of AmerGen's Responses

cc w/encl: See next page

September 12, 2002

Mr. John L. Skolds, President
and Chief Nuclear Officer
Exelon Nuclear
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: THREE MILE ISLAND NUCLEAR STATION, UNIT 1 (TMI-1), RE: ONCE-TROUGH STEAM GENERATOR (OTSG) SEVERED PLUGGED TUBE EVENT FOLLOW-UP (TAC NO. MB3305)

Dear Mr. Skolds:

On November 9, 2001, AmerGen Energy Company, LLC (AmerGen or the licensee), representatives met with the Nuclear Regulatory Commission (NRC) staff to present its root-cause assessment of the plugged tube severance in the OTSGs at TMI-1 discovered during the 2001 refueling outage. On November 21, 2001, the NRC staff issued its meeting summary of that meeting (accession no. ML013240523). Enclosure 4 of the meeting summary contained a number of NRC follow-up questions to which you responded by letters dated November 26, 2001, and January 18, 2002. The enclosure to this letter contains the NRC staff's review of your responses. Based on its review, the NRC staff concludes that, in general, your responses support your root-cause determination presented at the November 9, 2001, meeting. However, with respect to the analytical models for flow-induced vibration, the NRC staff does not agree that the licensee has provided a sufficient basis to support its conclusion that, given the conservatism of the analysis, a fluid-elastic stability margin (FSM) of 1.0 is appropriate as a design requirement. To properly support this conclusion, the licensee should demonstrate that any additional alternating stress associated with the stable fluid-elastic activity (due to excitation which begins before the FSM is reduced to 1.0) remains acceptable, or alternatively, the licensee can show that when the calculated FSM equals 1.0, that the actual FSM is sufficiently higher such that no fluid elasticity is taking place. The NRC staff notes that further industry analysis of the FSM design value and generic implications of the TMI-1 severed plugged tube event is ongoing. This completes our effort under TAC number MB3305.

If you have any questions, please contact me at 301-415-1402.

Sincerely,

/RA/

Timothy G. Colburn, Senior Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-289

Enclosure: Review of AmerGen's Responses

cc w/encl: See next page

DISTRIBUTION

PUBLIC	PDI-1 R/F	SRichards	RLaufer	TColburn	OGC
MO'Brien	EMurphy	BPlatchek, RGN-I	ACRS	LLund	

ACCESSION NO.: ML022460194

OFFICE	PDI-1\PM	PDI-2\LA	EMCB\SC	PDI-1\SC
NAME	TColburn	MO'Brien	LLund	RLaufer
DATE	9/9/02	9/9/02	9/9/02	9/11/02

OFFICIAL RECORD COPY

REVIEW OF AMERGEN'S RESPONSES TO

FOLLOW-UP QUESTIONS REGARDING THE ROOT CAUSE OF THE

PLUGGED TUBE SEVERANCE EVENT AT

THREE MILE ISLAND NUCLEAR STATION, UNIT 1

DOCKET NO. 50-289

1.0 INTRODUCTION

On November 9, 2001, AmerGen Energy Company, LLC (AmerGen or the licensee), met with the Nuclear Regulatory Commission (NRC) staff to present its root-cause assessment of the plugged tube severance in the once-through steam generators (OTSGs) at Three Mile Island Nuclear Station, Unit 1 (TMI-1). This plugged tube severance was discovered during the 2001 refueling outage steam generator (SG) inspection. On November 21, 2001, the NRC staff issued its meeting summary of this meeting (accession no. ML013240523). Enclosure 4 of the meeting summary documented a list of NRC follow-up questions from the meeting. AmerGen responded to these questions by letters dated November 26, 2001 and January 18, 2002. The NRC staff's review of these responses is documented herein.

2.0 BACKGROUND

On October 20, 2001, AmerGen reported the finding of a severed plugged tube (66-130) in the "B" OTSG at TMI-1. The tube was severed at the secondary (lower) face of the upper tube sheet (UTS). Under the action of superheated steam cross flow, the cantilevered section of tube below the severance contacted adjacent tubes which were still in service (i.e., not plugged) causing wear damage to these tubes. One of these tubes was determined to have a residual burst pressure capacity that was less than three times normal operating pressure and that was marginal with respect to main steam line break (MSLB) pressure. Additional factual circumstances associated with this occurrence include the following:

- The OTSGs at TMI-1 employ Alloy 600 MA tubing and 15 carbon steel tube support plates (TSPs). The tubing penetrations in the TSPs are generally of the broached, quatrefoil design with the exception of the outermost three penetrations around the periphery of the 15th TSP which are round, drilled holes.

- Tube 66-130 is located in the peripheral zone of the bundle where the 15th TSP has drilled hole penetrations. Peripheral tubes are exposed to high cross flow velocities of superheated steam between the 15th TSP and the secondary face of the UTS.

- Tube 66-130 was plugged in 1986 due to intergranular attack (IGA) on the inner diameter (ID) surface near the 5th TSP. No degradation was detected at that time by bobbin or 8X1 array surface coil probes at the secondary face of the UTS.

- The UTS plug in tube 66-130 was replaced in 1997.

- Tube 66-130 was swollen (enlarged diameter) along its length. The tube outer diameter (OD) is nominally 0.625-inch. The measured OD was 0.664- to 0.710-inch at the fracture face. This swelling exceeded the UTS and 15th TSP hole diameters causing the tube to be clamped at these locations.

2.1 Summary - Licensee's Postulated Failure Mechanism

- Subsequent to plugging in 1986, water leaked past the UTS plug into tube 66-130.

- Water was present in the tube when the UTS plug was replaced in 1997. This created a seal, trapping the water in the tube.

- During plant heatup, the water in the tube expanded and was not able to escape, leading to high pressure inside the tube and resultant tube swelling. The amount of swelling exceeded the diameter of the UTS and 15th TSP tube penetration holes. This swelling resulted in the tube being clamped at the 15th TSP and UTS, causing a significant reduction in the damping coefficient at the supports.

- Tube 66-130 is located in the peripheral region of the tube bundle. The uppermost span of this tube, between the 15th TSP and the UTS, is subject to high cross flow velocity super heated steam. The reduction in damping coefficient associated with swelling of the tube reduced the stability ratio from a nominal value of 2.9 to 1.1. A value of 1.0 or less is indicative of a fluid-elastic instability under which the vibration amplitude is limited only by the presence of adjacent structures.

- With a stability ratio of 1.1, tube 66-130 did not quite reach the point of instability. However, at a stability ratio of 1.1, some structural-fluid interaction was taking place such that the vibration amplitude significantly exceeded the nominal amplitude associated with turbulent- and vortex-induced vibration.

- The increased vibration amplitude significantly increased alternating stress in the tube relative to nominal. In addition, a mean stress was introduced as a result of the swollen tube having to neck down where it enters the UTS. This mean stress acted to reduce the number of cycles necessary to initiate fatigue for a given alternating stress level.

- The increased alternating stress and mean stress led to fatigue crack initiation and propagation. ODIGA (10 to 15% through-wall) was present at the location of fatigue crack initiation and assisted in the initiation process.

- Fatigue crack propagation continued until the remnant of the tube cross section failed ductilely, resulting in complete severance of the tube.

- The severed tube was now free to deflect and to impact or rub against adjacent tubes causing wear damage on these tubes just below the secondary face of the UTS.

2.2 Laboratory Exam Results - Pulled Tube Specimens

The licensee harvested a section from each of 3 tubes for laboratory examination, including tube 66-130, which was the plugged tube which severed, and two adjacent tubes which were damaged as a result of the severance of tube 66-130. The removed sections extended from the secondary face of the UTS (corresponding to the fracture surface for tube 66-130) to just above the 15th TSP.

NRC staff's questions 1 through 4 in Enclosure 4 of the November 21, 2001, meeting summary relate to the results of the laboratory examinations of these specimens and how these results relate to the licensee's postulated root cause conclusions.

Based on its review of the licensee's responses to these questions, the NRC staff concludes that the laboratory results are qualitatively consistent with the licensee's postulated failure mechanism. For tube 66-130, the licensee provided fractographs. Much of the fracture surface was smeared or burnished by the rubbing of the fracture surfaces against each other during the failure process. In addition, additional fracture surface was lost as a result of abrasion when the severed tube rubbed against the adjacent tubes. However, it was possible to pick out several regions of ODIGA fatigue damage (stage I and stage II), and ductile tearing.

The licensee was unable to answer the NRC staff's question regarding the level of alternating stress suggested by the fatigue striations or the number of alternating stress cycles to failure. This information would have been useful for validating the Framatome ANP flow-induced vibration (FIV) model. The licensee cited a number of difficulties including the limited number of clear features on the fracture surface, the presence of multiple initiation sites, the lack of crack propagation to applied stress data for strain hardened material, and a lack of information concerning the preferential bending direction or whether orbital motion was taking place.

ODIGA for the non-severed tube specimens was described as being typical for OTSGs in the steam space, approximately 1 or 2 grains deep. This ODIGA was not observed at the wear scars, indicating that this attack is of long standing and not highly active. ODIGA along the severed tube was also about 1 or 2 grains deep with occasional penetrations 3 or 4 grains deep (about 5% of the initial wall thickness). However ODIGA penetrations at the fracture surface ranged from 10 to 15% of the initial wall thickness. The licensee did not discuss why ODIGA is somewhat deeper near the fracture location than has generally been observed, or what role plugging the tube or swelling of the tube might have played in this regard. The NRC staff notes that in the absence of quantitative information on the stress levels that existed at the fracture location, it is not known whether the ODIGA is a necessary condition for initiating fatigue for the conditions that existed for tube 66-130, although it clearly reduces the time to fatigue crack initiation.

2.3 Denting

In response to the NRC staff's question 5, the licensee stated that denting is present in about 500 tubes at the secondary face of the UTS, about 1000 tubes at the secondary face of the lower tube sheet (LTS), and in 7 tubes at TSPs. Dent voltages range from a maximum of about 90 volts at the tube sheets and to about 4.5 volts at the TSPs. The licensee did not discuss these results further with respect to their relevancy to the severed tube occurrence.

The NRC staff notes that large dents could clamp the tubes and thus reduce damping from nominal values. The NRC staff believes denting is unlikely to have contributed to the severance of tube 66-130, since the tube was clamped anyway at the 15^{th} TSP and UTS as a result of swelling. However, the NRC staff believes that large dents could potentially affect the damping coefficients at the affected intersections of both plugged and unplugged tubes. From the information provided, there may be little likelihood at TMI-1 that a tube in the peripheral region of the bundle would contain sufficient denting to be clamped at both the 15^{th} support and the UTS.

2.4 Analytical Models for Flow-Induced Vibration

NRC staff's questions 6, 8, and 9 requested information on the analytical models used to predict cross flow velocities and the tube FIV response, including the major input parameters, model uncertainties, and how they were qualified. The NRC staff requested this information, not to perform a critical review of these models, but to have a general understanding of the models and assumptions used to assess the FIV response for nominal design conditions and the off-nominal conditions associated with the tube severance. Question 7 requested information on the flow-induced vibration mechanisms affecting three specific regions of the OTSGs; the peripheral zone of the uppermost span (region 1), the lane region of the uppermost span (region 2), and the peripheral zone of the lowermost span (region 3).

The licensee's vendor, Framatome ANP, uses a modified version of the Electric Power Research Institute's (EPRI's) PORTHOS code to perform detailed three-dimensional thermal hydraulic analyses of OTSGs, including cross flow velocities. The licensee enclosed a paper entitled, "Adaption of PORTHOS to the Once Through Steam Generator," with a description of the model and its verification. OTSGs have a triangular pitch, and NRC staff's question 6 inquired as to whether the PORTHOS model could account for different tube bundle flow resistance in the different azimuthal directions. The licensee did not answer this question directly, but stated that the tube bundle hydraulic resistance does not have significant azimuthal dependance. The smaller of the directional porosity values has been input for both the radial and azimuthal porosity in the PORTHOS model. This maximizes the velocities and is, therefore, conservative.

The licensee cites the 2000 Edition of the American Society for Mechanical Engineers (ASME) Standard and Guide on the Operation of Nuclear Plants, Part 11, Appendix A, and the 1998 Edition of the ASME Boiler and Pressure Vessel Code, Section III, Appendix N, Paragraph N-1300, as defining the potential FIV mechanisms affecting heat exchanger tube bundles; fluid-elastic instability, turbulence-induced vibration, and vortex-induced vibration. The Framatome ANP FIV analyses are described by the licensee as closely following the methodologies in ASME Code, Section III, Appendix N, for each of the FIV mechanisms. For each mechanism, the FIV model considers the entire length of a tube between the secondary face of the UTS to the secondary face of the LTS. The tube is assumed to be fixed at the tube sheet secondary faces. This ignores the small clearances that nominally exist at these locations which the license describes as being conservative since tube motion in the small annulus would increase damping and reduce vibratory response. The NRC staff notes that this assumption may actually be realistic for cases where denting exists at the tube sheets or if the tube is swelled. The tube is nominally assumed to be pinned at the 15 TSP locations. This allows axial and rotational movement of the tube relative to the TSPs, but not horizontal movement.

Cross flow velocities and fluid densities applied to the FIV model are taken from the PORTHOS results as a function of elevation and tube radial location. For each assumed radial location of the tube, these inputs are applied along the entire length of the tube model. The FIV responses are normally evaluated for different radial locations, in 3-inch radial increments.

The FIV model considers the axial load in the tube. Nominally, this load is compressive at 100% full power, varying as a function of radial location. Compressive load acts to reduce natural frequency, reducing the fluid-elastic stability ratio. For a swollen plugged tube, the Poisson's effect can lead to 1000 lbs tensile load in the tube which, in and of itself, increases natural frequency and increases the fluid-elastic stability ratio. For this off-nominal case, an induced axial force of both zero and 1000 lbs was considered.

A Connor's constant value of 3.3 is in the Framatome ANP model when evaluating fluid-elastic stability. The licensee states that this value provides a lower bound for 90% of published data and is lower than Framatome ANP's measured value based on the OTSG tube array geometry. In general, a lower value of the Connor's constant leads to a more conservative analysis.

Framatome ANP test data are said by the licensee to support a conclusion that most of the damping for a nominal OTSG tube originates from the relative motion between the tube and TSPs acting to dissipating energy. Damping ratios increase with increasing vibration amplitude.

Assumed damping ratios (values are proprietary) are based on tests with single tubes, tests of selected tubes in an actual OTSG, recent tests involving tubes with nominal TSP clearances, tightly supported tubes with no TSP clearances, and swollen tubes with pressure inside the tube, with pressure released and with air and water inside the tubes. The assumed values were determined such as to give lower 90% confidence estimates of fluid-elastic stability ratios. The damping ratios for swelled tubes are less than for non-swelled tubes. In addition, a lower bound value was also used for swollen plugged tubes as part of the root-cause analysis. Everything else being equal, damping ratios used in the lower part of the bundle (saturated two-phase flow) were 2% higher than in the upper (superheated steam) region of the bundle.

Assumed damping ratios for turbulence-induced FIV and vortex-induced FIV were smaller than were assumed for the fluid-elastic analyses, reflecting much smaller vibration amplitude than what would exist at the threshold of fluid-elastic instability.

The licensee's response contained a good summary description of uncertainties associated with some of the key input parameters. The licensee concludes that the FIV models are conservative. However, the NRC staff does not agree that the licensee has provided a sufficient basis to support its conclusion that given the conservatism of the analyses, a fluid-elastic stability margin (FSM) of 1.0 is appropriate as a design requirement. The difficulty is that fluid-elastic excitation begins to take place before the point of instability (i.e., FSM = 1.0) is reached. This excitation can cause a significant increase in vibration amplitude, alternating stress, and perhaps failure before FSM is reduced to 1.0. This behavior is not currently modeled in the licensee's analysis. To support the conclusion, the licensee should demonstrate that the additional alternating stress associated with the stable fluid-elastic activity remains acceptable or, alternatively, when the calculated FSM equals 1.0, the actual FSM is sufficiently higher than this value to assure that no fluid elasticity is taking place.

2.5 FIV Analyses - Results

NRC staff's questions 10, 11, and 12 related to the results of the FIV analyses.

For inservice tubes under nominal operating conditions, the FSM for tubes in region 1 (i.e., the upper span near the periphery, but away from the lane) is 2.9. Tube 66-130 was in region 1. Region 2 (i.e., the upper span near the periphery and adjacent to the lane) actually has the highest cross flow velocities, and the nominal FSM is somewhat lower, 1.9.

Plugged tubes which are not swollen generally have somewhat higher FSMs than inservice tubes since plugged tubes have negligible axial load. However, when the plugged tube is swollen, the FSM in region 1 is reduced to 1.1 to 1.3. The NRC staff notes that the FSM for a plugged swollen tube in region 2 is not described in the licensee's report, but would presumably be lower than for region 1, perhaps less than 1.0.

For the lower most span of tubing (region 3), the limiting FSM occurs about 52 inches from the center of the bundle. For in-service tubes at this location, the FSM is 4.0. For a plugged swollen tube at this location, the FSM reduces to 3.0.

For inservice tubes under normal operating conditions, the turbulence-induced FIV results in low alternating stress and negligible fatigue usage in regions 1, 2, and 3. Interestingly, region 3 is most limiting for turbulence-induced FIV. For plugged, swollen tubes, alternating stress is actually reduced due to the tensile force introduced by the swelling process. The same is true for vortex-induced FIV, although the alternating stress levels appear to be higher in regions 1 and 2 than from turbulence-induced FIV.

2.6 Response of Swelled, Plugged Tubes with Axial Burst

NRC staff question 13 inquired as to the potential for an axial burst of a plugged, swelled tube to propagate to a circumferential failure. The NRC staff asked this question because axial burst of plugged, swelled tubes is not uncommon, at TMI-1 or elsewhere. The licensee stated that the opening associated with such a burst would be limited to less than 90 degrees because of the limited energy of the trapped water inside the plugged tube. The licensee modeled such a burst as a 90-degree circumferential crack, 100% through-wall. The fatigue life at such a crack location would be limited by vortex-induced vibration to 44 years. As the NRC staff understands the licensee's explanation, the circumferential crack model is justified on grounds that the length of the axial burst would likely be small compared to the tube span length, thus having little impact on the overall tube stiffness. The fluid-elastic response of the tube would not be significantly affected. That a tube is or is not subject to significant fluid-elastic behavior would not be significantly affected by the presence of an axial burst. A swelled, plugged tube containing an axial burst could still sever at the UTS or 15th TSP if subject to significant fluid-elastic behavior. The NRC staff notes that the licensee's explanation is consistent with the fact that prior to the TMI-1 occurrence, plugged tubes which have been observed to contain axial bursts have generally not been observed to have been severed. Presumably, these tubes were stable, even after swelling and subsequent axial rupture.

2.7 Detection Threshold for Wear Scars

NRC staff's question 14 related to the detection threshold for wear scars. Framatome ANP recently performed a study on a sample set consisting of 11 pulled tubes and 20 specimens with machined flaws representative of OTSG wear at TSPs. This study indicated a detection threshold of less than 16% depth.

2.8 Wear Rate

Four of the six tubes adjacent to tube 66-130 (the severed tube) contained long wear scars just below the secondary face of the UTS. The 2 tubes on the lee side of tube 66-130, relative to the direction of cross flow, sustained the most damage with eddy current indicating maximum depths of 92 and 62%. The 2 tubes located perpendicular to tube 66-130, relative to the direction of cross flow, had wear scars measuring about 40% through-wall. The 2 adjacent tubes immediately upstream of tube 66-130 did not exhibit indications of wear. Thus, it appears that once severed, tube 66-130 was mainly vibrating against adjacent tubes downstream, with some lesser vibration perpendicular to the direction of flow.

The licensee estimated the wear rate for the most limiting tube by comparative analysis. Tests have established that the nominal wear rate for tubes inside the carbon steel TSP is 3.1 mils/5 EFPY (effective full power years). This wear rate was adjusted to reflect wear rates induced by the severed tube. The biggest adjustment applied by the licensee was to account for increased sliding velocity afforded by the larger gap between impacting surfaces (tube-to-tube gap vs. tube-to-TSP gap). Another adjustment accounted for a higher wear coefficient, based on recent test data) for an Inconel tube rubbing on another Inconel tube rather than on the carbon steel TSP. In addition, the licensee applied an adjustment to reflect an impact factor of 2. With these adjustments, the licensee estimates a wear rate of 224 mils/EFPY.

The NRC staff notes that an upper bound growth rate estimate is conservative from the standpoint of estimating the probability that an occurrence such as occurred at TMI-1 could lead to a tube rupture occurring as an initiating event, but is generally not conservative from the standpoint of assessing risk associated with such occurrences. This is because most of the risk from accident scenarios involving ruptured or leaking tubes stems from ruptures or leaks occurring as consequential events (NUREG-1570, "Risk Assessment of Severe Accident-Induced Steam Generator Tube Rupture," March 1998). Risk from consequential tube ruptures varies directly with the time period of vulnerability when the tube could rupture if challenged during an accident scenario. The time period of vulnerability varies inversely with flaw growth rate.

The licensee states that its 224 mils/EFPY estimate is accurate only to an order of magnitude, citing a large uncertainty associated with the sliding velocity. The NRC staff believes this to likely be a large over estimate. The licensee's estimate implies that such wear scars grow from zero to the point of leakage or rupture in only 6 weeks. For a 24-month fuel cycle, there is virtually no chance that such a flaw would first be detected by inspection. Given that the tube at TMI-1 with the deepest wear scar was found by inspection just short of the point of incipient ligament tearing and leakage, then the best estimate likelihood of finding a similar such flaw by inspection is greater than 50-50. Based on an assumed detection threshold of 16% and a 24-month inspection interval, the NRC staff estimates that a flaw with a depth growth rate of 27 mils/year would have a 50-50 chance of being detected by inspection. This estimate compares with an estimated flaw depth growth rate of 45 mils/year at Ginna where a severed plugged tube wore against an adjacent live tube which led to a steam generator tube rupture event in 1982.

3.0 CONCLUSIONS

The licensee's responses are supportive of the root-cause evaluation provided to the NRC staff at the November 9, 2001, meeting. However, the licensee has not provided a sufficient basis to support its conclusion that an FSM of 1.0 is appropriate as a design requirement. This does not change the NRC staff's assessment of the licensee's root cause, but the staff notes that further industry analysis of the FSM and other generic implications is ongoing. The industry is expected to complete the initial phase of its assessment of the generic implications of the TMI-1 plugged tube severance later this summer and to meet with the NRC staff at that time to present its findings.

Principal Contributor: E. Murphy

Date: September 12, 2002

Three Mile Island Nuclear Station, Unit No. 1

cc:

Site Vice President - Three Mile Island Nuclear Station Unit 1
AmerGen Energy Company, LLC
P. O. Box 480
Middletown, PA 17057

Senior Vice President Nuclear Services
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

Vice President - Mid-Atlantic Operations Support
Exelon Generation Company, LLC
200 Exelon Way, KSA 3-N
Kennett Square, PA 19348

Senior Vice President -
Mid Atlantic Regional Operating Group
Exelon Generation Company, LLC
200 Exelon Way, KSA 3-N
Kennett Square, PA 19348

Vice President -
Licensing and Regulatory Affairs
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

Regional Administrator
Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Chairman
Board of County Commissioners
of Dauphin County
Dauphin County Courthouse
Harrisburg, PA 17120

Chairman
Board of Supervisors
of Londonderry Township
R.D. #1, Geyers Church Road
Middletown, PA 17057

Senior Resident Inspector (TMI-1)
U.S. Nuclear Regulatory Commission
P.O. Box 219
Middletown, PA 17057

Director - Licensing - MId-Atlantic Regional Operating Group
Exelon Generation Company, LLC
Nuclear Group Headquarters
Correspondence Control
P.O. Box 160
Kennett Square, PA 19348

Rich Janati, Chief
Division of Nuclear Safety
Bureau of Radiation Protection
Deparment of Environmental Protection
Rachel Carson State Office Building
P.O. Box 8469
Harrisburg, PA 17105-8469

Three Mile Island Nuclear Station Unit 1
Plant Manager
AmerGen Energy Company, LLC
P. O. Box 480
Middletown, PA 17057

Regulatory Assurance Manager - Three Mile Island Unit 1
AmerGen Energy Company, LLC
P.O. Box 480
Middletown, PA 17057

John F. Rogge, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Michael A. Schoppman
Framatome ANP
Suite 705
1911 North Ft. Myer Drive

Rosslyn, VA 22209

Three Mile Island Nuclear Station, Unit No. 1

cc: continued

Vice President, General Counsel and Secretary
Exelon Generation Company, LLC
300 Exelon Way
Kennett Square, PA 19348

Dr. Judith Johnsrud
National Energy Committee
Sierra Club
433 Orlando Avenue
State College, PA 16803

Eric Epstein
TMI Alert
4100 Hillsdale Road
Harrisburg, PA 17112

Correspondence Control Desk
Exelon Generation Company, LLC
200 Exelon Way, KSA 1-N-1
Kennett Square, PA 19348

Manager Licensing - Oyster Creek and Three Mile Island
Exelon Generation Company, LLC
Nuclear Group Headquarters
Correspondence Control
P.O. Box 160
Kennett Square, PA 19348