March 11, 2019

Scott D. Portzline
Eric Epstein
Three Mile Island Alert
4100 Hillsdale Road
Harrisburg, PA 17112

Victor McCree
Executive Director for Operations
United States Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: 10 CFR 2.206 Petition on Apparent Violation of 10 CFR 50.9

Mr. McCree,

On behalf of Three Mile Island Alert ("TMI Alert"), I submit this petition pursuant to §2.206 in Title 10 of the Code of Federal Regulations (hereafter "10 CFR"). We petition the Nuclear Regulatory Commission ("NRC") to take enforcement action on regulation §50.59 in 10 CFR "Changes, tests and experiments" regarding an apparent design and manufacturing defect with the replacement steam generators at the Three Mile Island Unit 1 Nuclear Station (TMI) licensed to Exelon Generation Company LLC, the owner - Docket No. 50-289. The defects caused some of the steam tubes to become damaged by striking or rubbing against each other.

During normal operation, this does not appear to be a concern since the damaged steam tubes were taken out of service by plugging the ends. However, under reactor transient[1] conditions, this defect could result in a thermally induced steam generator tubes rupture event.[2] This in turn can result in a loss of coolant accident with core damage. "Steam generator ("SG") tubes ruptures are potentially risk-significant events because thermally induced SG tube failures caused by hot gases from a damaged reactor core can result in a containment bypass event and a large release of fission products to the environment."[3]

"Continuing development of the understanding of severe accidents raised the possibility that natural convection of steam and hydrogen through a degrading reactor core could impose heat loads threatening the integrity of steam generator tubes during accidents important to core

[1] "Reactor transient conditions" is defined as a change in the reactor coolant system temperature, pressure, or both, attributed to a change in the reactor's power output. For the purposes of this filing, the concern is for higher temperatures and pressures of the reactor coolant.
[2] "...natural convection of steam and hydrogen through a degrading reactor core could impose heat loads threatening the integrity of steam generator tubes...", Letter from the Advisory Committee on Reactor Safeguards to the U.S. NRC dated May 19, 2017 p1.
[3] NUREG-2195 p iii.

damage frequency such as those involving a station blackout ("SBO"). That is, core damage accidents initiated by other means could evolve to become containment bypass accidents."[4]

"Since publication of the Reactor Safety Study in 1975, it has been understood that accidents involving radionuclides through paths that bypass containment pose significant risk to the public health and safety. In bypass accidents, there is no mitigation of radionuclide release by natural and engineered processes in the reactor containment."[5]

It is from the three above citations that we (TMI Alert) stress the importance and urgency of this request for enforcement action including shutting down the reactor until the problem is understood and remediated.

**1. Violation and safety defect**

The analysis and testing used to confirm AREVA's new design of the Enhanced Once Through Steam Generators (EOTSGs) was insufficient when it came to identifying steam tube fluttering.[6] It is also unable to evaluate the extent of potential damage to the steam tubes caused by fluttering during reactor transient conditions. The analysis did state: "The only portions of the EOTSG significantly affected by the LBLOCA loads are the tube-to-tubesheet weld and tubes with potential wear degradations."[7]

But the tube-to-tube wear that took place during the first fuel cycle which had normal operating temperatures was not predicted. Therefore one can conclude that the adequacy of the methodology for determining the behavior of the steam tubes is also insufficient for evaluating the extent of tube damage or tube ruptures during a reactor transient. This inadequacy also raises the question of how many additional tubes will experience damage or rupturing during reactor transient conditions?

This assertion of design error is further backed by the statement of the NRC's Branch Chief of Nuclear Reactor Regulation:

> *"So, you know obviously for fuel and some other components, we look heavily at design. So, for the steam generators, the thermal hydraulic conditions on the secondary side are not something that's within our standard review plan...But at*

[4] Letter from the Advisory Committee on Reactor Safeguards to the U.S. NRC dated May 19, 2017 p1.
[5] Loc. cit.
[6] Fluttering of the steam tubes refers to unwanted lateral oscillations perpendicular to the axis of the tubes. If the fluttering is severe enough, two tubes will contact each other causing damage at the point of contact.
[7] Letter from Exelon to the NRC dated January 29, 2010, "Summary Report for Qualification of EOTSG for LBLOCA." p2 ML100290764

*the moment, the standard review plan doesn't have us looking in detail at thermal hydraulic conditions in the secondary side for the design basis."*[8]

Furthermore, no analysis of the tube fluttering behavior and the tube fluttering mechanism under reactor transient conditions with their associated higher temperatures and greater compressive loads on the steam tubes has been performed subsequent to learning that the thermal properties are contributing to the flutter.

Proper analysis and testing as required has not been performed on the new design used in the replacement steam generators installed at TMI Unit 1 in 2010 leaving the licensee, Exelon Generation Company LLC, in violation of its Reactor License. Regulation §50.59 states:

> (6) (2) A licensee shall obtain a license amendment pursuant to Sec. 50.90 prior to implementing a proposed change, test, or experiment if the change, test, or experiment would:
>
> - (i) Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the final safety analysis report (as updated);
> - (ii) Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component (SSC) important to safety previously evaluated in the final safety analysis report (as updated);
> - (iii) Result in more than a minimal increase in the consequences of an accident previously evaluated in the final safety analysis report (as updated);
> - (iv) Result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the final safety analysis report (as updated);
> - (v) Create a possibility for an accident of a different type than any previously evaluated in the final safety analysis report (as updated);
> - (vi) Create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the final safety analysis report (as updated);
> - (vii) Result in a design basis limit for a fission product barrier as described in the FSAR (as updated) being exceeded or altered;

Unexpected tube-to-tube wear[9] was found during the first outage and inspection of the new steam generators in October and November of 2011. Some of the steam tubes became

[8] Transcript of "Briefing on Steam Generator Tube Degradation" before the NRC Commissioners dated 2/7/2013, p23. ML13043A136

damaged when they struck or rubbed together caused by the fluttering of the tubes or banging into each other. This behavior was not expected and can lead to tube ruptures. The potential for this behavior was not discussed in the TMI Unit 1 updated Final Safety Analysis Report dated April 2014.

Analysis has been performed on steam tube loads during reactor transient conditions. But, that analysis did not examine the damaging behavior of steam tube flutter. In-situ pressure testing does not account for the fluttering behavior. The calculations for compressive and tensile loads also does not account for the fluttering behavior. Therefore, we conclude that a proper analysis and remediation is needed. Until remediated, this leaves Exelon out of compliance with its license.

## 2. The Defect

The steam generators steam tubes made from metal Alloy 690 are bowing under normal operating temperatures. This is because the steam generators are experiencing unexpected thermal expansion. This unwanted bowing was determined by the manufacturer AREVA to be attributable to three root causes:

Root Causes

1. The steam generators were not constructed as designed. Specifically, the preload tensile value was less than called for by the design specification. The result is that the tubes experience a higher compressive load than planned.[10]

2. The steam generator shell stays cooler than expected during normal operation. The steam tubes expand at a greater rate than the shell. The result is that the tubes experience higher compressive loads than planned for.[11]

3. The margin to buckling (bowing) was not conservative.[12]

[9] "Tube to tube wear (T-T) (not expected)." Three Mile Island Unit 1 Investigation of Tube-to-tube Wear, June 17, 2013, p4. ML13178A360

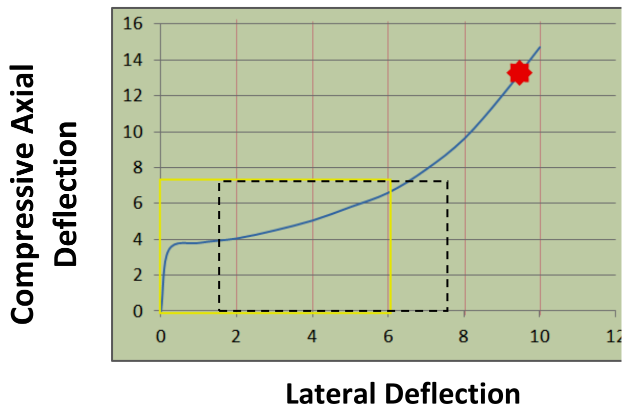
[10] "Tube preload less tensile than value in design analysis." Ibid. p13

[11] "EOTSG shell cooler than design value." Ibid.

[12] "The margin to buckling was non-conservative in design." Ibid.

The previous three points constitute a design error and a manufacturing error. They also call into question whether or not Exelon has met its obligation under the scope of "Regulation §50.59 section (6) (2)" as listed on the page 3 of this petition. TMI Alert asserts that the design changes and the material changes used for the replacement steam generators, along with the errors, constitute a violation of the "change" parameter of the regulation.

**Exelon's graph[13] on steam tube deflection + extended extrapolation**




Exelon's original graph is bound by the dotted lines on the lower left. The graph has been extended here by TMI Alert to extrapolate how great the lateral deflection or bowing would be if temperatures rose to 1100ºF degrees as marked by the red star. Normal operating temperatures are ~ 550ºF. An analysis is needed to determine the accuracy of this projection because the actual value is not known.

AREVA used an aggressive design intending to create steam more efficiently. The steam tubes are smaller in diameter and packed together tighter than in previous designs. The gap between the steam tubes is only 1/8 inch when the reactor is operating at normal temperatures.[14] It should also be noted that the tube walls are thinner than previous design. Therefore, these two conditions, created by design, make any tube-to-tube wear of somewhat greater concern regarding steam tube ruptures.

Steam generators act as a pressure and radiation barrier between the reactor coolant and the steam system. "This creates the possibility of a containment bypass situation, so they're of significant importance to us."[15] Steam generator tubes account for more than fifty percent of the primary pressure boundary surface of a Pressurized Water Reactor.[16] (TMI Unit 1 is a PWR.)

[13] Three Mile Island Unit 1 Investigation of Tube-to-tube Wear, June 17, 2013, p14. ML13178A360
[14] "The licensee also indicated that the tubes are in compression during operations (hot conditions) and that the gap between the tubes is 1/8 inches." Letter from the NRC to Exelon dated 12/28/2011. ML113550167
[15] Transcript of "Briefing on Steam Generator Tube Degradation" before the NRC Commissioners dated 2/7/2013, Statement of the NRC Reactor Systems Branch. p6
[16] https://www.ne.anl.gov/capabilities/cmm/highlights/sgt_integrity_program.html, p1.

The metal Alloy 690 has been known to cause safety related issues in components at other nuclear plants.[17] The only other nuclear plant (Arkansas Nuclear) to use the new AREVA steam generator design also experienced unexpected tube damage.[18]

### 3. Need for thorough Analysis

The steam tube flutter is actually created by the turbulent and chaotic flow [19] of the secondary system water between the tubes. Under higher temperatures, the energy of the flutter could become more damaging and chaotic. Also, "the SG geometry and the fluid flow rates in different SG designs can significantly influence the potential likelihood of consequential steam generator tube ruptures."[20]

"Although risk analyses have previously considered SG tube ruptures, they have typically not considered thermally induced consequential SG tube ruptures."[21]

The NRC's Advisory Committee on Reactor Safeguards concluded, "Initial concerns over bypass accidents focused on accidents initiated by a steam generator tube rupture, with additional failures that led to core degradation. Interfacing systems loss-of-coolant accidents were later recognized as other potential bypass accidents. Continuing development of the understanding of severe accidents raised the possibility that natural convection of steam and hydrogen through a degrading reactor core could impose heat loads threatening the integrity of steam generator tubes during accidents important to core damage frequency such as those involving a station blackout (SBO). That is, core damage accidents initiated by other means could evolve to become containment bypass accidents." [22]

The Electrical Power and Research Institute stated, "Potential contributing non-pressure primary loads created during accident conditions would be those bending loads resulting from dynamic conditions. Dynamic conditions include cross flow and other hydraulic and inertia forces from LOCA and seismic events. Both recirculating steam generators and once-through steam generators are subjected to these types of bending loads."[23]

---

[17] "The initial crack under fretting condition occurs at lower stress amplitude and lower cycles of cyclic loading than that under plain fatigue condition. The fretting damage, for example, can be observed in fossil and nuclear power plant, aircraft, automobile and petroleum chemical plants etc." A Study on Fretting Behavior in Room Temperature for Inconel Alloy 690, p. Abstract, http://adsabs.harvard.edu/abs/2006IJMPB..20.4303K

[18] "The characteristics of the tube-to-tube wear indications at ANO-1, including the depth and length are similar to those at TMI Unit 1." NRC Information Notice 2012-07: Tube-To-Tube Contact Resulting In Wear In Once-Through Steam Generators, p2.

[19] "Excitation forces created by either random turbulence of flutter." Three Mile Island Unit 1 Investigation of Tube-to-tube Wear, June 17, 2013, p4. ML13178A360

[20] NUREG-2195 p iii

[21] Ibid. p3-2.

[22] Letter from the Advisory Committee on Reactor Safeguards to the U.S. NRC dated May 19, 2017 p2.

[23] Steam Generator Integrity Assessment Guidelines Revision 3, Electric Power Research Institute,

“During a transient condition the thermal expansion can be significant....the largest tube axial tensile loads occur during postulated accident conditions....the resulting accident tube loads are plant specific.”[24]

The principal research engineer for Atomic Energy of Canada told U.S. NRC Commissioners, "So you get a situation where the more you have motion, the more you have hydraulic excitation. The more you have hydraulic excitation, the more you've got motion."[25]

Nuclear Regulatory Commissioner William D. Magwood stated that the thermodynamic behavior of the secondary flow has safety implications for steam tubes including fluid-elastic instability.[26] Fluid-elastic instability of tube bundles is defined as "when a tube bundle is subject to cross-flow with increasing velocity, it will come to a point at which the responses of the tubes suddenly rapidly increase without bound, **until tube-to-tube impacting** or other non-linear effects limit the tube motions. This phenomenon is known as fluid-elastic instability"[27]

An NRC Senior Level Advisor stated this about fluid-elastic instability, "That's a condition when the velocity of the water on the secondary side of the steam generator causes the tubes to become unstable and vibrate excessively in different modes of operation."[28]

TMI Alert contends that an analyses must be performed on the steam tubes for transient conditions based on direct input from the plant system thermal hydraulic analyses. There are newer modified software tools like those used to analyze the thermal hydraulics at San Onofre after the problems were discovered. Pressure testing and structural analysis are not able to predict or determine the fluttering behavior and the likelihood of steam tube ruptures. This fluttering under reactor transient conditions could cause the steam generators to self-destruct. We are also concerned about the effects harmonic resonance of the tube flutter.

Because TMI Unit 1 has already experienced unexpected steam tube damage, there appears to be a significant increased risk of an accident and a significant increase in the consequences as described in this filing.

---

October 2008, p3-6

[24] Ibid. p C-19

[25] Transcript of "Briefing on Steam Generator Tube Degradation" before the NRC Commissioners dated 2/7/2013, Statement of Michel Pettigrew. p88

[26] “But it [thermodynamic behavior] does have safety implications obviously because that leads to fluid elastic instability…,” Transcript of "Briefing on Steam Generator Tube Degradation" before the NRC Commissioners dated 2/7/2013, Statement of the NRC Reactor Systems Branch. p22.

[27] Flow Induced Vibration of Power and Process Plant Components: A Practical Workbook, M. K. Au-Yang 2001, Chapter 7, Fluid-Elastic Instability of Tube Bundles, abstract http://ebooks.asmedigitalcollection.asme.org/content.aspx?bookid=232&sectionid=39221296

[28] Transcript of "Briefing on Steam Generator Tube Degradation" before the NRC Commissioners dated 2/7/2013

Scott D. Portzline
Security Consultant to TMI Alert
3715 N. 3rd Street
Harrisburg, PA 17110
717-232-8863

Eric Epstein
Chairman of Three Mile Island Alert
4100 Hillsdale Road
Harrisburg, PA 17112
717-635-8615

## BACKGROUND [29]
## Steam Generator Tubes

"Steam generator (SG) tubes are small diameter, thin walled tubes that carry primary coolant through the primary to secondary heat exchangers. The SG tubes have a number of important safety functions. Steam generator tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied on to maintain the primary system's pressure and inventory. The SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In addition, as part of the RCPB, the SG tubes are unique in that they act as the heat transfer surface between the primary and secondary systems to remove heat from the primary system. This Specification addresses only the RCPB integrity function of the SG. The SG heat removal function is addressed by TS Section 3.4.

SG tube integrity means that the tubes are capable of performing their intended RCPB safety function consistent with the licensing basis, including applicable regulatory requirements.

Steam generator tubing is subject to a variety of degradation mechanisms. Steam generator tubes may experience tube degradation related to corrosion phenomena, such as wastage, pitting, intergranular attack, and stress corrosion cracking, along with other mechanically induced phenomena such as denting and wear. These degradation mechanisms can impair tube integrity if they are not managed effectively. The SG performance criteria are used to manage SG tube degradation."

note: In the 1979 TMI partial meltdown, the steam generators initially failed to remove heat from the reactor.

[29] Letter from NRC to Exelon dated 9/4/2012 p8 section 4-77. ML100480243