REVISED MEETING AGENDA DISCUSSION QUESTIONS

February 3, 2005

MEMORANDUM TO: Richard J. Laufer, Chief, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation (NRR)

FROM: Timothy G. Colburn, Senior Project Manager, Section 1 /***RA***/
Project Directorate I
Division of Licensing Project Management, NRR

SUBJECT: FORTHCOMING MEETING WITH AMERGEN ENERGY COMPANY, LLC (AMERGEN), REGARDING THE THREE MILE ISLAND NUCLEAR STATION, UNIT 1 (TMI-1), STEAM GENERATOR KINETIC EXPANSION INSPECTION ACCEPTANCE CRITERIA (TAC NO. MB6475)

DATE & TIME: Wednesday and Thursday, February 16-17, 2005
9:30 a.m. to 4:00 p.m., Wednesday, 9:00 a.m. to 4:00 p.m., Thursday

LOCATION: U.S. Nuclear Regulatory Commission
One White Flint North
11555 Rockville Pike
Rockville, Maryland
Room O-9B4 (February 16, 2005), Room O-10B4 (February 17, 2005)

PURPOSE: A working level meeting to discuss AmerGen's response to staff issues related to the TMI-1 steam generator kinetic expansion inspection acceptance and repair criteria report dated October 4, 2002, as supplemented August 16, 2004 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML022840503 and ML042370131).

CATEGORY 1:* This is a Category 1 Meeting. The public is invited to observe this meeting and will have one or more opportunities to communicate with the NRC after the business portion, but before the meeting is adjourned.

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* Commission's Policy Statement on "Enhancing Public Participation in NRC Meetings," (67 FR 36920), May 28, 2002

Participants from the NRC include members of the Office of Nuclear Reactor Regulation (NRR).

PARTICIPANTS:

NRC/NRR	AMERGEN
L. Lund	D. Distel
K. Karwoski	R. Freeman
T. Colburn	G. Navratil
E. Murphy	S. Leshnoff
C. Khan	et. al.
C. Liang	
R. Laufer	

Docket No. 50-289

Attachment: Meeting Agenda (ADAMS Accession Number ML050340182)

cc w/att: See next page

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DATE	2/3/2005	2/3/2005	2/3/2005

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MEETING AGENDA

FEBRUARY 16-17, 2005, CATEGORY 1 WORKING LEVEL MEETING WITH

AMERGEN ENERGY COMPANY, LLC (AMERGEN)

KINETIC EXPANSION (KE) INSPECTION ACCEPTANCE AND REPAIR CRITERIA

FOR THREE MILE ISLAND NUCLEAR STATION, UNIT 1 (TMI-1)

DOCKET NO. 50-289

The Nuclear Regulatory Commission (NRC) has reviewed the AmerGen October 4, 2002, and August 16, 2004 (the August 16 submittal was a complete revision of the October 4 submittal), submittals associated with the TMI-1 KE inspection, acceptance and repair criteria . The quality and clarity of the submittal has significantly improved such that the NRC staff has been able to review the proposal and identify a complete list of issues. Based on recent industry experience, NRC review of information related to similar alternate repair criteria and issuance of Generic Letter 2004-01, "Requirements for Steam Generator Tube Inspections," dated August 30, 2004, the NRC staff has identified concerns with the TMI-1 KE inspection, acceptance and repair criteria. The NRC staff is meeting with AmerGen to discuss these issues. The below agenda outlines the areas of concern that will be discussed during this working level public meeting.

1. Suitability of leaving circumferential flaws in-service
2. Leakage assessment including thermal hydraulic analysis model
3. Basis for assumption of no growth/no initiation of new flaws
4. Structural integrity assessment
   a. Basis for steamline break axial load
   b. Determination of the limiting accident
5. Consistency with recent industry experience
6. Inspection practices/techniques
7. Other issues

Specific issues under the above topics to be discussed include the following:

1. Suitability of Leaving Circumferential Flaws In-service

a. The large-break loss-of-coolant accident (LBLOCA) issue applies to TMI-1, (based on request for additional information (RAI) responses provided in the October 4, 2002, submittal) and circumferential flaws are of particular concern in this event due to the increase in axial loads. This issue is not currently addressed in the August 16, 2004, TMI-1 KE Report, ECR #02-01121, Revision 1, "Inspection Acceptance Criteria and Leakage Assessment Methodology for TMI OTSG [once-through steam generator] Kinetic Expansion Examinations," which is inconsistent with industry practice for joint repairs, therefore, this issue remains unresolved.

b. For a 0.64-inch long circumferential crack, what is the factor of safety against tube severance under a main steamline break (MSLB), based on elastic analysis? For this same crack, does the axial thermal stress behave as a primary or secondary stress? (For design in accordance with Section III of the American Society for Mechanical Engineers (ASME), *Boiler and Pressure Vessel Code* (Code), axial thermal stress is always considered secondary. For circumferentially cracked tubes, if the crack is large

enough such that deformation occurs largely at the crack rather than being relatively evenly distributed along the length of the tube, then the net section stress at this location is not "self limiting" and should be treated as primary. Industry representatives (e.g., Nuclear Energy Institute (NEI) have stated they are developing guidance for when thermal loads should be considered primary versus secondary). If the licensee has an alternative justification for the 0.64-inch circumferential crack criterion other than elastic analysis, the licensee is requested to provide that justification including a justification for the value of the safety factor assumed to the applied load.

c. Based on recent industry operating experience, circumferential flaws are likely to initiate in the KE expansion region at TMI-1. The NRC staff's concerns related to the "no growth" and "new initiation" statistical assessments have not been resolved and are discussed in more detail in Item 3 below.

d. No other domestic plant leaves circumferential flaws in service in the steam generator (SG) tube pressure boundary. Based on this and other issues associated with circumferential cracks (identified above), the NRC staff would like the licensee to discuss the suitability of its proposal to leave circumferential flaws in service.

2. Leakage Assessment Including Thermal Hydraulic Analysis Model

a. Did the leak tests, performed in support of this inspection/repair criteria, use deoxygenated water? Recent industry experience indicates tests not performed with deoxygenated water may be non-conservative.

b. The proposed leakage model assumes zero leakage from flaws located above the region of the tubing which is inspected and evaluated in accordance with the structural integrity criteria (i.e., the SG tube pressure boundary). This is inconsistent with industry practice for similar repair criteria. Recent industry experience indicates the leakage from this region may not be minimal, when assumed for all in-service tubes. Therefore, the assumption needs to be modified to reflect this.

c. Is the leakage assessment conservative for volumetric intergranular attack (IGA), given that the axial and circumferential extents (i.e., components) are independently assessed? Is there experimental and/or analytical evidence which indicates that summing the leak rates from projected axial and circumferential crack components of volumetric IGA indication give conservative leak-rate estimates?

d. Laboratory tests were performed by the licensee to develop a leakage reduction factor (LRF). Information from other sources indicates lower reductions in leakage due to pressure. 1) The licensee calculated the LRF by putting a clamp over an electro-discharge machined (EDM) notch and measuring the reduction in leakage. To account for internal pressure, the licensee used the zero-applied contact pressure results as the basis. Clarify how these adjustments (relative to the zero-applied contact pressure) were made and the basis for concluding they are conservative. 2) Discuss whether use of a notch is conservative when the results are applied to cracks/volumetric IGA given that cracks have much lower leak rates.

e. Based on the information submitted, the NRC staff understands the following: 1) Flaws are assumed to be 100% through-wall for the entire measured (via eddy current) extent for the structural analysis. 2) Flaws less than 67% through-wall are assumed not to leak for the leakage assessment. 3) Flaws exceeding 67% through-wall are assumed to be 100% through-wall for the entire measured (via eddy current) extent for the leakage assessment. Please confirm the above understanding. [Please note: Table

III-6 of the 2001 Steam Generator Tube Inservice Inspection Report for TMI-11(R14 SG Outage Report) implies that the length of assumed 100% through-wall crack length is less than the measured (via eddy current) crack length. Please clarify this discrepancy.]

Thermal Hydraulic Model for the Leakage Assessment

a. The NRC staff has determined that the thermal hydraulic model used for the leakage assessment is different than that used for the structural assessment and has not been reviewed by the NRC staff. Considering the resulting axial loads are significantly lower (i.e., 1310 pounds (lbs)) than those used for the structural assessment (i.e., 2400 lbs.), the NRC staff has concluded that the thermal hydraulic model used for the leakage assessment must be reviewed.

b. The leakage assessment is performed for a MSLB using revised tube loading conditions (based on use of a different thermal hydraulic model as discussed above). The loads on the tube for the revised MSLB analysis are lower than for other accidents (e.g., small and large break LOCA). As a result, it is not clear whether the MSLB is still the most-limiting accident in terms of assessing the consequences of leakage from these joints given the differences in loading conditions between the accidents and the different assumptions for assessing the radiological consequences of these accidents.

c. In the thermal hydraulic analysis for the leakage assessment, the licensee appears to have tried to maximize the cooldown rate to increase the axial tube loading. However, it is not clear whether this results in an overall conservative result given it may have decreased the differential pressure across the tubes (and the driving force for the leakage). In addition, it appears that the leakage assessment was performed based on the actual loads on the tube at various time intervals and the leakage over these intervals were summed. Regarding this approach, it is not clear how the licensee accounted for all of the uncertainties in all of the models (e.g., thermal hydraulics, tube material properties, PICEP, etc.) to ensure that the leakage estimates have high confidence (e.g., a 95% prediction interval at 95% confidence). Provide the details of how the leakage calculations are performed.

d. Please confirm that the methodology and input assumptions used for your MSLB analysis for generating inputs to define the OTSG tube load are consistent with that used in the MSLB analysis documented in Updated Final Safety Analysis Report (UFSAR), Section 14.1.2.9. Identify any deviations from the licensing basis methodology, analysis assumptions and initial conditions and provide proper justification for such deviations.

e. Provide the justification for why a reactor trip setpoint of 1900 psig plus a 30-psi error will result in a conservative calculation with respect to SG tube temperature for OTSG loads.

f. In the long-term analysis, operator actions are credited in the analysis. Please confirm that all operator actions credited in this analysis are consistent with the plant emergency operating procedures at TMI-1 and that the reactor operators are properly trained on the plant simulators for these operations. Justify that the time allowed for operator action is adequate, and has been verified on the plant simulator.

g. Please compare the transient curves between the new analyses for OTSG tube load and the licensing analysis in Section 14.1.2.9 of the UFSAR. Identify each

deviation and provide proper justifications (the NRC staff has noted quite a few differences).

h. In Section 14.1.2.9 of the UFSAR, it is concluded that the results of the analysis confirm that the maximum temperature differential that occurs in the OTSG does not produce excessive stress, and SG integrity is maintained. Discuss why this 100-second analysis supports such a conclusion and why your new analysis requires both a 10-minute duration and a long-term analysis to assess the SG tube integrity.

3. Assumption of No Growth/No Initiation of New Flaws

a. Statistical tests performed to determine whether flaws are growing compare data from the current outage to data from the prior outage. The NRC staff believes the licensee should use data from the outage during which the first rotating probe examination was performed of each KE. This would ensure that potential slow flaw growth rates would be more evident. It is requested that the licensee discuss its plans to perform the statistical tests in this manner.

b. An extreme value test is performed to identify possible outliers or erroneous data. Erroneous data is corrected prior to using that data in the subsequent statistical tests. Outliers (i.e., indications with large apparent growth rates) are used in the subsequent statistical tests. Industry experience indicates that when a population of flaws grow, some grow faster than others. Therefore, the NRC staff would like to discuss why the outliers are not, in and of themselves, considered evidence of flaw growth, and therefore, an invalidation of the no-growth assumption used to calculate the flaw acceptance criteria.

c. The NRC staff is not confident the threshold value of 0.05 new indications per KE examined is truly indicative of an active degradation mechanism. In addition, different criteria for circumferential and volumetric degradation may be appropriate since they are potentially two different populations. Industry guidance on this subject would indicate that one new crack results in a declaration of active degradation. Discuss why the size of the indication is not a consideration or why comparisons to prior data are not sufficient (i.e., if it cannot be seen with hindsight, it is new). Lastly, based on the information provided in Table B in Section 3.2.1.9 of the August 16, 2004, submittal, it could appear that degradation is active with an initiation rate of 0.03 indications per KE. Please provide a discussion of the above issues.

d. The licensee indicated that some KE indications "drop out", or disappear, each outage. These should be discussed in more detail including examples of several indications (e.g., largest, smallest, theory on reason for disappearance, etc.). If threshold-of-detection is ascribed to be the cause of disappearance, be prepared to discuss the criteria for the threshold-of-detection.

4. Structural Integrity Assessment

Basis for MSLB Axial Load

a. The licensee's August 16, 2004, report states that the 3140-lb. axial load corresponds to an axial membrane stress of 49.5 ksi (thousand pounds per square inch) and a design-basis tube strain of 0.16%. The licensee further states that tubes with a lower bound yield strength and nominal geometry will experience load relaxation (from 3140 lbs.) due

to yielding, resulting in an axial load of 2400 lbs. This load is used to determine the size of the needed "defect free zones" in the KE.

- The "design basis" tube strain of 0.16% was determined assuming that all tubes were behaving elastically (Topical Report BAW-10146). The corresponding axial membrane stress of 49.5 ksi exceeds the nominal yield strength of the tubing at an MSLB temperature of 235 degrees F. Had the actual stress/strain properties of the tubes been assumed in the licensee's analysis, rather than elastic properties, the resulting tube strain could exceed 0.16% since the tube bundle and tube sheet would provide less resistance to the tendancy of the SG shell (with temperature in the range of 520 to 575 degrees F) to expand axially relative to the tubes. If credit is taken for load relaxation in the tubes due to yielding (as is the case for TMI-1), why is it not necessary to also consider the corresponding increase in tube end displacements (and, thus, tube strain) when determining the axial loads in the tubes? What are the tube end displacements and tube strains under MSLB temperatures if a realistic distribution of stress/strain properties are assumed to exist within the tube population? What would be the effect on the axial loads and minimum required defect-free lengths assuming use of the realistic distribution of stress/strain properties?

- What factors of safety are applied to the axial loads to ensure the joints don't slip when determining the necessary size of the defect-free zones? What is the technical basis for the safety factors? [Note, a factor of safety of 1.0 is reasonable for thermal loads behaving as secondary, as acknowledged in the structural performance criteria in the latest Technical Specification Task Force (TSTF) submittal from NEI of the generic license change package and in the forthcoming revision to NEI 97-06, "Steam Generator Program Guidelines." But this safety factor criterion is based on the assumption of elastic analysis, recognizing that load relaxation will take place prior to failure. A safety factor of 1.0 is not appropriate if one is taking explicit credit for load relaxation, since components at the point of incipient failure under design-basis loadings would be contrary to ASME Code, Section III, and Section XI philosophy.]

b. The licensee states that the design-basis MSLB load for the SG tubes of 3140 lbs. was determined by assuming that all tubes remain fully elastic. It was necessary to adjust the results obtained for the high-yield strength tubes and greater wall thickness for consideration of minimum yield strength and nominal wall thickness tubes that may be present in the SGs. Additional details are required for the NRC staff to fully understand this adjustment.

- What are the dimensional tolerances for the nominal 0.625" diameter, 0.034 " thick tubing?

- What are the estimated nominal, upper bound, and lower bound yield strengths of the tubing at room temperature and at the temperature associated with the maximum MSLB load (i.e., 235 degrees F as reported in BAW-10146)?

- It is not evident to the NRC staff, based upon its review of BAW 10146, that the 3140-lb. MSLB load is based on a larger than nominal tube wall thickness as is suggested in the licensee's words above. For a nominal 0.625" outside diameter (OD) tube with a nominal 0.034" thick wall, the cross-sectional area of the tube is 0.0631 square inches. BAW-10146, Table 5-6, indicates that the 3140-lb. load is based on this same nominal cross-sectional area. Provide an explanation for this

apparent discrepancy. Provide a description of tube wall dimensions assumed in the calculation of the 3140-lb. MSLB load.

- The pullout resistance of the tube from the tube sheet is a function of the contact pressure caused by the expansion process, the effects of thermal tightening (differential thermal expansion between the tube and the tube sheet), tube internal pressure, and tube sheet bow. During a steamline break transient, the tube internal pressure and the tube temperature are changing. In addition, the yield strength of the tube changes with temperature. The yield strength of the tube affects the system response (e.g., the applied load due to load relaxation). It is not clear whether the analysis provided truly represents the most-limiting conditions of the transient. The licensee should confirm that the most-limiting point of the accident was evaluated for the most-limiting situation (high-yield strength tubing/low-yield strength tubing) using the most-limiting input parameters (lowest contact pressure/pullout resistance of any of the test data). This approach is consistent with how we have assessed other similar amendments. The goal is that all tubes have adequate integrity so worst-case assumptions are generally made.

Determination of the Limiting Accident

In determining the limiting accident, it is not clear what factors of safety were applied under all events considered (e.g., LOCA, normal operating, feedwater line break conditions, etc.). Question 4.a., above, focuses on the factor of safety used in assessing the MSLB accident; however, it is not clear whether another event may be more limiting if appropriate safety factors were used (this question assumes the correct safety factors were not applied).

5. Consistency with Recent Industry Experience

Analyses of kinetically expanded joints in other designed SGs have indicated that defect-free lengths greater than what is being proposed for TMI-1 are needed to ensure structural and leakage integrity. In addition, the contact pressures for the TMI-1 KEs appear to be significantly larger than those at other plants with KEs. Therefore, please compare and contrast the expansion process used at TMI-1 to the KE processes used at other plants to help the NRC staff understand the potential differences (e.g., joint tightness, resistance to cracking, etc.).

6. Inspection Practices/Techniques

a. On page 23 of the August 16, 2004, report, the licensee indicates that if "localized" degradation occurs at a KE, then the scope of the inspection will not be expanded to 100%. The specific example given was damage from a maintenance tool. If growth or new degradation is occurring, the scope should be expanded to 100%. It is not clear what other "localized" degradation could be occurring. Discuss the intent of this statement.

b. On page 23 of the August 16, 2004, report, the licensee states that if growth of existing degradation or initiation of new degradation in the KE region is detected, then an examination of 100% of the KEs in the affected generator(s) will be undertaken. However, the proposed statistical tests combine data from both SGs because there is a limited data population in the "B" SG. Therefore, the NRC staff assumes the 100% scope expansion would occur in both SGs. Please confirm this assumption.

c. The licensee states that the eddy current measurements always result in conservative overestimates of the flaw size. This is attributed to lead in and lead out affects and the flaw being small in comparison to the coil field. How did the licensee confirm that measurement uncertainty is not a function of flaw size (i.e., is there a flaw size beyond which the flaw size could be underestimated (at a 95% confidence level))? In addition, did the licensee confirm that the 95% confidence levels on uncertainty for cracks and for volumetric IGA (i.e., non-notch specimens) when analyzed separately from the notch data still result in overestimates of the flaw size?

d. Discuss the inspections performed of the parent tube/sleeve assembly in the upper tube sheet region. Describe the flaw acceptance criteria utilized for the portion of the sleeve/tube assembly located in the tube sheet.

7. Other Issues

a. The proposed reporting requirements should be supplemented to include the KE length and the tube sheet radius associated with each tube with degradation in the KE region.

b. Discuss the axial loads simulated during insitu pressure tests for the purposes of demonstrating structural and leakage integrity.

c. The number of tubes in Section 1.3.2.39 of the revised Updated Final Safety Analysis Report pages does not add up (i.e., the number of tubes in-service versus the number of tubes plugged do not correlate). Discuss this inconsistency.

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