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U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 2055

Three Mile Island, Unit 1
Operating License No. DPR-50
NRC Docket No. 50-289

Subject: Third Ten-Year Interval Inservice Inspection (ISI) Program
Response to Request for Additional Information
Risk-Informed Inservice Inspection Program
*Alternative to the ASME Boiler and Pressure Vessel Code*
Section XI Requirements for Class 1 and 2 Piping Welds

References:
1. Letter from M. P. Gallagher (AmerGen Energy Company, LLC) to U. S. Nuclear Regulatory Commission (USNRC), dated October 1, 2002
2. Letter from T. G. Colburn (USNRC) to J. L. Skolds (AmerGen Energy Company, LLC), dated May 28, 2003

In the Reference 1 letter, AmerGen Energy Company (AmerGen), LLC, submitted a proposed alternative to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components", requirements for the selection and examination of Class 1 and 2 piping welds. The Reference 2 letter provided a request for additional information. Attached is our response to this request.

If you have any questions, please contact us.

Very truly yours,

Michael P. Gallagher

Michael P. Gallagher
Director, Licensing and Regulatory Affairs
Mid-Atlantic Regional Operating Group
AmerGen Energy Company, LLC

Attachment – Response to Request for Additional Information

cc: H. J. Miller, Administrator, Region I, USNRC
USNRC Senior Resident Inspector, TMI
D. Skay, USNRC Senior Project Manager
File No. 02078

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## RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
## THREE MILE ISLAND, UNIT 1

References: 1. Letter from M. P. Gallagher (AmerGen Energy Company, LLC) to U. S. Nuclear Regulatory Commission (USNRC), dated October 1, 2002

**Question:**

1. In relief request RR-00-21 for implementing the RI-ISI program, the licensee stated that "In lieu of the evaluation and sample expansion requirements of EPRI TR-112657, Revision B-A, Section 3.6.6.2, 'RI-ISI Selected Examinations,' Three Mile Island, Unit 1 will utilize the requirements of Subarticle-2430, 'Additional Examinations,' which is contained in Code Case N-578-1." It should be noted that Code Case (CC) N-578-1 has not been approved by the NRC for generic use. The guidelines for sample expansion in CC N-578-1 are not consistent with the NRC staff's position as delineated in the EPRI TR-112657, Revision B-A, Report. The NRC staff considered that all required examinations due to sample expansion should be completed during the same refueling outage, and that the candidates for selecting elements in sample expansion should include all elements that are determined to be susceptible to the same root cause and degradation mechanism found in the initial samples. Therefore, for the evaluation and sample expansion, the licensee should follow the requirements of EPRI TR-112657, Revision B-A, Section 3.6.6.2, instead of CC N-578-1. Please revise your evaluation and sample expansion requirements, accordingly.

**Response:**

In the Reference 1 letter, we state that:

"If the additional required examinations reveal flaws or relevant conditions exceeding the referenced acceptance standards, the examination shall be further extended to include additional examinations.

(1) These examinations shall include all remaining piping elements whose postulated failure modes are the same as the piping structural elements originally examined.

(2) An evaluation shall be performed to establish when those examinations are to be conducted. The evaluation must consider failure mode and potential."

AmerGen Energy Company, LLC (AmerGen) will eliminate the requirement contained in item 2 with regards to performing an evaluation to establish when additional examinations are to be conducted. In lieu of item 2, AmerGen will consider all R-A category welds as ASME Class 1, and will follow the provisions of the 1995, through 1996 Addenda, version of the ASME Section XI Code, IWB-2430(b), with regards to the second sample expansion. The required additional examinations will be performed during the same outage that the relevant condition was detected.

Question:

2. The licensee identified in its RI-ISI Program that the following four augmented programs will not be subsumed into the RI-ISI Program and will remain unaffected:
   - Stagnant Borated Water Systems (IE Bulletin 79-17)
   - Service Water Integrity Program (Generic Letter (GL) 89-13)
   - Flow Accelerated Corrosion (FAC) (GL 89-08)
   - High Energy Line Breaks (USNRC Branch Technical Position MEB 3-1)

   a. The augmented program of High Energy Line Breaks was not referenced in the EPRI TR-112657, Revision B-A, Report. Discuss this augmented inspection program and its relationship to EPRI's RI-ISI methodology. What degradation mechanisms and systems are involved in this augmented program?

   b. The integration of augmented programs of Stagnant Borated Water Systems (IE Bulletin 79-17) and Service Water Integrity Program (GL 89-13) into the RI-ISI Program is discussed in EPRI TR-112657, Revision B-A, Report. Provide your reasons for not integrating these augmented programs into your proposed RI-ISI Program.

   c. Identify and discuss that the element numbers shown in Table 3 of your RI-ISI Program include the elements to be examined under the referenced augmented programs.

Response:

a. The augmented High Energy Line Break (HELB) inspection program is not addressed within the EPRI RI-ISI TR methodology. As such, this program will remain unchanged at TMI and will continue as an augmented ISI program in accordance with TMI's original commitment to USNRC Branch Technical Position MEB 3-1. This program was identified in the RI-ISI submittal simply to clarify that the HELB commitment is not subsumed under the RI-ISI Program and will remain unchanged.

b. The RI-ISI process at TMI evaluated several augmented inservice inspection programs for integration and inclusion with the risk informed program. No augmented inspections were performed under the current ISI Program that fell within the scope of the RI-ISI boundaries for IE Bulletin 79-17, Stagnant Borated Water Systems. The degradation mechanism assessment process outlined in the TR follows the criteria of Table 3-14. Under this table, PWR IGSCC would be assessed based on the applicable material properties and environmental conditions.

   Generic Letter 89-13, Service Water Integrity Program, also was not subsumed under the RI-ISI evaluation methodology at TMI. TR Section 3.6.5 states "Section 3.6.7 provides alternatives for assessing localized corrosion for those licensees wishing to address service water and other raw water systems as part of their RI-ISI application." The scope of the TMI RI-ISI Program does not address service water and raw water systems as a separate program. This program was identified in the RI-ISI submittal simply to clarify that the GL 89-13 program is not subsumed under the RI-ISI Program and will remain unchanged. The program was not subsumed because the scope of the

RI-ISI program did not address Class 3 systems such as service water and other raw water systems.

c. FAC is the only augmented program used in lieu of additional risk informed element selections at TMI. Table 3 of the Program Summary includes those elements where the only degradation mechanism identified was FAC. This information was included as these tables show the complete results of the risk evaluation. Table 4 of the Summary then identifies the total number of elements selected for examination under the RI-ISI Program. Per the EPRI TR methodology, elements only subject to FAC are not included in the population from which the RI-ISI Program selects and performs for-cause inspections.

In response to RAI request #2c, the following information is provided regarding those locations where the degradation mechanism assessment only identified FAC.

| Unit 1 | | | |
|---|---|---|---|
| Category | System | Count | Mechanism |
| 1 | FW | 86 | FAC-only |
| 3 | FW | 4 | FAC-only |

Thus, 90 of the elements evaluated under the RI-ISI process had FAC as the only degradation mechanism identified, and these locations are included in Table 3 (all within the FW system) of the Program Summary. In accordance with the TR, these elements are not included in the population selected for examination under the RI-ISI program as shown in Program Summary Table 4.

<u>Question</u>:

3. In Table 2 of your proposed RI-ISI Program, the piping element in the decay heat removal system is susceptible to primary water stress corrosion cracking (PWSCC) and erosion-cavitation (E-C). In Table 4, it is indicated that two elements and one element are scheduled for inspection in Category 2 and 5 in this system, respectively. Discuss how the inspection will be performed and discuss elements selected to detect degradation from both PWSCC and E-C.

<u>Response</u>:

Table 2 of the RI-ISI Program Summary is a compilation of all degradation mechanisms identified within each system evaluated under the TR methodology. This is meant to provide an overview and does not imply that each element with the given system is potentially subject to each mechanism identified. Table 4 provides the number of elements selected for each System / Risk Category and does not provide a link to the individual degradation mechanisms as they differ among the evaluated elements.

For the Decay Heat Removal (DH) system at TMI, sixteen (16) elements are selected for inspection. Two (2) of these selections are from Category 2, thirteen (13) are from Category 4, and one (1) is from Category 5. Of the two (2) Category 2 locations selected, one (1) has

PWSCC identified and the other has E-C identified, the Category 5 location has E-C as the identified degradation mechanism, and by definition, Category 4 locations are not susceptible to any known degradation.

In accordance with the TR and taking additional guidance from Code Case N-578-1, all elements with PWSCC will receive both volumetric and visual (VT-2) examinations, all elements with E-C will receive a volumetric examination to determine wall thickness, and all elements with no known mechanism will receive an expanded volumetric examination.

<u>Question</u>:

4. In Table 4 of your proposed RI-ISI Program, fifteen elements in the main steam system are scheduled for inspection in Risk Category 1. Category 1 is for inspection of flow accelerated corrosion and is inspected under a separate GL 89-08 program. Discuss what degradation mechanism this inspection is intended for in your RI-ISI Program.

<u>Response</u>:

The fifteen (15) locations selected for examination in the RI-ISI Program under Main Steam Risk Category 1 are all susceptible to multiple degradation mechanisms. For each element within this grouping, both the FAC and Thermal Fatigue (TT – Thermal Transients) degradation mechanisms were identified. While the FAC mechanism is inspected for under a separate augmented inspection program, these elements remain within the RI-ISI Program for the purpose of conducting for-cause inspections of the TT mechanism. In accordance with the TR, these fifteen (15) elements selected for inspection will receive an expanded volumetric examination under the RI-ISI Program.

<u>Question</u>:

5. In your RI-ISI Program, a large number of elements, 139, in Risk Category 4 (Table 4) are scheduled for inspection. Elements in this Category are not susceptible to any specific degradation mechanism. If degradation was found in the initial inspection of this Category, discuss the guidelines that you will follow to select elements for additional examination when a root cause of the degradation can not be determined. Should you also consider elements in other Categories for additional examination and modify your program (subject to NRC approval) due to the new finding for future inspection?

<u>Response</u>:

If service-induced degradation was found in the initial examination sample of Risk Category 4, an evaluation would be performed (as per EPRI TR-112657, Revision B-A, Section 3.6.6.2) and the root or probable cause of the degradation would likely be established. The evaluation would also include a review of other segments in Risk Categories 4, 6, and 7, where no degradation mechanisms were initially identified to determine their susceptibility to the same root/probable cause. Additional examination would then be implemented for the identified degradation mechanism in accordance with approved examination sample expansion guidelines. The applicable RI-ISI program would then be revised. However, the revised program would not be submitted to the NRC for approval unless the identified degradation is not defined by the EPRI RI-ISI methodology.

In the event that a degradation mechanism was found in the initial examination sample of Risk Category 4 and the root or probable cause cannot be determined, additional examinations would be performed on Risk Category 4 elements up to a number equivalent to the number of elements initially scheduled for the current outage. If unacceptable flaws are again found similar to the initial problem, the remaining elements in Risk Category 4 will be examined.

Question:

6. Through-wall cracking due to PWSCC was found in a hot-leg pipe weld made of Alloy 182/82 material at the V. C. Summer plant. Discuss and confirm that all your components that consist of Alloy 182/82 weld or butter have been evaluated and properly categorized for their susceptibility to PWSCC.

Response:

The RI-ISI methodology implemented for TMI followed the EPRI Topical Report for all aspects of the analysis including the Degradation Mechanism Evaluation. This evaluation followed the requirements of Section 3.4 in the TR. Degradation mechanisms were identified for each piping segment within the selected system boundaries in accordance with Section 3.4.2.1. The actual piping design, system functions, and operating conditions were compared to a set of material and environmental attributes. This process is summarized in TR Table 3-14.

Under the Stress Corrosion Cracking portion of Table 3-14, the attributes for PWSCC are defined. In addition, Section 3.4.2.2 requires consideration of applicable service experience. As such, any locations identified with Inconel buttering (Alloy 182/82) were characterized as a candidate for PWSCC. In accordance with this set of requirements for assessing the PWSCC mechanism, ten (10) locations were identified in the TMI RI-ISI program as potentially susceptible to this degradation. Based on the TR guidance and industry experience, TMI has conservatively scheduled both an expanded volumetric examination as well as a VT-2 visual examination for each of the components selected that are potentially subject to PWSCC.

Question:

7. Under what conditions would your RI-ISI Program be resubmitted to the NRC before the end of any 10-year interval?

Response:

In Section 4.0, "Implementation and Monitoring Program," of the Reference 1 letter, AmerGen stated that, the RI-ISI program is a living program requiring feedback of new relevant information to ensure the appropriate identification of high safety significant piping locations. Such relevant information would include major updates to the TMI, Unit 1 PRA model which could impact both the risk characterization and risk impact assessments, any new trends in service experience with piping systems at TMI, Unit 1 and across the industry, and new information on element accessibility that will be obtained as the risk informed inspections are implemented. The risk ranking of piping segments will be reviewed and adjusted on an ASME ISI "period" basis. This review will be documented internally, and the results need not be submitted to the NRC on the "period" frequency. The RI-ISI program may be resubmitted for NRC approval per the requirements of 10 CFR 50.55a if there is a deviation from the RI-ISI methodology described in the initial RI-ISI submittal to the NRC for that interval; or, if industry experience determines that there is a need for significant revision to the RI-ISI methodology as described in the initial RI-ISI submittal to the NRC for that interval.

Question:

8. On page 1 of your submittal, you state that the TMI Nuclear Station 2000 Probabilistic Risk Assessment (PRA) Model TMIL2RV2, dated August 2000, was used for your RI-ISI analysis. Please provide the baseline core damage frequency and baseline large early release frequency from this version of the PRA model.

   Your submittal also states that the TMI-1 PRA model used to support your RI-ISI Program is an update to the Individual Plant Examination (IPE) submitted to the NRC on May 20, 1993. In the NRC staff's evaluation of the IPE, dated December 19, 1996, the NRC staff states that the IPE submittal did not include a containment phenomena sensitivity analysis as requested by GL 88-20. Please state whether including the containment phenomena sensitivity analysis would have an impact on your RI-ISI analysis.

Response:

On page 1 of the submittal, AmerGen stated that the TMI Nuclear Station 2000 Probabilistic Risk Assessment (PRA) Model TMIL2RV2, dated August 2000, was used for the RI-ISI analysis. The baseline Core Damage Frequency (CDF) is 4.02e-5/year, and the baseline Large Early Release Frequency (LERF) is 2.81e-6/year from this version of the PRA model.

With regards to the second portion of the question, a Level 2 PRA sensitivity was completed in support of the TMI, Unit 1 IPE submittal and provided in response to IPE RAI Question 9 on the Level 2 PRA via GPUN Letter No. C311-95-2467, dated December 6, 1995.

In the RI-ISI analysis, the Conditional Core Damage Probability (CCDP) and Conditional Large Early Release Probability (CLERP) are calculated for a number of cases and are used to determine the Consequence Category. Any changes to the PRA model resulting from the Level 2 sensitivities could impact the CLERP values for these cases. Changes to the CLERP values would have to be large enough to move the welds into a higher Consequence Category to have any effect on the RI-ISI analysis results. With this in mind, there are four (4) groups of consequences that need to be examined further. The CLERP for two (2) of these groups is driven by conservative assumptions about the conditional probability of LERF given CDF and, therefore, would not be impacted by any sensitivities done for the Level 2 model. The largest contributor to CLERP for the other two groups is the failure of containment due to an early hydrogen burn. The IPE RAI response noted above includes a discussion of the examined sensitivity parameters suggested by the Gabor, Kenton & Associates (GKA) report prepared for EPRI and how they were addressed in the IPE submittal. It is noted in the response to IPE RAIs that "...the Oconee PRA did not rely on the burn models in MAAP for the CET quantification". Therefore, the examination of hydrogen burn parameters was not considered to be appropriate. The CLERP for the latter two groups would therefore not be impacted by any of these sensitivities done for the Level 2 model.

Additionally, the sensitivities described previously show very small changes to the LERF. The CLERP values calculated for the four groups of RI-ISI cases would need to increase by 70% to

200% to cause a change in their Consequence Category. Since the CLERP values for the first two (2) RI-ISI groups are dominated by conservative assumptions, it is highly unlikely that changes to the Level 2 input parameters could increase the CLERP sufficiently to alter the Consequence Category. Even though containment failures are a significant contributor to CLERP for the latter two (2) RI-ISI groups, the magnitude of the increase required to cause a change to the Consequence Category (70% - 200%) makes it highly unlikely that there could be any change in Consequence Category due to changes in the Level 2 input parameters.