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10 CFR 54
10 CFR 50

TMI-09-050
April 29, 2009

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

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

Subject: Exelon Generation Company (EGC) review of the Safety Evaluation Report with open items.

- References:**
1. Letter from M. Gallagher (AmerGen Energy Company, LLC) to U. S. Nuclear Regulatory Commission, "Three Mile Island Nuclear, Unit 1 License Renewal Application," dated January 8, 2008.
 2. Letter from B. Holian (U. S. Nuclear Regulatory Commission) to M. Gallagher Exelon Generation Company, "Safety Evaluation Report With Open Items Related to the License Renewal of Three Mile Island Nuclear Station, Unit 1," dated March 13, 2009.

In the Reference 1 letter, AmerGen Energy Company, LLC (AmerGen) submitted a License Renewal Application for Three Mile Island Nuclear Station, Unit 1, which would extend the term of the current operating license an additional 20 years.

In the Reference 2 letter, the U. S. Nuclear Regulatory Commission (USNRC) requested Exelon Generation Company (EGC) to review the Safety Evaluation Report with Open Items (SER) Related to the License Renewal of Three Mile Island Nuclear Station, Unit 1, verify its accuracy, and provide comments to the staff within 60 days from the issuance of the letter.

EGC has completed its review of the SER and is providing comments. Additionally, contained within the SER was one Confirmatory Item which had previously been discussed during a Conference call held between EGC and the staff on 3/11/09. EGC is providing to the staff the information required to close Confirmatory Item (CI) 4.3.2-1. Both of these items are contained within Attachment A.

If you have any questions, please contact Al Fulvio, Manager License Renewal, at 610-765-5936.

A131
MRR

I declare under penalty of perjury that the foregoing is true and correct.

Respectfully,

Executed on

04-29-2009



Michael P. Gallagher
Vice President, License Renewal
Exelon Generation Company, LLC

Attachment A:

1. EGC Comments on the Safety Evaluation Report with Open Items (SER) Related to the License Renewal of Three Mile Island Nuclear Station, Unit 1.
2. Confirmatory Item (CI) 4.3.2-1 response

cc: Regional Administrator, USNRC Region I, w/ Attachment
USNRC Project Manager, NRR - License Renewal, Safety, w/ Attachment
USNRC Project Manager, NRR - License Renewal, Environmental, w/o Attachment
USNRC Project Manager, NRR - TMIGS, w/o Attachment
USNRC Senior Resident Inspector, TMIGS, w/o Attachment

File No. 08001

Attachment – A

1. EGC Comments on the Safety Evaluation Report with Open Items (SER) Related to the License Renewal of Three Mile Island Nuclear Station, Unit 1.
2. Confirmatory Item (CI) 4.3.2-1 response

EGC Comments on the Safety Evaluation Report with Open Items (SER) Related to the License Renewal of Three Mile Island Nuclear Station, Unit 1.		
Page #	Section #	Comment
2-25	2.1.5.2.1	In the last sentence it states that heat exchangers were evaluated with the system associated with the process environment. This is not a correct statement. Heat exchangers and coolers are screened as follows: 1. With the exception of heat exchangers and coolers that are in scope only for 10 CFR 54.4 (a)(2) spatial interactions, the materials, environments and aging effects on both sides of the heat transfer surfaces are evaluated with the system that performs the cooling function. This convention was chosen because the significant aging effects and associated aging management program activities are generally associated with the cooling system side. 2. For heat exchangers and coolers that are in scope for 10 CFR 54.4 (a)(2) only, the portions of the heat exchanger or cooler with the potential for spatial interaction are a function of the design and the process fluid. Therefore, each side of the heat exchanger or cooler is evaluated separately with the system associated with the process environment. Refer to LRA Section 2.1.6.1 (pg 2.1-25) which describes this heat exchanger methodology.
2-73	2.3.3.25.2	The Staff's evaluation discusses the impact of RAI 2.3.3.25-1 on the scoping of the Water Treatment and Distribution System. There were two (2) additional changes made to the scoping of the Water Treatment and Distribution System that were not addressed in the Staff's evaluation. As described in AmerGen's response to RAI 2.1.5.2-3, Water Treatment and Distribution System components were added to the scope of License Renewal for spatial interaction. In addition, as described in letter 5928-08-20079 dated April 8, 2008, outdoor Water Treatment and Distribution System piping that was originally in-scope for structural support was removed from scope since this piping was determined to not provide a structural support intended function.
2-110	2.5.1.2	Last paragraph - 1st sentence wording implies that the "entire" substation is included in the scope of LR. The boundary is drawn at the 1st circuit breaker upstream of the aux transformers. Would read more accurately: " ... onsite circuits and up to and including the first circuit breakers in the substation ... "
2-228 to 2-241	Table 3.3-1 Various Items	The column "Staff Evaluation" does not include Exceptions when stated in the LRA. Examples: 3.3.1-3, 19, 20, 23, 24, 26, 33, 40, 43, 46, 47, 50, 52, 54, 55, 56, 57, 58, 59, 60, 61, 63, 65, 66, 67, 68, 71, 72, 76, 78, 80, 83, 86, 90 & 91 .

EGC Comments on the Safety Evaluation Report with Open Items (SER) Related to the License Renewal of Three Mile Island Nuclear Station, Unit 1.		
Page #	Section #	Comment
2-232	Table 3.3-1 Item 3.3.1-28	The column "AMP in LRA...etc" does not credit the program for One-Time Inspection. The section referenced in the "Staff Evaluation" column also does not discuss the program.
2-238	Table 3.3-1 Various Items	The column "AMP in LRA...etc" does not credit the program for Inspection of Internal surfaces in Miscellaneous Piping and Ducting Components. Examples include: 3.3.1-71 & 72. The section referenced in the "Staff Evaluation" column also does not discuss the program.
2-238	Table 3.3-1 Item 3.3.1-71	The column "AMP in LRA...etc" does not credit the program for Inspection of Internal surfaces in Miscellaneous Piping and Ducting Components. The section referenced in the "Staff Evaluation" column also does not discuss the program.
2-328	Table 3.4-1 Item 3.4.1-6	The "AMP in LRA" column does not discuss the Closed Cycle Cooling Water AMP. However, the section referenced in the "Staff Evaluation" does discuss this AMP.
2-330	Table 3.4-1 Item 3.4.1-16	The "AMP in LRA" column does not discuss the Water Chemistry and One-Time Inspection AMPs. However, the section referenced in the "Staff Evaluation" does discuss these AMPs.
3-6	3.0.3	GENERIC COMMENT: DSER Section 3.0.3, Table 3.0.3 - 1, column "Applicant Comparison to the GALL Report" identifies program exceptions and enhancements IAW LRA Appendix B which does not align with the final identification of exceptions and enhancements as revised by RAI's. For example, the One-Time Inspection program is identified as "Consistent with Exceptions" in Table 3.0.3 - 1. RAI B.2.1.18-1 eliminated the sole exception making this program consistent with GALL. In this case, the program would be discussed in DSER Section 3.0.3.1 instead of 3.0.3.2. A converse of this is the Bolting Integrity Program which was identified as Consistent in LRA Appendix B but changed to consistent with exceptions by RAI B.2.1.7-1.
3-6	Table 3.0.3	Reactor Head Closure Studs should be Consistent with Enhancement, not Consistent with Exceptions.
3-74	3.0.3.2.12	Revise the 2nd paragraph as follows: The applicant stated that the program provides preventive actions... The applicant also stated that contaminants are controlled and monitored in accordance with site technical specifications and applicable American Society for Testing and Materials (ASTM) standards and that the program manages loss of material due to general, pitting, crevice corrosion, and microbiologically-influenced corrosion, and biological fouling.

EGC Comments on the Safety Evaluation Report with Open Items (SER) Related to the License Renewal of Three Mile Island Nuclear Station, Unit 1.		
Page #	Section #	Comment
3-76	3.0.3.2.12	In the discussion of Exception 2 there is an incorrect reference to RAI B.2.1.16-2. The correct reference is RAI B.2.1.16-1.
3-80	3.0.3.2.12	Revise the 1st paragraph following the bulleted list of Enhancements as follows: The applicant committed to program enhancements that will a) add fuel oil sampling activities and increase sampling frequencies, b) provide for adherence to industry sampling standards, c) provide for biocide and inhibitor additions to fuel oil if required , d) provide for draining, cleaning and inspection of fuel tanks that had not previously been subjected to these activities and, e) use ultrasonic techniques to determine loss of material of tank bottoms should evidence of loss of material be identified during visual inspection activities.
3-80	3.0.3.2.12	Revise the 3rd paragraph in the Operating Experience Section as follows: In its response to the RAI dated October 20, 2008, the applicant stated that only the FO-T-1 fuel oil tank was subjected to cleaning and internal visual inspection in September 2007. The applicant discovered unacceptable pitting corrosion-. The pits, although small in diameter, were greater than 50% of the wall floor plate thickness, which was and were repaired in accordance with industry standard, American Petroleum Institute (API) 653 by welding patch plates over the affected areas. The applicant's AMP also provides for internal cleaning of the FO-T-1 fuel oil tank during the period of extended operation every ten years. The staff noted that all other fuel oil tanks will receive periodic cleaning and visual inspection of the tank interior or one-time external volumetric inspection...
3-81	3.0.3.2.12	Revise the 4th paragraph in the Operating Experience Section as follows: The staff noted that the documentation provided by the applicant during the onsite review supported the applicant's statements regarding operating experience and confirmed that the plant-specific operating experience did not reveal any degradation not bounded by industry experience, except for pitting corrosion discovered in the FO-T-1 fuel oil tank where acceptable corrective actions have been performed or implemented by the applicant. As worded, it sounds as if pitting corrosion was "not bounded" by industry experience which is not the case.

EGC Comments on the Safety Evaluation Report with Open Items (SER) Related to the License Renewal of Three Mile Island Nuclear Station, Unit 1.		
Page #	Section #	Comment
3-90	3.0.3.2.15	In the third paragraph on page 3-90, the SER states, "... the applicant stated that the diesel generator fuel storage 30,000 gallon tank will be subjected to a visual and UT examination in accordance with API Standard 1631..." The actual RAI B.2.1.20-1 response states that the tank, "...will be internally inspected in accordance with the guidance for assessing tank wall thickness contained in API Standard 1631..." The distinction is that the guidance of API 1631 will be used, not that API 1631 itself will be the standard of record for the inspection. The meaning is the same -- the inspection will be performed as described in the API 1631 standard, whether that standard or another applicable standard is used. The SER does correctly reflect this later when it says, "...loss of material will be detected using the UT examination methods of API Standard 1631."
3-113	3.0.3.2.22	In 1st line of Summary of Tech Info in the Application, the SER should cite this AMP as an existing (not new) AMP.
3-122	3.0.3.2.25	Add the following to the second paragraph of section 3.0.3.2.25: "Since these components would have fatigue usage that exceeds 1.0 if the transient cycle limits were increased to 1.5 times the current design limits, the program will maintain the current transient cycle design limits to manage fatigue during the period of extended operation."
3-128	3.0.3.3.1	Staff Evaluation, third paragraph. The sentence "The applicant indicated that the LRA Section B.2.2.1 will be amended, and its corresponding basis document will be updated based on the revised requirements and that the estimated completion date for these changes is April 30, 2009" should read "The applicant indicated that the LRA Section B.2.2.1 text would change slightly based on the new requirements. The corresponding basis document will be updated based on the revised requirements and that the estimated completion date for these changes is April 30, 2009." Our e-mail indicated how LRA Section B.2.2.1 would read and our commitment to change the basis document. We did not intend to submit an amendment to the LRA for the changes.
3-151	Table 3.1-1 Item 3.1.1-80	Line Item 3.1.1-80, missing Commitment Number 36 , missing Staff Evaluation.

EGC Comments on the Safety Evaluation Report with Open Items (SER) Related to the License Renewal of Three Mile Island Nuclear Station, Unit 1.		
Page #	Section #	Comment
3-161	3.1.2.1.3	Last sentence, sixth paragraph should read "The applicant stated that inspection of the corresponding weld off the D cold leg drain line was performed in 2003 (penetrant) with acceptable results....." not "The applicant stated that inspections of corresponding welds off the D cold leg drain line were performed in 2001 (volumetric) and in 2003 (penetrant) with acceptable results....."
3-165	3.1.2.2.3 (2)	Section 3.1.2.2.3 (2) first paragraph should include environment of reactor coolant and neutron flux .
3-185	Table 3.2-1 Item 3.2.1-8	The Column "AMP in LRA..." does not credit the One-Time Inspection Program. However, the section referenced in the "Staff Evaluation" column does discuss the program.
3-187	Table 3.2-1 Item 3.2.1-23	The Column "AMP in LRA..." also does not credit programs: Reactor Head Closure Studs and ASME Section XI Subsection IWE. The section referenced in the "Staff Evaluation" column also does not discuss the programs.
3-187	Table 3.2-1 Item 3.2.1-24	The Column "AMP in LRA..." does not credit the Appendix J program. The section referenced in the "Staff Evaluation" column also does not discuss the program.
3-189	Table 3.2-1 Item 3.2.1-38	"AMP in LRA" column of table should also include the Open Cycle Cooling Water System.
3-192	3.2.2.1	The second paragraph starts with "LRA Table 3.2.2-1 summarizes". This should read: "LRA Tables 3.2.2-1 to 6 summarize"
3-192, 3-193	3.2.2.1	Page 3-192, first paragraph after the bulleted items, and on page 3-193, third full paragraph, the SER refers to "LRA Table 3.2.2-1" as providing a summary for the AMR results for ESF, this should be "LRA Tables 3.2.2-1 through 3.2.2-6."
3-193	3.2.2.1	The tenth paragraph starts with "LRA Table 3.2.2-1, provides". This should read: "LRA Tables 3.2.2-1 to 6 provide"
3-197	3.2.2.1.1	Page 3-197, second full paragraph, fifth line: RAI-AMR-GENERIC-2 states that line item 3.2.1-37 is not applicable because it DOES NOT PREDICT fouling as an aging mechanism, not because it PREDICTS the additional aging affect/mechanism of loss of material/fouling. Suggested fix to SER is to replace "predicts" with "does not predict."
3-199	3.2.2.1.2 (1)	Should read "Steel Bolting" and not steel components.
3-200	3.2.2.1.2 (2)	Should read "Steel Bolting" and not steel components.
3-233 to 3-240	Table 3.3-1 Various Items	The column "Staff Evaluation" reads "Consistent with GALL" when the LRA reads "Not consistent with". Examples include items: 3.3.1-32, 48, 51 & 82. The section referenced in the "Staff Evaluation" column also does not discuss the programs.

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Page #	Section #	Comment
3-235	Table 3.3-1 Item 3.3.1-45	The column "AMP in LRA...etc" does not credit the programs for Structures Monitoring and Inspection of Overhead Heavy Load and Light Load Program. The section referenced in the "Staff Evaluation" column also does not discuss the programs.
3-237	Table 3.3-1 Item 3.3.1-60	The column "AMP in LRA...etc" does not credit the programs for External Surfaces Monitoring and Inspection of Overhead Heavy Load and Light Load Handling Systems. The section referenced in the "Staff Evaluation" column also does not discuss the programs.
3-237	Table 3.3-1 Item 3.3.1-63	The column "AMP in LRA...etc" does not credit the program for Inspection of Overhead Heavy Load and Light Load Handling Systems. The section referenced in the "Staff Evaluation" column also does not discuss the program.
3-237 to 3-239	Table 3.3-1 Various Items	The column "AMP in LRA...etc" does not credit the program for Structures Monitoring. Examples include: Table Items 3.3.1-65, 66, 67 & 78. The section referenced in the "Staff Evaluation" column also does not discuss the program.
3-239 to 3-240	Table 3.3-1 Various Items	The column "AMP in LRA...etc" does not credit the program for Open-Cycle Cooling Water. Examples include: 3.3.1-76 & 81.
3-259	3.3.2.1.11	Section 3.3.2.1.11 is for loss of material/pitting and crevice corrosion for stainless steel and nickel alloy components in raw water. The third paragraph of the section discusses a wetted air/gas environment that is not relevant. Delete sentence reading "The staff noted that the wetted air/gas environment..."
3-284	3.3.2.2.9	Second to last paragraph of 3.3.2.2.9 (3) states "susceptible steel heat exchanger components." TMI credited 3.3.1-21 for pumps and valves, not heat exchangers.
3-289	3.3.2.2.10 (6)	First paragraph states "The staff noted that only stainless steel ducting and components is applicable to TMI-1." TMI-1 does not have stainless steel ducting.
3-290	3.3.2.2.10 (7)	Section 3.3.2.2.10 (7) states the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components program manages loss of material/pitting and crevice corrosion in copper alloy components in the Fire Protection System. TMI did not credit this program to manage copper alloy components in the Fire Protection System.
3-302	3.3.2.3.5	Section 3.3.2.3.5 has a discussion about LRA Table 3.2.2-5 that is not relevant to the Containment Isolation System. The CI System is addressed in LRA Table 3.3.2-5.

EGC Comments on the Safety Evaluation Report with Open Items (SER) Related to the License Renewal of Three Mile Island Nuclear Station, Unit 1.		
Page #	Section #	Comment
3-316	3.3.2.3.13	SER Section 3.3.2.3.13 stated LRA Table 3.3.2-13 did not contain any plant-specific Notes F through J; however, TMI used Note G twice for this system. Refer to LRA Table 3.3.2-13, page 3.3-244.
3-322	3.3.2.3.21	Revise the 9th paragraph as follows: In LRA Table 3.3.2-21, the applicant proposed to manage loss of material due to pitting and crevice corrosion and fouling for titanium alloy material for tanks exposed...
3-324	3.3.2.3.22	SER Section 3.3.2.3.22 stated LRA Table 3.3.2-22 did not contain any plant-specific Notes F through J; however, TMI used Note I three times for this system. Refer to LRA Table 3.3.2-22, pages 3.3-332 and 334.
3-325	3.3.2.3.25	The Staff's evaluation for the Water Treatment and Distribution System AMR should also address the changes made to LRA Table 3.3.2-25 in RAI 2.3.3.25-1, RAI 2.1.5.2-3, and Exelon identified changes in letter 5928-08-20079 dated April 8, 2008.
3-331	Table 3.4-1, Item 3.4.1-19	The "AMP in LRA" column does not discuss the Lubrication Oil and One-Time Inspection AMPs. However, the section referenced in the "Staff Evaluation" does discuss these AMPs.
3-347, 2-348	3.4.2.2.6	This section incorrectly interpreted the LRA to conclude that aging effect "cracking due to stress corrosion cracking" is not applicable to TMI-1. The LRA states "Item number 3.4.1-13 is applicable to BWRs only and not used at TMI-1." The SER fails to recognize that the comparable Table 3.4.1 Item number 3.4.1-14 applies to PWRs for SCC.
3-357	3.4.2.3.2	Revise the 2nd paragraph of DSER Section 3.4.2.3.2 as follows: The applicant designated Note G for aluminum alloy filter housings and Note H for copper alloy (Zn less than 15%) piping and fittings exposed to a lubricating oil environment in the condensers & air removal system because the aging effect loss of material due to the mechanism of microbiologically influenced corrosion for the AMR line item component, material, and environment combination... The staff reviewed the GALL Report and found that the AMR line item, piping and fittings is not evaluated for a lubricating oil environment for loss of material due to pitting, crevice, microbiologically influence corrosion and accordingly Note H is appropriate...

EGC Comments on the Safety Evaluation Report with Open Items (SER) Related to the License Renewal of Three Mile Island Nuclear Station, Unit 1.		
Page #	Section #	Comment
3-358	3.4.2.3.3	Revise the 6th paragraph as follows: In LRA Table 3.4.2-3, the applicant designated Note G for aluminum alloy sight glass housings and Note H for copper alloy (Zn less than 15%) piping fittings and copper alloy (Zn less greater than 15%) sight glass housings exposed to a lubricating oil environment in the emergency feedwater system because the aging effect loss of material due to microbiologically influenced corrosion for the AMR line item component, material, and environment combination... The staff reviewed the GALL Report and found that the AMR line item, copper alloy piping, fittings, and sight glass housings is not evaluated for a lubricating oil environment for loss of material due to pitting, crevice, microbiologically influence corrosion and that the GALL Report does not address aluminum sight glass housings exposed to lubricating oil. The applicant...
3-362	3.4.2.3.5	Revise the 6th paragraph as follows: The applicant designated Note H for copper alloy (Zn less than 15%) piping and fittings and copper alloy (Zn less than 15%) exposed to a lubricating oil environment in the feedwater system because the aging effect loss of material due to microbiologically influenced corrosion for the AMR line item component, material, and environment combination... The staff reviewed the GALL Report and concluded that the AMR line item, copper alloy piping and fittings, and sight glass housings is not evaluated for a lubricating oil environment for loss of material due to pitting, crevice, and MIC microbiologically influenced corrosion . The applicant...
3-363	3.4.2.3.6	Revise the 2nd paragraph as follows: In LRA Table 3.4.2-6, the applicant designated Note H for copper alloy (Zn less than 15%) piping, fittings and valves exposed to a lubricating oil environment in the feedwater system because the aging effect loss of material due to microbiologically influenced corrosion for the AMR line item component, material, and environment combination... The staff reviewed the GALL Report and concluded that the AMR line item, copper alloy piping, fittings, and valves is not evaluated for a lubricating oil environment for loss of material due to pitting, crevice, microbiologically influence corrosion. The applicant...
3-375	Table 3.5-1 Item 3.5.1-1	Third column should include Boric Acid Corrosion Program which is discussed in referenced SER section.
3-378	Table 3.5-1 Item 3.5.1-17	Third column includes IWE which is not in the LRA for the associated line item.
3-379	Table 3.5-1 Item 3.5.1-23	Third column should include Boric Acid Corrosion Program which is discussed in referenced SER section.

EGC Comments on the Safety Evaluation Report with Open Items (SER) Related to the License Renewal of Three Mile Island Nuclear Station, Unit 1.		
Page #	Section #	Comment
3-379	Table 3.5-1 Item 3.5.1-24	Third column should include Boric Acid Corrosion Program which is discussed in referenced SER section.
3-380	Table 3.5-1 Item 3.5.1-47	In Table 3.5-1, Item 3.5.1-47, replace the reference to the "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components" with the External Surfaces Monitoring Program. This change will align Table 3.5-1 with the discussions in reference paragraphs 3.3.2.1.3 and 3.2.2.1.4 for Item 3.5.1-47 (the loss of material due to pitting and crevice corrosion in copper alloy exposed to outdoor air).
3-381	Table 3.5-1 Item 3.5.1-50	Table 3.5-1, Item 3.5.1-50, should also include the Bolting Integrity program which was used to manage the loss of material due to pitting and crevice corrosion in stainless steel bolting exposed to outdoor air in the Reactor Building Spray System.
3-381	Table 3.5-1 Item 3.5.1-34	Column 2 identifies 2 Aging Effect/Mechanisms's the first of which is NA because the environment is non-aggressive. This should be reflected in the last two columns.
3-383	Table 3.5-1 Item 3.5.1-46	Column 2 identifies 2 Aging Effect/Mechanisms's the first of which is NA because the environment is less than 140 deg F. This should be reflected in the last two columns.
3-391	3.5.2.1.5	The first paragraph is applicable to "cracking due to restraint shrinkage, creep, and aggressive environment." The end of the first paragraph refers to "the aging effects/mechanisms of reinforced concrete in an air with borated water leakage environment include cracking, loss of bond, and loss of material (spalling, scaling)/corrosion of embedded steel. These aging effects/mechanisms are managed by the Boric Acid Corrosion Program." This part should refer to the same Aging Effect/Mechanisms's as in the beginning.
3-397	3.5.2.2.1	In the 10th sentence down from the top of the page the wording "(weld repair) to full design thickness" should indicate "(weld repair) to full nominal thickness".
3-438	3.6.1	Section 3.6.1, last sentence states that the applicant's review of industry OE included the GALL report and OE issues identified since issuance of the GALL report. TMI Electrical OE reviews included issues (industry and plant specific) identified in the 5 years prior to the submittal of the LRA.
3-439	Table 3.6-1 Item 3.6.1-2	Column 5 (AMP in LRA) does not match TMI LRA AMP title (LRA AMP title matches GALL AMP title).
3-439	Table 3.6-1 Item 3.6.1-3	Column 5 (AMP in LRA) does not match TMI LRA AMP title per LRA Appendix A and Appendix B. Our LRA Table 3.6-1 AMP text quoted GALL table entries.
3-439	Table 3.6-1 Item 3.6.1-4	Column 5 (AMP in LRA) does not match TMI LRA AMP title (LRA AMP title matches GALL AMP title).

EGC Comments on the Safety Evaluation Report with Open Items (SER) Related to the License Renewal of Three Mile Island Nuclear Station, Unit 1.		
Page #	Section #	Comment
3-439 or 3-440	Table 3.6-1 Item 3.6.1-5	Missing table line item for 3.6.1-5, connector contacts for electrical connectors exposed to borated water leakage.
3-440	Table 3.6-1 Item 3.6.1-6	Line Item No. 3.6.1-6, column 6 incorrectly states that this AMP is "Not Consistent." In alignment with the LRA and other similar SER Table 3.x.1-y column 6 text, it is recommended that the column 6 text for this line item be changed to: "Not applicable to TMI. (See Section 3.6.2.3)"
3-441	Table 3.6-1 Item 3.6.1-11 Item 3.6.1-12	Column 6 (Staff Evaluation), does not indicate if program is consistent or not consistent with GALL.
3-445	Table 3.6-1 Item 3.6.1-7 Item 3.6.1-8	Column 5 (AMP in LRA) does not match TMI LRA AMP title (LRA AMP title matches GALL AMP title).
3-445	Table 3.6-1 Item 3.6.1-9	Column 2 (Aging Effect), states that the aging effect/mechanism is "Loss of Material/Pitting and Crevice Corrosion." The TMI LRA Table 3.6-1 identified this line item's aging effect/mechanism as "Loss of material due to general corrosion."
3-445	Table 3.6-1 Item 3.6.1-9 Item 3.6.1-10	Column 6 (Staff Evaluation) provides the reader a parenthetical reference to SER section 3.6.2.1, which discusses AMR results that are consistent with the GALL report. The AMP for these two line items in the Structures Monitoring Program SER section 3.6.2.1, does not discuss the Structures Monitoring Program. It is suggested that the parenthetical reference should be to SER section for the AMR results for the Structures Monitoring Program: 3.5.2.1.
4-2	4.1.1	The second paragraph (after the three bullets) should be deleted and replaced with: "The above exemption does not need to be continued for the period of extended operation because the 29 EFPY P-T limit curves for which the exemption was granted will not be used during the period of extended operation."
4-4	4.2.1.1	Under third bullet: Change "in-vessel" to "plant-specific cavity" (dosimetry). TMI-1 does not have in-vessel dosimetry.
4-5	4.2.1.2	In third paragraph, third sentence: change: "remain valid" to: "have been projected" and change: "10 CFR 54.21(c) (i)" to: "10 CFR 54.21(c) (ii)." This is consistent with Conclusion in 4.2.1.4.
4-11	4.2.4.1	In the second paragraph, last sentence, change: "55 EFPY" to "52 EFPY."

EGC Comments on the Safety Evaluation Report with Open Items (SER) Related to the License Renewal of Three Mile Island Nuclear Station, Unit 1.		
Page #	Section #	Comment
4-13, 4-14	4.2.5.2	The staff evaluation does not describe the TLAA method provided by TMI in the LRA. TMI used method (ii) to develop P-T Limit Curves (based upon ART values projected for 60 years). These P-T limit curves were then evaluated to demonstrate that adequate operating margin will exist at the end of the PEO.
4-16	4.3.1.1	Move the second sentence in the second paragraph down to the fourth paragraph, where it will appropriately describe the work performed for license renewal.
4-17	4.3.1.2	Revise first sentence. Replace: "the analyses remain valid for the period of extended operation" with: "fatigue of these components will be managed for the period of extended operation."
4-27 4-28	4.3.3.2 4.3.4.2	Revise first sentence. Replace: "the analyses remain valid for the period of extended operation" with: "fatigue of these components will be managed for the period of extended operation." This is consistent with the conclusions in paragraphs 4.3.3.4 and 4.3.4.4.
4-30, 4-33, 4-34	4.3.5.2, 4.3.7.2, 4.3.7.4	Revise first sentence. Replace: "the analyses remain valid for the period of extended operation" with: "the analyses have been projected to the end of the period of extended operation."
A-7	Appendix A; No. 32	In the last sentence of the first paragraph of commitment 32, the SER omitted the word initially. The commitment should read "...will be inspected for water collection, <i>initially</i> at least twice a year...". "Initially" was specifically added via RAI response B.2.1.32-1 and again identified within letter TMI-09-009 dated 01/26/2009.

Confirmatory Item (CI) 4.3.2-1 response

Staff Question #1. The assumed value for dissolved oxygen that the maximum F_{en} values were calculated at and the reasoning why the dissolved oxygen value is considered a bounding assumption?

Response:

The maximum F_{en} values computed for carbon steels and low-alloy steels are based upon the equations provided in NUREG/CR-6583. The maximum F_{en} values computed for stainless steels are based upon the equation provided in NUREG/CR-5704. For each of these F_{en} computations at TMI-1, the oxygen factor was determined based upon an assumed dissolved oxygen level of < 0.05 ppm.

For carbon and low alloy steels, the oxygen factor is only applicable for temperatures $\geq 150^{\circ}\text{C}$ (302°F) because the temperature factor is zero at temperatures < 150°C , eliminating the oxygen factor from the computation since the two factors are multiplied together in the equation. For stainless steels, the oxygen factor is only applicable for temperatures $\geq 200^{\circ}\text{C}$ because the temperature factor is zero at temperatures < 200°C , also eliminating the oxygen factor from the computation. Therefore, the TMI-1 F_{en} calculations are based upon an assumption that the reactor coolant dissolved oxygen content is < 0.05 ppm when the reactor coolant is at a temperature $\geq 150^{\circ}\text{C}$ (302°F), the bounding case.

The assumed dissolved oxygen value of < 0.05 ppm is bounding for TMI-1 operations because the actual reactor coolant dissolved oxygen values are reduced to < 0.05 ppm prior to reaching 150°C (302°F) and are also maintained < 0.05 ppm throughout power operations. This is demonstrated by actual monitoring data described further in the response to question 2, below.

Staff Question #2. A summary of TMI-1's operating experience that indicates the normal level of dissolved oxygen. Indicate how often surveillance for dissolved oxygen is completed and how long dissolved oxygen has been maintained at this level.

Response:

TMI maintains the oxygen concentration in compliance with Chemistry Procedure, CY-AP-120-105, *Reactor Coolant System Chemistry for Three Mile Island*. The limits and frequencies for dissolved oxygen control in the procedure are based on those in the *Pressurized Water Reactor Primary Water Chemistry Guidelines, Volume 1, Revision 6* (1014986) published by EPRI (EPRI Guideline) and on those in the TMI-1 Technical Specifications (Section 3.1.5).

During plant heat up at the beginning of an operating cycle, before reactor coolant temperature reaches 250°F , the dissolved oxygen concentration in the reactor coolant is limited to < 0.1 ppm, with a goal of < 0.05 ppm. During power operations, the dissolved oxygen concentration goal is < 0.005 ppm, which is achieved by maintaining a measurable concentration of hydrogen in the coolant. EPRI has concluded that with

more than 0.5 cc/kg of hydrogen in the coolant under operating conditions, the dissolved oxygen concentration in the coolant will approach zero. The lower limit for the hydrogen concentration in the RCS during power operation is 25 cc/kg, unless a plant shutdown is planned in the next 24 hours. This value is 50 times more than that required to control the dissolved oxygen concentration within the prescribed limits.

As an example, the dissolved oxygen in the RCS during the last start-up proceeded as follows:

At an RCS temperature of 216 degrees F, the dissolved oxygen concentration was 0.045 ppm.

At an RCS temperature of 241 degrees F, the dissolved oxygen concentration was 0.001 ppm.

These values are representative of previous startups.

During power operations in the last three operating cycles, the measured dissolved oxygen concentration has normally been < 0.005 ppm and has not exceeded 0.027 ppm. These values are representative of previous fuel cycles.

During power operations, Reactor Coolant dissolved oxygen analyses are performed a minimum of 5 times per week in accordance with the TMI Technical Specifications.

Historical Review

Background:

The presence of oxygen in the water in the reactor coolant system of a PWR during power operations is caused by the decomposition of the water through radiolysis. The addition of hydrogen to the water suppresses the chemical reaction that is involved in this process and results in a negligible dissolved oxygen concentration in the coolant. Appendix E of the *EPRI Pressurized Water Reactor Primary Water Chemistry Guidelines, Volume 1, Revision 6 (1014986)* states that, "Adding only a small amount of H₂ suppresses radiolytic decomposition and this is predicted to be essentially complete when the concentration reaches 0.5 cc (STP) kg⁻¹ H₂." EPRI also cites two experimental studies that showed that the decomposition of water through radiolysis stopped at 0.3 cc/kg, in one case, and less than 1 cc/kg, in another case. Plant level test results cited by EPRI indicated that decomposition of water was inferred when the hydrogen concentration was reduced to 1.3 to 5 cc/kg. Applying the worst of the modeling and the test data, the minimum hydrogen concentration in the RCS needed to essentially eliminate dissolved oxygen is 5 cc/kg. The existing chemistry program is designed to ensure an excess of hydrogen (≥ 25 cc/kg) is available in the coolant so the resultant concentration of oxygen is extremely small. The EPRI guideline references testing that showed that the maximum dissolved oxygen concentration in a main reactor loop was less than 0.00000001 ppm, if 25 cc/kg hydrogen was maintained in the loop. A review of the TMI-1 reactor coolant dissolved oxygen concentrations reported since the return of the unit to power operations in 1985 identified that for all cases where the measured values were 0.05 ppm or greater, the hydrogen concentration was greater than 15 cc/kg. Therefore it can be assumed that the actual concentration of dissolved oxygen in the

coolant remained below 0.05 ppm at operating temperatures, because this amount of hydrogen effectively suppresses the radiolytic decomposition of water.

Historical Review (Pre-shutdown 1974-1979)

Hydrogen Control

During the early period of operation, the control of oxygen was accomplished by maintaining a hydrogen concentration that was sufficient to prevent the radiolytic decomposition of the water in the reactor coolant. A review of the oldest revision of the B&W Water Chemistry Manual (1975), archived at TMI, identified the control band for hydrogen gas in reactor coolant to be 15 – 40 cc/kg. The B&W Water Chemistry Manual states, "With excess hydrogen in the coolant, no detectable amounts of dissolved oxygen should exist under normal power conditions; therefore, if the dissolved hydrogen in the coolant is above the minimum limit (15 std cc H₂/kg water), the amount of oxygen can normally be assumed to be negligible." A sampling of data from the time period 1974 to 1979 identified that the hydrogen concentration in the reactor coolant system was consistent with the establishment of the 15 cc/kg limit.

Dissolved Oxygen

A sampling of data for the period 1974-1979 identified that the dissolved oxygen concentration in the reactor coolant during power operations was less than 0.050 ppm.

Staff Question #3. Indicate the industry guidance currently followed to maintain dissolved oxygen.

Response:

Industry guidance for reactor coolant system chemistry, including dissolved oxygen is based on the *EPRI Guideline 1014986, Pressurized Water Reactor Primary Water Chemistry Guidelines, Volume 1, Revision 6*.

Staff Question #4. Indicate at what level corrective actions are taken if dissolved oxygen exceeds a certain level.

Response:

During Startup, Heat-up - Cool down and Hot Shutdown modes, corrective actions are taken if the dissolved oxygen concentration is > 0.05 ppm.

During Power Operations and Hot Standby modes, corrective actions are taken if the dissolved oxygen concentration is > 0.005 ppm.

Staff Question #5. Provide copies of pages of any procedures that indicate the normal level of dissolved oxygen and at what level corrective actions are implemented if dissolved oxygen exceeds a certain level.

Response:

These procedures were previously provided