Constellation Energy
Nine Mile Point Nuclear Station

P.O. Box 63
Lycoming, NY 13093

November 30, 2005
NMP1L 2004

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

SUBJECT: Nine Mile Point Units 1 and 2
Docket Nos. 50-220 and 50-410
Facility Operating License Nos. DPR-63 and NPF-69

Amended License Renewal Application (ALRA) – Responses to NRC Requests for Additional Information Regarding ALRA Parts 1, 2 and 3 (TAC Nos. MC3272
and MC3273)

Gentlemen:

By letter dated July 14, 2005, Nine Mile Point Nuclear Station, LLC (NMPNS) submitted an Amended License Renewal Application (ALRA) for the operating licenses of Nine Mile Point Units 1 and 2.

In a letter dated November 2, 2005, the NRC requested additional information regarding the ALRA Part 1 - Aging Management of Auxiliary Systems, Part 2 – Nine Mile Point Unit 1 ALRA Issue Regarding Control Rod Drive Stud Tube and Part 3 – Time-Limited Aging Analyses. The NMPNS responses to these requests for additional information are provided in Attachment 1. Attachment 2 provides a list of the regulatory commitments associated with this submittal.

If you have any questions about this submittal, please contact David Dellario, NMPNS License Renewal Project Manager, at (315) 349-7141.

Very truly yours,

James A. Spina
Vice President Nine Mile Point

JAS/MSL/sac

A107

STATE OF NEW YORK :
: TO WIT:
COUNTY OF OSWEGO :

I, James A. Spina, being duly sworn, state that I am Vice President Nine Mile Point, and that I am duly authorized to execute and file these responses on behalf of Nine Mile Point Nuclear Station, LLC. To the best of my knowledge and belief, the statements contained in this submittal are true and correct. To the extent that these statements are not based on my personal knowledge, they are based upon information provided by other Nine Mile Point employees and/or consultants. Such information has been reviewed in accordance with company practice and I believe it to be reliable.

James A. Spina
Vice President Nine Mile Point

Subscribed and sworn before me, a Notary Public in and for the State of New York and County of Oswego, this 30th day of November, 2005.

WITNESS my Hand and Notarial Seal:

Judith S. Proud
Notary Public

JUDITH S. PROUD
Notary Public, State of New York
No. 01PR4856665
Qualified in Oswego County
Commission Expires April 14, 2006

My Commission Expires:

4/14/06
Date

Attachments:

1. Responses to NRC Requests for Additional Information (RAI) Regarding ALRA Part 1 - Aging Management of Auxiliary Systems, Part 2 – Nine Mile Point Unit 1 ALRA Issue Regarding Control Rod Drive Stud Tube and Part 3 – Time-Limited Aging Analyses.
2. List of Regulatory Commitments

cc: Mr. S. J. Collins, NRC Regional Administrator, Region I
Mr. L. M. Cline, NRC Senior Resident Inspector
Mr. T. G. Colburn, Senior Project Manager, NRR
Mr. N. B. Le, License Renewal Project Manager, NRR
Mr. J. P. Spath, NYSERDA

ATTACHMENT 1

Nine Mile Point Nuclear Station

Responses to NRC Requests for Additional Information (RAI) Regarding Amended License Renewal (ALRA) Part 1 - Aging Management of Auxiliary Systems, Part 2 – Nine Mile Point Unit 1 ALRA Issue Regarding Control Rod Drive Stud Tube and Part 3 – Time-Limited Aging Analyses

This attachment provides the Nine Mile Point Nuclear Station, LLC (NMPNS) responses to the requests for additional information contained in the NRC letter dated November 2, 2005. Each NRC RAI is repeated, followed by the NMPNS response for Nine Mile Point Unit 1 (NMP1) and/or Nine Mile Point Unit 2 (NMP2), as applicable.

PART 1 – AGING MANAGEMENT OF AUXILIARY SYSTEMS

3.3 Aging Management of Auxiliary Systems

General RAI

a-RAI 3.3.2-1

One-Time Inspection (OTI) is appropriate where either an aging effect is not expected to occur but there is insufficient data to completely rule it out, or the aging effect is expected to occur very slowly so as not to affect the component intended function. In the Amended LRA (ALRA) submitted to the staff by letter dated July 14, 2005, the applicant proposed to use OTI program to manage aging effects for various materials exposed to various environments for majority of the components in the following two systems:

(a) ALRA Table 3.3.2.A-14, NMP1 Radioactive Waste System

(b) ALRA Table 3.3.2.B-14, NMP2 Floor and Equipment Drains System.

(1) Explain from system characteristics standpoint, why the OTI program [rather than periodic inspections] is proposed to manage the aging effects for those material/environment combinations having OTI as the sole AMP in these two systems; and

(2) Justify the use of OTI program for the following cases:

(a) In ALRA Table 3.3.2.A-14, Aging Effect Requiring Management (AERM) of cracking for Wrought Austenitic Stainless Steel (WASS) Heat Exchangers exposed to air, moisture or wetting, temperature greater or equal to 140 degree F, and for WASS valves exposed to treated water, temperature greater or equal to 140 degree F but less than 212 degree F.

(b) In ALRA Table 3.3.2.A-14, AERM of loss of material (LOM) for carbon or low alloy steel (yield strength less than 100 ksi), or WASS valves, and piping and fittings exposed to Demineralized Untreated Water (DUW).

(c) *In ALRA Table 3.3.2.A-14, AERM of LOM for carbon or low alloy steel (yield strength less than 100 ksi) valves exposed to either the DUW, low flow, or treated water, temperature greater or equal to 140 degree F but less than 212 degree F.*

(d) *In ALRA Table 3.3.2.B-14, AERM of cracking for WASS Drainers exposed to treated water, temperature greater or equal to 140 degree F but less than 212 degree F.*

(e) *In ALRA Table 3.3.2.B-14, AERM of LOM for aluminum pump, or carbon or low alloy steel (yield strength less than 100 ksi) strainers exposed to raw water.*

NMP Response

(1) The NMP1 Radwaste and NMP2 Floor and Equipment Drains Systems include the following subsystems:

- equipment drains in various buildings,
- floor drains in various buildings, and
- the piping, pumps, tanks, and valves in these subsystems.

The components in these systems are predominantly fabricated of carbon steel and the environment is generally water; however, exposure to water is not continuous. When tanks or sumps reach pre-set levels, the pumps automatically start in order to empty them and thus expose the downstream components to water.

Due to this non-continuous exposure, the One Time Inspection Program was chosen to manage aging since the identified aging effects were judged to occur at such a slow rate that the component intended functions would not be impacted during the period of extended operation. Based on further evaluation, which included benchmarking and review of the guidance from the most recent industry aging management documentation, it has been concluded that the Preventative Maintenance (PM) Program more effectively manages the aging of the carbon steel and gray cast iron components in these systems than the One Time Inspection Program. The PM Program is, therefore, substituted for the One Time Inspection Program to manage the aging of the carbon steel and gray cast iron components in these systems with the exception of the carbon steel piping and fittings and valves that are subjected to an internal fuel oil environment. Since these components are exposed to fuel oil drainage, it is possible that there is water contamination such that loss of material due to corrosion could occur. This is considered to be unlikely since there will be an oil film on the inside of these components; however, the One Time Inspection Program will ascertain whether or not loss of material is occurring. If it is occurring, the Corrective Action Program will be utilized to document and disposition the anomaly. The cast and wrought austenitic stainless steel (CASS and WASS), nickel-based alloy, and copper alloy (Zn ≤15%) components will continue to be managed by the One Time Inspection Program.

An extent of condition review was performed for the other NMP1 and NMP2 mechanical systems to determine if similar changes were needed relative to the application of the One Time Inspection Program for aging management. As a result of this review, there were two other changes that were identified as follows: (1) for the NMP1 Miscellaneous Non-Contaminated Vents and Drains System, the AMP for managing the internals of the system components (carbon steel Piping and Fittings in a Demineralized Untreated Water or Raw Water environment) is changed to the PM Program from the One Time Inspection Program and (2) for the NMP2 Standby Liquid Control System, the line item on p. 3.3-288 for WASS Valves in the Air, Moisture or Wetting, temperature <140°F Environment is deleted (line with Note H). The valves that were identified as being in that environment are actually wetted and are covered by the other wetted WASS Valve environments already included in the ALRA.

For the specific instances questioned:

(2)(a) The heat exchangers that are addressed by the line item in the ALRA are associated with the Radwaste System Concentrator 12. This Concentrator, and hence its associated components, are infrequently (less than once per operating cycle) used since other preferable methods for liquid waste processing are normally utilized (see USAR Section XII.2.2.1). As shown on Drawing LR-18045-C, Sheet 5, the heat exchangers associated with Concentrator 12 are the Concentrator Heat Exchanger, the Concentrator Distillate Sub-Cooler, the Concentrator Vent Condenser, and the Concentrator Vapor Condenser. The One-Time Inspection Program is considered to be the appropriate aging management program for these components since they are normally exposed to air and the rate of aging is judged to be so slow that their intended functions would not be impacted during the period of extended operation.

The valves in this system that are WASS in Treated Water ≥140°F, but <212°F are all ¾" valves (mostly ball valves) in either instrument lines or drain lines. As such, the applicable AERM of cracking was considered to be unlikely since there is normally no flow through these lines and it is very improbable that the water temperature is sustained at the high end of the indicated range. For this reason, the One Time Inspection Program was considered to be adequate for aging management of these valves, so it was credited.

(2)(b-d) As discussed in the response to the first part of this RAI, the aging management of the carbon steel and gray cast iron components within the NMP1 Radioactive Waste System is changed from the One Time Inspection Program to the PM Program. For the stainless steel components within the system, it is considered to be unlikely that they will experience the AERMs that have been identified for them. For this reason, the One Time Inspection Program is retained as the AMP.

(2)(e) These pumps are the sump pumps in the Control Building floor drain sump (see Drawing LR-66C-0). These pumps are non-safety-related pumps that are in scope for 10 CFR 54.4(a)(2) because they are located in the Control Building and there is safety-related equipment in the vicinity. Even though the Environment for these pumps has been identified as Raw Water, the water that enters the sump is treated or demineralized water that has leaked onto the floor and drained to the sump. Since there is no chemistry control of this water, it has been identified as Raw Water. The One Time Inspection Program has been credited for aging management since it is considered unlikely that the AERM of LOM would ever occur to the extent such that the loss of the intended function of the pumps would be lost.

For the carbon steel strainers, as discussed in the response to the first part of this RAI, the AMP is to be changed to the PM Program.

System Specific RAI

a-RAI 3.3.2.A-5-1

What does the Note "K" represent for Heat Exchangers, and valves and dampers in Table 3.3.2.A-5? Please explain why the LOM was not identified as an AERM for WASS Heat Exchangers exposed to DUW similar to the WASS Heat Exchangers in Table 3.3.2.A-14.

NMP Response

As part of the recovery effort which led to the submittal of the ALRA, NMP chose to convert the lettered plant-specific notes to the standard industry lettered notes. As discussed with the staff, NMPNS agrees that Note H should be substituted for Note K. The cited locations were the only locations within the ALRA that were impacted. For those two cited locations in Table 3.3.2.A-5, each Note K is, therefore, changed to Note H.

There is a similar Notes anomaly in ALRA Table 3.3.2.B-6 (p. 3.3-217). For the Component Type 'Piping and Fittings', the indicated Notes column entry of 'J' should be 'None' consistent with the other Notes column entries in this table and is changed accordingly.

Also as discussed with the staff, NMPNS agrees that the AERM of LOM should be applied to the WASS heat exchangers in a DUW environment. The following changes are therefore made:

(a) Consistent with Table 3.3.2.A-14, the HT and PB function for the WASS heat exchangers in Table 3.3.2.A-5 should have a line item for the AERM of LOM which is added. For this line item, the AMP is the Closed Cycle Cooling Water System Program with the Note of H. Additionally, the Note 9 for the LOHT AERM line item is removed.

(b) In Table 3.3.2.A-15, for the WASS heat exchangers with HT and PB intended functions in a DUW environment, a line item for the AERM of LOM is added. For this line item, the AMP is the Closed Cycle Cooling Water System Program with the Note of H. The Note 9 for the LOHT intended function line item is removed (additionally, for the LOHT line item, the One Time Inspection Program was removed in NMP letter NMP1L 1996, dated 11/17/05).

(c) In Table 3.3.2.A-17, for the WASS heat exchangers in a DUW environment, the AERM is changed from None to LOM, the AMP is changed from None to the Closed Cycle Cooling Water System Program, and the Note is changed from None to H.

(d) In Table 3.3.2.A-21, for WASS heat exchangers in a DUW environment, a line item for the AERM of LOM is added with the AMP of the Closed Cycle Cooling Water System Program, and Note H. Additionally, the Note 9 in the LOHT AERM line item is removed.

(e) In Table 3.3.2.B-27, for the WASS heat exchangers with the HT and PB intended functions in a DUW environment, a line item for the AERM of LOM is added with the AMP of the Closed Cycle Cooling Water System Program, and Note H. Additionally, the Note 9 in the LOHT AERM line item is removed.

PART 2 – NMP1 ALRA ISSUE REGARDING CONTROL ROD DRIVE STUB TUBE PENETRATIONS

a-RAI 3.1.2-1

In Table 3.1.1.A-1 of the Nine Mile Point (NMP) Amended License Renewal Application (ALRA), dated July 14, 2005, the applicant states, "Aging management of the CRD stub tube penetrations is managed in accordance with BWRVIP-47 of the BWR Vessel Internals Program, XI.M9, and plant-specific commitments contained in the NRC safety evaluation (SE) dated March 25, 1987." Then, by another letter dated July 14, 2005, the applicant provided its response to RAI 3.1.2-1, stating that, "NMP committed to implement a strategy whereby during the period of extended operation a leaking control rod drive (CRD) stub tube penetration would be roll repaired. If, following the roll repair, this stub tube was to leak within acceptable limits, then a weld repair would be effected no later than one operating cycle following discovery of the leakage."

The wording in Table 3.1.1.A-1 and in the applicant's response to RAI 3.1.2-1 imply that NMP Unit 1 will operate with CRD stub tube leakage for one operating cycle (2 years). The staff does not consider this is acceptable for the period of extended operation. The SE, dated March 25, 1987, as cited above, which allows NMP Unit 1 to operate with CRD stub tube leakage, was only acceptable as a temporary repair. Specifically, Item (6) of the staff's conclusions of the aforementioned SE, states that, "The proposed leakage criteria provides sufficient time to complete the final development of the prototype mechanical seal and associated tooling and investigate other methods such as weld repair."

Based on the information above, the staff requests that the applicant revise the Corrective Action statement on Page B2-25 and Commitment 36, in Table A1-4, of the NMP1 ALRA to commit to immediately repair any leaking CRD stub tubes, during the proposed period of extended operation, by the implementation of a permanent weld repair per approved ASME Code Cases with staff conditions, if any. In addition, the staff requests that the applicant revise the "Discussion" section of Item Number 3.1.1.A-30, "Penetrations," in Table 3.1.1.A for NMP1 (Page 3.1-29 of the ALRA) by deleting, "plant-specific commitments contained in the NRC safety evaluation dated March 25, 1987, and by adding, "plant-specific commitments for license renewal as indicated in Commitment 36 of Table A1-4."

NMP Response

NMP has revised ALRA Section B2.1.8, BWR Vessel Internals Program (NUREG 1801, Program XI.M9), Commitment 36 in ALRA Table A1-4, and ALRA Table 1 Item Number 3.1.1.A-30 to clarify its positions, as described below, related to the use of roll/expansion techniques for the repair of leaking NMP1 CRD stub tubes consistent with the request in the 3rd paragraph of the RAI above.

NMP is revising the commitment it made in its response to RAI 3.1.2-1 in NMP Supplemental Letter NMP1L 1928, dated 2/14/2005, as follows:

"As acknowledged by the NRC in the referenced RAI, the ASME Code Committee is evaluating the acceptability of roll/expansion techniques as a permanent repair for CRD stub tubes via Code Case N-730. NMP will continue to follow the status of the proposed ASME code case and will implement the final code case, as conditioned by the NRC, once it has been approved. If the code case is not approved by ASME, then NMP will seek NRC approval of the current draft of Code Case N-730 dated 10/19/05 on a plant specific basis as conditioned by the NRC.

During the period of extended operation, should a CRD stub tube rolled in accordance with the provisions of the code case resume leaking, NMP will implement one of the following zero leakage permanent repair strategies prior to startup from the outage in which the leakage was detected:

1. A welded repair consistent with BWRVIP-58-A, "BWRVIP Internal Access Weld Repair" and Code Case N-606-1, as endorsed by the NRC in Regulatory Guide 1.147.
2. A variation of the welded repair geometry specified in BWRVIP-58-A subject to the approval of the NRC using Code Case N-606-1.
3. A future developed mechanical/welded repair method subject to the approval of the NRC."

The 2nd paragraph of ALRA Table 3.1.1.A Item 3.1.1.A-30 (p. 3.1-29), Commitment 36 from ALRA Table A1.4 (p. A1-42), and the Corrective Action bullet in ALRA Section B2.1.8 (p. B2-25) is replaced with:

"If the 10/19/05 draft of Code Case N-730 is approved by the ASME, NMP1 will implement the final code case as conditioned by the NRC. If the code case is not approved by the ASME, NMP1 will seek NRC approval of the 10/19/05 code case draft on a plant specific basis as conditioned by the NRC.

During the period of extended operation, should a CRD stub tube rolled in accordance with the provisions of the code case resume leaking, NMP will implement one of the following zero leakage permanent repair strategies prior to startup from the outage in which the leakage was detected:

1. A welded repair consistent with BWRVIP-58-A, "BWRVIP Internal Access Weld Repair" and Code Case N-606-1, as endorsed by the NRC in Regulatory Guide 1.147.
2. A variation of the welded repair geometry specified in BWRVIP-58-A subject to the approval of the NRC using Code Case N-606-1.

3. A future developed mechanical/welded repair method subject to the approval of the NRC."

PART 3 – <u>TIME-LIMITED AGING ANALYSES</u>

<u>aRAI 4.2-1</u>

Under TLAA 4.2 on Reactor Vessel Neutron Embrittlement Analysis, the applicant needs to provide a description of the Reflood Thermal Shock Analysis of the Reactor Vessel and the Reflood Thermal Shock Analysis of the Reactor Vessel and the Reactor Vessel Core Shroud for both NMP 1 and NMP 2. If not, then the applicant needs to provide a justification in the application of why these analyses are not needed (i.e. for NMP Unit 1- no jet pumps).

NMP Response

Summary

"Reflood Thermal Shock" refers to an analysis of reactor vessel and/or reactor vessel internals integrity during a low pressure coolant injection event following a LOCA. A similar analysis (applicable to the RPV only) is described as "Analysis of RPV Structural Integrity During a Design Basis Accident" in the NMP2 USAR Section 5.3.3 (Reference 1). During the TLAA identification process for NMP2, NMPNS evaluated this analysis and determined that it did not meet all the criteria for a TLAA. For NMP1, the design of the recirculation loops precludes reflooding and this type of analysis does not exist for NMP1. Therefore, reflood thermal shock is not a TLAA for the NMP1 or NMP2 RPV.

No analyses for reflood thermal shock exist for the NMP1 or NMP2 reactor vessel internals (RVI). Fracture mechanics evaluations do exist for RVI justifying reinspection intervals for components where cracking was detected by inservice examination. These fracture mechanics evaluations account for reduction of fracture toughness of the RVI materials (mainly austenitic stainless steel) as required and consider design basis loading conditions. No such fracture mechanics evaluations are credited for the life of the plant. The longest interval between inspections for cracked RVI components is ten years. Therefore, reflood thermal shock is not a TLAA for the NMP1 or NMP2 RVI.

Basis

Section 5.3.3 of the NMP2 USAR indicates an analysis of the structural integrity of boiling water RPVs during a design basis accident (DBA) has been performed. The analysis ensured NMP2's compliance with Position C of Regulatory Guide (RG) 1.2, Rev. 0 (now withdrawn). While the analysis specifically addressed the BWR/6 vessels only, the analysis was determined applicable to the Unit 2 vessel (Reference 1).

The analysis included:

1. Description of the LOCA event.
2. Thermal analysis of the vessel wall to determine the temperature distribution at different times during the LOCA.

3. Determination of the stresses in the vessel wall including thermal, pressure, and residual stresses.
4. Consideration of radiation effect on material toughness (nil ductility transition temperature [NDTT] shift and changes in toughness).
5. Fracture mechanics evaluation of vessel wall for different postulated flaw sizes.

This analysis incorporated conservative assumptions, particularly in the areas of heat transfer, stress analysis, effects of radiation on material toughness, and crack tip stress intensity factor evaluation. The analysis concluded that even in the presence of large flaws, the vessel will have considerable margin against brittle fracture following a LOCA.

The purpose for the analysis was to address the requirements of RG 1.2 "Thermal Shock to Reactor Pressure Vessels." RG 1.2 states that a suitable program be followed to ensure that the RPV will behave in a non-brittle manner under LOCA conditions. However, RG 1.2 has been withdrawn. The withdrawal notice for the regulatory guide indicates it has been superseded by 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," and by Regulatory Guide 1.154, "Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors." Both of these are applicable only to PWRs. Additionally, 10 CFR 50 Appendix G ensures that reactor vessels will behave in a non-brittle manner under all operating and design conditions. The analyses of the conditional probability of reactor pressure vessel failure summarized in LRA Sections 4.2.3 and 4.2.4 are based on generic analyses for BWR vessels documented in BWRVIP-05 and the associated NRC Safety Evaluation and SER Supplement. These analyses demonstrate that the probability of brittle fracture of BWR vessels is acceptably low given that 100% of the vessel welds are not volumetrically examined during each 10-year inservice inspection interval. The discussion of operational issues in Section 5.0 of BWRVIP-05 indicates that Service Level C and D conditions, under which LOCA falls, are not limiting with respect to the potential for propagation of large flaws, because conditions that cause rapid vessel cooling also cause rapid depressurization. Leak test conditions are more limiting with respect to the propagation of deep flaws due to the low (150-200°F) temperature coupled with the high pressure (~1000 psig). Before issuing a safety evaluation for BWRVIP-05, the NRC staff also required the conditional failure probability of the RPV due to cold overpressure events (which may be outside design basis) be addressed. Some of these events are also more limiting than LOCA. The analyses described in LRA Section 4.2.3 and 4.2.4 consider the projected reduction in fracture toughness of the vessel due to neutron embrittlement at the end of the period of extended operation. Additionally, the adequacy of the upper shelf energy (USE) of the NMP1 and NMP2 reactor vessels is addressed in LRA Section 4.2.1. Plant specific equivalent margins analyses exist for NMP1, and a generic equivalent margins analysis exists that is applicable for NMP1 and NMP2, that considers Service Level A and B (normal and upset) and Service Level C and D (faulted and emergency) conditions. Therefore, generic and plant specific analyses exist and have been evaluated as TLAAs that adequately address the possibility of brittle and ductile fracture under all design basis conditions of the reactor pressure vessels for NMP1 and NMP2. As such, the analysis described in USAR Section 5.3.3 does not contribute meaningfully toward making a safety determination and was not considered a TLAA by NMPNS.

The design of the NMP1 recirculation loops precludes reflooding, so this type of analysis does not exist for NMP1. A LOCA thermal transient does exist for the NMP1 reactor vessel and was included under the Service Level C and D loadings evaluated in the equivalent margins analyses described above.

NMPNS understands that several other BWR license renewal applicants have determined that a reflood thermal shock analysis of the reactor vessel (based on the generic analysis in Reference 2) met the criteria for a TLAA. Therefore, although NMPNS does not consider this analysis a TLAA, NMPNS performed an information-only evaluation of the NMP2 RPV against the generic BWR/6 analysis as follows:

The generic analysis (Reference 2) postulated a ¼ T inner surface flaw in the vessel beltline region (where T is the vessel thickness not including cladding). The analysis assumed end-of-life material toughness. The peak stress intensity occurred 300 seconds into the event. At the time of the peak stress intensity, the temperature at the ¼ T depth was approximately 400°F. The analysis assumed a minimum fracture toughness (K_{IC}) of 200 ksi$(in)^{0.5}$ was available based on the temperature at the ¼ T location at the time of the peak stress intensity.

The projected adjusted reference temperature (ART, the RT_{NDT} value adjusted for irradiation) at the end of the period of extended operation (54 EFPY) for the limiting material in the NMP2 reactor vessel beltline, given in LRA Table 4.2-4, is 67.9°F. Therefore, the temperature at the critical location at the time of peak stress is 332.1°F greater than RT_{NDT}. Figure A-4200-1 of Reference 3, shows that at temperatures of 200°F above RT_{NDT} or greater, the fracture toughness would exceed 200 ksi $(in)^{0.5}$. Therefore, the fracture toughness (K_{IC}) at 400°F of the limiting beltline material is bounded by the K_{IC} used in the fracture mechanics analysis.

a-RAI 4.7-1

The applicant needs to include a TLAA in Section 4.7 for the Irradiation Assisted Stress Corrosion Cracking of Reactor Vessel Internals.

NMP Response

Summary

NMP1 and NMP2 have several analyses evaluating the extent of crack growth of known indications in reactor vessel internals components that may propagate due to IASCC. None of these analyses involve time-limited assumptions defined by the current operating term, i.e., 40 years. Therefore, the third criteria from 10 CFR 54.3 for a TLAA, "(3) Involve time-limited assumptions defined by the current operating term, for example, 40 years," is not satisfied for this issue.

Basis

NMP1 and NMP2 have performed several analyses that predict the extent of crack growth in core shroud welds in which indications have been found. A neutron fluence of $5x10^{20}$ n/cm^2 is considered the threshold for BWR IGSCC accelerated crack growth rate

(CGR) due to fluence in BWR SS components and the potential CGR and fracture toughness used to disposition flaws uses NRC approved guidance in BWRVIP-99 and BWRVIP-100. The industry BWR Vessel Internals Program (BWRVIP) considers the inspection scope and intervals for all components to remain valid for detection for internal components SCC (both IG and IA) including the components that exceed $5x10^{20}$ n/cm^2. The NRC license renewal SE's provide concurrence on this point. The only location for which an NRC LR SE identified the need for a TLAA related to IASCC was the top guide grid beam addressed in BWRVIP-26. This location has specifically been addressed at NMP. NMPNS is managing IASCC of this location through the inspection program rather than relying on a TLAA since the location is predicted to have fluence greater than the $5x10^{20}$ n/cm^2 for the 40 year term for both units and cracking has been identified and characterized as likely IASCC at NMP1.

NMP1 is not crediting plant specific flaw evaluations for core shroud operability. In accordance with BWRVIP-76 guidelines, NMP1 has applied conservative crack growth rate (CGR) assumptions for non-structurally significant indications and locations where inspection coverage was less than 50% of the weld, such as for vertical welds V3, V4, V7, and V8. These evaluations are not used to justify operation for the life of the plant, but to justify operation until the next scheduled inspection. The mechanical repairs installed on the NMP1 core shroud replace the structural function of cracked core shroud circumferential welds H1 through H8 and vertical welds V9 and V10; thus, crack growth analyses for these welds are not credited. The components of the mechanical repairs (tie rods for the horizontal welds and clamps for the vertical welds) generally will not be exposed to fluences of $5x10^{20}$ n/cm^2 or greater during the remaining plant life including the period of extended operation. The only possible exception is the inboard end of the vertical repair clamp threaded pin that could reach a fluence of approximately $5x10^{20}$ n/cm^2. There are currently no calculations or analyses evaluating IASCC associated with the vertical repair clamp threaded pins. IASCC was not specifically addressed in the design of the vertical repair clamps or the tie rods but the potential for IGSCC was addressed through the selection of resistant materials and fabrication processes. The same factors that influence IGSCC are believed to influence IASCC resistance as well. IASCC crack growth analyses and flaw evaluations are typically performed in response to indications discovered during inspections rather than to evaluate the potential for IASCC initiation.

NMP1 does currently credit a crack growth/flaw tolerance analysis for the H9 weld. The H9 weld analysis was a limit load type analysis that considered expected crack growth for 80,000 hours (10 years of operation) . The H9 weld is a low fluence weld so the cracking evaluated is considered to be intergranular stress corrosion cracking (IGSCC) rather than irradiation assisted stress corrosion cracking (IASCC).

For NMP2, IASCC-related analyses justifying reinspection intervals have been submitted to the NRC, and subsequently approved via NRC Safety Evaluation Reports. These analyses account for the effects of IASCC through the use of crack growth rate assumptions based on the predicted fluence at the end of the time period covered by the analysis. However, none of these analyses contain a time-limited assumption related to the life of the plant. These analyses were performed for time periods shorter than 40 years to justify inspection intervals for augmented inspection. The longest duration covered by any of the analyses for the NMP2 core shroud is 3 refueling cycles.

With respect to IASCC of the top guide, NMPNS has concluded that both NMP1 and NMP2 have exceeded their neutron fluence threshold for IASCC susceptibility. As such, the top guide grid beam inspections recommended in GE Service Information Letter (SIL) 554 have been incorporated into the BWRVIP Inspection Plans for both units. NMP1 has completed the first inspections and NMP2 will perform the first inspections during its next refueling outage. Therefore, evaluating the projected accumulated neutron fluence as a potential TLAA is considered unwarranted. NMP1 is not relying on a flaw evaluation to demonstrate acceptability of the top guide through the end of life, but rather plans enhanced inspections. Ultrasonic examination of 100% of the grid beams was performed during RFO-18 as an alternative to the EVT-1 examination recommended by SIL 0554. Indications were found, some of which were consistent with IASCC. The inspection findings are being addressed under the NMP Corrective Action Program and currently require reinspection during the next refueling outage. Therefore, no calculations for the life of the plant are relied upon to demonstrate continued acceptability of the top guide.

Conclusion: There are no calculations or analyses related to IASCC of the reactor vessel internals for NMP1 or NMP2 that involve time-limited assumptions defined by the current operating term, for example, 40 years. Therefore, criterion 10 CFR 54.3(a)(3) for a TLAA is not satisfied for this issue.

ATTACHMENT 2

List of Regulatory Commitments

The following table identifies those actions committed to by Nine Mile Point Nuclear Station, LLC (NMPNS) in this submittal. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

REGULATORY COMMITMENT	DUE DATE
NMP is revising the commitment it made in its response to RAI 3.1.2-1 in NMP Supplemental Letter NMP1L 1928, dated 2/14/2005, as follows: "As acknowledged by the NRC in the referenced RAI, the ASME Code Committee is evaluating the acceptability of roll/expansion techniques as a permanent repair for CRD stub tubes via Code Case N-730. NMP will continue to follow the status of the proposed ASME code case and will implement the final code case, as conditioned by the NRC, once it has been approved. If the code case is not approved by ASME, then NMP will seek NRC approval of the current draft of Code Case N-730 dated 10/19/05 on a plant specific basis as conditioned by the NRC. During the period of extended operation, should a CRD stub tube rolled in accordance with the provisions of the code case resume leaking, NMP will implement one of the following zero leakage permanent repair strategies prior to startup from the outage in which the leakage was detected: 1. A welded repair consistent with BWRVIP-58-A, "BWRVIP Internal Access Weld Repair" and Code Case N-606-1, as endorsed by the NRC in Regulatory Guide 1.147. 2. A variation of the welded repair geometry specified in BWRVIP-58-A subject to the approval of the NRC using Code Case N-606-1. 3. A future developed mechanical/welded repair method subject to the approval of the NRC."	NMP1: August 22, 2009