February 14, 2005
NMP1L 1928

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

License Renewal Application – Responses to NRC Requests for Additional
Information Regarding the Reactor Vessel and Reactor Vessel Internal
Components (TAC Nos. MC3272 and MC3273)

Gentlemen:

By letter dated May 26, 2004, Nine Mile Point Nuclear Station, LLC (NMPNS) submitted an
application to renew the operating licenses for Nine Mile Point Units 1 and 2.

In a letter dated January 13, 2005, the NRC requested additional information regarding the aging
management reviews and aging management programs for the reactor vessel and reactor vessel
internal components. 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 Peter Mazzaferro, NMPNS
License Renewal Project Manager, at (315) 349-1019.

Very truly yours,

[Signature]

James A. Spina
Vice President Nine Mile Point

JAS/DEV/sac
STATE OF NEW YORK
    \nCOUNTY 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 this supplemental information 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.

Subscribed and sworn before me, a Notary Public in and for the State of New York and County of Oswego, this 14th day of February 2005.

WITNESS my Hand and Notarial Seal:

[Signature]

SANDRA A. OSWALD
Notary Public

My Commission Expires: 02/04/05

Attachments:
1. Responses to NRC Requests for Additional Information (RAI) Regarding the Aging Management Reviews and Aging Management Programs for the Reactor Vessel and Reactor Vessel Internal Components
2. List of Regulatory Commitments

cc: Mr. S. J. Collins, NRC Regional Administrator, Region I
    Mr. G. K. Hunegs, NRC Senior Resident Inspector
    Mr. P. S. Tam, 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 the Aging Management Reviews and Aging Management Programs for the Reactor Vessel and Reactor Vessel Internal Components

This attachment provides the Nine Mile Point Nuclear Station, LLC (NMPNS) responses to the requests for additional information contained in the NRC letter dated January 13, 2005, regarding the reactor vessel and reactor vessel internal components. 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. Revisions to the License Renewal Application (LRA) are described where appropriate. The revisions are highlighted by shading unless otherwise noted.

Aging Management Review in Tables 3.1.2-A and B

RAI 3.1.2-1

LRA Table 3.1.2.A-1 indicated that the Reactor Vessel (RV) penetrations are made of carbon or low alloy steel, nickel based alloys, and wrought austenitic stainless steel. The applicant stated that for the vessel drains, made of carbon or low alloy steel, that loss of material is an applicable aging effect and that this aging effect will be managed through the implementation of the ASME Section XI Inservice Inspection (ISI) Program and the water chemistry program. Please explain how the ISI Program can adequately manage loss of material; i.e., provide details of what part of Section XI of the Code addresses loss of material of the vessel drains.

The staff has determined from the BWRVIP Report, BWRVIP-17, that the CRD stub tubes are fabricated from stainless steel, and that some portions of the stainless steel might have been procured in a sensitized condition. The BWRVIP-17 report also indicated that these components have experienced cracking and that through-wall leakage of reactor coolant had occurred. In 1987, the staff issued to NMP1 a temporary relief to perform a roll expansion repair of CRD housings to stop or limit RCS leakage. The staff is aware that the applicant has used this relief request (which allows the use of roll/expansion) as the basis for repairing/correcting through-wall leakage for these components at NMP. Based on the above staff review, please provide the following information:

(1) Identify the potential aging effects for these stainless steel/sensitized stainless steel CRD stub tubes. Identify the applicable aging management review (AMR) entry for these CRD stub tubes and provide information to explain how the new proposed AMR(s) will
manage all potential aging effects (including cracking by both thermal fatigue and stress corrosion cracking [SCC]) that are applicable to these components during the extended period of operation.

(2) With respect to implementing roll/expansion techniques as alternative repair methods, the staff is concerned that NMP will consider these techniques as permanent repair methods for through-wall flaws for the two 10-year ISI intervals in the period of extended operation for NMP1. The staff emphasizes that any relief requests submitted under current 10-year inservice inspection intervals are not applicable to the two 10-year ISI periods in the extended periods of operation unless a new relief request for the new intervals for these ISI intervals is approved through applicable provisions in 10 CFR 50.55a. At present the ASME Code is evaluating the acceptability of this type of repair as a permanent repair method.

The staff also needs to emphasize that the industry's most current basis for implementing roll/expansion repairs is given in the Boiling Water Reactor Vessel and Internals Project (BWRVIP) Report, "Roll/Expansion Repair of Control Rod Drive and In-Core Instrument Penetrations in BWR Vessels (BWRVIP-17)," with some modifications. In the staff's letter and safety evaluation on BWRVIP-17, dated March 13, 1998, the NRC staff articulated its position that the corrective action required by the ASME Code, upon discovery of an unacceptable flaw in a Class I pressure retaining boundary component, is to either repair the flaw or replace the flawed component in order to return it to a condition of ASME Code compliance. Thus, in the aforementioned SE, the staff took a position that repair of a crack in a CRD stub tube or in-core penetration would require an ASME Code-acceptable weld repair. In taking this position, the staff articulated, that although the BWRVIP roll/expansion method may, for some time period, control the symptom of the flaw (leakage), it would not remove the flaw either in its entirety or conforming to an ASME Section XI acceptable criterion; and therefore, the BWRVIP roll/expansion method would not meet the criteria or the intent of a permanent repair method.

Additionally, in issuing its SE on BWRVIP-17, the NRC staff established its position that the alternative roll/expansion method in the BWRVIP-17 report does not provide a sufficient basis for authorizing a permanent alternative pursuant to 10 CFR 50.55a(a)(3), and therefore had denied the BWRVIP-17 generic roll/expansion application as an alternative permanent repair for CRD stub tubes. Based on its position for denying the alternative roll/expansion methodology in the BWRVIP-17 report, the staff will not entertain submittal of a corresponding relief request for the two 10-year ISI intervals in the period of extended operation. Therefore, NMP will need to provide a commitment to perform the following actions no later than the first available opportunity in the extended period of operation for NMP1:

(A) Should the ASME Code determine that a roll expansion repair is an acceptable permanent repair and the NRC staff endorses the Code Case for this repair method, then NMP1 should comply with the requirements of the new Code Case.
(B) Should the ASME Code determine that a roll expansion repair is not an acceptable permanent repair method, then NMP1 should effect a permanent Code repair using a NRC-approved Code Case or other repair option acceptable to the NRC.

This commitment should also be stated in the updated final safety analysis report (UFSAR) supplement summarizing the applicable aging management program accordingly.

Response

Vessel Drains

The NMP1 reactor pressure vessel (RPV) drain line penetration is made from carbon steel and is exposed to an internal environment of high temperature treated water. The corresponding aging effects requiring management are cumulative fatigue damage and loss of material due to flow-accelerated corrosion. The applicable aging management program for cumulative fatigue damage is the Fatigue Monitoring Program. For the loss of material aging effect, the ASME Section XI Inservice Inspection and Water Chemistry Control Programs were originally assigned as the aging management programs since the nozzle penetration weld is inspected in accordance with ASME Section XI requirements and the quality of the reactor water is controlled by the Water Chemistry Control Program. This was determined to be sufficient due to the short length of pipe comprising the actual penetration. However, based upon related activities ongoing in the industry, NMPNS is revising the applicable aging management program for the loss of material aging effect to the Flow-Accelerated Corrosion Program. This revision is applicable to NMP1 and NMP2. As such, LRA Tables 3.1.2.A-1 and 3.1.2.B-1 are being revised to reflect this change in aging management programs.

LRA Revisions

(Note: Section, table, and page numbers cited for LRA Section 3.1 refer to the revised version of LRA Section 3.1 that was submitted by NMPNS letter NMP1L 1892, dated December 6, 2004.)

In LRA Sections 3.1.2.A.1 (page 3.1-5) and 3.1.2.B.1 (page 3.1-12), under the “Aging Management Programs” heading, “Flow-Accelerated Corrosion Program” is added.

LRA Tables 3.1.2.A-1 (page 3.1-43) and 3.1.2.B-1 (page 3.1-72) are revised to credit the Flow-Accelerated Corrosion Program for the RPV drain line penetrations, as shown on the pages at the end of the RAI response.

Control Rod Drive (CRD) Stub Tubes

NMPNS acknowledges that NMP1 has operating experience relative to CRD stub tubes that is being managed for the current operating term and will require management in the period of extended operation. In 1984, NMP1 experienced leakage from CRD stub tubes and received initial concurrence from the NRC to utilize a roll repair method to limit the leakage from the penetrations and assure safe plant operation. By letter dated March 25, 1987, the NRC approved
the roll repair as an alternative, per 10 CFR 50.55a(a)(3), to the requirements of ASME Section XI, paragraph IWA 5250(a)(2), with respect to testing and repair of the CRD stub tubes. The basis for this approval is documented in the NRC safety evaluation and is still applicable based upon operating experience to date. The alternative repair approved in 1987 has proven to be effective in complete stoppage of the identified leakage at all but one of the thirty-three (33) CRD penetrations roll repaired to date. The one outlier, which remained leak tight for ten (10) years, is scheduled to be roll repaired during the upcoming refueling outage in March/April 2005. In addition, since NMP1 implemented Noble Metals and Hydrogen Water Chemistry in year 2000, only one CRD penetration has experienced a leak (which was roll repaired in the spring of 2001). Based upon the technical and safety bases for the roll repair method documented in the NRC safety evaluation, NMP1 operating experience, and the lack of an expiration date on the safety evaluation, NMPNS considers the roll repair technique to be an acceptable long term repair method approved pursuant to NRC regulations.

The above RAI requested NMPNS to commit to one of two options for managing leakage from CRD penetrations in the period of extended operation based upon the outcome of an ASME Code Committee review of a roll repair method. Given that this issue is still under review, NMPNS cannot commit to an unknown item. However, using existing approved methods, NMPNS plans to implement a strategy whereby a leaking CRD stub tube penetration would be roll repaired. If, following the roll repair, this stub tube were to leak within acceptable limits, then a weld repair would be effected no later than one operating cycle following discovery of the leakage. The potential delay in up to one operating cycle before implementing a weld repair is based on the absence of a safety issue, the inability to discover a leak following roll repair until the end of an outage during the ASME Section XI pressure test, and avoidance of the increased radiation exposure caused by performing a weld repair on an emergent time schedule versus a planned schedule. Additionally, the contingency costs for having a weld repair available at every outage would be avoided. NMPNS considers this strategy to be a responsible one that maintains nuclear safety and is consistent with existing approved strategies. Future discussions with the NRC staff are requested to further explain the technical and safety bases of this strategy.

With respect to the information described in the LRA for the CRD stub tubes, the following provides the requested information. The CRD stub tubes are fabricated of wrought austenitic stainless steel with the stub-tube-to-vessel weld being a nickel based alloy. These components are exposed to an environment of high temperature treated water. The corresponding aging effects requiring management are cumulative fatigue damage (managed by the Fatigue Monitoring Program) and cracking due to stress corrosion cracking (managed by the BWR Vessel Internals and Water Chemistry Control Programs). These AMR results are included in LRA Table 3.1.2.A-1 (page 3.1-43) under Component Type “Penetrations: CRD Stub Tube” and are consistent with NUREG-1801, Volume 2, Items IV.A1.5-a and b guidance. In addition to these programs, NMPNS performs augmented inspections based upon commitments associated with NRC approval of the alternative repair described above. These inspections include a UT examination of at least two previously roll-repaired CRD stub tubes if made available through normal CRD drive maintenance each refueling outage, and periodic in-vessel visual inspections in the lower plenum when access is provided. All CRD housings and penetrations also receive a VT-2 examination for evidence of leakage during the ASME Section XI reactor vessel pressure test conducted each refueling outage, and a visual examination for leakage during each mid-cycle
shutdown when the drywell is de-inerted. Based upon the above, there is reasonable assurance that the NMP1 CRD stub tubes will be adequately managed during the period of extended operation.

LRA Revisions

- LRA Section A1.1.12 (page A1-5) is revised to add a second paragraph as follows:

  “In addition to the guidelines issued by the BWRVIP, the repair for NMP1 CRD stub tube leaks requires resolution for the period of extended operation. Using existing approved methods, NMPNS plans to implement a strategy whereby a leaking CRD stub tube penetration would be roll repaired. If, following the roll repair, this stub tube were to leak within acceptable limits, then a weld repair would be effected no later than one operating cycle following discovery of the leakage.”

- In LRA Section B2.1.8 (page B-21), under the “Program Description” heading, the following is added as the third paragraph:

  “In addition to the guidelines issued by the BWRVIP, the repair for NMP1 CRD stub tube leaks requires resolution for the period of extended operation. Using existing approved methods, NMPNS plans to implement a strategy whereby a leaking CRD stub tube penetration would be roll repaired. If, following the roll repair, this stub tube were to leak within acceptable limits, then a weld repair would be effected no later than one operating cycle following discovery of the leakage.”

- LRA Tables 3.1.1.A (page 3.1-27) and 3.1.2.A-1 (page 3.1-43) are revised to reflect that for the CRD stub tubes, the BWR Vessel Internals Program is credited for managing the aging effect of cracking due to stress corrosion cracking, as shown on the following pages.
Table 3.1.1.A NMP1 Summary of Aging Management Programs for the Reactor Vessel, Internals, and Reactor Coolant Systems Evaluated in Chapter IV of NUREG-1801

<table>
<thead>
<tr>
<th>Item Number</th>
<th>Component</th>
<th>Aging Effect/ Mechanism</th>
<th>Aging Management Programs</th>
<th>Further Evaluation Recommended</th>
<th>Discussion</th>
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<td>3.1.1.A-30</td>
<td>Penetrations</td>
<td>Crack initiation and growth due to SCC, IGSCC, and/or cyclic loading</td>
<td>BWR bottom head penetrations; water chemistry</td>
<td>No</td>
<td>Consistent with NUREG-1801 with exceptions (see Appendix B2.1.2). 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 dated March 25, 1987.</td>
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<td>Component Type</td>
<td>Intended Function</td>
<td>Material</td>
<td>Environment</td>
<td>Aging Effect Requiring Management</td>
<td>Aging Management Program</td>
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<td>Nozzle Safe Ends (cont'd)</td>
<td>PB (cont'd)</td>
<td>Wrought Austenitic Stainless Steel</td>
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<td>Cracking</td>
<td>BWR Stress Corrosion Cracking Program, Water Chemistry Corrosion Program</td>
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<td>Cumulative Fatigue Damage Program</td>
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<td>• CRD Stub Tube</td>
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<td>• Flux Monitor</td>
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<td>• Instrumentation</td>
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<td>• Vessel Drain</td>
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<td>Penetrations:</td>
<td>PB</td>
<td>Carbon or Low Alloy Steel</td>
<td>Treated Water or Steam; High Temperature - BWR Reactor Pressure Vessel</td>
<td>Cumulative Fatigue Damage</td>
<td>Fatigue Monitoring Program</td>
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<td>(Yield Strength &lt; 100 KSI)</td>
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<td>Nickel Based Alloys; Wrought Austenitic Stainless Steel</td>
<td>Treated Water or Steam, High Temperature - BWR Reactor Pressure Vessel</td>
<td>Cracking</td>
<td>BWR Penetrations Program, Water Chemistry Corrosion Program</td>
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Table 3.1.2.A-1 Reactor Vessel, Internals, and Reactor Coolant System
NMP1 Reactor Pressure Vessel – Summary of Aging Management Evaluation
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<th>Component Type</th>
<th>Intended Function</th>
<th>Material</th>
<th>Environment</th>
<th>Aging Effect Requiring Management</th>
<th>Aging Management Program</th>
<th>NUREG-1801 Volume 2 Item</th>
<th>Table 1 Item</th>
<th>Notes</th>
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<td>Penetrations:</td>
<td>Core Differential Pressure and Liquid Control</td>
<td>Carbon or Low Alloy Steel (Yield Strength &lt; 100 KSI)</td>
<td>Treated Water or Steam, High Temperature - BWR Reactor Pressure Vessel</td>
<td>Cumulative Fatigue Damage</td>
<td>Fatigue Monitoring Program</td>
<td>IV.A1.3-d</td>
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<td>CRD Stub Tubes</td>
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<td>Loss of Material</td>
<td>Flow-Accelerated Corrosion Program</td>
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<td>Nickel Based Alloys</td>
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<td>Treated Water or Steam, High Temperature - BWR Reactor Pressure Vessel</td>
<td>Cracking</td>
<td>BWR Penetrations Program</td>
<td>Water Chemistry Control Program</td>
<td>IV.A1.5-a</td>
<td>3.1.1.B-30</td>
<td>B</td>
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<tr>
<td>Wrought Austenitic Stainless Steel</td>
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<td>Treated Water or Steam, High Temperature - BWR Reactor Pressure Vessel</td>
<td>Cracking</td>
<td>BWR Penetrations Program</td>
<td>Water Chemistry Control Program</td>
<td>IV.A1.5-a</td>
<td>3.1.1.B-30</td>
<td>B</td>
</tr>
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</table>

Page 8 of 17
RAI 3.1.2-2

(Not used)

RAI 3.1.2-3

LRA Tables 3.1.1.A and 3.1.1.B identified crack initiation and growth due to SCC and/or IGSCC for the reactor vessel closure studs and stud assembly for NMP1 and NMP2. Please identify whether the reactor closure studs and stud assembly have experienced aging effects such as distortion/plastic deformation due to stress relaxation, and loss of material due to mechanical wear. If so, please provide information to explain how the reactor head closure stud program manages these aging effects, or identify other program(s) that will manage these aging effects.

Response

NMPNS has reviewed the results of examinations performed on the NMP1 and NMP2 reactor head closure studs and stud assemblies and has not identified any aging effects. The operating experience for these components indicates that nicks, scratches, gouges, and thread damage have occurred due to maintenance activities during refueling outages and were determined to be acceptable for continued service. There have been no deficiencies attributed to distortion/plastic deformation due to stress relaxation or loss of material due to mechanical wear. The NMP Reactor Head Closure Stud Program has been shown to be effective in managing the aging effects of the reactor head closure studs and stud assemblies.

RAI 3.1.2-4

The requirements of BWRVIP-48 apply to jet pump raiser brace attachment, core spray piping bracket attachment, steam dryer support and hold down brackets, feedwater spargers, guide rod and surveillance sample holder. Section 2.2.3 of BWRVIP-48 indicated that furnace-sensitized stainless steel vessel ID attachment welds are highly susceptible to IGSCC. Please provide information to identify whether there are any furnace-sensitized stainless steel attachment welds at both NMP1 and NMP2 units, and identify aging management program(s) for managing potential aging effects for any existing furnace-sensitized stainless steel attachment welds. Please also provide details on any additional augmented inspection program that is implemented for any existing furnace-sensitized stainless steel attachment welds at both NMP1 and NMP2 units.

Response

NMPNS implements the guidelines contained in BWRVIP-48, BWR Vessel and Internals Project, Vessel ID Attachment Weld Inspection and Flaw Evaluation Guidelines, for NMP1 and NMP2. NMP1 does not have jet pumps or attachments for the core spray piping. However, there are vessel attachment welds for the surveillance sample holder brackets (upper and lower), feedwater sparger brackets, guide rod (track bracket), and steam dryer brackets. The NMP2 design includes each of the components listed in the RAI and has corresponding attachment welds. The NMP1 attachment lugs are fabricated from stainless steel, SA-240, TP 304, and the
attachment welds are Inconel 182 welds except for the surveillance sample holders which are 308 SS filler metal. The NMP1 fabrication records are not definitive regarding the sequence of installation of the attachment welds (i.e., welded before or after vessel post-weld heat treatment). Therefore, these attachment welds are conservatively assumed to be furnace-sensitized. For NMP2, the fabrication records document that the stainless steel attachment welds were welded following vessel heat treatment and, therefore, are not furnace sensitized.

The BWR Vessel Internals Program is credited with managing the aging effects of the attachment welds for NMP1 and NMP2 in accordance with the guidelines contained in BWRVIP-48. For NMP1, an enhanced EVT-1 inspection is performed on the steam dryer bracket and feedwater sparger bracket attachment welds. The surveillance sample holder and guide rod (track bracket) attachment welds are inspected via a VT-1 exam for both the beltline region lower bracket and the non-beltline region upper bracket, which is consistent with the enhanced resolution inspection recommended by BWRVIP-48 guidelines for beltline locations. These inspections are in addition to those required by the ASME Section XI Inservice Inspection Program. For NMP2, the core spray piping bracket and jet pump riser brace bracket attachment welds receive an enhanced EVT-1 exam, while the remainder of the attachment welds are inspected via a VT-3 exam. There are no additional augmented inspection programs credited with aging management for the NMP1 and NMP2 reactor vessel inside diameter (ID) attachment welds.

RAI 3.1.2-5
(Not used)

RAI 3.1.2-6
(Not used)

RAI 3.1.2-7
(Not used)

RAI 3.1.2-8

In LRA Table 3.1.2.B-2, the applicant did not identify cracking due to stress corrosion cracking (SCC, including irradiated assisted stress corrosion cracking or IASCC) or loss of fracture toughness due to thermal aging as applicable aging effects for the jet pump assemblies or the orificed fuel supports. Jet pump assemblies and orificed fuel supports are both fabricated from cast austenitic steel (CASS) and are exposed to treated water or a steam high temperature environment. Please provide NMP2 basis of why cracking due to SCC (including IASCC) or loss of fracture toughness due to thermal aging is not considered to be applicable aging effect for the jet pump assemblies or the orificed fuel supports that are fabricated from CASS. If cracking due to SCC (including IASCC) or loss of fracture toughness due to thermal aging is considered to be applicable aging effect for these components, please identify an acceptable inspection-based aging management program or combination of programs to manage these aging effects. In
addition, the LRA did not appear to have addressed all components that are fabricated of cast austenitic stainless steel and exposed to treated water or steam, and high temperatures. Please provide information to indicate that other cast austenitic stainless steel components at NMP1 and NMP2 meet the material specification requirements as stated in the aging management programs, GALL XM12 or GALL XM13. If not, please commit to the GALL XM12 or GALL XM13 aging management programs for these components as required by 10 CFR 54.21(a)(1) and (a)(2).

Response

In the NRC license renewal safety evaluation (SE) for BWRVIP-41, the staff noted that neutron embrittlement and/or thermal embrittlement of CASS components becomes a concern only if cracks are present in the components. Therefore, if the individual applicant can show that cracks have not occurred in the CASS components, then the staff can conclude that loss of fracture toughness resulting from neutron embrittlement and/or thermal embrittlement will not be a significant aging effect. Based on this NRC license renewal SE, NMPNS has reviewed the existing BWRVIP inspection program and has concluded that adequate inspections of CASS components exist in the base BWRVIP inspection program, and that a unique inspection program is not required to satisfy the NRC license renewal SE with regard to CASS components. Existing inspections include the following:

- The cast elbow and nozzle components are joined to a wrought tube, referred to as a sleeve (ref. BWRVIP-41, Figure 2.3.6-2). BWRVIP-41 requires inspection of the wrought side of welds IN-1 and IN-2, and EVT-1 inspection of the cast side does take place. These inspections would identify cracking in the CASS components if it exists in the vicinity of the weld.
- In the case of: (a) Diffuser collar-to-diffuser shell weld, DF-1, and (b) Riser-to-Transition piece weld, RS-3, BWRVIP-41 requires EVT-1 (or UT) inspection of the wrought side of the weld. Inspection of the diffuser collar or transition piece casting in the vicinity of welds is performed and is able to detect cracking in this region.
- The restrainer bracket inspection per BWRVIP-41 requires inspection of the main wedge, WD-1, and the adjusting set screw locations, AS-1 and AS-2. The inspection includes the restrainer bracket.
- BWRVIP-47 requires inspection of weld CRGT-3. The weld on both the casting and wrought side gets EVT-1 inspected. Therefore, any cracking in the CASS component in the vicinity of the weld will be detected.
- In the case of the Orificed Fuel Support (OFS) casting, the casting gets removed during certain maintenance activities during refueling outages. During this activity maintenance procedures include overall inspection of the OFS. These inspections will be enhanced to include a sample VT-1 inspection of the casting and an EVT-1 inspection if any evidence of impact or mishandling is identified.

The inspections to date have not identified cracking in BWR CASS components. The above inspections are judged to be capable of detection of cracking in the CASS components. It is concluded that the existing BWRVIP inspection program provides adequate inspection of CASS
components that is consistent with the BWRVIP-41 and BWRVIP-47 NRC license renewal safety evaluations.

As stated in the NRC license renewal SE for BWRVIP-41, the BWRVIP and the NRC Office of Nuclear Regulatory Research (RES) is engaged in a joint confirmatory research program to determine the effects of high levels of neutron fluence on BWR internals. NMPNS will evaluate the results of the joint BWRVIP/RES program and determine the need for additional inspections of CASS jet pump components during the period of extended operation based on the results of this ongoing research.

The following reactor pressure vessel internals cast components exist at NMP1 and NMP2:

**NMP1 and NMP2:**
- Orificed Fuel Support (OFS)
- Control Rod Guide Tube Base

**NMP2 cast components applicable to Jet Pumps:**
- Transition Piece (BWRVIP-41, Fig. 2.3.5-1)
- Restrainer Bracket (BWRVIP-41, Fig. 2.3.8-3)
- Inlet Mixer Assembly (BWRVIP-41, Fig. 2.3.7-4)
- Elbow Casting (BWRVIP-41, Fig. 2.3.6-2)
- Nozzle (BWRVIP-41, Fig. 2.3.6-2)
- Diffuser collar/ Guides (BWRVIP-41, Fig. 2.3.9-4)

These internals components are constructed from ASTM A351, Grade CF8 (or CF3) material, with additional requirements imposed by General Electric Company. As indicated in Section 2.2.3 of BWRVIP-41, the cast stainless steel is a duplex structure consisting of austenite and up to 25 percent ferrite. Consistent with the BWRVIP-41 license renewal SE, the NMPNS BWRVIP program assessment is that there is potential susceptibility to embrittlement due to fluence and thermal aging for some of the jet pump CASS components and the OFS. However the impact of the embrittlement is loss of fracture toughness, not crack initiation. The NMP BWRVIP program includes evaluations of fracture toughness, which account for loss of fracture toughness from both fluence and thermal aging. Therefore, if cracking of CASS components is identified, adequate controls exist to ensure that potential loss of fracture toughness is accounted for.
RAIs on Aging Management Programs

RAI B2.1.19-1: Reactor Vessel Surveillance Program

The applicant stated in the LRA that NMP2 will implement the BWRVIP integrated surveillance program (ISP) BWRVIP-116, "BWR Vessel Internals Project Integrated Surveillance Program Implementation for License Renewal," which is currently being reviewed by the staff. If the BWRVIP-116 report is not approved by the staff, then the applicant must submit a plant specific surveillance program for each NMP unit, two years prior to the commencement of the extended period of operation. Please provide NMP's commitment to indicate that it will implement either BWRVIP-116, as approved by the staff, or if the ISP is not approved two years prior to the commencement of the license renewal period, a plant specific surveillance program for each NMP unit will be submitted. This commitment should also be stated in the updated final safety analysis report (UFSAR) Section A.1.25, "Reactor Vessel Surveillance Program," of the LRA.

Response

NMPNS addresses the reactor vessel surveillance program in LRA Sections A1.1.32, A2.1.32 and B2.1.19. These sections indicate that NMP1 and NMP2 commit to follow the guidelines of BWRVIP-116 for the period of extended operation. If the NRC does not approve BWRVIP-116, then NMPNS will submit a plant specific reactor vessel surveillance plan two years prior to commencement of the period of extended operation. Revisions to the UFSAR supplements for license renewal to clarify the NMPNS commitments are described below.

LRA Revisions

LRA Section A1.1.32

- Revise LRA Section A1.1.32, fourth sentence of the first paragraph (page A1-15), as follows:

  "NMPNS commits to implement the Integrated Surveillance Program (ISP) described in BWRVIP-116 (if approved by the NRC staff)."

- Revise LRA Section A1.1.32 (page A1-15) to add the following sentence to the end of the first paragraph:

  "Should BWRVIP-116 not be approved by the NRC, a plant specific reactor vessel surveillance program will be submitted to the NRC two years prior to commencement of the period of extended operation."

- Revise LRA Section A1.1.32, first bullet item (page A1-15), as follows:

  "Incorporate the requirements and elements of the ISP, as documented in BWRVIP-116 and approved by the NRC, or an NRC approved plant-specific program, into the Reactor Vessel Surveillance Program."

Page 13 of 17
LRA Section A2.1.32

- Revise LRA Section A2.1.32, fourth sentence of the first paragraph (page A2-14), as follows:
  "NMPNS commits to implement the Integrated Surveillance Program (ISP) described in BWRVIP-116 (if approved by the NRC staff)."

- Revise LRA Section A2.1.32 (page A2-14) to add the following sentence to the end of the first paragraph:
  "Should BWRVIP-116 not be approved by the NRC, a plant-specific reactor vessel surveillance program will be submitted to the NRC two years prior to commencement of the period of extended operation."

- Revise LRA Section A2.1.32, first bullet item (page A2-14), as follows:
  "Incorporate the requirements and elements of the Integrated Surveillance Program, as documented in BWRVIP-116 and approved by the NRC, or an NRC approved plant-specific program, into the Reactor Vessel Surveillance Program."

LRA Section B2.1.19

- Revise LRA Section B2.1.19 (page B-41), under the "Exceptions to NUREG-1801" heading, to add the following as the last sentence of the paragraph:
  "Should BWRVIP-116 not be approved by the NRC, a plant-specific reactor vessel surveillance program will be submitted to the NRC two years prior to commencement of the period of extended operation."

- Revise LRA Section B2.1.19 (page B-41), footnote 2, as follows:
  "NRC review of BWRVIP-116 is not complete. When NRC issues a final safety evaluation report for BWRVIP-116, NMPNS will address any open items and complete the SER Action Items."

- Revise LRA Section B2.1.19 (page B-42), under the "Enhancements" heading, as follows:
  "Program Elements Affected

  Revise applicable existing procedures to ensure that the procedures address the following elements:

  First paragraph of NUREG-1801 Program Description – incorporate the requirements and elements of the ISP, as documented in BWRVIP-116 and approved by NRC, or an NRC approved plant-specific program, into the Reactor Vessel Surveillance Program."
RAIs on BWRVIP Documents

RAI BWRVIP-1

The NRC staff has approved the applicable BWRVIP reports and has approved other applicable reports as required license renewal applicant action items, in accordance with 10 CFR Part 54.

Each license renewal applicant is to verify that its plant is bounded by the applicable reports. Further, the renewal applicant is to commit to programs described as necessary in the BWRVIP reports to manage the effects of aging during the period of extended operation. Applicants for license renewal will be responsible for describing any such commitments and identifying how such commitments will be controlled. Any deviations from the aging management programs within these BWRVIP reports described as necessary to manage the effects of aging during the period of extended operation and to maintain the functionality of the components or other information presented in the report, such as materials of construction, will have to be identified by the renewal applicant and evaluated on a plant-specific basis in accordance with 10 CFR 54.21(a)(3) and (c)(1).

10 CFR 54.21(d) requires that an FSAR supplement for the facility contain a summary description of the programs and activities for managing the effects of aging and the evaluation of TLAAs for the period of extended operation. Those applicants for license renewal referencing the applicable BWRVIP report shall ensure that the programs and activities specified as necessary in the applicable BWRVIP reports are summarily described in the FSAR supplement.

10 CFR 54.22 requires that each application for license renewal include any technical specification changes (and the justification for the changes) or additions necessary to manage the effects of aging during the period of extended operation as part of the renewal application. The applicable BWRVIP reports may state that there are no generic changes or additions to technical specifications associated with the report as a result of its aging management review and that the applicant will provide the justification for plant-specific changes or additions. Those applicants for license renewal referencing the applicable BWRVIP reports shall ensure that the inspection strategy described in the reports does not conflict with or result in any changes to their technical specifications. If technical specifications changes do result, then the applicant must ensure that those changes are included in its application for license renewal.

If required by the applicable BWRVIP report, the applicant referencing a particular report for license renewal should identify and evaluate any potential TLAAs issues and/or commitments to perform future inspections when inspection tooling is made available.

Based on the above stated requirements, please provide the necessary commitments, information and changes as described above for each of the following applicable BWRVIP reports, if applicable:

- BWRVIP-75
- BWRVIP-78
- BWRVIP-86
- BWRVIP-42
- Other reports applicable to license renewal for NMP1 and NMP2.

Response

NMPNS is an active participant in the BWR Vessel and Internals Project (VIP). The NMPNS BWRVIP Program controls the evaluation and implementation of each BWRVIP report when issued by the BWRVIP and, as applicable, approved by the NRC. For each of the BWRVIP reports listed in the above RAI, NMPNS provides the following summary of current and future implementation.

BWRVIP-75, "BWR Vessel and Internals Project, Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules," proposed revisions to the extent and frequencies for piping inspection contained in Generic Letter (GL) 88-01. The revisions were based on the consideration of inspection results and service experience gained by the industry since the issuance of GL 88-01, and included additional knowledge regarding the benefits of improved BWR water chemistry. The report also provided justification for the respective conditions of normal water chemistry (NWC) and hydrogen water chemistry (HWC). BWRVIP-75 was accepted for use by the NRC as described in its final safety evaluation, dated May 14, 2002.

NMP1 and NMP2 have incorporated the revised inspection frequencies of BWRVIP-75 for those welds addressed by GL 88-01. There are seven categories of welds, lettered A through G, defined by GL 88-01. Of these, four are applicable to NMP1 (Categories A, D, F and G) and three for NMP2 (Categories A, D and E). Neither plant has welds in the remaining categories. The category A welds have been incorporated within the Alternate Risk-Informed Inservice Inspection Program for both plants. NMPNS currently implements the revised scope and frequency based upon normal water chemistry but the program allows switching to the hydrogen water chemistry frequencies once the criteria have been met. NMPNS implements its commitments to GL 88-01, as modified by BWRVIP-75, for the current operating term and will continue to do so for the period of extended operation.

BWRVIP-78, "BWR Vessel and Internals Project, BWR Integrated Surveillance Program Plan," and BWRVIP-86, "BWR Vessel and Internals Project, BWR Integrated Surveillance Program Implementation Plan," describe the technical basis for the development and implementation of an integrated surveillance program (ISP) intended to support operation of all U.S. BWR reactor pressure vessels (RPV) through the completion of each facility's current operating license. By letter dated February 1, 2002, the NRC issued its safety evaluation accepting the use of these reports as an alternative to existing BWR plant-specific RPV surveillance programs. In October 2002, the BWRVIP committee issued BWRVIP-86-A, "BWR Vessel and Internals Project, Updated BWR Integrated Surveillance Program (ISP) Implementation Plan," which incorporated changes to the above two reports to incorporate revisions resulting from the regulatory review process.

By letters dated January 9, 2004 (NMP1L 1804 and NMP2L 2109), NMP1 and NMP2 requested license amendments to transfer from plant-specific RPV surveillance programs to the Integrated Surveillance Program (ISP) as documented in BWRVIP-78 and BWRVIP-86-A. These letters
also included proposed changes to the Updated Final Safety Analysis Reports (UFSARs). On November 8, 2004, the NRC issued the requested amendments. As such, NMP1 and NMP2 will implement the BWRVIP RPV ISP for the remainder of the current operating license to demonstrate compliance with the requirements of 10 CFR 50 Appendix H. For the period of extended operation, NMP1 and NMP2 intend to implement the ISP in accordance with BWRVIP-116, “BWR Vessel and Internals Project, Integrated Surveillance Program (ISP) Implementation for License Renewal,” once approved by the NRC, as stated in revised LRA Sections A1.1.32, A2.1.32, and B2.1.19.

BWRVIP-42, “BWR Vessel and Internals Project, LPCI Coupling Inspection and Evaluation Guidelines,” contains generic guidelines to BWRVIP members on inspection and flaw evaluation of low pressure coolant injection (LPCI) couplings. The implementation of the guidelines associated with this report, including the license renewal applicant action items and open item regarding inaccessible weld inspections, was addressed in NMPNS letter NMP1L 1888, dated November 19, 2004. This letter indicates that BWRVIP-42 is only applicable to NMP2, since NMP1 does not have an LPCI system, and includes the necessary changes to the LRA Sections A2.1.13 and B2.1.8.

Based upon the above responses to each of the BWRVIP reports listed in the RAI, there are no additional changes required to the LRA.

RAI-Steam Dryer

(Not used)
**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.

<table>
<thead>
<tr>
<th>REGULATORY COMMITMENT</th>
<th>DUE DATE</th>
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<tr>
<td>Using existing approved methods, NMPNS plans to implement a strategy whereby a leaking CRD stub tube penetration would be roll repaired. If, following the roll repair, this stub tube were to leak within acceptable limits, then a weld repair would be effected no later than one operating cycle following discovery of the leakage.</td>
<td>Following roll repair of a CRD stub tube penetration, if leakage exceeds acceptable limits.</td>
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</table>
| Maintenance procedures for inspection of the Orificed Fuel Support (OFS) casting will be enhanced to include a sample VT-1 inspection of the casting and an EVT-1 inspection if any evidence of impact or mishandling is identified. | NMP1: August 22, 2009  
NMP2: October 31, 2026 |
| NMPNS commits to implement the Integrated Surveillance Program (ISP) described in BWRVIP-116 (if approved by the NRC staff). When the NRC issues a final safety evaluation report (SER) for BWRVIP-116, NMPNS will address any open items and complete the SER Action Items. | Following NRC issuance of a final SER for BWRVIP-116 |
| Should BWRVIP-116 not be approved by the NRC, a plant-specific reactor vessel surveillance program will be submitted to the NRC two years prior to commencement of the period of extended operation. | NMP1: August 22, 2007  
NMP2: October 31, 2024 |