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10 CFR Part 54

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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Monticello Nuclear Generating Plant
Docket 50-263
License No. DPR-22

Response to Requests for Additional Information Regarding the Monticello Nuclear Generating Plant License Renewal Application (TAC No. MC6440)

References:
1) NMC letter to NRC, "Application for Renewed Operating License," dated March 16, 2005 (ADAMS Accession No. ML050880241)
2) NRC letter to NMC, "Request for Additional Information for the Review of the Monticello Nuclear Generating Plant License Renewal Application (TAC No. MC6440)," dated September 28, 2005 (ADAMS Accession No. ML052730175)

Pursuant to 10 CFR Part 54, the Nuclear Management Company, (NMC) LLC submitted a License Renewal Application (LRA) (Reference 1) to renew the operating license for the Monticello Nuclear Generating Plant (MNGP).

On September 28, 2005, the U.S. Nuclear Regulatory Commission (NRC) issued a Request for Additional Information (RAI) (Reference 2) regarding the LRA for the MNGP.

Enclosure 1 provides the NRC RAI followed by the NMC response.

This letter contains no new commitments or changes any previous commitments.

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

Executed on October 28, 2005.

John T. Conway

John T. Conway
Site Vice President, Monticello Nuclear Generating Plant
Nuclear Management Company, LLC

Enclosure

cc: Administrator, Region III, USNRC
Project Manager, Monticello, USNRC
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ENCLOSURE 1

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION DATED SEPTEMBER 28, 2005

A. STRUCTURES, SCOPING AND SCREENING

1. <u>NRC RAI 2.4.2-1</u>

The two component groups: (I) Concrete in air/gas (foundation, walls, slabs) and (2) Concrete in air/gas (foundation, walls, slabs, grout) have identical intended functions in Tables 2.4.2-1, 2.4.3-1, 2.4.4-1, 2.4.7-1, 2.4.8-1, 2.4.11-1, 2.4.12-1, 2.4.14-1, 2.4.15-1, 2.4.16-1, and 2.4.17-1. Explain the difference in the foundations, walls, and slabs between the two component groups. If there is no difference, explain the need for having the first component group since the second component group has already covered the first component group.

In addition to the two component groups mentioned, concrete in air/gas (walls, slabs) is another component group listed with the same intended function as the previous two component groups in Tables 2.4.3-1, 2.4.4-1, 2.4.8-1, 2.4.12-1, and 2.4.17-1. Explain the difference in the walls and slabs listed in this component group and the previous two component groups. If there is no difference, explain the need for having this component group since other component groups have already covered walls and slabs.

<u>NMC Response</u>

To explain the difference between component groups, first an explanation of how Table 2.4.x-1 was assembled is needed. Tables in Section 2.4 of the License Renewal Application (LRA) were assembled by copying the component group and intended functions for each 3.5.2-x Table line entry into the 2.4.x-1 Table. This format was consistently used throughout the LRA. Many line entry component group descriptions appear similar but there are subtle differences that are evident upon review of the 3.5.2-x Table aging management program (AMP), aging effects/mechanisms, material, etc.

For this specific question, Table 2.4.x-1 component groups, "concrete in air/gas (foundation, walls, slabs)" and "concrete in air/gas (foundation, walls, slabs, grout)" have different component group descriptions (one component group includes grout and the other did not). Review of Table 3.5.2-x reveals that the component group without grout is evaluated for the aging effect, "cracking, loss of bond, loss of material due to corrosion of embedded steel." The mechanism of corrosion of embedded steel is not applicable to grout but is applicable to reinforced concrete.

For part two of this question, the Table 2.4.x-1 component groups, "concrete in air/gas (walls, slabs)," "concrete in air/gas (foundation, walls, slabs)," and "concrete in air/gas (foundation, walls, slabs, grout)" have different component group descriptions. The same rationale discussed above is applicable here as well.

2. <u>NRC RAI 2.4.4-1</u>

Table 2.4.4-1 lists two identical component groups, "Concrete in atmosphere/weather (walls, slab)," with identical intended functions. Explain the need for listing the same component group twice.

<u>NMC Response</u>

Due to page format/spacing limitations, Table 3.5.2-4 was unable to include all the component information on the same page, and therefore it became necessary to repeat the component group, intended functions, etc. on the following page. Table 2.4.4-1 could have omitted this duplication but a decision was made not to interfere with the process used to assemble the 2.4.x-1 Table.

3. <u>NRC RAI 2.4.5-1</u>

Table 2.4.5-1 lists three identical component groups, "Non-metallic fire proofing in air/gas (...cementitious fireproofing, ...)," with identical intended functions. Explain the need for listing the same component group three times.

<u>NMC Response</u>

To explain the difference between component groups, first an explanation of how Table 2.4.x-1 was assembled is needed. Tables in Section 2.4 of the LRA were assembled by copying the component group and intended functions for each 3.5.2-x Table line entry into the 2.4.x-1 Table. This format was consistently used throughout the LRA. Many line entry component group descriptions appear similar but there are subtle differences that are evident upon review of the 3.5.2-x Table AMP, aging effects/mechanisms, material, etc.

Table 2.4.5-1 component groups, "non-metallic fire proofing in air/gas (cementitious fireproofing for coating structural steel and miscellaneous components)," "non-metallic fire proofing in air/gas (fibrous fire wraps, cementitious fireproofing (i.e., pyrocrete, etc.))," and "non-metallic fire proofing in air/gas (fibrous fire wraps, cementitious fireproofing (i.e., pyrocrete, etc.), rigid board (i.e., gypsum board, etc.))" have different component group descriptions. Review of Table 3.5.2-5 reveals that component groups were evaluated for aging effects/mechanisms that were not applicable to all components groups.

4. **NRC RAI 2.4.6-1**

After the "System Function Listing," the applicant provides reference to Sections 12.2.1.2 and 12.2.1.3 of the Updated Safety Analysis Report (USAR) for additional hangers and supports commodity group details. A review of the SAR sections indicates that the systems, structures and components (SSCs) are classified as Class I and Class II with the definitions of Class I and Class II noticeably different from Criteria 1, 2, and 3 in 10 CFR 54.4. The applicant is requested to provide a discussion of how the current licensing basis (CLB) classification has been reconciled in the system function listing in Section 2.4.6.

NMC Response

The following excerpt from Section 2.1.4.2 of the LRA and the additional comment provide clarification on how the CLB classification of SSCs was reconciled. LRA Section 2.1.4.2 entitled, "SSC Functions" states,

> "Numerous sources, including the MNGP USAR, docketed correspondence with the NRC, Maintenance Rule documents, and DBDs [Design Basis Documents] provided system and structure-level function information. Documentation of references used in this process was included for each system function as appropriate. The process used at the MNGP identified all system-level and structure-level functions. If the functions met any of the criteria specified in 10 CFR Part 54.4 (a)(1), (2), or (3), then the system or structure was in-scope for LR [License Renewal]."

Even though USAR Class I and II designations are significantly different than 10 CFR 54.4 designations, SSCs were still scoped in accordance with the criteria in 10 CFR 54.4.

5. **NRC RAI 2.4.6-2**

In Table 2.4.6-1 line item "carbon steel, low-alloy steel in air/gas," the applicant identified a number of supports/anchorages as ASME Class MC supports and some are identified as non-ASME support components. The applicant is requested to provide information regarding the classification of component supports inside the torus (some may be non-ASME), and the supports outside the torus, specifically, the classification of the support system supporting the torus. It appears that the torus support system is classified as Class MC supports, and all its components are and will be inspected by the requirements of subsection IWF. After reviewing LRA Table 3.5.2-6, it was not obvious how the applicant treated these supports. The applicant is requested to provide clarification.

NMC Response

The torus supports include torus columns, torus saddles, and torus seismic restraints (see LRA page 3-675). These supports, located on the outside of the torus, are classified as American Society of Mechanical Engineers (ASME) Class MC and will be managed by the ASME Section XI, Subsection IWF Program.

For the remaining torus system support components, an abridged list, extracted from Table 3.5.2-6, is provided below and includes the classifications and locations.

Torus system components/supports inside torus with classification:

Supports for light fixtures, Junction boxes (J-boxes) -	Non-ASME
Conduit -	Non ASME
Light fixtures and J-boxes -	Non-ASME
Supports for vent system -	ASME Class MC
Downcomer bracing -	ASME Class MC
Supports for piping, conduit and components -	Non-ASME
Supports for High Pressure Coolant Injection (HPC), - Reactor Core Isolation Cooling (RCI) System Sparger, Safety Relief Valve (SRV) T-Quencher, Emergency Core Cooling System (ECCS) Suction Strainer	Non-ASME

Torus system components/supports outside torus with classification:

Supports for lighting fixtures, J-boxes, racks, panels, cabinets -	Non-ASME
Cable tray, conduit, tube track -	Non-ASME
Light fixtures, J-boxes, racks, panels, and cabinets -	Non-ASME
Supports for cable trays, conduit, tube track, Heating Ventilation and Air Conditioning (HVAC) - ducts, instrumentation tubing and piping	Non-ASME

6. **NRC RAI 2.4.6-3**

Table 2.4.6-1 lists "Carbon steel, low-alloy steel in atmosphere/weather (bolted connections and anchorage)" as a component group and "Carbon steel, low-alloy steel in atmosphere/weather (bolted connections and support anchorage)" as another component group. The only difference between the two component

groups is that one group lists "anchorage" and the other "support anchorage." Explain the difference between "anchorage" and "support anchorage." The staff is uncertain of the nature of anchorage and support anchorage, and therefore requests the applicant to provide examples of each.

NMC Response

The line entry in LRA Table 2.4.6-1, page 2-216 containing the word, "anchorage" also could have included the word, "support" before "anchorage" in the description in order to be consistent with other similar entries. The word, "anchorage" alone however is sufficient to convey the intent. "Anchorage" and "support anchorage" refer to components used to secure (i.e. anchor) the support to the concrete surface and includes concrete anchors of various types, and associated components such as nuts and washers.

7. **NRC RAI 2.4.13-1**

It appears that the supports and components included in Code PCT-04 of the system function listing are not within the scope of license renewal. The applicant is requested to provide a summary listing of these supports and components, and a confirmation that their failure under earthquake induced loads will not affect the functioning of the safety-related SSCs.

NMC Response

System function Primary Containment (PCT) -04 referred only to non-safety related components that could not affect safety related SSCs. Components associated with this function are not in scope of license renewal. The function for non-safety related components that could affect safety related components is Primary Containment-Non-Safety Affecting Safety (PCT-NSAS), evaluated on page 2-255 of the LRA. Functions associated with component supports are further addressed in Section 2.4.6, "Hangers and Supports Commodity Group." The PCT-04 function was evaluated against the criteria of 10 CFR 54.4(a) and found not to meet any of its requirements. Consequently, the function was provided for information and completeness only, since it did not form a basis for including the primary containment structure within the scope of the Rule. This scoping methodology was consistently used through Section 2 of the LRA.

8. **NRC RAI 2.4.13-2**

The second and third line items in Table 2.4.13-1 list almost identical components subjected to the same material and environment combination, and identical intended functions. A similar redundancy is noted on the first two line items on p. 2-257. The applicant is requested to clarify these redundancies.

NMC Response

To explain the apparent redundancies between components, first an explanation of how Table 2.4.13-1 was assembled is needed. Tables in Section 2.4 of the LRA were assembled by copying the component group and intended functions for each 3.5.2-x Table line entry into the 2.4.x-1 Table. This format was consistently used throughout the LRA. Many line entry component group descriptions appear similar but there are subtle differences that are evident upon review of the 3.5.2-x Table AMP, aging effects/mechanisms, material, etc..

For this specific question, Table 2.4.13-1 component groups, "carbon steel, low alloy steel in air/gas (drywell, torus, drywell head, drywell head bolts, torus ring girder, downcomers, vent lines, vent header, bellows assemblies, ECCS suction header)" and "carbon steel, low alloy steel in air/gas (drywell, torus, drywell head, drywell head bolts, torus ring girder, downcomers, vent lines, vent header, bellows assemblies, vent header deflectors, ECCS suction header)" have different component group descriptions (one component group includes vent header deflectors and the other does not). Review of Table 3.5.2-13 reveals that the component group without vent header deflectors is managed by 10 CFR 50, Appendix J while the group with vent header deflectors is managed by the Primary Containment In-Service Inspection Program. This is because the vent header deflectors do not perform a pressure retaining function associated with an Appendix J test.

For part two of this question, Table 2.4.13-1 component groups, "carbon steel, low alloy steel in air/gas (personnel airlock, equipment hatch, [Control Rod Drive] CRD hatch, seismic restraint inspection ports)" and "carbon steel, low alloy steel in air/gas (personnel airlock, equipment hatch, CRD hatch, seismic restraint inspection ports, including locks, hinges and closure mechanisms)" have different component group descriptions. Review of Table 3.5.2-13 reveals that the component group with locks, hinges, and closure mechanisms is managed for a different aging effect, "loss of leak tightness in closed position" in accordance with NUREG-1801 line item II.B4.2-b.

9. NRC RAI 2.4.13-3

Table 2.4.13-1 lists the component group as "lubrite in air/gas," with the drywell head included as a component. In the description of drywell head, the applicant states, "The head is held in place by bolts and sealed with a double gasket arrangement." The applicant is requested to clarify where the lubrite bearings are used in the drywell head.

NMC Response

Lubrite type material is not used for the drywell head or downcomers. Table 2.4.13-1, page 2-260 included this component group, "lubrite in air/gas (drywell head and downcomers)" because tables in Section 2.4 of the LRA were assembled by copying the component group and intended functions for each Table 3.5.2-13 line entry into Table 2.4.13-1. Table 3.5.2-13 included this entry to demonstrate that NUREG-1801 line item II.B1.1.1-e was evaluated for applicability. The evaluation provided in Table 3.5.2-13 stated that, "The drywell head and downcomer pipes are carbon steel material. Graphite plate material is not used for these components and therefore the aging effect is not applicable" (see LRA note 556). Therefore the description of the drywell head in the LRA, page 2-251 is consistent with note 556 in Table 3.5.2-13.

10. NRC RAI 2.4.17-1

Table 2.4.17-1 lists "Carbon steel, low-alloy steel in air/gas (fire rated doors)" as a component group with the intended function being fire barrier, and "Carbon steel, low-alloy steel in air/gas (... doors,...)" as another component group, with one of the intended functions also being fire barrier. Explain the difference in the doors between the two component groups.

NMC Response

Table 2.4.17-1 component group, "carbon steel, low alloy steel in air/gas (fire rated doors)" refers to doors that provide a fire barrier intended function and are managed for aging by the Fire Protection Program. Table 2.4.17-1 component group, "carbon steel, low alloy steel in air/gas (structural steel, steel embeds, doors, etc.)" refers to those doors that were assigned a fire barrier function as discussed above but also perform at least one other intended function such as high energy line break (HELB) barrier and/or flood barrier. Consequently doors with a fire barrier intended function that also perform additional functions are managed by the Structures Monitoring Program in addition to the Fire Protection Program in accordance with NUREG-1801.

B. Structures, Aging Management

1. NRC RAI 3.5.2-1

Concrete in below grade (Foundation, Walls) is listed as requiring no AMP in Tables 3.5.2-2, 3.5.2-3, 3.5.2-4, 3.5.2-6, 3.5.2-7, 3.5.2-9, 3.5.2-11, 3.5.2-12, 3.5.2-13, 3.5.2-15, 3.5.2-16, and 3.5.2-17. The applicant states that an AMP is not required to manage aging because, as described in Note 501, plant documents confirm that the concrete had an air content between 3 percent and 6

percent, and inspections performed on concrete in accessible areas did not exhibit degradation related to freeze-thaw. It is the staff's position that the existing concrete also have a water-to-cement ratio of 0.35 - 0.45 to ensure there is no aging degradation related to freeze-thaw. The applicant is requested to verify the water-to-cement ratio is 0.35 - 0.45, or provide an appropriate AMP for this component group.

The same question applies to Concrete in below grade (Foundations, Walls, Lean Concrete) listed in Table 3.5.2-8.

<u>NMC Response</u>

Note: This response applies to Tables 3.5.2-2, 3.5.2-3, 3.5.2-4, 3.5.2-6, 3.5.2-7, 3.5.2-8, 3.5.2-9, 3.5.2-10, 3.5.2-11, 3.5.2-12, 3.5.2-14, 3.5.2-15, 3.5.2-16, 3.5.2-17 and 3.5.2-18 but is not applicable to Table 3.5.2-13 (i.e. not applicable since there is no below grade concrete inside drywell).

The criteria used to evaluate concrete located below grade were consistent with NRC Staff final position issued for Interim Staff Guidance (ISG)-03 (see LRA Section 2.1.4.3). ISG-03 provided the following criteria for the aging effects loss of material (spalling, scaling) and cracking due to freeze-thaw for concrete in inaccessible areas (i.e. exterior locations below grade and foundations).

> "Inaccessible Areas:
>
> Evaluation is needed for plants that are located in moderate to severe weathering conditions (weathering index>100 day-inch/yr) (NUREG-1557). Documented evidence to confirm that the in-place concrete had the air content between 3 percent to 6 percent and the subsequent inspections performed did not exhibit degradations related to freeze-thaw should be considered a part of the evaluation."

LRA Note 501 and Table 1, item 3.5.1-20, and further evaluation for 3.5.2.2.2.1 included the following statement.

> "MNGP is located in a severe weathering region according to Figure 1 of ASTM C33-90, and therefore freeze-thaw evaluation is required. Plant documents confirm that the concrete had an air content between 3 percent and 6 percent, and subsequent inspections performed on concrete in accessible areas did not exhibit degradation related to freeze-thaw. This evaluation satisfies NUREG-1801 and ISG-03 condition requirements for concrete in inaccessible areas, and therefore loss of material and cracking due to freeze-thaw do not require aging management."

All criteria of ISG-03 have been satisfied including an aging effect evaluation due to severe weather location, documentation confirming that the concrete had an air content between 3 percent and 6 percent, and plant inspection findings that did not exhibit degradation related to freeze-thaw.

Additionally, ISG-03 was one of five ISG guidance documents issued for implementation by the NRC prior to the LRA submittal. Under this guidance, the Staff removed the water-to-cement ratio and replaced it with the statement, "subsequent inspections performed did not exhibit degradations related to freeze-thaw". This substitution is consistent with ACI 318, Durability Requirements that state, "Since it is difficult to accurately determine the water-cementitious materials ratio of concrete during production, the f'c specified should be reasonably consistent with the water-cementitious materials ratio required for durability. Selection of an f'c that is consistent with the water-cementitious ratio selected for durability will help ensure that the required water-cementitious materials ratio is actually obtained in the field." ACI 318 also states that the quality and production of concrete must be considered. At MNGP, all in place concrete met or exceeded the design required strength (f'c). Concrete inspections continue to show no evidence of degradation due to freeze-thaw. Plant concrete design specifications (Specification for Purchase of Off-Site Concrete for the MNGP, Specification for Forming, Placing, Finishing and Curing of Concrete, and Specification for Materials Testing Services, etc.) include requirements that satisfy ACI 318 standards for materials, durability, concrete quality, mixing, and placing. Plant documents confirm that the concrete was constructed in accordance with the recommendations in ACI 201.2R for durability and therefore able to resist freeze-thaw and other age related degradation. Materials used in the concrete mix design conformed to ASTM specifications (C-94, C-150, etc.) that ensured consistent, proportional, non-porous concrete of quality materials. Aggregates conformed to the requirements of ASTM C-33 and were accepted based on ASTM C-295 (petrographic) C-289 (reactivity) and other tests. Mixing and delivering of concrete was in accordance with ACI standards for hot and cold weather conditions (ACI 305, ACI 306) and appropriate air entrainment, adequate curing, and special attention to construction practices were maintained with reference to ASTM C-260, C-494 and C-618. Utilizing industry construction standards ensured good workmanship and quality control practices (i.e., the requirements of ACI 304, 308, 309, ASTM C-94, etc.).

Compliance with the above industry code requirements and guidelines ensures that freeze-thaw is not significant as proven by the absence of freeze-thaw degradation.

2. NRC RAI 3.5.2-2

Concrete in below grade (Foundation, Walls) is listed as requiring no AMP in Tables 3.5.2-4, 3.5.2-6, 3.5.2-7, 3.5.2-9, 3.5.2-11, 3.5.2-12, 3.5.2-13, 3.5.2-15, 3.5.2-16, and 3.5.2-17. The applicant states that an AMP is not required to manage aging because, as described in Note 506: (1) the plant initial licensing basis did not include a program to monitor settlement, (2) no significant settlement has been observed, and (3) de-watering systems are not used. The applicant's claim for not requiring an AMP is inconsistent with ISG-03, which requires a "Structural Monitoring Program" based on the requirement of 10 CFR 50.65 (Maintenance Rule) for accessible areas and a "plant-specific program for inaccessible areas." Therefore, the applicant is requested to provide an appropriate AMP for this component group.

The same question applies to Concrete in below grade (Foundation, Walls, Lean Concrete) listed in Table 3.5.2-8.

NMC Response

Note: This response applies to Tables 3.5.2-3, 3.5.2-4, 3.5.2-6, 3.5.2-7, 3.5.2-8, 3.5.2-9, 3.5.2-10, 3.5.2-11, 3.5.2-12, 3.5.2-14, 3.5.2-15, 3.5.2-16, 3.5.2-17 and 3.5.2-18 but is not applicable to Table 3.5.2-13 (i.e. not applicable since there is no below grade concrete inside drywell).

ISG-03 did not include an evaluation for the aging effects cracks, distortion, and increase in component stress level due to settlement and therefore NUREG-1801 (2001) was used to evaluate the plant specific applicability of this aging effect. Settlement is a condition that directly affects the concrete foundation components (see NUREG-1557 page B-154 and Electric Power Research Institute (EPRI) 103842 page 4-88), and thus applicable to inaccessible concrete. NUREG-1801 (2001) states that,

> "The initial Licensing Basis for some plants included a program to monitor settlement. If no settlement was evident during the first decade or so, the NRC may have given the licensee approval to discontinue the program. However, if a de-watering system is relied upon for control of settlement, then the licensee is to ensure proper functioning of the de-watering system through the period of extended operation."

LRA Note 506, Table 3.5.1 item number 3.5.1-25, and further evaluation 3.5.2.2.1.2 included the following statement.

> "The plant initial Licensing Basis did not include a program to monitor settlement. No significant settlement has been observed on any major structure and de-watering systems are not used. This satisfies NUREG-1801

condition requirements on concrete settlement, and therefore cracks, distortion, and increase in component stress levels due to settlement do not require aging management."

The NUREG-1801 criteria used to evaluate concrete for settlement consists of an initial Licensing Basis program to monitoring settlement, evident of settlement during the first decade, and a de-watering system relied upon for settlement control. All are not applicable to the MNGP. The initial licensing basis did not include a settlement monitoring program, no significant settlement has ever been observed on any major structure, and de-watering systems were not used (see discussion on the Diesel Fuel Oil Transfer House that follows).

Additionally, the MNGP USAR, Section 12.2.1.11 includes a discussion on foundation design and construction. The USAR provides conclusions on foundation designs based on soil bearing values. Major building structures were constructed on bedrock, compacted granular fill, or stiff clay. No unusual or unforeseen foundation construction problems were encountered. A survey traverse of points on the buildings was established to monitor foundation settlement. This survey determined that settlement was uniform and within the predicted values. The reactor building, the turbine building and other structures are supported on mat foundations. The stack and control and cable spreading building are also on mat foundations, and the emergency diesel generator building is constructed on a continuous footing foundation. The emergency filtration train building rests on a combination of mat and caisson foundations.

The MNGP review is consistent with NUREG-1801 condition requirements for settlement, similar to the review performed in the Dresden and Quad Cities Safety Evaluation Report (SER), NUREG-1796. NUREG-1796 stated that, "no aging management is required" and the Staff responded with, "The Staff finds the applicant's explanation to be acceptable because there has been no requirement to monitor settlement as part of the licensing basis for all four units, and there are no de-watering systems in place."

Although not managed for cracks, distortion, and increase in component stress level due to settlement (except for the Diesel Fuel Oil Transfer House, see below), all accessible concrete is managed for aging effects including cracks, loss of material, etc. Additionally, whenever an inaccessible area is excavated, exposed or modified, an inspection is performed.

As stated above, settlement has not been observed at any major structures. The Diesel Fuel Oil Transfer House, a small structure north of the Emergency Diesel Generator Building has experienced settlement. The structure is rectangular with external dimensions of 11'-6" (N-S) x 14' (E-W) x 13'-6" high. Walls are 1'-6" thick. It is a moderate weight structure exerting a mean bearing pressure of about 1,100 lb. / $ft.^2$ on the underlying foundation material. The foundation material is compacted granular backfill underlain by stiff clay lenses and

sandstone bedrock, and should not be susceptible to settlement under the load imposed. However the Diesel Fuel Oil Transfer House has undergone significant differential settlement. Based on plant records and settlement data, settlement of the Diesel Fuel Oil Transfer House occurred rather rapidly following construction and was probably due to washout after a rainstorm and was long ago effectively complete. Settlement data recorded annually since 1992 continues to show no significant settlement of the structure. A Nonconformance Report root cause analysis concluded there was insufficient foundation support to prevent settling. Measurements taken in 1976, 1979 and 1991 show settlement has continued over the years since construction but at an extremely low rate. Surveys determined that the entire building has settled about ¾" to 1" on the east side and about 5 ¼" to 5 ½" on the west side. No evidence of cracks or distortion has been observed by the Structures Monitoring Program inspections performed in 1996 and 2002. The Diesel Fuel Oil Transfer House is monitored for the aging effects of cracks, distortion, and increase in component stress level due to settlement now on an annual basis as part of the Structures Monitoring Program and will be managed throughout the period of extended operation.

In conclusion, cracks, distortion, and increase in component stress level due to settlement is not an applicable aging mechanism for any in scope structure with the exception of the Diesel Fuel Oil Transfer House which is managed for aging effects due to settlement.

3. <u>NRC RAI 3.5.2-3</u>

Concrete in below grade (Diesel Fuel Oil Storage Tank Deadmen) is listed as requiring no AMP in Table 3.5.2-6. The applicant states that an AMP is not required to manage aging because, as described in Note 552, "NUREG-1801 lists inside or outside containment as the environment. Consider that this environment includes atmosphere/weather and below grade." The applicant's claim for not requiring an AMP is inconsistent with ISG-03, which requires a "Structural Monitoring Program" based on the requirement of 10 CFR 50.65 (Maintenance Rule) for accessible areas and a "plant-specific program for inaccessible areas." Therefore, the applicant is requested to provide an appropriate AMP for this component group.

<u>NMC Response</u>

The buried Diesel Fuel Oil Storage Tank is anchored to a concrete foundation. To account for this condition, NUREG-1801 line item III.B4.3-a, "building concrete at locations of expansion and grouted anchors" was used. Although probably not the best usage of line III.B4.3-a, it was chosen to address this unique condition. Note 552 was provided to clarify the different environment used than that specified in NUREG-1801 (i.e. below grade rather than the NUREG-1801 environment, inside or outside containment). The LRA AMP also differed from that specified in NUREG-1801. Generic Note "I" was used to

describe these differences since it was considered the "best" note available to describe this unique condition. The aging effect, "reduction in concrete anchor capacity due to local degradation" for the inaccessible location was evaluated by considering all possible concrete degradation mechanisms including freeze-thaw (III.A3.1-a), leaching of calcium hydroxide (III.A3.1-b), reaction with aggregate (III.A3.1-c), corrosion of embedded steel (III.A3.1-e), aggressive chemical attack (III.A3.1-g), settlement (III.A3.1-h), and erosion of porous concrete sub-foundations (III.A3.1-h). The evaluation concluded that these mechanisms did not require aging management (see page 3-686 of the LRA). Service induced cracking was also not applicable since vibration/movement of the tank is not expected and the fact that the tank is surrounded by soil/fill material. Therefore reduction in concrete anchor capacity due to local degradation was determined insignificant and aging management not required.

4. **NRC RAI 3.5.2-4**

Concrete in below grade (Pedestal) is listed as requiring no AMP in Table 3.5.2-10. The applicant states that an AMP is not required to manage aging because, as described in Notes 501 and 506: (1) the plant initial licensing basis did not include a program to monitor settlement, (2) no significant settlement has been observed, (3) de-watering systems are not used, and (4) plant documents confirm that the concrete had an air content between 3 percent and 6 percent and inspection performed on concrete in accessible areas did not exhibit degradation related to freeze-thaw. It is the staff's position that the existing concrete also have a water-to-cement ratio of 0.35 - 0.45 to ensure there is no aging degradation related to freeze-thaw. The applicant is requested to verify the water-to-cement ratio is 0.35 - 0.45.

The applicant's claim for not requiring an AMP is inconsistent with ISG-03, which requires a "Structural Monitoring Program" based on the requirement of 10 CFR 50.65 (Maintenance Rule) for accessible areas and a "plant-specific program for inaccessible areas." Therefore, the applicant is requested to provide an appropriate AMP for this component group.

The same question applies to Concrete in below grade (Foundation, Walls, Slabs, Grout) listed in Table 3.5.2-18.

NMC Response

For response to RAI 3.5.2-4 (i.e., questions on freeze-thaw and settlement), see response to RAI 3.5.2-1 and RAI 3.5.2-2.

5. NRC RAI 3.5.2-5

In describing the intended functions of the three items in Table 3.5.2-13, related to concrete in air/gas, the applicant states one of the intended functions is "non-safety support." The applicant is requested to clarify this characterization in terms of the CLB safety classification as well as in terms of 10 CFR 54.4 definitions, and provide examples of how the components are providing non-safety support.

NMC Response

The primary containment in scope concrete components subject to an AMR with the intended function, "non-safety support" were identified in Table 2.4.13-1 with AMR results provided in Table 3.5.2-13. Table 2.1-1 of the LRA included the definition as,

> "Provide structural support to non-safety related components whose failure could prevent satisfactory accomplishment of any of the required safety related functions."

This component-level intended function was the specific function of the component that supported system-level functions that formed the basis for including the primary containment structure within the scope of license renewal. The scoping methodology utilized by the NMC for the MNGP was consistent with the guidance provided by the NRC in NUREG-1800 and by the industry in Nuclear Energy Institute (NEI) 95-10.

In terms of the MNGP CLB, this function was characterized in a license renewal technical report as,

> "The MNGP CLB includes a number of topics that identify NSR SSCs credited for preventive or mitigative functions in support of safe shutdown for special events (e.g., external floods) or whose failure could prevent satisfactory accomplishment of a Scoping Criterion 1 function (e.g., Seismic II/I considerations). Based on a review of the CLB, those topics that meet Scoping Criterion 2 are, High Energy Line Break (HELB)...Flooding Events...Missile Hazards...Overhead Handling Systems...Seismic Interaction."

In terms of 10 CFR 54.4, this function was characterized in a technical report for license renewal as,

> "[Non-safety related] NSR SSCs directly connected to Scoping Criterion 1 SSCs: The in-scope boundary for license renewal extends into the NSR portion of the piping and supports up to and including the first equivalent anchor beyond the safety/nonsafety interface. For Monticello, the first equivalent anchor is that point beyond which failure of the piping system will

not prevent the satisfactory accomplishment of the Scoping Criterion 1 function of the connected SSCs.

NSR structures attached to, or next to, Scoping Criterion 1 structures are in scope for license renewal if their failure could prevent a Scoping Criterion 1 SSC from performing its intended function.

NSR SSCs that are not directly connected to Scoping Criterion 1 SSCs: The NSR SSCs may be in-scope if their failure could prevent the performance of a Scoping Criterion 1 function."

The license renewal technical report included further discussion on NSR SSCs that are not directly connected to Scoping Criterion 1 SSCs with detailed information on the identification process of spatial interactions, a process to determine which NSR conduits, trays, junction boxes, and lighting fixtures to consider in scope for license renewal, and the process for determining in scope NSR HVAC ducts and supports.

An example of how concrete components provide non-safety support would be an attachment to the concrete of NSR light fixtures or NSR HVAC duct routed near/above scoping Criteria 1 components.

6. <u>NRC RAI 3.5.2-6</u>

In Table 3.5.2-13, under the component type "concrete in air/gas," a number of structural components (e.g., drywell equipment foundation, bioshield wall, RPV pedestal) are listed. Section 3.5.2.2.1.3 and Note 508 describe the elevated temperature situation around the reactor vessel, and justify the existence of the elevated temperatures in these areas, based on the estimated temperatures in the MNGP drywell. The applicant is requested to provide the following information related to this component type:

a. Please provide a summary description of the cooling system installed to control the temperatures inside the drywell.

b. Please provide the operating experience related to the effectiveness of the cooling system. Are the shield wall temperatures, or any other parameter monitored that would detect the malfunctioning of the cooling system?

c. Based on the discussion of the elevated temperature condition in and around the bioshield wall in Section 3.5.2.2.1.3, the staff agrees that the concrete properties will not be significantly affected, if the actual temperatures around the shield wall remain within the estimated limits. However, additional shrinkage and loss of moisture due to radiation could

degrade the concrete on a long term basis. In this context, please provide a summary of the results of the last two inspections performed for: (1) the bioshield wall, (2) RPV pedestal, (3) anchorages of seismic stabilizer frame, and (4) masonry walls (if any) inside the drywell.

NMC Response

a. Drywell fan coolers are used to control temperatures inside the drywell. USAR Section 5.2 states that, "The primary containment ventilating and cooling system consists of four air coolers which cool the atmosphere to below a 135°F bulk average drywell temperature during normal plant operation. The drywell atmosphere is circulated through the drywell and the air coolers by fans, and the reactor building closed cooling water system is employed to remove heat from the air coolers."

b. Plant daily operating data confirms that the general area maximum normal operating temperature inside drywell is below the NUREG-1801 limit of 150°F. Therefore the drywell fan coolers have proven their effectiveness in controlling the drywell air temperature. Plant calculations determined that the biological shield wall pipe penetrations were sufficiently designed in size, insulation characteristics, and air gap to limit the local area maximum normal operating temperatures to 179° F, less than the NUREG-1801 threshold local area temperature of 200°F.

c. Results of the 1996, 1998 and 2002 Periodic Structural Inspection Reports found all concrete at the RPV Pedestal to be acceptable with no deficiencies observed. The bioshield wall is complete encased in steel and therefore cannot be inspected. Drywell structural steel components were found acceptable with no deficiencies observed including stabilizer attachment welds to the plated bioshield wall. USAR Section 12 states that the primary function of the bioshield wall is, "to protect equipment inside the drywell against radiation and thermal effects. The structure is capable of transmitting loads due to seismic and jet forces acting on it. The biological shield is composed of two steel cylinders interconnected with 27 WF (177 lb/ft) columns and is filled with concrete. Because of the radiation and temperature effects on the concrete only the lower 12 feet of concrete, up to the 959 foot elevation, has been designed as structural concrete capable of resisting forces and shears. Above the 959 foot elevation the two steel cylinders and 27 WF columns are structurally adequate and the concrete fill has not been considered as adding to the support."

7. NRC RAI 3.5.2-7

The applicant identifies two line items in Table 3.5.2-06 related to carbon steel and low-alloy steel embedded in concrete as not requiring aging management. Note 549 states: "Requirements specified in NUREG-1801 for concrete quality,

inspections and housekeeping are satisfied for steel elements in inaccessible areas." Based on the industry-wide experience related to corrosion of drywell shell in sand pocket region, the applicant is requested to provide more information regarding the inspections and housekeeping, and why these activities should not be part of an existing AMP. The purpose of including the embedded items in an existing program is to look for evidence of environment change (e.g., sand drains not working properly) in accessible areas that would indicate potential degradation in inaccessible areas.

The same question applies to carbon steel and low-alloy steel embedded in concrete listed in Table 3.5.2-13.

<u>NMC Response</u>

Components in Table 3.5.2-6 refer to the embedded anchorage of the drywell support skirt, the embedded components of the female stabilizers, and embedded conduit. The evaluation of the embedded portions of the drywell shell and drywell support skirt is provided in Table 3.5.2-13. See the following discussion.

NUREG-1801, Chapter II.B, line item II.B1.1.1-a documents four (4) condition requirements. If all four requirements are satisfied, "loss of material due to corrosion is not significant." NUREG-1801 condition requirements are provided below.

NUREG-1801 condition requirements for line II.B1.1.1-a:

1. Concrete meeting the requirements of ACI 318 or 349 and the guidance of 201.2R was used for the containment concrete in contact with the embedded containment shell or liner.
2. The concrete is monitored to ensure that it is free of penetrating cracks that provide a path for water seepage to the surface of the containment shell or liner.
3. The moisture barrier, at the junction where the shell or liner becomes embedded, is subject to aging management activities in accordance with IWE requirements.
4. Borated water spills and water ponding on the containment concrete floor are not common and when detected are cleaned up in a timely manner.

Plant Evaluation:

1. Concrete meeting the following requirements was used for the containment concrete in contact with the embedded containment shell and skirt. The MNGP concrete design specifications for Purchase of Off-Site Concrete for the MNGP, Specification for Forming, Placing, Finishing and Curing of Concrete, Specification for Materials Testing Services and others specifications include requirements that satisfy ACI 318 standards for materials, durability, concrete quality, mixing and placing. Plant documents confirm that the concrete was constructed in accordance with the recommendations in ACI 201.2R for durability and therefore able to resist weathering action, chemical attack, abrasion, leaching of calcium hydroxide, corrosion of reinforcement, and chemical reactions of aggregates. Materials used in the concrete mix design conformed to ASTM specifications (C-94, C-150, etc.) that ensure consistent, proportional, non-porous concrete of quality materials. Aggregates conformed to the requirements of ASTM C-33 and were accepted based on ASTM C-295 (petrographic) C-289 (reactivity) and other tests. Mixing and delivering of concrete was in accordance with ACI standards for hot and cold weather conditions (ACI 305, ACI 306) and appropriate air entrainment, adequate curing, and special attention to construction practices were maintained (ASTM C-260, C-494 and C-618). MNGP construction specifications ensure good workmanship and quality control practices (ACI 304, 308, 309, ASTM C-94, etc.).

2. For accessible concrete inside drywell, the Structures Monitoring Program inspects for cracking adjacent to the moisture barrier, at the concrete floor, and RPV Pedestal. Inspections ensure that the concrete is free of penetrating cracks that provide a path for water seepage to the containment shell. The bioshield wall is completely encased in steel and therefore inaccessible for inspection.

   AMPs will be used to manage the drywell to reactor building refueling seal bellows assembly located between the drywell outer shell and the reactor building concrete (See Table 3.5.2-15). By managing this assembly for aging degradation and water leakage (during refueling activities), any water seepage past the assembly to the drywell shell, sand pocket, embedded shell, and embedded skirt and will be prevented, or detected and corrected. Therefore, loss of material due to corrosion for these components will be insignificant. Aging managed for the drywell to reactor building refueling seal bellows assembly will be provided by the Primary Containment In-Service Inspection Program and the Structures Monitoring Program. These Programs ensure that degradation of the assembly or any water leakage past the assembly will be detected and corrected before loss of intended function. These programs ensure that,

- the drywell air gap drain outlets and sand pocket drain outlets are not obstructed prior to refueling,
- the drain lines that are incorporated into the refueling bellows assembly are monitored for leaks,
- the drywell air gap drain outlets and drywell sand pocket drain outlets are inspected for signs of leakage during refueling, and
- aging effects including loss of material are detected.

The Plant Chemistry Program is also used to manage the assembly by ensuring that the water chemistry remains within design parameters.

MNGP operating history showed no evidence of refueling seal leakage, no water observed in air gap during construction, and no water used to extinguish a fire in the air gap or for any other reason. Plant engineering and maintenance personnel confirmed the absence of leakage at the drywell air gap drains, and the sand pocket drains. Plant specific operating history has proven that inspection and monitoring activities adequately manage aging effects to ensure no loss of intended function.

3. The Primary Containment In-Service Inspection Program manages aging effects associated with the moisture barrier. See LRA Table 3.5.2-13, line item II.B4.3-a.

4. Borated water inside drywell is not a concern for BWR plants. (The presence of boron in the drywell is not applicable since the only system to utilize a boron solution is the Standby Liquid Control (SLC) System. This solution of sodium pentaborate resides in the SLC Tank (T-200) which is external to the drywell. When this standby system is tested, it is flushed with demineralized water after completion of testing and prior to being returned to its standby mode.)

In summary, all four requirements are met and therefore loss of material due to corrosion is not significant.

8. <u>NRC RAI 3.5.2-8</u>

Recent experience with torus cracking at Fitzpatrick indicates high-pressure coolant injection (HPCI) discharge configuration in the torus as one of the reasons for the cracking. The applicant is requested to provide information regarding the HPCI configuration at MNGP that could affect torus integrity during the period of extended operation.

NMC Response

The HPCI Turbine Exhaust discharge pipe configuration at MNGP is significantly different than that at Fitzpatrick.

- The HPCI Turbine Exhaust at Fitzpatrick is located near the torus ring girder (end bay) while the MNGP HPCI Turbine Exhaust is located approximately mid-bay between torus ring girders.

- The Fitzpatrick HPCI exhaust pipe incorporates a 90 degree elbow, is relatively short with no holes and discharges vertically in the vicinity of the torus ring girder and stiffener plates that are part of the torus external column support system. The Fitzpatrick torus crack was observed adjacent to these stiffener plates. The MNGP exhaust pipe is considerably longer with a submerged sparger that consist of a pipe, capped at one end, with a large number of holes in the lower portion which increases the area available for steam condensation compared to a straight pipe discharge. The MNGP pipe is not a vertical run but instead incorporates a 60 degree elbow.

Based on the above comparison, the HPCI Turbine Exhaust configuration and location at MNGP is unlike that at Fitzpatrick. The MNGP pipe configuration and location are believed to be less susceptible to cracking and therefore less likely to affect torus integrity during the period of extended operation.

9. NRC RAI 3.5.2-9

Three line items in Table 3.5.2-6 indicate that lubrite plates have been used at several locations in hangers and supports. It is the staff's position that an inspection of the accessible portion of the bearing is needed to ensure proper functioning during postulated environmental conditions. Therefore, the applicant is requested to incorporate an examination of the accessible portion of the lubrite bearings in an appropriate AMP.

NMC Response

Industry guidance provided in EPRI-1002950, Aging Effects for Structures and Structural Components, Revision 1 dated August 2003 for Lubrite or similar material states that, "An extensive search of industry operating experience did not identify any instances of Lubrite plate degradation or failure to perform its intended function." Additionally EPRI states that, "Lubrite material resists deformation, has a low coefficient of friction, resists softening at elevated temperatures, absorbs grit and abrasive particles, is not susceptible to corrosion, withstands high intensities of radiation, and will not score or mar. Lubrite products are solid, permanent, completely self lubricating, and require no

maintenance for the design life of the product. The Lubrite lubricants used in nuclear applications are designed specifically for the environments to which they are exposed."

Review of the MNGP pipe support and MC component support drawings where Lubrite or similar material was used reveals that the sliding surfaces are sandwiched between plates and therefore inaccessible for inspection. Plant specific operating experience found no evidence of age related degradation for Lubrite or similar material during support overhaul activities, and no pipe or pipe support failures attributed to the inability of sliding surface material to function as designed.

In accordance with EPRI-1002950, aging effects for Lubrite or similar material are not significant and although no aging management is required, supports in LRA Table 3.5.2-6 which incorporate the use of Lubrite or similar materials are inspected by the ASME Section XI, Subsection IWF Program.

10. NRC RAI 3.5.2-10

Recent experience concerning breakage of T-Quencher support bolts at Hatch indicates that the Plant Chemistry Program (PCP) that controls the chemistry of treated water may not be adequate for managing the aging of submerged support components. The applicant is requested to discuss the adequacy of PCP, by itself, to manage the aging degradation of the submerged supports. This RAI is applicable to all line items in Table 3.5.2-13, where PCP has been identified as the only AMP.

NMC Response

All line items Table 3.5.2-13 in a treated water environment (submerged) are managed by the Plant Chemistry Program in addition to the MNGP Primary Containment In-Service Inspection Program, and in many cases are also managed by a third program, 10 CFR 50, Appendix J. Therefore this RAI is not applicable to MNGP.

Additionally, the MNGP Primary Containment In-Service Inspection Program includes activities that perform periodic visual inspections by divers (when the torus is not drained) and by engineers (when drained) for submerged components including their support members, bolted connections, and welds. Components inspected include such items as T-quenchers, SRV piping and supports, ECCS strainers, vent header supports, catwalk supports, and other submerged piping and supports not included in the IWE, VT-3 inspection. Inspections are conducted periodically at intervals not to exceed five (5) years. The MNGP Primary Containment In-Service Inspection Program manages aging effects for visible degradation such as deformation, cracks, corrosion, loose bolts, etc.

11. NRC RAI 3.5.2-11

For a number of items in Table 3.5.2-13, 10 CFR 50, Appendix J has been identified as the AMP to manage aging. Option B of Appendix J would permit the applicant to conduct Type B leakage rate tests of penetrations at 10 year intervals. The applicant is requested to discuss the plant-specific process (e.g., test frequency and operating experience) that is credited to manage degradation and leak tightness of the pressure boundary penetrations, including vent bellows.

NMC Response

Interval of Appendix J, Type B Test:

Type B tests, which are conducted at performance based intervals not exceeding 120 months (plus an extension of 15 months if required by the refueling schedule), are performed to assess leakage through individual penetration isolation barriers other than valves. Per NEI 94-01, air lock tests must be performed at intervals not exceeding 30 months and at other times as determined by air lock use. Also, bolted access-way cover seals are always tested following end of outage closures of the access-ways. The default interval between Type B tests is 30 months. The interval may be extended to 60 months following two (2) consecutive tests with results that meet performance leakage acceptance criteria and to 120 months following three (3) consecutive tests that meet these criteria. The interval reverts to the default interval following a test failure.

The MNGP operating history on bellow leakage/replacement is limited to one, 2-ply bellow. Leakage was identified during LLRT testing and not a result of cracks observed during a visual examination. Leakage was identified at the outer most bellows from a small failure underneath the outer collar of the expansion joint and consequently the bellow was replaced. No cracks in the weld metal were identified. Industry operating history has also identified cracked bellows but no cracks in the weld metal.

12. NRC RAI 3.5.2-12

Three line items in Table 3.5.2-13 indicate that lubrite plates have been used at several locations in the MNGP primary containment. In Note 556, the applicant states that graphite plate material is not used for drywell head and downcomers. The applicant is requested to clarify how lubrite plates are associated with drywell head and downcomers. In Note 559, the applicant states that beam seats in the drywell consist of carbon steel plate over a bronze plate lubricated with graphite packed into trepanned depressions. The steel plate covers the graphite packing and protects it from particulate contaminants. The staff believes that, if the lubrite bearing, in general, is qualified for use in the sustained

temperatures and radiation existing in the drywell, the inspection should consist of the examination of the accessible part of the bearing to ensure proper functioning during postulated environmental conditions. In view of this discussion, the applicant is requested to incorporate an examination of the accessible portion of the lubrite bearings in an appropriate AMP.

NMC Response

Lubrite type material is not used for the drywell head or downcomers. Table 3.5.2-13 line entry was necessary to demonstrate that aging effects in NUREG-1801 line item II.B1.1.1-e were evaluated. The evaluation provided in Table 3.5.2-13, note 556 stated that, "The drywell head and downcomer pipes are carbon steel material. Graphite plate material is not used for these components and therefore the aging effect is not applicable."

NUREG-1801 line item II.B1.1.1-e also includes fretting due to mechanical wear of carbon steel. EPRI-1002950, Structural Tools, evaluated fretting as loss of material occurring as a result of the relative motion between two components. EPRI concluded that thermal cycling during plant heat-up, cool down (refueling operations) and normal operation have insufficient relative motion and frequency to result in significant wear. EPRI concluded that wear of carbon steel is a design issue that incorporates sliding surfaces into the design. In accordance with the EPRI evaluation, the drywell head and downcomers do not require aging management for fretting. Note that the drywell head and downcomers are managed for loss of material due to corrosion consistent with NUREG-1801 line II.B1.1.1-a. See Table 3.5.2-13 for this evaluation.

Lubrite material incorporated into radial beam seat connections is used inside drywell to connect platform steel to the drywell shell. The beam seat Lubrite plate is sandwiched between larger steel plates which overhang it and therefore inspection is not possible. These beam seats are well over 20 feet from the reactor pressure vessel and outside the bioshield wall. This is the only application of Lubrite type material in use inside the drywell where the higher gamma radiation levels are expected. According to EPRI-1002950 the only aging effect for Lubrite is change in material properties, and only if Lubrite is exposed to at least 10^4 Rads. Radiation levels outside the bioshield wall at the perimeter of the drywell would be significantly less that the EPRI threshold limit. All other Lubrite type material applications are outside the drywell where radiation levels would typically be significantly less than inside the drywell.

C. Reactor Vessel Surveillance

1. NRC RAI B2.1.29-1

The scope of the Reactor Vessel Surveillance Program indicates that the Monticello Nuclear Generating Plant (MNGP) Reactor Vessel Surveillance Program monitors radiation embrittlement of the reactor pressure vessel (RPV). MNGP is part of the Boiling Water Reactor Vessel and Internals Project (BWRVIP) integrated surveillance program (ISP) that uses data from boiling water reactor (BWR) member surveillance programs to select the "best" representative material to monitor radiation embrittlement for a particular plant. The BWRVIP ISP monitors capsule test results from various member plants. MNGP plans to use the BWRVIP ISP during the period of extended operation by implementing the requirements of BWRVIP-116. BWRVIP-116 indicates the "best" representative plate material for the MNGP RPV is the plate (C2220) in the MNGP RPV surveillance capsules. Table 2-3 in BWRVIP-116 indicates the last capsule to be removed from MNGP RPV will be at 40 effective full power years (EFPY) where the estimated neutron fluence for the capsule will be 1.98×10^{18} n/cm^2 (E>1.0 MeV).

a. Provide the lead or lag factors (the ratio of the neutron fluence for the surveillance capsule to the peak neutron fluence for the reactor vessel at the 1/4 thickness depth {1/4T}) for all the surveillance capsule locations in the Monticello reactor vessel. Identify the lead or lag factor for the surveillance capsule that is planned to be removed at 40 EFPY.

b. Explain why the neutron fluence for the surveillance capsules lags the peak neutron fluence for the MNGP RPV.

c. Can the surveillance capsule that is planned to be removed at 40 EFPY be moved to another location in the MNGP RPV so that it receives a greater amount of neutron fluence? Explain: If it can be moved, what is the impact of moving it on the estimated neutron fluence for the capsule?

d. Since BWRVIP-116 is presently under review by the staff, the staff requests that the applicant add to Commitment 41 the following:

> "The Reactor Vessel Surveillance Program will be enhanced to ensure that any additional requirements that result from the NRC review of BWRVIP-116 will be addressed prior to the period of extended operation."

NMC Response

a. The capsules in the MNGP reactor vessel are located at 30, 120 and 300 degrees. The lead factor (the ratio of the neutron fluence for the surveillance capsule to the peak neutron fluence for the reactor vessel at the 1/4 thickness depth) for the 30-degree capsule was calculated to be 0.41, as reported in Battelle report BCL-585-84-2 Revision 1, Examination, Testing, and Evaluation of Irradiated Pressure Vessel Surveillance Specimens from the Monticello Nuclear Generating Plant. The lead factor for the capsules at 120 and 300 degrees are also expected to be 0.41 due to symmetry. Consequently, for the next surveillance capsule specimens tested, a lead factor of 0.41 is predicted based on analyses described in the Battelle report.

b. The neutron fluence for the surveillance capsules lags the peak neutron fluence for the MNGP RPV due to the arrangement of fuel bundles in the reactor core and the azimuthal position of the capsule. The vessel fluence at various azimuthal positions highly depends on the relative distance of the peripheral bundles to the vessel surface and the power density of the peripheral bundles. For the MNGP RPV, the vessel surface at 0 to 15 degrees azimuth is significantly closer to the peripheral bundles than the capsule at the 30-degree azimuth. Thus, the peak vessel fluence near the 0-degree azimuth is significantly higher than the capsule fluence.

c. The surveillance capsule specimen holders in the MNGP reactor vessel are located at the 30, 120 and 300 degree azimuths. As a result of examination of specimens removed from the 30° location in 1981 an azimuthal correction factor was determined to be approximately 4.0. Consequently, moving the capsule to the peak azimuth would be expected to increase the capsule flux by a factor of 4. The net capsule fluence for that position at a later test date would, therefore, depend on the total accumulated Effective Full Power Years (EFPY) before the move and the remaining EFPY after the move.

   With regard to actual movement of the specimen holders, a major obstacle is the method of attachment. The current method makes use of brackets welded to the surface of the reactor vessel. Attachment of new brackets at another location would be severely limited due to the highly irradiated, axial location within the annulus. In addition, sensitization of the aged low alloy base material is a consideration. Although possible, given these limitations and the fact that fluence analysis methodology is recognized as yielding reasonable and conservative fluence profiles, movement of the specimen holder to a new location is not considered prudent.

d. The "Compilation of Commitments Related to License Renewal Aging Management for the Monticello Nuclear Generating Plant" included as

Enclosure 3 to NMC Letter No. L-MT-05-014 to the NRC (Application for Renewed Operating License, dated March 16, 2005) was subsequently revised and re-submitted on June 10, 2005 (ADAMS Accession No. ML051680145).

The June 10, 2005 submittal resulted in renumbering of this commitment from 41 to 42 and inserted "NMC" into the response as follows:

> "NMC intends to use the Integrated Surveillance Program for MNGP during the period of extended operation by implementing the requirements of BWRVIP-116, which is currently being reviewed by the NRC."

Inherent in the above commitment is that the NMC will address any additional requirements prior to the period of extended operation resulting from the staff's review that are applicable to MNGP and that have been identified within a time frame which allows such a review and response. The Reactor Vessel Surveillance Program, described in LRA Section B2.1.29, will be enhanced to ensure that these additional requirements are addressed prior to the period of extended operation.

D. Reactor Coolant System

1. NRC RAI 3.1.2-1

Table 3.1.2-2 of the MNGP LRA indicates that the Top Head Dollar Plate, Top Head Flange and the Top Head Torus are susceptible to Crack Initiation and Growth due to Stress Corrosion Cracking and Intergranular Stress Corrosion Cracking, and their intended function is pressure boundary. The material is identified as alloy steel (A533-65 Grade B Class and A508 Class 2) and clad (308/309). Identify which materials provide the pressure boundary function and provide your basis, including any operating experience, for concluding that these materials are susceptible to Crack Initiation and Growth due to Stress Corrosion Cracking and Intergranular Stress Corrosion Cracking.

NMC Response

The base material of the top head enclosure components provides the pressure boundary function. These materials are identified in LRA Table 3.1.2-2 as A533-65 Grade B, Class 1 for the top head dollar plate and the top head torus, and A508 Class 2 for the top head flange. For corrosion resistance all components were clad with stainless steel (308/309). Although not contained as a GALL line item, these components have been identified as susceptible to crack initiation and growth due to stress corrosion cracking and intergranular cracking because of operating experience at other plants that indicate this susceptibility.

This operating experience was initially identified to BWR owners in RICSIL 50 and is also described in BWRVIP-74-A. In response to RICSIL 50, MNGP did perform UT inspection of the RPV head welds and a visual inspection on the interior surfaces in 1991. No evidence of cracking was observed. Although the cracking found at other plant sites was observed primarily in the cladding, crack penetration into the base material was also observed. As a result, MNGP included this aging effect in Table 3.1.2-2 for these components with the note that "Aging effect not in NUREG-1801 for this component, material, and environment combination."

E. Neutron Embrittlement of the Reactor Pressure Vessel and Internals

1. NRC RAI 4.2-1

Section 4.2.1 of the LRA indicates that neutron fluence was calculated for the MNGP RPV for the extended 60-year (54 EFPY) licensed operating period based on 3.90×10^8 megawatt hour (MWh) through Cycle 22 at 1775 megawatts thermal (MWt) plus 4.76×10^8 MWh at 1880 MWt. This results in a peak neutron fluence of 5.17×10^{18} n/cm^2 (E>1.0 MeV) and a peak 1/4T fluence of 3.82×10^{18} n/cm^2 (E>1.0 MeV) for the RPV and a neutron fluence at the inside of the shroud of 3.84×10^{21} n/cm^2 (E>1.0 MeV) at the end of the extended operating period. To support the projected MNGP neutron fluence values, provide the following information:

a. Identify the current operating cycle for MNGP. Clarify your intent with respect to uprating the power level of MNGP and how any intended future power uprate corresponds to the assumptions noted above regarding MNGP operating cycles beyond Cycle 22.

b. Provide the MNGP capacity factor at which MNGP has operated over the current and three preceding operating cycles.

c. State the capacity factor and neutron flux that were assumed in the MNGP LRA for all future MNGP operating cycles through the end of the period of extended operation. If different capacity factors were assumed for future operating cycles at 1775 MWt and at 1880 MWt, provide the assumed capacity factors for each power level.

d. If the capacity factors from the response to (b) are different than those in response to (c), provide justification for using the capacity factors from (c) for determining the projected neutron fluence in the MNGP LRA neutron embrittlement analyses.

NMC Response

a. The current operating cycle for MNGP is 23. Currently there are no approved plans in place for another power uprate, although NMC continues to review the feasibility and benefits of an Extended Power Uprate Project at the MNGP for business planning purposes. The 1880 MWt power level in the fluence evaluation was used for conservatism and exceeds the current licensed power level of 1775 MWt. This was done for conservatism and consistency with most calculations that were performed in support of the 1998 uprate from 1670 MWt to 1775 MWt. Since fluence is a function of the total estimated energy, any requested increase beyond the current fluence basis for license renewal of 1880 MWt will require a revision to the RG 1.190 neutron flux and fluence evaluation.

b. The MNGP flux/fluence evaluation for 54 Effective Full Power Years (EFPY) is independent of capacity factors. As described in (c) the maximum flux was determined based on Cycle 22 core data. Then, based on a conservative assessment of the respective EFPYs for the MNGP licensed power levels the total projected fluence was determined. Although this methodology is limited by the 54 EFPY parameter which is a function of the overall capacity factor MNGP has incorporated a multiplier on the flux/fluence calculation to account for operational variances which encompasses the effects on end-of-life EFPY.

   In addition, see NMC's response to a previous RAI regarding "historic and predicted capacity factors" (TAC No. MC6440), dated June 10, 2005 (ADAMS Accession No. ML051680145).

c. Flux estimates for the MNGP were performed in accordance with the General Electric methodology for neutron flux calculation documented in Licensing Topical Report (LTR) NEDC-32983P-A which has been approved by the NRC. In general, this methodology adheres to the guidance in Regulatory Guide 1.190 for neutron flux evaluation. A key input to this calculation was the total integrated power (MWDth) through the first 22 cycles of operation. In addition, Cycle 22 core data was used as a basis for the calculation. Flux profiles were generated from this data and, using the maximum flux, the integrated fluence at 54 EFPY was determined. Fluence estimates at 54 EFPY were conservatively determined using 1775 MWt for Cycles 1 through 22 (previous to rerate implementation in the fall of 1998 the rated power was 1670 MWt) and 1880 MWt for the remainder of the license renewal period of extended operation (54 EFPY). This resulted in EFPYs of 25.09 and 28.91 respectively.

In addition to the conservative methodology described above, a bias adjustment derived from extensive benchmarking of the methodology against measured data as well as an uncertainty related to the flux calculation was incorporated. To account for variations in operation (e.g. capacity factor, core design, etc.), a multiplier of 1.3 was applied to the reactor pressure vessel to obtain a bounding fluence. See response to b) above.

Based on the preceding, the MNGP estimate of fluence is considered conservative and bounding for the license renewal extended period of operation.

d. The methodology described in (c) to project fluence used total integrated power (a function of capacity factor) from past cycles as the basis for projecting future operations. The basis has been consistently applied and, to account for potential variations in operations, also includes an additional multiplier of 1.3. Since the bases (integrated power/capacity factor) are consistent, no further justification for this aspect of the flux and fluence evaluation is required.

2. NRC RAI 4.2-2

Tables 4.2.1-1 and 4.2.1-2 contain the equivalent margins analysis (EMA) for the limiting MNGP plate and weld. These tables do not include an evaluation of surveillance plate and weld data. Surveillance data was submitted to the NRC in a letter dated December 21, 1998. The letter contains report SIR-97-003, Revision 2, "Review of the Results of Two Surveillance Capsules, and Recommendations for the Materials Properties and Pressure-Temperature Curves to be Used for the Monticello Reactor Pressure Vessel," which indicates unirradiated upper shelf energy (USE) data was available for surveillance plates, but not available for surveillance welds. Therefore, USE evaluations using surveillance data could be performed for the plates but not the welds. Based on the surveillance plate data in SIR-97-003, Revision 2, determine the impact of the surveillance plate data on the limiting beltline plate USE and evaluate what impact, if any, this data has on validity of the plate EMA.

NMC Response

Using the "1st Capsule" data for plate C2220-2 identified in Table 2-1 of SIR-97-003, Revision 2 results in a measured decrease of 18.3% as opposed to an 11.5% predicted decrease using RG 1.99 Figure 2 as noted in LRA Table 4.2.1-1 at a fluence of 2.93 x 10^{17} n/cm^2. Correspondingly, at the 54 EFPY 1/4T fluence of 3.82 x 10^{18} with an 18.3% measured decrease the RG1.99 Position 2.2 adjusted decrease is 33.5% which exceeds the margin to safety requirement of 23.5% defined in BWRVIP-74-A.

As opposed to this use of EMA criteria, a more representative and conservative approach is to use the 1st capsule initial USE (71 ft-lb) in conjunction with the 54 EFPY fluence and %Cu (0.17). This results in an RG1.99 Figure 2 decrease of 21% or 56 ft-lb 54 EFPY USE which exceeds the minimum acceptable value of 50 ft-lbs at 54 EFPY. Further, it can be demonstrated that the adjusted % decrease obtained using RG1.99 Position 2.2 also results in a 54 EFPY USE greater than 50 ft-lbs. As described above, using the data from SIR-97-003 results in a Position 2.2 decrease of 33.5%. With a transverse unirradiated USE of 86.5 ft-lbs (0.65 times 133 ft-lbs), a decrease of 33.5% results in a 54 EFPY USE of 57.5 ft-lbs, which also exceeds the 50 ft-lb minimum identified in 10 CFR 50, Appendix G.

This substantiates the conclusion presented in LRA Table 4.2.1-1 and confirms that surveillance plate heats C2220-1 and -2 have adequate upper shelf energy for safe operation through the license renewal period of extended operation.

3. <u>NRC RAI 4.2-3</u>

Using the generic weld and plate data in Appendix B of EPRI Topical Report (TR) -113596, "BWR Vessel and Internals Project, Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines (BWRVIP-74)," determine the projected Charpy USE for the limiting weld and plate in the reactor vessel beltline at the 1/4T depth using the neutron fluence at the end of the period of extended operation.

<u>NMC Response</u>

The following table provides the calculated 54 EFPY USE values for the limiting plate, nozzle, and weld materials. The MNGP LRA submittal provided an Equivalent Margin Analysis (EMA) because there is insufficient unirradiated USE data for the beltline materials. Unirradiated USE data exists for Plate Heat C2220-1 and -2; however, there is no USE data for the weld materials. Therefore, for the limiting weld material, which is Shielded Metal Arc Weld (SMAW) material, the Mean – $\kappa\sigma$ value (95/95 confidence) from Section B.3.1 of BWRVIP-74 is used as the unirradiated USE value in conjunction with the Monticello weld chemistry and fluence in the calculations presented in the following:

USE Determination for MNGP 54 EFPY Limiting Beltline Plate and Weld Materials

Location	Heat	Initial USE (ft-lbs)	1/4T Fluence (n/cm^2)	%Cu	% Decrease USE per RG1.99	54 EFPY USE (ft-lbs)
Lower-Intermediate Shell Plate 1-15[1]	C2220	71 [2]	3.82×10^{18}	0.17	21	56
Beltline Weld	Unknown SMAW	84.5 [3]	3.82×10^{18}	0.10	19.5	68

Notes:

1. Although this material meets the requirements of 10 CFR 50 Appendix G for 54 EFPY, sufficient unirradiated USE data is not available for the beltline materials (see Section 4.2.1 of the MNGP LRA submittal); therefore, an EMA evaluation was provided to support the MNGP LRA submittal.

2. The value of 71 ft-lbs is cited in a letter from B.A. Wetzel (NRC) to R.O. Anderson (NSPC), "Generic Letter (GL) 92-01, Revision 1, 'Reactor Vessel Structural Integrity' – Monticello Nuclear Generating Plant (TAC No. M83485)", dated April 5, 1994, and represents 65% of the measured USE (109 ft-lbs) for the Charpy plate specimens tested from the first MNGP surveillance capsule. Evaluation of unirradiated data for plate Heat C2220-2 demonstrates that the initial USE is 133 ft-lbs longitudinal, which translates to 86.5 ft-lbs transverse. A conservative value of 71 ft-lbs is used in this calculation.

3. There is no information available to support an unirradiated USE value for the limiting MNGP weld material. Therefore, the initial USE considered in this table is obtained from Section B.3.1 of BWRVIP-74 (ADAMS Accession No. ML012920549). This value represents the Mean - $\kappa\sigma$ (95/95 confidence) USE value for the 41 available SMAW data points.

4. <u>NRC RAI 4.2-4</u>

The peak fluence at the RPV wall for the MNGP RPV is 5.17×10^{18} n/cm^2 (E>1.0 MeV) for 54 EFPY of operation. Based on this fluence value, the previous Reflood Thermal Shock Analysis of the RPV is not bounding for the period of extended operation. The original analysis has been superseded by an analysis for BWR-6 RPVs that is applicable to the MNGP BWR-3 RPV. The BWR-6 RPV

analysis is applicable to MNGP because it uses a bounding main steam line break event and an RPV thickness similar to the MNGP RPV. This analysis assumes end-of-license (EOL) material toughness, which in turn depends on the EOL adjusted reference temperature (ART). The critical location for the fracture mechanics analysis is at 1/4 of the RPV thickness (from the inside, 1/4T). For the main steam line break event, the peak stress intensity occurs at approximately 300 seconds after initiation of the event. The analysis shows that at 300 seconds into the thermal shock event, the temperature of the vessel wall at 1.5 inches deep (which is the 1/4T depth for the BWR-6 RPV) is approximately 400°F. For the MNGP vessel, the 1/4T depth is 1.26 inches. Provide the fracture toughness (peak stress intensity value) required to prevent fracture of the MNGP RPV due to reflood thermal shock.

NMC Response

Paper G1/5, "Fracture Mechanics Evaluation of a Boiling Water Reactor Vessel Following a Postulated Loss of Coolant Accident," Ranganath, S., Fifth International Conference on Structural Mechanics in Reactor Technology, Berlin, Germany, August 1979, defines the basis for this evaluation. As noted in the MNGP LRA submittal, the BWR/6 example in the paper referenced above, bounds the conditions at the MNGP. This was demonstrated in the submittal by comparison of the parameters for the BWR/6 case versus the plant-specific MNGP case. As shown in the submittal, the plant-specific temperature at 1/4T depth into the vessel wall was determined to be 370°F at 300 seconds into the thermal shock event. It was also stated that using the highest 60 year Adjusted Reference Temperature (ART), the beltline material reaches upper shelf (200 ksi√in) at 261°F. Since this temperature is significantly lower than 370°F, it is assured that the beltline material remains at upper shelf at 300 seconds into the thermal shock event. Figure 5 of "Fracture Mechanics Evaluation of a Boiling Water Reactor Vessel Following a Postulated Loss of Coolant Accident," Ranganath, S., Fifth International Conference on Structural Mechanics in Reactor Technology, Berlin, Germany, August 1979, Paper G1/5, further demonstrates that at 300 seconds into the thermal shock event and at 1/4T depth into the vessel wall, the maximum applied stress intensity is 103 ksi√in. Therefore, there is sufficient margin to prevent fracture due to reflood thermal shock.

F. Stress Relaxation of Rim Holddown Bolts

1. NRC RAI 4.8-2

The Time-Limited Aging Analysis (TLAA) for Stress Relaxation of Rim Holddown Bolts is discussed in Section 4.8 of the LRA. To more accurately address Stress Relaxation of Rim Holddown Bolts at MNGP, a plant-specific calculation was performed that incorporated the MNGP core plate geometry, an operating temperature of 288° C (550° F), and an MNGP fluence calculation that was

performed in accordance with guidance provided in Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," March 2001, (LRA Section 4.2.1). The maximum fluence applicable to the bolts of the core plate was determined to be 2.2 x 10^{19} n/cm^2 (E>1.0 MeV) at the end of the 60-year operating license. The resultant relaxation was determined to be based on GE Design Documents. The analysis assumed that all of the bolts were at this fluence even though many bolts experience a lower fluence depending on their specific location. This plant-specific analysis is bounded by the original analysis that conservatively assumed a higher value of 19 percent relaxation.

a. Provide the stress-relaxation curves and their supporting data that were utilized to determine the percent relaxation of the rim holddown bolts at MNGP. Explain how this data was utilized to establish the curves.

b. The staff requests that the applicant provide information regarding the material type, (i.e., type 304 or type 316, etc.), values of neutron flux and temperature related to the data cited in response to RAI 4.8-2(a) and compare them to the type of material, neutron flux and temperature values of the rim holddown bolts at MNGP. If the type of material, neutron flux or temperature values for the rim holddown bolts at MNGP are different than that for the data, evaluate the impact of these differences on the predicted stress relaxation values of the rim holddown bolts at MNGP.

c. Describe the load on the specimens used to develop the stress relaxation curves in response to RAI 4.8-2(a). Explain why these specimens are applicable for use in determining the axial stess relaxation in the rim holddown bolts.

d. Based on the stress relaxation curves provided in response to RAI 4.8-2(a) describe how the value of 8 percent stress relaxation was determined for the rim holddown bolts at MNGP.

e. The bending stresses in the holddown bolts result from the horizontal loads acting on the core plate. Some of these loads may depend on the preloading of the holddown bolts. The core plate is also subjected to vertical loads, which could cause portions of the core plate rim to separate from the shroud support as a result of smaller bolt preloads.

Show that the axial and bending stresses for the mean and highest loaded holddown bolts will not exceed the American Society of Mechanical Engineers Boiler and Pressure Vessel Code Section III allowable stresses for P_m and P_m+P_b as a result of the loss of preload at the end of the period of extended operation. State clearly the assumptions on which the analysis was based.

NMC Response

a. Stress-relaxation properties of irradiated austenitic steels and nickel alloys were studied extensively at GE in the late 1970s and early 1980s, and mean and 95-95 limit curves for use in design were developed and issued in the GE BWR Materials Handbook. The data used to develop this design curve were stress relaxation measurements on irradiated austenitic steels and nickel alloys in laboratory and in-core specimens. The specimen types for these measurements were springs and bent-beam specimens.

   Figure 1, "The Relaxation of Irradiated Austenitic Steels and Ni," shows the GE mean design curve along with the data used for its derivation. The curve is based on a model that assumed a stress-linear, primary plus secondary creep law form, and was fit to the shown data using stepwise multiple regression. The data presented as closed symbols are from measurements made on springs, and the data shown as open symbols are measurements made using bent-beam specimens.

b. The core plate hold-down bolts used in MNGP are Type 304 stainless steel. The data used to develop the curve in Figure 1 includes several austenitic materials. However, based on the following discussion, the use of data from other austenitic material is considered appropriate, as shown in Figure 1.

   High energy radiation produces a number of simultaneous effects in materials, mostly originating with the displacement of atoms from their original lattice position to relatively distant locations, usually as an interstitial. The interstitial atoms and the associated vacancies group into interstitial and vacancy clusters (hardening), migrate to grain boundaries, and relax constant displacement stresses with interaction with and displacement by dislocations. These effects in austenitic stainless steels are most strongly influenced by the face-centered cubic (FCC) structure of the materials.

   Relaxation of irradiated, structural materials from radiation creep is much less sensitive to "normal material variations" (e.g., in austenitic stainless steels) than other radiation properties. Radiation segregation and hardening characteristics are similar for all austenitic stainless steels, although some experience presegregation (from annealing). Also, neutron relaxation is among the most consistent and reproducible phenomenon, and little variation is observed in stainless steel (e.g., 304, 316, 321, 347/8, L-grade and nuclear grade). The relaxation behavior of these stainless steels is often used for many different austenitic alloys such as Nitronic 50, Alloy X-750 and Alloy 718.

To further support this observation, see Figure 7-17 in the BWRVIP-99 report, "Crack Growth Rates in Irradiated Stainless Steels in BWR Internal Components" (repeated here as Figure 2). This figure shows stress relaxation data from wedge loaded DCB specimens in 288°C water at Halden that are exposed to neutron fluences of approximately 4.4 to 6 x 10^{20} n/cm^2 (>1 MeV) (i.e. ~0.6 to 0.9 dpa). This data shows stress relaxation levels clustered between 28% and 36% for DCB specimens fabricated from 304/316/348 stainless steels. It should be noted that this data is for fluence levels nearly 10 times higher than that predicted for the MNGP bolts, and that the effects at lower fluences would be expected to be less pronounced.

To further demonstrate the applicability of the GE design curve, Figure 3, "The Relaxation of Irradiated Austenitic Steels, GE Mean Design Curve and Additional Data," shows the relaxation data from J.P. Foster and Halden with the curve. It is seen that the GE design curve predicts higher relaxation levels (i.e. lower fraction of load remaining) than observed from the Foster and Halden data, and is thus conservative compared to this data.

More than 80% of the tests, shown in Figure 1 were conducted at a temperature of 550°F and a majority of these were conducted in an operating BWR environment. The other tests, were conducted at either 570 or 600°F which is expected to produce more relaxation. Since such a large portion of the data was conducted at typical BWR operating conditions, the data temperature is considered fully representative of the core plate bolts.

While the cumulative fluence information was available as part of the original test reports and the GE Design Curve documentation, the flux conditions were not directly available. Many of the tests were associated with springs which had reached fluences ranging from ~ 8 x 10^{20} to 8 x 10^{21}n/cm^2. Based on a reasonable time of exposure, the flux would be expected to range from ~ 7 x 10^{12} to 9 x 10^{13}n/cm^2/s. The fluxes that were defined for two of the smaller sets of test data were 2.7 x 10^{14} and 2 x 10^{17}n/cm^2/s, respectively. A review of the data over these 4 orders of magnitude showed no discernable flux dependence. However, the neutron flux levels were at least 100 times higher than that experienced by the core plate bolts.

As described above, the temperature data are representative for use in the MNGP core plate bolt evaluation. The neutron flux data, however, were measured in specimens subjected to fluxes ranging from ~1 x 10^{13} to 2 x 10^{17}n/cm^2/s. This is higher than the 8.5 x 10^9 n/cm^2/s average flux experienced by the core plate bolts themselves. Given the large range of higher flux for which the properties are the same, the impact of the lower flux to which the bolts are exposed is viewed to be negligible.

The effect of radiation on microstructure is the driving force for stress relaxation with neutron fluence. The primary effect of radiation is to harden the material through the creation of vacancy and interstitial defects in the crystal lattice. These defects affect creep and segregation within the material which will increase the strength, produce stress relaxation and lead to very narrow changes in the local composition at grain boundaries (G.S. Was and P. L. Andresen, "Irradiation-Assisted Stress Corrosion Cracking in Austenitic Alloys," Journal of Metals, April 1992). Additionally, the data on materials under Light Water Reactor (LWR) conditions: i.e. temperature less than 350°C, at fluences in the 0.5 to 5 dpa (~6 x 10^{19} to ~6.6 x 10^{20} n/cm^2) and attributable to a LWR neutron spectrum is limited (P. Scott: "A review of irradiation assisted stress corrosion cracking," Journal of Nuclear Materials, Vol. 211, pp. 101-122, 1994). Below this fluence level, damage is just starting to occur. Following a short-term transient, the creep strain at constant load is very linear with fluence (which in turn is integrated flux over time) ("BWRVIP-99: BWR Vessel and Internals Project: Crack Growth Rates in Irradiated Stainless Steels in BWR Internals Components," TR-1003018, December 2001). Scott showed that the LWR operating flux/temperature region is broadly associated with processes of limited microstructural dimension that in turn is associated with radiation-induced segregation (RIS) (Figure 4, "Relationship of Microstructural Radiation Induced Processes as a Function of Flux and Temperature"). The continuity of these processes to lower flux levels is illustrated in Figure 4 for LWR material changes. Therefore, extrapolation of the higher flux data to the core plate region is fully justified and appropriate. Consistent with other microstructural processes, higher temperature and higher fluxes would be expected to produce bigger changes. This data, generated at higher flux values, would therefore be expected to be conservatively representative relative to the lower flux conditions experienced by the MNGP core plate bolts with respect to relaxation.

c. As discussed in the response to b, the GE design curve is based on data from different austenitic materials and specimen-types. Relaxation data from springs (torsion) and bent-beam specimens (tension) make up the database used to derive the GE design curve, and are shown in Figure 1. Data reported later by Halden (in-core DCBs / tension) and Foster (springs / shear) and shown in Figure 3 exhibit the same behavior with increasing fluence as the GE design curve, but at a lower value of relaxation (i.e., higher fraction of remaining load). Relative to this data, the GE design curve is somewhat conservative. The data in both Figures 1 and 3 do not exhibit any apparent effect of specimen or loading type. Therefore, the data used to derive the GE design curve is applicable to the MNGP loading configuration.

d. A MNGP specific bolt average fluence of 2.2×10^{19} n/cm^2 (E>1MeV) was calculated at the peak fluence location. Given that the fluence on the core plate bolts varies depending on the azimuthal location, with a significant fraction of the bolts at less than the peak fluence, this provides an additional conservatism for most bolt locations. The MNGP specific fluence of 2.2×10^{19} n/cm^2 (E>1MeV) was then compared to the original, hand drawn GE design curves to determine a value of 8% relaxation. Figure 1 is a re-plotted version of the GE design curve that shows a computer generated curve fit of the same data. The value of 8% relaxation is conservative relative to the statistical curve fit shown in Figure 1.

e. A MNGP specific evaluation has been completed which includes the following key results: 1) high margin exists in the preload in the core plate bolts at the end of the period of extended operation, 2) the high retained preload provides assurance that separation of the core plate or sliding will not occur, and 3) since separation and sliding of the core plate are precluded, the bolts are subject to only axial loads (e.g. bending is negligible).

Key assumptions included in the MNGP evaluation that demonstrate conservatism include: 1) although the core plate is held down by two sizes of bolts, 2.0" and 2.5" diameter, the calculation of bolt stress assumes all bolts are 2.0" in diameter, 2) the calculated values of bolt stresses for the faulted condition differential pressure and seismic loads are conservatively compared with normal condition allowable stress, and 3) the calculation takes no credit for the core plate aligner pins.

Demonstration that separation of the core plate rim and shroud support does not occur requires calculation of the net vertical load and horizontal load due to operating and seismic conditions postulated for MNGP. The maximum (faulted condition) resultant net vertical external load was calculated and found to be approximately 30% of the total initial preload in the bolts. The preload in the bolts was further reduced due to radiation-induced relaxation. The relaxation of this preload due to irradiation is estimated to be approximately 8% (see response to d above). There is adequate margin available in the preload even after relaxation at the end of the period of extended operation. Since the retained bolt preload exceeds the maximum external vertical loads for the faulted condition, separation of the core plate will not occur.

To demonstrate that core plate sliding does not occur, the maximum horizontal load on the core plate was calculated and compared to the holddown load.

Since the frictional resistance based on minimum hold down loads exceeds the maximum horizontal seismic loads, sliding of the core plate is not expected to occur.

Bolt stresses are the highest for the initial preload conditions in conjunction with the maximum postulated vertical loads. Conservatively, the maximum membrane stress was calculated by combining the maximum (initial) preload in the bolts with the maximum postulated vertical loads, and by considering the tensile stress area based on 2.0" diameter bolts. This calculated stress (10,470 psi) was found to be within the ASME Code allowable stress of 16,950 psi. Since it has been shown that sliding is not predicted, the only significant stress is the membrane stress.

Based on the above, there is adequate stress margin available in the bolts compared to the code allowable stresses. Also there is adequate end-of-life pre-load in the bolts to prevent sliding of the core plate and separation of the core plate from the shroud.

ENCLOSURE 1

Figure 1
Relaxation of Irradiated Austenitic Steels & Ni-GE Mean Design Curve

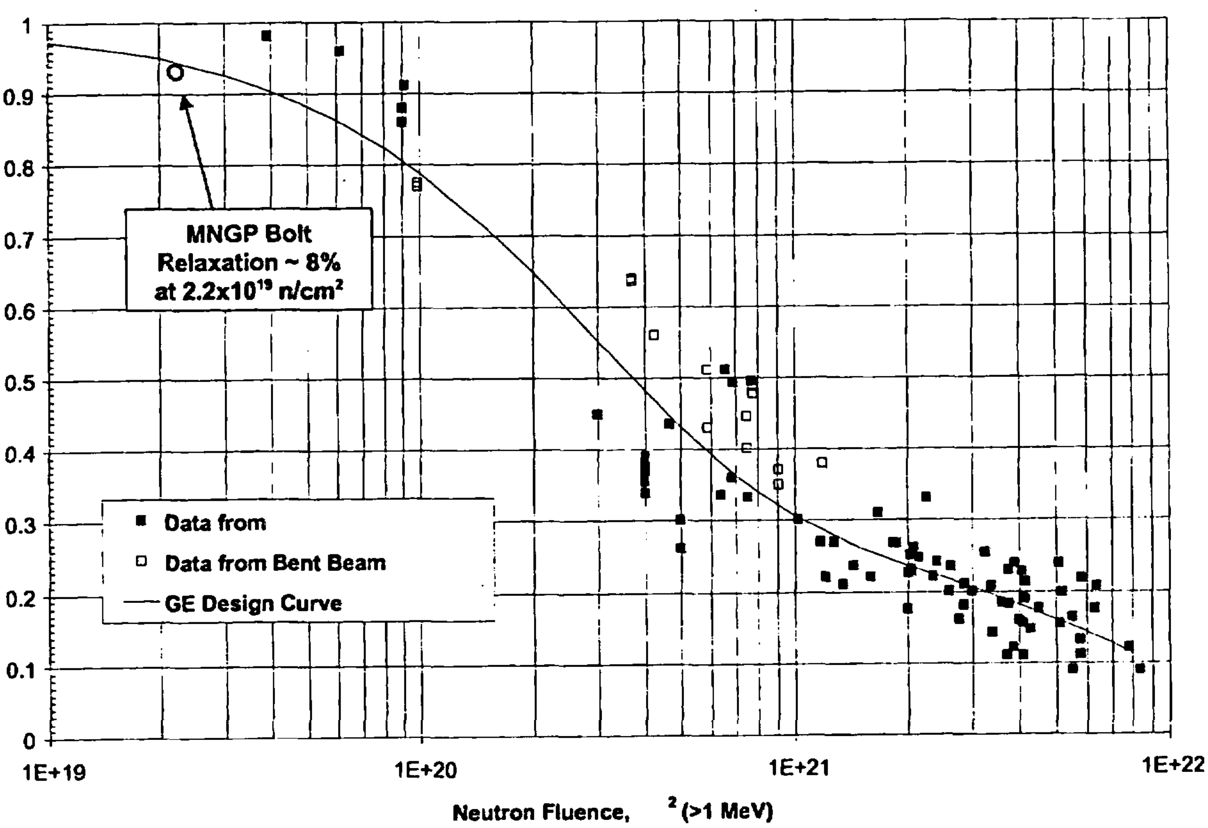

Figure 2
Stress Relaxation Data from Halden
(Shown as Figure 7-17 in BWRVIP-99).

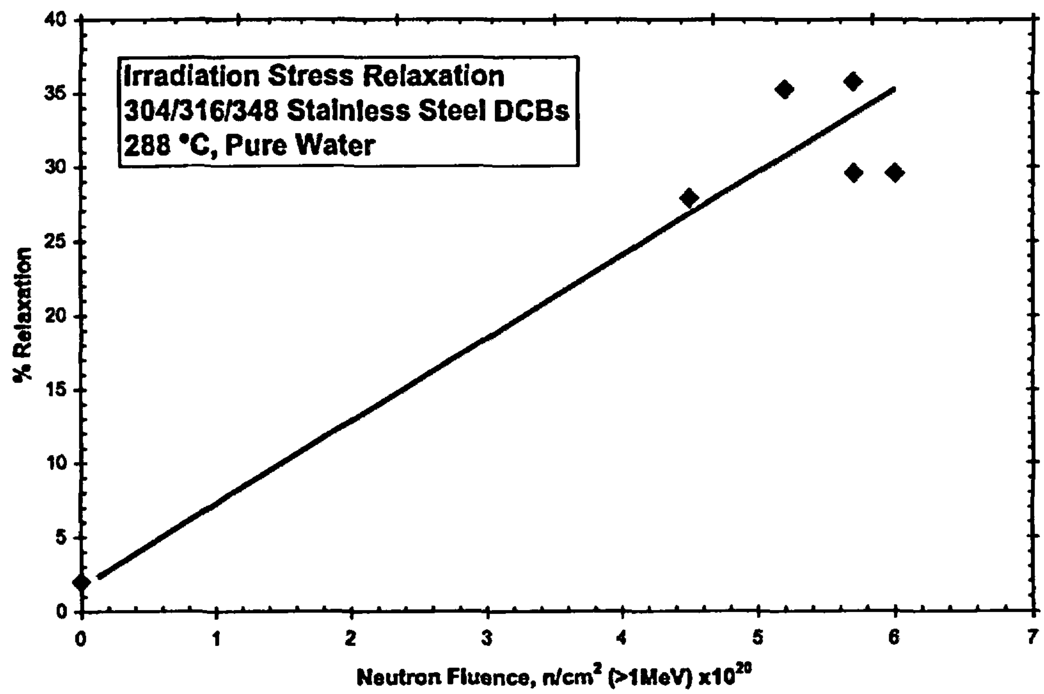

Figure 7-17
Stress relaxation in wedge-loaded DCB specimens at Halden that exhibited minimal or no crack growth [7-22].

ENCLOSURE 1

Figure 3
Relaxation of Irradiated Austenitic Steels
GE Mean Design Curve and Additional Data

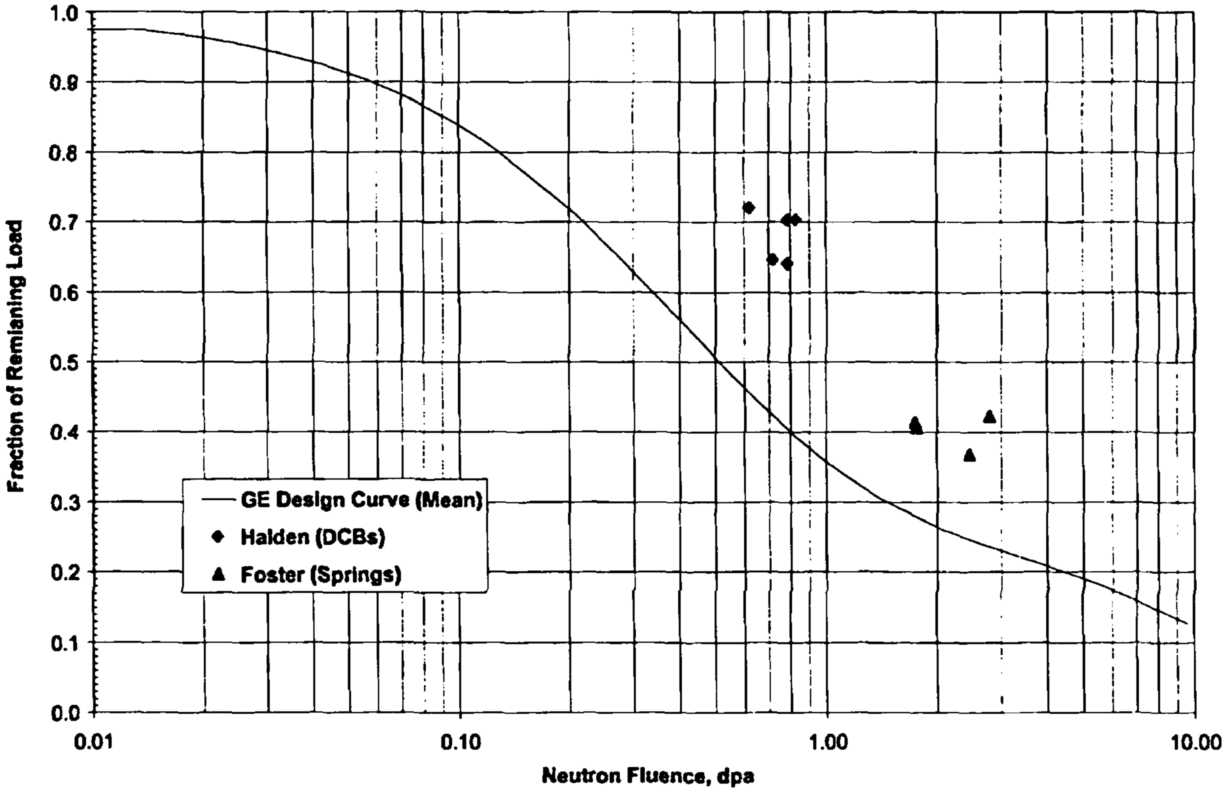

Figure 4
Relationship of Microstructural Radiation Induced Processes as a Function of Flux and Temperature