P.O. Box 63
Lycoming, New York 13093

Constellation Energy Group

Nine Mile Point Nuclear Station

January 9, 2004
NMP2L 2109

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

SUBJECT: Nine Mile Point Unit 2
Docket No. 50-410
Facility Operating License No. NPF-69

License Amendment Request: Revision to the Reactor Pressure Vessel Material Surveillance Program

Gentlemen:

Pursuant to 10 CFR 50.90, Nine Mile Point Nuclear Station, LLC (NMPNS) hereby requests an amendment to Nine Mile Point Unit 2 (NMP2) Operating License NPF-69. The proposed change revises the NMP2 licensing basis by replacing the current plant-specific reactor pressure vessel (RPV) material surveillance program with the Boiling Water Reactor Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP). Specifically, the proposed change revises the NMP2 Updated Safety Analysis Report (USAR) to reflect the following key elements:

- NMP2 participation in the ISP, whose program documents consist of BWRVIP-78, dated December 1999, and BWRVIP-86-A, dated October 2002, and

- The use of a methodology for determination of RPV and/or surveillance capsule neutron fluences that is in accordance with the recommendations of Regulatory Guide (RG) 1.190.

By letter dated February 1, 2002, the NRC issued a Safety Evaluation (SE) approving the BWRVIP ISP as an acceptable alternative to all existing BWR plant-specific RPV surveillance programs for the purpose of maintaining compliance with the requirements of 10 CFR 50 Appendix H, "Reactor Vessel Material Surveillance Program Requirements." In Regulatory Issue Summary (RIS) 2002-05, "NRC Approval of Boiling Water Reactor Pressure Vessel Integrated Surveillance Program," dated April 8, 2002, the NRC stated that licensees who elect to participate in the ISP shall submit a license amendment request to incorporate this program into their licensing basis. This license amendment request is consistent with the guidance contained in the referenced NRC SE and the RIS.

A008

Similar requests have previously been approved for the AmerGen Energy Company's Clinton Power Station, Unit 1, by NRC letter dated August 12, 2003 (TAC No. MB6998), and for Exelon Generation Company's Quad Cities Nuclear Power Station, Units 1 and 2, by NRC letter dated August 28, 2003 (TAC Nos. MB7008 and MB7009).

NMPNS requests approval of the proposed amendment within one year. Once approved, the amendment shall be implemented within 90 days. This letter contains no new regulatory commitments, as reflected in Attachment 3.

Pursuant to 10 CFR 50.91(b)(1), NMPNS has provided a copy of this license amendment request and the associated analyses regarding no significant hazards consideration to the appropriate state representative.

Very truly yours,

Peter E. Katz

Peter E. Katz
Vice President Nine Mile Point

PEK/DEV/bjh

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

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

Peter E. Katz

Subscribed and sworn before me, a Notary Public in and for the State of New York and County of Oswego, this 9th day of January 2004.

WITNESS my Hand and Notarial Seal:

Sandra A. Oswald
Notary Public

My Commission Expires:

1/9/04
Date

SANDRA A. OSWALD
Notary Public, State of New York
No. 01OS6032276
Qualified In Oswego County
Commission Expires 10/25/05

Attachments:
1. Evaluation of Proposed Changes
2. Proposed Changes to Updated Safety Analysis Report Pages (Mark-up)
3. List of Regulatory Commitments

cc: Mr. H. J. Miller, NRC Regional Administrator, Region I
Mr. G. K. Hunegs, NRC Senior Resident Inspector
Mr. P. S. Tam, Senior Project Manager, NRR (2 copies)
Mr. J. P. Spath, NYSERDA

ATTACHMENT 1

EVALUATION OF PROPOSED CHANGES

Subject: ***License Amendment Request: Revision to the Reactor Pressure Vessel Material Surveillance Program***

1.0 DESCRIPTION

This letter is a request to amend Nine Mile Point Unit 2 (NMP2) Operating License NPF-69. The proposed change revises the NMP2 licensing basis by replacing the current plant-specific reactor pressure vessel (RPV) material surveillance program with the Boiling Water Reactor Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP). Specifically, the proposed change revises the NMP2 Updated Safety Analysis Report (USAR) to reflect the following key elements:

- NMP2 participation in the ISP, whose program documents consist of BWRVIP-78, dated December 1999 (Reference 2), and BWRVIP-86-A, dated October 2002 (Reference 3), and

- The use of a methodology for determination of RPV and/or surveillance capsule neutron fluences that is in accordance with the recommendations of Regulatory Guide (RG) 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," dated March 2001.

2.0 PROPOSED CHANGE

The NMP2 USAR, Section 5.3.1.6, "Material Surveillance," describes the current plant-specific RPV material surveillance program. This USAR section is revised to document NMP2 participation in the ISP, including references to appropriate BWRVIP program documents and the NRC safety evaluation (SE) dated February 1, 2002 (Reference 1). Other sections of the USAR that contain information related to the RPV material surveillance program are also revised as appropriate.

A description of the methods of analysis used to determine reactor vessel neutron flux and fluence is contained in USAR Section 4.1.4.5, "Neutron Fluence Calculations," and USAR Section 4.3.2.8, "Vessel Irradiations," as referenced by USAR Section 5.3.1.6.2, "Neutron Flux and Fluence Calculations." A summary of NMP2 conformance with NRC regulatory guides is contained in USAR Section 1.8. These USAR sections are revised, as appropriate, to describe the use of a neutron fluence calculational methodology that is in accordance with the recommendations of RG 1.190.

The proposed changes to the USAR are provided in Attachment 2. Following NRC approval of the license amendment request, the USAR will be updated to incorporate the changes identified in Attachment 2 in accordance with 10 CFR 50.71(e).

3.0 BACKGROUND

Appendix H to 10 CFR 50 requires that reactor pressure vessels have their beltline regions monitored by a surveillance program that complies with American Society for Testing and Materials (ASTM) E 185, except as modified by Appendix H. ASTM E 185 provides guidelines for designing a minimum surveillance program, selecting materials, and evaluating test results for light-water cooled nuclear power reactor vessels. It also provides recommendations for the minimum number of surveillance capsules and their withdrawal schedule. 10 CFR 50, Appendix H further requires that the proposed withdrawal schedule be submitted to and approved by the NRC prior to implementation.

The NMP2 RPV material surveillance program was developed in accordance with 10 CFR 50, Appendix H, and ASTM E 185-73. The program is described in NMP2 USAR Section 5.3.1.6, "Material Surveillance." The current NMP2 RPV surveillance capsule withdrawal schedule is contained in USAR Table 5.3-3.

The BWRVIP ISP was developed in response to an issue raised by the NRC regarding the potential lack of adequate unirradiated baseline Charpy V-notch (CVN) data for one or more materials in plant-specific RPV surveillance programs at several boiling water reactors (BWRs). The lack of baseline properties would inhibit a licensee's ability to effectively monitor changes in the fracture toughness properties of RPV materials in accordance with 10 CFR 50, Appendix H. The BWRVIP ISP, as approved by the NRC in its SE dated February 1, 2002 (Reference 1), resolves this issue.

Implementation of the ISP will provide additional benefits. When the original surveillance materials were selected for plant-specific surveillance programs, the state of knowledge concerning RPV material response to irradiation and post-irradiation fracture toughness was not the same as it is today. As a result, many facilities did not include what would be identified today as the plant's limiting RPV materials in their surveillance programs. Hence, this effort to identify and evaluate materials from other BWRs, which may better represent a facility's limiting materials, should improve the overall evaluation of BWR RPV embrittlement. Also, the inclusion of data from the testing of BWR Owners' Group (BWROG) Supplemental Surveillance Program (SSP) capsules will improve overall quality of the data being used to evaluate BWR RPV embrittlement. Further, occupational radiation exposure will be reduced due to elimination of the need for some units (including NMP2) to remove material specimens. Overall, the combined benefits of the ISP are substantial. Finally, implementation of the ISP is also expected to reduce the cost of surveillance testing and analysis because surveillance materials that are of little or no value will no longer be tested.

4.0 TECHNICAL ANALYSIS

In BWRVIP-78 (Reference 2) and BWRVIP-86-A (Reference 3), the technical basis is described for the development and implementation of an ISP intended to support operation of all U. S. BWR RPVs through the completion of each facility's current 40-year operating license.

In its SE dated February 1, 2002 (Reference 1), the NRC concluded that the ISP proposed by the BWRVIP, if implemented in accordance with the conditions of the NRC SE, is an acceptable alternative to all existing BWR plant-specific RPV surveillance programs for the purpose of maintaining compliance with the requirements of 10 CFR 50 Appendix H through the end of the current facility 40 year operating licenses. The NRC SE requires that each licensee electing to participate in the ISP (1) provide information regarding what specific neutron fluence methodology will be implemented as part of participation in the ISP, and (2) address neutron fluence methodology compatibility as it applies to the comparison of neutron fluences calculated for its RPV versus the neutron fluences calculated for surveillance capsules in the ISP which are designated to represent its RPV. This information is provided in the following discussion.

With respect to the specific fluence methodology, NMPNS has used the methodology described in NMPNS letter NMP2L 2096 dated August 15, 2003 (Reference 4) to calculate the most recent fluence values. This calculation was performed to support proposed revisions to the NMP2 RPV pressure-temperature limit curves that were submitted to the NRC in letter NMP2L 2096. As noted in letter NMP2L 2096, details regarding the fluence methodology were included in reports provided to the NRC in NMPNS letters NMP2L 2015 dated March 8, 2001, and NMP1L 1708 dated January 15, 2003 (References 5 and 6). Supplemental information was submitted in letter NMP1L 1749 dated July 31, 2003 (Reference 7). The methodology is in accordance with the recommendations of RG 1.190 and was approved by the NRC in a letter dated October 27, 2003 (Reference 8). The NMP2 USAR is being revised (Attachment 2) to reflect that an NRC-approved fluence methodology will be used which conforms with RG 1.190. Use of an NRC-approved fluence methodology satisfies the first condition contained within the NRC SE (Reference 1).

Regarding neutron fluence methodology compatibility, at the August 29, 2002 "Workshop on the BWRVIP RPV Integrated Surveillance Program," the NRC staff stated that methodology compatibility is satisfied if the surveillance capsules and the RPVs are evaluated with an NRC-approved methodology that complies with RG 1.190. The requirement to use an NRC-approved fluence methodology that is consistent with RG 1.190 is being included in the NMP2 USAR (Attachment 2). Use of an NRC-approved fluence methodology that is consistent with RG 1.190 satisfies the second condition contained within the NRC SE (Reference 1).

In accordance with the existing plant-specific RPV material surveillance program, the first NMP2 surveillance capsule has been withdrawn and tested. As discussed in USAR Section 5.3.1.6, the existing program also utilized data from LaSalle County Station, Units 1 and 2, and Columbia Generating Station (formerly WNP-2) to supplement the NMP2 surveillance data. Under the ISP, NMP2 is not identified as a host plant. The representative materials for the NMP2 limiting RPV plate and weld materials, and their associated withdrawal schedules, are identified in Reference 3. Thus, in accordance with the ISP, future withdrawal and testing of NMP2 surveillance capsules will be permanently deferred.

5.0 REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration

Nine Mile Point Nuclear Station, LLC (NMPNS) is proposing to revise the licensing basis for Nine Mile Point Unit 2 (NMP2) by replacing the plant-specific reactor pressure vessel (RPV) material surveillance program with the Boiling Water Reactor Vessel Internals Project (BWRVIP) Integrated Surveillance Program (ISP). This change is acceptable because the BWRVIP ISP has been approved by the NRC staff as meeting the requirements of paragraph III.C of Appendix H to 10 CFR 50 for an integrated surveillance program.

NMPNS has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change implements an ISP that has been evaluated by the NRC as meeting the requirements of paragraph III.C of Appendix H to 10 CFR 50. The proposed change provides the same assurance of RPV integrity as has always been provided. Implementation of an ISP is not a precursor or initiator of any accident previously evaluated. No physical changes to the plant will result from the proposed change. The proposed change will not cause the RPV or interfacing systems to be operated outside of any design or testing limits, and will not alter any assumptions or initial conditions previously used in evaluating the radiological consequences of accidents.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed change revises the NMP2 licensing bases to reflect participation in the BWRVIP ISP. The ISP was approved by the NRC staff as an acceptable material surveillance program that complies with 10 CFR 50, Appendix H. No physical changes to the plant will result from the proposed change. The proposed change does not affect the design or operation of any system, structure, or component. As an alternate monitoring program, the ISP cannot create a new failure mode involving the possibility of a new or different kind of accident.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   The proposed change has no impact on the margin of safety of any TS. There is no impact on safety limits or limiting safety system settings. The change does not affect any plant safety parameters or setpoints. No physical or operational changes to the plant will result from the proposed change.

   The RPV material surveillance program requirements contained in 10 CFR 50, Appendix H provide assurance that adequate margins of safety exist during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the reactor coolant pressure boundary may be subjected over its service lifetime. The BWRVIP ISP has been approved by the NRC staff as an acceptable material surveillance program that complies with 10 CFR 50, Appendix H. The ISP will provide the material surveillance data that will assure that the safety margins required by the NRC regulations are maintained.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above evaluation, NMPNS concludes that the proposed amendment involves no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

10 CFR 50 Appendix G, "Fracture Toughness Requirements," which is invoked by 10 CFR 50.60, "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation," specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary, including RPVs. In order to support evaluations to demonstrate that compliance with these requirements will be maintained, information regarding irradiated RPV material properties and the neutron fluence level of a licensee's RPV is necessary. Therefore, 10 CFR 50.60 also invokes 10 CFR 50, Appendix H, which requires licensees to implement an RPV material surveillance program.

An alternative provided in Appendix H to 10 CFR 50 is to implement an ISP. An Appendix H requirement for an ISP is that "the representative materials chosen for surveillance for a reactor are irradiated in one or more other reactors that have similar design and operating features." Appendix H, Paragraph III.C, "Requirements for an

Integrated Surveillance Program," sets forth specific criteria upon which approval of an ISP shall be based. In its safety evaluation dated February 1, 2002 (Reference 1), the NRC documented that the BWRVIP ISP met the criteria specified in 10 CFR 50, Appendix H, Paragraph III.C.

Conformance with the NRC General Design Criteria (GDC) for Nuclear Power Plants, Appendix A to 10 CFR 50, is described in Section 3.1 of the NMP2 USAR. In particular, GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," requires, in part, that the reactor coolant pressure boundary be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions, the boundary will behave in a non-brittle manner and the probability of a rapidly propagating fracture is minimized. Adoption of the ISP does not alter the NMP2 USAR statement of conformance with GDC 31.

In conclusion, based on the considerations discussed above and in Section 4.0, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed adoption of an integrated surveillance program for reactor material specimen surveillances at NMP2 would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 REFERENCES

1. Letter from U. S. NRC to C. Terry (BWRVIP), "Safety Evaluation Regarding EPRI Proprietary Reports 'BWR Vessel and Internals Project, BWR Integrated Surveillance Program Plan (BWRVIP-78)' and 'BWRVIP-86: BWR Vessel and Internals Project, BWR Integrated Surveillance Program Implementation Plan,'" dated February 1, 2002

2. BWRVIP-78, "BWR Vessel and Internals Project, BWR Integrated Surveillance Program Plan," Final Report, December 1999

3. BWRVIP-86-A, "BWR Vessel and Internals Project, Updated BWR Integrated Surveillance Program (ISP) Implementation Plan," Final Report, October 2002

4. NMPNS Letter to the NRC, NMP2L 2096 dated August 15, 2003, "License Amendment Request Pursuant to 10 CFR 50.90: Revision of Reactor Pressure Vessel Pressure-Temperature Limits"

5. NMPNS Letter to the NRC, NMP2L 2015 dated March 8, 2001, "10CFR50, Appendix H, Reactor Vessel Material Surveillance Program Requirements, Report of Test Results"

6. NMPNS Letter to the NRC, NMP1L 1708 dated January 15, 2003, "Transmittal of Neutron Transport Calculations Benchmarking Report (TAC Nos. MB6687 and MB6703)"

7. NMPNS Letter to the NRC, NMP1L 1749 dated July 31, 2003, "Request for Additional Information (RAI) – Amendment Application Re: Pressure-Temperature Limit Curves (TAC Nos. MB6687 and MB6703)"

8. NRC Letter to NMPNS dated October 27, 2003, "Nine Mile Point Nuclear Station, Unit No. 1 – Issuance of Amendment Re: Pressure-Temperature Limit Curves and Tables (TAC No. MB6687)"

ATTACHMENT 2

Proposed Changes to Updated Safety Analysis Report Pages (Mark-up)

The current versions of the following Updated Safety Analysis Report (USAR) pages have been marked-up by hand to reflect the proposed changes. The USAR will be updated to incorporate these proposed changes in accordance with 10 CFR 50.71(e).

Pages	Tables	Figures
4.1-13	1.8-1, 49 of 49	4.3-1
4.1-15	4.3-2	4.3-2
4.3-3	4.3-3	
5.3-5	5.3-2a	
5.3-6	5.3-2b	
5.3-7	5.3-3	
5.3-11	5A-1, 2 of 2	
5.3-19	5A-2, 1 of 2	
5.3-20	5A-2, 2 of 2	
5A-2	5A-4	
5B-1		
5B-3		

REGULATORY GUIDE 1.150, REVISION 1 (FEBRUARY 1983) (cont'd.)

examination were 7.5 percent of the welds in the RHR system, HPCS system, and LPCS system, normally excluded from preservice volumetric examination.

The degree of compliance with RG 1.150 Revision 1 is provided in Table 1.8-1a. This document presents the alternate method of compliance with the regulatory guide, how the compliance is achieved (under the column "Response"), and what documents or procedures reference the compliance implementation (under the column "Procedures and References"). The Degree of Compliance position is incorporated in the RPV examination procedures. A technique qualification shall be performed using the ultrasonic examination systems that will be employed during the automated PSI examinations of the RPV. Actual RPV and nozzle segments containing known size reflectors located in the ID surface shall be used. This qualification shall be witnessed by the licensee, its Agents, and the Unit 2 ANII, as a minimum. The qualification shall demonstrate that flaws of the maximum allowable limits are detectable.

REGULATORY GUIDE 1.155 (AUGUST 1988) - STATION BLACKOUT

FSAR Section 8.3.1.5

Position The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

Unit 2 is evaluated against the requirements of the Station Blackout Rule, 10CFR50.63, using the guidance contained in NUMARC 87-00, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," NUMARC 87-00 Supplemental Questions/Answers, dated December 27, 1989, and NUMARC 87-00 Major Assumptions, dated December 27, 1989, except where RG 1.155 takes precedence. Table 1 of RG 1.155 provides a cross-reference between the regulatory guide and NUMARC 87-00. Any exceptions to the NUMARC guidance taken by Unit 2 are identified in the SBO documentation (see Letter No. NMP2L 1230, dated April 3, 1990, to NRC, TAC No. 68571).

Regulatory Guide 1.163 (September 1995) - Performance-Based Containment Leak-Test Program

FSAR Section 6.2.6

Position The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide with the following exceptions:

1. The MSIVs measured leakage is excluded from the combined leakage rate of 0.6 La, and as-found testing is not required to be performed on the MSIVs.

2. Primary containment airlocks door seals are tested prior to reestablishing containment integrity when something has been done that would bring into question the validity of the previous airlock door seal test.

Regulatory Guide 1.190 (March 2001) - Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence

FSAR Section 4.1.4.5

Position Evaluations of Unit 2 reactor vessel neutron fluence will use a method that complies with the Regulatory Position (Paragraph C) of this guide.

Add

Extent of Application

ANSYS is used extensively in GE/NEG for elastic and elastic-plastic analysis of the RPV, core support structures, reactor internals, and fuel.

4.1.4.2 Fuel Rod Thermal Analysis

Fuel rod thermal design analyses are described in Section 4 of GESTAR II[10].

4.1.4.3 Reactor Systems Dynamics

The analysis techniques and computer codes used in reactor system dynamics are described in Section 5.4 of GESTAR II[10]. Section 4.4.4 also provides a complete stability analysis for the reactor coolant system (RCS).

4.1.4.4 Nuclear Engineering Analysis

The analysis techniques are described and referenced in Section 3 of GESTAR II[10].

4.1.4.5 Neutron Fluence Calculations

Neutron vessel fluence calculations were originally carried out using a one-dimensional, discrete ordinate, S_N transport code with general anisotropic scattering. This code is a modification of a widely-used discrete ordinate code which solves a wide variety of radiation transport problems. The program solves both fixed source and multiplication problems. Slab, cylinder, and spherical geometries are allowed with various boundary conditions. The fluence calculations incorporate, as an initial starting point, neutron fission distributions prepared from core physics data as a distributed source. Anisotropic scattering was considered for all regions. The cross sections were prepared with 1/E flux weighted, $P_{(3)}$ matrices for anisotropic scattering but did not include resonance self-shielding factors. Fast neutron fluxes at locations other than the core midplane were calculated using a two-dimensional, discrete ordinate code. The two-dimensional code is an extension of the one-dimensional code.

For power uprate conditions, a full two-dimensional calculation of vessel neutron fluences was performed using the DORT[12] discrete ordinates transport code which is an updated version of the DOT code[11]. The core was modeled by specifying two homogeneous regions representing the central and outer portions of the core in each of 25 axial intervals. The Los Alamos National Laboratory 80-group cross sections were used as the source of the microscopic cross sections data.

Insert 1

4.1.5 References

1. Beitch, L. Shell Structures Solved Numerically by Using a Network of Partial Panels, AIAA Journal, Vol. 5, No. 3, March 1967.

2. Wilson, E. L. A Digital Computer Program For the Finite Element Analysis of Solids With Non-Linear Material Properties, Aerojet General Technical Memo No. 23, Aerojet General, July 1965.

3. Farhoomand, I. and Wilson, E. L. Non-Linear Heat Transfer Analysis of Axisymmetric Solids, SESM Report SESM71-6, University of California at Berkeley, Berkeley, CA, 1971.

4. McConnelee, J. E. Finite-Users Manual, General Electric TIS Report DF 69SL206, March 1969.

5. Clough, R. W. and Johnson, C. P. A Finite Element Approximation For the Analysis of Thin Shells, International Journal Solid Structures, Vol. 4, 1968.

6. A Computer Program For the Structural Analysis of Arbitrary Three-Dimensional Thin Shells, Report No. GA-9952, Gulf General Atomic.

7. Burgess, A. B. User Guide and Engineering Description of HEATER Computer Program, March 1974.

8. Young, L. J. FAP-71 (Fatigue Analysis Program) Computer Code, GE/NED Design Analysis Unit R. A. Report No. 49, January 1972.

9. Rashid, Y. R. Users Manual for CRPLS01 Computer Program, NEDO-23538, December 1976.

10. General Electric Standard Application for Reactor Fuel, including United States Supplement, NEDE-24011-P-A and NEDE-24011-P-A-US (latest approved revision).

11. RSIC Computer Code Collection - DOT IV Version 4.3 - One and Two-Dimensional Transport Code System, CCC-429, Oak Ridge National Laboratory Report, November 1983.

12. RSIC Computer Code Collection - TORT Three-Dimensional Discrete Ordinates Transport, CCC-543, Oak Ridge National Laboratory, October 1991.

Insert 2

compositions for the stainless steel in the shroud and the carbon steel in the vessel contain the mixtures by weight as specified in the ASME material specifications for ASME SA-240, 304L, and ASME SA-533 Grade B. In the region between the shroud and the vessel, the presence of the jet pumps was ignored. A simple diagram showing the regions, dimensions, and weight fractions is shown on Figure 4.3-1.

The distributed source used for this analysis was obtained from the gross radial power description. The distributed source at any point in the core is the product of the power from the power description and the neutron yield from fission. By using the neutron energy spectrum, the distributed source is obtained for position and energy. The integral over position and energy is normalized to the total number of neutrons in the core region. The core region is defined as a 1-cm thick disc with no transverse leakage. The power in this core region is set equal to the average power in the axial direction. The radial power distribution is shown on Figure 4.3-2.

The neutron fluence is determined from the calculated flux by assuming that the plant is operated 90 percent of the time at 90 percent power level for 40 yr, or equivalent to 1×10^9 full power seconds. The neutron fluence will be re-evaluated as required by the plant operating history. The originally calculated fluxes and fluence are listed in Table 4.3-2. The calculated neutron flux leaving the cylindrical core is listed in Table 4.3-3.

Vessel neutron fluences were reevaluated for power uprate conditions using a full two-dimensional calculation method, as described in Section 4.1.4.5. The results indicate that the original neutron fluence basis described above continues to bound operation of the unit at the uprated power level of 3,467 MWt.[3]

Insert 3

4.3.3 Analytical Methods

See Section A.4.3.3 of GESTAR II[1].

4.3.4 Changes

See Section A.4.3.4 of GESTAR II[1].

ORIGINAL

TABLE 4.3-2

CALCULATED NEUTRON FLUXES
USED TO EVALUATE VESSEL IRRADIATION

Neutron Energy (MeV)	Average Flux in Core (n/cm^2-sec)	Flux at Core Boundary (n/cm^2-sec)	Flux at Inside Surface Vessel (n/cm^2-sec)	Flux at 1/4 T Vessel (n/cm^2-sec)
>3.0	1.5×10^{13}	5.4×10^{12}	3.3×10^{8}	1.6×10^{8}
1.0-3.0	3.3×10^{13}	1.2×10^{13}	2.8×10^{8}	2.3×10^{8}
0.1-1.0	5.3×10^{13}	1.7×10^{13}	4.9×10^{8}	6.7×10^{8}

NOTES:

1. The calculated flux is a maximum in the axial direction but average over the azimuthal angle.

2. Maximum fluence >1.0 MeV = 1.1×10^{18} n/cm^2 at 1/4 T of vessel. The maximum fluence is calculated using 1×10^{9} full power seconds. The fluence includes an azimuthal peaking factor and a factor to cover analytical uncertainties. The azimuthal peaking factor is derived from the results of a two-dimensional transport calculation. The two-dimensional analysis models the reactor bundle pattern in (R,θ) geometry. Fluxes are calculated at the inside radius of the vessel. The peaking factor used is 1.4.

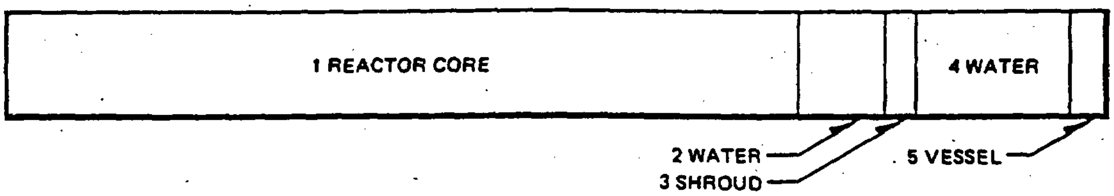
   In addition to the angular peaking factor, a safety factor of 2 was applied to ensure that the predicted values are conservative.

ORIGINAL

TABLE 4.3-3

CALCULATED NEUTRON FLUX AT CORE EQUIVALENT BOUNDARY

Group	Lower Energy Bound (ev)	Flux (n/cm^2-sec)
1	10.0×10^6	4.6×10^{10}
2	6.065×10^6	6.1×10^{11}
3	3.679×10^6	2.1×10^{12}
4	2.231×10^6	4.2×10^{12}
5	1.353×10^6	4.4×10^{12}
6	8.208×10^5	3.9×10^{12}
7	4.979×10^5	4.0×10^{12}
8	3.020×10^5	2.8×10^{12}
9	1.832×10^5	2.3×10^{12}
10	1.111×10^5	1.8×10^{12}
11	6.732×10^4	1.4×10^{12}
12	4.087×10^4	1.1×10^{12}
13	2.478×10^4	1.0×10^{12}
14	1.503×10^4	1.0×10^{12}
15	9.119×10^3	9.6×10^{11}
16	5.531×10^3	9.4×10^{11}
17	3.355×10^3	9.4×10^{11}
18	2.034×10^3	9.1×10^{11}
19	1.010×10^3	1.3×10^{12}
20	2.492×10^2	2.5×10^{12}
21	5.560×10^1	2.6×10^{12}
22	1.240×10^1	2.5×10^{12}
23	0.625	4.0×10^{12}
24	0.0	2.5×10^{13}



MATERIAL		RADIUS INCHES	MATERIAL	MATERIAL DENSITY
NO.	NAME		WATER	0.274 g/cm^3
			UO_2	2.642 g/cm^3
1	REACTOR CORE	93.56	ZIRCONIUM	0.896 g/cm^3
2	WATER	101.4	WATER	0.74 g/cm^3
3	SHROUD	103.4	304L STAINLESS STEEL	FROM ASME SA 240
4	WATER	125.5	WATER	0.74 g/cm^3
5	VESSEL	131.68	CARBON STEEL	FROM ASME 533
6	AIR		AIR	1.3×10^{-3} g/cc

ORIGINAL

FIGURE 4.3-1

MODEL FOR ONE DIMENSIONAL TRANSPORT ANALYSIS OF VESSEL FLUENCE

NIAGARA MOHAWK POWER CORPORATION
NINE MILE POINT-UNIT 2
FINAL SAFETY ANALYSIS REPORT

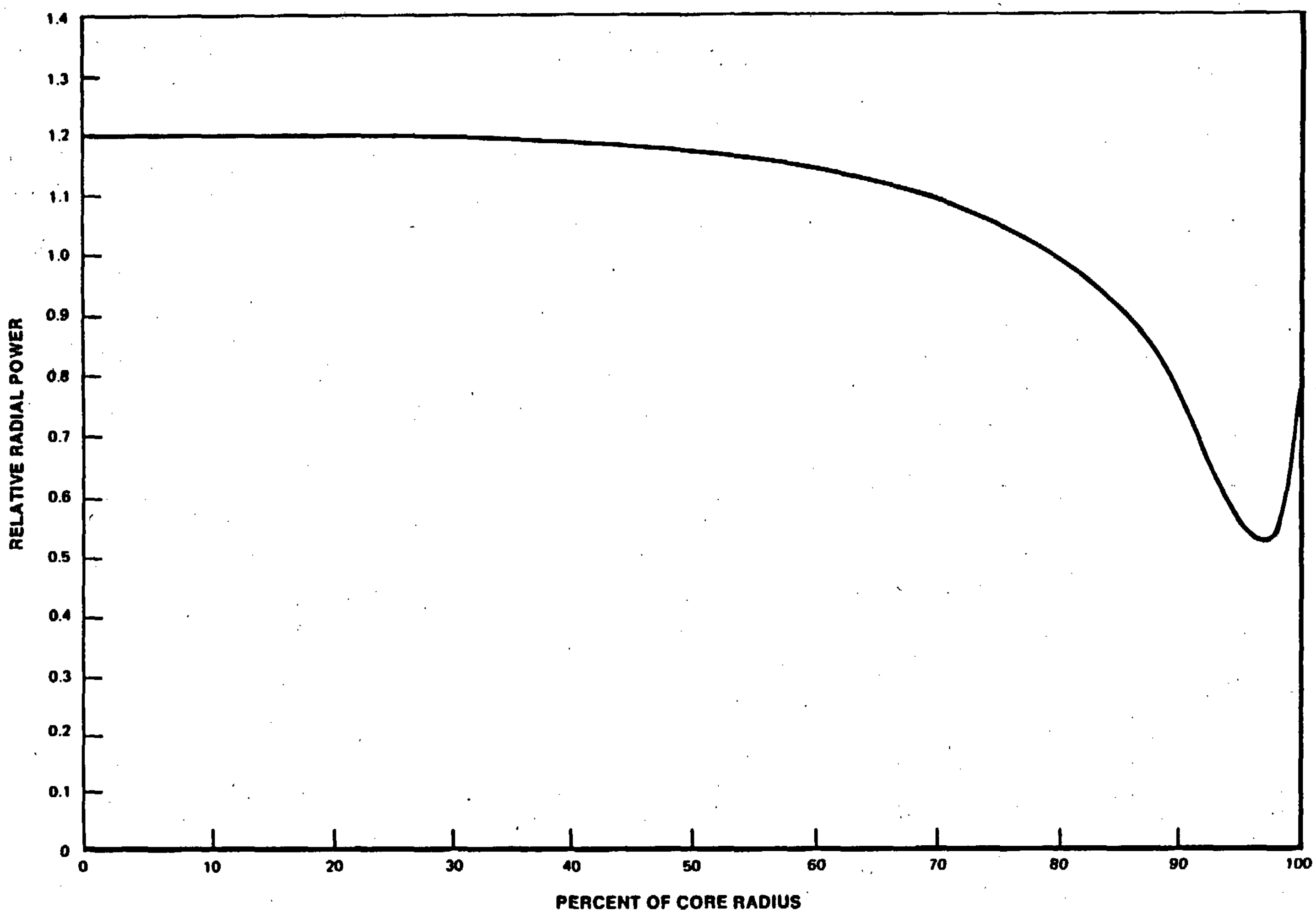


FIGURE 4.3-2

RADIAL POWER DISTRIBUTIONS USED IN THE VESSEL FLUENCE CALCULATION

ORIGINAL

NIAGARA MOHAWK POWER CORPORATION
NINE MILE POINT-UNIT 2
FINAL SAFETY ANALYSIS REPORT

Since the stress intensity factor is greatest at the outside surface of the flange-to-shell and head joints, a flaw can also be detected by outside surface examination techniques.

Fracture Toughness Margins in the Control of Reactivity

Appendix G of ASME Section III was used in determining pressure-temperature (P-T) limitations for all phases of plant operation. The additional requirements of 10CFR50 Appendix G, May 1983, are included in the P-T limitations.

Results of Chemical Analysis and RT_{NDT} Evaluations

See Tables 5.3-1, 5.3-2a, and 5.3-2b.

5.3.1.6 Material Surveillance

5.3.1.6.1 Compliance with Reactor Vessel Material Surveillance Program Requirements

The materials surveillance program monitors changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region resulting from their exposure to neutron irradiation and thermal environment.

The original Were

Reactor vessel materials surveillance specimens ~~are~~ provided in accordance with requirements of ASTM E185-73 and 10CFR50 Appendix H, except for material selection indicated in Section 5.3.1.5.1. Materials for the program ~~are~~ selected to represent materials used in the reactor beltline region. Specimens ~~are~~ manufactured from a plate actually used in the beltline region, and a weld typical of those in the beltline region, and thus represent base metal, weld material, and the weld HAZ material. The plate and weld ~~are~~ Were heat treated in a manner that simulates the actual heat treatment performed on the core region shell plates of the completed vessel.

Each in-reactor surveillance capsule contains 36 Charpy V-notch specimens. The capsule loading consists of 12 specimens each of base metal, weld metal, and HAZ material. A set of out-of-reactor baseline Charpy V-notch specimens and archive material are provided with the surveillance test specimens.

~~The three capsules will be withdrawn in accordance with Table 5.3-3. The lead factors for the surveillance capsules originally were 0.29 for the inside surface of the reactor vessel and 0.41 for the 1/4 T position. These lead factors are based on a fluence of 1.7 x 10^{18} n/cm^2 at the vessel inside surface and 1.1 x 10^{18} n/cm^2 at 1/4 T. These lead factors were calculated assuming that the vessel was symmetrical. This assumption was made because the vessel qualification program does not provide measurements of vessel radii to identify the angular locations where the vessel inside diameter is larger. It is possible that a surveillance capsule could be located at an~~

Insert 4

~~extended radius position. This would provide surveillance sample test results lower than calculated. This is nonconservative when the peak fluence is estimated from the capsule data using lead factors.~~

~~Vessel neutron fluences were reevaluated for power uprate conditions as described in Section 4.1.4.5. The lead factor was determined to be .46 at 1/4 T for the uprated plant conditions. However, the fracture toughness analysis, which was performed in accordance with RG 1.99 Rev. 2 (see Sections 5.3.1.6.3 and 5.3.2.1.3) is not impacted by a change in the value of the 1/4 T lead factor.~~

~~Fracture toughness testing of irradiated capsule specimens is in accordance with requirements of 10CFR50 Appendix H.~~

~~Unit 2 will monitor material radiation damage using the Unit 2 capsules and test data from all La Salle 1 and 2 and WNP-2 capsules. This program is described in the Reactor Vessel Material Surveillance Program submitted to the NRC in a letter dated September 30, 1985.~~

5.3.1.6.2 Neutron Flux and Fluence Calculations

A description of the methods of analysis is contained in Sections 4.1.4.5 and 4.3.2.8. The peak fluence of 1/4 T depth of the vessel beltline material is 1.1×10^{18} n/sq cm. All predictions of radiation damage to the reactor vessel beltline material were made using peak fluence values.

5.3.1.6.3 Predicted Irradiation Effects on Vessel Beltline Materials

Estimated maximum changes in RT (initial reference temperature) and upper shelf fracture energy, as a function of the EOL fluence at the 1/4 T depth of the vessel beltline materials, are listed in Tables 5.3-2a and 5.3-2b. The predicted peak EOL fluence at the 1/4 T depth of the vessel beltline is 1.1×10^{18} n/sq cm after 40 yr of service. Transition temperature changes and changes in upper shelf energy were calculated in accordance with the guidance of RG 1.99. Reference temperatures were established in accordance with 10CFR50 Appendix G, and Subsubarticle NB-2330 of ASME Section III.

5.3.1.6.4 Positioning of Surveillance Capsules and Methods of Attachment

Surveillance specimen capsules are located at three azimuths at a common elevation in the core beltline region. The sealed capsules are not attached to the vessel but are mechanically retained by capsule brackets welded to the vessel cladding as shown on Figure 5.3-1. Since reactor vessel specifications require that all low-alloy steel pressure vessel boundary material be produced to fine grain practice, underclad cracking

In areas where brackets, such as the surveillance specimen holder brackets, are located, additional nondestructive examinations are performed on the vessel base metal and stainless steel weld-deposited cladding or weld buildup pads during vessel manufacture. The base metal is ultrasonically examined by straight beam techniques to a depth at least equal to the thickness of the bracket being joined. The area examined is the area of the subsequent attachment weld plus a surrounding band of width equal to at least half the thickness of the part joined. The required stainless steel weld-deposited cladding is similarly examined. The full penetration welds are liquid penetrant examined to ASME Section III standards. Cladding thickness is required to be at least 1/8 in. The above requirements have been successfully applied to a variety of bracket designs that are attached to weld-deposited stainless steel cladding or weld buildups in many operating BWR RPVs.

ISI examinations of core beltline pressure-retaining welds are performed from the outside surface of the RPV. If a bracket for mechanically-retaining surveillance specimen capsule holders were located at or adjacent to a vessel shell weld, it would not interfere with the straight beam or half node angle beam ISI ultrasonic examinations performed from the outside surface of the vessel.

NOTE: Surveillance specimen capsule at 3° azimuth location was ~~has been~~ removed for testing ~~to comply with Technical Specification 4.4.6.1.3 requirements.~~

5.3.1.7 Reactor Vessel Fasteners

The reactor vessel closure head (flange) is fastened to the reactor vessel shell flange by multiple sets of threaded studs and nuts. The lower end of each stud is installed in a threaded hole in the vessel shell flange. A nut and washer are installed on the upper end of each stud. The proper amount of preload can be applied to the studs by a sequential tensioning using hydraulic tensioners. The design and analysis of this area of the vessel is in full compliance with all ASME Section III, Safety Class 1, code requirements. The material for studs, nuts, and washers is SA-540 Grade B23 or B24 at the 130,000 psi-specified minimum yield strengths level.

Hardness tests are performed on all main closure bolting to demonstrate that heat treatment has been properly performed. A minimum of 45 ft-lb Charpy V-notch (C_V) energy and 25 mils lateral expansion is required at 70°F. The maximum reported

in accordance with the original plant-specific material surveillance program.

determined from Charpy data since 50 ft-lbs was not achieved within 60°F of the NDT. Therefore, σ_I was determined by fitting the Charpy data using a nonlinear least squares regression analysis and using the confidence band data to estimate σ_I. Using this approach, σ_I was taken to be 14.5°F.

original calculated

In order to assess the ART_{NDT} at the 1/4 T and 3/4 T positions, RG 1.99 Revision 2 requires an assessment of the peak fast (E>1Mev) neutron flux at the ID surface of the pressure vessel. Table 4.3-2 reports the fast neutron flux to be 6.1 x 10^8 n/cm^2/sec at the ID surface of the vessel. This table also states that an azimuthal peaking factor of 1.4 should be applied to the flux estimates. In addition, a safety factor of 2 is applied to ensure that the predicted values are conservative. Therefore, a fast neutron flux of 1.708 x 10^9 n/cm^2/sec was used in the P-T curve analysis.

5.3.2.1.4 Reactor Vessel Annealing

In-place annealing of the reactor vessel because of radiation embrittlement is unnecessary because the predicted value of the adjusted reference temperature does not exceed 200°F.

5.3.2.1.5 Predicted Shift in RT_{NDT}

The allowable internal vessel pressure for a specific coolant temperature is a function of several key variables including the ART_{NDT}. The ART_{NDT} for the vessel beltline region enters the P-T calculations directly via the reference stress intensity factor relation (K_{IR}). Therefore, it is necessary to provide reasonable and conservative estimates of the shift in ΔRT_{NDT} for the period of time for which the P-T calculations will be used. The ART_{NDT} was calculated using Revision 2 to RG 1.99 (Figure 5.3-3).

5.3.2.2 Operating Procedures

By comparison of the pressure versus temperature limit in Section 5.3.2.1 with intended normal operating procedures for the most severe upset transient, it is shown that these limits are not exceeded during any foreseeable upset condition. Reactor operating procedures have been established in such a manner that actual transients are less severe than those for which the vessel design adequacy has been demonstrated. Of the design transients, the upset condition producing the most adverse temperature and pressure condition anywhere in the vessel heads and/or shell areas has a minimum fluid temperature of 250°F and a maximum pressure peak of 1,180 psig. Scram automatically occurs as a result of this event, prior to the reduction in fluid temperature, so the applicable operating limits are given on Figures 5.3-2a through 5.3-2e. For a temperature of 250°F, the maximum allowable pressure exceeds 1,600 psig for the intended margin against nonductile failure. The maximum transient pressure of 1,180 psig is within the specified allowable limits.

vessel has also been designed to withstand a limited number of transients caused by Operator error. For abnormal operating conditions where safety systems or controls provide an automatic temperature and pressure response in the reactor vessel, the reactor vessel integrity is maintained since the most severe anticipated transients have been included in the design conditions. Therefore, it is concluded that vessel integrity will be maintained during the most severe postulated transients, since all such transients are evaluated in the design of the reactor vessel. The postulated transient for which the vessel has been designed is discussed in Section 5.2.2.

5.3.3.7 In-service Surveillance

ISI of the RPV is discussed in Section 5.2.4.

The materials surveillance program monitors changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region resulting from their exposure to neutron irradiation and thermal environment. ~~Specimens of actual reactor beltline material are exposed in the reactor vessel and periodically withdrawn for impact testing.~~ Operating procedures will be modified as necessary in accordance with ~~test~~ program results to assure adequate brittle fracture control.

The ISI program is in accordance with applicable ASME Code requirements, and provides assurance that brittle fracture control and pressure vessel integrity are maintained throughout the service lifetime of the RPV.

5.3.4 References

1. Cooke, F. E., et al. Transient Pressure Rises Affecting Fracture Toughness Requirements for Boiling Water Reactors, NEDO-21778-A, December 1978.

2. Watanabe, H. Boiling Water Reactor Feedwater Nozzle/Sparger Final Report. (Supplement 2), NEDE-21821-02, August 1979.

Insert 5

TABLE 5.3-2a

ADJUSTED RT_{NDT} FOR NINE MILE POINT UNIT 2 BELTLINE MATERIALS

Plates - Beltline

Heat No.	Wt. % Cu	Wt. % Ni	ASME NB-2300 Start RT_{NDT} (°F)	12.8 EFPY[3]			32 EFPY[3]		
				RT_{NDT} (°F)	Margin (°F)	ART_{NDT} (°F)	RT_{NDT} (°F)	Margin (°F)	ART_{NDT} (°F)
C3065-1	0.06	0.63	-10	13	24	27	20	28	38
C3121-2	0.09	0.65	0	20	28	48	31	37	68
C3147-1	0.11	0.63	0	26	39	65[2]	40	45	85[2]
C3147-2	0.11	0.63	0	26	39	65[2]	40	45	85[2]
C3066-2	0.07	0.64	-20	15	25	20	24	31	35
C3065-2	0.06	0.63	+10	13	24	47	20	28	58

NOTE: Peak EOL fast (E>1Mev) fluence is 1.724×10^{18} n/cm^2 at vessel ID surface. All calculations based on Revision 2 to Regulatory Guide 1.99.

(Deleted)

(1) ~~These materials are also in the reactor vessel surveillance program.~~
(2) Limiting plate.
(3) Calculations performed at vessel ID surface using peak beltline flux.

See Section 5.3.1.6 for a description of the reactor vessel material surveillance program.

TABLE 5.3-2b

ADJUSTED RT_{NDT} FOR NINE MILE POINT UNIT 2 BELTLINE MATERIALS

Welds - Beltline

Weld Seam	Heat/Lot No.	Wt. % Cu	Wt. % Ni	ASME NB-2300 Start RT_{NDT} (°F)	12.8 EFPY[5]			32 EFPY[5]		
					RT_{NDT} (°F)	Margin (°F)	ART_{NDT} (°F)	RT_{NDT} (°F)	Margin (°F)	ART_{NDT} (°F)
BD,BB,BF	5P5657/0931[~~1~~,2]	0.07	0.71	-60	33	39	12	51	55	46[4]
	5P5657/0931[~~1~~,3]	0.04	0.89	-60	19	27	-14	29	35	4
BA,BB,BC	5P6214B/0331[2]	0.02	0.82	-50	9	22	-19	14	25	-11
	5P6214B/0331[3]	0.014	0.70	-40	8	21	-11	12	23	-5
AB	4P7465/0751[2]	0.02	0.82	-60	9	22	-29	14	25	-21
	4P7465/0751[3]	0.02	0.80	-60	9	22	-29	14	25	-21
	4P7216/0751[2]	0.06	0.85	-50	28	35	13[4]	44	48	42
	4P7216/0751[3]	0.04	0.83	-80	19	27	-34	29	35	-16

NOTE: Peak EOL fast (E>1Mev) fluence is 1.724×10^{18} n/cm^2 at vessel ID surface. All calculations based on Revision 2 to Regulatory Guide 1.99.

(Deleted)

(1) ~~These materials are also in the reactor vessel surveillance program.~~
(2) Single wire submerged arc process.
(3) Tandem wire submerged arc process.
(4) Limiting weld.
(5) Calculations performed at vessel ID surface using peak beltline flux.

See Section 5.3.1.6 for a description of the reactor vessel material surveillance program.

THIS TABLE HAS BEEN DELETED

TABLE 5.3-3

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE*

Capsule Number	Vessel Location	Lead Factor @ 1/4 T	Withdrawal Time (EFPY)
1	3°	0.46	8**
2	177°	0.46	20
3	183°	0.46	Spare

* Any changes to the withdrawal schedule require NRC approval to verify conformance to 10CFR50 Appendix H per Administrative Letter 97-04.

** Withdrawn in RFO7 (Refuel Outage).

1. RT_{NDT} no greater than +10°F for the shell course, head, and closure flange.

2. RT_{NDT} no greater than -20°F for nozzle forgings.

3. RT_{NDT} no greater than -20°F for low alloy weld metal used to join base or weld materials requiring impact testing.

4. Vessel main closure studs meet the Charpy requirement of 45 ft-lb and 25 mils at +10°F.

The use of these toughness limits and values to establish vessel operating limits is described in Section 5.3.2.

5A.3 OTHER FERRITIC REACTOR COOLANT PRESSURE BOUNDARY MATERIALS (NSSS)

The subject materials were impact tested, and are considered to be in compliance with 10CFR50 Appendix G. Specific components, applicable code requirements, and impact test temperatures are as follows:

1. SRV (8 x 10 in) - ASME Section III, 1974 and Summer 1976 Addenda, +60°F maximum.

2. HPCS isolation valve - ASME Section III, 1971 and Winter 1973 Addenda, +40°F maximum.

3. MSIV - ASME Section III, 1977 and Summer 1977 Addenda, +60°F maximum.

5A.4 ORIGINAL PLANT-SPECIFIC REACTOR PRESSURE VESSEL SURVEILLANCE SPECIMENS

Surveillance specimen materials are identified, with properties, in Tables 5A-1, 5A-2, 5.3-2a, and 5.3-2b. It can be seen in Tables 5.3-2a and 5.3-2b that all beltline materials are resistant to radiation degradation of toughness. One of the limiting plates (in terms of EOL RT_{NDT}) was used to fabricate surveillance specimens. The weld materials (Table 5.3-2b) used for the surveillance specimen weld had a predicted EOL RT_{NDT} of 11°F to 13°F lower than the limiting weld material in this respect, but had the highest predicted shift in RT_{NDT} (which is the current basis in ASTM E185-79 (Table 5A-2) for surveillance program design). This is not considered significant because of the insensitivity of these materials to neutron radiation damage. HAZ specimens were taken from the HAZ of the weldment fabricated from the materials indicated in Table 5.3-2a.

The weld procedure used to prepare the surveillance specimen weldment is shown as Attachment 5A-1. Although stick electrodes (shielded metal arc weld) were used to seal backup bars, as shown on Figure 5A-2, these materials were removed by backgouging. The weld joint design (Figure 5A-2) shows that the weld specimens are

TABLE 5A-1 (Cont'd.)

Weld Seam	Type	Heat No.	Lot No. or Flux No.	Drop-weight NDT(°F)	Charpy Toughness			
					Charpy Temp (°F)	Charpy Energy (ft-lb)	Lateral Expansion (mils)	% Shear
	1NMM (tandem wire, trade name Raco)	4P7465	0751 (Linde 124)	-60	-80 -60 0 +10 +40 +212	15,21 62,56,43 79,83,74 71,72,74 76,85 94,106 123,110,111	11,20 47,45,36 66,60,54 69,73,72 77,76 74,88 70,76,77	0,0 30,30,25 30,30,25 30,35,35 40,40 60,75 100,100,100
	1NMM (single wire, trade name Raco)	4P7216	0751 (Linde 124)	-60	-80 -60 0 +10 +40 +212	5,7 19,38,24 41,59,58 64,60,72 66,60 61,63 90,89,94	4,6 22,35,22 32,46,47 48,41,60 53,41 41,44 83,77,68	5,0 15,20,15 20,20,30 70,30,35 45,40 45,40 100,100,100
	1NMM (tandem wire, trade name Raco)	4P7216	0751 (Linde 124)	-80	-100 -80 -20 +10 +40 +212	22,9 31,23,28 62,73,84 84,92,95 89,87 96,97 102,101,98	17,7 15,11,19 40,51,56 57,61,60 51,64 74,68 87,66,66	0,5 5,5,5 30,40,60 60,95,90 95,90 100,100 100,100,100

* ~~Weld materials 5P5657/0931, both single and tandem wire processes, are also in the reactor vessel surveillance program.~~

Limiting weld. See Section 5.3.1.6 for a description of the reactor vessel material surveillance program.

TABLE 5A-2

BELTLINE PLATE TOUGHNESS DATA

(SA-533 Grade B, Class 1 Plate - Lukens Steel Company)

Plate	Heat No./ Slab No.	Drop-weight NDT (Top/Bottom) (°F)	Charpy V-Notch Toughness (Top/Bottom)				
			Orientation (L or T)	Charpy Test Temp. (ft-lb)	Energy (ft-lb)	Lateral Expansion (mils)	% Shear
No. 2 Shell (lower intermediate) 22-1-1	C3065-1	-30/-30	T	+30	44,49,50/54,54,46	46,36,41/36,41,31	40,40,40/30,30,30
			T	+40	55,60,63/41,55,54	48,46,48/40,45,44	50,50,50/40,40,40
			L	+30	63,61,87/74,76,70	41,61,50/56,61/63	60,60,60/70,70,70
			T	+50	70,50,50	42,52,41	50,50,50
			T	+212	100,97,94	76,74,74	99,99,99
				+150	100,106,100	81,84,85	99,99,99
				+70	54,51,54	46,45,47	50,50,50
				-30	40,34,21	16,24,31	30,30,30
				-70	7,10,10	4,6,6	1,1,1
				-100	7,8,6	4,3,2	1,1,1
22-1-2	C3121-2	-30/-50	T	+30	40,60,56	50,54,36	50,50,50
			T	+40	50,51,50/45,42,44	46,41,44/41,41,40	50,50,50/40,40,40
			L	+30	88,86,56/81,53,56	46,61,56/48,42,62	50,50,50/50,50,50
			T	+10	34,30,40	31,30,27	30,30,30
			T	+60	50,53,50	46,46,45	40,40,40
			L	+10	58,44,32	44,31,36	40,40,40
			L	+20	58,36,42	45,31,35	40,40,40
			T	+212	78,76,71	66,64,61	99,98,99
				+100	73,65,73	61,61,65	90,90,90
				0	45,38,38	34,32,39	40,40,40
				-10	35,36,31	31,31,30	30,30,30
				-30	14,15,24	16,10,11	10,10,10
				-100	10,10,6	8,9,2	1,1,1
No. 2 Shell 22-1-3	C3147-1* Add	-20/-30	T	+40	38,40,41	33,40,36	40,40,40
			T	+50	40,62,54/40,54,44	38,41,48/43,41,38	50,50,50/40,40,40
			T	+60	50,51,50/50,50,52	45,44,41/46,44,42	40,40,40/40,40,40
			L	+40	64,64,74	56,50,50	50,50,50
			T	+30	42,42,45	35,41,38	40,40,40
			L	+30	76,80,96	60,61,56	60,60,60
			T	+212	72,70,81	63,66,63	99,99,97
				+100	68,75,86	61,61,68	70,70,70
				+70	66,61,61	53,58,59	80,80,80
				0	39,38,36	36,35,36	40,40,40
				-20	17,23,20	18,20,15	20,20,20
				-150	2,2,2	1,1,1	1,1,1

TABLE 5A-2 (Cont'd.)

Plate	Heat No./ Slab No.	Drop-weight NDT (Top/Bottom) (°F)	Charpy V-Notch Toughness (Top/Bottom)				
			Orien- tation (L or T)	Charpy Test Temp. (ft-lb)	Energy (ft-lb)	Lateral Expansion (mils)	% Shear
No. 1 Shell (lower) 21-1-1	C3147-2*	-20/-30	T	+40	50,51,41	36,38,41	40,40,40
			T	+50	47,56,40	40,42,36	50,50,50
			T	+60	52,50,50	48,44,44	60,60,60
			L	+40	91,96,96	70,65,68	70,70,70
			T	+30	51,56,51	48,51,48	60,60,60
			L	+30	70,80,90	65,74,80	90,90,90
			T	+212	93,86,86	85,79,81	99,99,99
				+100	78,78,80	64,71,66	90,90,90
				+50	54,56,50	48,40,45	50,50,50
				0	34,35,32	31,33,32	30,30,30
				-20	31,23,31	21,21,16	20,20,20
				-150	5,4,6	2,3,2	1,1,1
No. 1 Shell 21-1-2	C3066-2	-30/-40	T	+30	45,38,43/38,54,43	36,38,38/40,44,36	40,40,40/40,40,40
			T	+40	58,72,58/55,52,51	58,48,48/45,42,45	50,50,50/40,40,40
			L	+30	78,80,90	65,61,66	70,70,70
			T	+20	60,56,36	34,50,59	50,50,50
			L	+20	77,84,80	62,51,60	60,60,60
			T	+212	80,91,86	67,69,71	99,99,99
				+150	88,93,84	76,84,79	99,99,99
				+70	58,65,64	45,56,50	50,50,50
				+30	56,50,40	43,34,46	40,40,40
				-30	33,33,26	21,28,29	30,30,30
				-100	9,7,7	5,4,4	1,1,1
21-1-3	C3065-2	-10/-40	T	+50	33,34,49/41,48,45	30,40,31/40,39,36	30,30,30/40,40,40
			T	+60	43,53,50	37,41,42	40,40,40
			T	+70	56,56,60/51,53,51	50,48,46/46,45,43	50,50,50/50,50,50
			L	+50	63,71,73	56,59,52	60,60,60
			T	+20	36,41,44	31,36,36	30,30,30
			T	+40	44,50,51	36,42,42	40,40,40
			L	+20	63,62,66	51,51,50	40,40,40
			T	+212	91,88,83	84,81,85	99,99,99
				+150	75,77,70	63,68,64	90,90,90
				+100	65,60,65	56,55,56	60,60,60
				+40	56,43,58	44,46,39	40,40,40
				-10	15,23,25	11,18,21	20,20,20
				-100	3,5,5	2,2,3	1,1,1

* ~~This material is also in the reactor vessel surveillance program~~

Limiting plate. See Section 5.3.1.6 for a description of the reactor vessel material surveillance program.

TABLE 5A-4

SURVEILLANCE CAPSULE CONTENTS AND LOCATIONS

Capsule No.	Number of Transverse Charpy Specimens				Number of Specimens Flux Wires	
	Azimuth	Base	HAZ	Weld	Fe	Cu
1	3°	12	12	12	2	2
2	177°	12	12	12	2	2
3	183°	12	12	12	2	2

Note: Surveillance specimen capsule at 3° azimuth location ~~has been~~ was removed for testing ~~to comply with Technical Specification 4.4.6.1.3 requirements.~~ in accordance with the original plant-specific material surveillance program.

APPENDIX 5B

LEAD FACTORS FOR SURVEILLANCE CAPSULES

(ORIGINAL PLANT-SPECIFIC MATERIAL SURVEILLANCE PROGRAM)

CONCERN

During a NRC conference call with NMPC, the NRC indicated that NMPC needed to provide some additional information regarding the lead factors for the Unit 2 surveillance coupon. Additionally, the NRC wanted some information relative to the justification for the lead factors for Unit 2 and their compliance with 10CFR50 Appendix H; whether test results from another reactor could be utilized for Nine Mile Point; and whether there were constraints on relocating the Unit 2 surveillance capsules. These were followed by a letter dated November 16, 1984, which had specific requests. The information below addresses the staff concerns regarding the Unit 2 lead factors.

RESOLUTION

The Unit 2 neutron materials surveillance samples provide a reactor vessel neutron lead factor of 0.29 for the inside surface of the reactor vessel and 0.41 for the 1/4 T position.

There should be no significant temperature difference between the capsule and RPV inner wall. The downcomer fluid flow, during normal operation, is very turbulent and well mixed before it reaches the vessel beltline.

There is no significant neutron spectrum difference between the surveillance material and RPV inner wall. The calculated shift for any energy group above 1.0 MEV is ±2.5 percent max.

Currently, 10CFR50 Appendix H requires that "surveillance specimen capsules must be located near the inside vessel wall in the beltline region so that the radiation history duplicates to the extent practical within the physical constraints of the system, the neutron spectrum, temperature history, and maximum neutron fluence experienced by the reactor vessel inner surface."

For Unit 2, surveillance specimen baskets are located about the core midplane at azimuths (i.e., 3 deg, 177 deg, and 183 deg) that are physically advantageous for specimen withdrawal and yet duplicate as much as possible the neutron spectrum and temperature history of the vessel inner surface. These locations were specifically located to ease removal and, thus, reduce occupational radiation to the technicians removing the sample. Specifically, the holder was located to avoid interferences from the jet pumps, core spray lines, and other reactor vessel internals to ensure that the vessel sample could be removed expeditiously.

Finally, the NRC has previously accepted the current locations of another similar plant previously licensed. This includes a BWR 6 plant which has a lead factor of 0.4.

In conclusion, the current location of the capsule meets the requirements of 10CFR50 Appendix H. ~~However,~~ Unit 2 ~~commits~~ originally committed to supplement the data from Unit 2 with data from other operating BWR 5 251 series vessels. This supplemental data ~~will~~ was to be used to provide a trending estimate for Unit 2. The supplemental data evaluation ~~will~~ was to consider operational history, fluence values, neutron spectrum, and material similarity.

~~Unit 2 will monitor material radiation damage using the Unit 2 capsules and test data from all LaSalle 1 and 2 and WNP-2 capsules. This program is described in the Reactor Vessel Material Surveillance Program submitted to the NRC in a letter dated September 30, 1985.~~

Insert 6

Inserts for NMP2 USAR

Insert 1 (for USAR Page 4.1-13)

Subsequent to the above-described initial and power uprate calculations, reactor vessel neutron fluence has been evaluated using a method in accordance with the recommendations of Regulatory Guide (RG) 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," dated March 2001. Future evaluations of reactor vessel fluence will be completed using a method in accordance with the recommendations of RG 1.190 (as noted in Reference 13). NRC approval of the Unit 2 neutron fluence calculational methodology is documented in Reference 14.

Insert 2 (for USAR Page 4.1-15)

13. [NRC Letter approving Unit 2 participation in the ISP]

14. NRC Letter to NMPNS dated October 27, 2003, "Nine Mile Point Nuclear Station, Unit No. 1 – Issuance of Amendment Re: Pressure-Temperature Limit Curves and Tables (TAC No. MB6687)"

Insert 3 (for USAR Page 4.3-3)

Subsequent to the above-described initial and power uprate evaluations, reactor vessel neutron fluence has been evaluated using a method in accordance with the recommendations of Regulatory Guide (RG) 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," dated March 2001. Future evaluations of reactor vessel fluence will be completed using a method in accordance with the recommendations of RG 1.190, as noted in Section 4.1.4.5.

Inserts for NMP2 USAR (Cont'd)

<u>Insert 4</u> (for USAR Page 5.3-6)

In Reference 6, the NRC approved Unit 2 participation in the BWR Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP), as described in BWRVIP-78 (Reference 3) and BWRVIP-86-A (Reference 4). The NRC approved the ISP for the industry in their safety evaluation dated February 1, 2002 (Reference 5). The ISP meets the requirements of 10 CFR 50, Appendix H. Participation in the ISP replaces the Unit 2 plant-specific vessel material surveillance program.

The current surveillance capsule withdrawal schedule for Unit 2 representative materials is based on the latest NRC-approved version of BWRVIP-86 (Reference 4). No capsules from the Unit 2 vessel are included in the ISP. Capsules from other plants will be removed and specimens will be tested in accordance with the ISP implementation plan. The results from these tests will provide the necessary data to monitor embrittlement of the Unit 2 vessel.

<u>Insert 5</u> (for USAR Page 5.3-20)

3. BWRVIP-78, "BWR Vessel and Internals Project, BWR Integrated Surveillance Program Plan," Final Report, December 1999

4. BWRVIP-86-A, "BWR Vessel and Internals Project, Updated BWR Integrated Surveillance Program (ISP) Implementation Plan," Final Report, October 2002

5. Letter from U. S. NRC to C. Terry (BWRVIP), "Safety Evaluation Regarding EPRI Proprietary Reports 'BWR Vessel and Internals Project, BWR Integrated Surveillance Program Plan (BWRVIP-78)' and 'BWRVIP-86: BWR Vessel and Internals Project, BWR Integrated Surveillance Program Implementation Plan,'" dated February 1, 2002

6. [NRC Letter approving Unit 2 participation in the ISP]

<u>Insert 6</u> (for USAR Page 5B-3)

Subsequent to development of the Unit 2 plant-specific surveillance program, the BWR Vessel and Internals Project (BWRVIP) developed an integrated surveillance program (ISP) to comply with the requirements of 10CFR50 Appendix H. No capsules from the Unit 2 vessel are included in the BWRVIP ISP. Capsules from other plants will be removed and specimens will be tested in accordance with the ISP implementation plan. The results from these tests will provide the necessary data to monitor embrittlement of the Unit 2 vessel. See Section 5.3.1.6 for further description of the BWRVIP ISP.

ATTACHMENT 3

List of Regulatory Commitments

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

REGULATORY COMMITMENT	DUE DATE
None	N/A