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Lycoming, New York 13093

Constellation Energy Group

Nine Mile Point
Nuclear Station

July 31, 2003
NMP1L 1749

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

SUBJECT: Nine Mile Point Unit 1
Docket No. 50-220

Request for Additional Information (RAI) - Amendment Application Re: Pressure-Temperature Limit Curves (TAC Nos. MB6687 and MB6703)

Gentlemen:

By letter NMP1L 1697, dated November 15, 2002, as supplemented by letter NMP1L 1708, dated January 15, 2003, Nine Mile Point Nuclear Station, LLC, (NMPNS) transmitted an application for amendment to the Nine Mile Point Unit 1 (NMP1) Technical Specifications (TSs). The proposed amendment would revise the Reactor Coolant System Pressure-Temperature (P-T) limit curves and associated limit tables specified in TS 3/4.2.2, "Minimum Reactor Vessel Temperature for Pressurization." The amendment application also included a request for exemption from the requirements of 10 CFR 50.60, "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation," since the alternative methodology of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Case N-640 was used to develop the new P-T limit curves and tables.

On April 24, 2003, the NRC staff transmitted by e-mail a list of issues for discussion regarding the proposed TS amendment and supporting Reactor Pressure Vessel (RPV) benchmarking report (included in the January 15, 2003 supplemental submittal) and requested a telephone conference call with the NMPNS staff to discuss the identified issues. The RPV benchmarking report was submitted for NRC review and approval pursuant to Regulatory Guide 1.190 for plant-specific (applicable to NMP1 and Nine Mile Point Unit 2 (NMP2)) qualification of the calculational methodology for determining the RPV neutron fluence values used in the development of the P-T limit curves. As requested, a telephone conference was held on April 30, 2003 between NRC and NMPNS staff representatives. As agreed during the telephone conference, a formal RAI was issued on May 6, 2003 to allow for a docketed response to the issues discussed. Attachment 1 provides the NMPNS responses to the RAI in summary form. The technical report contained in Attachment 2 provides more detailed response information. Note that the applicable section(s) of the report are referenced in the summary responses.

A001

In addition, as indicated in a subsequent telephone conference held on July 30, 2003 with the NRC staff, an update to the RPV benchmarking report will be prepared and submitted to the NRC for review. This revision to the benchmarking report will update the data previously reported in the January 15, 2003 submittal consistent with that reported in the attached NMPNS responses to the RAI. The revised benchmarking report will also include updated core shroud boat sample calculation to measurement (C/M) ratios for NMP1 Operating Cycle 12 since new neutron transport results are now available.

It is our understanding that upon receipt of the attached RAI response the NRC staff will resume review of the amendment application for NMP1 to effectuate our request for approval of the P-T limit curves and associated tables for 28 EFPY. In addition, as discussed in the April 30, 2003 telephone conference, revised (22 EFPY) P-T limit curves for NMP2, which were also developed using the N-640 Code Case, are currently in internal review. The RPV benchmarking report and supplemental information provided in this RAI response support qualification of the neutron fluence methodology (pursuant to Regulatory Guide 1.190) used to develop the P-T limit curves for both NMP1 and NMP2. Submittal of the NMP2 TS amendment application and Code Case relief request, similar to the NMP1 submittal, is scheduled for August 2003. Therefore, NRC staff approval of the benchmarking report and plant-specific (applicable to NMP1 and NMP2) qualification of the fluence methodology will also support approval of the NMP2 amendment application for revised (22 EFPY) P-T limit curves.

The current NMP2 P-T limit curves are valid for 12.8 EFPY and will expire mid-cycle (during Cycle 10) following startup from the Spring 2004 refueling outage (RFO9). Accordingly, approval of the NMP2 P-T curve amendment application and Code Case relief request is needed for RFO9 to support plant heatup and cooldown and system leakage testing.

I declare under penalty of perjury that the foregoing is true and correct. Executed on July 31, 2003.

Sincerely,

Peter E. Katz
Vice President Nine Mile Point

PEK/CDM/bjh

Attachments:
1. Response to Request for Additional Information
2. Report No. MPM-703782

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. John P. Spath, NYSERDA

ATTACHMENT 1

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

On April 30, 2003, the NRC staff held a telephone conference with the Nine Mile Point Nuclear Station, LLC, (NMPNS) staff to discuss previously e-mailed issues regarding the proposed Technical Specification (TS) amendment for the Reactor Coolant System (RCS) Pressure-Temperature (P-T) limit curves and supporting Reactor Pressure Vessel (RPV) benchmarking report. As agreed during the telephone conference, the NRC staff issued a formal Request for Additional Information (RAI) on May 6, 2003 to allow for a docketed response to the issues discussed. The NMPNS responses to the RAI are provided below in summary form. More detailed response information is provided in the technical report contained in Attachment 2. For clarity, the applicable section(s) of the report are referenced in the summary responses.

Note that each identified issue in the RAI is repeated verbatim (italics added), followed by the NMPNS response.

NRC Issue

Benchmarking - The licensee derived the neutron sources using the power distribution and the ORIGEN 2.1 code. It is not clear to the NRC staff why this code was used since fuel composition as a function of burnup should be available from reload reports. Please justify the derivation of the source. For example, calculate a cycle for which you know the source would compare with results using ORIGEN 2.1.

NMPNS Response

Detailed data on the breakdown of fission by isotope was not available at the time previous calculations were performed for Nine Mile Point Unit 1 (NMP1). To address this, the ORIGEN (version 2.1) computer code was used to calculate the fission distribution by isotope as a function of fuel burnup for determination of the fission source.

To validate the use of ORIGEN, NMP1 Cycle 9 was calculated using fission fractions determined both by Casmo-Simulate and by ORIGEN. The ORIGEN calculation resulted in a higher fraction of fissions in Pu, and thus, the normalization of the transport runs for the ORIGEN derived neutron source was higher by about 0.6%. Transport calculations of the neutron flux above 1 MeV at the vessel inner radius at the maximum axial elevation were performed to determine the impact at the vessel. The ORIGEN source produces a calculated fluence rate that is uniformly higher by 1.3%. Thus, the effect of the difference in neutron spectrum is about 0.7%.

Based on these results, it is concluded that the use of ORIGEN to determine the fraction of fissions from fuel isotopes is acceptable in lieu of using plant-specific data from fuel depletion codes. If ORIGEN is used, differences within the range of burnups considered are less than previously estimated uncertainties in results arising from the uncertainties in neutron spectrum and source normalization, and the error is in a conservative direction.

For more detailed response information regarding this issue, refer to Sections 2.0 and 3.0 of the technical report provided in Attachment 2.

NRC Issue

Poolside Critical Assembly (PCA) Benchmark - The arrangement analyzed has a 12-cm gap between the core and the thermal shield and a 13-cm gap between the thermal shield and the vessel. However, the core-to-shroud distances in Unit 1 range from 10 to about 40 cm and the shroud-to-vessel distance is about 40 cm. The arrangement chosen is not representative of a boiling-water reactor geometry. In addition, the test excluded the Rh-103 and U-238 detectors. The licensee's justification was that the Rh-103 detector is not commonly used due to its short half-life; therefore, the cross sections are not well known. Based on NUREG/CR-6115 (ORNL/TM-13205), the NRC staff disagrees with that justification. The basis for rejecting the U-238 dosimeter was due to cross section difficulties. The purpose of benchmarking is to demonstrate the ability of the methodology as an integrated tool to calculate the dosimeter activation. Based on this review, the benchmarking appears not successful because it analyzed the wrong geometry and rejected dosimeters without a reasonable justification.

NMPNS Response

The pool critical assembly (PCA) pressure vessel simulator was constructed to provide a well-characterized geometry that is a mockup of typical reactor geometries. Measurements were made with this simulator arranged in a variety of geometries, including, in some cases, simulated surveillance capsules, but the recommended benchmark consists of a single geometry which includes a 12 cm gap between the reactor core and thermal shield plate and a 13 cm gap between the thermal shield and the vessel simulator. While this geometry is more typical of pressurized water reactor (PWR) geometries than that of boiling water reactors (BWRs), these are the largest water gaps studied in the PCA. It is felt that this PCA geometry comes the closest of any recommended benchmarks (including recommendations both from Regulatory Guide 1.190 and from American Society for Testing and Materials (ASTM) Standard E2006), and thus, is the most applicable for benchmark testing of BWR calculations as applied to the reactor vessel and in-vessel structures. The PCA tests neutron transport calculations both through significant water regions and through steel regions somewhat thicker than BWR vessels. Moreover, the water regions in the PCA are effectively thicker than 12-13 cm, since water in the PCA has a density of about 1 compared to about 0.75 in the BWR bypass and downcomer region. Use of the PCA benchmark is also desirable to supplement measurements in actual reactor geometries because it allows validation of neutron transport calculations in regions where dosimetry measurements cannot normally be made. Thus, it is concluded that use of the PCA benchmark to test BWR calculational

methodology is valuable and satisfies the benchmarking requirement of Regulatory Guide 1.190.

In the previous analysis of the PCA, some difficulty was encountered in the Rh103 and U238 dosimetry comparisons. This was attributed to either poor cross sections (Rh103) or problems with treatment in the threshold region (U238). The latter problem is postulated to arise because the collapse of some cross sections to the 47-group BUGLE96 structure is expected to be spectrum dependent. In this case, use of the same collapsed set, or a few sets (in fact, BUGLE supplies two sets), ignores changes that would occur in the cross sections as the neutron spectrum changes (i.e., for the best accuracy, the dosimetry cross sections should be specifically collapsed for each dosimetry point).

In ASTM Standard E1018, a recommended dosimetry cross section set is specified in 640 energy groups (SAND group structure). BUGLE has 15 energy groups of varying width between 1 and 10 MeV, while SAND has 90 evenly spaced groups. Accordingly, it was decided to use the SAND code to evaluate the PCA dosimetry. The calculation to measurement (C/M) ratios for both the old (BUGLE) results and the new (SAND) results indicate a difference of less than 2% for most of the reactions. Notable changes did occur for the U238 fission reaction and the Rh(n,n') reaction. In the case of the Rh reaction, the cause for the change is clear. The BUGLE cross section for the Rh reaction is derived from ENDF/B-VI and the recommended dosimetry cross section is taken from IRDF-90. The IRDF-90 cross section is based on an evaluation that uses dosimetry results to establish its validity. Therefore, it is not surprising that use of this cross section produces agreement with measurements consistent with the other results.

Results for the U238 fission reaction show significant improvement using the new procedure, but the deviation of the U238 C/M ratios from unity still exceeds that for all the other results. They are, however, very close to the C/M ratios for U238 calculated by Oak Ridge National Laboratory (ORNL). This indicates that bias remains in the U238 measurements or cross section compared to the other dosimetry reactions, but the error is within acceptable limits.

For more detailed response information regarding this issue, refer to Sections 2.0 and 4.0 of the technical report provided in Attachment 2.

NRC Issue

NMP1 Surveillance Capsule at 210-Degree Azimuthal - The NRC staff reviewed the NMP1 210-degree surveillance capsule report (the capsule was removed at the end of cycle 12 at 16.81 EFPYs). This report does not include any information on the methodology of neutronic calculations, and in addition, indicates substantial discrepancies between the copper, iron, and nickel dosimeters. The report recommends the fluence value based on the copper results, yet copper activation represents a small part of the spectrum compared to iron and nickel. Iron and nickel results are also available. Because the neutronic methodology was not presented and because of very large discrepancies in the dosimeter response, the NRC staff finds that the NMP1

210-degree surveillance capsule results do not support the benchmarking of the fluence calculational methodology.

NMPNS Response

The neutron transport calculational methodology used for analysis of the NMP1 surveillance capsule dosimetry, including the 210-degree capsule dosimetry, is described in Section 2.0 of the technical report provided in Attachment 2. The dosimetry analysis results are summarized in the response to the NRC issue below, which required additional analysis of the available NMP1 dosimetry and a re-analysis of the 210-degree capsule dosimetry. As indicated, a comparison of the C/M results shows good agreement, well within the ±20% requirement of Regulatory Guide 1.190 for methods qualification.

For more detailed response information regarding the re-analysis of the 210-degree capsule dosimetry, refer to Section 5.0 of the technical report provided in Attachment 2.

In addressing this issue and the issue below, additional analyses were required, which included supplemental neutron transport calculations for Cycles 9, 10, and 11, the re-analysis of Cycle 7, and the re-analysis of the 5 cases to represent Cycle 12. The analyses also included a review of the core shroud boat sample retrospective dosimetry, the results of which were previously included as part of the plant-specific methods qualification discussed in the RPV benchmarking report submitted to the NRC on January 15, 2003. As indicated in a telephone conference held on July 30, 2003 with the NRC staff, an update to the benchmarking report will be prepared and submitted to the NRC for review. This revision to the benchmarking report will update the data previously reported in the January 15, 2003 submittal consistent with that reported in the NMPNS responses to the RAI included in this attachment and in Attachment 2. The revised benchmarking report will also include updated Cycle 12 core shroud boat sample C/M ratios since new neutron transport results are now available.

For the Cycle 12 core shroud boat samples, the revised benchmarking report will show that the average C/M for all iron and nickel dosimeters improved for the upper shroud sampling location (V9 boat sample), while the calculated nickel response over-predicts the measured value for the lower core shroud location (V10 boat sample). The lower boat sample C/M for nickel is greater than 20%, while the C/M for iron remains within 20%. Review of the boat sample data has led to the conclusion that the end-of-cycle power shift toward the top of the core was significant in Cycle 12, causing a decrease in the flux below core mid-plane during the final 6 months of the cycle. This shift is not captured by the current neutron transport end-of-cycle case used for Cycle 12. Because of its short half-life, nickel is very sensitive to these end-of-cycle power shift effects and these effects are not considered significant relative to predicting the cycle average flux as is demonstrated by the good agreement for the iron dosimeter from the same location. Therefore, the overall C/M for the V10 core shroud boat sample location will exclude (disqualify) the nickel data. Accordingly, the revised benchmarking report will show that all C/M ratios (with V10 boat sample nickel excluded) are within the ±20% requirement of Regulatory Guide 1.190 for methods qualification.

NRC Issue

Surveillance Capsules Used in the Proposed Benchmarking - The licensee removed, tested, and analyzed a total of five surveillance capsules from both Nine Mile Point units. However, only two capsules were used in the submitted benchmarking analysis. As stated in RG 1.190, one of the objectives of benchmarking is to determine potential bias in the calculation of the best-estimate which requires that all of the existing data be used. The licensee ignored the existence of three Nine Mile Point capsules and made no effort to determine the existence of a bias. The benchmarking effort appears incomplete.

Because you requested approval of a plant-specific methodology, the NRC staff recommends that you concentrate your efforts on the Nine Mile Point surveillance capsules and re-think your effort to calculate the PCA in view of the successful outcome of your NUREG-6115 calculation.

NMPNS Response

Surveillance dosimetry removed from NMP1 consists of the following:

- Dosimetry wires attached to a surveillance capsule removed after the first cycle of operation in April 1973.
- Dosimetry wires contained in the 30-degree surveillance capsule removed after Cycle 5 in March 1979.
- Dosimetry wires contained in the 300-degree surveillance capsule removed during a Cycle 7 shutdown in March 1982.
- Dosimetry wires contained in the 210-degree surveillance capsule removed after Cycle 12 in March 1997.

Detailed analyses of the first three dosimeter sets were not made as part of the previous submittal because the amount of data on the earlier fuel cycles was not nearly as extensive as that on Fuel Cycle 12. The detailed data needed for this work has recently been provided by Constellation and these sets have now all been analyzed and comparisons made to provide additional plant-specific benchmark validation for NMP1 and Nine Mile Point Unit 2 (NMP2) calculations. In addition, updated analyses of the Cycle 12 dosimetry were carried out to ensure that all the analyses are consistent and use the latest methodology and most recent information.

To perform the dosimetry analysis, updated models of the NMP1 reactor were constructed. These models include the latest information available on the reactor geometry. Changes include inclusion of plant-specific data for regions above and below the core, and definition of the regions beyond the vessel into the biological shield. Moreover, a more precise definition of the outer core edge and outer row of fuel bundles was incorporated into the model.

The Cycle 1 dosimetry set was located in a tube at the right rear of the 30-degree surveillance capsule. The set consisted of wires which extended vertically and which were centered at axial midplane. Three samples each of copper and iron wires were counted. Nickel counts are not available because of the long delay which occurred between the end of irradiation and counting. Since no specific location information is available on the individual wire samples, they were averaged for each dosimeter type. The average results were then compared with calculated dps/mg for each dosimeter type. Comparisons indicate good agreement between measurement and calculation with an overall average C/M of 0.937.

The dosimetry set for the 30-degree capsule was removed from the reactor in 1979, but the dosimetry was not analyzed until 1984. Thus, no meaningful data on the nickel wires could be obtained. The capsule had 3 packets of Charpy specimens, and each packet had dosimeter wires located at the top of the packet running horizontally across the capsule. It is assumed that the wires are located near to the capsule radial centerline. The axial wire location is determined from the capsule layout, and the resulting locations of the 3 sets of dosimetry wires span about 5 inches centered slightly above axial midplane. The differences in flux between the axial positions are relatively small. Results show very good agreement between calculation and measurement with an overall average C/M of 0.996.

The 300-degree capsule is similar to the 30-degree capsule except that only two Charpy packets are present. In this case, the nickel wires could be counted to supplement the iron and copper data. The results again show very good agreement with a C/M of 1.073.

Re-analysis of the 210-degree capsule, which was removed at the end of Cycle 12, was performed. For this cycle, large changes in power shapes were noted due to changing control rod patterns. To obtain an accurate dosimetry evaluation, Cycle 12 was broken into 5 subcycles and each was analyzed in detail to obtain the relative flux in the capsule. In addition, Cycles 9 through 11 were also analyzed to obtain relative flux over the time span for the most significant dosimeter response. The results indicate that on the average there is very good agreement with the calculation, although there is a consistent trend of C/M with dosimeter half life indicating that the detailed Cycle 12 analysis may still not completely reflect the changing flux levels that occurred during the cycle. However, all of the dosimeters fall well within ±20% of the measurement and an overall C/M of 0.997 was obtained.

All of the calculated reaction rates for the NMP1 dosimetry sets fall well within ±20% of the measurement. Moreover, if the four sets are averaged, the average C/M result is very close to 1.0. This indicates that for the NMP1 plant, the calculations do not exhibit any significant bias. Thus, no bias factor will be applied to the calculations.

For more detailed response information regarding this issue, refer to Sections 2.0 and 5.0 of the technical report provided in Attachment 2.

ATTACHMENT 2

REPORT NO. MPM-703782

Response to NRC Request for Additional Information: Nine Mile Point Unit 1 P-T Limit Curves

July, 2003

MPM
....SERVING CLIENT NEEDS
THROUGH ADVANCED TECHNOLOGY

Report Number MPM-703782

Final Report

entitled

Response to NRC Request for Additional Information: Nine Mile Point Unit 1 P-T Curves

prepared for

Constellation Generation Group
Nine Mile Point Units 1 and 2
Lake Road
Lycoming, NY 13093

by

MPM Technologies, Inc.
2161 Sandy Drive
State College, PA 16803-2283

July, 2003

Preparer	**Checker**
7/31/03	**7/31/03**
Date	**Date**

MPM Approval

7/31/03

Date

Nuclear Quality Assurance Certification

This document certifies that MPM has performed all work under Nine Mile Point Station Purchase Order Number 01-35807-031 in accordance with the requirements of the Purchase Order. All work has been performed under the MPM Nuclear Quality Assurance Program.

M P Manahan, Sr.

M. P. Manahan, Sr.
President
7/31/03

Date

Richard P. Erhard

R. Erhard
QA Manager
7/31/03

Date

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> "Response to NRC Request for Additional Information: Nine Mile Point Unit 1 P-T Curves," Report MPM-703782, MPM Technologies, Inc., 2161 Sandy Drive, State College, PA 16803-2283, July, 2003.

MPM Technologies, Inc.

M. P. Manahan, Sr.
President

Executive Summary

In March 2001, the Nuclear Regulatory Commission (NRC) issued Regulatory Guide (RG) 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." Although specifically developed to address calculation of fluence to the vessel, the guide can be considered to apply to other reactor components such as the shroud or surveillance capsule. One of the requirements of RG 1.190 is the benchmarking of the methodology used in the fluence determination. MPM Report Number MPM-402781 documents calculations performed to qualify the MPM methodology as applied for fluence determination for Nine Mile Point Unit 1 and Unit 2. The NRC has reviewed this report and issued a letter on May 6, 2003 which requests additional information. This report has been prepared to provide the NRC with detailed information in response to the RAI. The results of analyses to address the NRC request are briefly summarized with detailed discussions provided throughout the report.

Fission Source Using ORIGEN 2.1

Detailed data on the breakdown of fission by isotope was not available at the time previous calculations were performed for NMP-1 and NMP-2. To overcome this deficiency, the ORIGEN (version 2.1) computer code was used to calculate the fission distribution by isotope as a function of fuel burnup for determination of the fission source.

To validate the use of ORIGEN, NMP-1 cycle 9 was calculated using fission fractions determined both by Casmo-Simulate and by ORIGEN. The ORIGEN calculation resulted in a higher fraction of fissions in Pu, and thus the normalization of the transport runs for the ORIGEN derived neutron source was higher by about 0.6%. Transport calculations of the neutron flux above 1 MeV at the vessel inner radius at the maximum axial elevation were performed to determine the impact at the vessel. The ORIGEN source produces a calculated fluence rate that is uniformly higher by 1.3%. Thus, the effect of the difference in neutron spectrum is about 0.7%.

Based on these results, it is concluded that the use of ORIGEN to determine the fraction of fissions from fuel isotopes is acceptable in lieu of using plant-specific data from fuel depletion codes. If ORIGEN is used, differences within the range of burnups considered are less than previously estimated uncertainties in results arising from the uncertainties in neutron spectrum and source normalization, and the error is in a conservative direction.

Additional PCA Benchmark Analyses

The pool critical assembly (PCA) pressure vessel simulator was constructed to provide a well-characterized geometry that is a mockup of typical reactor geometries. Measurements were made with this simulator arranged in a variety of geometries, including in some cases simulated surveillance capsules, but the recommended benchmark consists of a single geometry which includes a 12 cm gap between the reactor core and thermal shield plate, and a 13 cm gap between the thermal shield and the vessel simulator. While this geometry is more typical of

PWR geometries than that of BWRs, these are the largest water gaps studied in the PCA. It is felt that this PCA geometry comes the closest of any recommended benchmarks (including recommendations both from RG 1.190 and from ASTM Standard E2006), and thus is the most applicable for benchmark testing of BWR calculations as applied to the reactor vessel and in-vessel structures. The PCA tests neutron transport calculations both through significant water regions and through steel regions somewhat thicker than BWR vessels. Moreover, the water regions in the PCA are effectively thicker than 12-13 cm, since water in the PCA has a density of about 1 compared to about 0.75 in the BWR bypass and downcomer region. Use of the PCA benchmark is also desirable to supplement measurements in actual reactor geometries because it allows validation of neutron transport calculations in regions where dosimetry measurements cannot normally be made. Thus, it is concluded that use of the PCA benchmark to test BWR calculational methodology is valuable and satisfies the benchmarking requirement of RG 1.190.

In the previous analysis of the PCA, some difficulty was encountered in the Rh103 and U238 dosimetry comparisons. This was attributed to either poor cross sections (Rh103) or problems with treatment in the threshold region (U238). The latter problem is postulated to arise because the collapse of some cross sections to the 47-group BUGLE96 structure is expected to be spectrum dependent. In this case, use of the same collapsed set, or a few sets (in fact, BUGLE supplies two sets), ignores changes that would occur in the cross sections as the neutron spectrum changes (i.e., for the best accuracy, the dosimetry cross sections should be specifically collapsed for each dosimetry point).

In ASTM Standard E1018, a recommended dosimetry cross section set is specified in 640 energy groups (SAND group structure). BUGLE has 15 energy groups of varying width between 1 and 10 MeV, while SAND has 90 evenly spaced groups. Accordingly, it was decided to use the SAND code to evaluate the PCA dosimetry. The calculation to measurement (C/M) ratios for both the old (BUGLE) results and the new (SAND) indicate a difference of less than 2% for most of the reactions. Notable changes did occur for the U238 fission reaction and the Rh(n,n') reaction. In the case of the rhodium, the cause for the change is clear. The BUGLE cross section for the rhodium reaction is derived from ENDF/B-VI and the recommended dosimetry cross section is taken from IRDF-90. The IRDF-90 cross section is based on an evaluation that uses dosimetry results to establish its validity. Therefore, it is not surprising that use of this cross section produces agreement with measurements consistent with the other results.

Results for the U238 fission reaction show significant improvement using the new procedure, but the deviation of the U238 C/M ratios from unity still exceeds that for all the other results. They are, however, very close to the C/M ratios for U238 calculated by ORNL. This indicates that bias remains in the U238 measurements or cross section compared to the other dosimetry reactions, but the error is within acceptable limits.

Previous Surveillance Capsules

The NRC has requested that NMP-1 surveillance capsule dosimetry data be analyzed and

reported. Surveillance dosimetry removed from NMP-1 consists of the following:

- Dosimetry wires attached to a surveillance capsule removed after the first cycle of operation in April 1973.
- Dosimetry wires contained in the 30° surveillance capsule removed after cycle 5 in March 1979.
- Dosimetry wires contained in the 300° surveillance capsule removed during a cycle 7 shutdown in March 1982.
- Dosimetry wires contained in the 210° surveillance capsule removed after cycle 12 in March 1997.

Detailed analyses of the first three dosimeter sets were not made as part of the previous submittal because the amount of data on the earlier fuel cycles was not nearly as extensive as that on fuel cycle 12. The detailed data needed for this work has recently been provided by Constellation and these sets have now all been analyzed and comparisons made to provide additional plant-specific benchmark validation for NMP-1 and NMP-2 calculations. In addition, updated analyses of the cycle 12 dosimetry were carried out to ensure that all the analyses are consistent and use the latest methodology and most recent information.

To perform the dosimetry analysis, updated models of the NMP-1 reactor were constructed. These models include the latest information available on the reactor geometry. Changes include utilization of plant-specific data for regions above and below the core, and definition of the regions beyond the vessel into the biological shield. Moreover, a more precise definition of the outer core edge and outer row of fuel bundles was incorporated into the model.

The Cycle 1 dosimetry set was located in a tube at the right rear of the 30 degree surveillance capsule. The set consisted of wires which extended vertically and which were centered at axial midplane. Three samples each of copper and iron wires were counted. Nickel counts are not available because of the long delay which occurred between the end of irradiation and counting. Since no specific location information is available on the individual wire samples, they were averaged for each dosimeter type. The average results were then compared with calculated dps/mg for each dosimeter type. Comparisons indicate good agreement between measurement and calculation with an overall average C/M of 0.937.

The dosimetry from the 30° capsule was removed from the reactor in 1979, but the dosimetry was not analyzed until 1984. Thus, no meaningful data on the nickel wires could be obtained. The capsule had three packets of Charpy specimens, and each packet had dosimeter wires located at the top of the packet running horizontally across the capsule. It is assumed that the wires are located near to the capsule radial centerline. The axial wire location is determined from the capsule layout, and the resulting locations of the three sets of dosimetry wires span about 5 inches centered slightly above axial midplane. The differences in flux between the axial positions are relatively small. Results show very good agreement between calculation and measurement with an overall average C/M of 0.996.

The 300° capsule is similar to the 30° capsule except that only two Charpy packets are

present. In this case, the Ni wires could be counted to supplement the Fe and Cu data. The results again show very good agreement with a C/M of 1.073.

Re-analysis of the 210° capsule, which was removed at the end of cycle 12, was performed. For this cycle, large changes in power shapes were noted due to changing control rod patterns. To obtain an accurate dosimetry evaluation, cycle 12 was broken into 5 subcycles and each was analyzed in detail to obtain the relative flux in the capsule. In addition, cycles 9 through 11 were also analyzed to obtain relative flux over the time span for the most significant dosimeter response. The results indicate that on the average there is very good agreement with the calculation, although there is a consistent trend of C/M with dosimeter half life indicating that the detailed cycle 12 analysis may still not completely reflect the changing flux levels that occurred during the cycle. However, all the dosimeters fall well within ±20% of the measurement and an overall C/M of 0.997 was obtained.

All of the calculated reaction rates for the NMP-1 dosimetry sets fall well within ±20% of the measurement. Moreover, if the four sets are averaged, the average C/M result is very close to 1.0. This indicates that for the NMP-1 plant, the calculations do not exhibit any significant bias. Thus no bias factor will be applied to the calculations.

In summary, it is concluded that the RG 1.190 requirement for qualification of the MPM methodology used for Nine Mile Point Units 1 & 2, by comparisons to measurement and calculational benchmarks, has been fully satisfied.

Contents

1.0 Introduction

In March 2001, the Nuclear Regulatory Commission (NRC) issued Regulatory Guide (RG) 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" [1]. This guide is the final version of two previous draft guides, DG-1053 and DG-1025. The guide was developed to provide state-of-the-art calculational and measurement procedures that are acceptable to the NRC staff for determining pressure vessel fluence. Although specifically developed to address calculation of fluence to the vessel, the guide can be considered to apply to other reactor components such as the shroud and the surveillance capsules.

One of the requirements of RG 1.190 is the benchmarking of the methodology used in the fluence determination. Specifically, RG 1.190 has the following requirement:

> *Methods Qualification. The calculational methodology must be qualified by both (1) comparisons to measurement and calculational benchmarks and (2) an analytic uncertainty analysis. The methods used to calculate the benchmarks must be consistent (to the extent possible) with the methods used to calculate the vessel fluence. The overall calculational bias and uncertainty must be determined by an appropriate combination of the analytic uncertainty analysis and the uncertainty analysis based on the comparisons to the benchmarks.*

Fluences in various reactor components for Nine Mile Point Units 1 and 2 (NMP-1 and NMP- 2) have been evaluated in several reports [2,3,4] prepared prior to the issuance of RG 1.190 in March 2001. In addition, an update of the NMP-2 shroud fluence evaluation has been issued [5,6]. The fluence analyses in these reports were fully compliant with RG 1.190 except for part of the benchmarking requirement for methods qualification. MPM Report Number MPM-402781[7] was prepared and submitted to the NRC to address this requirement of the RG 1.190. The NRC issued an RAI [8] on May 6, 2003 which pertains to report MPM-402781 as part of its review of the NMP-1 P-T curve amendment application. This report has been prepared to document in detail the MPM response to the RAI.

The issues raised by NRC in the Reference [8] RAI are given below:

Benchmarking - The licensee derived the neutron sources using the power distribution and the ORIGEN 2.1 code. It is not clear to the NRC staff why this code was used since fuel composition as a function of burnup should be available from reload reports. Please justify the derivation of the source. For example, calculate a cycle for which you know the source would compare with results using ORIGEN 2.1.

Poolside Critical Assembly (PCA) Benchmark - The arrangement analyzed has a 12-cm gap between the core and the thermal shield and a 13-cm gap between the thermal shield and the vessel. However, the core-to-shroud distances in Unit 1 range from 10 to about 40 cm and the shroud-to-vessel distance is about 40 cm. The arrangement chosen is not representative of a boiling-water reactor geometry. In addition, the test excluded the Rh-103 and U-238 detectors.

The licensee's justification was that the Rh-103 detector is not commonly used due to its short half-life; therefore, the cross sections are not well known. Based on NUREG/CR-6115 (ORNL/TM-13205), the NRC staff disagrees with that justification. The basis for rejecting the U-238 dosimeter was due to cross section difficulties. The purpose of benchmarking is to demonstrate the ability of the methodology as an integrated tool to calculate the dosimeter activation. Based on this review, the benchmarking appears not successful because it analyzed the wrong geometry and rejected dosimeters without a reasonable justification.

NMP1 Surveillance Capsule at 210-Degree Azimuthal - The NRC staff reviewed the NMP1 210-degree surveillance capsule report (the capsule was removed at the end of cycle 12 at 16.81 EFPYs). This report does not include any information on the methodology of neutronic calculations, and in addition, indicates substantial discrepancies between the copper, iron, and nickel dosimeters. The report recommends the fluence value based on the copper results, yet copper activation represents a small part of the spectrum compared to iron and nickel. Iron and nickel results are also available. Because the neutronic methodology was not presented and because of very large discrepancies in the dosimeter response, the NRC staff finds that the NMP1 210-degree surveillance capsule results do not support the benchmarking of the fluence calculational methodology.

Surveillance Capsules Used in the Proposed Benchmarking - The licensee removed, tested, and analyzed a total of five surveillance capsules from both Nine Mile Point units. However, only two capsules were used in the submitted benchmarking analysis. As stated in RG 1.190, one of the objectives of benchmarking is to determine potential bias in the calculation of the best-estimate which requires that all of the existing data be used. The licensee ignored the existence of three Nine Mile Point capsules and made no effort to determine the existence of a bias. The benchmarking effort appears incomplete.

In the report sections which follow, the MPM transport methodology is reviewed and each of the NRC issues are addressed. In addition, since new transport data have been developed to address the RAI issues, data reported for the NMP-1 210 Degree Capsule [9] have been updated and are included here. The updated transport results have been extrapolated to verify that the current best estimate fluences are bounded by the fluences used in the current NMP-1 P-T curve amendment.

2.0 Neutron Flux Calculational Method

2.1 Neutron Transport Model

The neutron exposure of reactor structures is determined by a neutron transport calculation, or a combination of neutron transport calculations, to represent the distribution of neutron flux in three dimensions. The calculation determines the distribution of neutrons of all energies from their source from fission in the core region to their eventual absorption or leakage from the system. The calculation uses a model of the reactor geometry that includes the significant structures and geometrical details necessary to define the neutron environment at locations of interest.

The transport calculations reported here were carried out using the DORT two-dimensional discrete ordinates code [10] and the BUGLE-96 cross-section library [11]. Other codes used included the DOTSOR and DOTSYN codes (available as part of the LEPRICON code package [12]). DOTSOR was used to convert core power distributions from X,Y to R,θ coordinates and place the source in each mesh cell. DOTSYN is used to synthesize the two-dimensional calculations to obtain a calculated neutron spectrum and integrated activities at any desired locations within the modeled reactor geometry. In some cases the ORIGEN 2.1 code [13] was used to calculate the effects of burnup on the neutron source when detailed isotopic fission data was not available. This code was not used for NMP-1 calculations. The computer codes and data libraries were obtained from the Radiation Safety Information Computational Center (RSICC) at Oak Ridge National Laboratory (ORNL). Each code was then compiled on the computer used by MPM for the calculations and a series of test cases were run to verify the code performance. The test cases all agreed within allowable tolerance with established results. This verification was conducted under the MPM Nuclear Quality Assurance Program.

The DORT code is an update of the DOT code which has been in use for this type of problem for many years. It is routinely used and has been used by others for benchmarking calculations [14,15,16,17]. In the analyses, anisotropic scattering was treated with a P_3 expansion of the scattering cross-sections, and the angular discretization was modeled with an S_8 order of angular quadrature. These procedures are in accordance with RG 1.190 and ASTM Standard E-482 [18].

The BUGLE-96 library is a 47 energy group ENDF/B-VI based data set produced specifically for light water reactor shielding and pressure vessel dosimetry applications (an update of the earlier SAILOR library). The energy group boundaries for the 47 groups are given in Table 2-1. This library contains cross-sections collapsed using flux spectra from both BWR and PWR reactor models. Within the core region, cross section sets are collapsed using a PWR core spectrum and a BWR core spectrum. Outside the core, cross sections are produced using a PWR downcomer spectrum, a PWR vessel 1/4 T spectrum, and a concrete spectrum. Reference [19,20] details the data testing of the BUGLE libraries and validation of their applicability for LWR shielding calculations.

As indicated above, the DORT code calculates the neutron transport in two dimensions. In order to estimate the fluence rate in the three-dimensional geometry, the following equation was used to synthesize the flux ϕ for each case of cylindrical geometry:

$$\phi(R,\theta,Z) = \phi(R,\theta) * \phi(R,Z) / \phi(R)$$

In this equation, $\phi(R,\theta)$ is taken from the DORT R,θ calculation (normalized to the power at midplane in the model region), and $\phi(R,Z)$ is from the R,Z calculation normalized to the power in the entire core. A third calculation determined $\phi(R)$ using a one-dimensional cylindrical model normalized at core midplane. The one-dimensional calculation uses the same radial geometry and source distribution as the R,Z calculation at core midplane. In the case of the PCA benchmark, the calculation was carried out in X, Y, Z geometry and a similar synthesis equation was used.

For each calculation, a detailed model of the reactor geometry was developed. The models contain all of the significant reactor structures and use a mesh structure that is fine enough to give good flux convergence. Typically, the radial mesh will be the most critical because of the large flux attenuation of neutrons as they are transported from the core to the vessel region. Most radial mesh intervals outside the core are smaller than 1 cm. This is particularly important in steel structures for calculation of the flux above 1 MeV. Models of large reactor geometries, such as for BWRs, will have 140 to 200 radial mesh points. In the azimuthal direction models have 40 to 80 mesh points to cover an octant of the reactor, depending on the structures to be defined. In the axial direction, models may have as many as 150 mesh points, or more. Modeling smaller reactor geometries, such as the PCA, do not require as many points.

2.2 Compliance with RG 1.190

Regulatory Guide 1.190 covers recommended practices for neutron transport calculations and applies to other reactor components in addition to the primary emphasis on the pressure vessel. The regulatory positions in the guide that pertain to calculational methodology are summarized in Table 2-2 which is taken directly from the guide. The table references paragraphs in the guide that give more detailed information on each position. The compliance of the MPM calculational methodology with the guide is summarized below.

Fluence Determination: All calculations were performed using an absolute fluence calculation. Meets guide requirement.

Modeling Data: All the data used in the models are documented and verified. Meets guide requirement.

Nuclear Data: The calculations use the BUGLE-96 cross section set which is based on the latest version (VI) of the Evaluated Nuclear Data File (ENDF/B). The BUGLE-96 set has undergone extensive testing and benchmarking to ensure its validity for LWR calculations.

Meets guide requirement.

Cross-Section Angular Representation: The calculations use a P3 angular expansion in accordance with the guide.
Meets guide requirement.

Cross-Section Group Collapsing: The calculations use the BUGLE-96 library without additional collapsing. Benchmarking has shown that the 47 group structure is adequate for LWR neutron transport calculations.
Meets guide requirement.

Neutron Source: Isotopic variation is accounted for in the neutron spectrum, neutrons per fission, and energy per fission within the modeling limitations. Moderator density is included in detail.
Meets guide requirement.

End-of-Life Predictions: No fluence projections are made in this benchmarking effort. Fluence projections for plant analyses use best-estimate fuel loadings.
Meets guide requirement.

Spatial Representation: Mesh intervals are adopted to ensure that flux changes within intervals are small enough to allow for accurate results. Radial intervals in the outer core region and in the region between the core and the outside of the vessel are generally about 1 cm except near boundaries where a finer mesh is used in some cases. Inside the core, where flux changes are small, larger intervals are used. In the azimuthal direction, between 40 and 80 meshes are used, depending on the complexity of structures to be modeled. In the axial direction, a coarse mesh is acceptable in regions where the flux changes slowly. Finer meshing is used near boundaries. The quadrature used was S8.
Meets guide requirement.

Multiple Transport Calculations: It was not necessary to use bootstrapping for these calculations so this requirement does not apply.

Point Estimates: This requirement only applies to Monte Carlo calculations which are not used here.

Statistical Tests: This requirement only applies to Monte Carlo calculations which are not used here.

Variance Reduction: This requirement only applies to Monte Carlo calculations which are not used here.

Spectral Effects on RT_{NDT}: This requirement only applies to extrapolation through the vessel and does not affect the benchmark calculations. However, when fluence within the vessel is required, the displacement per atom (dpa) methodology is applied to vessel calculations as

specified in RG 1.99, Revision 2 [21] (see, for example, Reference [3]).
Meets guide requirement.

Cavity Calculations: With the exception of the one dosimetry measurement at the rear of the PCA vessel, no cavity results have been applied for this benchmarking effort. In the event that cavity dosimetry measurements are analyzed in the future, it will be necessary to ensure that the quadrature is adequate. Utilization of cavity flux calculations is not anticipated at this time.
Meets guide requirement.

Methods Qualification: These calculations and comparisons provide the required methods qualification. This includes verification of vessel fluence calculations using the PCA simulator measurements, the BWR calculational benchmark, and comparisons with plant specific BWR measurements. No uncertainty analysis was performed for the PCA calculation or for the BWR calculational benchmark since no uncertainty data is given with these problems. Comparisons with measurements or with standard results provide validation of the accuracy of the calculations. A complete analytical uncertainty analysis was carried out in accordance with the guide for the NMP-1 and NMP-2 calculations. This uncertainty analysis indicated that calculational errors for vessel fluences in the beltline region were about 15%, well within the 20% accuracy requirement specified by RG 1.190.
Meets guide requirement.

Fluence Calculational Uncertainty: An extensive evaluation of all contributors to the uncertainty in the calculated fluence was made for the NMP-1 and NMP-2 calculations. This evaluation indicated that the uncertainty in calculated fluences in the reactor beltline region is below 20% as specified in the guide. In addition, the comparisons with measurements indicate agreement well within the 20% limit. Thus, fluence evaluations using the present methodology applied to NMP-1 and NMP-2 will use the calculated results with no bias applied.
Meets guide requirement.

Table 2-1 Neutron Energy Group Structure – 47 Groups.

Energy Group	Upper Energy (MeV)	Energy Group	Upper Energy (MeV)
1	1.733E+01	25	2.972E-01
2	1.419E+01	26	1.832E-01
3	1.221E+01	27	1.111E-01
4	1.000E+01	28	6.738E-02
5	8.607E+00	29	4.087E-02
6	7.408E+00	30	3.183E-02
7	6.065E+00	31	2.606E-02
8	4.966E+00	32	2.418E-02
9	3.679E+00	33	2.188E-02
10	3.012E+00	34	1.503E-02
11	2.725E+00	35	7.102E-03
12	2.466E+00	36	3.355E-03
13	2.365E+00	37	1.585E-03
14	2.346E+00	38	4.540E-04
15	2.231E+00	39	2.145E-04
16	1.920E+00	40	1.013E-04
17	1.653E+00	41	3.727E-05
18	1.353E+00	42	1.068E-05
19	1.003E+00	43	5.044E-06
20	8.208E-01	44	1.855E-06
21	7.427E-01	45	8.764E-07
22	6.081E-01	46	4.140E-07
23	4.979E-01	47	1.000E-07
24	3.688E-01		1.000E-11

Table 2-2 Summary of Regulatory Positions on Fluence Calculation Methods [1].

	Regulatory Position
Fluence Determination. Absolute fluence calculations, rather than extrapolated fluence measurements, must be used for the fluence determination.	1.3
Modeling Data. The calculation modeling (geometry, materials, etc.) should be based on documented and verified plant-specific data.	1.1.1
Nuclear Data. The latest version of the Evaluated Nuclear Data File (ENDF/B) should be used for determining nuclear cross- sections. Cross-section sets based on earlier or equivalent nuclear-data sets that have been thoroughly benchmarked are also acceptable. When the recommended cross-section data change, the effect of these changes on the licensee-specific methodology must be evaluated and the fluence estimates updated when the effects are significant.	1.1.2
Cross-Section Angular Representation. In discrete ordinates transport calculations, a P_3 angular decomposition of the scattering cross-sections (at a minimum) must be employed.	1.1.2
Cross-Section Group Collapsing. The adequacy of the collapsed job library must be demonstrated by comparing calculations for a representative configuration performed with both the master library and the job library.	1.1.2
Neutron Source. The core neutron source should account for local fuel isotopics and, where appropriate, moderator density. The neutron source normalization and energy dependence must account for the fuel exposure dependence of the fission spectra, the number of neutrons produced per fission, and the energy released per fission.	1.2
End-of-Life Predictions. Predictions of the vessel end-of-life fluence should be made with a best-estimate or conservative generic power distribution. If a best estimate is used, the power distribution must be updated if changes in core loadings, surveillance measurements, or other information indicate a significant change in projected fluence values.	1.2
Spatial Representation. Discrete ordinates neutron transport calculations should incorporate a detailed radial- and azimuthal-spatial mesh of ~2 intervals per inch radially. The discrete ordinates calculations must employ (at a minimum) an S_8 quadrature and (at least) 40-80 intervals per octant.	1.3.1
Multiple Transport Calculations. If the calculation is performed using two or more "bootstrap" calculations, the adequacy of the overlap regions must be demonstrated.	1.3.1

Table 2-2 Summary of Regulatory Positions on Fluence Calculation Methods [1] (Continued).

	Regulatory Position
Point Estimates. If the dimensions of the tally region or the definition of the average-flux region introduce a bias in the talley edit, the Monte Carlo prediction should be adjusted to eliminate the calculational bias. The average-flux region surrounding the point location should not include material boundaries or be located near reflecting, periodic or white boundaries.	1.3.2
Statistical Tests. The Monte Carlo estimated mean and relative error should be tested and satisfy all statistical criteria.	1.3.2
Variance Reduction. All variance reduction methods should be qualified by comparison with calculations performed without variance reduction.	1.3.2
Capsule Modeling. The capsule fluence is extremely sensitive to the geometrical representation of the capsule geometry and internal water region, and the adequacy of the capsule representation and mesh must be demonstrated	1.3.3
Spectral Effects on RT_{NDT}. In order to account for the neutron spectrum dependence of RT_{NDT}, when it is extrapolated from the inside surface of the pressure vessel to the T/4 and 3T/4 vessel locations using the > 1-MeV fluence, a spectral lead factor must be applied to the fluence for the calculation of ΔRT_{NDT}.	1.3.3
Cavity Calculations. In discrete ordinates transport-calculations, the adequacy of the S_8 angular quadrature used in cavity transport calculations must be demonstrated.	1.3.5
Methods Qualification. The calculational methodology must be qualified by both (1) comparisons to measurement and calculational benchmarks and (2) an analytic uncertainty analysis. The methods used to calculate the benchmarks must be consistent (to the extent possible) with the methods used to calculate the vessel fluence. The overall calculational bias and uncertainty must be determined by an appropriate combination of the analytic uncertainty analysis and the uncertainty analysis based on the comparisons to the benchmarks.	1.4.1, 1.4.2, 1.4.3
Fluence Calculational Uncertainty. The vessel fluence (1 sigma) calculational uncertainty must be demonstrated to be 20% for RT_{PTS} and RT_{NDT} determination. In these applications, if the benchmark comparisons indicate differences greater than ~20%, the calculational model must be adjusted or a correction must be applied to reduce the difference between the fluence prediction and the upper 1-sigma limit to within 20%. For other applications, the accuracy should be determined using the approach described in Regulatory Position 1.4, and an uncertainty allowance should be included in the fluence estimate as appropriate in the specific application.	1, 1.4.3

3.0 Fission Source Using ORIGEN 2.1

Detailed data on the breakdown of fission by isotope was not available at the time the previous calculations were performed for NMP-1. To overcome this deficiency, the ORIGEN (version 2.1) computer code [13] was used to calculate the fission distribution by isotope as a function of fuel burnup for use in the determination of the fission source. This application of the ORIGEN code was tested by comparisons of heavy element amounts with those from fuel depletion codes for other plants to verify that the ORIGEN output as a function of burnup is reasonable. However, while no previous comparison of the transport results bases on ORIGEN data with data from plant-specific core follow calculations was made in the past, the impact of using the ORIGEN code was included in the uncertainty analysis. The comparison calculations have now been made and are reported in this section to justify the use of the ORIGEN code for this purpose. Constellation is currently using Studvik's Casmo-Simulate code [22] to provide core follow output, including isotopic data as a function of burnup.

ORIGEN uses a matrix exponential method to solve a large system of coupled, linear, first-order ordinary differential equations with constant coefficients in order to calculate the buildup, decay, and processing of radioactive materials. Supplied with the code are libraries for standard and extended burnup calculations [23]. Since ORIGEN uses a single cross section for each reaction at each time step, cross sections must be collapsed for the specific geometry being calculated. ORIGEN does vary cross sections with fuel burnup, but such effects as void fraction and other variables that can affect reaction rates are not taken into account. Comparisons with isotopic distributions from fuel depletion codes for burnup of BWR fuel in the typical ranges encountered in the outer fuel bundles in most BWR plants indicated that the best results were obtained with the BWR extended burnup library. This library was therefore selected for all BWR applications made to date.

Fuel encountered in BWR applications is initially uranium enriched to a few percent U235. As burnup occurs, U235 is removed and Pu239 is produced. Gradually more and more of the fission events occur in Pu239 and the higher plutonium isotopes. This has two effects on transport calculations. First, fissions in plutonium have a slightly more energetic fission spectrum, and thus a higher fraction of neutrons in the higher energy groups are produced. Differences in transport due to the change in spectrum will vary, depending on distance from the source. Second, both the neutrons per fission and the energy per fission increase with burnup, with the neutrons per fission increasing faster. Since the normalization of the transport calculation is proportional to the number of neutrons and inversely proportional to the energy per fission (since the calculation produces the neutron flux at a given reactor power), the high energy neutron flux increases with fuel burnup.

To validate the use of ORIGEN, NMP-1 cycle 9 was calculated using fission fractions determined both by Casmo-Simulate and by ORIGEN. For this case, the fuel in the outer two rows of bundles had an initial average enrichment of 2.77% and an average burnup in the outer rows of about 22000 MWd/MTU. The ORIGEN calculation resulted in a higher fraction of fissions in Pu for this burnup and thus the normalization of the transport runs for the ORIGEN

derived neutron source was higher by about 0.6%. Results for the two calculations are shown in Figure 3-1 which presents the neutron flux above 1 MeV at the vessel inner radius at the maximum axial elevation. It is observed that the curves are in close agreement, with the ORIGEN source producing a calculated fluence rate that is uniformly higher by 1.3%. Thus the effect of the difference in neutron spectrum is about 0.7% when the 0.6% difference in normalization is subtracted.

The 1.3% difference may be compared with the estimated uncertainty in vessel fluence due to the spectrum and normalization uncertainty previously evaluated for NMP-1 in Reference [2]. This estimate was 2.9% for the vessel inner radius maximum fluence rate. Thus, the difference between ORIGEN and Casmo-Simulate sources falls well within the previously estimated uncertainty. Moreover, it should be noted that the ORIGEN result differs in a conservative direction, i.e. use of the ORIGEN data produce a higher calculated exposure.

Based on these results, it is concluded that the use of ORIGEN to determine the fraction of fissions from fuel isotopes is acceptable in lieu of using plant-specific data from fuel depletion codes. If ORIGEN is used, differences within the range of burnups considered are less than estimated uncertainties in results arising from the uncertainties in neutron spectrum and source normalization, and the error is in a conservative direction.

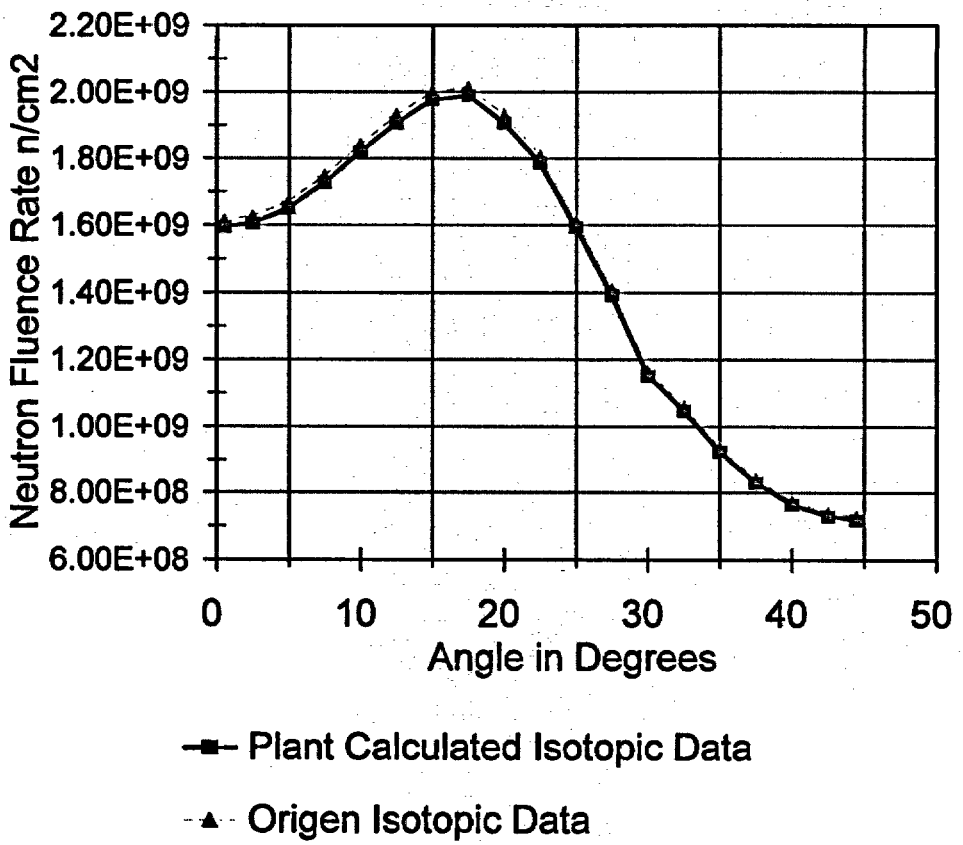

Figure 3-1 Average Fluence Rate for NMP-1 for Cycle 9 as a Function of Azimuthal Angle at the Maximum Axial Height Calculated Using Neutron Source Data from ORIGEN and from Casmo-Simulate

4.0 Additional Analyses for PCA Benchmark

The pool critical assembly (PCA) pressure vessel simulator was constructed to provide a well-characterized geometry that is a mockup of typical reactor geometries. Measurements were made with this simulator arranged in a variety of geometries, including in some cases simulated surveillance capsules, but the recommended benchmark described in Reference [14] consists of a single geometry which includes a 12 cm gap between the reactor core and thermal shield plate, and a 13 cm gap between the thermal shield and the vessel simulator. While this geometry is more typical of PWR geometries than that of BWRs, these are the largest water gaps studied in the PCA. It is felt that this PCA geometry comes the closest of any recommended benchmarks (including recommendations both from RG 1.190 and from ASTM Standard E2006) and thus is the most applicable for benchmark testing of BWR calculations as applied to the reactor vessel and in-vessel structures. The PCA tests neutron transport calculations both through significant water regions and through steel regions somewhat thicker than BWR vessels. Moreover, the water regions in the PCA are effectively thicker than 12-13 cm since water in the PCA has a density of about 1 compared to about 0.75 in the BWR bypass and downcomer region. Use of the PCA benchmark is also desirable to supplement measurements in actual reactor geometries because it allows validation of neutron transport calculations in regions where dosimetry measurements cannot normally be made. Thus it is concluded that use of the PCA benchmark to test BWR calculational methodology is valuable and satisfies the benchmarking requirement of RG 1.190.

In the previous analysis of the PCA, some difficulty was encountered in the dosimetry comparisons. This was attributed to either poor cross sections (Rh103) or problems with treatment in the threshold region (U238). The latter problem is postulated to arise because the collapse of some cross sections to the 47-group BUGLE96 structure may be quite spectrum dependent. In this case, use of the same collapsed set or a few sets (in fact, BUGLE supplies two sets) ignores changes that would occur in the cross sections as the neutron spectrum changes (i.e. for the best accuracy the dosimetry cross sections should be specifically collapsed for each dosimetry point).

In ASTM Standard E1018, a recommended dosimetry cross section set is specified in 640 energy groups (SAND group structure). In particular, BUGLE has 15 energy groups of varying width between 1 and 10 MeV, while SAND has 90 evenly spaced groups. Accordingly, it was decided to use the SANDII code [24] to evaluate the PCA dosimetry. While SANDII is a code developed to perform spectrum adjustment, no adjustment is being made here. The code is used merely to calculate dosimeter activities using the recommended dosimetry library. To carry this out, the assumption is made that the actual neutron spectrum in the energy region above about 0.1 MeV is fairly smooth. This assumption is justified because the effect of resonances in this region is small. Using this assumption, the SANDII code expands the 47-group calculated neutron spectrum into the SAND finer group structure using a spline procedure to fit the integral fluence rate. The expanded spectrum can then be multiplied by the 640 group dosimetry cross sections to give the reaction rate. This process minimizes the effect of the rapidly varying dosimetry cross section in the threshold region for all the threshold dosimetry reactions. This is

particularly important when the threshold region accounts for a large part of the dosimeter response.

Results using the SAND procedure are presented in Table 4-1. This table contains the updated calculated values for the equivalent fission fluxes using the SAND procedure with the same transport results as the earlier calculation. The calculation to measurement (C/M) ratios for both the old (BUGLE) results and the new (SAND) results are shown. For most of the reactions, the difference is less than 2%. Notable changes did occur for the U238 fission reaction and the Rh(n,n') reaction. In the case of the rhodium, the cause for the change is clear. The BUGLE cross section for the rhodium reaction is derived from ENDF/B-VI and the recommended dosimetry cross section is taken from IRDF-90 [25]. The IRDF-90 cross section is based on an evaluation that uses dosimetry results to establish its validity. Therefore, it is not surprising that use of this cross section produces agreement with measurement consistent with the other results.

Results for the U238 fission reaction show significant improvement using the new procedure, but the deviation of the U238 C/M ratios from unity still exceeds that for all the other results. They are, however, very close to the C/M ratios for U238 calculated by Remec [14]. This indicates that bias remains in the U238 measurements or cross section compared to the other dosimetry reactions, but the error is within acceptable limits. The change from the BUGLE to SAND results is about 8% which is about 4 times the change for the other reactions when similarly derived cross sections are used. This may indicate a problem with the collapsing of the BUGLE cross section for this reaction, but it is sufficient here to conclude that the SANDII procedure using the recommended U238(n,f) cross section produces results within acceptable bounds and consistent with the benchmark document [14].

Use of the SANDII procedure for BWR dosimetry applications has also been tested for the dosimetry reactions of Ni(n,p), Fe(n,p), and Cu(n,α). Results for these reactions all indicate small changes between the BUGLE and SAND results similar to those observed in the PCA for dosimeters other than Rh and U238. It is intended to apply the SAND procedure for BWR dosimetry analysis in the future in order to obtain the most consistent results.

A summary of the C/M ratios for each PCA location is presented in Table 4-2. It is seen that the C/M ratios are very consistent among the positions with the calculation consistently low by an average of 6%. There is no obvious trend to the bias in going from the location nearest the core to the one at the back of the vessel. Table 4-2 also contains C/M ratios calculated by Remec [14] using BUGLE-93. The C/M ratios are seen to be very consistent except at the A7 position which only has the Np237 reaction. Results calculated using BUGLE-96 are also reported in Reference [16]. Results in this reference using the synthesis approach show a slight increase in bias going through the pressure vessel, and a three-dimensional calculation was made which eliminated this bias. The latter reference did not use the BUGLE-96 dosimetry cross sections and also made comparisons with a slightly different set of measured data. The fact that the three-dimensional calculation eliminated some of the bias illustrates that the synthesis method may contribute part of the 6% bias observed. This is not unexpected since the PCA is not a large enough mockup to completely eliminate effects arising from outside the symmetric

model region. Such 3-D effects will be much smaller in the beltline region of a power reactor where axial dimensions are much larger.

On the basis of the above discussion, the following conclusions are drawn:

- The PCA benchmark is applicable for testing BWR calculational methodologies because it contains similar materials and neutron transport distances as compared with BWRs. The PCA neutron path distances are large enough to confirm the adequacy of the methodology and transport cross sections for BWR applications.
- Successful calculation of the PCA as demonstrated by agreement with other accepted calculations and by agreement with measurements within acceptable tolerance has been achieved. This satisfies the RG1.190 requirement for benchmarking against a measurement benchmark.
- A consistent bias (averaging 6%) was observed in the PCA calculations and this is in agreement with other calculations using the synthesis method. This bias is not likely to be present in calculations of operating plants because of geometry differences.
- Discrepancies in C/M ratios observed using the BUGLE 47-group dosimetry cross sections can be lessened or eliminated by using the dosimetry cross sections contained in the ASTM standard cross section data file.

Table 4-1 Updated Comparison of Calculated and Measured Results for PCA.

	Equivalent Fission Flux (n/cm^2-s)		C/M	
Reaction	Measured Results	New Calculated Results	Old	New
Location A1				
Np237(n,f)	6.64E-06	6.05E-06	0.891	0.911
U238(n,f)	note 1	5.21E-06	note 1	note 1
Rh103(n,n')	5.54E-06	5.37E-06	1.109	0.969
In115(n,n')	5.61E-06	5.17E-06	0.921	0.921
Ni58(n,p)	5.83E-06	5.47E-06	0.937	0.939
Al27(n,α)	7.87E-06	7.28E-06	0.920	0.925
Location A2				
Np237(n,f)	note 1	6.86E-07	note 1	note 1
U238(n,f)	note 1	5.65E-07	note 1	note 1
Rh103(n,n')	note 1	6.17E-07	note 1	note 1
In115(n,n')	6.06E-07	5.64E-07	0.929	0.931
Ni58(n,p)	6.18E-07	5.88E-07	0.949	0.952
Al27(n,α)	1.02E-06	9.47E-07	0.919	0.928
Location A3				
Np237(n,f)	2.27E-07	2.25E-07	0.970	0.992
U238(n,f)	note 1	2.00E-07	note 1	note 1
Rh103(n,n')	note 1	2.01E-07	note 1	note 1
In115(n,n')	1.99E-07	1.94E-07	0.973	0.973
Ni58(n,p)	2.31E-07	2.27E-07	0.977	0.982
Al27(n,α)	4.48E-07	4.35E-07	0.959	0.971
Location A4				
Np237(n,f)	9.27E-08	8.50E-08	0.916	0.916
U238(n,f)	6.11E-08	5.48E-08	0.814	0.897
Rh103(n,n')	7.74E-08	7.75E-08	1.844	1.002
In115(n,n')	5.87E-08	5.80E-08	0.984	0.989
Ni58(n,p)	5.3E-08	5.06E-08	0.951	0.954
Al27(n,α)	1.02E-07	9.71E-08	0.938	0.952
Location A5				
Np237(n,f)	5.18E-08	4.69E-08	0.906	0.906
U238(n,f)	2.74E-08	2.36E-08	0.778	0.863
Rh103(n,n')	4.35E-08	4.18E-08	2.158	0.960
In115(n,n')	2.76E-08	2.65E-08	0.955	0.961
Ni58(n,p)	2.09E-08	1.99E-08	0.948	0.950
Al27(n,α)	4.1E-08	3.91E-08	0.937	0.953

Table 4-1 Updated Comparison of Calculated and Measured Results for PCA (Continued).

	Equivalent Fission Flux (n/cm^2-s)		C/M	
Reaction	Measured Results	New Calculated Results	Old	New
Location A6				
Np237(n,f)	2.70E-08	2.38E-08	0.879	0.881
U238(n,f)	1.12E-08	9.49E-09	0.764	0.847
Rh103(n,n')	2.19E-08	2.08E-08	2.454	0.950
In115(n,n')	1.17E-08	1.13E-08	0.962	0.969
Ni58(n,p)	7.43E-09	7.35E-09	0.987	0.989
Al27(n,α)	1.54E-08	1.48E-08	0.946	0.964
Location A7				
Np237(n,f)	7.25E-09	6.82E-09	0.926	0.941
U238(n,f)	note 1	2.56E-09	note 1	note 1
Rh103(n,n')	note 1	5.85E-09	note 1	note 1
In115(n,n')	note 1	3.09E-09	note 1	note 1
Ni58(n,p)	note 1	2.05E-09	note 1	note 1
Al27(n,α)	note 1	5.05E-09	note 1	note 1

Note 1: Reference 14 does not report measurements at these locations.

Table 4-2 Updated Comparison of Calculated and Measured Results for PCA.

PCA Location	MPM Average C/M Ratio [a]	ORNL Average C/M Ratio [b]
A1	0.933	0.93
A2	0.937	0.92
A3	0.980	0.96
A4	0.952	0.95
A5	0.932	0.92
A6	0.933	0.91
A7	0.941	0.84
Overall Average	0.943	0.93

a. Results from this work including all measurements.
b. Results from Reference [14]. Np results are excluded from the averages at the A1 and A3 positions.

5.0 NMP-1 Dosimetry Analyses

The NRC has requested that NMP-1 surveillance capsule dosimetry be analyzed and reported. This section of the report provides the results from the requested analyses. Surveillance dosimetry removed from NMP-1 consists of the following:

- Dosimetry wires attached to a surveillance capsule removed after the first cycle of operation in April 1973.
- Dosimetry wires contained in the 30° surveillance capsule removed after cycle 5 in March 1979.
- Dosimetry wires contained in the 300° surveillance capsule removed during a cycle 7 shutdown in March 1982.
- Dosimetry wires contained in the 210° surveillance capsule removed after cycle 12 in March 1997.

Detailed analyses of the first three sets on this list were not made as part of the previous submittal. These sets were not analyzed in detail in the past because the amount of data on the earlier fuel cycles is not nearly as extensive as that on fuel cycle 12. The detailed data needed for this work has been provided by Constellation and these sets have now all been analyzed and comparisons made to provide additional plant specific benchmark validation for NMP calculations. In addition, updated analyses of the cycle 12 dosimetry were carried out to ensure that all the analyses are consistent and use the latest methodology and most recent information.

To perform the dosimetry analysis, updated models of the NMP-1 reactor were constructed. These models include the latest information available on the reactor geometry. Changes from earlier models include the use of plant specific data on regions above and below the core and definition of the regions beyond the vessel into the biological shield. Moreover, a more precise definition of the outer core edge and outer row of fuel bundles was incorporated into the model.

The R-θ model is shown in Figure 5-1. The model included 176 mesh points in the radial direction covering the range from the center of the core to about 12 inches into the concrete biological shield. In the azimuthal direction, 66 mesh points were used to model a single octant of the reactor. Inspection of the fuel loading patterns indicated that only minor deviations from an octant symmetry were present and thus use of a single octant in the model was justified. The core region used a homogenized material distribution which includes the fuel, fuel cladding, and the water. The water region in the fuel was divided into two parts: one with the water in the fuel bundle outside the pins, and one that included the water in the water pins and water in the channels outside the fuel bundle. The latter water was assumed to be all at the base water density with no steam fraction. The other water volume contains both liquid water and steam. The fraction occupied by steam is known as the void fraction and varies by fuel bundle and by axial position within the fuel bundle.

Inspection of the void fraction values indicated that significant variation in the void fraction occurred, but that some groups of neighboring fuel bundles had close to the same void fraction distribution. To model the void fraction variation, the outer row of fuel bundles was divided into six regions of approximately uniform water material density, and the average water density for the assemblies in each of these regions was calculated by multiplying the base water density by 1.0 minus the void fraction. An additional 5 regions contain fuel bundles in the next to outer row. The third to outer row is represented by a single region, and a final region encompasses the remainder of the core.

Since the NMP-1 BWR has large water channels between some of the fuel bundles in order to accommodate the control rods, the core model explicitly accounted for the water channel at the inside and outside of the outer row of fuel bundles. The volume fractions of the outer two rows of fuel bundles were adjusted from the average for the homogenized core to exclude these regions. The water radially outside the outer row of bundles is included in the bypass region, and the water inside this row is included as a water channel.

For the R,Z model, the core was divided into 4 radial regions. Three of these regions consisted of each of the outer three rows of assemblies averaged over the octant. The fourth region consisted of the inner part of the core. The neutron source in each of these regions was calculated using a radial source averaged over the octant together with an average axial power shape for each region. The axial power distribution was supplied for each assembly in 24 or 25 nodes, each representing about 6 inches of core height.

Each of the four radial region was also divided into axial regions according to variation in void fraction. The void fraction was also given for each assembly in 24 or 25 axial nodes. Except for nodes near the bottom of the core which had zero void fraction, each node was modeled as a separate region for the calculation. This resulted in a total of 88 (or 92 for the 25 node case) varying regions in the core, each with a distinct cross section set. For the R-Z model, the core radius was taken to be that which gave the equivalent core volume. Regions above and below the core were modeled with smeared regions representing water volumes and stainless steel and zirconium regions. The model had 176 mesh points in the radial direction identical to the R-θ model. In the axial direction, the model had 139 (or 142) mesh points with 91 (94) in the core region.

Analysis of the dosimetry removed from the reactor before the end of cycle 7 was analyzed using the cycle 7 fuel power distribution and axial void fraction distribution in the three dimensions. This cycle was chosen to be most representative of the early operation of NMP-1 and the choice was necessary because information to define the power distribution and void fraction details for the earlier cycles is not available. Use of the approximation that cycles 1 to 6 are identical to cycle 7 produces additional uncertainty in the calculation of dosimetry results. Based on information available for these earlier cycles and on variations in NMP-1 and other BWR plants, this uncertainty is estimated to be 9% (1σ).

Results for the cycle 1 dosimetry are presented in Table 5-1. This dosimetry set was located in a tube at the right rear of a dosimetry capsule. The set consisted of wires which

extended vertically and which were centered at axial midplane. Three samples each of copper and iron wires were counted. Nickel counts are not available because of the long delay which occurred between the end of irradiation and counting. The results shown in Table 5-1 have been decay corrected to the end-of-irradiation date of April 14, 1973.

Since no specific location information is available on the individual wire samples, they were averaged for each dosimeter type. The average results were then compared with calculated dps/mg for each. Comparisons indicate good agreement between measurement and calculation with an overall average C/M of 0.937.

The dosimetry results for the 30° capsule are presented in Table 5-2. The capsule was removed from the reactor in 1979, but the dosimetry was not analyzed until 1984. Thus no meaningful data on the nickel wires could be obtained. The capsule had 3 packets of Charpy specimens and each packet had dosimeter wires located at the top of the packet running horizontally across the capsule. It is assumed that the wires are located near to the capsule radial centerline. The axial wire location is determined from the capsule layout, and the resulting locations of the 3 sets span about 5 inches centered slightly above axial midplane. As can be seen from the calculated results in Table 5-2, the differences in flux between the axial positions are relatively small.

Table 5-2 contains results for 3 Fe dosimeters and 2 Cu dosimeters. The P1 packet did not contain a copper wire. The measured results in the table are decay corrected to an end-of-irradiation date of March 3, 1979. Results show very good agreement between calculation and measurement with an overall average C/M of 0.996.

The 300° capsule is similar to the 30° capsule except that only two Charpy packets are present. In this case, the Ni wires could be counted to supplement the Fe and Cu data. Results are shown in Table 5-3 which have been decay corrected to March 19, 1982. The results again show very good agreement for the Cu monitor, but deviations increase for the Fe and Ni monitors. This may be due, in part, to the fact that the Ni monitors are almost entirely dependent on the flux level at the beginning of fuel cycle 7, as compared to the Cu which integrates over past cycles. Since the calculation uses conditions at the middle of cycle 7 to represent an average over the cycle, it is not surprising that part of the cycle may show deviations in flux level from the average.

Results from the 210° capsule removed at the end of cycle 12 are shown in Table 5-4. For this cycle, large changes in power shapes were noted due to changing control rod patterns. To get an accurate dosimetry evaluation, cycle 12 was broken into 5 subcycles and each was analyzed in detail to obtain the relative flux in the capsule. In addition, cycles 9 through 11 were also analyzed to obtain relative flux over the time span for the most significant dosimeter response. The results indicate that on the average there is very good agreement with the calculation, although there is a consistent trend of C/M with dosimeter half life indicating that the detailed cycle 12 analysis may still not completely reflect the changing flux levels that occurred during the cycle. However, all the dosimeters fall well within $\pm$20% of the measurement.

All of the calculated reaction rates for the NMP-1 dosimetry sets fall well within ±20% of the measurement. Moreover, if the four sets are averaged, the average C/M result is very close to 1.0. This indicates that for the NMP-1 plant, the calculations do not exhibit any significant bias. Thus no bias factor will be applied to the calculations.

Table 5-1 Comparison of Calculated and Measured Results for NMP-1 Dosimetry Removed After Cycle 1

Dosimeter	Measured dps/mg	Calculated dps/mg	C/M
Fe	95.37	Note 1	Note 1
Fe	94.54	Note 1	Note 1
Fe	103.87	Note 1	Note 1
Fe Average	97.93	94.92	0.969
Cu	5.93	Note 1	Note 1
Cu	5.89	Note 1	Note 1
Cu	5.98	Note 1	Note 1
Cu Average	5.935	5.368	0.905
Overall Average of Cu and Fe			0.937

End-of-irradiation-April 14, 1973.
Note 1: Since no specific location information is available on the individual wire samples, they were averaged for each dosimeter type.

Table 5-2 Comparison of Calculated and Measured Results for NMP-1 Dosimetry Removed from 30° Capsule

Dosimeter	Set ID	Measured dps/mg	Calculated dps/mg	C/M
Fe	P1	188.30	175.92	0.934
Fe	P2	177.72	175.78	0.989
Fe	P3	173.49	175.69	1.013
Fe Average				0.979
Cu	P2	18.05	17.61	0.975
Cu	P3	16.76	17.59	1.050
Cu Average				1.012
Overall Average of Cu and Fe				0.996

End-of-irradiation-March 3, 1979

Table 5-3 Comparison of Calculated and Measured Results for NMP-1 Dosimetry Removed from 300° Capsule

Dosimeter	Set ID	Measured dps/mg	Calculated dps/mg	C/M
Fe	P7	174.10	178.24	1.024
Fe	P8	157.46	178.14	1.131
Fe Average				1.078
Ni	P7	2737	3069.19	1.121
Ni	P8	2641	3067.33	1.161
Ni Average				1.141
Cu	P7	20.80	20.78	0.999
Cu	P8	20.80	20.76	0.998
Cu Average				0.999
Overall Average of Fe, Ni, and Cu				1.073

End-of-irradiation-March 19, 1982.

Table 5-4 Comparison of Calculated and Measured Results for NMP-1 Dosimetry Removed from 210° Capsule

Dosimeter	Measured dps/mg	Calculated dps/mg	C/M
Cu-1	20.66	18.10	0.876
Cu-2	20.81	18.10	0.870
Cu-3	19.43	18.10	0.932
Cu Avg	20.3	18.10	0.892
Fe-1	149.4	139.0	0.930
Fe-2	138.6	139.0	1.003
Fe-3	135.7	139.0	1.024
Fe Avg	141.2	139.0	0.984
Ni-1	1725	1838	1.066
Ni-2	1636	1838	1.124
Ni-3	1592	1838	1.155
Ni Avg	1651	1838	1.113
Capsule Average			0.997

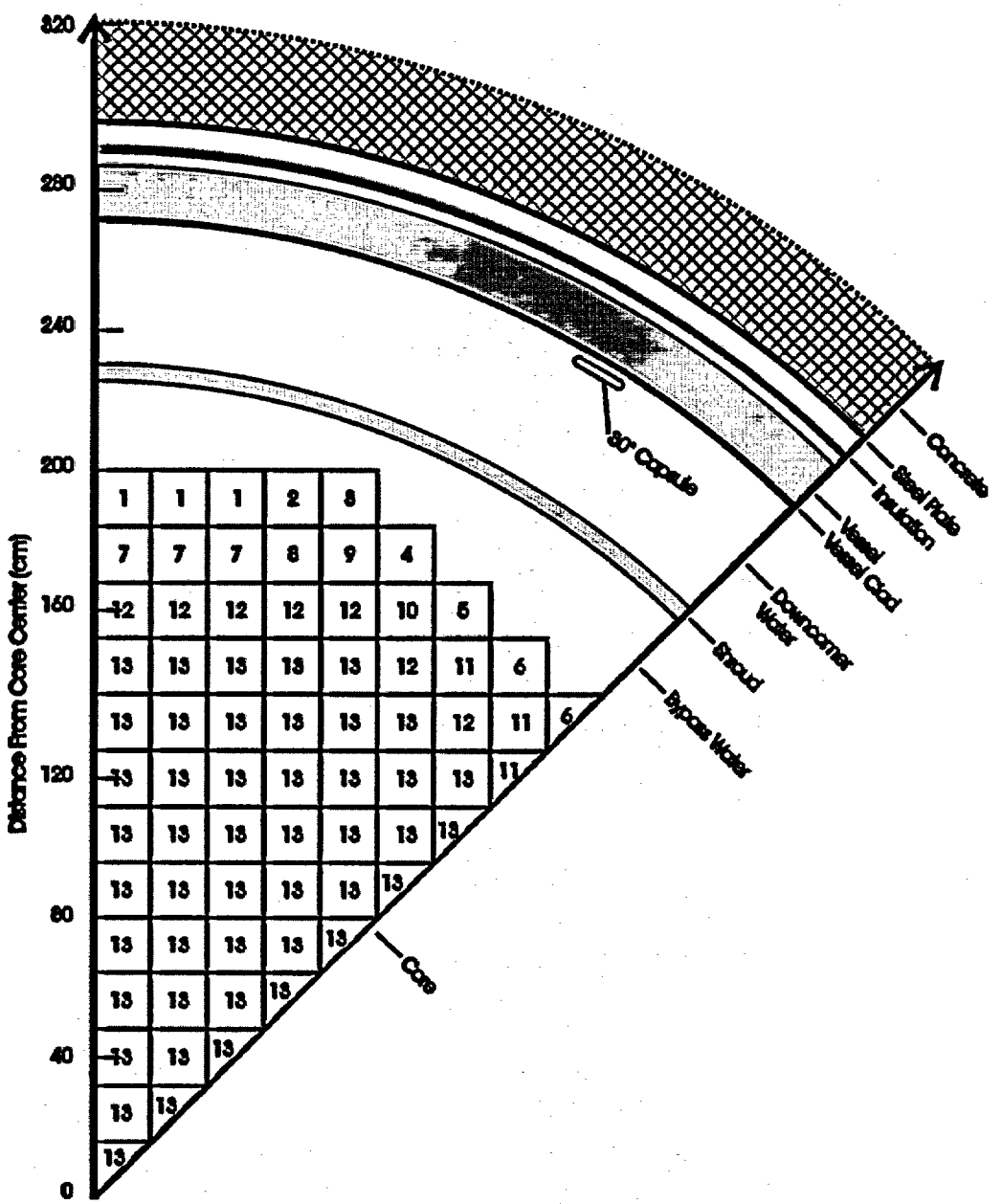

Figure 5-1 Nine Mile Point Unit 1 R-θ Geometry Used in the DORT Calculations

6.0 Updated Fluence Extrapolation

Two dimensional discrete ordinates transport calculations for NMP-1 were reported in Reference [26]. The cycle 12 peak vessel IR fast (E > 1 MeV) flux was determined to be 3.054 x 10^9 n/cm^2/sec. This flux, which is 3% higher than the peak cycle 7 flux, was assumed to represent the exposure from plant startup through 28 EFPY for the upper shell course plates. Therefore, using the cycle 12 flux at 28 EFPY, the peak ID surface fast fluence of 2.70 x 10^{18} n/cm^2 was used in the development of the current NMP-1 P-T curves.

Using the present set of calculations, which include detailed analysis of cycle averages for cycles 7, 9, 10 and 11, as well as a more detailed analysis of through cycle analysis for cycle 12, the new calculated fluence at 28 EFPY is 1.96E18 n/cm^2. Thus a more detailed calculation has resulted in a lower fluence. Therefore, P-T curves based on the earlier fluence are conservative. The fluence results from the current work are summarized in Tables 6-1 and 6-2.

Table 6-1 Estimated Maximum Exposure in the Reactor Vessel Wall of the Nine Mile Point Unit 1 Reactor

	Fluence (E>1 MeV) (n/cm^2)	Fluence (E>0.1 MeV) (n/cm^2)	dpa
Maximum Vessel Exposure at end of Cycle 12 (March 1997)			
IR	1.24E18	2.43E18	1.92E-03
1/4 T	7.85E17	2.13E18	1.30E-03
½ T	4.38E17	1.63E18	8.17E-04
3/4 T	2.33E17	1.14E18	4.99E-04
Maximum Vessel Exposure at 28 EFPY			
IR	1.96E18	3.84E18	3.03E-03
1/4 T	1.24E18	3.37E18	2.05E-03
½ T	6.92E17	2.56E18	1.29E-03
3/4 T	3.68E17	1.80E18	7.86E-04

Table 6-2 Maximum Vessel Fluence (E > 1 MeV) Using dpa Attenuation

Position	16.76 EFPY Fluence (n/cm^2)		28 EFPY Fluence (n/cm^2)	
	Calculated dpa	RG1.99(2)	Calculated dpa	RG1.99(2)
IR	1.24E18	1.24E18	1.96E18	1.96E18
1/4 T	8.36E17	8.08E17	1.32E18	1.28E18
½ T	5.27E17	5.27E17	8.29E17	8.27E17
3/4 T	3.22E17	3.43E17	5.07E17	5.42E17

7.0 Summary and Conclusions

In summary, it is concluded that the RG 1.190 requirement for qualification of the MPM methodology used for Nine Mile Point Units 1 & 2, by comparisons to measurement and calculational benchmarks, has been fully satisfied. Based on the work reported here, the following specific conclusions have been made.

- In cases where plant-specific data from fuel depletion codes are not available, it is concluded that the use of ORIGEN to determine the fraction of fissions from fuel isotopes is acceptable and slightly conservative. If ORIGEN is used, differences within the range of burnups considered are less than estimated uncertainties in results arising from the uncertainties in neutron spectrum and source normalization.
- It has been concluded that use of the PCA benchmark, as applied for testing BWR calculational methodologies, is valuable and satisfies the benchmarking requirement of RG 1.190. The PCA contains similar materials and transport distances as compared with BWRs. The neutron transport paths are large enough to confirm the adequacy of the methodology and transport cross sections.
- Successful calculation of the PCA as demonstrated by agreement with other accepted calculations and by agreement with measurements within acceptable tolerance has been achieved. This satisfies the RG1.190 requirement for benchmarking against a measurement benchmark.
- A consistent bias (averaging 6%) was observed in the PCA calculations and this is in agreement with other calculations using the synthesis method. This bias is not likely to be present in calculations of operating plants because of geometry differences.
- Discrepancies in C/M ratios observed using the BUGLE 47-group dosimetry cross sections can be lessened or eliminated by using the dosimetry cross sections contained in the ASTM standard cross section data file.
- All of the calculated reaction rates for the dosimetry sets fall well within $\pm$20% of the measurement. Moreover, if the four sets are averaged, the average C/M result is very close to 1.0. This indicates that for the NMP-1 plant, the calculations do not exhibit any significant bias. Thus no bias factor will be applied to the calculations.
- Using the present set of calculations, the new calculated fluence at 28 EFPY is 1.96E18 n/cm^2. The new model includes detailed analysis of cycle averages for cycles 7, 9, 10 and 11, as well as a more detailed through cycle analysis of cycle 12. Thus, a more detailed calculation has resulted in a lower fluence. Therefore, P-T curves based on the earlier fluence estimates are conservative.

8.0 Nomenclature

BWR	boiling water reactor
C/M	calculated to measured ratio
dpa	displacements per atom
EFPS	effective full power seconds
ID	inner diameter
IR	inner radius
LWR	light water reactor
MPM	MPM Technologies, Inc.
NMPC	Niagara Mohawk Power Corporation
NMP-1	Nine Mile Point Unit 1
NMP-2	Nine Mile Point Unit 2
NRC	U. S. Nuclear Regulatory Commission
OD	outer diameter
OR	outer radius
ORNL	Oak Ridge National Laboratory
PCA	pool critical assembly
PWR	pressurized water reactor
RG	Regulatory Guide
RSICC	Radiation Safety Information Computational Center
T	vessel wall thickness

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