November 14, 2000

Mr. Nathan L. Haskell
Director, Licensing and Performance Assessment
Palisades Plant
27780 Blue Star Memorial Highway
Covert, MI 49043

SUBJECT: PALISADES PLANT - REACTOR VESSEL NEUTRON FLUENCE EVALUATION AND REVISED SCHEDULE FOR REACHING PRESSURIZED THERMAL SHOCK SCREENING CRITERIA (TAC NO. MA8250)

Dear Mr. Haskell:

By letter dated February 21, 2000, as supplemented April 21, July 6, and August 31, 2000, you forwarded Westinghouse Report WCAP-15353, "Palisades Reactor Pressure Vessel Neutron Fluence Evaluation," and requested that the Nuclear Regulatory Commission (NRC) review and approve it. You also requested approval of a new estimate for reaching the pressurized thermal shock (PTS) screening criteria in 10 CFR Part 50.61. By a separate letter dated April 27, 2000, you have requested an amendment to exclude construction time from the authorized operating term by changing the operating license expiration date from March 14, 2007, to March 24, 2011. The evaluation provided by your February 21, 2000, letter, as supplemented, considered your April 27, 2000, request and addressed the reactor vessel PTS issue for the existing and proposed license expiration dates.

The NRC staff has completed its review of WCAP-15353 and supporting documentation. Enclosed is our safety evaluation in which we conclude that: (1) the planned operation with the Ultra-Low Leakage core design will provide a substantial reduction in the reactor vessel fluence during the remaining fuel cycles (Cycles 16-22) of the current operating license with its proposed extension, (2) the WCAP-15353 methodology is acceptable for predicting the reactor vessel fluence, and (3) the calculations and updates of the Palisades vessel fluence are acceptable. The NRC staff also concludes that the properties of the Palisades Plant reactor vessel beltline materials will comply with the requirements of 10 CFR Part 50.61 for continued operation of the facility through both the end of the facility's current operating license (to March 14, 2007) and through the proposed period of operation (to March 24, 2011).

The reactor vessel fluence reduction associated with the Ultra-Low Leakage core design requires that the flux suppression assemblies be loaded in specific core design locations and oriented with the stainless steel rods facing the reactor vessel. The NRC staff recognizes that your procedures, during refueling, will assure that the placement and orientation of the flux suppression assemblies are consistent with that assumed in the WCAP-15353 fluence analysis. The NRC staff wishes to be informed promptly in the unlikely event you should discover any mispositioned or misoriented flux suppression assembly.

This letter does not authorize operation beyond the current license expiration date of March 14, 2007. Your application for amendment, dated April 27, 2000, to this end will be addressed by separate correspondence in the near future.

If you have questions regarding this letter or the enclosure, contact me at (301) 415-3049.

Sincerely,

/RA/

Darl S. Hood, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-255

Enclosure: Safety Evaluation

cc w/encl: See next page

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cc:

Mr. Thomas J. Palmisano
Site Vice President
Palisades Plant
27780 Blue Star Memorial Highway
Covert, MI 49043

Mr. Robert A. Fenech, Senior Vice President
Nuclear, Fossil, and Hydro Operations
Consumers Energy Company
212 West Michigan Avenue
Jackson, MI 49201

Arunas T. Udrys, Esquire
Consumers Energy Company
212 West Michigan Avenue
Jackson, MI 49201

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 Warrenville Road
Lisle, IL 60532-4351

Jerry Sarno, Supervisor
Covert Township
P. O. Box 35
Covert, MI 49043

Office of the Governor
P. O. Box 30013
Lansing, MI 48909

U.S. Nuclear Regulatory Commission
Resident Inspector's Office
Palisades Plant
27782 Blue Star Memorial Highway
Covert, MI 49043

Drinking Water and Radiological
  Protection Division
Michigan Department of
  Environmental Quality
3423 N. Martin Luther King Jr Blvd
P. O. Box 30630 CPH Mailroom
Lansing, MI 48909-8130

Michigan Department of Attorney General
Special Litigation Division
630 Law Building
P.O. Box 30212
Lansing, MI 48909

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO REACTOR PRESSURE VESSEL FLUENCE REEVALUATION

CONSUMERS ENERGY COMPANY

PALISADES PLANT

DOCKET NO. 50-255

1.0 INTRODUCTION

By letter dated February 21, 2000 (Reference 1), Consumers Energy Company (the licensee) submitted information and requested Nuclear Regulatory Commission (NRC) approval of a revised reactor pressure vessel fluence value for the end of the operating license[1] for the Palisades Plant. In its submittal, the licensee also included a revised estimate of the date for reaching the pressurized thermal shock (PTS) screening criteria of 10 CFR Part 50.61, "Fracture Toughness Requirements For Protection Against Pressurized Thermal Shock Events," for the limiting reactor vessel material and requested NRC staff approval. The technical basis for the revised fluence submittal was provided in the Westinghouse report WCAP-15353, "Palisades Reactor Pressure Vessel Fluence Evaluation," which includes an ultra low leakage loading scheme and several improvements and updates for the fluence prediction through the current fuel cycle, Cycle 15 (Reference 2). These improvements include an updated core thermal power measurement, a revised water temperature estimate for the peripheral fuel assemblies, an improved CASMO-4/SIMULATE-3 calculation of the core neutron source distribution, the inclusion of the effects of axial variation of the leakage between the core and the cavity, and the effect of the presence of the baffle formers on the vessel fluence. In this reevaluation, the fluence prediction is based on a detailed model calculation and the dosimetry measurement data is used to qualify the calculational model, rather than to provide an overall renormalization of the fluence calculation.

WCAP-15353 provides a detailed description of the in-vessel (surveillance capsule) and cavity dosimetry measurement programs, the dosimetry evaluation methods, the dosimetry results through the end of Cycle 14, and prediction for Cycle 15 which is assumed to be the equilibrium cycle for the remaining plant operating time. The vessel fast neutron exposure is provided in terms of: (1) neutron energy E > 1.0 MeV, (2) E > 0.1 MeV, and (3) iron displacements-per-atom (dpa). The report also provides the models of the discrete ordinates DOORS neutron transport code used to calculate the vessel fluence through Cycle 15 and the projections to future Palisades Plant statepoints (Reference 3). The FERRET least-squares adjustment

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[1] The current operating license will expire March 14, 2007 (i.e., after Cycle 19). By application dated April 27, 2000, the licensee requested an amendment to exclude the construction period and allow for 40 years of operation by extending the expiration date to March 24, 2011. Therefore, this NRC staff safety evaluation also addresses the reactor vessel for the proposed operation until March 24, 2011 (i.e., through Cycle 22).

methodology (Reference 4) is used to combine the calculations and dosimetry measurements to determine the fluence uncertainty. Comparisons of the FERRET adjusted calculations and measurements are provided, as well as comparisons of the unadjusted measured and calculated dosimeter reaction rates. The measurement-to-calculation (M/C) comparisons are used to demonstrate the validity of the DOORS fluence calculational methodology. However, only the calculated values are used in the fluence evaluation.

The licensee submitted additional information in letters dated April 21, July 6, and August 31, 2000 (References 5, 6, and 11, respectively). The Palisades' methodology and analysis are summarized in Section 2.1 below, and the NRC staff's evaluation of the technical issues raised during this review is presented in Sections 2.2 to 2.7. The NRC staff's evaluation of the revised schedule for reaching the PTS screening criteria is presented in Section 2.8. The NRC staff's conclusions are presented in Section 3.0. References are identified in Section 4.0.

2.0 EVALUATION

2.1 Neutron Dosimetry Measurements and Analysis

In the submittal, the vessel fluence is determined by a detailed Discrete Ordinates Radiation Transport (DORT, a component of the DOORS set of codes, Reference 3) neutron transport calculation. The in-vessel (surveillance capsule) and cavity dosimetry measurements are used to qualify this methodology for application to the Palisades Plant. The in-vessel dosimetry includes five surveillance capsules, withdrawn during the first thirteen cycles of operation. The in-vessel dosimetry included two accelerated surveillance capsules located on the outside surface of the support barrel and three capsules located on the inside surface of the reactor pressure vessel. The capsules removed after Cycles 2, 5, and 10 were installed before Cycle 1 operation and provided a continuous measurement up to the time of removal. The capsule removed after Cycle 9 measured only the single cycle, while the capsule removed after Cycle 13 measured both Cycles 12 and 13.

To provide a reliable measurement of the Palisades Plant's fluence, multiple foil sensor sets were installed at selected azimuthal and axial locations in the reactor vessel cavity. In addition, stainless steel "gradient" chains were installed in the cavity to provide a measurement of the axial fluence gradient. Several chains were symmetrically located and were used to confirm azimuthal fluence symmetry. After irradiation, the gradient chains were segmented and used to provide Fe-54 (n, p), Ni-58 (n, p), and Co-59 (n, γ) activities.

The neutron dosimetry provides a direct measurement of the specific activity (disintegrations per second per gram or dps/gm) associated with the individual dosimeter-specific reactions. (The specific activity measurements for the Palisades Plant are provided in Appendices A-D of Reference 2, together with the associated reactor irradiation history). The measured dosimeter activities were converted to specific full-power reaction rates (reactions per second per nucleus or rps/nucleus) averaged over the period of irradiation. This conversion accounted for the physical characteristics of the sensor (e.g., weight of the target isotope in the sample), the operating history of the reactor, the energy response of the sensor (e.g., reaction cross section), decay of the target isotope, decay time following irradiation and, in the case of fission dosimeters, the number of product atoms produced per reaction. In order to allow comparison of the measured and calculated dosimeter reaction rates, the measured reaction rates were corrected for target depletion and interference from competing reactions.

2.2 Calculation of the Vessel Fluence and Neutron Dosimetry Measurements

WCAP-15353 provides a description of the DORT calculation used to determine the vessel fluence, as well as the calculations used to predict the Palisades Plant's measured dosimetry which validates the transport model. The calculational model includes a detailed representation of the peripheral fuel assemblies, the core internals, the downcomer, and vessel geometry. A two-zone representation of the core was used to allow the modeling of the reduced temperature (and increased density) of the coolant in the peripheral fuel assemblies. The stainless steel shield pins in the peripheral fuel assemblies were explicitly modeled in the calculation. The calculation was performed for an azimuthal quadrant of the geometry and employed a relatively fine (r, α, z) spatial mesh. The discrete ordinates calculation was performed using an S_{16} angular quadrature approximation.

The BUGLE-96 ENDF/B-VI, 47-neutron, 20-photon group cross sections and nuclear data library (Reference 7) were used in the DORT calculations. The scattering cross sections were represented using a P_5 Legendre expansion. The detailed pin-wise power distribution in the peripheral fuel assemblies was used to determine the core neutron source for input to DORT. The pin-wise core power distribution was calculated with the CASMO-4/SIMULATE-3 code system (Reference 8). The calculations were performed in (r, α), (r, z) and (r) geometries. A synthesis technique was used to determine the three-dimensional fluence distribution and account for the effect of axial leakage between the core and the cavity.

The calculations of the in-vessel and cavity dosimetry measurements were performed with the DORT model used to calculate the vessel fluence. The surveillance capsules and cavity sensors were modeled in detail at their corresponding radial and azimuthal locations. The measured dosimeter reaction rates were calculated using the dosimeter specific reaction cross sections. The E > 1.0 MeV fluence at the dosimeter location was also calculated to allow comparison with the dosimeter fluence determined with FERRET. The calculated dosimeter response was determined for the irradiation period up to the time the capsule was withdrawn. Coupled DORT neutron-photon calculations were performed to determine the photo-fission contribution to the U-238 (n, f) and Np-237 (n, f) dosimeter reaction rates.

2.3 Measurement-to-Calculation (M/C) Comparisons and Analysis

The in-vessel and cavity dosimetry measurements were used to validate the Palisades Plant's specific DORT calculational model. The validation includes comparisons of both: (1) measurements and unadjusted calculations and (2) FERRET adjusted calculations and unadjusted measurements. These comparisons recognize that the calculations and measurements are uncertain and include a least-squares adjustment of the calculated values. Comparisons are provided for the fluence E > 1.0 MeV, the fluence E > 0.1 MeV, and the dpa. Reaction rate comparisons are made for the following fast neutron reactions: Cu-63 (n, α), Ti-46 (n, p), Fe-54 (n, p), Ni-58 (n, p), U-238 (n, f), and Np-237 (n, f). Comparisons are provided for the five in-vessel capsules and the six cavity sensors (each at a different azimuthal location) for Cycles 8, 9, 10, and 11.

2.4 Pressure Vessel Fluence Calculations

The model used in the calculation of the reactor vessel fluence and dosimetry is generally state-of-the-art and consistent with the methods of the Draft Regulatory Guide DG-1053 entitled, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," (Reference 9). The core neutron source is based upon a cycle-dependent, pin-wise source in the peripheral fuel assemblies and accounts for the axial and radial source gradients on the core periphery. The increase in the number of neutrons per fission and the hardening of the neutron spectrum due to the increase in the number of plutonium fissions with fuel burnup, is also included in the determination of the source. (Neglecting either of these effects would result in a nonconservative prediction of the reactor vessel fluence). This modeling has been evaluated and tested independently at Brookhaven National Laboratory (BNL) and found to be in excellent agreement with BNL calculations (Reference 10).

The WCAP-15353 calculational methodology is an extension and improvement of the methodology previously described in WCAP-14557 (Reference 11). There are several important changes in the methodology. These include: (1) the reduction in the magnitude of the core neutron source during Cycles 1-13, (2) the reduction in the coolant temperatures in the peripheral fuel assemblies and in the bypass region, (3) a reevaluation of the core neutron source with the updated CASMO-4/SIMULATE-3 core performance code, and (4) the inclusion of the former plates in the DORT calculational model. Each of these changes result in an incremental reduction in the vessel fluence. The reduction in the core power and neutron source during Cycles 1-13 was the result of a loss of calibration in the plant's Feedwater Flow System (FFS) that occurred early in Cycle 1. The loss of calibration was discovered when the FFS was recalibrated in the middle of Cycle 13. As a result, during this period the plant was operating at a core power more than 2 percent (of rated power) lower than assumed in the WCAP-14557 analysis. Since the core neutron source is directly proportional to the core power, this change results in a proportional reduction in the vessel fluence and the prediction of the dosimetry measurements.

As noted in Reference 6, the licensee performed additional Ultrasonic Flow Measurements (UFMs) of the Palisades Plant's feedwater flow to determine the effect of the loss of calibration on the heat balance and core power. These measurements provided for extensive comparisons between the existing venturi feedwater flow measurement system and the UFMs. A statistical analysis of the flow comparisons showed that the loss of calibration of the venturi feedwater flow system resulted in a systematic overprediction of the core power level by more than 2 percent. To establish confidence in this conclusion, a detailed uncertainty analysis of both the venturi and UFM systems was performed. This analysis confirmed the conclusion and indicated that it is highly probable (> 95 percent) that the power/source overprediction is somewhat greater than 2 percent. Since the WCAP-15353 fluence calculations only take credit for a 2 percent overprediction of the core power level, the NRC staff considers the calculations to be conservative.

In the WCAP-14557 model, the temperature of the coolant in the peripheral fuel assemblies and in the region between the baffle and the core support barrel was taken to be the core-average coolant temperature. This is a conservative approximation made by Westinghouse and has been independently evaluated by Brookhaven National Laboratory (BNL). In the updated model of WCAP-15353, this conservatism has been removed and the coolant temperatures are determined using SIMULATE-3. However, the licensee indicated in

Reference 6 that the method used to determine the coolant temperature in the peripheral fuel assemblies includes additional conservatism.

The CASMO-3/SIMULATE-3 core performance code-system was used to determine the DORT input core neutron source in the initial Westinghouse analysis. However, the more recent CASMO-4/SIMULATE-3 version of this code system has been used to determine the source for the WCAP-15353 fluence calculations. This new version of CASMO-4 includes several improvements which could have a significant effect on the core neutron source and, consequently, the reactor vessel fluence predictions. These include: (1) the incorporation of additional Pu and Gd isotopes in the resonance region calculation and (2) the explicit modeling of Gd in the pin-cell calculation. Since the modeling changes incorporated in the CASMO-4/SIMULATE-3 model are intended to improve the basic physics treatment of the fuel cell neutronics, and have not been specifically designed to reduce the power and neutron source in the peripheral fuel assemblies, the use of CASMO-4/SIMULATE-3 should reduce the uncertainty in the source and improve the accuracy of the fluence predictions.

To confirm the accuracy of the CASMO-4/SIMULATE-3 model for application to the Palisades Plant, the licensee made extensive comparisons of the CASMO-4/SIMULATE-3 predictions with the Palisades Plant's Incore Monitoring System (PIDAL) measurements. This verification included the comparison of the power in the fuel assemblies on the periphery of the core which determine the neutron source used in the DORT fluence calculation. The results of the comparisons for Cycles 4-14 are provided in Reference 11 and indicate that the CASMO-4/SIMULATE-3 predictions are in good agreement with the PIDAL measurements.

The Palisades Plant's core baffle and support barrel are separated by seven radial former plates which are distributed axially over the height of the core. These former plates tend to shield the vessel from the neutron fluence emanating from the core at an angle with respect to the horizontal. While the initial WCAP-14457 analysis did not include the effect of the former plates, the WCAP-15353 calculational model includes a detailed description of these plates (Reference 12). In response to an NRC request for additional information (RAI), the effect of the former plates has been evaluated (Section 1.3 of Attachment 1 to Reference 11) and found to be consistent with the guidance of DG-1053. In addition, since the inclusion of the former plates in the DORT model reduces the fluence, BNL has independently calculated the effect of the former plates. The BNL calculations and the results of Reference 11 agree to within a few percent.

DG-1053 (Reference 9) requires that an analytic-uncertainty analysis be performed to estimate the uncertainty in the DORT fluence prediction. WCAP-15353 includes an analytic-uncertainty for the Palisades Plant's fluence prediction and the details of this analysis are provided in Reference 6 (see response to RAI 14). The results of this analysis are in agreement with previous BNL analyses (References 10 and 11) and indicate that the calculated vessel inner-wall fluence uncertainty (E > 1.0 MeV) is less than 20 percent.

2.5 <u>The Palisades Plant's Ultra-Low Leakage Core Design</u>

The reactor vessel fluence is proportional to the fast-neutron leakage from the fuel assemblies on the periphery of the core. In Cycle 8 and subsequent cycles, the Palisades Plant's core and fuel assembly designs have included various low leakage features (e.g., substitution of fuel rods with stainless steel rods in the outer rows of some peripheral fuel assemblies) for reducing the rate of fluence accumulation on the reactor vessel. The current Cycle 15 core is an Ultra-Low Leakage design which provides an even larger reduction in the core neutron leakage and the vessel fluence than Cycle 11 (Reference 12). The licensee indicates in Reference 11 (Section 1.6 of Attachment 1) that future cores will also utilize Ultra-Low Leakage designs providing fluence accumulation rates equivalent to that obtained during Cycle 15. As a result of the planned operation with the Ultra-Low Leakage core design, the vessel fluence accumulated during the remaining fuel cycles (Cycles 16-22) will be substantially lower than was determined in the initial WCAP-14557 analysis (Reference 12).

The fluence reduction is accomplished by the placement of flux suppression fuel assemblies in specific peripheral core locations opposite the limiting locations on the vessel. Since the fuel assembly design includes stainless steel rods on the side of the assembly facing the vessel, the Ultra-Low Leakage design requires that the flux suppression assemblies be loaded in specific core design locations and oriented with the stainless steel rods facing the vessel. The NRC staff recognizes that the licensee's procedures, during refueling, will assure that the placement and orientation of the flux suppression assemblies are consistent with that assumed in the WCAP-15353 fluence analysis. The NRC staff wishes to be informed promptly in the unlikely event the licensee should discover any mislocated or misoriented flux suppression assembly.

2.6 <u>Neutron Dosimetry Measurements and Analysis</u>

The Palisades Plant's dosimetry measurement data base is extensive and includes in-vessel and cavity measurements for a standard set of six fast neutron threshold dosimeters. Except for the Cycle 2 accelerated capsule, all the dosimeter activity measurements were made by the Westinghouse Analytical Services Laboratory. Following removal and transport to the laboratory, the samples were prepared, weighed, and then counted using a lithium drifted germanium, Ge(Li), gamma spectrometer. These dosimetry activity measurements were performed according to the established American Society for Testing and Materials (ASTM) procedures (e.g., Reference 13).

The full power reaction rates (which are used in the benchmarking) are proportional to the measured specific activities and include adjustments for: (1) the actual Palisades Plant operating history and (2) the decay of the product isotope during plant operation and up to the time of the activity measurement. The operating history for Cycles 1-10 was taken from NUREG-0020, while the operating history for Cycles 11-14 was taken from the plant records. The integration over the power history is performed using a monthly time interval. This approximation has been evaluated by BNL in Reference 10 and shown to be adequate for the dosimeters used at the Palisades Plant.

In determining the reaction rates for the U-238 (n, f) and Np-237 (n, f) fission dosimeters, several corrections are required. The contribution of U-235 (n, f) thermal fission due to the presence of U-235 impurities in the dosimeter sample must be removed from the measured activity. The fission contribution from Pu that was accumulated during the irradiation period must also be removed from the measured activity. The WCAP-15353 analysis includes corrections for both the U-235 impurities and Pu build-up. Since the gamma field is significant at the dosimeter locations, the photo-fission contribution must also be removed from both the U-238 and Np-237 measured activities. The WCAP-15353 analysis includes photo-fission corrections for both the U-238 and Np-237 dosimeters. BNL has independently evaluated these corrections and found them to be acceptable.

The fluence axial shape in the cavity was determined using stainless steel gradient chains. The composition of the gradient chains used to interpret the activation measurements was determined using samples certified by the National Institute for Standards and Technology. This determination was confirmed by comparison of the activation of the segmented gradient chains and the activation of the high purity iron, nickel and cobalt foils (Reference 6, response to RAI 5).

2.7 Measurement-to-Calculation (M/C) Comparisons and Analysis

The WCAP-15353 pressure vessel neutron fluence evaluation includes benchmarking of the methodology based on a set of plant-specific in-vessel and cavity-dosimetry measurements performed over the first thirteen cycles of operation. The measurements include an extensive set of six dosimeter types with broad spectral coverage; i.e., energy thresholds ranging from about 4.7 MeV down to about 0.6 MeV. The dosimeters were located in accelerated locations on the outside wall of the support barrel, on the inside wall of the reactor vessel and in the cavity between the vessel and biological shield. In addition, special gradient chains were used to provide a measurement of the axial distribution of the fluence in the cavity. The dosimetry provided continuous fluence measurements from initial operation through Cycle 2, Cycle 5, and Cycle 10. In addition, dosimetry measurements were made during Cycle 8, Cycle 9, Cycles 10-11, and Cycles 12-13. The extensive spatial distribution, spectral coverage and cycles of operation of these dosimetry measurements provide a relatively complete benchmarking data base and validation of the calculational methodology over the expected range of operation.

The benchmarking reported in WCAP-15353 includes several types of comparisons of the dosimetry measurements and calculations. Comparisons are first made for the measured dosimetry reaction rates; i.e., Cu-63 (n, α), Ti-46 (n, p), Fe-54 (n, p), Ni-58 (n, p), U-238 (n, f), and Np-237 (n, f). The in-vessel and cavity measurements provide one-hundred and one (101) M/C comparisons. Of these reaction rates, twenty-four are in-vessel and seventy-seven are cavity measurements. This set of comparisons yields an average M/C value 1.08 ± 0.086.

In addition to the reaction rate comparisons, the FERRET code was used to combine the energy-dependent reaction rate measurements and determine several spectrum integrated quantities; i.e., E > 1.0 MeV fluence, E > 0.1-MeV fluence, and dpa. The values determined by FERRET include a least-squares statistical adjustment of the measurements and the calculated input spectrum (based on the estimated uncertainty). The dosimetry measurements were combined to determine seventeen values of each of these quantities. The M/C values for the E > 1.0 MeV fluence, E > 0.1 MeV fluence, and dpa are 1.05 ± 0.06, 1.04 ± 0.06, and 1.05 ± 0.05, respectively. As indicated in Reference 6 (response to RAI 15(b)), if only the in-vessel capsule data are used (as recommended by previous independent reviewers), the average M/C value is 1.003.

The M/C agreement for both the comparison of the reaction rates and the comparison of the FERRET adjusted integral quantities is well within the estimated calculation and measurement uncertainties provided in WCAP-15353.

2.8 Pressurized Thermal Shock Screening Criteria

In its February 21, 2000 letter, the licensee requested that the NRC consider the information that has been discussed above regarding the licensee's proposed methodology for establishing reactor vessel fluence values and reevaluate the status of the Palisades Plant's reactor vessel with respect to the requirements in 10 CFR Part 50.61 on PTS. Specifically, the licensee requested that the NRC staff "endorse the new date at which the reactor vessel is estimated to reach the PTS screening criteria." The new date cited by the licensee is the year 2014.

Based upon the NRC staff's approval of the licensee's fluence calculation methodology, the NRC staff reviewed the issue of the compliance of the Palisades Plant's reactor vessel with the requirements of 10 CFR Part 50.61. 10 CFR Part 50.61 requires that end of license (EOL) fluence values be established and used to define the PTS reference temperature (RT_{PTS}) for each reactor vessel beltline material. These material RT_{PTS} values are then compared to the screening criteria given in 10 CFR Part 50.61 (300 °F for circumferential welds or 270 °F for axial welds, plates, and forgings). If all reactor vessel beltline material RT_{PTS} values are below the screening criteria of 10 CFR Part 50.61, it is concluded that the reactor vessel will maintain sufficient fracture toughness against PTS events through the end of the facility's operating license.

In its July 6, 2000, response to the NRC staff's RAI (e.g., response to RAI 17), the licensee provided a PTS assessment of each reactor vessel beltline material for two EOL conditions. The first condition presumed that the Palisades Plant's operating license would expire on its current expiration date of March 14, 2007. The second condition was based on a submittal by the licensee to the NRC on April 27, 2000, to recapture the facility construction period and extend the operating license to March 24, 2011. Table 1 below reflects the information provided by the licensee for each material relative to the current March 14, 2007, EOL date, while Table 2 reflects the same information relative to the proposed March 24, 2011, EOL date.

The NRC staff has reviewed the information shown in Tables 1 and 2. The NRC staff has confirmed that the information on the chemical contents (Cu% and Ni%), initial properties ($RT_{NDT(U)}$), and margin terms to be assessed in the evaluation of each Palisades Plant reactor vessel beltline material are consistent with values previously reported to (Reference 14), and found acceptable by the NRC (Reference 15). The projected "EOL fluence" values shown in each table were also found to be acceptable since they were determined using a calculational methodology acceptable to the NRC staff (e.g., see Section 3.0 below). Finally, when the methodology specified in 10 CFR Part 50.61 was applied to determine RT_{PTS} values from these input parameters, the NRC staff concluded that the RT_{PTS} values reported by the licensee (and given in the final column of Tables 1 and 2) are correct. The data shown in Table 1 will be incorporated into the NRC staff's Reactor Vessel Integrity Database (RVID). Similarly, if the NRC should approve the licensee's construction recapture submittal, the data shown in Table 2 would be used to update the RVID.

On the basis of the material RT_{PTS} values given in Tables 1 and 2, the NRC staff concludes that each Palisades Plant reactor vessel beltline axial weld and plate has been demonstrated to remain below the PTS screening criteria of 270 °F on March 14, 2007, and March 24, 2011. The NRC staff also concludes that the Palisades Plant reactor vessel beltline circumferential weld has been demonstrated to remain below the PTS screening criteria of 300 °F on March 14, 2007, and March 24, 2011. Therefore, the NRC staff concludes that the properties of the Palisades Plant reactor vessel beltline materials will comply with the requirements of 10 CFR Part 50.61 for continued operation of the facility through both the end of the facility's current operating license and through the proposed period of operation (to March 24, 2011) which includes construction recapture.

Table 1: Assessment of the Palisades Plant's Reactor Vessel Material Properties on the Current Facility Operating License Expiration Date (March 14, 2007)

RPV Material	Material Heat #	Cu%	Ni%	EOL Fluence (n/cm^2)	$RT_{NDT(U)}$ (°F)	Margin (°F)	RT_{PTS} (°F)
Axial Welds 2-112A/C	W5214	0.213	1.01	1.373×10^{19}	-56	65.5	261
Axial Welds 3-112A/C	W5214	0.213	1.01	1.373×10^{19}	-56	65.5	261
	34B009	0.192	0.980	1.373×10^{19}	-56	65.5	246
Circumferential Weld 9-112	27204	0.203	1.018	1.873×10^{19}	-56	65.5	275
Plate D-3803-1	C-1279	0.24	0.50	1.873×10^{19}	-5	17	192
Plate D-3803-2	A-0313	0.24	0.52	1.873×10^{19}	-30	34	192
Plate D-3803-3	C-1279	0.24	0.50	1.873×10^{19}	-5	17	192
Plate D-3804-1	C-1308A	0.19	0.48	1.873×10^{19}	0	34	185
Plate D-3804-2	C-1308B	0.19	0.50	1.873×10^{19}	-30	34	158
Plate D-3804-3	B-5294	0.12	0.55	1.873×10^{19}	-25	34	105

Table 2: Assessment of the Palisades Plant's Reactor Vessel Material Properties on the Proposed Facility Operating License Expiration Date (March 24, 2011)

RPV Material	Material Heat #	Cu%	Ni%	EOL Fluence (n/cm^2)	$RT_{NDT(U)}$ (°F)	Margin (°F)	RT_{PTS} (°F)
Axial Welds 2-112A/C	W5214	0.213	1.01	1.492×10^{19}	-56	65.5	266
Axial Welds 3-112A/C	W5214	0.213	1.01	1.492×10^{19}	-56	65.5	266
	34B009	0.192	0.980	1.492×10^{19}	-56	65.5	251
Circumferential Weld 9-112	27204	0.203	1.018	2.061×10^{19}	-56	65.5	281
Plate D-3803-1	C-1279	0.24	0.50	2.061×10^{19}	-5	17	195
Plate D-3803-2	A-0313	0.24	0.52	2.061×10^{19}	-30	34	196
Plate D-3803-3	C-1279	0.24	0.50	2.061×10^{19}	-5	17	195
Plate D-3804-1	C-1308A	0.19	0.48	2.061×10^{19}	0	34	188
Plate D-3804-2	C-1308B	0.19	0.50	2.061×10^{19}	-30	34	161
Plate D-3804-3	B-5294	0.12	0.55	2.061×10^{19}	-25	34	107

3.0 CONCLUSIONS

The NRC staff completed its review of WCAP-15353 entitled, “Palisades Reactor Pressure Vessel Neutron Fluence Evaluation,” and supporting documentation provided in References 1, 5, 6, and 11. Based on this review, the NRC staff concludes that: (1) the planned operation with the Ultra-Low Leakage core design will provide a substantial reduction in the reactor vessel fluence during the remaining fuel cycles (Cycles 16-22) of the current operating license with its proposed extension, (2) the WCAP-15353 methodology is acceptable for predicting the reactor vessel fluence, and (3) the calculation and updates of the Palisades vessel fluence are acceptable. The NRC staff also concludes that the properties of the Palisades Plant reactor vessel beltline materials will comply with the requirements of 10 CFR Part 50.61 for continued operation of the facility through both the end of the facility’s current operating license (to March 14, 2007) and through the proposed period of operation (to March 24, 2011) which includes construction recapture.

The reactor vessel fluence reduction associated with the Ultra-Low Leakage core design requires that the flux suppression assemblies be loaded in specific core design locations and oriented with the stainless steel rods facing the reactor vessel. The NRC staff recognizes that the licensee’s procedures, during refueling, will assure that the placement and orientation of the flux suppression assemblies are consistent with that assumed in the WCAP-15353 fluence analysis. The NRC staff wishes to be informed promptly in the unlikely event the licensee should discover any mislocated or misoriented flux suppression assembly.

4.0 REFERENCES

1. Letter from D. G. Malone, Consumers Energy Company, to US NRC “Palisades Reactor Vessel Neutron Fluence Reevaluation,” dated February 21, 2000.

2. WCAP-15353, “Palisades Reactor Pressure Vessel Fluence Reevaluation,” Westinghouse Electric Company, by G.K. Roberts et. al., January 2000, (see Attachment 2 of Reference 1 above).

3. DOORS 3.1, “One-, Two- and Three-Dimensional Discrete Ordinates Neutron/Photon Transport Code System, Version 3.1," Radiation Safety Information Computational Center (RSICC) Computer Code Collection CCC-650, August 1996.

4. HEDL-TME–79-40, “FERRET Data Analysis Code,” by F.A. Schmittroth, Hanford Engineering Development Laboratory, Richland, Washington, September 1979.

5. Letter from D.G. Malone, Consumers Energy Company, “Request for Approval of Revision to Incore Monitoring Code (PIDAL-3),” dated April 21, 2000 (and as supplemented by Reference 11 below).

6. Letter from N.L. Haskell, Consumers Energy Company, to US NRC, “Reply to Request for Additional Information Regarding Reactor Pressure Vessel Neutron Fluence Evaluation,” dated July 6, 2000.

7. DLC-185, “BUGLE-96, Coupled 47-Neutron, 20-Gamma Ray Group Cross Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications,” RSICC Data Library Collection, March 1996.

8. “CASMO-4; A Fuel Assembly Burnup Program, Users Manual,” Studsvik/SOA-95/1, Revision 0; and “SIMULATE-3; Advanced Three-Dimensional Two-Group Reactor Analysis Code, Users Manual,” Studsvik/SOA-95/15, Revision 0.

9. Draft Regulatory Guide DG-1053, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,” US NRC, dated September 1999.

10. Enclosure to letter from A. Hsia, US NRC, to R. Fenech, Consumers Power Company, “Palisades Plant - Transmittal of Technical Evaluation Report,” dated September 2, 1994. The enclosure is a memorandum from J. Carew et al., Brookhaven National Laboratory, to L. Lois, US NRC, “Palisades Pressure Vessel and Cavity Fluence Evaluation,” dated March 15, 1994.

11. Letter from N.L. Haskell, Consumers Energy Company, to US NRC, “Palisades Plant Additional Information Regarding Reactor Vessel Fluence Projections,” dated August 31, 2000.

12. WCAP-14557, Revision 1, “Consumers Power Company Reactor, “ Vessel Neutron Fluence Measurement Program for Palisades Nuclear Plant - Cycles 1 through 11," by J.D. Perock et al., Westinghouse Electric Company, March 1996.

13. ASTM Designation E263-93, “Standard Test Method for Measuring Fast Neutron Reaction Rates by Radioactivation of Iron,” in ASTM Standards, Section 12, American Society for Testing and Materials, Philadelphia, PA, 1997.

14. Letter from N.L. Haskell, Consumers Energy Company, to US NRC, responding to request for additional information regarding Generic Letter 92-01, Revision 1, Supplement 1, “Reactor Vessel Structural Integrity,” dated September 8, 1998.

15. Letter from R.G. Schaaf, US NRC, to N.L. Haskell, Consumers Energy Company, “Palisades Plant - Response to the Requests for Additional Information Regarding Generic Letter 92-01, Revision 1, Supplement 1, “Reactor Vessel Structural Integrity,” dated September 15, 1999.

Principal Contributors: L. Lois
Y. Orechwa
M. Mitchell
J. Carew (BNL)

Date: November 14, 2000