Consumers Power

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General Offices: 1945 West Parnall Road, Jackson, MI 49201 • (517) 788-0550

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DOCKET 50-255 - LICENSE DPR-20 - PALISADES PLANT -
COMPLIANCE WITH PRESSURIZED THERMAL SHOCK REGULATION 10CFR50.61 AND REGULATORY
GUIDE 1.99 REVISION 2 (TAC NO. 59970)

Consumers Power Company (CPC) submittal on April 3, 1989 provided a revised report on reactor vessel fluence for Cycles 1 - 8. Attached is the vessel fluence reduction report describing the effect of incorporating low-leakage fuel management for the Cycle 9 core loading pattern. In this proposed Cycle 9 design, 16 thrice-burned fuel assemblies with zircaloy-clad hafnium absorber rods will be used at the selected core peripheral locations to protect the vessel axial welds from neutron fast flux E>1.0 MeV. Remaining core peripheral locations will be loaded with twice-burned fuel assemblies. All once-burned and fresh fuel assemblies will be inside the core away from the peripheral locations.

This report reflects results based upon the development of in-house methodology utilizing the DOT 4.3 discrete ordinates transport code and Reactor Engineering Analyses performed during the period of 1987-1990. It concludes that the PTS screening criteria will be exceeded at the axial welds in September, 2001, as opposed to the previously reported exceed date of March, 2002. The difference reflects an improvement in vessel flux reduction in Cycle 9 relative to Cycle 8 and slightly higher vessel flux levels calculated by the refined in-house transport methodology relative to the Westinghouse methodology previously utilized. Thus, the previously

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derived conclusion that the flux reductions achieved in the Cycle 8 and 9 core loading patterns are, by themselves, insufficient to allow plant operation to the current expected end of life in 2011 remains valid. Further measures, eg, greater flux reduction, Regulatory Guide 1.154 analysis, vessel shielding etc, are necessary to allow plant operation to the nominal end of plant life and beyond.

Richard W Smedley
Staff Licensing Engineer

CC Administrator, Region III, USNRC
NRC Resident Inspector - Palisades

ANALYSIS OF THE REACTOR PRESSURE VESSEL FAST NEUTRON FLUENCE AND PRESSURIZED THERMAL SHOCK REFERENCE TEMPERATURES FOR THE PALISADES NUCLEAR PLANT

May 1990

Performed by the
Reactor Engineering Department
Palisades Nuclear Plant
Consumers Power Company

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Cycle 8. The remaining core peripheral locations will be loaded with twice burned fuel assemblies. All of the new fuel assemblies will be located within the core interior.

In this report, cycle-specific calculations have been performed for Cycles 1 through 9. Results presented address the accumulated vessel fluence through the end of Cycle 7 as well as the flux reductions obtained for the Cycle 8 (currently in operation) and Cycle 9 (under design) low leakage loading patterns. Vessel fluence limits based on the 10CFR50.61 PTS screening criteria and both the 10CFR 50.61 and Regulatory Guide 1.99, Revision 2, reference temperature correlations are calculated based on the vessel material chemistries. Vessel lifetimes are calculated relative to the fluence limits assuming the flux-reduction fuel management for Cycle 9 and beyond utilizing the Regulatory Guide 1.99, Revision 2 reference temperature correlations. In addition, details are provided about the in-house methodology and data [3] and the status of Consumers' in-house flux reduction and measurement program.

1.0 INTRODUCTION

Consumers Power Company previously submitted to the NRC a report describing Cycle 8 fluence reduction measures for the Palisades Nuclear Plant reactor pressure vessel [1]. It was committed that an additional fluence report reflecting Cycle 9 fuel management and extrapolated to nominal plant end-of-life would be submitted to the NRC. The information contained herein is intended to address the fluence re-evaluation and reduction program as previously committed to and to describe the methodology utilized for determining vessel incident fast fluxes and fluence levels.

In order to accurately calculate pressure vessel fluence levels, in-house methodology was developed utilizing the DOT 4.3 discrete ordinates transport code as the base model. Training in the use of DOT 4.3 and associated cross section libraries and support codes was obtained from Combustion Engineering. The scope of the training included code usage and model development as well as results evaluations. In-house methodology of flux calculations was further refined via consultation with Westinghouse Electric Company, Radiation and Systems Analysis-Nuclear Technology Division. Westinghouse determined that Consumers Power's neutron transport methodology represented state-of-the-art practice consistent with Westinghouse methodology [2,8].

The modeling of the vessel and fluence analysis was performed using DOT 4.3 and the SAILOR cross-section library. Cycles 1 through 7 core loading patterns were typical of out-in fuel management in that the fresh fuel was placed on the core periphery. This approach results in the maximum overall core neutron leakage and flux to the reactor pressure vessel. The Cycle 8 core was loaded with thrice burned fuel assemblies with stainless steel shielding rods located near the axial weld locations. In the previously submitted report [1], flux reductions of a factor of two were achieved at the axial weld locations from the Cycle 8 loading pattern. The design goal for Cycle 9 was to meet or exceed the flux reductions achieved in Cycle 8. The proposed Cycle 9 loading pattern consists of thrice burned fuel assemblies with hafnium absorbers located at the same core peripheral locations that utilized stainless steel shielding rods in

2.0 SUMMARY

Neutron transport calculations were performed using the DOT 4.3 computer code and SAILOR cross section library. The 2D R-θ neutron fluxes (E>1.0 MeV) were computed using DOT 4.3 with consideration of axial flux peaking. For each of Cycles 1 through 9, cycle specific DOT runs have been made. For Cycles 1 through 7 on-line core monitoring energy generation data and actual cycle operational history data were utilized for vessel flux and fluence calculations; calculations for Cycles 8 and 9 utilized predictive core simulator data. A comparison between calculated and measured fluxes at the W-290 wall capsule location, analyzed at the end of Cycle 5, was made. It was found that the calculated fluxes were about 4% higher than the measured values, thus assuring reasonable flux predictions for the models.

Flux levels for Cycle 8 were compared with that of Westinghouse methodology [1]. The in-house model indicates a positive bias in the flux calculations relative to the Westinghouse methodology and this bias varied with the azimuthal locations. Maximum variation was on the order of about 12% at 45° location [3].

Pressure vessel fluence limits based on the PTS screening criteria of 10CFR50.61 were calculated using the reference temperature (RT) correlations of both 10CFR50.61 and Regulatory Guide 1.99, Rev 2 using the vessel chemistries provided in Reference 4. The results are summarized in Table 2.1 and show the dramatic reduction in the vessel weld fluence limits with the use of the Regulatory Guide 1.99 RT correlation. With the pending issuance of a revised 10CFR50.61 incorporating the Regulatory Guide 1.99, Rev 2 RT correlation, the more restrictive Regulatory Guide fluence limits were utilized in this study.

Core loading patterns for Cycles 8 and 9 are designed to provided substantial flux reduction at the axial weld locations in comparison to previous cycles. The associated flux reductions for the primary vessel materials are shown in Table 2.2. Fast flux (E>1.0 MeV) reductions of more than 50% were obtained at the axial weld locations for Cycles 8 and 9 in comparison with Cycle 7. For Cycle 8, at the circumferential weld and base metal (peak) locations, about 20%

flux reduction was obtained. However, for Cycle 9 the flux reductions are on the order of 48% at these locations.

Vessel lifetimes based on when the PTS screening criteria are met were determined for fuel management schemes with flux reduction for Cycles 8, 9, and beyond. Operation beyond end of Cycle 8 (September 1990) was assumed to occur at 75% capacity. With no flux reduction utilized, the PTS screening criteria would be exceeded at the axial welds in 1995; utilizing Cycle 8 flux reductions, this would be extended to 2000. With flux reduction incorporated in Cycle 9 and beyond, the PTS limit would be exceeded at the axial welds again, but not until about September, 2001. These predicted dates are far short of the assumed nominal plant operating license expiration date of March, 2011.

While the flux reduction obtained in Cycles 8 and 9 substantially reduced the axial weld flux levels, the reduction is insufficient to remain within the PTS screening criteria through the minimum plant life (nominal end of operating license). Some additional flux reduction will be possible through more aggressive low-leakage fuel management in Cycle 10 and beyond. However, in order to allow plant operation at least until the nominal license expiration date, additional PTS-addressing measures will have to be implemented (eg, Regulatory Guide 1.154 analysis, vessel shielding, etc). Activities are currently underway with the Combustion Engineering Owners' Group in the areas of additional vessel surveillance data and model development for a Regulatory Guide 1.154 analysis; initial conceptual discussions are underway with other vendors for incorporation of weld specific vessel shielding in Cycle 10.

An ex-vessel dosimetry program was developed by Westinghouse and hardware installation was completed during the end of Cycle 7 refueling outage. This program would supplement the existing surveillance program. In addition to the ex-vessel program, Combustion Engineering will install an in-vessel dosimetry capsule at the W-290 capsule holder vacated following Cycle 5. These in-vessel and ex-vessel dosimetry programs will provide measured data for use in vessel wall and vessel support fluence evaluations.

Updates on vessel fluence levels and adjusted reference temperatures will be provided to the NRC as actual operational data including vessel dosimetry information is obtained. In addition, developments in fuel management, vessel materials information, vessel shielding and other PTS-related areas that substantially impact the vessel lifetime will be reported as required in 10CFR50.61.

3.0 METHODOLOGY

3.1 Overview

The pressure vessel fast neutron fluence levels (E>1.0 MeV) were calculated utilizing available historical and predictive fuel cycle information.

The primary analytical model was based on a two dimensional (R,θ) discrete ordinates code DOT 4.3 representation [5] of the Palisades reactor vessel configuration. The representation includes a model of the core/vessel geometry, the neutron source distribution, and nuclear interactions as represented by cross section data. Measurement data was available for comparison from an analysis of radiometric dosimeters irradiated in the W-290 vessel wall surveillance capsule [6], which was removed at the end of Cycle 5. The measured fast neutron flux as calculated from the measured activities using reactor power history, dosimetry cross sections and basic nuclear data was used to compare the DOT calculated neutron fluxes for Cycles 1 through 5. Individual DOT calculations for remaining Cycles 6 through 9 were also made. To-date fluence levels were calculated and end-of-life fluence levels were extrapolated based upon anticipated capacity factors for the remaining life of the Palisades Plant.

3.2. Fuel Management

Palisades followed a standard out-in fueling scheme through Cycle 7 (Figure 3.1). In this scheme, only fresh fuel was placed around the core periphery. This approach results in the maximum overall core neutron leakage and fast flux to the reactor vessel, but minimizes power peaking and generally provides the greatest thermal margin.

Utilization of the Regulatory Guide 1.99, Rev 2, reference temperature correlations for comparison to the 10CFR50.61 PTS screening criteria determined that the axial welds would be responsible for limiting the life of the Palisades reactor vessel. It was decided to alter the fuel management strategy to distribute the power away from these critical weld locations for Cycle 8 operation. A low leakage loading pattern was adopted to improve the neutron economy and to reduce the fluence levels at the axial welds.

A total of 16 thrice-burned stainless steel shielded assemblies were installed at the core periphery. In addition, eight twice burned assemblies were placed on the core periphery. The remaining 24 peripheral locations were filled with fresh fuel assemblies (Figure 3.2). With this arrangement, it was anticipated that the reduced power in the peripheral assemblies would reduce the primary source of fast neutrons reaching the reactor vessel axial welds.

Design of the Cycle 9 core is based upon 52 fresh, 60 once-, 76 twice-, and 16 thrice-burned fuel assemblies. All thrice burned assemblies will have zircaloy-clad hafnium rods placed in eight guide tube locations. These assemblies will be placed on the edge of the core near critical weld locations (Figure 3.3). Hafnium is an effective absorber primarily for neutrons in the thermal through epithermal energy ranges. It is anticipated that the power in these thrice-burned fuel assemblies will be greatly reduced along with the neutron source. Therefore, there will be fewer neutrons reaching the vessel at the critical weld locations.

3.3 Geometry

The Palisades reactor exhibits one-eighth (1/8) core symmetry, thus only a zero to 45 degree sector has been included in the DOT model (Figure 3.4). In this figure two surveillance capsules attached to the inner vessel wall are shown. A plan view of the Palisades capsule arrangement is shown in Figure 3.5, with specific surveillance capsules dimensions shown in Figure 3.6. Figure 3.5 shows that four of the 45 degree sectors do not have any capsules. Two other sectors have one accelerated (attached to core support barrel) and one wall capsule. The remaining two sectors have two vessel wall capsules at the 10° and 20° locations. The utilized DOT model contains two wall capsules at the 10° and 20° locations. This model utilizes 99 radial and 98 azimuthal intervals for a total of 9702 meshes in polar (R, θ) geometry. Fine mesh detail has been utilized as necessary in setting up the geometry model to accurately represent the reactor core, shroud, bypass flow, core support barrel, inlet flow, surveillance capsules, vessel clad and the vessel wall regions. A total of 15 outer assemblies have been modeled to represent the detailed core; the total model mesh extends to just outside the vessel in the reactor cavity area. Various regions of the DOT model are represented in such a way that their volumes are close to that of the physical volumes of the reactor internals.

3.4 Material Cross Sections

The DOT model analysis employed a P_3 expansion of the scattering cross sections. The microscopic cross sections used in the analysis were obtained from the SAILOR cross section library. Macroscopic cross sections were calculated for each region in the model using the computer code GIP. Plant specific material compositions and the corresponding atomic densities were used for this analysis.

3.5 Neutron Source

Assembly-wise radial power distributions were obtained from the Palisades incore monitoring system (INCA) for Cycles 1 through 7; fuel vendor-generated discrete PDQ bundle power data were used for Cycles 8 and 9. Average energy generated by fuel assemblies was obtained from the exposure data and the heavy metal weight of the assemblies to calculate cycle average assembly powers. Figures 3.7 through 3.15 exhibit the fifteen (15) outer peripheral normalized bundle powers for Cycles 1 through 9. Cycle 8 actual assembly power data to date is adequately modeled by utilizing the predictive core simulator information. Local pin power distributions were derived from discrete PDQ model calculations. The local pin power distributions and the average assembly powers were combined to determine core normalized pin power distributions.

Axial peaking was accounted for by applying the bundle specific axial peaking factors to the normalized pin powers of the fifteen modeled fuel assemblies. This approach conservatively defines the axial variation of the vessel incident neutron/source. Axial power information was obtained from INCA core monitoring data for Cycles 1 through 7 and 3D XTG core simulator models for Cycles 8 and 9. The core power distributions were initially calculated in Cartesian (x,y) geometry from the original data sources. The Cartesian geometry was converted to a polar (R,θ) geometry using an algorithm that maintained equivalent average source strength over the affected surface area between coordinate systems.

3.6 Burnup Corrections

As the fuel starts to deplete in the core during plant operation, exposure of the individual fuel assemblies increases and a build up of plutonium isotopes occurs. Plutonium isotopes have higher υ (neutrons/fission) and κ (energy/fission) values and exhibit fission spectra shifted towards the higher energies (harder spectra) than uranium isotopes. The contributions of the individual isotopes U235, U238, Pu239 and Pu241 to the core neutron source have been accounted for in the present set of flux calculations. Since the fission spectra and effective neutron yield differs for the above isotopes, the core neutron source and the vessel wall flux will generally increase with the fuel depletion for given peripheral assembly power levels. This is especially important for the twice and thrice burned fuel at the core periphery for Cycles 8 and 9. Composite fission spectra for each of Cycles 1 through 9 have therefore been developed. Individual isotopic fission spectra were obtained from ENDF-B/V for the uranium and plutonium isotopes. The spectra were collapsed to 47 energy groups similar to the SAILOR Library [7]. The exposure dependent neutron source for each cycle was then determined by weighting the individual group-wise neutron yields with the corresponding exposure dependent isotopic fission fractions based on the cycle average exposure of five peripheral assemblies. Only 19 groups above 1 MeV have been employed in the DOT model for the fast flux calculations. Cycle specific fission spectra are shown in Table 3.1 in comparison with the SAILOR Library fission spectra. Fission spectra are normalized to one (1) neutron in the 47 groups, similar to the SAILOR Library. From Table 3.1, it can be noted that high energy neutron groups have higher yields for Cycles 8 and 9, compared to the previous seven cycles.

In the DOT model, cycle-specific υ/κ ratios for fifteen (15) fuel assemblies were obtained from CASMO lattice depletion code data for a standard Palisades fuel type, utilizing middle-of-cycle exposure values. The effect of neutron yield and energy generated in these assemblies were incorporated in the neutron source. These effects are more important on the fast flux at the reactor vessel for Cycles 8 and 9 as compared to previous Cycles 1 through 7.

3.7 Neutron Transport Analysis

The spatial distribution of neutron flux in the reactor was calculated using the DOT 4.3 computer code. The DOT program solves the Boltzman transport equation in two-dimensional geometry using the method of discrete ordinates. Third order scattering (P_3) and S_8 angular quadratures were used. The cycle-by-cycle neutron flux distributions were calculated using the cycle-dependent neutron sources and material compositions.

3.8 Vessel Fluence Calculations

Fluence levels of a given cycle were obtained by multiplying the flux at the clad-base metal interface by the effective full power seconds at 2530 MWTH for that cycle. Accumulated fluence values at the EOC 8 were calculated by adding the fluence for all the Cycles 1 through 8. Further extrapolation to end-of-life fluence is based upon the estimate that the plant will operate at 75% capacity factor after EOC 8 at the calculated fluence rate for the Cycle 9 proposed core loading scheme.

3.9 Fluence Limits/Reference Temperature Calculations

Target fluence limits for pressure vessel welds and base metals are calculated using the 10CFR50.61 correlation for RT_{PTS} and the vessel material PTS screening criteria. The reference temperature correlation is given as:

$$RT_{PTS} = I + M + (-10 + 470Cu + 350CuNi)f^{0.270}$$

where:

RT_{PTS} is the adjusted reference temperature for pressurized thermal shock considerations (°F)

I is the initial reference temperature (°F)

M is the margin term (°F)

Cu, Ni are the copper and nickel content (in weight percent), respectively

f is the accumulated fluence (E>1.0 MeV)) in units of $10^{19} n/cm^2$

The corresponding fluence limits are determined by solving the RT correlation for the fluence value. Initial reference temperature and chemistry information and corresponding fluence limits are shown in Table 3.2.

Target fluence limits for pressure vessel welds and base metals are also calculated using Regulatory Guide 1.99, Rev 2 reference temperature correlation and the 10CFR50.61 PTS screening criteria. The adjusted reference temperature for each material in the beltline is given as:

$$ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin}$$

or

$$ART = I + M + \Delta RT_{NDT}$$

where: $\Delta RT_{NDT} = (CF * f^{(0.28-0.10 \log f)})$

I, M and f have the same meaning as above. The chemistry factor, CF (°F) depends on the content of copper and nickel in the belt line materials. This factor is provided in Regulatory Guide 1.99, Rev 2. The corresponding fluence limits are determined by solving the RT correlation for the fluence value and are shown in Table 3.3.

For each Cycles 1 through 9, fluence values were obtained for the base metal, and axial and circumferential weld materials. Using the parameters of Table 3.3 and the accumulated fluence at the end of each cycle, the corresponding adjusted reference temperatures were calculated.

4.0 RESULTS

4.1 Comparison to Measured Data

The W-290 surveillance capsule was removed at the end-of-Cycle 5 and was analyzed by Westinghouse [6]. The measured average flux at the W-290 capsule was corrected for a discrepancy in the power irradiation history data (Figure 4.1) versus that utilized in Reference 6. The corrected measured flux at the W-290 capsule was $6{\cdot}73\times10^{10}$ n/cm^2-sec. DOT calculations for Cycles 1 through 5 provide a cycle-energy averaged flux at the W-290 locations of $7{\cdot}02\times10^{10}$ n/cm^2-sec [3], 4% higher that the measured value. It was also noted that the lead factors obtained for Cycles 1 through 5 from DOT calculations were fairly constant (between 1·24-1·27). These facts indicate that calculated flux values from the in-house DOT results can be directly used for reasonable end-of-life fluence calculations. It should be noted that these results are a slight improvement over the DOT calculations utilized in Reference 1, which exhibited a +11% bias relative to the measured W-290 fluxes.

4.2 Flux/Fluence Distribution

For Cycles 1 through 9, the maximum fast flux occurs at the azimuthal interval between 16·44°-17° at the clad-base metal interface (Table 4.1). Flux distributions for Cycles 3 through 7 are very similar. Comparison of flux distribution between different cycles is presented in Figures 4.2 and 4.3. These figures confirm that the maximum flux occurs around 17°. Wall capsules at 10° and 20° exhibit an attenuating effect in their immediate vicinities, but do not affect the peak fluxes. For Cycles 1 though 7, a second peak occurs around the 32° azimuthal location. For Cycles 8 and 9, this peak is eliminated as a result of the implementation of low leakage fuel management schemes. Substantial flux reduction for the low leakage fuel management schemes relative to the high neutron leakage loading patterns is apparent. Radial flux distributions at the 0, 17 and 30 degree azimuthal locations for Cycles 7 (representative of previous cycles), 8 and 9 are presented in Appendix 7.1.

Accumulated fast fluence distributions at the end of Cycle 9 and EOL at the clad-base metal interface is shown in Figure 4.4. Based upon Reg. Guide 1.99, Revision 2, fluence limits corresponding to base metal, axial, and circumferential welds are also presented in Figure 4.4. From this figure it can be noted that the fluence values at the axial welds at 0° and 30° are limiting the life of the Palisades reactor pressure vessel.

Table 4.2 summarizes the cycle specific fluence ($\Delta\Phi$) and accumulated fast fluence ($\Sigma\Phi$) at the clad-base metal interface for each of Cycles 1 through 9. For the selected azimuthal locations: 0° (axial weld location), 17° (maximium of peak at base metal), 30° (axial weld location) and 45°, effective full power years (ΔEFPY) for each cycle and the accumulated EFPY's are also presented. Table 4.3 provides the fluence limit violation dates with Cycle 9 fluence rates for plant operations beyond the end of Cycle 8 date of September, 1990.

4.3 Calculational Uncertainty

A number of factors contribute to the uncertainty in the projected peak fast fluence at the reactor vessel wall. These factors are due to the conversion of measured activity data to fluxes, uncertainties in material composition, neutron cross sections, power distributions, as-built core/vessel dimensions and cycle-by-cycle variation in the fast flux lead factors. An uncertainty of ±25% is estimated in the calculated vessel wall fluence, typical of current neutron transport methodology uncertainties. The calculated +4% flux bias relative to actual W-290 measured fluxes indicates that vessel wall flux predictions are reasonable given the inherent uncertainty in the methodology.

4.4 Adjusted Reference Temperatures and Screening Limits

Adjusted reference temperatures (ARTs) as a function of effective full power years (EFPYs) corresponding to the fluence values at the end of Cycle 1 through 9 and projected to plant EOL, have been plotted in Figure 4.5. PTS screening limits for each of the beltline materials are provided. This figure suggests that the axial welds are the limiting material for the Palisades reactor pressure vessel relative to PTS limits. Table 4.4 provides the summary of PTS adjusted reference temperatures for base metal, axial and circumferential weld materials. Note that for the licensed end-of-life date of March, 2011, ARTs for the axial welds at 30 degrees exceed the PTS screening limit of 270°F.

5.0 DISCUSSION

5.1 Impact of Results

Modifications to the Cycles 8 and 9 loading patterns substantially reduce the flux at the critical weld locations and delays exceeding the PTS screening criteria to about September 2001, as opposed to in 1995 if no flux reduction measures are taken. The flux reduction is insufficient, however, to allow operation of the plant within the PTS screening criteria until the minimum expected plant life, corresponding to the expiration of the pending full term operating license in March, 2011.

In-house flux calculations have a positive bias with respect to Westinghouse model [1], mainly due to the slightly larger core size in the in-house model. The bias ranges from +0.5% at 0° and increases to about +11.7% at the 45° location. In addition, more realistic plant-specific design and operational data have been utilized in the in-house model. This approach therefore does not depend very heavily on assumptions used for the flux calculations, but relies on the plant specific parameters.

In order to maximize vessel lifetime, further measures must be taken in the areas of greater flux reduction, Reg Guide 1.154 analysis to properly define the real Palisades PTS risk, and possible vessel annealing/shielding actions to reduce the accumulated vessel embrittlement rate.

5.2 Additional Flux Reduction

The most straightforward method of reducing the vessel fast flux level is reduction of the source itself, which has been initially addressed with the incorporation of low-leakage fuel management and stainless steel shield rods in Cycle 8 and thrice burned fuel with hafnium absorbers for Cycle 9. While flux reduction gains are predicted for Cycle 9, some further reductions are believed to be obtainable via fuel management alone. Cycle 9 will be the first cycle with the new steam generators installed. The new generators are expected to provide substantially higher primary coolant flow than the current generators.

The increased flow, which can be quantified accurately during Cycle 9 operation, will provide additional core operating thermal margin and thus allow higher power peaking limits to be utilized in developing the Cycle 10 loading pattern. The higher peaking will provide additional fuel management flexibility and support more aggressive low-leakage fuel management for further reductions in vessel wall fluxes.

Additionally, Cycle 9 will be the first cycle to incorporate a new high thermal performance (HTP) spacer grid design in the fresh reload fuel. Insertion of a second reload of fuel with the HTP spacers in Cycle 10, along with development of a Palisades-specific DNB correlation for the HTP fuel, will provide additional allowable peaking factor increases to be utilized in Cycle 10.

A third area design to allow greater fuel management flexibility in the Cycle 10 core design will be the installation, utilization, and optimization of a new full core power monitoring system beginning in Cycle 9. This monitoring system will allow the Cycle 10 loading pattern design to utilize 1/4 core symmetry, as opposed to current 1/8 core symmetry utilized in Cycles 1-9, and will provide more options for reducing power and flux levels in peripheral fuel assemblies.

Discussions have been held with NSSS vendors on the possibility of installing critical material area neutron shields. A shield between the fuel and the vessel wall would act to reflect, slow down, or absorb high-energy neutrons before they could reach the vessel wall. Stainless steel shielding pads could be designed to mount near the core support barrel to maximize the attenuation of the high energy neutrons of concern. The possibility exists to use other hybrid materials which are better neutron shielding then stainless steel and therefore provide further neutron flux reduction beyond that attainable with low leakage fuel management alone. It is estimated that internal vessel shielding could reduce the flux at the critical axial weld locations a minimum of 25%.

5.3 Refined Flux Measurement

In order to benchmark vessel fluence calculations, an upgraded vessel dosimetry program has been initiated to supplement the existing surveillance capsule program. An ex-vessel dosimetry program was developed by Westinghouse and hardware installation occurred during the end of Cycle 7 refueling outage. The dosimetry installed will provide detailed azimuthal and axial mapping of the 270-360 degree vessel quadrant, with gradient chains installed in the other three quadrants to provide accurate axial and cross-quadrant mapping. It is intended to exchange this dosimetry at the end of Cycle 8 with similar sets of dosimeters for the Cycle 9 irradiation period. The dosimetry will provide measured data for use in vessel wall and supports fluence evaluations. In addition to the ex-vessel program, Combustion Engineering has been contracted to fabricate and install a replacement in-vessel dosimetry capsule to be inserted into the W-290 capsule holder vacated following Cycle 5. Installation will occur during the next refueling outage (Fall 1990). When installed, this capsule will provide an excellent through-wall correlation with the ex-vessel dosimetry installed in the same quadrant.

In addition to implementing the supplemental dosimetry program, efforts will be made to extend the DOT model up to the reactor cavity area to analyze the ex-vessel dosimeters. A further enhancement planned to the DOT model will be to synthesize a 3-D model for flux calculations to remove some of the inherent conservatism in the calculations due to utilization of the bundle-specific peak axial power over the entire core axial height.

5.4 Other PTS Activities

Planned flux reduction measures do not appear to fully solve the vessel fluence issue relative to PTS. Consumers Power Company is pursuing a methodology through the Combustion Engineering Owners Group (CEOG) to augment plant data by correlating surveillance material and data from other plants to Palisades vessel materials. Such data could allow Palisades to reduce operating restrictions caused by Regulatory Guide 1.99, Rev 2/10CFR50.61 default margin terms and initial reference temperatures for generic weld material in absence of actual Charpy weld test specimen data.

A detailed risk evaluation based on Regulatory Guide 1.154 analysis is also being pursued through CEOG. Such analysis will identify and summarize the potential risk of a PTS event occurring. This risk would be based on the known operating activities or transients which could lead to a PTS event. The program is being undertaken in a phased approach with the currently in-progress Phase I dealing with generic model development only. The analysis, if actually needed, would be completed at least three years prior to the predicted exceed date of the PTS screening criteria.

6.0 REFERENCES

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4. Letter from K W Berry to NRC, "Response to Request for Additional Information - Pressurized Thermal Shock (PTS) Rule 10CFR50.61," August 7, 1986.

5. RSIC Computer Code Collection DOT IV Version 4.3 (Report No. CC-429).

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7. RSIC Library Collection SAILOR DLC-76.

8. Telecopy of E.P. Lipincott (Westinghouse) to O.P. Jolly (CPCo), "Final Report on Westinghouse Review of Consumers Power PTS Calculations," April 20, 1990.