Consumers Energy

A CMS Energy Company

Palisades Nuclear Plant
27780 Blue Star Memorial Highway
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U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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DOCKET 50-255 - LICENSE DPR-20 - PALISADES PLANT
PALISADES REACTOR VESSEL NEUTRON FLUENCE REEVALUATION

In a letter dated April 4, 1996, Consumers Energy provided an updated analysis of reactor vessel fluence and a revised estimate of the date at which the screening criteria of 10 CFR 50.61 would be reached for the Palisades Plant. The NRC concurred, in part, with this submittal in an interim SER issued on December 20, 1996. This SER provided NRC's estimate that the 10 CFR 50.61 screening criteria would be reached in approximately 2003 (based on then-existing estimates of fluence accumulation rates for subsequent cycles), and stated that NRC review of the remainder of the submittal would continue. Later projections by Consumers Energy, as reported in a letter dated September 8, 1998, indicate that the limiting reactor vessel weld would be expected to reach the screening criteria in early 2004.

Subsequently, during a meeting on October 19, 1998, the NRC and Consumers Energy agreed: (1) that Palisades reactor vessel fluence values determined by calculations adjusted to plant specific dosimetry measurements were not likely to be approved by the NRC staff, and (2) NRC might consider approval of fluence calculations adjusted to industry average data. At a meeting on December 7, 1998, it was further agreed that Consumers Energy would make additional submittals modifying the fluence calculation to incorporate industry average data and to reflect changes in plant parameters which have defined physical bases.

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A Consumers Energy submittal dated March 25, 1999, provided a revised estimate of fluence that was determined by adjusting the calculated Palisades fluence using industry average data. Subsequent communications regarding this submittal resulted in the conclusion that fluence calculations adjusted to industry average data were not likely to be approved without additional justification. As a result, further activities in this area were suspended.

This submittal provides new fluence estimates based on refined calculations that incorporate changes to plant parameters which have defined physical bases. In addition, recommendations of NRC and industry experts have been implemented in this submittal to incorporate axial leakage effects in the determination of the reactor pressure vessel fluence and the response of in-vessel and reactor cavity dosimetry. This submittal also provides new fluence projections which account for fluence reduction methods which have been implemented as of the current fuel cycle (Cycle 15).

This submittal also updates the reactor vessel fluence analysis of record in order to include data from an additional surveillance capsule and from operating cycles 12, 13, and 14 completed since our April 4, 1996, submittal. This additional dosimetry data from SA-60-1 supplemental surveillance capsule, is part of Palisades' measurement database which is used to validate calculations performed to estimate the fracture toughness of the Palisades reactor vessel.

The Palisades reactor vessel remains limited by the beltline axial welds fabricated with weld wire heat number W5214. It was estimated in our September 8, 1998, submittal that the PTS screening criterion of 10 CFR 50.61 will be reached for limiting weld material at an accumulated fluence of 1.58×10^{19} n/cm^2. It is assumed in this evaluation that fluence will continue to accumulate in future operating cycles at a rate comparable with that determined for Cycle 15. Based on this submittal and the Cycle 15 fluence accumulation rate, the Palisades reactor vessel is not expected to reach the PTS screening criteria until 2014.

Attachment 1 provides an overview of changes incorporated into WCAP-15353, "Palisades Reactor Pressure Vessel Neutron Fluence Evaluation," Revision 0, with respect to our previous fluence submittal dated April 4, 1996 (WCAP-14557, Revision 1, "Consumers Power Company Reactor Vessel Neutron Fluence Measurement Program for Palisades Nuclear Plant - Cycle 1 Through 11").

Attachment 2 contains WCAP-15353, Revision 0.

Consumers Energy requests that the NRC review and approve WCAP -15353 which utilizes the improved inputs and methods discussed. It is also requested that NRC endorse the new date at which the reactor vessel is estimated to reach the PTS screening criteria. Approval is requested by August 31, 2000. Consumers Energy would be pleased to meet with the staff to discuss this submittal at your convenience.

The NRC is also requested to discontinue any remaining activity related to our previous submittal of March 25, 1999 (TAC MA5242). Any remaining issues regarding this previous submittal should be considered superseded, and the TAC number may be closed.

SUMMARY OF COMMITMENTS

This letter contains no new commitments and no revisions to existing commitments.

Daniel G. Malone
Acting Director, Licensing

CC Administrator, Region III, USNRC
Project Manager, NRR, USNRC
NRC Resident Inspector - Palisades

Attachments

ATTACHMENT 1

CONSUMERS ENERGY COMPANY
PALISADES PLANT
DOCKET 50-255

February 21, 2000

SUMMARY OF CHANGES
INCORPORATED INTO WCAP-15353
REVISION 0
WITH RESPECT TO CONSUMERS ENERGY'S FLUENCE
SUBMITTAL DATED APRIL 4, 1996

6 Pages

SUMMARY OF CHANGES INCORPORATED INTO WCAP-15353 REVISION 0 WITH RESPECT TO CONSUMERS ENERGY'S FLUENCE SUBMITTAL DATED APRIL 4, 1996

Additional Operating Data

Palisades has completed three additional operating cycles (Cycles 12 - 14) since the previous fluence submittal of April 4, 1996. Further, one additional accelerated surveillance capsule dosimetry set has been added to the measurement data base, increasing the plant-specific total to five in-vessel dosimetry sets and thirteen ex-vessel dosimetry sets.

Core Modeling and Source Term Changes

In our April 4, 1996, submittal the fluence calculations utilized Studsvik's CASMO-3 transport cross-section generator code and the SIMULATE-3 advanced nodal diffusion core simulator code to create the source term for DORT calculations. The Studsvik CASMO/SIMULATE codes were chosen to generate DORT input data for several reasons:

- To achieve consistency among the multi-cycle calculations
- Because of their widespread use throughout the US and international nuclear industry
- Due to the better predictive capabilities of this code package compared to those previously used at Palisades

In this submittal, DORT source terms were generated using CASMO-4 and SIMULATE-3. According to Studsvik, CASMO-4 models light water reactor fuel more accurately than CASMO-3 as a result of the following model changes:

- Expanded resonance calculations which include Gadolinium (Gd) isotopes and additional Plutonium (Pu) isotopes
- New 2-D transport solution -- Method of Characteristics with heterogeneous geometry (CASMO-3 used homogenized cells)
- Expanded pin cell calculation which models Gd explicitly (Previously, Gd was modeled using MICBURN with an approximated fuel buffer region)
- Improved 2-D macrogroup calculation
- Expanded depletion chains for Gd and Erbium

Examination of the differences between results from CASMO-3/SIMULATE-3 and those from CASMO-4/SIMULATE-3 revealed that the incorporation of additional Pu isotopes into the resonance calculations is one physical basis for the changes in the cycle-burnup-averaged relative radial power distribution. Changes in the cycle-burnup-averaged relative radial power distribution ultimately affect the reactor vessel fluence. Palisades observed more power being generated in the interior of the core (on a cycle-averaged basis) using CASMO-4/SIMULATE-3 compared to CASMO-3/SIMULATE-3. This effect was more pronounced for the early out-in loading pattern cycles (1 – 8) where most of the higher burnup fuel (containing relatively large amounts of Pu) resided in the interior of the core.

Another reason for changes in the cycle-burnup-averaged core radial power distribution using CASMO-4/SIMULATE-3 was due to assemblies with large amounts of Gadolinium. These assemblies reached peak k_∞ faster than those assemblies modeled using CASMO-3/SIMULATE-3. This is a product of the expanded resonance calculations and the explicit modeling of the Gadolinium in CASMO-4. These items result in faster Gadolinium burnout and hence shorter reactivity hold-down in fresh assemblies. Cycles 9 –15, which contain large numbers of fresh assemblies with significant amounts of gadolinium in the interior of the core, manifest this small, but physical effect. The vessel fluence, therefore, is affected by increasing the cycle-burnup-averaged assembly relative power (compared to CASMO-3/SIMULATE-3 models) for assemblies in the interior core region.

A less significant change to the SIMULATE-3 model used to generate the latest source terms is the incorporation of a more accurate representation of the assemblies located on the peripheral flats. These assemblies are no longer artificially expanded into an artificially large peripheral planar node, as was done in the past. Specifically, SIMULATE-3 can only model one planar node size. Palisades, however, has different water gap sizes (wide and narrow water gaps) which surround the assemblies adjacent to control blades. The assemblies on the peripheral flats of the core are not adjacent to control blades, and hence have the same (narrow) water gap on all sides. Palisades' SIMULATE-3 model had to use the larger water gap to model these peripheral flat assemblies. Instead of adding more homogenized water-fuel material into the larger node size, an air box was added around the assembly to make up for the additional area.

Changes to Fluence Calculation Methodology and DORT Inputs

An important change incorporated in WCAP-15353, Revision 0, "Palisades Reactor Pressure Vessel Neutron Fluence Evaluation," relative to our April 4, 1996, submittal is the use of a 3-D Flux Synthesis technique to construct the 3-D flux profiles at the reactor vessel clad base metal interface and dosimetry locations. Dr. Richard E. Maerker, (Oak Ridge National Laboratory, retired), and the NRC have encouraged the incorporation of axial leakage effects into the DORT transport models in order to reconcile the biases seen between measurements and calculations. The three-dimensional flux solution constructed using a 2-D/1-D synthesis technique was utilized, since it adheres to the methodology described in Section 1.3.4 of DG-1053

"Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence", and is well accepted in the industry. In order to generate a DORT model which utilized this method, Palisades provided cycle-specific relative axial power data and the geometry of core shroud assemblies to Westinghouse.

In addition to the change in the basic methodology for generating the vessel fluence values, numerous changes to the DORT inputs were made. While the April 4, 1996, submittal used cross-sections and fission spectra based on BUGLE-93 in all DORT calculations, WCAP-15353 used the corresponding data from BUGLE-96. BUGLE-96 has corrected some deficiencies that have been discovered in BUGLE-93 regarding the self-shielding of steel constituents. Since deep penetration through steel is needed to reach the ex-vessel dosimetry, this change is important for accurately determining the neutron spectra at the ex-vessel dosimetry. These neutron spectra are utilized as initial estimates for the least-square spectrum adjustments made via the FERRET/SAND-II computer codes.

Further, while the current and past submittals both used S_{16} quadrature, WCAP-15353 increased the order of the Legendre Scattering Polynomial Expansions from P_3 to P_5. This change was made to enhance the group flux solution determined via the transport calculations, in particular, for the analyses involving R,Z geometry.

In addition, the DORT model in WCAP-15353 differs from that used in our April 4, 1996, submittal since a "two-zone" core is used instead of one homogenized core region. The following discussion is taken directly from WCAP-15353:

> "The geometric model that was applied in the discrete ordinates radiation transport calculations utilizes a "two-zone" core to accurately reflect known differences in the physical parameters between the radially inboard or "center" fuel assemblies versus the radially outboard or "peripheral" fuel assemblies. This modeling approach is important for plants that utilize low-leakage loading patterns since the center assemblies typically operate at an average relative power slightly greater than 1.0 and the peripheral assemblies operate at a considerably lower value. With less power being generated in the peripheral assemblies, less heat is transferred to the local water region resulting in a lower temperature rise; thus the water density of the peripheral assemblies is some what greater than the center assemblies. Hence, utilization of a two-zone core model should reduce the neutron fluence calculated at the Clad-Base Metal Interface (CBMI) of the RPV since the cooler water that surrounds the peripheral assemblies provides enhanced neutron shielding."

The "two-zone" core-water and bypass-water region densities are now calculated using data from the SIMULATE-3 code's embedded thermal hydraulic model. Previously, core average water temperatures were used for the core-water and bypass regions for Cycles 1 - 8, and bypass-water region densities for Cycles 9 - 11 were calculated using a formula based on the curve fit of incore power vs. core exit temperatures. The new modeling technique is significantly easier than the previous curve fit method and produces similar results for out-in loading patterns. In addition, the SIMULATE-3 method allows for a consistent model to be used for all cycle-specific DORT inputs (i.e. relative pin powers, water densities, ν/κ data, etc.).

An additional DORT model change made in WCAP-15353, which differs from that in our April 4,1996, submittal, includes more explicit modeling of the Cycle 10 and Cycle 11 shield rod regions. In the April 4, 1996, submittal only the outside 2 rows of stainless pins in the Shield Assembly Reload N (SAN) bundles were modeled. However, the 2 rows of stainless pins on both sides of the SAN assemblies are now modeled in WCAP-15353.

Finally, the reactor vessel cladding is now modeled as stainless steel Type 308/309 rather than Type 304 as previously assumed. This change was not previously incorporated in the determination of Cycle 1-12 fluence analyses, and did not significantly affect fluence.

<u>The PTS Screening Criterion</u>

The limiting fluence for both the base metal, the circumferential weld, and axial weld material can be calculated using the following formula derived from the 10 CFR 50.61 equation:

$$f = 10^{\left(\frac{0.28 - \sqrt{0.0784 - 0.4\log\left[\frac{(RT_{PTS} - RT_{NDT(u)} - M)}{CF}\right]}}{0.2}\right)}$$

Where,

RT_{PTS}	=	270 °F for Axial Welds and Shell Plates
	=	300 °F for Circumferential Welds
$RT_{NDT(u)}$	=	0 °F for the Limiting Shell Plate
	=	-56 °F for Axial and Circumferential Welds
M	=	34 °F for the Shell Plate
	=	66 °F for the Axial and Circumferential Welds
CF	=	158 °F for the Limiting Shell Plate (D-3803-3 heat # C-1279-1)
	=	227 °F for the Intermediate/Lower Circumferential Weld (heat #27204)

= 231 °F for the Limiting Axial Welds (heat # W5214)

The following PTS screening criteria limiting fluence values can be calculated using the formula and data previously listed.

Limiting Shell Plate Fluence[1]	=	8.58×10^{19} [n/cm^2]
Limiting Circumferential Weld Fluence	=	2.84×10^{19} [n/cm^2]
Limiting Axial Weld Fluence	=	1.58×10^{19} [n/cm^2]

[1] The Shell Plate material and the Circumferential Weld materials both are exposed to the peak vessel fluences at the 15° and 75° azimuths of the Westinghouse model. This means that the Circumferential Weld material will reach its screening criteria prior to the Shell Plate material.

Projected PTS Screening Criteria Dates

The approximate date at which the PTS screening criteria will be reached for the limiting reactor vessel beltline materials can now be determined. The method used to determine the PTS screening criteria is consistent with that used and submitted to the NRC on April 4, 1996.

The PTS screening criteria dates are determined based on the following formula:

Date = Date (EOC14) + (Limiting Fluence - Accumulated Fluence (EOC14))/Fluence Rate

Circ. Weld = $1999.95 + (2.84 \times 10^{19} - 1.56 \times 10^{19})/(1.67 \times 10^{10} \times 365.25 \times 24 \times 3600 \times 0.89)$
= 2027.2

Axial Weld = $1999.95 + (1.58 \times 10^{19} - 1.18 \times 10^{19})/(1.06 \times 10^{10} \times 365.25 \times 24 \times 3600 \times 0.89)$
= 2013.3

Projected PTS Screening Criteria Dates Including Core Power Correction

Palisades has historically used the flow indicated by two feedwater flow venturies (one on each feedwater loop) as inputs to the heat balance power calculation. The heat balance power calculation is the primary (and most accurate) method available at the plant to determine the actual reactor thermal power. Accurate measurements of the feedwater flow are required to adequately determine reactor power for the calibration of nuclear instrumentation.

The two feedwater flow venturies were calibrated prior to operation of the Plant in Cycle 1. Shortly after the beginning of Cycle 1, a flow straightener detached from one of the feedwater flow lines and impacted the downstream flow venturi. The damaged venturi was removed, repaired, and calibrated offsite prior to reinstallation at the plant. Both flow straighteners were removed from the two feedwater lines to prevent a similar occurrence.

The undamaged flow venturi, however, was not calibrated following the removal of its upstream flow straightener. This resulted in an inaccurate measurement of the feedwater flow to the associated steam generator and an over prediction of the true reactor power level. The plant operated in this condition until mid-Cycle 13 when an Ultrasonic Flow Measurement (UFM) system was placed in service. Extensive feedwater flow measurements were conducted using the highly accurate UFMs. These measurements were input into the heat balance calculation, revealing that the reactor had been operating, on average, at more than 2% below rated power when indicated power was 100%.

In order to determine the PTS Screening Criteria dates, and include the correction for operating at less than 98% actual power through mid-cycle 13, the end of cycle (EOC) burnups for Cycles 1- 13 should be adjusted. New PTS Screening criteria dates, with "UFM-correction", were determined by reducing each of the cycle lengths [EFPD] for Cycles 1 - 12 by 2%. It was elected not to incorporate an adjustment for part of Cycle 13. Cycle 14 and subsequent cycles do not require correction because nuclear instrument calibrations are now based on heat balances which use accurate values of feedwater flow from the UFMs. Following are the new projected dates at which the PTS screening criteria will be reached for the limiting Reactor vessel materials.

Circ. Weld $= 1999.95 + (2.84\times10^{19} - 1.53\times10^{19})/(1.67\times10^{10} \times 365.25 \times 24 \times 3600 \times 0.89)$
$= 2027.8$

Axial Weld $= 1999.95 + (1.58\times10^{19} - 1.16\times10^{19})/(1.06\times10^{10} \times 365.25 \times 24 \times 3600 \times 0.89)$
$= 2014.0$