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January 23, 1986

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Nuclear Reactor Regulation
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DOCKET 50-255 - LICENSE DPR-20 - PALISADES PLANT -
RESPONSE TO 10CFR 50.61 - FRACTURE TOUGHNESS REQUIREMENTS FOR PROTECTION AGAINST PRESSURIZED THERMAL SHOCK EVENTS

The final rule concerning fracture toughness requirements for protection against pressurized thermal shock events was published in the Federal Register (50FR 29944) on July 23, 1985 as 10CFR 50.61. The rule requires a submittal by January 23, 1986 of projected values of the reference temperature for pressurized thermal shock (RT_{PTS}) (at the inner beltline vessel surface) of reactor vessel beltline materials as calculated by the method specified in 10CFR 50.61, paragraph b.2.

SUMMARY

Calculations performed by Consumers Power Company show that the present maximum fluence on the Palisades reactor vessel wall inner diameter is estimated to be $1.25 \times 10^{19} n/cm^2 \geq 1Mev$ and that the bounding projection of End-of-Life (EOL) fluence is $6.8 \times 10^{19} n/cm^2 \geq 1Mev$. Employing the generic weld metal data of section b.2.i of 10CFR 50.61 results in an RT_{PTS} of 259°F for the longitudinal weld material and an RT_{PTS} of 273°F for the circumferential weld material. These values are less than the respective screening criteria of 270°F for longitudinal and 300°F for circumferential welds. Calculations of the RT_{PTS} for the reactor vessel base metal show it to

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be 271^{o}F which is essentially the screening criteria of 270^{o}F. The value of 271^{o}F for RT_{PTS} of the base metal was arrived at by substituting a margin value ("M" in equation 1 of 10CFR 50.61) of 10^{o}F rather than the 48^{o}F value supplied in 10CFR 50.61. The basis for the 10^{o}F margin is presented in the discussion section below. The results presented above assume that for the remaining Plant life, the Plant will operate at an 80 percent capacity level and that no significant changes will be made in the current core configuration. Therefore, although low leakage reactor cores for the Palisades Plant have been studied, no specific programs have as yet been adopted. Consumers Power Company will however, continue to review the Palisades reactor vessel wall fluence through the implementation of its reactor vessel surveillance program and will take such steps as are necessary to ensure the continued safe operation of the Palisades plant. Because the RT_{PTS} values for the Palisades reactor vessel do not exceed the screening criteria of 10CFR 50.61, the report required by section b.3 for reactors exceeding the screening criteria will not be made. Consumers Power Company will, however, inform the NRC of any significant changes or developments that may occur in the future.

DISCUSSION

Combustion Engineering report P-NLM-019, "Summary Report on Manufacture of Test Specimens and Assembly of Capsules for Irradiation Surveillance of Palisades Reactor Vessel Materials", April 1, 1971 (reference 1) is the basis document for the Palisades Reactor vessel surveillance program. This document provides fabrication data, chemical properties, and some mechanical properties for the vessel materials as well as the surveillance program materials. "Palisades Pressure Vessel Irradiation Capsule Program: Unirradiated Mechanical Properties", April 25, 1977 (reference 2) provides a description of the mechanical properties of the surveillance program materials. The unirradiated reference temperature (RT_{NDT}) for longitudinally oriented Palisades reactor vessel base metal was established per section NB-2331 of the ASME Code as -10^{o}F. A lack of drop weight specimens for other samples (base traverse, HAZ, and weld metal) precluded a firm determination of the initial

RT_{NDT} values for all of the reactor vessel materials. However, by comparing full Charpy impact curves for the base metal longitudinal and traverse specimens, the RT_{NDT} of the base metal traverse sections was estimated to be $0^{o}F$. Initial RT_{NDT} values for the weld metal could not be accurately estimated, although Charpy impact data indicated that the RT_{NDT} values should be much lower than the values obtained for the base metal.

Battelle Columbus Laboratories Report, BCL-585-12, "Final Report on Palisades Nuclear Plant Reactor Pressure Vessel Surveillance Program Capsule A-240", March 13, 1979, (reference 3) documented the dosimetry and mechanical properties of the first surveillance capsule removed from the Palisades reactor vessel. This capsule was an accelerated capsule. The fluence measured for the Charpy test specimens taken from this capsule was 4.5 X $10^{19} n/cm^2 \geq 1 Mev$. The dosimetry calculations at that time indicated a maximum wall fluence of 3.5 X 10^{19} $n/cm^2 \geq 1Mev$ at the end of 32 effective full power years (EFPY). The shift in the 30 ft-lb Charpy energy curves for both the longitudinal and traverse base metal specimens was $205^{o}F$. The weld metal specimens exhibited a shift of $350^{o}F$. No chemical analysis was conducted on either the baseline specimens or the irradiated specimens taken from this capsule.

"Analysis of Capsules T-330 and W-290 From The Consumers Power Company Palisades Reactor Vessel Radiation Surveillance Program", WCAP-10637 (reference 4) was submitted to the NRC as part of Consumers Power Company letter dated October 31, 1984. This report provided chemical analysis, mechanical testing data and dosimetry calculations for the first reactor vessel wall capsule removed as part of the Palisades reactor vessel surveillance program. This capsule was removed after five fuel cycles and at 4.975 EFPY. The specimen fluence determined from flux wire measurements was 1.09 X 10^{19} $n/cm^2 \geq 1Mev$. The calculated fluence for these test specimens at the capsule location was 1.25 X 10^{19} $n/cm^2 \geq 1Mev$. The shifts in the 30 ft-lb Charpy energy curves were $155^{o}F$ for base metal traverse specimens, $175^{o}F$ for base metal longitudinal specimens and $290^{o}F$ for weld metal specimens. Based on the dosimetry calculations for capsule fluence, the maximum end-of-life fluence for the reactor vessel wall was then determined to be 6.3 X 10^{19} n/cm^2 ≥ 1Mev. This value assumed 32 EFPY and averaged power cycles similar to the previous cycles. Thus the fluence calculated from the analysis of capsule W-290 was approximately 80 percent greater than the maximum wall fluence calculated from the earlier accelerated capsule (A-240).

Examination of the weld chemistry data from this report (WCAP-10637) suggested that the weld material present in the surveillance capsules was not representative of the weld material used in the fabrication of the reactor vessel. A thorough review of the reactor vessel and surveillance capsule fabrication records documented in Attachment III to Consumers Power Company letter dated June 14, 1985 (reference 5) indicated that although the base metal in the surveillance capsules was of the reactor vessel, the weld specimens did not represent the vessel weld material. This letter described the true chemical composition of the weld material in the Palisades reactor vessel and showed that the screening criteria of 10CFR 50.61 as proposed at that time could be met for a maximum wall fluence of 5.9 X 10^{19}n/cm^2 ≥ 1Mev. This fluence was projected using an average of the measured and calculated values taken from WCAP-10637.

Consumers Power Company letter dated June 14, 1985 clearly established our intent to employ generic chemistry for both circumferential and longitudinal welds metal. For the longitudinal weld, the important chemistry values for incorporation into equation 1 of 10CFR 50.61 are .19 wt. percent copper and 1.10 wt. percent nickel. For the girth weld, the values are .21 wt. percent copper and .98 wt. percent nickel. The -56^{o}F initial average (I) RT_{NDT} for LINDE 1092 flux welds is appropriate for use with those average chemistries. In addition, the 59^{o}F margin value M in equation 1 is also appropriate in determining the adjusted reference temperature for the weld material.

Data obtained from references 1 and 4 show that the reactor vessel base metal contains between .24 and .25 wt. percent copper and between .52 and .55 wt. percent nickel. Substituting the upper values for copper and nickel into equation 1 of 10CFR 50.61, the calculated shifts at 1.25 X $10^{19} n/cm^2 \geq 1 Mev$ and at 4.5 X $10^{19} n/cm^2 \geq 1 Mev$ are $165^{o}F$ and $234^{o}F$ respectively for the base metal. The measured shifts for the base metal at these fluences average $165^{o}F$ at 1.25 X $10^{19} n/cm^2 \geq 1 Mev$ and are $205^{o}F$ at 4.5 X 10^{19} $n/cm^2 \geq 1 Mev$. Employing the measured upper limits for copper and nickel, a 4.5 X 10^{19} $n/cm^2 \geq 1 Mev$ fluence, an initial RT_{PTS} of $0^{o}F$ and the $48^{o}F$ margin term in equation 1 of 10CFR 50.61 results in a calculated RT_{PTS} of $282^{o}F$. However, using an initial value of $0^{o}F$ plus the measured shift provided by reference 3 ($205^{o}F$) results in a measured, adjusted RT_{NDT} of $205^{o}F$. Consumers Power Company does not believe that the $48^{o}F$ margin term of equation 1 was intended to create such a large difference between the calculated values of RT_{PTS} and the measured value of RT_{NDT}. In view of the fact that the base metal of the reactor vessel is properly represented by the surveillance program and that reliable chemical and mechanical property data exist, Consumers Power Company has concluded that, in determining the RT_{PTS} for the reactor vessel base material, a margin factor of $10^{o}F$ should be appropriate. In addition, the Safety Evaluation Report (reference 7) written for the Technical Specification Change Request regarding pressure-temperature limits for the Palisades Plant reactor vessel (reference 5) suggested that the calculated fluence rather than the lower, measured fluence provided by reference 4 would be conservative in estimating the EOL fluence to the Palisades reactor vessel. Therefore, using a margin term of $10^{o}F$ in equation 1 would accurately reflect the actual RT_{PTS} of the Palisades reactor vessel base metal in light of the measured shift data and the use of upper limit values for copper and nickel and calculated fluence in the equation for RT_{PTS}.

The data provided by reference 4 indicate that the change in the authorized reactor power level after the second fuel cycle has increased the rate of fluence accumulation over subsequent cycles. Based on the calculated fluence values from reference 4 it was estimated that the maximum wall fluence which presently exists is 1.25 X 10^{19} $n/cm^2 \geq 1 Mev$. Assuming an 80 percent capacity level over the duration of the remaining operating life of the Plant and continued operation at the current power level, it is estimated that the maximum EOL fluence to the Palisades reactor vessel inner wall will be 6.8 X 10^{19} n/cm^2 $\geq$1Mev. (This value for EOL fluence is being used to evaluate low leakage core designs.)

By employing equation 1 of 10CFR 50.61, the projected EOL fluence, and generic chemistry data, the calculated RT_{PTS} for the longitudinal weld metal is 259^{o}F and 273^{o}F for the circumferential weld. By employing equation 1 of 10CFR 50.61, the projected EOL fluence, a 10^{o}F margin term, and the upper limit chemistry data for copper and nickel, the RT_{PTS} for the base metal is 271^{o}F. Therefore, the RT_{PTS} values for the Palisades reactor vessel weld metal are below the appropriate screening criteria, and the RT_{PTS} for the vessel base metal is essentially equal to the screening criterion of 270^{o}F.

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References:

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(2) Perrin, J S and E O Fromm, "Palisades Pressure Vessel Irradiation Capsule Program: Unirradiated Mechanical Properties", August 25, 1977.

(3) Perrin J S, D R Framelo, RG Jung and E O Fromm, "Palisades Nuclear Plant Reactor Pressure Vessel Surveillance Program: Capsule A-240", March 13, 1979.

(4) Kunka, M K and C A Cheney, "Analysis of Capsules T-330 and W-290 from the Consumers Power Company Palisades Reactor Vessel Radiation Surveillance Program WCAP-10637", September 1984.

(5) Letter from B D Johnson (CPCo) to D M Crutchfield (NRC) - Attachment III, "Summary of Findings Relative to Palisades Plant Reactor Vessel Material", June 14, 1985.

(6) 10CFR 50.61 "Fracture Toughness Requirements for Protection Against Thermal Shock Events", 50FR 29944, July 23, 1985

(7) Letter from J A Zolinski (NRC) to D J VandeWalle (CPCo) - "Revised Pressure Temperature Limits", August 21, 1985.