3.6 ORIGINAL FUEL DESIGN EVALUATION

3.6.1 FUEL DESIGN AND ANALYSIS

The fuel rod cladding is designed to satisfy the design bases given in Section 3.2.3.5. The effects of irradiation on UO_2 and cladding materials are considered in the design calculations. The predicted effects of anticipated transients are also considered in the design process.

As stated in Section 3.2.3.5, the fuel rod cladding is designed to the following bases:

Basis 1

Maximum primary stress during steady state operation, expected transients, and depressurization is limited to two-thirds of the minimum yield strength of the material at operating temperature.

Basis 2

Predicted permanent hoop strain of the cladding at the end of fuel life is less than 1.0%.

These bases are conservative and the calculations used to demonstrate their satisfaction were conducted for limiting cases using limiting assumptions. This is considered advisable in the prediction of long-term fuel behavior under irradiation.

Maximum tensile stress in the fuel cladding occurs during a depressurization transient near EOL when internal gas pressure is highest. Clad thickness is such that under the anticipated transient conditions, this stress does not exceed two-thirds of the unirradiated value of yield stress of the clad material at its operating temperature. An unirradiated value is used for conservatism.

The satisfaction of Basis 2, the long-term total strain limit, was demonstrated as follows:

a. Clad stress-strain behavior was based on a stress analysis which includes the effect of creep. The loads considered were those due to fuel thermal and fission growth, fission gas pressure and external coolant pressure.

b. The fuel thermal and fission growth was calculated considering the fuel as a solid rod with unrestrained thermal expansion and a volumetric growth rate of 0.16% for 10^{20} fissions/cm^3 (Reference 1), and a LHR of 17.5 kW/ft. The fission gas pressure was calculated for a 31% fission gas release (which was based on the derivation of Lewis (Reference 2) considering the change in plenum volume due to the thermal expansion and growth of the rod).

c. The analysis was based upon an incremental approach, which divided the three-year fuel life span into discrete time intervals and evaluated the clad stress and strain, including the effect of creep, during these intervals. The relation between the incremental creep and the actual stress state is expressed by the Prandtl-Reuss formulae. The basis for creep is given by the von Mises criterion and the relation between creep rate and generalized stress is that given by Holmes (Reference 3). The rapidly convergent iterative technique was employed to solve the resulting non-linear equations.

d. For the nominal fuel-to-clad gap, at about 775 hours after BOL, the fuel has expanded to completely fill the fuel-to-clad gap and to restore the clad to a circular shape after its initial collapse onto the fuel. The fuel was subsequently assumed to swell unrestrained with the clad following. Based upon this conservative assumption, the final strain after three years service was 0.5%. That is, for

average fuel-to-clad gap at peak power density, Basis 2 was satisfied without credit for fuel strain under load.

e. For the most adverse initial condition, i.e., minimum clad ID, maximum pellet OD coincident with the point of maximum power density which was assumed to be sustained over lifetime, application of the unrestrained fuel growth model resulted in a computed strain at the end of the third cycle (EOC3) of 0.8%. However, as is well known (References 4, 5, and 6), the effect of restraint from the exterior, cooler regions of the fuel pellet, the clad, and the external pressure result in a significant limitation on radial swelling with corresponding flow of pellet material into the dish provided.

These analyses were conducted throughout with design BOL power density, although it was known that for fuel in its third burnup cycle, LPD would be substantially below these values. Thus, the LPD increase which might be associated with overpower transients near end of fuel life was conservatively considered. Further consideration of EOL power density is provided in subsequent paragraphs together with a summary of data justifying the maximum linear heat ratings and peak burnups. Table 3.6-1 contains typical maximum linear heat ratings as a function of burnup. The maximum linear heat rating for the first core was 17.5 kW/ft at BOL. The maximum heat rating near EOC3 was 14.9 kW/ft, resulting in a BOL/EOC3 ratio of 1.18. This was greater than the value of 1.12 for the ratio of maximum transient to steady state heat ratings. Thus, use of BOL power densities in these calculations for EOC3 transients provided considerable margin.

Studies by Notely, et al (References 5 and 6) in which 27 fuel elements were irradiated without failure, reported measured clad strains up to 3.33%.

In a series of experimental element irradiations, Westinghouse (Reference 4) reported strain values at failure for Zr-4 clad fuel elements of 0.78 to 2.6% depending on the fuel properties assumed. Also, Lustman (Reference 7) noted that failures in pile have occurred at strain values between 0.5 to 1.0%. However, these results are based on relatively low Zr-4 cladding temperatures as compared to contemporary, large, commercial PWRs. It is known (Reference 8) that permissible strain values for Zircaloy increase above 650°F. In the zone of interest, the average Zr-4 cladding temperature is about 720°F; this should result in increased ductility and thus a higher strain limit to failure.

For the AREVA/Framatome design, compliance was demonstrated using the NRC-approved methodology using the RODEX2 code.

3.6.2 ANALYSIS OF BURNUP AND LINEAR HEAT RATINGS

Prior to a discussion of the experimental bases for justifying the initial maximum linear heat ratings and burnups, it is necessary to relate these parameters so that they may be viewed in the proper perspective. The maximum linear heat rating was reached but not exceeded only during approximately the first 28,000 MWD/MTU of peak burnup. The maximum linear heat rating decreased with additional burnup beyond this value.

Typical values at the time of initial design are shown in Table 3.6-1, which contains an analysis of burnup, total nuclear peaking factors, and the corresponding maximum linear heat rating (including consideration of the combination of total nuclear and mechanical peaking factors), for the most adverse equilibrium core.

Table 3.6-2 contains a comparison of maximum heat ratings for a number of plants of that period. Peak linear heat ratings for this plant were consistent with current practice and were considered as slightly conservative with respect to a number of the designs.

Although it was believed that fuel rods could operate satisfactorily with a small amount of fuel melting, the initial design did not permit fuel melting even under conditions imposed by anticipated transients. Cycle 1 design offered considerable margin with respect to the core linear heat rating of 24 kW/ft for melting (BOL value; typical EOC3 value was about 23 kW/ft), even when expected transients (112%) were considered.

3.6.3 SUMMARY OF PERTINENT FUELS IRRADIATION INFORMATION

The LHRs specified in this section are as they appeared in the referenced literature and represent total core heat rates.

3.6.3.1 High Linear Heat Rating Irradiations

The determination of the effect of linear heat rating and fuel-cladding gap on the performance of Zircaloy-clad UO_2 fuel rods was the object of two experimental capsule irradiation programs conducted in the Westinghouse Test Reactor (WTR) (Reference 9). In the first program, 18 rods containing 94% TD UO_2 pellets were irradiated at 11, 16, 18 and 24 kW/ft with cold diametral gaps of 0.006", 0.012" and 0.025". The wall thickness to diameter ratio (t/OD) of the Zircaloy-cladding was 0.064 which is slightly higher than the value of 0.059 of Cycle 1. Although these irradiations were short duration (about 40 hours), significant results applicable to Cycle 1 design were obtained. No significant dimensional changes were found in any of the fuel rods. Only one rod, which operated at 24 kW/ft with an initial diametral gap of 0.025", experienced center melting. Rods which operated at 24 kW/ft with cold gaps of 0.006" and 0.012" did not exhibit center melting on these bases. The initial gap of 0.0085" and the maximum linear heat ratings for this design (Table 3.6-1) provided adequate margin against center melting even when 12% overpower conditions were considered. These results also indicated that an initial diametral gap of 0.0085" was adequate to accommodate radial thermal expansion without inducing cladding dimensional changes, even at 24 kW/ft. This margin with respect to thermal expansion, decreased with increasing burnup at a rate of 0.16% ΔV per 10^{20} fissions/cm^3. However, the linear heat rating also diminished with burnup (Table 3.6-1). Since the diametral thermal expansion (assuming BOL maximum heat ratings) is almost twice as great as the swelling diametral growth (on the EOC3 burnup), these data added considerable weight to the conservative treatment of the influence of transients on fuel element integrity.

Further substantiation of the capability of operation at maximum linear heat ratings in excess of those in the first cycle design was obtained from later irradiation tests in WTR (Reference 9). Thirty-eight-inch long and 6" long fuel rods were irradiated at linear heat ratings of 19 kW/ft and 22.2 kW/ft to burnups of 3450 and 6250 MWD/MTU. The cold diametral gaps in these Zircaloy-clad rods containing 94% dense UO_2 were 0.002", 0.006" and 0.012". The cladding t/OD was 0.064. No measurable diameter changes were noted for the 0.006" or 0.012" diametral gap. Only small changes were observed for the rods with a 0.002" diametral gap.

Additional successful radiations had been performed with SS cladding in Saxton at 23 kW/ft and in Plum Point at 22 to 25 kW/ft.

3.6.3.2 Shippingport Blanket Irradiations

Zircaloy-clad fuel rods operated successfully (three defects had been observed which were a result of fabrication defects) in the Shippingport blanket with burnups of about 37,000 MWD/MTU and maximum linear power ratings of about 13 kW/ft (References 9, 10, and 11). Although higher linear heat ratings at lower burnups would be experienced, swelling (primarily burnup-dependent) and thermal

expansion (linear heat rating dependent) provide the primary forces for fuel cladding strain at the damage limit. Thus, Shippingport irradiations demonstrated that Zircaloy-clad rods with a cladding t/OD comparable to that for this plant (0.059) should successfully contain the swelling associated with 37,000 MWD/MTU burnup, while at the same time containing the radial thermal expansion associated with heat ratings of the time. Irradiation test programs in support of Shippingport in in-reactor loads demonstrated successful operation of burnups of 40,000 MWD/MTU and linear heat ratings of about 11 kW/ft with cladding t/OD ratios as low as 0.053 (compared with 0.059 for this plant) (Reference 12).

3.6.3.3 NRX Irradiations (AECL - Canada)

Eleven Zircaloy-clad, large diameter fuel elements (approximately .750" OD) with clad thicknesses of .016", .024", and 0.037" (t/OD = .021, .031, and .047 corresponding to TD percentages of 94.3, 94.3 and 93.7, respectively) were irradiated in the NRX pressurized loop facility of AECL, Canada (Reference 13) at loop pressures of 2000 to 3000 psi. The cold diametral gaps for the test elements were .0035" and .0040", and the fuel was UO_2 sintered pellets (0.700" diameter) loaded in an argon atmosphere.

The elements were operated for 535 full power days to an average burnup of 10,280 MWD/MTU at a maximum linear power output of 14.8 kW/ft. These elements experienced 308 power cycles. No failures were reported for these elements, and the final dimensions of the rods were reported to be virtually unchanged from pre-irradiation values.

The successful operation of these elements with considerable lower clad-to-diameter ratios than those for Cycle 1 demonstrated the capability of safe operation of Zircaloy-clad elements with thin cladding for many power cycles.

Additional tests on similar elements were then in progress at NRX involving test elements with UO_2 and (U, Pu) O_2 (PuO_2 = 2.4 wt%) at average linear heat ratings of 11.4 and 17.2 kW/ft. Those elements had accumulated burnups of 6,400 and 28,700 MWD/MTU without failure.

3.6.3.4 Saxton Irradiations

UO_2-PuO_2 fuel rods containing pellets of 94% TD and clad with Zircaloy-4 had been successfully irradiated in Saxton to burnups approaching 25,000 MWD/MTU at 16 kW/ft under USAEC Contract AT(30-1)-3385 (Reference 14). The t/OD of the cladding was 0.059 which is equivalent to the Cycle 1 design. The amount of PuO_2, 6.6%, was considered as insignificant with respect to providing any differences in performance when compared with that for UO_2. In fact, the higher thermal expansion coefficient for this PuO_2-UO_2 composition than that for UO_2 would induce greater cladding strain under equivalent irradiation conditions. Subsequent tests on two of the above rods (18,600 MWD/MTU at 10.5 kW/ft) successfully demonstrated the capability of these rods to undergo power transients from 16.8 kW/ft to 18.7 kW/ft.

3.6.3.5 Vallecitos Boiling Water Reactor - Dresden

The combined Vallecitos Boiling Water Reactor (VBWR) - Dresden irradiation of Zircaloy-clad oxide pellets (Reference 15 and 16) provided additional confidence with respect to the design conditions for the fuel rods for Cycle 1 core. Ninety-eight rods irradiated in VBWR to an average burnup of about 10,700 MWD/MTU were assembled in fuel assemblies and irradiated in Dresden to a peak burnup

greater than 48,000 MWD/MTU. The reported maximum heat ratings for these rods was 17.3 kW/ft, which occurred in VBWR. The t/OD cladding ratio of 0.052, pellet TD of 95%, and the external pressure of about 100 psi are conditions which are all in the direction of less conservatism with respect to fuel rod integrity when compared with the design values of 0.059 cladding t/OD ratio and an external pressure of 2250 psi. Ten of these VBWR - Dresden rods representing maximum combinations of burnup, heat rating and pellet density had been selected for detailed destructive examinations as part of an AEC program. The remaining 88 rods were returned to Dresden and successfully irradiated to the termination of the program.

3.6.3.6 Large Seed Blanket Reactor Rods

Two rods operated in the B-4 loop at the Materials Testing Reactor provided a very interesting simulation for contemporary PWR designs (Reference 4, 17, and 18). Both rods were comprised of 95% TD pellets with dished ends clad in Zircaloy. The first of these, No. 79-2, was operated successfully to a burnup of 12.41×10^{20} f/cc (approximately 48,000 MWD/MTU) through several power cycles which included linear power from 5.6 to 13.6 kW/ft. The second fuel pin, No. 79-25, operated successfully to 15.26×10^{20} f/cc (approximately 60,000 MWD/MTU). The basic difference in this rod was the 0.028" wall thickness, as compared to 0.016" (t/OD 0.058) in the first rod. All other parameters were essentially identical. The linear heat rating ranged from 7.1 to 16.0 kW/ft. After the seventh interim examination, the rod operated at a peak linear power of 12.9 kW/ft at a time when the peak burnup was 49,500 MWD/MTU. These high burnups were achieved with fuel elements which were assembled by shrinking the cladding onto the fuel. This indicated that a comparable irradiation of the fuel elements for this reactor would allow a considerable increase in swelling life at a given clad strain.

One additional rod irradiated in Materials Testing Reactor as part of the Large Seed Blanket Reactor (LSBR) series (rod 79-18) demonstrated the effect of clad restraint on the swelling behavior of a UO_2-Zircaloy-clad rod (Reference 19). A starting fuel density of 81.4% of theoretical was used in conjunction with a zero cold gap and a 0.060 cladding t/OD ratio. The rod was irradiated to 49,000 MWD/MTU with no measurable change in rod diameter.

3.6.3.7 Central Melting in Big Rock

As part of a Joint U.S. - Euratom Research and Development Program, Zircaloy-clad UO_2 pellet rods with 95% of TD had been irradiated under conditions designed to induce central melting in the Consumers Big Rock Point Reactor (Reference 20). The test included 0.7" diameter fuel rods (cladding t/OD = 0.061, fuel-to-clad gap of about 0.011") at maximum linear heat ratings of about 27 kW/ft and 22 kW/ft with peak burnups up to 20,000 MWD/MTU. Results of these irradiations provided a basis for incorporating linear heat ratings well in excess of those calculated for this reactor (Reference 21). These results showed that the presence of localized regions of fuel melting were not catastrophic to the fuel assembly.

3.6.3.8 Peach Bottom 2

General Electric (GE) had successfully irradiated fuel pins of the Peach Bottom 2 design to burnups in excess of 42,000 MWD/MTU at peak linear heat ratings of 23 kW/ft. An interim examination at 32,500 MWD/MTU indicated a satisfactory condition (Reference 22).

3.6.4 EVALUATION

It was concluded from the above information that heat ratings as high as 23 to 24 kW/ft could be achieved in the fuel elements without fuel centerline melting. Linear heat ratings in the Cycle 1 core design fell significantly below this limit even at the 112% overpower condition.

Heating ratings and burnups for this design were well demonstrated by the existing technology. Nevertheless, it was felt fruitful to consider the question of what constitutes a fuel element failure. For one, the cladding must be violated. On the subject of the influence of expected transients, a conservative analysis had been presented of the factors which influence cladding performance during such transients. The fuel rod cladding was designed on a conservative basis and the calculations considered limiting cases and limiting assumptions. Consideration of peaking factor reductions shown in Table 3.6-1 increased the conservatism of these analyses.

The analyses had been conducted throughout with design BOL power density, although it was known that for fuel in its third burnup cycle, LPD would be substantially below these values. Thus, the LPD increase which might be associated with overpower transients near end of fuel life had been conservatively considered. Cladding integrity had been demonstrated even under these adverse conditions. Consideration of peaking factor decreases noted in Table 3.6-1 made this analysis even more conservative.

Present heat rating limits are based on LOCA/Emergency Core Cooling System stored energy considerations and are included in Section 14.17.

3.6.5 REFERENCES

1. M.L. Bleiberg, R.M. Berman, and B. Lustman, "Effects of High Burnup on Oxide Ceramic Fuel," WAPD-T-1455, March 1962
2. B. Lewis, "Engineering for the Fission Gas in UO_2 Fuel," Nuclear Applications, Vol. 2, No. 2, April 1966
3. J.J. Holmes, J.A. Williams, D.H. Nyman, and J.C. Tobin, "In-Reactor Creep of Cold Worked Zircaloy Z," Flow and Fracture of Metals and Alloys in Nuclear Environments, ASTM-STP-380, 1965
4. E. Duncombe, J.E. Meyer, and W.A. Coffman, "Comparison with Experiment of Calculated Dimensional Changes and Failure Analysis of Irradiated Bulk Oxide Fuel Test Rods Using the CYGRO-I Computer Program," WAPD-TM-583, September 1966
5. Notely, Bain, and Robertson, "The Longitudinal and Diametral Expansion of UO_2 Fuel Elements," AECL-2143, November 1964
6. M.J.F. Notely and J.R. MacEwan, "The Effect of UO_2 Density on Fission Product Gas Release and Sheath Expansion," AECL-2230, March 1965
7. B. Lustman, "Fuel Clad Design Basis for Thermal Reactors," Bettis Atomic Power Laboratory, May 1966
8. P.J. Pankaskie, "Creep Properties of Zircaloy-2 for Design Application," HW-75267, October 1962
9. Indian Point Nuclear Generating Unit No. 2, Preliminary Safety Analysis Report, Appendix A
10. J.T. Stiefel, H. Feinroth, and G.M. Oldham, "Shippingport Atomic Power Station Operating Experience, Developments and Future Plans," WAPD-TM-390, April 1963

11. Question V.B.2, Prairie Island Nuclear Generating Plant, Preliminary Safety Analysis Report, (Docket No. 50-306)

12. T.D. Anderson, "Effects of High Burnup on Bulk UO_2 Fuel Elements," Nuclear Safety, Vol. 6, No. 2 Winter 1964-1965, p. 164-169

13. R.D. MacDonald, et al, "Zircaloy-2 Clad Fuel Elements Irradiated to a Burnup of 10,000 MWd/MTU," AECL-1952, 1964

14. R.S. Miller, et al, "Operating Experience with the Saxton Reactor Partial Plutonium Core-II" paper presented at AEC Plutonium Meeting in Phoenix, August 1967

15. C.J. Baroch, J.P. Hoffmann, H.E. Williamson, and T.J. Pashos, "Comparative Performance of Zircaloy and Stainless Steel Clad Fuel Rods Operated to 10,000 Mwd/T in the VBWR," GEAP-4849, April 1966

16. F.H. Megerth, "Zircaloy-Clad UO_2 Fuel Rod Evaluation Program, Quarterly Progress Report No. 2, February 1968 - April 1968," GEAP-5624 (May 1968)

17. R.M. Berman, H.B. Meieran, and P. Patterson, "Irradiation Behavior of Zircaloy-Clad Fuel Rods Containing Dished-End UO_2 Pellets," (LWBR-LSBR Development Program), WAPD-TM-629, July 1967

18. J.T. Engel, et al, "Performance of Fuel Rods Having 97 Percent Theoretical Density UO_2 Pellets Sheathed in Zircaloy-4 and Irradiated at Low Thermal Ratings," (LSBR/LWBR Development Program), WAPD-TM-631, July 1968

19. J.E. McCauley, et al, "Evaluation of the Irradiation Behavior of a Zircaloy-4 Clad Fuel Rod Containing for Density UO_2 Fuel Pellets," LWBR-LSBR Development Program, WAPD-TM-596, January 1968

20. J.P. Blakely, "Action on Reactor and Other Projects Undergoing Regulatory Review of Consideration" Nuclear Safety, Vol. 9, No. 4, p. 326 (July-August 1968)

21. S.Y. Ogawa, Final Report, "Power Reactor High Performance UO_2 Program," Joint US-Euratom Research and Development Report, GEAP-10042, June 1969

22. Summary description of Peach Bottom Atomic Power Station Units No. 2 and No. 3 and Review of Considerations Important to Safety, Docket No. 50-277 and 50-278

TABLE 3.6-1

TYPICAL PEAK BURNUP - MAXIMUM HEAT RELATIONSHIP

MAXIMUM LOCAL EXPOSURE MWD/MTU	TOTAL NUCLEAR PEAKING Factor	MAXIMUM HEAT RATING kW/ft
24,200	2.86	17.5
24,200 - 36,000	2.86	17.5
36,000 - 48,500	2.42	14.9

TABLE 3.6-2

COMPARISON OF MAXIMUM HEAT RATINGS

REACTOR	kW/ft
Maine Yankee	16.7
Fort Calhoun	17.1
Calvert Cliffs, Unit 1	17.5
Calvert Cliffs, Unit 2	17.5
Hutchinson Island, Unit 1	17.8
Millstone Unit 2	17.8
Turkey Point	17.3
Surrey	17.5
Prairie Island	17.4
Three Mile Island	17.5
Oconee	17.5
Indian Point, Unit 2	18.5
Diablo Canyon	18.9
Browns Ferry	18.5
Sequoyah	18.8
San Onofre, Units 2 and 3	18.5