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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

April 16, 1980

Additional question: what insurance inspections are being evaluated & planned for late in life conditions. (post 24000 EFPH)

MEMORANDUM FOR: Robert A. Benedict, Senior Project Manager
Light Water Reactors, Branch No. 2
Division of Project Management

FROM: Ralph O. Meyer, Leader
Reactor Fuels Section
Core Performance Branch
Division of Systems Safety

SUBJECT: SHIPPINGPORT LWBR EXTENDED BURNUP

I have discussed Admiral Rickover's letter on continued operation of the Light Water Breeder Reactor with members of the Fuels Section. We have identified six areas that ought to be considered in connection with extending the fuel burnup:

1. Fission Gas Release
2. Cladding Collapse
3. Pellet/Cladding Interaction
4. Fuel Rod Bowing
5. Crud Buildup
6. Clearance for axial growth

It is probable that Bettis has already performed the evaluation we would recommend, so our comments will be very brief.

1. Fission gas will accumulate at the higher burnups and contribute to the internal rod pressure. Since the system pressure is simultaneously being reduced, a check should be made to ensure that (hot) fuel rod pressure does not get larger than the system pressure.

2. The earlier cladding collapse analysis gave marginal results for higher burnups and should be reevaluated. The reduction of system pressure will help here.

3. The increased accumulation of corrosive fission products and the relative thin cladding will cause this fuel to be susceptible to PCI failure. Power

changes should therefore be gentle. We would recommend on-line radiation monitoring for increased activity (fuel failures) so this information could be correlated with the power history to aid in post-operation diagnostics.

4. Fuel rod bowing increases with burnup and affects heat transfer correlations. This effect should be taken into account.

5. Crud buildup on the fuel could also affect heat transfer. Core pressure drops and previous Shippingport experience should be able to indicate any significance of this effect.

6. Irradiation growth of the fuel assembly must be accommodated.

Ralph O. Meyer

Ralph O. Meyer, Leader
Reactor Fuels Section
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NRC Question #1

Fission gas will accumulate at the higher burnups and contribute to the internal rod pressure. Since the system pressure is simultaneously being reduced, a check should be made to ensure that (hot) fuel rod pressure does not get larger than the system pressure.

Bettis Response

Criterion: Maintain fuel rod internal pressure below external system pressure.

Bases

A plenum is provided at the top of the fuel stack in each LWBR fuel rod for the purpose of accepting gases released from the fuel pellets without unacceptable increase in rod internal gas pressure or reduction in fuel-cladding gap conductance. In addition, a large plenum volume provides space for a large volume of helium fill gas which dilutes the fission and volatile gases released from the fuel during irradiation, thereby minimizing the reduction of gas thermal conductivity.

Assessment

The maximum design rod internal gas pressure at the worst case calculated plenum temperature of 650°F has been calculated for the highest depleted fuel rod in each region. Results are shown in Table 1. The calculational model accounts for initial fill gas (helium), volatile gas release, helium produced by ternary fission and fission gas release, all in a conservative manner. The calculated internal pressures are much lower than the system pressure at any time in life. This is the result of the rod plenum space having been designed to avoid high internal pressures during normal operation. The fuel elements are not pre-pressurized, and fission gas release is low because of the low fuel operating temperature and because of the low power level.

TABLE 1

Calculated Internal Rod Gas Pressure Compared to System Pressure - PSIA

Region	Calculated Gas Pressure 18,000 EFPH	Calculated Gas Pressure 24,000 EFPH	System Pressure 18,000 EFPH	System Pressure 24,000 EFPH
Seed	429	565	1815	1615
PF Blanket	202	230	1815	1615
Blanket	173	195	1815	1615
Reflector	189	200	1815	1615

NRC Question #2

The earlier cladding collapse analysis gave marginal results for higher burnups and should be reevaluated. The reduction of system pressure will help here.

Bettis Response

Criterion: Cladding integrity is maintained.

Bases

Large deformation potentially can occur at unsupported lengths of blanket and reflector rod cladding (non-free-standing cladding) which may be created by relative pellet axial movement within the fuel rods. Gaps were observed to occur primarily in the out-of-flux plenum regions of LWBR irradiation test fuel rods. With much lesser probability, they can occur within the fuel rod stack. In the SAR assessment, LWBR fuel rod cladding collapse performance was evaluated using the BUSHL analysis procedure (Reference 1). This procedure was used to calculate critical lengths for short tube collapse as a function of fast fluence and temperatures. These results were used to develop comparisons between maximum calculated gap between pellets along the rod length and critical gap length for collapse for the fuel rod with highest fluences in each of the three blanket regions. Results indicated that no collapses were expected.

Cladding deformation over axial gaps is presently analyzed using a much improved procedure based on the ACCEPT computer program (Reference 2) which was not available for the SAR assessments. ACCEPT differs from BUSHL in that it calculates progressive deformation of the cladding over axial gaps and accounts for

(a) Progressive axial gap growth due to fuel densification and fuel cladding axial extension.

(b) Pellet-to-cladding radial clearances.

(c) Foundation deformation (pellet shrinkage and ovalling).

(d) Incremental, time dependent cladding deformation, vice plastic stability calculations performed in the BUSHL program using perturbation analysis and isochronous stress-strain curves to account for creep.

Time to collapse is calculated using ACCEPT with input parameters fitted conservatively to predict both out-of-pile and in-pile test data. The procedure includes the projected reductions in primary system pressure, temperature and reactor power.

NRC Question #2 (continued)

Assessment

Plenum Axial Gaps

Cladding deformation over the worst case plenum gap for the standard blanket fuel rod experiencing the highest fluence and assuming worst case conditions (geometry, material creep, etc.) has been calculated using the ACCEPT analysis procedure. With the planned power, pressure, temperature history, a collapse time in excess of 24,000 EFPH is predicted for the limiting fuel rod worst case conditions. Analysis of this behavior to 30,000 EFPH is now in progress.

In-Flux Axial Gaps

Formation of axial gaps within the fuel rod stack of blanket rods is expected to be a low probability event since formation of such axial gaps requires pellet hangup. Only 2.4% of the representative LWBR irradiation test rods were found to have formed axial gaps within their fuel stacks. In each of these, the gaps between any two pellets were only a fraction of the total available gap when summed over the total length of the rod.

Cladding stability over axial gaps in the fuel stack of blanket rods has been evaluated statistically. Deformation of cladding over within-stack axial gaps was analyzed on several bases. On a best estimate basis (i.e., assuming best estimate material and geometry parameters), peak ovality of the most limiting blanket rod was predicted to be only 9 mils after 24,000 EFPH operation. On a probabilistic basis considering statistical variation of relevant parameters, the collapse fraction after 24,000 EFPH was predicted to be 0.0003 (about 2 out of 6815 blanket rods). Analysis of this behavior to 30,000 EFPH is now in progress.

Effects of Pressure Reduction

The planned sequence of system pressure reductions, from 2000 psia at beginning-of-life to 1615 psia for operation beyond 18,000 EFPH has a strong effect on reducing cladding deformation due to the highly non-linear effect of stress upon creep rate (about 60% reduction due to fourth power stress dependancy). Cladding creep rates are decreased further by reduced temperatures and core power later in life.

References

1. A. L. Thurman, "BUSHL - A Computer Program for the Inelastic Buckling of Shells of Resolution Under External Pressure and Axial Compression," WAPD-TM-890, October 1970.

2. D. N. Hutula, B. E. Wianko, "ACCEPT: A Three-Dimensional Finite Element Program for Large Deformation Elastic-Plastic Creep Analysis of Pressurized Tubes," WAPD-TM-1383, March 1980

NRC Question #5

The increased accumulation of corrosive fission products and the relative thin cladding will cause this fuel to be susceptible to PCI failure. Power changes should therefore be gentle. We would recommend on-line radiation monitoring for increased activity (fuel failures) so this information could be correlated with the power history to aid in post-operation diagnostics.

Bettis Response

Criteria: Avoid plastic instability and halogen stress corrosion cracking of fuel rod cladding.

Bases

Up-power transient capability of LWBR fuel rods is specified in terms of allowable operating conditions which will avoid damage caused by pellet-cladding interaction (PCI). The PCI criterion is defined by cladding stress levels which have been shown in test experience to lead to fuel rod damage from either stress corrosion cracking or plastic instability. Stress corrosion cracking is the limiting concern for normal power increases within the range of allowable operating power levels. This mechanism becomes more significant with extended lifetime as the fission-produced iodine, believed to be the corrosive agent, accumulates in the fuel rods. Plastic instability is the limiting concern in the evaluation of overpower transients and becomes more significant with extended lifetime operations in LWBR since the peak overpower level increases, cladding thickness is reduced by wear, and rod-to-rod contact resulting in relatively high local cladding temperatures becomes more likely.

Assessment

Fuel rod damage by PCI is avoided in LWBR fuel rods by restrictions on operating power maneuvers and on the scram setpoint level. The assessment of fuel rod capability to withstand up-power transients without PCI damage has included the following effects which become significant with extended lifetime.

(a) Increased overpower levels. Whereas the peak overpower level which must be shown to be acceptable was 125% to 12,000 EFPH and 135% to 18,000 EFPH, the peak overpower level which must be shown to be acceptable is increased to 135% to 21,000 EFPH and to 135% to 24,000 EFPH in assessing PCI limitations.

Rod-to-rod contact. As rod bowing increases with extended lifetime, the clearance between rods decreases such that contact or close proximity may occur. The upper bound analysis of worst case fuel rods assumes contact after 18,000 EFPH. The resulting higher cladding temperatures and additional corrosion buildup at the contact locations are included in the up-power transient assessments. Rod-to-rod contact is discussed further in a subsequent section.

It has been shown using the same conservative analysis procedure used in the SAR assessment that the extended lifetime operating plan is acceptable for up-power transient concerns and will not lead to fuel rod damage by either PCI mechanism, stress corrosion cracking (SCC) or plastic instability. Although stress limits for SCC decline with increasing lifetime due to higher iodine release and potentially higher cladding temperatures, the planned power reductions restrict peak stresses below design limits with margins equal to or greater than the margins prior to 18,000 EFPH.

Irradiation testing of reference fuel rods has included up-power transients. In addition to the tests described in Section 4.2.1 of the SAR, additional up-power transient tests were performed subsequently at higher depletion levels. Two seed reference rods experienced overpower transients of approximately 170% at depletion levels close to 10 x 10^{20} fissions/cc. No fuel rod damage resulted even though predicted stresses exceeded design limits. Three blanket reference rods experienced overpowers. Two of these rods experienced overpower transients of 129% and 140%, respectively, without cladding damage. The third test rod experienced cladding stresses exceeding the LWBR stress corrosion cracking stress limits during a 157% overpower transient and suffered cladding damage as expected.

The overpower experience and depletion levels of the irradiation test rods involved in the up-power transient tests are summarized in Table 2. Although the peak depletion levels at 24,000 EFPH have not been bounded by the test rod depletion levels, the test levels are considered sufficient to provide meaningful checks on the analysis procedure. The results of these tests have been used in support of the stress corrosion cracking design limits and have enabled the specification of plant operating guidelines which will maintain fuel rod integrity under up-power transient conditions.

Coolant Activity Monitoring

Delayed neutron monitoring is done using on line equipment and recorded on strip charts. This equipment is described in Chapter 9, Section 9.3.5, of the SAR.

To detect fuel failures coolant samples are taken about twice a week during power operation. These samples are analyzed for various fission product isotopes with emphasis on gross iodine, I-131 and I-133. An increased number of samples are taken during swingload operation. In addition, about six samples are taken during each startup as the reactor attains various power levels.

TABLE 2

LWBR Irradiation Test Rods in Up-Power Transient Testing

Test Rod	Overpower Transient (%)	Fuel Depletion (10^{20} f/cc)	Test Result
Seed Reference Rods			
79-621	173	10.4	No defect
79-630	167	9.9	No defect
LWBR Seed Rod Peak Depletion at 24,000 EFPH		12.0	
Blanket Reference Rods			
79-575	139	5.1	No defect
79-467	140	2.7	No defect
	105	5.5	No defect
	134	0.9	No defect
79-587	157	2.3	Defected
LWBR Blanket Rod Peak Depletion at 24,000 EFPH		5.6	

NRC Question #4

Fuel rod bowing increases with burnup and affects heat transfer correlations. This effect should be taken into account.

Bettis Response

Criterion: Bowing minimized to assure that heat transfer is adequate and cladding temperature does not exceed limits.

Bases

Rod bowing sufficient to cause local close proximity or contact between rods can result in local hot spots on the cladding surfaces and can restrict coolant flow in the channel between the bowed rods. The design basis is to minimize rod bowing (and other effects which can result in reduced rod-rod clearance, e.g., cladding wear, diameter shrinkage) so that any potential contacts do not cause unacceptable thermal/hydraulic performance and do not result in cladding temperatures at the point of contact which exceed limits based on cladding strength and corrosion resistance.

Rod bowing behavior is sensitive to fuel element design parameters, to power level and operating environment, to grid system design, and to local fuel-cladding interactions. The fuel element design parameters contributing to bowing are cladding wall thickness eccentricity and overall helical variations (wave-length), initial rod straightness, fuel pellet configuration (fuel pellet dish, chamfer and endface non-perpendicularity), fuel-cladding gap size, and plenum spring force. Grid system design parameters contributing to rod bowing are grid spring force, alignment and nonperpendicularity of dimple contacts in the seed.

Assessment

As discussed in Section 4.2.1 of the SAR, fuel element and grid design parameters were specified to minimize rod bowing. In addition, seed and blanket fuel rods were selectively located in grid cells such that the rod-grid interaction configurations resulted in reduced bowing potential and/or increased initial rod-rod clearance. However, extended lifetime operation beyond 18,000 EFPH may result in reduced rod-rod clearances due to increased bowing and due to increased cladding wear and rod diameter shrinkage at grid contacts. On a worst case basis, rod-rod contact in both the seed and blanket is calculated to occur prior to 30,000 EFPH. Although the calculated number of occurrences is small, it has been conservatively assumed that rod-rod contact occurs at 18,000 EFPH at the most limiting seed and blanket locations. The power reductions (and the pressure reductions) specified for operation beyond 18,000 EFPH assure that thermal/hydraulic performance is acceptable and that cladding temperatures at contact points do not exceed limits.

NRC Question #4 (continued)

The cladding temperature at rod-rod contact points has been calculated using a procedure developed since the SAR was written based on comparison to irradiation tests of rods in contact for periods up to 20,000 hours of irradiation and to out-of-pile tests in which the cladding temperature was directly measured. These test results indicate that although in the worst case location accelerated oxide corrosion occurs at the reduced power and pressure during beyond 18,000 EFPH operation no more than 1 mil of corrosion would be expected. The elevated temperatures at rod contact points result in reduction of cladding strength, and this effect has been included in assessment of fuel rod performance during overpower transients and for stress corrosion cracking. In addition, out-of-pile flow tests which included rods intentionally bowed to contact indicate that fretting wear at rod-rod contact locations is acceptably small.

The condition of rods touching leads to a concern for reduced CHF capability. Rods touching heat transfer tests performed by Bettis and elsewhere (e.g., see Reference 1) have shown CHF degradations for rods touching varying from zero to greater than 30% depending on fluid conditions. These data show that at the LWBR CHF limiting condition (relatively low flux and high enthalpy) there is no power penalty. At the high flux low enthalpy region, which is the point where CHF power penalties have been measured, clad temperatures at the touching point can be above those under normal heat transfer conditions. The Bettis rods touching data has been used to define cladding temperatures and the onset of CHF in the event of rods touching. In addition, the allowances for those effects which reduce hot subchannel flow area along the entire length of the fuel region (i.e., rod bow, grid pitch and fuel rod ovality, shrinkage, and wear) have been increased for beyond 18,000 EFPH analyses. A power penalty of approximately 17% is being used to account for these effects. The increased hot channel factors and reductions in core power, plant average temperature and pressure for operation beyond 18,000 EFPH provide the necessary margin to ensure acceptable fuel element performance under rods touching conditions.

Reference

1. ASME Publication 77-HT-91, August 1977.

NRC Question #5

Crud buildup on the fuel could also affect heat transfer. Core pressure drops and previous Shippingport experience should be able to indicate any significance of this effect.

Bettis Response

Criteria: The effect of CRUD deposition must be included in thermal performance evaluations.

Basis

During the latter half of PWR Core 2 Seed 1 operation and during PWR Core 2 Seed 2 operation when pH was maintained at 10.2 ± 0.1 using ammonia chemistry, as is similarly specified for LWBR, the instrumented PWR-2 fuel assemblies showed a stable long-term flow condition within the ±2 percent accuracy of the flow instrumentation. Data from PWR operations forms part of the basis for the design CRUD model and for the specification of LWBR primary coolant chemistry control.

Assessment

The PWR Core 2 experience tends to support extrapolation of the current LWBR flow base for extended operation; that is, no radical flow change due to crud deposition is anticipated. Measurements taken to date on LWBR show that the seed and blanket flows have decreased by approximately 2% while core pressure drop has gone up approximately 5%. The flow decrease has been fairly steady with lifetime and is equal to a rate of flow decrease of 1% per year. Continued monitoring of LWBR seed and blanket flowrates and core pressure drop provides a confirmation that core thermal and hydraulic performance capability is not degraded by crud deposition beyond that provided by the crud allowances incorporated in the LWBR extended life performance analysis. The 5% preferential crud factor employed for setting operating limits thus far will continue to be used for extended operations.

NRC Question #6

Irradiation growth of the fuel assembly must be accommodated.

Bettis Response

A. Fuel Rods

Criteria: Rod elongation limited to prevent interference with baseplate.

Bases

A limit is placed on the allowable maximum rod elongation for the LWBR fuel rods to ensure that there is no rod-to-baseplate interference and that no unacceptable flow-induced rod vibration will occur.

Assessment

The elongations of the LWBR fuel rods were calculated on a worst case design basis using an analysis procedure incorporating the CYGRO-4 computer program which has been qualified to the measured elongations of test rods in the LWBR irradiation test program.

Listed in Table 3 are the maximum calculated elongations of the LWBR fuel rods at 18,000 and 24,000 EFPH. The table also lists the associated minimum rod-to-baseplate clearances and the free end rod overhang increases at the selected EFPH values.

Results indicate no rods interfere with the baseplate through 24,000 EFPH. The calculated overhang increases at the free end grids are satisfactory also since such overhangs have been included in flow tests with no indication of unusual flow-induced vibration or accelerated cladding wear due to length increases of these magnitudes.

B. Fuel Assemblies

Criteria: No axial interference between modules or module stub tubes and the bottom plate structures is permitted.

Bases

The criterion of no axial interference is invoked, because column loading of these modules could induce transverse deflection of the modules in a column buckling mode. In this way module to module clearances could be reduced, as well as fuel rod to fuel rod clearances.

Assessment

Table 4 indicates the large margin for clearance that exists in LWBR for axial length change of the various module types during 24,000 EFPH of operation. For instance, among the seed modules the worst case length increase is .537" as contrasted with a minimum beginning of life clearance of 2.090". The worst case length increase model conservatively includes maximum creep and growth materials properties and also conservatively assumes that grid spring forces never relax. This imposes the maximum axial tensile stress in the shell and post retaining structure and hence the axial extension rate at an unrealistic maximum.

TABLE 3

Summary of Worst Case Rod Elongation Calculations

	Calculated Elongation (inches)	Calculated Rod-to-Baseplate Clearance (inches)	Calculated Rod Free-End Overhang Increase (inches)
		Seed	
18,000	1.519	.400	1.410
24,000	1.751	.188	1.622
		Regular Blanket	
18,000	1.242	1.788	1.142
24,000	1.467	1.576	1.354
		Power Flattening Blanket	
18,000	1.190	1.840	1.090
24,000	1.382	1.661	1.269
		Reflector	
18,000	.517	1.475	.455
24,000	.599	1.397	.533

TABLE 1

Fuel Assembly Length Increase and Axial Clearance at 24,000 EFPH

	Worst Case Length Increase	Best Estimate Length Increase	Minimum Axial Clearance at EOL
Seed	.537	.330	2.090"
Blanket	.343	.225	2.901"
Reflector	.144	.107	.637"

Question

What inservice inspections are being evaluated and planned for late in life conditions (post 24,000 EFPH)?

Answer

An inservice inspection program was performed for the Shippingport Plant prior to LWBR core operations. As noted in the SAR, the requirements of Section XI of the 1974 edition of the ASME Code were utilized as a guide in selecting items for the inservice inspection program. There were no indications found which required repair as a result of the Shippingport inservice inspection program.

Bettis is presently developing an inservice inspection program for post 24,000 EFPH. This program will include a sampling of the welds inspected prior to LWBR initial criticality to verify there is no degradation of these welds; welds of each size used in the plant (these will include welds exposed to thermal transients such as the pressurizer surge line); branch connections off main coolant loops; a sampling of welds previously inspected for pipe whip and dissimilar metal welds; a sampling of coolant loop welds not isolable from the vessel. In addition, specific hydrostatic and visual examinations will be performed.