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

SAFETY EVALUATION AND ENVIRONMENTAL IMPACT APPRAISAL BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 65 TO FACILITY OPERATING LICENSE NO. DPR-38

AMENDMENT NO. 65 TO FACILITY OPERATING LICENSE NO. DPR-47

AMENDMENT NO. 62 TO FACILITY OPERATING LICENSE NO. DPR-55

DUKE POWER COMPANY

OCONEE NUCLEAR STATION, UNITS NOS. 1, 2 AND 3

DOCKETS NOS. 50-269, 50-270 AND 50-287

Introduction

By the applications dated April 20, 1978[13] and June 26, 1978[1], as supplemented April 27, 1978[14], August 21, 1978[12], August 28, 1978[2], September 6, 1978[3], September 22, 1978[4], and September 26, 1978[11], Duke Power Company (the licensee) proposed to change the common Technical Specifications (TS) for the Oconee Nuclear Station, Units Nos. 1, 2 and 3 in connection with the refueling of Unit No. 1 for Cycle 5 operation. The refueling consists of the replacement of 61 burned fuel assemblies by 56 fresh assemblies and five previously burned assemblies. The five previously burned assemblies were last irradiated in Cycle 4 of Oconee Unit No. 1. These assemblies will be irradiated for a fourth cycle as part of a joint Duke Power/Babcock & Wilcox (B&W)/Department of Energy program to demonstrate reliable fuel performance at extended burnups and to obtain post-irradiation data.

Because of performance anomalies observed at other B&W plants, orifice rod assemblies will not be used in Cycle 5.

Cycle 5 will nominally extend for one year. The design cycle length is 320 effective full power days (EFPD). The mode of operation will be feed-and-bleed. Operation of the reactor was converted from the rodded mode to feed-and-bleed to increase operating margin because of a quadrant tilt problem in Cycle 4. The Cycle 5 fuel shuffle pattern was designed to minimize the effects of any power tilt present in Cycle 4.

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Reactivity control during Cycle 5 will be accomplished using the 61 full-length Ag-I_n-Cd control rods and by soluble boron shim. In addition to the full-length control rods, eight axial power shaping rods are provided for additional control of the axial power distribution. Neither control rod interchange nor burnable poison rods are necessary for Cycle 5.

Analyses performed for the Cycle 5 reload core design were based on the following assumptions:

1) Cycle 4 operation is terminated at 250 EFPD.

2) Cycle 5 operation will not exceed 320 EFPD.

The licensee has proposed the following changes to the Technical Specifications for Unit 1. These changes are in accord with the analysis used to support Cycle 5 operation.

1) Revise the protective system maximum allowable setpoint contained in Specifications 2.1 and 2.3 respectively.

2) Revise the xenon reactivity hold Specification.

3) Change the steady state quadrant power tilt limit to 5.00%.

4) Change Specification Figures 3.5.2-1A1, 3.5.2-1A2, 3.5.2-2A1, 3.5.2-2A2, Rod Position Limits, Figures 3.5.2-3A1, 3.5.2-3A2, Power Imbalance Limits, and Figures 3.5.2-4A1, 3.5.2-4A2, Axial Power Shaping Rod (APSR) Position Limits.

5) Add requirements on High Pressure Injection pump operability and operating procedures.

I. Safety Evaluation

Fuel Mechanical Design

The batch 7 fresh fuel uses the Mark B4 fuel assembly design reviewed and accepted by us for use during Cycle 3. Also, these types of fuel assemblies are currently operating in Oconee 3 and Arkansas Nuclear One, Unit No. 1 (ANO-1).

The batch 7 fuel assembly design is based upon established concepts and utilizes standard component materials. Therefore, on the bases of the analyses presented and previous successful operations with equivalent fuel, we conclude that the fuel mechanical design of this fuel is acceptable and does not decrease the safety margin.

Five batch 4D Mark B3 assemblies will remain in the core for their fourth cycle of irradiation and will experience burnups up to approximately 41,000 MWD/MTU. This is part of a joint Duke Power/B&W/ Department of Energy program to demonstrate extended burnup feasibility in light water reactors (LWR's). The fuel is predicted to maintain its structural integrity with these burnups. The licensee states that the fuel parameters most affected by amount of irradiation are fuel rod and assembly growth and fuel swelling. These parameters will remain within the original batch 4D design limits during the Cycle 5 irradiation, as the Final Safety Analysis Report (FSAR) design basis burnup is 44,000 MWD/MTU, significantly greater than the planned 41,000 MWD/MTU exposure. The licensee's evaluation of post irradiation data from two cycles of operation in the Oconee 1 reactor indicate the fuel holddown spring force, which is affected by residence time as well as burnup, will meet performance requirements through the fourth cycle of irradiation.

Creep collapse time of the cladding was calculated to be in excess of 30,000 effective full power hours (EFPH) which is longer than the maximum fuel design exposure for Cycle 5 of 28,469 EFPH for batch 4D fuel. The calculation of creep collapse time was performed using the power history of the limiting fuel assembly. As was done in previous analyses, the CROV computer code was used to predict the collapse time(5). The licensee stated(6) and we agree that the CROV code conservatively predicts cladding collapse.

Additional conservatisms used in the CROV calculations were that no credit was taken for fission gas release; the cladding thickness used in CROV was the lower tolerance limit (LTL) of the as-built measurements; and the lowest as-fabricated pellet densities were assumed to be located in the worst case power region of the core.

The fuel cladding strain analysis was performed using a number of conservative assumptions: maximum allowable fuel pellet diameter and density; lowest permitted tolerance for the cladding inner diameter; conservatively high local pellet burnup; and conservatively high heat generation rate. This insures that the 1.0% limit on cladding plastic circumferential strain is not violated.

We find that the licensee's evaluation of the batch 4D fuel assemblies provides reasonable assurance that the fuel can safely be irradiated for a fourth cycle. Furthermore, coolant activity TS are based upon the equivalence of 1% failed fuel in the reactor. This specification would halt operation of the reactor in the unlikely event that predictions of a low failure rate for the batch 4D fuel are grossly in error. Since the activity corresponding to failure of 1% of the fuel remains a limiting condition for operation of the reactor, irradiation of the five batch 4D fuel assemblies does not result in a reduction of safety margin for Unit 1 Cycle 5 operation.

Fuel Thermal Design

The batch 7 fuel produces no significant differences in fuel thermal performance relative to the other fuel remaining in the core. As was done in earlier Oconee reload calculations, the linear heat rate (LHR) capability was calculated using the TAFY-3 computer code(7). The nominal LHR for Cycle 5 is 5.80 kw/ft and the LHR capability is 20.15 kw/ft.

During the last several years, data have become available that indicate the fission gas release rate from LWR fuel pellets increases with burnup. The effect of enhanced fission gas release on Emergency Core Cooling System (ECCS) performance was significant for B&W fuel. Enhanced release at high burnup affects the fuel rod internal pressure and the pellet volumetric average temperature which are important inputs to the B&W Loss of Coolant Accident (LOCA) analyses. These inputs were calculated for the Oconee 1(5) reload using the TAFY-3(7) fuel performance code which was approved prior to identification of enhanced fission gas release at high burnup. Another B&W fuel performance code, TACO, includes the effects of enhanced release and was also approved by the NRC staff. B&W states that both the rod pressure and volumetric average fuel temperature calculated by TAFY-3 conservatively envelope those calculated by TACO between 2,000 and 42,000 MWD/T fuel rod burnup. We have reviewed this application of the TACO code and concur in the results. The limiting LOCA calculation for this cycle of Oconee 1 occurs at a burnup within this range. Thus, the use of TAFY-3 to calculate the fuel rod pressure and volumetric average temperature input for the LOCA analysis conservatively bounds the effects of enhanced fission gas release.

Nuclear Analysis

The reactor core physics parameters for Oconee 1 Cycle 5 operation were calculated using a PDQ computer code. Since the core has not yet reached an equilibrium cycle, there were minor differences in the physics parameters between the Cycle 5 and Cycle 4 cores.

The licensee proposed a change in the plant Technical Specifications increasing the allowable steady state quadrant tilt from 3.41% to 5.00%. This tilt allowance was appropriately accounted for in the licensee's derivation of rod position, axial shape index, and minimum reactor trip setpoint analyses, and is therefore acceptable.

There was a quadrant flux tilt present in the Oconee 1 reactor during Cycle 4(8). This tilt was 2.4% when full power operation was achieved, and burned out to an insignificant level during the cycle. The shuffle pattern for Cycle 5 was designed to minimize the carry over of any Cycle 4 tilt to Cycle 5. In response to our questions, the licensee provided(2) details of the new shuffle pattern. We have reviewed this information and agree the shuffle pattern will effectively minimize tilt carry over effects from one cycle to the next.

The original Technical Specification tilt limit for Cycle 4 was 3.41%. Early in Cycle 4, a tilt anomaly occurred resulting in the core quadrant tilt exceeding the then current limit. The staff reviewed and approved an increased tilt limit of 6%, with concommittant compensating changes to the Technical Specifications. As the cycle proceeded the tilt decreased. After extensive discussion and study, the licensee proposed and the staf accepted a reduction of the limit to its original value of 3.41% with again concommittant compensating changes to the Technical Specifications. The licensee has established and verified the cause of the Cycle 4 tilt. The tilt was attributed to asymmetry at the end of Cycle 3 burnup distribution which was accentuated by the core loading pattern for Cycle 4. The licensee has revised his methods of selection of the core loading scheme in order to reduce the future potential for core isotopic asymmetries and resultant quadrant tilt. We have reviewed the licensee's analysis of this situation and find it acceptable. The Cycle 4 core exhibited a decreasing tilt with increasing core burnup. During the latter part of Cycle 4, the tilt magnitude was in the order of less than or equal to 1% (the normal range of expected measured tilt).

The licensee has now proposed to increase the current quadrant tilt Technical Specification limit to 5%. The quadrant tilt Technical Specification in conjunction with the control rod insertion limit and power imbalance limit Technical Specifications ensure that plant limiting conditions for operation are not exceeded. These conditions ensure that limiting values of linear heat generation rate and peak enthalpy rise assumed in the safety analysis are not exceeded. These limiting values are not altered by the proposed Technical Specification change. The margin to safety and operating limits have not been altered; hence the Oconee 1, Cycle 5, core is not anticipated to exhibit future anomalous tilt behavior. The change does not alter the probability that the core will exhibit anomalous behavior. Hence, the change is acceptable. The increased tilt limit permits greater operating flexibility with no decrease in safety margin.

The licensee proposed a change to TS 3.5.2.6, Xenon Reactivity. This specification will limit potential Xenon reactivity transients and the associated change in transient power distribution during power operation by restricting the nonequilibrium Xenon reactivity. During steady state operation and power maneuvers at or near rated power, transient Xenon power distribution effects would be compensated for by a proposed 5% allowance in the power imbalance analyses, TS 3.5.2.7, and in the control rod position limit analyses, TS 3.2.2.5. In response to staff questions the licensee has shown the adequacy of the 5% allowance(2). The magnitude of the nonequilibrium Xenon reactivity is calculated by the reactor operator as a function of fuel burnup, core power and power history.

This TS is common to reactors that use the feed and bleed operational mode such as Oconee 1. This change is intended to limit transient Xenon reactivity. Section 3 of the TS limits power operation below the power level cut-off point until "the reactor has operated within a range of 87 to 92% of rated thermal power for a period exceeding two hours in the soluable poison control mode." This TS ensures that plant operation will be in conformance with the assumptions of the analyses described above. Based on the licensee response in Reference 2 and on the fact that this specification has been acçepted for use, for the discussed purpose, at other operating reactors (e.g., Rancho Seco), the staff finds this change acceptable.

We find that, based on our review of the licensee's nuclear analysis techniques and their commitment to perform acceptable physics startup testing, the Oconee 1 nuclear analysis is acceptable. The proposed Technical Specifications of APSR position limits and the usual regulating control rod and imbalance limits, which assure that the LOCA LHR limits are not exceeded, are acceptable because the licensee has determined these limits using appropriate parameters for Cycle 5 and analysis techniques approved for earlier cycles of the Oconee reactors.

Thermal-Hydraulic Analyses

The licensee is proposing to remove all the Orifice Rod Assemblies (ORA) and has revised the thermal-hydraulic analysis accordingly[3]. The core bypass flow has increased to 10.4% (106 ORAs removed) from the 8.34% value used for Cycle 4 analysis (44 ORAs removed).

To offset the increase in core bypass flow, the reference design radial times local peaking factor ($F_{\Delta h}$) has been reduced from 1.78 to 1.71. The most limiting transient, the loss of two reactor coolant pumps, has been reanalyzed with an $F_{\Delta h}$ of 1.71 and the minimum Departure from Nucleate Boiling Ratio (DNBR) remains above 1.3, with the trip setpoints previously established for Cycles 3 and 4. The ORAs were also removed from Oconee Unit 3. This was recently approved for the Unit 3 reload[9].

We have reviewed the licensee's analyses and conclude that the thermal hydraulic analyses for Oconee 1 cycle 5 are acceptable.

Accident and Transient Analysis

The accident and transient analysis provided by the licensee demonstrates that the Oconee FSAR analyses conservatively bound the predicted conditions of the Oconee Unit 1 Cycle 5 core and are, therefore, acceptable. Each FSAR accident analysis has been examined, with respect to changes in Cycle 5 parameters, to determine the effects of the reload and to insure that performance is not degraded during hypothetical transients. The core thermal parameters used in the FSAR accident analysis were design operating values based on calculated values plus uncertainties. FSAR values of core thermal parameters were compared with those used in the Cycle 5 analysis. The effects of fuel densification on the FSAR accident results have been evaluated and are reported in the Oconee Unit 1 fuel densification report[10]. Since Cycle 5 reload fuel assemblies contain fuel rods with theoretical density higher than those considered there, the conclusions derived in that report are valid for Oconee Unit 1 Cycle 5. The limited conditions of the analyses for transients in Cycle 5 are bounded by the initial conditions for previous analyses performed in either the FSAR, the fuel densification report or previous reload submittals. Calculational techniques and methods for Cycle 5 analyses remain consistent with those used for the FSAR. No new dose calculations were performed for Cycle 5 operation. The dose considerations in the FSAR are based on maximum peaking and burnup for all core cycles; therefore, the dose considerations are independent of the reload batch.

ECCS Analysis

This matter has been separately considered by the staff and is discussed in the NRC's Order in the captioned matter dated April 26, 1978, and in the NRC's Exemption in the captioned matter dated October 23, 1978, which accompanies this Safety Evaluation.

Physics Startup Tests

The physics startup test program for Cycle 5 as stated in Section 9 of the reload submittal has been reviewed. The physics startup test program includes zero power measurements of critical boron concentration, temperature coefficients, ejected control rod worth and control rod group reactivity worth. Power distribution, temperature coefficient and power coefficient measurements will be made at higher powers. The acceptance criteria and the actions to be taken if the acceptance criteria are not met were reviewed as well as the tests. The licensee has stated[11] that the action to be taken if the sum of the worth of groups 5, 6 & 7 differs from the predicted by more than ±10%, is to measure group 4 and that if the sum of the worths of groups 4, 5, 6 and 7 differs from the predicted by more than ±10%, additional measurements, as well as evaluation of the discrepancy, will be made.

A summary of the results of this test program will be submitted to the NRC. This entire program has been reviewed by the NRC staff and found to be acceptable.

Effects of Fuel Demonstration Program on Accident Analysis

Irradiating the entire core to extended burnups of about 41,000 MWD/MTU, not just the five demonstration fuel assemblies, would increase the amount of long-lived fission products in the core. The only significant long-lived radioisotope of concern with respect to the potential consequences of the postulated design basis accidents is the noble gas

Krypton 85. Even if the entire core burnup were extended to 44,000 MWD/MTU, the FSAR assumption for Design Basis Accidents, the amount of Krytpon 85 generated would not show an increase; therefore, the potential consequences of the postulated design basis accidents given in our Safety Evaluation (SE) dated December 29, 1970, for Oconee Unit 1 will not change because fuel assemblies in the core will be irradiated to burnups of only 41,000 MWD/MTU, and only for five fuel assemblies not an entire core of 177 fuel assemblies.

Conclusion on Safety

Based on our evaluation of the reload application and available information, we conclude that it is acceptable for the licensee to proceed with Cycle 5 operation of Oconee 1 in the manner proposed.

We have reviewed the proposed changes to the Technical Specifications and find them acceptable. These consist of all the changes requested by the licensee in his letter of June 26, 1978[1], except for Figure 2.3-2A which was acceptably revised in the supplement of September 6, 1978[3], and the submittal of April 20, 1978[13], as supplemented April 27, 1978[14], which provides both for timely operator action and maintenance of all the High Pressure Injection pumps in an operable condition in the unlikely event of a small break LOCA during plant operation. The TS for the Oconee Nuclear Station, in terms of radioactivity in the primary coolant and radioactivity releases from the station need not be revised for the five batch 4D Mark B3 demonstration fuel assemblies. These TS are based on a 44,000 MWD/MTU burnup, while the demonstration assemblies will experience only about 41,000 MWD/MTU burnup.

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

II. Environmental Conclusion Regarding Cycle 5 Reload Excluding Fuel Demonstration Program

We have determined that this action does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that this change involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this change.

III. Environmental Considerations of Fuel Demonstration Program

By letter dated October 23, 1978[15], the Department of Energy (DOE) the cognizant Federal agency for the fuel demonstration program, of which Oconee Unit No. 1 is a small portion, stated that an environmental review of possible future DOE funded extended fuel burnup work and widespread utilization of the process is not required at this time.

We have considered the effect of irradiating five fuel assemblies to extend burnups in Oconee 1 on the environmental impacts from the uranium fuel cycle and from shipping fuel and waste to and from Oconee Unit 1. We conclude that these five assemblies will have no significant effect on these environmental impacts over the operating lifetime of the plant. The licensee is not expecting at this time to change the amount of uranium or the number of fuel assemblies shipped to and from the plant by irradiating the five assemblies to extended burnups. The licensee will add five fewer new fuel assemblies than normal to the core for Cycle 5 and will add five more new fuel assemblies than normal to the core for Cycle 6. The remaining cycles, as now planned, will have the normal number of new assemblies. Irradiating these five fuel assemblies to extended burnups does not increase the number of fissions in any fuel cycle for Oconee Unit 1 or over the operating lifetime of the plant, therefore, the amount of fission products generated by Oconee Unit 1 over its operating lifetime does not change. There will be more than the normal amount of long-lived fission products in the core during Cycle 5 and fewer during Cycles 6 and 7. Therefore, on the average, each fuel assembly will have the same magnitude of fission products as if these five assemblies were not irradiated to extended burnups.

The proposed action will therefore not significantly increase normal radiological effluents from the plant. It will also not allow the licensee to discharge concentrations greater than the maximum allowed nor to discharge more activity in a year than the maximum allowed. Compliance with the present TS will adequately control releases such that there will be no appreciable effect on the environment due to operation under these proposed changes.

Conclusion and Basis for Negative Declaration

On the basis of the NRC evaluation and information supplied by the licensee, it is concluded that the proposed action will have no appreciable impact on the environment due to radiological effluents from the plant and will not affect the cost/benefit balance.

Having reached these conclusions, the Commission has determined that an environmental impact statement need not be prepared for this proposed change and that a Negative Declaration to that effect should be issued.

Dated: October 23, 1978

REFERENCES

1. Ltr. from William O. Parker, Jr., Duke Power Company (DPC) to R. Reid, U. S. Nuclear Regulatory Commission (NRC), 6/26/78, forwarding the Oconee Nuclear Station, Unit No. 1, Cycle 5 Reload Report, BAW-1493.

2. Ltr. from William O. Parker, Jr., (DPC) to R. Reid (NRC), 8/28/78, forwarding additional information.

3. Ltr. from William O. Parker, Jr., (DPC) to R. Reid (NRC), 9/6/78, forwarding Revision 1 to BAW-1493.

4. Ltr. from William O. Parker, Jr., (DPC) to R. Reid (NRC), 9/22/78, forwarding additional information.

5. BAW-1493, "Oconee Unit 1, Cycle 5, Reload Report," 7/78.

6. BAW-10084, Rev. 1, "Program to Determine In-Reactor Performance of B&W Fuels - Cladding Creep Collapse," 11/76.

7. BAW-1004, "TAFY-Fuel Pin Temperature and Gas Pressure Analysis," 5/72.

8. BAW-1477, "Oconee 1 Cycle 4 Quadrant Flux Tilt," 1/78.

9. Ltr. from R. Reid (NRC) to William O. Parker, Jr., (DPC), 7/6/78.

10. BAW-1388, "Oconee 1 Fuel Densification Report, Revision 1," 7/73.

11. Ltr. from William O. Parker, Jr., (DPC) to R. Reid (NRC), 9/26/78.

12. Ltr. from William O. Parker, Jr., (DPC) to R. Reid (NRC), 8/21/78, ORA Removal and Exemption Request.

13. Ltr. from William O. Parker, Jr., (DPC) to R. Reid (NRC), 4/20/78, HPI Pump Operability.

14. Ltr. from William O. Parker, Jr., (DPC) to R. Reid (NRC), 4/27/78, Operating Procedures Related to Small Break LOCA.

15. Ltr. from Peter Lang, Department of Energy to R. Reid (NRC), 10/23/78, Environmental Considerations of Fuel Demonstration Program.

APPENDIX A

LOST STEAM GENERATOR TUBE PLUGS AT OCONEE UNIT 1
SAFETY EVALUATION REPORT
ENGINEERING BRANCH, DIVISION OF OPERATING REACTORS

I. BACKGROUND

On Thursday, October 19 Duke Power Company (the licensee) informed the NRC that two steam generator tube plugs had been lost at the Oconee Unit 1 Nuclear Power Plant and were believed to be loose in the primary coolant system. The two plugs were lost during tube plugging operations in the Unit 1, B steam generator.

One plug being installed in the top of a tube in the "1B" steam generator failed to detonate. In order to remove the faulty plug the tube was pressurized from the unplugged end and the plug was forced out of the tube. However, when an attempt was made to retrieve the plug it could not be located in the upper head region and it is therefore believed that the plug may have entered the hot leg pipe from the reactor vessel. The hot leg pipe runs horizontally from the reactor pressure vessel and then vertically through the "candy cane" configuration into the upper steam generator head.

A second plug was discovered missing during review of photos of the lower tube sheet which are taken to confirm that the tube plugging operation has been properly completed. One tube which was thought to have been plugged was determined to be unplugged from the photograph. The licensee has suggested that a tube might have been double plugged. This means that the "jumper" who inserts the tube plugs during the plugging process could have possibly inserted a second plug in a previously plugged tube rather than the tube intended for plugging. Since the plugs are inserted deep into the tube sheet this is a possibility. However, since there is no way to confirm this scenario, it must be assumed that the plug has been lost in the lower steam generator head or in the cold leg of the reactor coolant system. However, the licensee has indicated that all plugs that were placed in lower tube sheet had properly detonated. Thus it is believed that the plug lost in the lower head was detonated. The lost plugs are approximately 3 inches in length, one half inch in diameter and 1/2 pound in weight.

In their October 19, 1978 submittal and in telephone conversations on October 19 and October 20 the licensee has addressed concerns regarding (1) potential for detonation of the undetonated plug (2) consequences of plug detonation, and (3) the consequences of loose parts in the primary coolant system.

II. DISCUSSION

A. Significance of Undetonated Plug

1. Potential for Detonation of Undetonated Plug

Babcock and Wilcox has run several tests to assess the potential for detonation of the undetonated lost plug. These tests were conducted with the same type of plugs that were lost at Oconee Unit 1 which are delivered pre-assembled by the manufacturer. Two plugs were heated in pressurized (2250 psi) reactor coolant grade water (6000 ppm H_3BO_3, 1.0 ppm LiOH) to 620°F at approximately 64°F per hour. This testing indicated no evidence of detonation and examination at the conclusion of the testing indicated that the explosives had dissolved from the plug and no solids remained in the plugs. Babcock and Wilcox consultation with duPont Explosive Products Division and military explosive personnel confirmed that decomposition of the chemical explosives will occur when the plugs are heated at temperatures and rates comparable to those existing during reactor coolant system heatup. Four plugs were also heated in air as high as 980°F with no evidence of detonation. The explosive vaporizes at 290°F and therefore would not be in an explosive geometry beyond this temperature.

A second set of testing included impact testing with dry and wet plugs. Impact testing with dry material indicated that detonation could occur at an impact energy of 25 Ft-lbs. Under wet conditions impact energy as high as 185 Ft-lbs. did not cause detonation. Calculations by Babcock and Wilcox indicate that 185 ft-lbs. bounds the impact energy which a plug could be subjected to in the RCS.

2. Consequences of Detonation

Although the licensee maintains that the probability of the unrecovered, undetonated plug not decomposing and subsequently detonating during operation is negligible, they addressed the consequences of such an event in the October 19 submittal. We have been informed that a tube plug was detonated in air by the licensee. As a result of the detonation the walls of the hollow plug flared open in three sections. No shrapnel effects were observed. If a plug detonated within a steam generator tube outside of the tubesheet area, the affected tube and approximately ten surrounding tubes could be affected. The basis of this scenario is that the tube containing the plug might burst and that it could then cause damage to the immediately adjacent tubes. The primary to secondary leak which might result would be promptly detected and the unit brought to cold shutdown.

If the plug is postulated to detonate in the vicinty of the fuel assemblies, several rods could be affected. It is not believed that the explosive energy of the plug would be sufficient to damage more than a limited number of rods.

B. Significance of Detonated and Undetonated Plugs As Loose Parts

1. Loose Parts Monitoring Capability

The Loose-Parts Monitoring System (LPMS) installed at Oconee 1 is an early model of the system marketed by B&W. The LPMS uses piezoelectric crystal accelerometers to detect the sounds or vibrations associated with a loose part impacting in the primary system. The B&W system differs from that of other vendors in that the low frequency natural resonances of the pressure vessesl ("bell" frequencies) are utilized for detection, whereas other LPMS vendors use much higher, ultrasonic frequencies.

The design of the system assumes that debris in the primary coolant loop will rapidly migrate to natural collection areas, in this case the inlet plena of the reactor vessel and the two steam generators. Therefore, only these areas are instrumented with LPMS sensors. However, actual experience has shown that impacts at a considerable distance from the sensors can still be detected, although with somewhat reduced sensitivity. For example, an identical LPMS on Oconee 2 was able to detect a loose surveillance capsule tube in 1976.

The sensitivity of the LPMS is limited by the false alarm rate. At the alarm levels now in use at Oconee 1, false alarms occur at the rate of one or two per day. However, by checking loose part alarms against known events such as control rod stepping, most of these alarms can be discounted by the operations personnel. The remainder are investigated by manual monitoring using headphones or a loudspeaker. The licensee has been using this system for nearly six years, and has become quite skilled in its use. The LPMS on Unit 2 was successfully used to detect loose parts in 1974 and 1976. There was one incident on Unit 3 in 1976 where the LPMS failed to detect two small objects. However, the two objects were found lodged in place, and therefore would not be expected to trigger an LPMS alarm.

It should also be noted that a similar LPMS was used in 1978 to detect ejected burnable poison assemblies in the Crystal River 3 reactor. Since the Crystal River incident, B&W has recommended to its customers that extra attention be given to the LPMS.

The question of greatest interest for Oconee 1 is: will the LPMS detect a loose steam generator plug? Regulatory Guide 1.133 requires new plants to install systems capable of detecting impacts of energies of 1/2 ft-lb within 3 feet of a sensor. LPMS manufacturers claim no difficulty with this sensitivity, provided the background noise of the reactor is sufficiently low. Although detailed data on the Oconee system's signal to noise ratio is not readily available, it is expected that the system sensitivity is of this order. Therefore, the LPMS is probably capable of detecting a loose plug wandering randomly in an inlet plenum, since that is where the detectors are. More importantly, the system is almost certain to detect impacts energetic enough to cause damage provided some of these impacts involve the outer vessel wall or some other component with a direct acoustic path to a sensor.

2. Consequences to Reactor Internals

   a. Mechanical damage

      A steam generator plug weighs approximately 1/2 lb. If it is moving with the coolant (≈15 ft/sec.), it will have a kinetic energy on the order of 1 3/4 ft-lb. No data on the threshold for impact damage is available for B&W fuel. However, another reactor vendor has found that one fuel rod can absorb either one ft-lb. of bending energy, or about 250 ft-lbs. of compression loading before cladding failure. The B&W fuel rod should not be greatly different in behavior. It is not credible that a steam generator plug could enter the fuel lattice and still posess enough transverse velocity to apply 1 ft-lb. to bend a fuel rod. Nor is it credible that the plug could hit the end of a rod with sufficient velocity to cause failure due to compression loading. This does not take credit for the additional protection supplied by the grid spacers and upper and lower tie plates.

      Damage to the control rods is also not credible. The control rods are protected by guide tubes when withdrawn, and are better protected than the fuel rods when inserted. It is instructive to note that the control rod guide tubes successfully protected the control rods from the considerably more massive burnable poison rod assemblies during the recent Crystal River incident.

      The remainder of the internals should not be damaged by impacts of less than 2 ft-lbs. The steam generator plug should be able to travel freely about the plenum, thus there is no concern for fatigue due to repeated impacts at one location.

b. Flow blockage

Because of the small size of the steam generator plug and the relatively high cross flow within the core, it should not be possible for the plug to cause departure from nucleate boiling, even during a transient by blocking flow at the core inlet.

If the loose plug should enter the fuel lattice, which is quite improbable considering the size and weight of the object and the size of the openings in the lower tie plates, it still will probably not cause DNB. Safety analyses of such situations in the past (generally borrowed from fuel rod bowing calculations) have shown that the decreased neutron moderation caused by displacement of the moderator by the object will lower power in the immediate vicinity of the object and maintain margin to DNB. The steam generator plugs are hollow and therefore do not displace as much moderator as a solid object would.

In any case, the steam generator plug would have to travel to a high power area of the core to cause any concern with DNB, which would require the penetration of several but not all grid spacers. Moreover, only four rods would be affected. Therefore, it is concluded that flow blockage induced DNB is not a concern.

c. Mechanical interference.

The only moving parts within the reactor vessel are the control rods and the vent valves. Since the vent valves remain closed during normal operation and are needed only in the event of a LOCA, and since a loose part is not likely to remain in the upper plenum (and even less time in the downcomer), mechanical interference with the operation of the vent valves is not a problem. Interference with control rods is somewhat more serious in that control rods are moved more often, but is still not a problem because:

- interference should be detected by control rod exercise programs already in the Technical Specifications,
- The direction of flow at the slots in the control rod weldments is outward, making it difficult for a loose object to enter,
- even under the worst-case conditions of a steam line break at end-of-cycle when the reactivity defects are at their maximum, the safety analyses assume the worst rod stuck out of the core, and
- under anticipated transient conditions, it is known from calculations carried out for the ATWS investigations that the reactor will still scram even if 5 clustered rods fail to insert.

Therefore, it is concluded that mechanical interference with moving parts within the reactor vessel is not a problem.

3. Consequences to Steam Generator

If a plug is in the reactor outlet portion of the RCS, it may be carried into the upper head of the steam generator. Experience with loose objects in the steam generator upper head has shown that the plug would not become lodged but would continue to impact the upper tubesheet. Recent experience at Crystal River has shown that impacting by loose parts, much larger than a tube plug, did not result in significant damage to the twenty four inch thick tubesheet, tubesheet cladding or tube to tube sheet joints. Any significant impact would be detected by the installed Loose Parts Monitoring System and the unit would promptly be brought to shutdown condition for retrieval of the plug. Thus, any damage to the steam generator would be expected to be minimal. Furthermore, the 0.3 gpm steam generator primary to secondary leakage rate technical specification limit would require prompt corrective action in the improbable event of primary system degradation resulting from damage imparted by a loose tube plug.

4. Consequences To The Reactor Coolant Pump

The primary coolant recirculation pump is a single stage centrifugal type pump with a diffusser. The diameter of the impeller is approximately 30 inches. The manufacturer (Westinghouse) was contacted to determine what would happen to this pump if the steam generator tube plug could reach the suction and be ingested into the pump internals. They indicated that the diffusser and impeller vane passages have adequate clearance for the plug to flow through. If the tube plug were to impact the pump internals, minor damage would be incurred. He further indicated that if the plug were to be lodged within a vane passage of the impeller that there would be higher detectable vibrational levels within the pump, but that the pump would not catastrophically destruct since the pump was designed for unbalanced rotor operation.

In view of the above information, even if the plug were to flow within this pump there is reasonable assurance that pump pressure boundary integrity would be maintained and that major damage to pump internals would not occur. Furthermore, the loss of one reactor coolant pump is an event determined to be acceptable in the licensee's transient accident analysis.

5. Similar experiences at Westinghouse plants

Of the Westinghouse experience, the most similar event occurred at Turkey Point 4 in June, 1977. During a steam generator inspection and tube plugging operation, it was discovered that twelve of the steam generator tubes presumed to have been plugged during the previous outage were not plugged. A check of the plant records was unable to produce definite proof that the steam generator tube plugs had indeed been installed. The reactor was defueled and both the reactor and main coolant pipes were searched by TV cameras. No plugs were found. It was concluded that the plugs had never been installed, and the reactor was reassembled. At this point, an LPMS was installed. When the reactor coolant pumps were started, the LPMS detected a loose part impacting the lower vessel head. Subsequent testing indicated impacting only at less than full-flow conditions. During the testing, the impact indications stopped, presumably because the loose part had jammed or found a low-flow area. Analysis of the data tapes indicated that there was only one loose part moving randomly about the lower plenum. After further pump testing, which failed to dislodge the loose part and appropriate safety evaluations, the reactor was returned to service. The loose part is still in the vessel, and was heard on the LPMS during pump tests after refueling in September, 1978.

III. EVALUATION

Based on the above discussion the staff has reached the following conclusions:

1. Detonation of the undetonated plug is highly improbable. B&W has conducted sufficient testing to establish that the explosive in the plug will disintegrate in the primary coolant system environment.

2. The consequence associated with the unlikely event of the plug exploding are not unacceptable. Damage to the steam generator or reactor intervals would be minimal.

3. The significance of the tube plugs as loose parts is minimal. Loose plugs will not unacceptably affect the reactor internals, steam generators, or reactor coolant pumps. The licensee has an excellent LPMS for monitoring any activity of the loose parts.

4. Similar events in other plants have not resulted in unacceptable consequences.

It is therefore our conclusion that operation of Oconee Unit 1 with the loose plugs in the primary coolant system is acceptable.