Attachment 1

03C16 Reload Report Revision

7. ACCIDENT AND TRANSIENT ANALYSIS

7.1 General Safety Analysis

Each FSAR[1] accident analysis has been examined with respect to changes in Cycle 15 parameters to determine the effect of the Cycle 16 reload and to ensure that thermal performance during hypothetical transients is not degraded. The effects of fuel densification on the FSAR accident results have been evaluated and are reported in Reference 6. Since Batch 18 reload fuel assemblies contain fuel rods with a theoretical density higher than those considered in Reference 6, the conclusions in that reference are still valid.

No new dose calculations were performed for the reload report. The dose considerations in Reference 14 are characteristic for Oconee 3 Cycle 16.

7.2 Accident Evaluations

The key parameters that have the greatest effect on determining the outcome of a transient can typically be classified in three major areas: core thermal parameters, thermal-hydraulic parameters, and kinetics parameters, including the reactivity feedback coefficients and control rod worths.

Fuel thermal analysis parameters for each batch in Cycle 16 are given in Table 4-2. Table 6-1 compares the Cycle 15 and 16 thermal-hydraulic maximum design conditions. The Oconee 3 Cycle 16 key physics and kinetics parameters have been compared against the corresponding parameters assumed in the Oconee FSAR analyses. For the parameters that were non-conservative with respect to the corresponding FSAR values, the respective accident analyses have been evaluated to confirm that the FSAR licensing basis analyses still remains bounding.

The combination of average fuel temperature as a function of LHR and the lifetime pin pressure data used in the BAW-1915 LOCA limits analysis[13] are conservative compared to those calculated for this reload. Thus, the analysis and the LOCA limits reported in BAW-1915 provide conservative results for the operation of Oconee 3 Cycle 16 fuel[5].

The cycle specific LOCA kW/ft limits for Oconee 3 Cycle 16, using Mark-B10T fuel assemblies, have been determined utilizing the approved LOCA evaluation model methods outlined in References 18 and 19 with fuel performance input based on TACO3[7]. The Oconee allowable

LOCA LHR limits for the first cycle of Mark-B10T fuel (BOL-25 GWd/mtU) are listed in Table 7-1 below.

Table 7-1 Mark B10T fuel assembly LHR Limits for Oconee 3 Cycle 16

Core Elevation, ft	Mark-B10T LOCA LHR Limit BOL - 25 GWd/mtU
2	17.2
4	17.0
6	17.0
8	16.5
10	17.0

7.3. Fuel Dimensional and Weight Changes

The new fuel loaded for Oconee Unit 3 Cycle 16 is B&W Mark B10T fuel. This fuel is characterized by a number of dimensional changes from the Mark B10 fuel loaded for the previous cycle. The fuel pellets in the Mark B10T fuel have a slightly larger outside diameter (0.3735 inches) than the Mark B10 fuel (0.3700 inches). The fuel to clad gap is decreased to 6.5 mils from the Mark B10 fuel to clad gap of 7.0 mils. The Mark B10T fuel cladding is thinner than the Mark B10 fuel (25 mils vs. 26.5 mils). Also, the active fuel stack height for the Mark B10T fuel (142.29 inches) is taller than the Mark B10 fuel (140.6 inches). The impact of these and future fuel design changes on the FSAR Chapter 15 safety analyses was evaluated. Specifically, an evaluation of the reduction in core average heat flux due to the longer fuel stack height and the greater fuel-to-coolant heat transfer capability due to a thinner clad and gap was performed for each of the Oconee FSAR Chapter 15 Accidents. The greater fuel-to-coolant heat transfer capability will cause lower fuel temperatures on power increases and will tend to mitigate the amount of fuel heatups. This will have the greatest effect on transients involving (1) fuel heatups, e.g., flow reductions, and (2) power increases due to narrow ranges of positive reactivity insertions, e.g., rod ejections rather than rod withdrawals. The flow reductions are the most affected transients. The specific mechanism is that the negative fuel reactivity (Doppler) feedback is not as high since the fuel does not heat up as much. All FSAR Chapter 15 transients are evaluated below to determine the affect of the dimensional changes associated with the Mark B10T fuel and future fuel types.

15.1 Uncompensated Operating Reactivity Changes

The effect of the fuel dimensional changes may effect the physics parameters assumed in the basis for operating reactivity changes and other FSAR Chapter 15 analyses. Any effect on the cycle specific physics parameters are evaluated each cycle, as part of the reload process. The Oconee cycle specific physics parameters are evaluated each cycle to ensure that they are conservative with respect to the current Oconee FSAR analysis values. Therefore, any operating reactivity changes will be bounded as long as the cycle specific physics parameters are determined to be conservative with respect to the licensing basis analyses values.

15.2 & 15.3 Startup Accident & Rod Withdrawal Accident at Rated Power

For this analysis, the design criteria which must be met are, reactor coolant system thermal power shall be limited to < 112% full power and reactor coolant system pressure shall not exceed code pressure limits. The thinner gap and clad of the Mark B10T fuel give better heat transfer from the fuel to the moderator, this will result in lower fuel temperatures and less Doppler feedback during the neutron power increase. With less Doppler feedback the neutron power will increase at a slightly higher rate than would occur in the Mark B10 fuel and result in a high neutron flux level trip sooner for fast reactivity insertion rates. The reactor coolant system thermal power will be limited to less than 112% full power by the high neutron flux level trip for fast reactivity insertion rates and the high pressure trip for slow reactivity insertion rates. Per the Oconee FSAR a sensitivity analysis was performed to determine the effects of varying the reactivity addition rate. The reactivity addition rate was varied from more than an order of magnitude below the nominal single rod group withdrawal rate used for the reference analysis to a rate above that for simultaneous withdrawal of all control rods. The reference FSAR analysis demonstrated that the reactor is completely protected against any startup accident involving the withdrawal of any or all control rods An FSAR analysis of the reactivity insertion rate spectrum was also performed to confirm that the high flux level and high pressure trips will limit reactor thermal power to less than the peak power acceptance criteria for rod withdrawal at rated power accidents.

The thinner gap and clad of the Mark B10T fuel results in better heat transfer, so a given reactivity insertion will result in a slightly steeper pressure rise due to a faster temperature rise in the coolant. Since the pressure rise is directly related to the amount of power generated by the reactivity addition, the reactivity addition rate spectrum will cover the

small changes in reactivity addition for these transients. The most limiting RCS overpressure transient for rod bank withdrawal events has been determined to be the control rod bank withdrawal event at zero power (Startup Accident). For the startup accident, the licensing basis peak pressure analysis assumes a reactivity addition rate at the maximum rate which will not result in a high flux trip in order to maximize the RCS pressure. Therefore, any increase in effective reactivity addition rate due to less Doppler feedback above that assumed in the licensing basis analysis for this event will result in a reactor trip on the high flux level trip. Since the maximum pressure transient occurs at the transition point between tripping on the high pressure trip and the high flux level trip, the peak pressure transient will be limited by the high flux level trip. The rod withdrawal accident at rated power exhibits the same behavior and is bounded by the startup accident peak pressure analysis

Therefore, it is concluded that the high pressure trip and the high flux level trip adequately protect the reactor against any startup or rod withdrawal at rated power accident.

15.4 Moderator Dilution Accident

This transient is characterized by low reactivity addition rates and slow transient behavior. The changes in reactivity addition and fuel to moderator heat transfer rates will be negligible for this transient. The changes in the thermal power and peak RCS pressure will be very small; thus, the thermal power will not exceed 112% rated power and the system pressure will not exceed code allowable limits. Therefore, it is concluded that the high reactor coolant temperature trip or the high pressure trip will continue to adequately protect the reactor against a moderator dilution accident.

15.5 Cold Water Accident

The protection criteria for this accident are that the minimum DNBR be greater than the acceptance criterion for the correlation used and that the system pressure limits not be exceeded. The thinner gap and clad of the Mark B10T fuel give better heat transfer from the fuel to the moderator, this will result in lower fuel temperatures and less Doppler feedback during the neutron power increase. With less Doppler feedback the neutron power will increase at a slightly higher rate than would occur in the Mark B10 fuel and result in a high neutron flux level trip sooner. The FSAR analysis shows that the flux trip is actuated very quickly and the thermal power only reaches 62%. The change in neutron power will have

very little effect on thermal power since the neutron flux trip occurs very quickly. Therefore, any increase in thermal power and decrease in DNBR due to the Mark B10T fuel will be negligible, and the cold water accident will continue to be non-limiting with respect to DNBR. The thinner gap and clad of the Mark B10T fuel results in better heat transfer, so a given reactivity insertion will result in a slightly steeper pressure rise due to a faster temperature rise in the coolant. The increase in heat transfer and associated pressure rise will result in a slight increase in the total RCS pressure result. However , this transient is not limiting with respect to the pressure code limits and any slight increase in RCS pressure will be within the available margin and continue to be within the pressure code limits.

15.6 Loss of Coolant Flow Accident

The loss of coolant flow accident consists of several different possible scenarios, a partial loss of flow (failure of one or more pumps), full loss of flow (loss of electrical power to all pumps), locked rotor (mechanical failure of one pump resulting in locked rotor or sheared shaft). The reactor protection criterion for loss of coolant flow conditions resulting from electrical malfunction of reactor coolant pumps or their power supply is that the minimum DNBR experienced by the core shall be greater than the acceptance criteria for the correlation used. The worst DNB case has been determined to be the partial loss of coolant flow (two pump coastdown) transient. This transient was evaluated for the new Mark B10T fuel and the DNBR results were found to be conservative with respect to the generic Mark B8 fuel limits. The overall effect on this transient of the Mark B10T fuel dimensional changes is a DNB benefit. Therefore, the full or partial loss of coolant flow will continue to meet the DNBR acceptance criteria using Mark B10T fuel.

The locked rotor and sheared shaft event are affected by the increased neutron and thermal power prior to reactor trip like the full or partial loss of coolant accident. Unlike the loss of full coolant flow transient, which assumes a reactor trip on the detection of loss of power to the pumps, the locked rotor transient is tripped by the flux/flow imbalance trip. With a slightly increased neutron power following the locked rotor the flux/flow trip will occur slightly sooner. In the FSAR analysis the DNBR reaches the acceptance criteria limit very quickly (0.9 seconds). Since the DNBR limit with the Mark B10T fuel will be reached slightly sooner, the assumed switch from nucleate to film boiling will begin sooner. The increase in thermal power as a result of the increase in neutron power will be small. The Mark B10T fuel has better fuel to moderator heat transfer which will remove more heat from the fuel resulting in a lower fuel temperatures. Therefore, the peak fuel clad

temperature change will be small and the effect on the results of the locked rotor and sheared shaft accidents will be negligible.

15.7 Control Rod Misalignment Accidents

Control rod misalignment accidents consist of two main varieties, the statically misaligned rod where a control rod becomes stuck at some position during rod movement or stuck fully withdrawn during a reactor trip and the dropped rod where a control rod drops partially or fully into the core. The protection criteria for this accident are that the minimum DNBR be greater than the acceptance criterion for the correlation used and that the system pressure limits not be exceeded. The use of Mark B10T fuel will not effect the static misalignment of a control rod events since these events do not exhibit the transient phenomena of concern. The main concern for stuck rod events is that the stuck rod could result in insufficient negative reactivity on reactor trip to maintain the reactor shutdown. This is prevented by core design criteria. The effect of the new fuel on the dropped rod event will be on the rate of changes in neutron and thermal power during the event. The magnitude of the analysis results will not be affected and the transient will terminate at the same power levels, pressures, etc.. Therefore, the dimensional changes for the Mark B10T fuel will not adversely affect the dropped rod event.

15.8 Loss of Electric Load Accidents

The protection criteria for this accident are that the minimum DNBR be greater than the acceptance criterion for the correlation used and that the system pressure limits not be exceeded. The loss of load accident is not a limiting DNBR or peak pressure transient and is bounded by other more limiting moderator heatup transients. This event is protected by the high reactor coolant temperature and high pressure trips should the temperatures and pressure approach these setpoints.

15.9 Steam Generator Tube Rupture Accident

The steam generator tube rupture accident is a radioactivity dose release analysis where the protection criteria is that the dose shall be within the limits specified in 10CFR100. Therefore, the fuel dimensional changes will not affect the results of this transient.

15.10 Waste Gas Tank Rupture

This accident is a radioactive dose release analysis which does not involve a reactor coolant system transient and is therefore not affected by fuel dimensional changes.

15.11 Fuel Handling Accidents

This accident is a radioactive dose release or criticality analysis which does not involve a reactor coolant system transient and is therefore not affected by fuel dimensional changes.

15.12 Rod Ejection Accident

The protection criteria for the rod ejection accident are that the accident will not further damage the RCS and that the offsite dose will be within the 10CFR100 limits. The first criterion is met if the reactivity excursion does not result in fuel vaporization and therefore the potential for an explosion in the core. This is shown by demonstrating that the peak fuel pellet enthalpy does not approach the threshold 280 cal/g. Similar to the rod withdrawal accident at power, the rod ejection accident will have lower fuel temperatures and less Doppler feedback during a neutron power increase since the Mark B10T fuel will give better heat transfer from the fuel to the moderator, this will result in a slightly increased neutron power excursion. However, the better fuel to moderator heat transfer will reduce the peak fuel pellet enthalpy ensuring that the fuel pellet enthalpy will remain below the 280 cal/g threshold. With a slightly increased neutron power following the rod ejection the reactor trip will occur slightly sooner than the analysis in the Oconee FSAR. The change in thermal power as a result of the increase in neutron power and earlier reactor trip will be negligible. Therefore, the change to the number of fuel pins in DNB, from the analysis results in the FSAR rod ejection accident, will be negligible. In addition, it was determined that a reactivity addition of 1.52 % $\Delta k/k$ would be required to compromise reactor vessel integrity. Since the maximum rod worth has not changed from that used in the FSAR analysis (0.65 % $\Delta k/k$) reactor vessel integrity will be maintained. Therefore, no additional damage to the RCS will occur since fuel vaporization will not occur. In addition, any increase in offsite dose will be negligible and continue to be well within the 10CFR100 dose limits.

15.13 Steam Line Break Accident

The protection criterion of interest for this event is that the core will remain intact for effective core cooling, assuming minimum tripped rod worth with a stuck rod. During a steam line break event the RCS is overcooled resulting in a positive reactivity insertion. This reactivity addition causes the neutron power to increase until the reactor is tripped on the high flux level trip. A slight increase in neutron and thermal power will occur prior to reactor trip since less reactivity feedback occurs during fuel heatup. This will be somewhat offset by an earlier trip on neutron power and the decrease in positive reactivity by the greater transfer of heat from the fuel to the moderator. Therefore, any changes in RCS response as a result of the Mark B10T fuel dimensional changes will be negligible and the protection criterion will continue to be met.

15.14 Loss of Coolant Accidents

In order to determine the acceptability of the performance of the ECCS in mitigating a loss of coolant accident (LOCA), 10CFR50.46 requires that the results of the LOCA analysis meet the following acceptance criteria:

- The calculated fuel element cladding temperature shall not exceed 2200 °F.
- The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.
- The calculated amount of hydrogen generated from the cladding-water reaction shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders were to react.
- Calculated changes in core geometry shall be such that the core remains amenable to cooling.
- After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for an extended period of time.

The changes to the fuel dimensions for the Mark B10T fuel will result in lower average linear heat rates. This, combined with the greater heat transfer from fuel to moderator, will result in lower average fuel and clad temperatures and ensure that the effects of the new fuel dimension changes will not adversely affect the FSAR LOCA analyses. The fuel temperature, cladding oxidation and hydrogen generation results for the Mark B10T fuel

will be conservative with respect to the current FSAR analysis. In addition, the core will continue to maintain a coolable geometry and long term core cooling capability. Therefore, the Mark B10T fuel dimensional changes will not adversely affect the Oconee FSAR LOCA analysis results and the acceptance criteria of 10CFR50.46 will continue to be met.

From the examination of Cycle 16 core thermal properties and kinetics properties with respect to acceptable licensing basis values, it is concluded that this core reload will not adversely affect the safe operation of Oconee 3 during Cycle 16. Considering the previously accepted licensing basis analyses documented in the Oconee FSAR, the Cycle 16 transient evaluation demonstrates that the transient analyses for this cycle are identical to or conservative with respect to previously accepted analyses. The initial conditions of the transients in Cycle 16 are bounded by the FSAR and/or the fuel densification report[6].

REFERENCES

1. Oconee Nuclear Station, Units 1, 2, and 3 Final Safety Analysis Report, Docket Nos. 50-269, 50-270, and 50-287.

2. Core Thermal-Hydraulic Methodology Using VIPRE-01, DPC-NE-2003A, Duke Power Company, Charlotte, NC, July 1989.

3. Oconee Unit 3, Cycle 15 - Reload Report, DPC-RD-2023, Duke Power Company, February 1994.

4. TACO2 - Fuel Performance Analysis, BAW-10141P-A, Rev. 1, Babcock & Wilcox, June 1983.

5. Letter from A. C. Thadani (NRC) to C. H. Turk (B&WOG), Acceptance for Referencing Topical Report BAW-1915 "Bounding Analytical Assessment of NUREG-0630 Models on kW/ft Limits with the use of FLECSET", October 12, 1987.

6. Oconee 1 Fuel Densification Report, BAW-1388, Rev. 1, Babcock & Wilcox, July 1973.

7. TACO3 - Fuel Performance Analysis, BAW-10162P-A, , Babcock & Wilcox, November, 1989.

8. Oconee Nuclear Station Reload Design Methodology II, DPC-NE-1002A, Duke Power Company, Charlotte, North Carolina, October 1985.

9. Correlation of 15x15 Geometry Zircaloy Grid Rod Bundle CHF Data with the BWC Correlation, BAW-10143, Part 2, Babcock & Wilcox, Lynchburg, Virginia, March 1980.

10. 51-1232718-00, LOCA Evaluation of Mk-B9 Axial Blankets - Oconee 2, Babcock & Wilcox, August, 1994.

11. Program to Determine In-Reactor Performance of B&W Fuels - Cladding Creep Collapse, BAW-10084P-A, Rev. 2, Babcock & Wilcox Co., Lynchburg, Virginia, October 1978.

12. Letter from H. B. Tucker (Duke Power Company) to J. F. Stolz (NRC), April 23, 1986.

13. Bounding Analytical Assessment of NUREG-0630 Models on LOCA kW/ft Limits with use of FLECSET, BAW-1915, Babcock & Wilcox, April 1986.

14. Oconee 1 Cycle 9 - Reload Report, BAW-1841, Babcock & Wilcox, August 1984.

15. Nuclear Design Methodology Using CASMO-3/SIMULATE-3P, DPC-NE-1004A, Duke Power Company, Charlotte, North Carolina, November 1992.

16. Fuel Enrichment Upgrade Package, Duke Power Company, October 1994.

17. Letter from T. N. Wampler, B&W Fuel Company to R. R. St. Clair Duke Power Company, October 19, 1994.

18. B&W's ECCS Evaluation Model, BAW-1014P-A, Rev. 5, Babcock & Wilcox, November, 1986.

19. ECCS Analysis of B&W's 177-FA Lowered-Loop NSS, BAW-1013A, Rev. 3, Babcock & Wilcox, July 1977.