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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 75 TO FACILITY LICENSE NO. DPR-71 AND

AMENDMENT NO. 101 TO FACILITY LICENSE NO. DPR-62

CAROLINA POWER & LIGHT COMPANY

BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2

DOCKET NOS. 50-325 AND 50-324

1.C Introduction

By letters dated June 26, 2984 (Reference 1, NLS-84-219 and NLS-84-274) the Carolina Power & Light Company (the licensee) submitted proposed changes to the Technical Specifications appended to Facility Operating License Nos. DPR-71 and DPR-62 for the Brunswick Steam Electric Plant (BSEP), Units 1 and 2.

The proposed amendments would change the Technical Specifications (TSs) to permit operation of Unit 2 for Cycle 6. The changes incorporate revised minimum critical power ratio (MCPR) values, revise maximum average planar linear heat generation rate (MAPLHGR) values for the new BP8DRB299 fuel type, and inclusion of a footnote to TS 3.3.6.2, End-of-Cycle Recirculation Pump Trip (EOC-RPT) System Instrumentation to reflect the manual bypassing of the EOC-RPT system during Cycle 6 operation for Unit 2. In addition, TS 5.3.1 is being revised to reflect that reload fuel has a maximum enrichment of 2.99 weight percent U-235. The Reload Licensing Submittal incorporates the results of analyses supporting the use of full-arc admission without Recirculation Pump Trip. This reload is based on the same General Electric generic report used in the last Unit 1 reload submittal. This report has been reviewed and approved by the staff. The fuel enrichment is identical to that for Unit 1 during the last reload for which the Commission issued Amendment No. 56 on June 28, 1983 and Amendment No. 71 on June 5, 1984.

The proposed amendments would also revise Section 5.3.2 of the Technical Specifications to reflect the use of hybrid design hafnium control rod assemblies. These assemblies will be used to replace standard control rod assemblies during the current Unit 2 refueling outage and will be used as replacements during upcoming Unit 1 refuelings. The changes made to Section 5.3.2 of the Brunswick-1 and Brunswick-2 Technical Specifications reflect the use of hybrid design hafnium control rod assemblies to replace existing control rod assemblies. The Hybrid I Control Rod (HICR) Assembly has been designed by General Electric (GE) to be used as direct replacement for the present control rod assemblies. The original control rods contained only boron carbide, B_4C, as the absorbing material. The new assembly design uses B_4C absorber cubes and three solid hafnium rods in the outside edge of each wing. This new design will lengthen control rod lifetime.

In the core-related areas of fuel design and safety analyses, thermal-hydraulic design and safety analysis, nuclear design including power distributions and reactivity analyses as well as safety analyses of postulated BWR accidents and transients, the licensee has relied on the results presented in the approved GE topical report NEDE-24011, "General Electric Standard Application for Reactor Fuel", or GESTAR II (Ref. 3).

In addition, the licensee submitted a supplemental reload licensing document (Ref. 2) which provides results of analyses necessary to justify Cycle 6 operation but not included in GESTAR II.

2.0 Fuel System Design

2.1 Fresh Fuel Assemblies BP8DRB299

Fresh fuel assemblies (BP8DRB299), which are prepressurized 8x8 retrofit barrier fuel assemblies with an average enrichment of 2.99 w/o in U-235, will be loaded for Cycle 6 operation. Since (1) the prepressurized 8x8 retrofit barrier fuel has been previously approved (Ref. 3), and (2) the average enrichment of the fresh fuel is less than that of the approved maximum enrichment state in Reference 3, we conclude that fuel assemblies are acceptable for Cycle 6 operation.

2.2 Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Limits

The licensee's submittal provided MAPLHGR limits for the 8x8, 8x8R, P8x8R and BP8x8R fuel assemblies in the Cycle 6 core. Although the methodology used is generically applicable for the MAPLHGR limit determination, the staff was concerned that the effects of enhanced fission gas release at high burnup (i.e., greater than 20 MWd/kgU) were not adequately considered in the fuel performance model. In response to this concern, GE requested (Ref. 4 and 5) that credit for approved, but unapplied, ECCS evaluation model changes and calculated peak cladding temperature margin be used to avoid MAPLHGR penalties at higher burnups. This proposal was found acceptable (Ref. 6) provided that certain plant-specific conditions were met. The licensee has stated (Ref. 7) that the GE proposal is applicable to both the Brunswick Unit 1 and Unit 2 safety analysis. We have reviewed the basis for the licensee's finding and conclude that the proposed MAPLHGR limits are appropriate for Cycle 6 operation.

2.3 Use of Hybrid I Control Rods

The licensee proposed to begin replacing the standard control rod blades with the new Hybrid I control rod (HICR) blades. These blades are designed to have the same worth and weight as the existing blades. The differences in design are in the cladding and absorber material and serve to improve blade lifetime. The use of these control rods in BWRs has been reviewed and approved by the staff (Safety Evaluation letter dated August 22, 1983) and we conclude that their use is acceptable in Brunswick-2.

The details of the design and materials will not be included in the revised Technical Specifications. Since descriptions of the standard blades exist in the FSAR and of the HICR blades in approved topical report NEDE-22290-A, and the safety design criteria which control rods must meet are contained in the FSAR and in other Technical Specifications, we conclude that this is acceptable.

3.0 Nuclear Design

The nuclear design and analysis of the proposed reload has been performed by the methods described in Reference 3. Reference 3 has been approved for use in the design and analysis of reloads in BWR reactors and its use is acceptable for this reload. We have reviewed the results of the nuclear design analysis for Brunswick Unit 2 Cycle 6 and have determined that since they are consistent with those for similar reloads and are done with acceptable methods, they are acceptable.

4.0 Thermal Hydraulic Design

The objective of the review of the thermal-hydraulic design of the core for Cycle 6 operation is to confirm that the thermal-hydraulic design has been accomplished using acceptable methods, and to assure an acceptable margin of safety from conditions which could lead to fuel damage during normal operation and anticipated transients, and to assure that the core is not susceptible to thermal-hydraulic instability.

The review includes the following areas: (1) operating limit minimum critical power ratio (MCPR) and the related changes to the Technical Specifications, and (2) thermal-hydraulic stability. Discussion of the review concerning the thermal-hydraulic design for Cycle 6 operation follows:

4.1 Operating Limit MCPR and the Related Technical Specification Changes

A safety limit MCPR has been imposed to assure that 99.9 percent of the fuel rods in the core will not experience boiling transition during normal operation and anticipated operational transients. As stated in Reference 3, the approved safety limit MCPR for reload cores is 1.07. A safety limit of 1.07 was used for the Cycle 6 analyses.

To assure that the fuel cladding integrity safety limit MCPR will not be violated during any abnormal transient, the most limiting events have been reanalyzed for this reload (Ref. 2) by the licensee, in order to determine which event results in the largest reduction in the minimum critical power ratio. The operating limit MCPRs for each fuel type were then established by adding the largest reduction in the minimum critical power ratio and the uncertainties associated with the calculational methods to the safety limit MCPR.

We find that since approved methods (Ref. 3) were used and the results show an acceptable margin of safety from conditions which could lead to fuel damage during any anticipated operational transient that the thermal-hydraulic design of the Cycle 6 core is acceptable. The corresponding Technical Specification (3/4.2.3) changes are also acceptable since they are consistent with the Cycle 6 safety analysis.

4.2 Thermal-Hydraulic Stability

The results of thermal-hydraulic analyses (Ref. 2) show that the maximum core stability decay ratio is 0.67 for Cycle 6. We find that (1) the calculated decay ratio for Cycle 6 is less than that for similar reload cores (for example, the calculated decay ratio for the Unit 1, Cycle 4 core is 0.72) and (2) the Technical Specifications prohibit normal operation in the natural circulation mode in which the core would be less stable. We therefore conclude that the thermal-hydraulic stability results are acceptable for Cycle 6 operation.

5.0 Transient and Accident Analyses

The Postulated Uncontrolled Rod Withdrawal Error, Fuel Misorientation Event and Rod Drop Accident have been analyzed for this cycle. The cycle specific Rod Drop Accident analysis was necessary because certain parameters (accident reactivity shape function and scram shape function in the cold startup mode) were not bounded by the generic analysis. The results of the cycle specific analysis (220 calories per gram peak enthalpy) meets our acceptance criterion for this event and is therefore acceptable.

On the basis that approved methods have been used to perform the analyses and to obtain input parameters for them and that the results of the accident analyses are acceptable for Cycle 6, we conclude that the analyses of the three cited events are acceptable. Core-wide transient analyses are discussed in Section 4.1 above.

6.0 Technical Specification Changes

Various revisions to the Technical Specifications have been proposed. The results of our review are as follows:

Changes were made in Figures 3.2.1-1 through 3.2-1-6 of the Technical Specifications in order to specify the MAPLHGR limits. We conclude that these changes regarding the proposed MAPLHGR limits are acceptable based on the discussion in Section 2.1 of this SER.

Section 3/4.2.3 and Table 3.2.3.2-1 of the Technical Specifications have been revised to include the proposed operating limit MCPRs for Cycle 6 operation. We find that the proposed operating limit MCPRs have been established using approved methods to avoid violation of the safety limit MCPR during any anticipated operational transient. We conclude that the Technical Specification changes related to the operating limit MCPRs are acceptable based on the discussion in Section 4.1 of this SER.

In Technical Specification 3/4.2.3, the change was made in the ACTION requirements for violation of the MCPR limits. This is acceptable since the changed Technical Specification is consistent with that for the approved Unit-1 Technical Specifications.

Changes were made in Technical Specification 3/4.2.2, Table 3.3.3-2 and Bases 3/4.2.2 related to the design total peaking factors (TPF). The changes reflect the value of design TPF for Cycle 6 core and are acceptable.

Technical Specification 3/4.2.4 and Technical Specification 5.3.1 were revised to include the BP8x8R for the Cycle 6 operation and are acceptable as discussed in Section 2.2 of this report.

A note is added to Technical Specification 3.3.6.2 to indicate that during Cycle 6 operation the EOC recirculation pump trip system will be inoperable. This is acceptable since no credit is taken for this trip in the plant safety analysis.

A change was made in Technical Specification 4.2.4 which specifies the surveillance requirements, for operation within the LHGR limit. This is acceptable since the change is to make the statement consistent with that of the standard Technical Specification and is an editorial change.

7.0 Evaluation Summary

For the basis of our review which is described above, we conclude that the Brunswick-2 reactor may be operated for Cycle 6 and that both Unit 1 and Unit 2 may be operated with the new control rods without undue risk to the health and safety of the public. This conclusion is based on the fact that acceptable methods and procedures were used to perform the design and analysis of the cycle and the new control rods and that the Technical Specifications have been correctly based on the results of that analysis.

8.0 Environmental Considerations

The amendments involve changes in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes in surveillance requirements. The staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

9.0 Conclusions

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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Dated: September 22, 1984

8.0 References

1. Letters from A. Cutter (CP&L) to D. Vassallo (NRC, Request for Revision to Technical Specifications (Fuel Cycle No. 6-Reload Licensing), June 26, 1984. (NLS-84-219, NLS-84-274).

2. 23A1765, Supplemental Reload Licensing Submittal for Brunswick Steam Electric Plant Unit 2, Reload 5, May 1984.

3. NEDE-24011-P-A-6, General Electric Boiling Water Reactor Generic Reload Fuel Applications, April 1983.

4. R. E. Engel (GE) letter to T. A. Ippolito (NRC) dated May 6, 1981.

5. R. E. Engel (GE) letter to T. A. Ippolito (NRC) dated May 28, 1981.

6. L. S. Rubenstein (NRC) memorandum for T. M. Novak (NRC) on "Extension of General Electric Emergency Core Cooling System Performance Limits" dated June 25, 1981.

7. P. W. Howe (CP&L) letter to D. B. Vassallo (NRC) dated June 7, 1982.