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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 29 TO FACILITY LICENSE NO. DPR-71

CAROLINA POWER & LIGHT COMPANY

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO. 1

DOCKET NO. 50-325

A. Brunswick Steam Electric Plant, Unit No. 1, Operating Cycle No. 3 - Reload Application

By letter dated May 23, 1980, Carolina Power and Light Company (CP&L or licensee) requested revisions to the Technical Specifications to complete the second refueling of Brunswick Steam Electric Plant, Unit No. 1 (BSEP) and begin Cycle 3 operation.

The staff was assisted in the Safety Evaluation of the BSEP 1 reload licensing analysis by our technical consultant, Brookhaven National Laboratory (BNL). The following evaluation was submitted by BNL on June 19, 1980.

I. INTRODUCTION

In a recent letter[1] to the NRC Carolina Power and Light (CP&L) Company has requested revisions to the Technical Specifications for its Brunswick Steam Electric Plant (BSEP) Unit No. 1, and submitted General Electric's (GE) "Supplemental Reload Licensing Submittal for BSEP Unit 1 Reload 2".[2]

The above documents containing plant specific data, along with GE's BWR generic reload document[3] and NRC's Safety Evaluation Report[4] (SER) on the generic reload document have been reviewed. Additional bundle data describing basic nuclear characteristics[5] of one of the new bundle types used in the BSEP-1 Reload-2 core, recently submitted by GE, have also been reviewed.

This report presents a summary of our safety evaluation based on our review of the above documents.

CP&L's BSEP 1 is a BWR-4 plant. The Cycle 3 core is expected to contain 560 8 x 8 bundles including 156 fresh assemblies. These fresh assemblies are of the prepressurized retrofit type and would constitute 28% of the core.

Our evaluation of the BSEP 1 Reload 2 is limited to the items discussed in the following sections. Our acceptance of the results discussed in these sections is strictly limited to the criteria set forth by the USNRC in USNRC's own SER's referred to in this report. Brookhaven National Laboratory (BNL), acting as technical consultants to the USNRC, has not performed independent analyses to verify either the methods or the results and accuracy of the GE analyses. To establish acceptance of the results of GE's calculations, BNL has relied on NRC's SERs.

2. EVALUATION

2.1 Nuclear Characteristics

There are two types of fresh bundles planned for reload in the Brunswick 1 Cycle 3 core: 16 Reload 2 bundles designated as P8DRB265H and 140 Reload 2 bundles labelled P8DRB285. Reference 2 lists the types and numbers of the previously irradiated fuel assemblies. Figure 1 of Reference 2 shows the reference core loading pattern. We note that in near-central locations as well as near the periphery there are four-bundle control cells in which two out of the four assemblies are fresh. The beginning of cycle (BOC) cold eigenvalue with the strongest control rod fully withdrawn and all other rods fully inserted is reported to be 0.972. Technical Specifications require that adequate cold shutdown margin be demonstrated at BOC-3 with the highest worth rod withdrawn. Results shown in sections 4 and 5 indicate that both the control rod system and the standby liquid control system will have adequate shutdown margins under the most reactive conditions of the core.

2.2 Thermal Hydraulics

2.2.1 Fuel Cladding Integrity Safety Limit

The calculated safety limit MCPR of 1.07 for BWR reload cores such as Brunswick 1 Reload 2 has been found to be acceptable for the 8 x 8R (Reference 4) and P8 x 8R (Reference 5) fuels. This safety limit implies that during a transient characterized by an MCPR of 1.07, 99.9% of the fuel rods in the core are expected to avoid boiling transition.

2.2.2 Operating Limit MCPR (OLMCPR)

To insure that the fuel cladding integrity safety limit is not violated during any abnormal operational transient, the most limiting transients have been re-analyzed for Brunswick 1 Reload 2. The OLMCPR is obtained by adding to the safety limit the maximum CPR value for the most limiting transient for each fuel type. The OLMCPR values for the 8 x 8, 8 x 8R and P8 x 8R fuel types are given for the two exposure ranges in Section II of Reference 2.

2.2.2.1 Transient Analysis Methods

The methods employed for the transient calculations have been described in Reference 3. NRC approval of these methods has been documented in Reference 4. Inputs and initial condition parameters for the transient analysis calculations are given in the tables of Sections 6 and 7 of Reference 2. NRC's evaluation of the methods used to generate these reload-unique values is also included in Reference 4.

2.2.2.2 Transient Analysis Results

Transient events analyzed were the generator load rejection without bypass, feedwater controller failure, loss of 100°F feedwater heating and control rod withdrawal error. Reload-unique initial conditions and transient input parameters were assumed to be those listed in Sections 6 and 7 of Reference 2. Results of these analyses are listed in Sections 9 and 10. We have not verified independently the results of these analyses. However, the differences between these results and those of Brunswick 2 are small and consistent with the two designs. Also, as mentioned in Section 2.2.2.1 above, the generic methods employed in carrying out the calculations[3] have received approval by the NRC.[4]

2.3 Accident Analysis

2.3.1 ECCS Appendix K Analysis

In a supplement[4] to the earlier Safety Evaluation Report of GE's Licensing Topical Report of the Generic Reload Application,[3] application of the ECCS-LOCA (Appendix K) models used in the 8 x 8 retrofit reload fuel which was found to be "generically acceptable" has been extended to cover the P8 x 8R fuel. Based on that SER,[4] the proposed MAPLHGR limits for the prepressurized 8 x 8 retrofit fuel are found to be acceptable.

2.3.2 Control Rod Drop Accident

Results of the control rod drop accident analysis are shown in Figures 9 through 13 of Reference 2. These figures are intended to demonstrate that the curves plotted are appropriately covered by the bounding analysis. The latter is based on the assumption that the reactivity excursion caused by the rod drop will not result in a fuel enthalpy greater than 280 cal/gm at any axial fuel location in any fuel rod. The methods[3] used in carrying out these analyses have been approved by the NRC (Section 7.3 of Reference 4). We find these results to be acceptable.

2.3.3 Fuel Loading Error

Using the NRC approved methodology for the analysis of misoriented and misloaded bundles,[3] the GE Supplemental Reload Licensing document[2] reports that in the limiting event which results from a rotated P8 x 8R bundle, there is adequate margin to insure no loss in fuel integrity. We thus find the results of this analysis to be acceptable.

2.3.4 Overpressure Analysis

The NRC has determined that the effects of fuel prepressurization are well accounted for in vessel overpressurization analyses.[4] Accordingly, we agree that there is sufficient margin between the peak calculated vessel pressure and the design limit pressure.

2.5 Technical Specifications

The Technical Specifications have been changed to include specifications associated with the new, prepressurized type bundles as well as the corresponding surveillance requirements, regarding the Average Planar Linear Heat Generation Rates (APLHGR's), the APRM and Rod Block Monitor setpoints. These Technical Specifications changes reflecting the introduction of the new type of bundles have been reviewed and found acceptable.

2.6 Densification Power Spiking

It is acceptable to remove the 8 x 8, 8 x 8R and P8 x 8R spiking penalty factor from the Technical Specification of those BWR's for which it can be demonstrated that the predicted worst case maximum transient LHGR's, when augmented by the power spike penalty, do not violate the exposure-dependent safety limit LHGR's. The Brunswick plant meets the above criterion. Section 10, Rod Withdrawal Error and Appendix E Linear Heat Generation Rate for Bundle Loading Error, of Reference 2 include the densification effect in the reported LHGR value for all 8 x 8 type assemblies. On the basis of these data, we find that the Licensee meets the requirements on the densification power spiking.

2.7 Thermal Power Monitor

Operation of Brunswick 1 Cycle 3 with the Thermal Power Monitor (TPM) feature is acceptable provided the USNRC has already approved this option in the previous cycle for this plant. It was agreed in a recent conference call[6] that CP&L will provide the USNRC with the details of the earlier TPM approval for this plant.

2.8 Startup Plans

In the supplemental submittal[2], no mention is made of a startup test program for Brunswick 1.

We were informed[6] that CP&L plans to follow at Brunswick 1 the same startup test plans as those detailed in an earlier letter regarding the startup of the last two cycles of Brunswick 2. We received a verbal commitment from CP&L that the latter will inform the USNRC by letter on the startup test plans for the new cycle.

References

1. Letter, E. E. Utley (Carolina Power and Light Company) to T. A. Ippolito (USNRC) May 23, 1980 and Revised Technical Specifications for Brunswick Steam Electric Plant Unit 1.

2. "Supplemental Reload Licensing Submittal for Brunswick Steam Electric Plant Unit 1 Reload 2," NEDO-24239, General Electric Company, January 1980.

3. "General Electric Boiling Water Reactor Generic Reload Application," NEDE-24011-P-A.

4. Letter, T. A. Ippolito (USNRC) to R. Gridley (GE) dated April 16, 1979 and enclosed SER.

5. Letter, R. E. Engel (GE) to T. A. Ippolito (USNRC), "Specific Bundle Enrichment Nuclear Characteristics," April 30, 1980.

6. Telephone Conference Call, CP&L, GE, USNRC and BNL, June 19, 1980.

3.0 Transient Analysis Methods

In a recent Safety Evaluation* the staff concluded that the 8x8R GEXL correlation used by GE in the reload analysis for non-equilibrium cores has conservatisms which are equivalent to the 7x7 and 8x8 GEXL correlations previously approved by the staff. However, the data supporting the application of GEXL to 8x8R fuel have never been submitted for staff review in accordance with established procedures. We will require that this data base be submitted so that the staff can complete its review and that this issue be formally resolved prior to operation in future cycles.

For future cycles also, the REDY code will not be acceptable for use in calculating core response to pressurization transients. Reference NRC letter to G. G. Sherwood (GE) from Dick Denise dated January 23, 1980.

4.0 Conclusion

By letter dated June 25, 1980, CP&L confirmed that the Thermal Power Monitor feature previously approved for BSEP-1 will be used this operating cycle. By the same letter, CP&L confirmed that the startup physics test program previously approved and followed for the previous BSEP-1 cycle will be used for this operating cycle also.

Based on our review of the consultant's Safety Evaluation and the CP&L letter of June 25, 1980, we find the proposed operation in cycle 3 to be acceptable.

*Amendment No. 62 to Facility Operating License No. DPR-46 for the Cooper Nuclear Station, Dated May 20, 1980.

B. Brunswick Steam Electric Plant, Unit No. 1, Safety Relief Valve Setpoints

1.0 Introduction

By letter dated May 30, 1980, the licensee requested a temporary change in the setpoint values of 3 of the 11 BESP-1 safety-relief valves. This change was necessitated by the postponement of a major Mark I Containment modification effort until the fall of 1980. The modification involved the installation of T-quenchers in the torus to replace the existing paired discharge line design. System reserve requirements for the late summer of 1980 forced deferment of the planned Mark I Containment modification program. CP&L has previously committed to install the T-quencher modification for both Brunswick Units in the spring of 1980. (Letter dated January 30, 1980.)

2.0 Discussion

During a visual inspection of inaccessible snubbers performed in December 1979, damaged snubbers were found on the safety relief valve F013H tailpipe. It is believed the damage occurred following a reactor scram on November 20, 1979 when safety relief valves (SRV's) F013F, G, and H automatically lifted. SRV's F013F and H share one of the 5 paired discharge headers in the torus. The eleventh SRV (F013K) discharges directly into the torus through a single header. SRV's F013F and H had a setpoint differential pressure spread of 10 psi. Subsequent analysis indicated that the damage may have been caused by a water slug in the exhaust line of the paired discharge header.

The Mark I torus modifications will rearrange the SRV exhaust lines in the torus such that each valve will have a separate T-quencher. By eliminating the shared discharge headers, the likelihood for future tailpipe damage is reduced.

In lieu of the T-quencher modification, the licensee is proposing to increase the setpoint differential pressure spread for each of the paired SRV's to 20 psi. Since there have been no cases of simultaneous or near-simultaneous liftings of SRV pairs with a 20 psi setpoint differential, the licensee feels that this change will provide adequate assurance of SRV tailpipe integrity until the Mark I T-quencher modifications are installed in fall 1980.

3.0 Evaluation

To determine the adequacy of the proposed SRV setpoint change, we reviewed the staff's SER for BSEP Units 1 and 2 Supplement No. 2 dated December 23, 1974; Amendment No. 31 to DPR-62 dated October 6, 1977; and Amendment No. 14 to DPR-71 dated September 11, 1978.

The staff's SER approved an SRV setpoint pressure range from 1080 to 1100 psig. Amendment No. 31 to DPR-62 approved increasing the SRV lift setpoints for BSEP 2 by 25 psig. Amendment No. 14 to DPR-71 approved increasing the SRV lift setpoints for BSEP 1 by 25 psig.

The proposed setpoint range falls within the range of previously approved SRV setpoints, and thus is acceptable as far as overpressure protection is concerned.

We also considered the effects of the new SRV setpoints on thermal-hydraulic stability under transient conditions. We compared the previous 3-stage setpoints with the proposed 4-stage setpoints and the geometrical positioning of each valve group's discharge piping in the torus. No adverse dynamic effects from the proposed SRV setpoints were identified.

4.0 Conclusion

We find the proposed SRV setpoint change to be acceptable. We understand that upon completion of the T-quencher modifications, the SRV setpoints will be restored to their previous values. At that time it would be appropriate to delete the temporary change to the Technical Specifications.

C. Environmental Consideration

We have determined that this amendment 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 amendment 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, negative declaration, or environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

D. Conclusion

We have concluded, based on the considerations discussed above, that: (1) because this amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: July 1, 1980