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

SUPPLEMENT NO. 2 TO THE MAY 20, 1974 SAFETY EVALUATION REPORT

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

IN THE MATTER OF

CAROLINA POWER AND LIGHT COMPANY

H. B. ROBINSON STEAM ELECTRIC PLANT UNIT NO. 2

POWER INCREASE

DOCKET NO. 50-261

INTRODUCTION

By letter dated February 1, 1974, the Carolina Power and Light Company (CP&L) applied for an amendment to its Facility Operating License No. DPR-23 for the H. B. Robinson Steam Electric Plant Unit No. 2 (Robinson 2) which would allow a power increase from 2200 MWt to 2300 MWt (Reference 1). The Nuclear Regulatory Commission staff (NRC staff or staff) issued a Safety Evaluation Report (SER) of the power increase on May 20, 1974 (Reference 2). During its 170th meeting, June 6-8, 1974 the Advisory Committee on Reactor Safeguards (ACRS) reviewed the request by CP&L for a power increase and reported its findings to the NRC (Reference 3). The staff considered the ACRS comments and recommendations on the power increase and issued a supplement to the original power increase SER (Reference 4).

The purpose of this supplement is to report our evaluation of supplemental information and facts not considered in our original SER or Supplement No. 1 thereto.

BACKGROUND

Our May 20, 1974 SER evaluated reactor design and performance, containment and engineered safety features, accident analysis, conduct of operations, and plant Technical Specifications.

Supplement No. 1 to the Safety Evaluation (July 31, 1975) addressed concerns expressed by the ACRS in its June 1974 review of the proposed power increase. The items covered by that supplement were: re-evaluation of the operating limits, use of the axial power density monitoring system (APDMS), seismic shutdown requirements, turbine overspeed control system, effect of debris on operation of engineered safeguard systems, and heatup and cooldown pressure temperature limits. With the exception of the review of operating limits for which a new ECCS model was being evaluated, we concluded that the issues raised by the ACRS had been satisfactorily addressed.

Since SER Supplement No. 1 was issued, there have been changes in the facility (reloads, fuel vendor) and safety issues (fuel rod bowing, upper head temperature) that should be addressed in connection with the proposed power increase.

Starting with the reload for Cycle 4 operation, CP&L began using Exxon fuel in Robinson 2. The Exxon ECCS analysis package is based on implementation of the NRC staff's Water Reactor Evaluation Model (WREM). The generic Exxon ECCS-Evaluation Model (ECCS-EM) (Reference 5) describes the ECCS-EM as it applies to Robinson 2.

We reviewed the Exxon ECCS-EM package and concluded that it was in conformance with certain of the requirements of Appendix K to 10 CFR 50 (Reference 6). We also concluded that the codes and their application to Robinson 2 appeared to be acceptable with final approval contingent on a break spectrum calculation for Robinson 2. The break spectrum calculations were performed by Exxon and submitted in Reference 7. In all cases, both Westinghouse and Exxon performed the ECCS and other safety calculations at the 2300 MWt power level.

Cycles 4 and 5 reload submittals were documented in References 8 and 9, respectively. Appendix A to the Safety Evaluation of the Cycle 4 reload concluded that the ECCS-EM model was appropriate for use at Robinson 2 and that all issues, for which we had required Exxon to make clarifications or changes to its ECCS modeling, had been acceptably resolved (Reference 10). Our Cycle 4 Safety Evaluation also concluded that the reload core consisting of Exxon and Westinghouse fuel was acceptable and fully met the requirements of 10 CFR 50.46.

The Cycle 5 reload was considered by the licensee to involve no unreviewed safety questions and was considered to be a plant modification not requiring NRC staff prior approval as allowed under 10 CFR 50.59 (References 9 and 11). We stated that because of a change to Exxon's neutronics codes the Cycle 5 reload would have to be reviewed by the staff (Reference 12). Exxon proceeded to publish its revised neutronics methods topical report (Reference 13). We reviewed and approved Reference 14, the neutronics topical report. As a result of that separate review we concluded that the use of the revised neutronics methods in the physics calculations did not constitute an unreviewed safety question and supported the licensee's conclusion that Cycle 5 reload need not be reviewed by the staff (Reference 15).

Prior to the time of the Cycle 5 refueling it was discovered that the water temperature in the upper head region of Westinghouse designed reactors, including Robinson 2, could be significantly higher than had been assumed in the previously approved ECCS analysis and that this could increase the calculated peak clad temperature (PCT) in the event of a LOCA. Our analysis of the upper head temperature issue was presented in Reference 16. We concluded that the updated ECCS analysis corrected for this higher water temperature, was performed using models wholly conforming to 10 CFR 50.46 and Appendix K and yielded an acceptable PCT. The power level used in this new analysis was 2300 MWt.

Although all safety issues had been addressed final licensing action was not taken because the environmental aspects of continued operation (the power increase proceeding had been consolidated by the Commission with the proceeding under Section B to Appendix D, 10 CFR Part 50) was still under review. When the sole intervenor (in both the Section B and power increase aspects of the proceeding) withdrew and the Licensing Board issued a Partial Initial Decision on environmental matters, the Staff requested that CP&L address those issues which had developed in the two and one half years since the issue of Supplement 1 to the SER for impact on the proposed 2300 MWt power level (Reference 17).

CP&L replied that there were no open items (Reference 18). The ECCS reevaluation, use of Exxon fuel, rod bow, and the upper head temperature problem have been resolved as discussed below.

In addition to the items discussed by the licensee in Reference 18, we have also considered several other generic items for relevance to this action. This consideration is reported in an Appendix to this Supplement.

EVALUATION

The issues of ECCS reevaluation, use of Exxon Fuel, rod bow, and upper head temperature problem were resolved as discussed below.

ECCS

On December 27, 1974, the Atomic Energy Commission issued an Order for Modification of License implementing the requirements of 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors". The Order required the licensee to reevaluate the ECCS cooling performance and to provide Technical Specifications to implement the evaluation results. The licensee complied with this order and on October 17, 1975, the NRC issued license Amendment No. 13. At that time we evaluated the licensee's ECCS analysis and concluded it was acceptable at 2300 MWt operation.

Exxon Fuel

The use of Exxon fuel was evaluated for Cycle 4 operation. The licensee performed analyses, including departure from nucleate boiling (DNB) calculations, transient analyses, ECCS performance, safety margins, rod bow, densification, etc. at 2300 MWt. Our safety evaluation supporting Amendment No. 15, dated December 3, 1975, found the use of Exxon fuel acceptable.

Rod Bow

Our letter dated January 7, 1977, stated that, based on our review of rod bow data and the thermal margins available to offset the DNB ratio (DNBR) reduction due to fuel rod bowing, no change to the Robinson 2 Technical Specifications was required. We requested CP&L to provide a list of thermal margin credits which were used to offset any reduction in the DNBR penalty. CP&L, by letter dated February 11, 1977, stated that the only credit applied was the reduction in the margin between the DNBR calculated for the worst anticipated transient (1.68) and the minimum acceptable DNBR (1.3) as described in our January 7, 1977 letter. These transient DNBR calculations were performed using an operating power level of 2300 MWt. We have reviewed the rod bow situation at Robinson and have concluded that this matter is resolved.

Upper Head Temperature

On August 27, 1976, we issued an Order which required a reanalysis of the ECCS performance in view of probable higher water temperatures in the upper reactor vessel head than were previously assumed in the ECCS analyses. Our Safety Evaluation dated December 3, 1976 found the CP&L reanalysis performed at 2300 MWt to be acceptable.

We conclude that the additional matters related to operation at 2300 MWt have been resolved and that the conclusions of the May 20, 1974 Safety Evaluation remain unchanged.

Startup Tests

We have reviewed the licensee's startup tests which will be performed prior to startup. We find these tests acceptable.

Technical Specifications

The licensee submitted Technical Specification changes which will be required to allow operation at 2300 MWt (Reference 19). These changes were reviewed and found to be acceptable and appropriate for the requested power increase to 2300 MWt.

Accident Analyses

In 1974 accident analyses were performed for the proposed power increase to 2300 MWt. The results were reported in our Safety Evaluation dated May 20, 1974. Since several years have elapsed since that evaluation, we have reconfirmed the adequacy of those prior analyses, using current dose analysis methods, plant data, and recent site data.

We reanalyzed only those design basis accidents whose offsite consequences would be directly affected by the power level increase to 2300 MWt. These would be the steam generator tube failure, the steam line failure, the control rod ejection accident, the fuel handling accident, and the loss-of-coolant accident. The doses we estimated for these design basis accidents were all within current dose guidelines. The licensee has agreed to adopt the appropriate Technical Specification limits on reactor coolant and secondary coolant activities in the May 1974 SER. We relied upon those limits in making our estimates.

Steam Line Failure, Steam Generator Tube Failure and Control Rod Ejection Accident

The assumptions used in our analysis of the steam generator tube failure, the steam line failure, and the control rod ejection accident were based on information in the Robinson Final Safety Analysis Report, the Robinson Technical Specifications and communications between CP&L, Westinghouse and the NRC staff during the previous review of the 2300 MWt application. We conclude that the consequences of the steam generator tube failure and the steam line failure are within current dose guidelines with the agreed upon limits on coolant acitivity.

The limits are 1 microcurie dose-equivalent-Iodine-131 per gram in the reactor coolant, with a 60 microcurie per gram peak limit to accommodate iodine spiking; a limit of 0.1 microcurie dose-equivalent-Iodine-131 per gram in the secondary coolant; and a limit on gross $\beta + \gamma$ activity in the reactor coolant of $100/\bar{E}$ microcuries per gram, where $\bar{E}$ is defined as the average energy released per disintegration, in units of MeV. The results of our analysis of these accidents are given in Table 1. Our assumptions are shown in Table 2.

Fuel Handling Accident

Our analysis of the fuel handling accident with credit for iodine removal by the fuel building ventilation system charcoal filter and 100 hours' decay prior to the first fuel movement, shows that the doses for the exclusion area boundary would be well within 10 CFR 100 dose guidelines. The Robinson 2 Technical Specifications already include requirements which allow credit for iodine removal by the filters and the fuel decay time assumed above. The results of our analysis of this accident are given in Table 1 and are equally applicable to a fuel handling accident inside the containment. Our assumptions are shown in Table 3.

Loss-of-Coolant Accident

We also reanalyzed the potential doses from a Loss-of-Coolant Accident (LOCA) at 2300 MWt. Our analyses of iodine removal credit and meteorology in the determination of LOCA doses have changed from those used in our 1974 SER.

Using current models, we reanalyzed the containment spray parameters and mixing of the containment atmosphere and found that iodine removal credit greater than previously given the Robinson 2 containment spray system by the staff is appropriate.

We also reanalyzed the meteorological parameters for the Robinson site. In our previous analysis, we had used the available onsite meteorological data (April 1967 - April 1969) and the direction-independent accident dispersion model (Regulatory Guide 1.4) to estimate relative atmospheric concentrations (χ/Q). In our reanalysis, we used more recent onsite data (January 1975 - December 1977) obtained from an upgraded meteorological measurements program. We also used a new, direction-dependent meteorology model for χ/Q estimation. The new model more accurately reflects actual site meteorology as discussed below. For comparison, we also performed calculations based on the R.G. 1.4 model.

This direction-dependent model produces χ/Q values which are expected to be exceeded in the "worst" direction sector no more than 0.5% of the time. Using this model and the new Robinson data, the calculated LOCA doses are within 10 CFR Part 100 guidelines at the increased power level of 2300 MWt. The cumulative probability of the sector χ/Q's being exceeded in all directions using this model is less than 5%.

The Regulatory Guide 1.4 meteorology model produces χ/Q values which are expected to be exceeded no more than 5% of the time in all directions at a distance equal to the shortest site boundary distance or at the LPZ distance. We performed comparison calculations using this model and the new data. The resulting calculated exclusion area boundary thyroid dose would exceed somewhat the 10 CFR Part 100 guidelines. We did another calculation which shows that, if a site χ/Q probability level of 8% (instead of 5% as above) were used, this model would result in doses within the guidelines.

The direction-dependent model includes consideration of plume meander, directionally variable site boundary distances, and directionally variable dispersion conditions, while the Regulatory Guide 1.4 model assumes no meander, a circular boundary with radius equal to the shortest exclusion area boundary distance, and directionally uniform dispersion conditions. The direction-dependent model is particularly appropriate at the Robinson site where the poorest atmospheric dispersion conditions occur when the wind is blowing in the direction of the shortest exclusion area boundary distances. We have used the direction-dependent model for our reanalysis.

In the FSAR, the exclusion area boundary is defined as a circle with radius 430 meters. Using the direction-dependent meteorology model, 10 CFR Part 100 dose limits are not exceeded anywhere (in any direction) outside the 430 meter circle. Therefore, there is no need to redefine the exclusion area boundary.

With these changes from the staff's previous analyses and including the dose contributed by leakage from ECCS recirculation components outside containment, we estimated that the potential exposures at the exclusion area and low population zone boundaries would be less than 10 CFR 100 dose guidelines. The results of our analysis for the LOCA are given in Table 1 and our assumptions, which included operation at 2300 MWt, are shown in Table 3.

<u>Analysis of Control Room Habitability During Postulated Accidents</u>

We found during our review that neither the licensee nor the staff had previously documented a review of the habitability of the Robinson control room during postulated accidents. Using data obtained from the licensee, Table 4, and plant design data from the Robinson 2 Final Safety Analysis Report, we estimated the potential doses to control room operators from a design basis LOCA with the methods of U. S. NRC Standard Review Plan 6.4, "Habitability Systems". We calculated that, with a 90% iodine removal efficiency assumed for the recirculation charcoal filter in the control room ventilation system, the potential accident doses the operators would receive would be approximately 30 rem to the thyroid and approximately 1 rem to the whole body.

The acceptance criteria as given in General Design Criterion No. 19 and as amplified in SRP 6.4 for the new plants are 30 rem thyroid and 5 rem whole body. We, therefore, conclude that the Robinson control room design provides the operators adequate protection from radioactivity releases which might result from design basis accidents at core power levels up to 2300 MWt. The assumptions made in our analysis of the Robinson control room operator doses are given in Table 5. The results are detailed in Table 6.

The current Technical Specifications in the Robinson 2 license do not include operability and surveillance requirements necessary to support our assumption about automatic control room isolation on a safety injection signal or high radiation signal or about the iodine removal efficiency of the charcoal filter used in the recirculation mode of the control room ventilation system. The licensee has proposed Technical Specifications on these two existing systems to assure the assumptions for our control room dose analysis are appropriate. From our analysis, we find the radiation protection provisions of the Robinson control room acceptable.

SUMMARY

In summary, since issuance of the May 20, 1974 Safety Evaluation Report, we have evaluated additional issues which could impact the power increase. These issues have been resolved. We have also reanalyzed the radiological consequences of design basis accidents affected by the increase in power from 2200 MWt to 2300 MWt and estimated the potential offsite doses from these accidents. The estimated consequences are all appropriately within 10 CFR 100 dose guidelines. We have also addressed additional items which we believe should be brought to the attention of the Atomic Safety and Licensing Board considering this matter. We do not believe any of these additional items require resolution before acting on the proposed power increase.

Date:

Table 1

ESTIMATED OFFSITE DOSES FROM POSTULATED ACCIDENTS

AT H. B. ROBINSON UNIT 2

Accident	Offsite Doses (rem)			
	0-2 Hour EAB		LPZ	
	Thyroid	Whole Body	Thyroid	Whole Body
1. Steam Generator Tube Failure with Loss of Offsite Power				
a. With consequent iodine spike	13	< 1	1	<0.1
b. With prior iodine spike	87	< 1	4	<0.1
2. Steam Line Failure with Loss of Offsite Power				
a. With consequent iodine spike	1.7	<0.1	<1	<0.1
b. With prior iodine spike	3.6	<0.1	<1	<0.1
3. Control Rod Ejection Accident with Loss of Offsite Power				
a. With all releases through the steam generators	2.6	<0.1	0.2	<0.1
b. With all releases to containment	1.1	<0.1	0.3	<0.1
4. Fuel Handling Accident	58	1	2	<0.1
5. Loss-of-Coolant Accident (Total Dose)	283	7	56	<1
a. Containment leakage contribution	267	7	21	<1
b. ECCS leakage contribution	16	<0.1	4	<0.1
c. Purge contribution	0	0	31	<0.1

Table 2

ASSUMPTIONS USED FOR ANALYSIS OF H. B. ROBINSON, UNIT 2

SECONDARY SYSTEM ACCIDENTS

A. Steam Generator Tube Failure

1. U. S. NRC Standard Review Plan 15.6.3, "Radiological Consequences of a Steam Generator Tube Failure (PWR)," NUREG-75/087, November 24, 1975.

2. Core power level = 2300 MWt.

3. Normal reactor coolant activity = 1 microcurie dose equivalent I-131 per gram and $100/\bar{E}$ microcuries gross $\beta + \gamma$ activity per gram.

4. Maximum initial reactor coolant activity during accidents assumed to occur while an iodine spike is in progress (i.e., the "prior iodine spike" case) = 60 μCi/g dose equivalent I-131.

5. Normal secondary coolant iodine activity = 0.1 μCi/g dose equivalent I-131.

6. Normal steam generator reactor coolant leak rate = 1 gallon per minute.

7. Initial mass of reactor coolant = 422,800 lb.

8. Mass of coolant leaked through failed tube = 70,000 lb.

9. Time for reactor coolant and secondary system pressures and temperatures to equilibrate = 30 minutes.

10. Iodine Source spiking factor = 500 times equilibrium iodine release rate.

11. Iodine partition factor between steam and water = 10.

12. Loss of offsite power follows the tube failure, so that plant cooldown is accomplished through steam releases through secondary system relief valves.

13. 0 - 2 hour $\chi/Q = 7.4 \times 10^{-4}$ sec/m^3 at the exclusion area boundary of 430 meters from Figure 1, Regulatory Guide 1.5, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors." 0-8 hours χ/Q at the outer boundary of the low population zone = 3.6×10^{-5} sec/m^3.

TABLE 2 (continued)

B. Steam Line Failure

14. Appendix to U. S. NRC Standard Review Plan 15.1.5, "Radiological Consequences of Main Steam Line Failures Outside Containment (PWR)."

15. Steam generator secondary side volume = 4729 cubic feet.

16. Duration of steam generator 1 gpm reactor coolant leak = 30 minutes.

17. χ/Q as in A.13 above.

C. Rod Ejection Accident

18. Appendix to U. S. NRC Standard Review Plan 15.4.8, "Radiological Consequences of Control Rod Ejection Accident (PWR)."

19. Fuel pin centerline melting in 10% of the fuel in one fuel assembly.

20. For analysis of the doses from releases to containment, no containment spray actuation was assumed and χ/Q's as given in Table 3, B.8.

21. Containment leak rate = 0.1%/day.

22. Containment free volume = 2.1 million cubic feet.

23. For analysis of the doses from releases through the steam generators, a steam generator reactor coolant leak rate of 1 gallon per minute is assumed to continue for two hours after the accident. χ/Q's as in A.13.

TABLE 3

ASSUMPTIONS USED FOR ANALYSIS OF THE H. B. ROBINSON, UNIT 2

FUEL HANDLING ACCIDENT AND LOSS-OF-COOLANT ACCIDENT

A. Fuel Handling Accident

1. Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors."

2. Fuel handling begins 100 hours after reactor shutdown.

3. Refueling building filters remove 90% of the elemental iodine, 70% of the organic iodine before exhausting to the environment.

4. 0 - 2 hour EAB $\chi/Q = 1.7 \times 10^{-3}$ sec/m^3. 0-8 hour LPZ $\chi/Q = 3.6 \times 10^{-5}$ sec/m^3.

B. Loss-of-Coolant Accident

1. U. S. NRC Standard Review Plan 15.6.5, Appendix A, "Radiological Consequences of a Design Basis Loss-of-Coolant Accident: Containment Leakage Contribution," and Appendix B, for "Leakage from Engineered Safety Features Components Outside Containment."

2. Containment free volume = 2.1×10^6 cubic feet.

3. Containment leak rate = 0.1% per day (0 - 24 hrs), 0.05% per day (after 24 hrs).

4. Containment spray iodine removal constant = 10 hr^{-1} for elemental iodine and 0.3 hr^{-1} for particulate iodine in a sprayed volume of 1.7×10^6 cubic feet.

5. Unsprayed containment volume communicating with the sprayed region at an air exchange rate of 10 volumes per hour = 2.64×10^5 cubic feet.

6. Unsprayed containment volume communicating poorly with the other unsprayed region at an air exchange rate of 1.65 volumes per hour and not in communication with the sprayed region = 1.28×10^5 cubic feet.

TABLE 3 (Continued)

B. Loss-of-Coolant Accident (Continued)

7. Leakage to the environment is possible only from the sprayed region and the poorly-communicating unsprayed region.

8. The 0 - 2 hour EAB $\chi/Q = 1.7 \times 10^{-3}$ sec/m^3; 0 - 8 hour LPZ $\chi/Q = 3.6 \times 10^{-5}$ sec/m^3; 8 - 24 hour LPZ $\chi/Q = 2.3 \times 10^{-5}$ sec/m^3; 24 - 96 hour LPZ $\chi/Q = 9.0 \times 10^{-6}$ sec/m^3; 96 - 720 hour LPZ $\chi/Q = 2.3 \times 10^{-6}$ sec/m^3.

9. Maximum elemental iodine reduction in the containment = 100.

10. ECCS recirculation system leakage assumed to begin 20 minutes after LOCA.

11. ECCS recirculation system leak rate = 4 gallons per hour (twice the technical specification limit of 2 gallons per hour).

12. Recirculation fluid volume = 200,000 gal (0 - 60 min); = 300,000 gal (>60 min).

13. Twenty cfm continuous containment purge initiated at 23 days after the LOCA.

Table 4

Information Requested and Obtained from CP&L on the Robinson Control Room

1. Where is the control room air intake? How high off the ground and how far away from the containment?

   CP&L Response: 25 feet above ground, 115 feet from containment, on wall of auxiliary building away from containment.

2. What signals automatically isolate the control room? Mechanically, how is the control room isolated? What must the operator do to manually isolate the control room?

   CP&L Response: Automatic isolation of the control room air intake will occur following a safety injection signal or a high radiation signal from the monitor in the inlet line to the control room HVAC. The control room is isolated by dampers in the inlet line, which can be closed manually from the main control board.

3. The 1967 Safety Evaluation Report indicates there are charcoal filters in the control room HVAC. What is the depth of the charcoal? What is the flow rate through the filter in the emergency mode of operation? What is the general layout of the control room HVAC?

   CP&L Response: The charcoal is 2 inches deep and preceded by a HEPA filter. The flow rate through the charcoal is 5000 cfm during the emergency mode of operation. Normally the control room HVAC recirculates 4400 cfm and draws in 600 cfm outside air.

Table 4 (Continued)

which is mixed with the return air from the control room. 600 cfm is exhausted from the control room through a vent in one corner of the control room. On control room isolation signals or manual operator action the intake and exhaust close.

4. What is the volume of the control room? What other areas share the control room HVAC? What is the total volume of the rooms served by the control room HVAC?

CP&L Response: The control room volume and the total volume covered by the control room HVAC is 15,500 cubic feet (dimensions 41 feet x 42 feet x 9 feet).

Table 5

Assumptions Made in NRC Staff Estimate
of Control Room Operator Doses in a Design Basis
LOCA at 2300 MWt at Robinson 2

1. Dose Analysis methods of references 20 and 21

2. Containment projected area = 1377 $meters^2$

3. Distance from containment to control room = 115 feet.

4. X/Q at control room air intake at wind speed of 1 meter/sec = 4.0×10^{-3} sec/m^3

5. Control Room X/Q Adjustments:

Time Interval	Occupancy	Wind Speed	Wind Direction	Adjusted X/Q (sec/m^3)
0 - 8 hrs	1	1	1	4×10^{-3}
8 - 24 hrs	1	.67	.88	2.4×10^{-3}
24 - 96 hrs	.6	.5	.75	9.3×10^{-4}
96 - 720 hrs	.4	.37	.5	2.7×10^{-4}

6. Control room charcoal filter iodine removal efficiency = 90% for all forms

7. Filtered recirculation flow rate = 5000 cfm

8. Unfiltered air inleakage rate = 20 cfm

9. Iodine protection factor (dimensionless) = 226

10. Control room volume = 15,500 ft^3 (dimensions: 41 x 42 x 9 feet)

11. Finite cloud whole body dose reduction factor = 45

Table 6

Estimated Doses to Robinson 2
Control Room Operators for the Course of
the Design Basis LOCA at 2300 MWt

	Total Dose, rem	
	Thyroid	Whole Body
Loss of Coolant Accident	31	1
a. Containment Leakage contribution	13	< 1
b. ECCS leakage contribution	2	< 0.1
c. Purge contribution	16	< 0.1

References

1. Letter, Jones (CP&L) to O'Leary (DL), NG-74-144, dated February 1, 1974.

2. United States Atomic Energy Commission Safety Evaluation by the Directorate of Licensing, Docket No. 50-261, Carolina Power & Light Company, H. B. Robinson Steam Electric Unit No. 2 Power Increase, dated May 20, 1974.

3. Letter, Stratton (ACRS) to Ray (NRC), "Report on H. B. Robinson Unit No. 2," June 11, 1974.

4. Supplement No. 1 to the Safety Evaluation Report by the Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission, in the matter of Carolina Power & Light Company, H. B. Robinson Steam Electric Plant Unit No. 2 Power Increase, Docket No. 50-261, July 31, 1975.

5. XN-74-41: Volume I ENC-WREM - Based Generic PWR-ECCS Evaluation Model, July 25, 1975

   Volume II Model Justification, August 1, 1975

   Volume III Revision 2 Small Break Model, August 20, 1975

   Volume II, Appendix 3-Loop Westinghouse Sample Problem, August 1, 1975

   Supplement 1: Further Definitions and Justification to Reflood Heat Transfer Models, August 14, 1975.

   Supplement 2: Supplementary Information Related to Blowdown Analysis and Back Pressure During Reflood, August 14, 1975.

   Supplement 3: Supplementary Information Related to Blowdown and Heatup Analysis, August 20, 1975.

   Supplement 4: Supplementary Information Related to Blowdown and Heatup Analysis, August 20, 1975.

Supplement 5: XN-75-41/43

6. Safety Evaluation Report Regarding Review of the Exxon Nuclear Company Pressurized Water Reactor Generic ECCS Codes and the H. B. Robinson Reactor ECCS Evaluation Model For Conformance to All Requirements of Appendix K to 10 CFR 50 by the Office of Nuclear Reactor Regulation, September 11, 1975.

7. XN-75-57, Rev. 1, "H. B. Robinson Unit No. 2 LOCA Analysis Using the ENC WREM Based PWR ECCS Evaluation Model (September 26, 1975 Version)," November 9, 1975.

8. XN-75-38, "H. B. Robinson Unit 2 Cycle 4 Reload Fuel Licensing Data Submittal," August, 1975.

9. XN-76-39, "H. B. Robinson Unit 2 Cycle 5 Reload Safety Analysis Report," August 25, 1976.

10. Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 15 to License No. DPR-23, Change No. 40 to Technical Specifications, Carolina Power and Light Company, H. B. Robinson Steam Electric Plant Unit No. 2, Docket No. 50-261, forwarded by letter, Reid (NRC) to Jones (CP&L), Docket No. 50-261, December 5, 1975.

11. Letter, Utley (CP&L) to Reid (NRC), NG-76-1199, Docket No. 50-261, September 2, 1976.

12. Letter, Reid (NRC) to Jones (CP&L), Docket No. 50-261, November 9, 1976.

13. XN-75-27, Supplement 1, Exxon Nuclear Neutronic Design Methods for Pressurized Water Reactors, April 1977.

14. Letter, Lear (NRC) to Nechodom (Exxon), March 31, 1977.

15. Letter, Reid (NRC) to Jones (CP&L), Docket No. 50-261, December 3, 1976.

16. Letter, Reid (NRC) to Jones (CP&L), Docket No. 50-261, December 3, 1976, forwarding Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 25 to License No. DPR-23, Carolina Power & Light Company, H. B. Robinson Steam Electric Plant Unit No. 2.

17. Letter, Reid (NRC) to Jones (CP&L), Docket No. 50-261, January 24, 1978.

18. Letter, Utley (CP&L) to Case (NRC), NG-3514(R), Docket No. 50-261, February 6, 1978.

19. Letter, Utley (CP&L) to Reid (NRC), NG-77-1481, Docket No. 50-261, December 29, 1977.

20. USNRC Standard Review Plan 6.4, "Habitability Systems."

21. Murphy, K. G. and Dr. K. M. Campe, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19, " Proceedings of the Thirteenth AEC Air Cleaning Conference, CONF-740807, Melvin W. First, Editor, March 1975.

APPENDIX

ASLB BOARD NOTIFICATION ITEMS

INTRODUCTION

In addition to those items discussed in the body of Supplement 2 to the May 20, 1974 SER, several items have been identified which may warrant the attention of the Atomic Safety Licensing Board. These are discussed below:

DISCUSSION

Accumulator Gas Model

Actual accumulators may deliver ECCS water to the reactor coolant system faster than is predicted by some computer programs used to predict ECCS performance. This could mean that sufficient accumulator water would not be available at the time it is needed. Attention was focused on this problem when comparisons of accumulator delivery calculations were made between RELAP4 (NRC) and SATAN VI (Westinghouse) as part of the Upper Head Injection (UHI) review. Comparisons to the LOFT experimental data indicated that the Westinghouse model might be underpredicting accumulator delivery flow water. The key factors influencing delivery rates are the gas expansion model and the effective delivery line resistance. However, the EXXON model does not underpredict accumulator delivery and this issue is not a problem for cases using the Exxon Nuclear Company model.

Since Robinson 2 now has all Exxon fuel, we conclude that the proposed power increase would not be affected by this item.

Semiscale Experiment

Semiscale experiment S-07-6 was run on September 12, 1978. It was the first integral systems test for the MOD-3 system which utilized an externally piped downcomer and 12 foot heated core. It was intended to model an integral blowdown-refill-reflood scenario for a double-ended cold-leg pipe break. On September 21, 1978 the NRC staff was briefed on the test results.

Some of the results were unanticipated. For example, the heated core simulator was projected (by Semiscale) to quench at 110 seconds. Instead it dried out again and went through several cycles of dryout and rewet. Other portions of the cladding temperature transient showed similar discrepancies wherein test temperatures were somewhat below predicted. During the test the downcomer voided several times in the time span 100-400 seconds. This was not predicted. During these periods of downcomer voiding negative (downward) flow was observed from the heated core to the lower plenum.

Experiment S-07-6 yielded results relative to downcomer voiding after downcomer fill and successive dryout and rewet of the core over the extended reflood cycle which typical Appendix K evaluation model calculations do not show.

While the cause of these hydraulic oscillations has been established as downcomer heat transfer-induced, the typicality of these oscillations to large scale PWRs has not been esablished. Preliminary evaluations by the NRC staff indicate that the occurrence of oscillations in large scale PWRs under certain conditions cannot be ruled out at this time, however, the role of neither the small scale of the test apparatus nor the one-dimensionality of the semiscale configuration are clearly understood as contributors to the occurrence of hydraulic oscillations. The present judgment is that experimental atypicalities, in particular the stored energy in the downcomer and the one-dimensionality of the apparatus, have produced the atypical and unanticipated results.

Additional semiscale tests are planned and the atypicality for large PWRs is currently being assessed by the staff. Until this issue is resolved, we do not propose to reopen the question of PWR vendor evaluation model adequacy or approvals. It is the staff view that semiscale test S-07-6 does not place in a new or different light the adequacy of PWR bottom-flooding ECCS performance. If the staff determines at the completion of its assessment that the hydraulic oscillations are a new phenomena, it will address its position regarding ECCS model approvals. Until such time, the staff believes that the results of semiscale test S-07-6 do not significantly affect Robinson Unit 2 and that the power increase should not be withheld.

Containment Electrical Penetrations

Inspection and Enforcement Bulletin 77-06 was transmitted to licensees of operating plants on November 2, 1977, requesting information on the prevalence of GE penetration assemblies and measures which have been taken to ensure that containment electrical penetrations perform their design function in the event of a LOCA.

By letter dated December 5, 1977, the licensee stated that Robinson 2 penetrations are not similar to the GE type. Based on this response and discussions with the licensee, we consider this matter resolved for Robinson.

Containment Electrical Connectors

Inspection and Enforcement Bulletins 77-05 and 77-05A requested licensees to determine if containment electrical connectors are used in safety systems required to function in a LOCA environment inside containment.

By letter dated December 7, 1977, the licensee stated that Robinson 2 does not use connectors in the containment electrical penetrations associated with safety related equipment. We consider this item resolved for Robinson.

Aging of Cable

Sandia Laboratories has advised us that a specific type of polyethylene electrical cable, i.e., installed at the Savannah River Plant, will degrade under certain extreme combinations of temperature and radiation at a rate sufficient to warrant examination and possible replacement after a period of ten to fifteen years. The damage is evident by excessive embrittlement of the cable. The electrical properties of the cable do not appear to be significantly affected; however, it is possible that the embrittled cable may fail during a LOCA because cracking of the insulation could result in subsequent loss of insulation resistance when subjected to the conductive coolant spray. The type of cable cited has probably not been manufactured and sold during the last several years. Embrittlement seems to be well understood by cable manufacturers in a qualitative way, but there is neither sufficient understanding of embrittlement of polyethylene cable nor data available to assess the degree of degradation by simply knowing the cable purchase specifications.

To determine the cause of this problem, i.e., embrittlement, naturally aged polyethylene cable was obtained by Sandia from a Savannah River reactor in February 1977. This cable was in a degraded condition and it was then determined by analysis that the degree of degradation was dependent on the radiation and thermal environment to which the cable was exposed. Recent Sandia tests have determined that cable damage, as evident by the loss of elongation (embrittlement), occurs with increased exposure to the combined effects of radiation and thermal environments and that these elongation measurements give a quantitative measure of the brittleness of the cable insulation. The insulation system degradation is the result of a reaction between the copper and polyethylene which occurs when the antioxidents used in the cable insulation materials are depleted with age. The problem is further complicated by the use of a myriad of pigment and other additives used by different suppliers and by secondary chemical reactions with the polyvinyl chloride (PVC) which is usually used as a cable jacket material.

An Inspection and Enforcement circular is being recommended to inform licensees of this problem. We consider this to be sufficient action at this time. This finding is based primarily on the unusually severe environment, relative to nuclear power plants, to which the Savannah River cabling was exposed and the extended period of time (10-15 years) that the cables had been in this severe service.

A preliminary investigation by the licensee did not identify any cables of the type of concern. Therefore, we have no reason to believe that a problem exists at Robinson at this time.

We conclude that the power increase should not be impacted by this ongoing review.

CP&L Management Capabilities

On September 6, 1978, the Commission ordered that the proceeding on the Shearon-Harris Nuclear Power Plant be remanded to the ASLB for further hearing on the management capabilities of CP&L to construct and to operate Shearon Harris.

The NRC staff testimony in this case supports the position that the management performance of CP&L with respect to its presently operating reactor facilities at Robinson 2 and Brunswick 1 & 2 is acceptable.

Neutron Dosimetry

Regulatory Guide 8.14 "Personnel Neutron Dosimeters" indicates that licensees should supply personnel monitoring equipment to those employees whose exposure to neutrons is likely to exceed 300 mrem in a quarter. The Guide provides criteria for acceptable devices and techniques for neutron personnel monitoring. NTA film, a neutron dosimeter used throughout the nuclear industry, is not sensitive to neutrons below about 0.7 MEV. Therefore, depending upon the spectrum, the dose equivalent can be grossly underestimated. On the other hand, albedo dosimeters, which are not quite as widely used as NTA among power reactor licensees, are quite sensitive to low energy neutrons and can overestimate the dose equivalent by factors of 20 to 50 (again depending on the neutron spectrum and calibration technique). Since most licensees do not routinely measure the neutron spectral distribution at their facilities, the devices worn by the workers, although acceptable by R.G. 8.14, may be providing inaccurate dose estimates.

Accurate measurement of the neutron spectrum requires specialized nuclear instrumentation and methods generally not available to the licensee, except through consultants. Therefore, few attempts have been made by licensees to determine spectral distribution. A few pressurized water reactors have had neutron streaming problems inside containment and are installing additional neutron shielding. This problem is generic, and considerable staff time has been devoted to its resolution.

Based on neutron surveys made in the Robinson containment to date, there is no indication of a neutron streaming problem. Further, the small increase in power represented by the proposed actions (from 2200 to 2300 MWt) would not significantly affect either neutron streaming or the selection of neutron monitoring methods used to protect workers.

Asymmetric LOCA Loads

On May 7, 1975, the NRC was informed by Virginia Electric & Power Company that an asymmetric loading on the reactor vessel supports resulting from a postulated reactor coolant pipe rupture at a specific location (e.g., the vessel nozzle) had not been considered by Westinghouse or Stone and Webster in the original design of the reactor vessel support system for North Anna, Units 1 and 2. In the event of a postulated LOCA at the vessel nozzle, asymmetric LOCA loading could result from forces induced on the reactor internals by transient differential pressures across the core barrel and by forces on the vessel due to transient differential pressures in the reactor cavity. With the advent of more sophisticated computer codes and the accompanying more detailed analytical models, it became apparent to Westinghouse that such differential pressures, although of short duration, could place a significant load on the reactor vessel supports, thereby affecting their integrity. Although first identified at the North Anna facility, this concern has generic implications for all PWRs.

We have concluded that there is reasonable assurance that continued operation, including the power increase, does not constitute an undue risk to public health and safety because the likelihood of an initiating event of sufficient magnitude to seriously challenge the integrity of the vessel support members is very low.

The NSSS Vendor (Westinghouse) has submitted a topical report which describes analytical methods for assessing these loads. These methods have been approved by the staff. The staff's goal is to complete this review by December 30, 1979.

Cavity Seal Ring

During the course of responding to the staff's review of an application for license amendment on another facility, that licensee informed the NRC that the reactor cavity annulus seal ring (used as a water seal during refueling operations, and not removed during normal operations) and associated biological shielding over the reactor vessel cavity could become missiles in the event of a loss of coolant accident (LOCA) pipe break inside the reactor vessel cavity. On February 2, 1978, each PWR licensee was requested to provide its status of the cavity seal ring. CP&L notified NRC on February 22, 1978, that the seal ring will be removed before reactor startup from the then current refueling outage. Since the seal ring will no longer be in the reactor cavity during operation, this item is resolved for Robinson 2.

Fire Test

On September 15, 1978, a fire test of a full-scale vertical cable tray array was conducted at the Underwiters Laboratory near Chicago, Illinois. It was part of the NRC-expedited fire protection research program requested in the Commision's Order of April 13, 1978. The purpose of the test was to demonstrate the effectiveness of area sprinklers and mineral wool blanket type cable tray fire barriers in preventing damage to cables as a result of an exposure fire created by igniting two gallons of heptane. The test resulted in damage to some of the electrical cables.

Our Fire Protection Safety Evaluation (FPSE) for H. B. Robinson Unit 2 was issued on February 28, 1978 with Amendment No. 31 to the facility operating license. At that time our review of the fire protection features at Robinson was substantially completed. We are continuing to review the design details of certain modifications, and items for which information from the licensee was not available when we issued the FPSE. Based on our review we find that the configurations of cables and fire protection features and the method of actuating these features in the September 15, 1978 test were not similar to any existing or proposed configuration at Robinson 2. Therefore, the results of these tests are not applicable to this facility.

Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts

While performing inservice inspections during a March-April 1978 refueling outage at Millstone Unit 1, structural failures of piping supports for safety equipment were observed by the Millstone licensee. Subsequent licensee inspections of undamaged supports showed a large percentage of the concrete anchor bolts were not tightened properly.

Deficiency reports, in accordance with 10 CFR 50.55(e), filed by Long Island Lighting Company on Shoreham Unit 1, indicate that design of base plates using rigid plate assumptions has resulted in underestimation of loads on some anchor bolts. Initial investigation indicated that nearly fifty percent of the base plates could not be assumed to behave as rigid plates. In addition, licensee inspection of anchor bolt installations at Shoreham has shown over fifty percent of the bolt installations to be deficient.

Vendor Inspection Audits by NRC at Architect Engineering firms have shown a wide range of design practices, and installation procedures have been employed for the use of concrete expansion anchors. The current trends in the industry are toward more rigorous controls and verification of the installation of the bolts.

The data available on dynamic testing of the concrete expansion anchors show fatigue failures can occur at loads substantially below the bolt static capacities due to material imperfections or notch type stress risers. The data also show low cycle dynamic failures at loads below the bolt static capacities due to joint slippage.

The problem is currently under active review and an Inspection and Enforcement Bulletin No. 79-02 was recently issued which requested licensees to verify that certain requirements are met and provide documentation. If the requirements are not met, the licensees are to provide plans and schedules for resolution.

With respect to Robinson 2 there have been no observed failures or damage of pipe support base plates using concrete expansion anchor bolts of the type found at Millstone Unit 1 and at Shoreham Unit 1. Therefore, continued operation is deemed warranted pending completion of the verification review called for in the above bulletin.

Post LOCA Hydrogen Generation

Following a LOCA in a light water reactor nuclear power plant, hydrogen may accumulate inside the primary containment as a result of (1) metal-water reaction involving the fuel rod cladding; (2) radiolytic decomposition of the water in the reactor core and containment sump; (3) corrosion of materials inside the primary containment, such as aluminum and zinc (in the form of galvanized steel and metal-rich paints); and (4) thermal, chemical, and radiolytic decomposition of organic components of protective coating systems.

In the past, we have generally not considered either the corrosion of materials such as zinc, or the decomposition of organic materials as significant sources of hydrogen generation inside containment following a LOCA. Currently, there is no basis for a safety concern on this matter at Robinson 2. The staff will, on a generic basis, further assess the behavior of zinc and organic materials in a post-LOCA environment. However, since these materials are not believed to be major contributors of post LOCA hydrogen generation, the effect of less than a 5% power increase on such generation is not likely to be significant.

Potential Deficiency in Refueling Water Storage Tank (RWST)

A recent design deficiency report submitted for the Seabrook Station identified an inadequacy in the RWST capacity. The deficiency is related to the remaining capacity in the RWST following transfer from the injection to the recirculation mode, and the required operator action time needed relative to the remaining tank capacity. The current Standard Review Plan for ECCS does not specify requirements for particular sizing allowances in the RWST beyond stating that adequate volume should exist.

The results of the Seabrook design deficiency report review indicated that sufficient design capacity may not have been provided in the RWST to perform the required operator actions to realign the safety injection and charging pumps prior to running out of water in the RWST. Since this design deficiency may have generic implications which could result in the loss of these pumps (due to cavitation), which may be needed to mitigate the consequences of a LOCA following switchover, a supplemental design review will be conducted to determine whether procedural or design changes are needed to assure adequate RWST capacity and pump protection as was done on Seabrook.

An Inspection and Enforcement Circular is being sent to operating reactor licensees citing the Seabrook report and requesting that design information and present operating procedures be reviewed to assess any potential deficiency in the design or procedures.

We do not believe that the proposed power increase needs to be withheld pending completion of this review for three reasons: (1) There is no basis for assumming that Robinson in fact needs to make any changes in design or procedures, (2) there is little likelihood of a LOCA occuring at Robinson in the time required to complete the review and (3) the less than 5% power increase would not significantly affect the consequences of a LOCA.

The Potential for Stress Corrosion Cracking in PWRs

The potential for stress-corrosion cracking in PWR reactor coolant system piping is extremely low because the ingredients that produce Inter-granular Stress Corrosion Cracking (IGSCC) are not all present. The use of hydrazine additives and a hydrogen overpressure keep the oxygen in the coolant at very low levels. Other impurities that might cause stress-corrosion cracking, such as halides or caustics, are also rigidly controlled. Only for brief periods during reactor shutdown when the coolant is exposed to the air and during the subsequent startup are conditions even marginally capable of producing stress-corrosion cracking in the reactor coolant systems of PWRs. Operating experience in PWRs supports this determination. To date, no stress-corrosion cracking has been reported in the reactor coolant piping or safe ends of any PWR.

However, PWRs are not completely immune to stress-corrosion cracking in other piping systems. Several cases, both transgranular and intergranular, have been reported in low-pressure piping not part of the reactor coolant system. Residual heat removal, safety injection and containment spray piping, as well as various cross lines between systems and line connections to refueling water storage tanks have been affected.

These incidences of stress-corrosion cracking have generally occurred in the heat-affected zones of welds in austenitic stainless-steel pipe, but they have also been reported in base material that was sensitized. In these cases the corrodent has not always been clearly established.

However, some cracking appeared to be associated with the concentration of chloride contaminants or chemical additives in stagnant solutions of boric acid, generally occurring in piping that was not adequately vented. The oxygen level in these solutions would be expected to be relatively high. Those stresses that contributed most to the IGSCC probably resulted from welding or fabrication.

Cracking in operating PWR plants has not been widespread. NRC has, nevertheless, initiated action to better define the problem and stimulate industry efforts to control it better. We conclude that licensing action on the proposed power increase should proceed independently of these efforts to further reduce the potential for stress corrosion cracking.

Iodine Behavior

The technical report (NUREG 0409) entitled "Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture" by A. K. Postma and P. S. Tam was published in January 1978. This report hypothesizes several chemical transformation and transport processes in an effort to better describe and analyze the accident. The models presented in this report tend to predict iodine releases in the early part of the postulated accident that are greater than those predicted by the model currently used by the staff.

On the basis of this information, the staff has initiated effort through experimental research to attempt to acquire data on drop size and analytical research to determine the sensitivity of the various parameters. If the investigation results in a finding of non-conservatism in the staff's previous analysis, they will be reviewed in light of the new information.

Since these predictions are hypothetical we have no basis at this time to apply them to Robinson. We, therefore, conclude that the Robinson power increase should not be withheld pending completion of efforts to attempt to validate these hypotheses.