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RECEIVED OCT 26 1981 U.S. NUCLEAR REGULATORY COMMISSION

Docket Nos.: 50-413/414

Mr. William O. Parker, Jr.
Vice President - Steam Production
Duke Power Company
P.O. Box 33189
Charlotte, North Carolina 28242

Dear Mr. Parker:

Subject: Request for Additional Information

In the performance of the Catawba station licensing review, the staff has identified concerns with regard to the following areas:

1. Effluent Treatment Systems - Safety (Enclosure 1)
2. Hydrologic Engineering - Environmental (Enclosure 2)
3. Emergency Preparedness (Enclosure 3)
4. Chemical Engineering - Corrosion Engineering (Enclosure 4)
5. Chemical Engineering - Chemical Technology (Enclosure 5)
6. Accident Evaluation - Systems Analysis (Enclosure 6)

Our review in other areas will be completed in the near future; and we will send you separate requests for additional information related to those areas. We request that you provide the information herein requested no later than December 7, 1981. If you require any clarification of this request, please contact the project manager, Kahtan Jabbour, at (301) 492-7821.

Sincerely,

Elinor G. Adensam, Branch Chief
Licensing Branch #4
Division of Licensing

Enclosures: As stated

cc: See next page

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SURNAME	KJabbour/eb	MDuncan	EAdensam			
DATE	10/20/81	10/21/81	10/22/81			

NRC FORM 318 (10/80) NRCM 0240 OFFICIAL RECORD COPY USGPO 1980-329-824

CATAWBA

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EFFLUENT TREATMENT SYSTEMS BRANCH

460.2 (Table 1.9-1) In Item II.F.1 of Table 1.9-1 it is not clear how many monitors are provided to cover the range from 10^{-7}uCi/cc to 10^{5} uCi/cc. Discuss the range covered by each monitor. Also, it is indicated that "these" monitors are attached to the outside of the unit vent. Are all detector components attached to the outside of the vent, as indicated? If the detector is mounted outside of the vent, the thickness of the vent is enough to preclude detection and measurement of gamma photons of less than 100 Kev energy. Justify the statement that "these monitors are sensitive to the 80 Kev range of noble gases...".

460.3 (Table 1.9-1) Item II.F.1 of Table 1.9-1 states that procedures will involve the use of "area radiation monitors". This statement is not sufficiently explicit for us t determine whether the proposal is acceptable. Area monitoring detection systems are considered as being acceptable for this purpose provided that: (1) detectors are "dedicated" as steam relief valve monitors, (2) that detectors are directionally shielded to limit background radiation interference from other sources, and (3) that procedures are developed and implemented to convert the monitor readout, usually in R/hr, to steam concentration values in uCi/cc.

460.4 (Table 1.9-1) The design for sampling and analysis of radioiodine and particulates in effluents must comply with the requirements for sampling and analysis of plant effluents as described in Attachment 2 of Section II.F.1 in Enclosure 3 of NUREG-0737.

460.4 (cont'd) The submittal does not address this item. Provide the information showing the compliance with the above requirement.

460.5 (Table 1.9-1) In Item III.D.1.1, "Integrity of Systems Outside Containment..", the containment atmosphere sampling and reactor coolant (post-accident) sampling systems should be included in the systems to be leak tested. Commit to provide initial leak rate measurement results to the staff. Also a summary description of the test procedures and the acceptance criteria used, should be submitted to NRC and implemented prior to full power license.

460.6 (Table 1.8-1) Table 1.8-1 indicates that Catawba design is in compliance with the requirements of Regulatory Guide 1.140 with the exceptions of C.2.b, C.3.1, and other sections. For position C.2.b, describe and justify your alternate approacn to the design. For position C.3.1, include the justification for non-compliance.

460.7 (Tables 1.8-1 11.2, 11.3, 11.4) Table 1.8-1 indicates that Regulatory Guide 1.143, Rev. 1, 10/79 (formerly Branch Technical Position ETSB 11-1, Rev. 1), "Design Guidance for Radioactive Waste Management Systems, Structures and Components Installed in Light-Water-Cooled Nuclear Power Plants" is not applicable to Catawba. This is not acceptable. Compare your design of liquid, gaseous and solid radwaste systems to each position in Regulatory Guide 1.143

and list the items of non-compliance and the justification for it for the purpose of evaluating your design.

460.8 (11.2) In Section 11.2.3 for estimating the liquid releases, credit is taken for waste evaporator and condensate demineralizer in processing the floor drain tank contents. From the description of the floor drain tank subsystem operation (11.2.2.7.1.5) it is not clear that evaporator and condensate demineralizers will be used frequently. Full credit may not be taken unless both of these pieces of equipment are used continuously. Similarly, the full credit should not be taken for waste evaporator condensate demineralizer for processing waste collected by waste evaporator feed tank. Please justify your assumption.

460.9 (11.3) Provide the normal operating pressure at which the waste gases are stored in the gas decay tanks. This pressure is used for calculating the holdup time in the gas decay tanks.

460.10 (11.3) Radiological Effluent Technical Specifications require that whenever the hydrogen concentration is higher than 4 % by volume, the oxygen concentration should be less than 2%. In the event the system exceeds the technical specification limit, immediate actions are required to reduce the oxygen

460.10 (cont'd) concentration below 2%. Some of the automatic features which should be included are discussed in Standard Review Plan 11.3. These automatic features are:

(1) for systems designed to preclude explosions by maintaining oxygen below 4%, the source of hydrogen or oxygen (as appropriate) should be automatically isolated from the system (valve should fail in closed position); (2) for systems using recombiners, if the downstream hydrogen and/or oxygen concentration exceeds 4% (as appropriate), acceptable control features include automatically switching to an alternate recombiner train; and (3) injection of diluents to reduce concentrations below the limits specified. Describe how your system will reduce the potential for explosion, should it exceed 4% oxygen limit with high concentration of hydrogen. As for example, this situation may occur due to air leaking into the system.

460.11 (11.4) As part of the solid radwaste system review, the staff needs a detailed description of the process control program; which discusses solidifying different kinds of waste. This may include reference to a topical report which has been approved by the staff or may be a detailed description of the program. This program will be used to verify the solidification process by taking representative test specimens, at specified intervals, 460.11 (cont'd) of each type of waste. This description should discuss verification of the absence of free water inside the container after the solidification process is completed. Recently, some of the shipments have been determined to have free liquid in excess of the limit specified by the burial site requirement.

460.12 (11.4) What will be the provision for storage in the event that shipment of compacted or other waste is not possible for 30 days due to circumstances beyond your control?

HYDROLOGIC ENGINEERING ENVIRONMENTAL QUESTIONS
CATAWBA NUCLEAR STATION

240.5 (ER) 2.4.1.1 The average annual flow of the Catawba River at the USGS gaging station 1460 near Rock Hill, South Carolina is given in paragraph 3, page 2.4-1. Please provide the average monthly, maximum monthly and minimum monthly flows recorded at the Rock Hill gaging station.

240.6 (ER) 2.4.1.1 Please update Table 2.4.1-3, Lake Wylie Minimum Surface Water Elevations, to include period from 1973 to present.

Provide a table similar to Table 2.4.1-3 giving the annual maximum recorded Lake Wylie water surface elevations.

240.7 (ER) 2.4.1.1 Please provide a table showing the minimum and maximum average monthly Lake Wylie water surface elevations during the period of record

240.8 (ER) 2.4.1.1 Please develop a water budget for Lake Wylie using average and minimum inflows, required discharges from Wylie Dam, natural and forced evaporation, and present and projected consumptive water use.

NOTE: The following questions are on the FSAR but are of an environmental nature. Therefore, responses are required in order to complete the environmental review.

240.9 (FSAR) 2.4.13.2.2 An unspecified number of water level recorders are mentioned in Section 2.4.13.2.2 of the FSAR. Please provide information on the number, locations, and periods of use of these recorders. Please provide representative copies of the hydrographs.

240.10 (FSAR) 2.4.13.2.4 Groundwater levels in borings varied several feet during subsurface exploration. Please provide an analysis of this variation, its magnitude and probable causes.

EMERGENCY PREPAREDNESS LICENSING BRANCH

We have completed our review of the Catawba Nuclear Station Emergency Plan dated August 1980. Your Plan was reviewed against the criteria set forth in "Criteria for Preparation and Evaluation of Radiological Emergency Plans and Preparedness in Support of Nuclear Power Plants," Rev. 1, November 1980. This document addresses the standards set forth in the revised emergency planning regulations of 10 CFR 50 and Appendix E thereto which became effective November 3, 1980.

Our review has indicated that additional information and commitments are required before we can conclude that your onsite emergency preparedness program meets these criteria. Enclosed are our comments for which resolution is necessary. Your Plan should be revised to address these comments in accordance with the provisions of 10 CFR 50.

As stated in paragraph 50.47(a)(2), of the rule, the NRC will base its findings of adequacy on a review of the Federal Emergency Management Agency (FEMA) findings and determinations as to whether State and local emergency plans are adequate and capable of being implemented, and on its assessment as to whether the applicant's onsite emergency plans are adequate and capable of being implemented. In addition, an emergency response exercise with State and local government designed to test the integrated capability of the emergency preparedness plans must be conducted before issuance of an Operating License.

REVIEW COMMENTS
CATAWBA NUCLEAR STATION
Docket Nos. 50-413, 414

The following comments apply to the Catawba Nuclear Station Emergency Plan (Plan) dated August 1980, and identify, in parentheses, the applicable evaluation criteria of NUREG-0654, Rev. 1:

1. General)

   o The South Carolina, North Carolina and applicable county emergency plans must be submitted for review.

   o The applicant's procedures which implement the Catawba Plan must be submitted for review.

   o The corporate Crises Management Plan must be submitted for review.

2. (A.3) Written agreements do not exist for all agencies/organizations identified in Section 5.3.2 of the Plan which are relied upon for support and services. Provide these agreements.

3. (A.4) The Plan should clearly state that the capability exists for 24-hour/day operations for a protracted period and specify the individual responsible for assuring continuity of resources.

4. (B.3) Identify a line of succession for the Emergency Coordinator position.

5. (B.4) The functions of the Emergency Coordinator which may not be delegated should be identified in the Plan.

6. (B.5) The Plan should provide for specific augmentation of the on-shift staff in less than one hour in accordance with Table B-1 of NUREG-0654.

7. (B.8) The Plan should identify the offsite organizations which would be relied upon for assistance.

8. (B.9) Refer to item A.3.

9. (C.1) The Plan contains insufficient information with respect to Federal assistance. Provide information to meet this criterion.

10. (C.2) The Plan should indicate that a licensee representative be present at the principal governmental EOC.

11. (D.1) The Plan should identify the parameter values and equipment status for each emergency class.

12. (D.2) An initiating condition for a Site Area Emergency should be included which states: "transient requiring operation of shutdown systems with failure to scram."

13. (E.1) The procedures for notification of response organizations should be provided for review.

14. (E.2) The procedures for alerting, notifying, and mobilizing personnel should be provided for review.

15. (E.6) The Plan should provide for prompt alerting, notifying and instructing the public in a manner which meets the criteria of Appendix 3, NUREG-0654.

16. (F.2) The Plan should provide for a communications link for fixed/mobile medical support facilities.

17. (F.3) The Plan should address periodic testing of emergency communications systems.

18. (G.1, G.2) The actual means utilized for dissemination of information to the public must be identified in the Plan. The brochure, etc., should meet the specified criteria and should be submitted for review.

19. (G.3, .4, .5) Neither the Plan nor the referenced Crises Management addresses interaction with the news media. Provide this information.

20. (H.1) The TSC and OSC should conform to the criteria of NUREG-0696.

21. (H.2) The EOF should conform to the criteria of NUREG-0696.

22. (H.8) Meteorological instrumentation and procedures should conform to the criteria of Appendix 2, NUREG-0654.

23. (H.11) The emergency equipment/supplies identified in Appendix 10.4 should be expanded to include all of the categories of this criterion.

24. (I.1) The Plan should identify plant system and effluent parameter values which are characteristic of a spectrum of off-normal conditions and accidents.

25. (I.2) Post-accident sampling capability and containment radiation monitoring should be addressed in the Plan.

26. (I.3) The methods/techniques for determining the source term of a release and the magnitude of the release should be addressed in the Plan.

27. (I.4) The relationship between effluent monitor readings and onsite and offsite exposures/contamination for meteorological conditions should be addressed.

28. (I.5) The capability of acquiring and evaluating meteorological information which meets the criteria of Appendix 2, NUREG-0654 should be addressed.

29. (I.6) The Plan should address a method for determining release rate/projected doses if instrumentation is offscale or inoperative.

30. (I.10) The Plan should provide for relating measured parameters to dose rates for key isotopes and gross radioactivity measurements and for estimating integrated dose from actual and projected dose rates and comparing these estimates with PAGs.

31. (J.8) The time estimates provided in the Plan for evacuation within the plume exposure EPZ should be revised to satisfy the criteria of Appendix 4, NUREG-0654.

32. (J.10) Maps have not been included in the Plan. Provide these maps to meet this criterion.

33. (K.3) The Plan should address 24-hour/day capability to determine doses received by emergency workers and the reading of dosimeters and record keeping for emergency workers.

34. (K.5) Decontamination of supplies, instruments and equipment and waste disposal should be addressed in the Plan.

35. (L.1) The Plan states that the Charlotte Memorial Hospital is the backup medical facility; however, no written agreement is included for this facility. Provide this agreement.

36. (N.1) Periodic exercises should simulate an emergency which results in offsite radiological releases.

37. (N.2) Communication with Federal emergency response organizations and States within the ingestion pathway should be tested quarterly.

38. (N.4) The Plan should state that offciial observers from Federal, State and/or local governments will observe and evaluate the exercise.

39. (O.3) The Plan should address that first aid training is equivalent to Red Cross Multi-Media.

40. (O.4) The Plan should state that specialized training and retraining programs will meet the criteria of 4.a-4.j.

41. (O.5) Retraining should be accomplished on an annual basis.

42. (P.2) The Plan should identify the individual, by title, with the overall authority and responsibility for emergency planning.

43. (P.3) The responsibilities of the Emergency Planning Coordinator should include development and updating of emergency plans.

44. (P.4) The Plan should be updated as needed and also reviewed and certified to be current on an annual basis. Provide this commitment.

45. (P.5) The Plan should address this criteria concerning administration of the Plan.

46. (P.6) The Plan should contain a list of supporting plans.

47. (P.7) The Plan should identify the actual procedures required to implement the Plan.

48. (P.8) The cross-reference to NUREG-0654, which is included in the Plan should be accurate and specific.

49. (P.9) An independent review of the emergency preparedness program is to be conducted every 12 months. Provide this commitment in the Plan.

50. (P.10) The Plan should address updating of telephone numbers in emergency procedures.

DUKE POWER COMPANY
CATAWBA NUCLEAR STATION, UNITS 1 AND 2
DOCKET NOS. STN 50-413/414
REQUEST FOR INFORMATION
CHEMICAL ENGINEERING BRANCH
CORROSION ENGINEERING SECTION

282.0
282.1
(10.3.5)

The secondary water chemistry monitoring and control program as you provided in the FSAR is incomplete. Provide a complete secondary water chemistry monitoring and control program following the guidance of Branch Technical Position MTEB 5-3 attached to SRP 5.4.2.1, Revision 2, July 1981.

ADDITIONAL INFORMATION NEEDED BY CHEMICAL ENGINEERING BRANCH FOR COMPLETING REVIEW ON CATAWBA NUCLEAR STATION, UNIT NOS. 1 AND 2

CHEMICAL TECHNOLOGY SECTION

281.1 (1.8, 6.1.2) (a) You stated that partial compliance with Regulatory Guide 1.54 will be made and that exception is taken to the applicability of ANSI Standards that are referenced in ANSI N101.4-1972. Acceptance criteria in Standard Review Plan Section 6.1.2 indicate that, to be acceptable, a coating system inside containment must meet the regulatory positions in Regulatory Guide 1.54, which endorses ANSI N101.4, and the standards of ANSI N101.2, which is referenced in ANSI N101.4. It is our position that you either meet the criteria of Reg. Guide 1.54 or propose specific alternatives to Regulatory Guide 1.54 and ANSI N101.2 with justification for each of these alternatives.

281.1 (6.1.2) (b) In the discussion of compliance with Regulatory Guide 1.54, you referenced Westinghouse four categories of equipment but discussed only Categories 1, 2, and 4. Provide a discussion on Category 3 of the Westinghouse nuclear steam supply system equipment inside containment that are covered with protective coatings and indicate the total exposed surface area and approximate thickness of the protective coatings on these equipment (Table 6.1.2-1 gives the painted surface area for small equipment as F1300 ft^2).

281.1 (6.1.2) (c) In reference NS-CE-1352 which you cited, Westinghouse Corporation did not give credit to the protective coatings on Category 2 equipment for adherence to surfaces under post-accident environment inside containment Verify that the same position applies to Catawba Nuclear Station and indicate the total exposed surface area and approximate thickness of the protective coatings on these equipment.

281.1 (6.1.2) (d) Indicate those paints listed in Table 6.1.2-2 that do not meet either the regulatory positions of Regulatory Guide 1.54 or the standards in ANSI N101.2, consistent with the alternatives that you may take to Regulatory Guide 1.54 and ANSI N101.2.

281.1 (6.1.2) (e) In Table 6.1.2-2, you stated that the total weight of the organic materials in the electrical cable insulation has not yet been determined. In order for the staff to estimate the rate of combustible gas generation vs time because of exposure of organic cable insulation to DBA condition inside containment, provide the following information for each of the four types of cable insulation materials: (1) the quantity (weight and volume) of uncovered cable and cable in closed metal conduit or closed cable trays. We will give credit for beta radiation shielding for that portion of cable that is indicated to be in closed conduit or trays. (2) A breakdown of cable diameters and associated conductor cross sections, or an equivalent cable diameter and conductor cross section that is representative of total cable surface area associated with the quantity of cable identified in (1) above.

281.2 (9.1.3) Describe the samples and instrument readings and their frequency of measurement that will be performed to monitor the Spent Fuel Pool (SFP) water purity and need for SFP cleanup system demineralizer resin and filter replacement. State the chemical and radiochemical limits to be used in monitoring the SFP water and initiating corrective action. Provide the basis for establishing these limits. Your response should consider variables such as: boron concentration, gross gamma and iodine activity, demineralizer and/or filter differential pressure, demineralizer decontamination factor, pH, and crud level.

281.3 (9.3.2, 11.5) (a) Acceptance criter... ..a in Standard Review Plan Section 9.3.2 indicates that sumps inside containment should be sampled. Describe provisions to sample sump water inside the containment in accordance with the requirements of General Design Criterion 64 in Appendix A to 10 CFR Part 50.

281.3 (9.3.2) (b) Acceptance criterion 3.e in Standard Review Plan Section 9.3.2 indicates that isolation valves should fail in the closed position. Verify that isolation valves in the process sampling lines that penetrate the containment will fail in the closed position.

281.4 (1.9, II.B.3) Provide information that satisfies the attached proposed license conditions for post-accident sampling.

SAFETY EVALUATION
BY THE OFFICE OF NUCLEAR REACTOR REGULATION
CATAWBA NUCLEAR STATION, UNIT NOS. 1 AND 2
DUKE POWER COMPANY
DOCKET NOS. 50-413/414

NUREG-0737, II.B.3 - Post Accident Sampling Capability

REQUIREMENT

Provide a capability to obtain and quantitatively analyze reactor coolant and containment atmosphere samples, without radiation exposure to any individual exceeding 5 rem to the whole body or 75 rem to the extremities (GDC-19) during and following an accident in which there is core degradation. Materials to be analyzed and quantified include certain radionuclides that are indicators of severity of core damage (e.g., noble gases, iodines, cesiums and non volatile isotopes), hydrogen in the containment atmosphere and total dissolved gases or hydrogen, boron and chloride in reactor coolant samples in accordance with the requirements of NUREG-0737.

To satisfy the requirements, the applicant should (1) review and modify his sampling, chemical analysis and radionuclide determination capabilities as necessary to comply with NUREG-0737, II.B.3, (2) provide the staff with information pertaining to system design, analytical capatilities and procedures in sufficient detail to demonstrate that the requirements have been met.

EVALUATION AND FINDINGS

The applicant has committed to a post-accident sampling system that meets the requirements of NUREG-0737, Item II.B.3 in Amendment 20, but has not provided the technical information required by NUREG-0737 for our evaluation. Implementation of the requirement is not necessary prior to low power operation because only small quantities of radionuclide inventory will exist in the reactor coolant system and therefore will not affect the health and safety of the public. Prior to exceeding 5% power operation the applicant must demonstrate the capability to promptly obtain reactor coolant samples in the event of an accident in which there is core damage consistent with the conditions stated below.

1. Demonstrate compliance with all requirements of NUREG-0737, II.B.3, for sampling, chemical and radionuclide analysis capability, under accident conditions.

2. Provide sufficient shielding to meet the requirements of GDC-19, assuming Reg. Guide 1.4 source terms.

3. Commit to meet the sampling and analysis requirements of Reg. Guide 1.97, Rev. 2.

4. Verify that all electrically powered components associated with post accident sampling are capable of being supplied with power and operated, within thirty minutes of an accident in which there is core degradation, assuming loss of off site power.

5. Verify that valves which are not accessible for repair after an accident are environmentally qualified for the conditions in which they must operate.

6. Provide a procedure for relating radionuclide gaseous and ionic species to estimated core damage.

7. State the design or operational provisions to prevent high pressure carrier gas from entering the reactor coolant system from on line gas analysis equipment, if it is used.

8. Provide a method for verifying that reactor coolant dissolved oxygen is at $<$ 0.1 ppm if reactor coolant chlorides are determined to be $>$ 0.15 ppm.

9. Provide information on (a) testing frequency and type of testing to ensure long term operability of the post accident sampling system and (b) operator training requirements for post-accident sampling.

In addition to the above licensing conditions the staff is conducting a generic review of accuracy and sensitivity for analytical procedures and on-line instrumentation to be used for post-accident analysis. We will require that the applicant submit data supporting the applicability of each selected analytical chemistry procedure or on-line instrument along with documentation demonstrating compliance with the licensing conditions four months prior to exceeding 5% power operation, but review and approval of these procedures will not be a condition for full power operation. In the event our generic review determines a specific procedure is unacceptable, we will require the applicant to make modifications as determined by our generic review.

ACCIDENT ANALYSIS BRANCH
SYSTEMS ANALYSIS SECTION

450.01 (6.4) Provide the following information required for the Control room habitability evaluation:

(1) control room air volume
(2) charcoal absorber filter efficiencies in the emergency pressurization and recirculation lines
(3) control room personnel capacity (normal and emergency)
(4) data, assumptions, and methodology used to calculate the control room X/Q's.

450.02 (6.5) The applicant has submitted an analysis of the ice condenser which results in a time dependent efficiency rate. The analysis included an assumption that the steam - air mixture coming into the ice condenser is greater than 90% steam by volume. This assumption is contrary to the staff position established in a memorandum from Harold Denton to Richard DeYoung dated November 24, 1972.

The memorandum fowarded an attachment which contained the staff review of WCAP-7426 and said in part "The above considerations have led the licensing staff to assume an elemental iodine removal efficiency of 30% for sodium tetraborate impregnated ice. This efficiency corresponds to an inlet steam-air mixture which is 40% steam (by volume). The above iodine removal efficiency (30%) and the analytical description described in Appendix A of WCAP-7426 were used in the accident evaluation of...."

Therefore, in view of the previous analysis and the established staff practice, we will allow credit for an efficiency of 30% during the effective ice condenser removal capability.

In order for us to complete our analysis, we request that you provide the length of operation of the ice condenser (including the start and end times) and the delay time before startup of the recirculation fans.

450.03 (15.4.8) For the accident of rod ejection with loss of offsite power, the following parameters are needed for our analysis:

(1). The time required for the pressures of primary side and secondary side to equalize.

(2). The time required to start shutdown cooling.

450.04 (15.6.3) The radiological consequence analysis for the steam generator tube rupture accident assumes that the affected steam generator can be isolated in 30 minutes. Based on figures 15.6.3-2 and 15.6.3-4 it seems very unlikely that the affected steam generator could be isolated. Therefore, provide additional information which supports the assumption that the affected steam generator can be isolated within 30 minutes or provide additinal steam release volumes from the affected steam generator resulting from the steam dump until the affected steam generator can be isolated.

Also provide both the amount of primary to secondary leakage in the affected steam generator prior to the reactor trip and the amount of primary to secondary leakage after reactor trip up to the time of steam generator isolation.

450.05 (15.6.5) In the analysis of the offsite radiological consequences of the Design Basis Accident (FSAR Section 15.6.5.4.1), provide the following information:

(1) operating characteristics of the Annulus Ventilation System following a LOCA, including iodine filter efficiencies, total system flow, recirculation flow, pressurization flow, time at which total, pressurization and recirculation flows are initiated, description of time dependent characteristics of recirculation and pressurization flows, and a schematic showing leakage paths, flow rates, filter efficiencies.

(2) free volumes of the upper compartment, lower compartment, and annulus.

450.06 (15.6.5) Assumption 10 in FSAR Section 15.6.5.4.1 states that containment air return fan is assumed to fail. Describe the significance of the containment air return fan in the calculation of the postulated offsite radiological consequences and the effect the failure of this fan has on the doses.

450.07 (15.6.5) Assumption 13 in FSAR Section 15.6.5.4.1 states that after 300,000 seconds the annulus discharge is no greater than the annulus inleakage of 676 CFM. Provide an explanation of the significance of this assumption in your analysis.

450.08 (15.7.4) In the Evaluation of the postulated fuel handling accident inside containment, describe the type of air cleanup units which incorporate an iodine absorber with a removal decontamination factor of 6.7. If the air cleanup units are not ESF grade, evaluate this accident without this decontamination factor. The staff position is that no credit can be taken in the dose analysis for non-ESF grade air cleanup units.

450.09 (15.7.5) Provide the maximum postulated spent fuel cask drop distance. If the spent fuel cask drop distance exceeds 30 feet, evaluate the design basis radiological analysis in accordance with SRP 15.7.5.