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Docket Nos.: 50-413/414

To Distribution and Service List

SUBJECT: NRC Instrumentation and Control Systems Branch (ICSB) Agenda Items for Discussion with Duke on Catawba Station

Gentlemen:

The attached pages (i.e. 3 through 23) were inadvertently missing from Enclosure 1 to the letter dated December 21, 1981 addressed to Mr. William O. Parker, Jr. from Elinor G. Adensam.

Sincerely,

/S/

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

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Enclosure:
As stated

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SURNAME	KJabbour	MDuncan	EAdensam				
DATE	1/20/82	1/20/82	1/20/82				

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5. Provide an overview of the plant electrical distribution system, with emphasis on vital buses and separation divisions, as background for addressing various Chapter 7 concerns.

6. Indicate whether Catawba has the Westinghouse "General Warning Alarm System". If so, provide a list of conditions resulting in a general warning alarm and update the FSAR to include the input to the RTS from the general warning alarm.

7. Using detailed schematics describe the operation of circuits used for the NSW system. Discuss the design criteria for the instrumentation and control (i.e., indicators available, testability, automatic switchover). Also discuss the interface with the bypass and inoperable status panel.

8. Using detailed schematics describe the operation of circuits used for isolation of NSW to the air compressors. Discuss periodic testing and indicate which components (including sensors) are located in seismic qualified structures.

9. For the RTS and ESFAS, revise the FSAR Sections 7.2.2.1 and 7.3.2.1 to include the basis, assumptions, and results of the referenced FMEA.

10. Confirm that the FMEA, referenced in Section 7.3.2.1, for ESFAS includes (1) all BOP scope and (2) design changes subsequent to the design analyzed in the WCAP.

11. Section 6.2.1.1.3.2.2 states that "Instrumentation provided to monitor and record the containment pressure and temperature and sump temperature during the course of an accident within the containment is discussed in Chapter 7". In our review of Chapter 7 we find some information on the containment pressure in Section 7.3.1.1.2 and Tables 7.5.1-1 and 7.5.1-2; however, no information on the instrumentation for the containment and sump

temperatures could be found. Clarify the apparent discrepancy.

12. Identify all plant safety-related systems, or portions thereof, for which the design is incomplete.

13. Identify where microprocessors, telemetry systems, multiplexers, or computer systems are used in or interface with safety-related systems.

14. Identify any sensors or circuits used to provide input signals to the protection system or perform a function required for safety which are located or routed through non-seismically qualified structures. This should include sensors or circuits providing input for reactor trip, emergency safeguards equipment such as auxiliary feedwater system and safety-grade interlocks. Verification should be provided that the sensors and circuits meet IEEE-279 and are seismically and environmentally qualified. Identify any testing or analyses performed which insure that failures of non-seismic structures, mountings, etc. will not cause failures which could interfere with the operation of any other portion of the protection system.

15. Describe how the effects of high temperatures in reference legs of steam generator water level measuring instruments subsequent to high energy breaks are evaluated. Identify and describe any modifications planned or taken in response to IEB 79-21. Describe the level measurement errors due to environmental temperature effects on level instruments (excluding steam generator level) including reference legs.

16. Describe features of the Catawba environmental control system which insure that instrumentation sensing and sampling lines for systems important to safety are protected from freezing during extremely cold weather. Discuss the use of environmental monitoring and alarm systems to prevent loss of, or damage to systems important to safety upon failure of the environmental

control system. Discuss electrical independence of the environmental control system circuits.

17. Identify the lead-lag constants used in the RPS and ESF's.

18. Provide and describe the following information for NSSS and BOP safety related setpoints: (a) Provide a reference for the methodology used. Discuss any differences between the referenced methodology and the methodology used for Catawba. (b) Verify that environmental error allowances are based on the highest value determined in qualification testing. (c) Identify protection channels where the Technical Specification setpoint, with allowance for channel statistical error, falls within 5% of the instrument range limit or within 5% of the range between level measurement taps. For those cases, specify the remaining margin to the end of the range. (d) Document the environmental error allowance that is used for each reactor trip and engineered safeguards setpoint. (e) Identify any time limits on environmental qualification of instruments used for trip, post-accident monitoring or engineered safety features actuation. Where instruments are qualified for only a limited time, specify the time and basis for the limited time. (f) Address the effect of test equipment accuracy on setpoint errors. (g) As an example, derive the setpoints for the low-low steam generator level trip.

19. The information provided in Section 2.2.1 (including Table 2.2-1) and Tables 3.3-1, 3.3-2, and 4.3-1 of the Tech. Specs. include a trip for Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level. This trip is not discussed in Section 7.2.1.1.2. Clarify this apparent discrepancy.

20. Table 15.0.6-1 lists the following limiting trip points assumed in the accident analysis:

    a. Power range high neutron flux, high setting --- 118%

b. Low pressurizer pressure -- 1921 psig.

c. Low-low steam generator level -- 25% of narrow range level span.

The trip setpoints shown in Table 2.2-1 of the Tech. Specs. for items b. and c. above are ≥1865 psig (allowable values ≥1855 psig) and ≥10% (allowable value ≥9%) respectively. These values are considerably lower (less conservative) than those used for the accident analysis, rather than being more conservative to account for instrument errors. Also, replacing the 109% value used in Table 15.0.7-1 with the allowable value of 110% of Table 2.2-1, would result in a change in maximum overpower trip setting from 118% to 119%.

Discuss how the results of the accident analysis would be affected if the allowable trip settings given in the Tech. Specs. would be used in lieu of the values shown in Table 15.0.6-1. Revise the allowable trip settings, if needed, to prevent the safe operating limits from being exceeded.

21. The trip setpoints specified in Section 7.2.1.1.2 for item 4b, Reactor Coolant Pump Undervoltage, is 70% of rated voltage and for item 4c, Reactor Coolant Pump Underfrequency, is 56 Hz. No values have been provided for these setpoints in Table 2.2-1 of the Tech. Specs.

    Specify the trip setpoints and the allowable values for the above parameters in Table 2.2-1. Describe also the equipment used for the monitoring, and discuss how the periodic tests on these monitors are performed.

22. Page 7.3-7; provide a discussion of accuracy, or a reference to supplement the "typical" accuracy information given. Relate the accuracy requirements of the plant, such as for the safety analyses, to demonstrated equipment accuracy.

23. Tables 3.3-3, 3.3-4 and 4.3-2 of the Tech. Specs. include the following parameters for ESFAS:

- Differential Pressure Between Steam Lines
- Steam Flow in Two Steam Lines -- High, Coincident with T_{avg}---Low or Steam Line Pressure -- Low

These parameters are not discussed in Section 7.3. Clarify this apparent discrepancy.

24. Tables 3.3-3, and 3.3-4, Engineered Safety Feature Actuation System Instrumentation, of Technical Specifications do not include the loss of both Main FW Pumps as a parameter to start the auxiliary feedwater pumps. We find, however, the loss of both Main FW Pumps listed as a parameter to start the motor driven pumps in the description of the Auxiliary Feedwater System in Section 10.4.9, and also in Section 7.3.2.3. Clarify the apparent discrepancy.

25. Tables 3.3-9 and 4.3-6 in the Tech. Specs. specify the following readouts on the Auxiliary Shutdown Panel:

- Reactor Coolant Temperature - W/R hot leg
- Reactor Coolant Flow Rate

Tables 7.4.7-1 through 7.4.7-2 do not show monitors for the reactor coolant flow rate and show monitors for the reactor coolant temperature W/R to cold leg vice hot leg. Clarify the apparent discrepancies.

26. In the Safety Evaluation Report (and supplements) issued for the Catawba construction permit the following items required resolution during the operating licensee review:

    (1) Include activation of the RHR and safety injection pumps as an integral part of the ECCS periodic tests.

    (2) Include in the design the capability to test the reactor trip from a safety injection signal without being restricted

or limited by power operation of the reactor.

Discuss the status of the above items.

27. Identify where instrument sensors or transmitters supplying information to both a protection channel and control channel or to more than one control channel are located in a common instrument line or connected to a common instrument tap. The intent of this item is to verify that a single failure in a common instrument line or tap can neither defeat required separation between control and protection nor cause multiple control system actions not bounded by analyses contained in Chapter 15 of the FSAR. For control systems, the discussion can be limited to channels used for control of reactivity, reactor coolant pressure, reactor coolant temperature, reactor coolant flow, reactor coolant inventory, secondary system pressure, steam generator feedwater flow and steam generator steam flow.

28. Describe the scram response time testing.

29. Portions of paragraph 7.3.1.2.6, subparagraph 1, appear not to apply to ESFAS response times. In particular, the discussion on reactor trip breakers, latching mechanisms, etc. should be replaced by a discussion of ESF pump and valve time responses. The applicant should provide a revised discussion for ESFAS (a) defining specific beginning and end points for which the quoted times apply and (b) relating these times to the total delay for all equipment and to the accident analysis requirements.

30. The information provided in Section 7.2.2.2.3 on testing of the power range channels of the Nuclear Instrumentation System, covers only the testing of the high neutron flux trips. Testing of the high neutron flux rate trips is not included.

Provide a description of how the flux rate circuitry is tested periodically to verify its performance capability.

31. In the discussion of the auxiliary shutdown control in Section 7.4.7.2 it is stated that "The safety evaluation of achieving and maintaining hot shutdown with the controls available at the auxiliary shutdown panels includes consideration of transients whose consequences might jeopardize the safe shutdown conditions".
Provide a detailed discussion on the type and severity of transients considered, and describe the effect these transients have on the auxiliary shutdown control.

32. Provide a table showing safe shutdown display information and identify safety grade items.

33. Discuss the capability of achieving hot and cold shutdown from outside the control room including:
    a) list of qualified displays, location and basis for selection.
    b) description and location of auxiliary shutdown panel or equivalent.
    c) description of required controls.
    d) description of isolation, separation, and transfer/override provisions. Discuss a typical transfer scenario.
    e) description of any communications system required to coordinate operator actions, including redundancy, separation, and environmental qualification for local environment.
    f) description of control room annunciation of remote control or overridden status of devices under local control.

g) description of any auxiliary system essential to remote shutdown capability.

h) description of control of access to the displays and controls located outside the control room.

i) testing during reactor operation.

j) means for ensuring that cold shutdown can be accomplished before Technical Specification limits on hot shutdown are exceeded.

k) address statement at bottom of page 7.4-20: "Cold shutdown conditions can be reached from outside the control room with some temporary instrumentation and control modification."

34. Section 7.4.3.1 states that "There are no bypasses capable of preventing the Component Cooling Water System from performing its safety function; however, in the event system control must be transferred to the auxiliary shutdown panel, <u>some</u> automatic signals are defeated (refer to Section 7.4.7)".

Provide a list of the automatic signals that are defeated. Discuss which controls and interlocks, if any, that originate or pass through the control room and/or cable room, remain active.

35. As discussed in Section 7.2.2.3, isolated output signals from protection system channels are utilized to generate a control signal to automatic control systems, such as rod drive system, pressurizer pressure and level control, and others. The control signal is derived by auctioneering the redundant protection system channels to select the high or low signal, whichever is chosen based on consideration of safety in case of a failure.

Discuss what steps, if any, are taken to prevent unnecessary control action, such as opening of pressurizer relief valves, during the testing of protection system channels with a signal from a test source.

36. Will a test be performed to verify the capability of maintaining the plant in a safe shutdown condition from outside the control room?

37. Page 7.3-16 Testing During Shutdown. Describe provisions for insuring that the "isolation valves" discussed here are returned to their normal operating positions after test.

38. Describe compliance with R. G. 1.118 and IEEE Std. 338-1975. Confirm that tech. specs. provide detailed instructions which insure that blocking of a selected protection function actuator circuit is returned to normal operation after testing. Confirm that tech. specs. include RTS and ESFAS response times for reactor trip functions. Confirm that tests include all components, from sensor to operation of final actuation device, and describe a typical response time test. Indicate any area of non-compliance with basis for each.

39. Page 7.1-8 The statement is made, "The Westinghouse design of protection systems incorporates overcurrent devices to prevent malfunctions in one circuit from causing unacceptable influences on the functioning of the protection system." Provide information on the specific places where this is done and the basis that this design does not compromise protection channel independence. Discuss conformance with R.G. 1.75 and IEEE 384-1974.

40. Describe the design criteria and tests performed on the isolation devices in the NSSS and BOP. Address results of analysis or tests performed to demonstrate proper isolation between separation groups.

41. The discussion in Section 7.1.2.2 states that Westinghouse tests on the Series 7300 PCS system covered in WCAP-8892 are considered applicable to Catawba. As a result of these tests, Westinghouse has stated that the isolator output cables will be allowed to be routed with cables carrying voltages not exceeding 580 volts a.c. or 250 volts d.c. The discussion of isolation devices in Section 7.5.3.3.9 of the FSAR, however, considered the maximum credible fault accidents of 118 volts a.c. or 140 volts d.c. only. Also, the statement in Section 7.7.2.1 implies that the isolation devices were tested with 118 volts a.c. and 140 volts d.c. only. In order to clarify the apparent nonuniformity, provide the following:

(a) Specify the type of isolation devices used for Catawba Process Instrumentation System. If they are not the same as the Series 7300 PCS tested by Westinghouse, specify the fault voltages for which they are rated and provide the supporting test data.

(b) Provide information requested in (a) above for the isolation devices of the Nuclear Instrumentation System. As implied in WCAP-8892, the tests on Series 7300 PCS did not include the Nuclear Instrumentation System.

(c) Describe what steps are taken to insure that the maximum credible fault voltages which could be postulated in Catawba, as a result of BOP cable routing design, will not exceed those for which the isolation devices are qualified.

42. Page 7.1-14. The section covering compliance with R.G. 1.53 addresses only Westinghouse equipment and associated topical reports. Provide the equivalent information for the BOP portions of plant safety systems and auxiliary systems required for support of safety systems.

43. Pages 7.1-14. The section covering compliance with R.G. 1.47 addresses only ESF systems. State compliance to R.G. 1.47 for other Catawba safety related systems.

44. Describe the implementation of the bypassed and inoperable status indication provided for ESF and compliance with R.G. 1.47. Discuss types of status displays and alarms. Discuss computer utilization and software verification and validation techniques.

45. The discussion on the operating bypasses of the Reactor Trip System (RTS) in Section 7.2.2.2.3 (Item 13), refers the bypass indication to Section 7.8. In our review of Section 7.8 we find Section 7.8.3, titled ESF Bypass Indication, but we can find no discussion on the RTS bypass indication.

    Describe the RTS bypass indication system, provide a list of systems for which bypass indication is provided, and discuss how the bypass indication system conforms to the requirements of Regulatory Guide 1.47.

46. Page 7.1-16. Provide a schedule for developing IEEE-338 reliability goals and demonstrating the adequacy of test frequencies.

47. Discuss the plans and schedule for complying with R.G. 1.97, Rev. 2. Describe the conformance of the present design.

48. Describe how separation criteria for protection channel circuits, protection logic circuits, and non-safety related circuits complies with R.G. 1.75. Indicate the separation method between these circuits. Discuss a typical example for each type circuit including an intra-panel wiring circuit.

49. Discuss the method of redundantly tripping the turbine following receipt of reactor protection signals requiring turbine trip.

50. Discuss the diverse features of the undervoltage and shunt trips of the reactor trip breakers. Indicate if they can be tested independently.

51. Using detailed schematics, describe the design of the pressurizer PORV control and the block valve control. Does the current design still provide for actuation of the pressurizer spray or relief valves upon a single instrument failure. Identify and describe design features which ensure that the RCS pressure is safely controlled during low temperature operation to include parameters utilized and monitored for alarm indication. Discuss the degree of redundancy in the logic for the low temperature interlock for the RCS pressure control.

52. Describe the design features used to provide direct indication of pressurizer safety and relief valve positions in the control room.

53. Page 7.2-29. The fourth paragraph implies that a turbine trip may open the pressurizer code safety valves. Please discuss.

54. Provide analysis indicating whether the pressurizer PORV will be actuated following a turbine trip below the power setpoint of P-9. The analysis should cover core physics parameters bracketing those expected throughout the core lifetime.

55. Describe the electrical power supply arrangement, air supply design features, and any interlocks associated with control and operation of the steam generator PORV.

56. Section 7.7.1.7 and Figure 7.7.1-6 conflict concerning the basis for programming steam generator water level. Also there is no input label for the filter in Figure 7.7.1-6. Correct FSAR as appropriate.

57. Describe the steam generator level instrumentation. Identify the

channels for protection functions, control functions, and post-accident monitoring.

58. Provide an analysis indicating the time between reaching each high steam generator level alarm setpoint and filling the steam generator with water assuming failure of the level channel used for control in the low direction. Since only two out of three logic is used for high steam generator level, the remaining two channels do not meet the single failure criteria. Assume that the isolation function does not occur. The initial plant power level resulting in the most rapid steam generator filling should be assumed. The applicant should be prepared to discuss the probable consequences of filling the steam generator and causing water to flow into the steam piping and the consequences of a steam generator level control channel failure.

59. Using detailed schematics describe the protection for locked rotor or sheared shaft of the reactor coolant pumps.

60. Describe the procedures to borate the primary coolant from outside the control room when the main control room is inaccessible. How much time is there to do this?

61. The discussion of the philosophy of protection for the reactor coolant pumps which is presented in FSAR Section 7.3.2.3 is inadequate. Therefore:

    (a) Identify any situations in which a control room alarm will not be actuated upon loss of component cooling flow to the reactor cooling pumps.

    (b) Quantify the time delay between a loss of component cooling to the reactor coolant pumps and the initiation of an alarm.

    (c) Describe how the operator corrects for a loss of component cooling water flow to the reactor coolant pumps during a seismic event or at any other time flow is lost.

(d) Quantify the time it will take an operator (after an alarm is received) to attempt to take the corrective action which is identified in response to part (e) above and the time which is required to evaluate the results of this attempt.

(e) Describe the consequences of a failure to take effective corrective action (including reactor trip) within 10 minutes.

62. Using detailed system schematics, describe the power distribution for the accumulator valves and associated interlocks and controls including bypass indicator light arrangement.

63. The discussion of the accumulator motor-operated valves in Section 7.6.4 does not include information on the valve position indication. Provide this information and discuss how the requirements 2 and 3 of the Branch Technical Position ICSB 4 regarding the valve position indication and alarms, are followed.

64. In the event of a boron dilution transient, describe the operation of the detection system and the interface arrangement with the protection system for valve actuation. Indicate if the neutron detector is qualified both environmentally and seismically. Confirm quality of detectors is Category I.

65. Confirm that the reactor coolant pump breakers are designed and qualified to meet all criteria applicable to equipment performing a safety function. If not, provide the basis for determining that tripping the pump breakers on underfrequency is not a safety function.

66. Using detailed schematics, verify that no single failure will preclude reactor coolant system letdown capability.

67. Discuss whether the motor-operated valves in the safety injection pump lines from RWST receive an automatic signal following SI initiation.

68. Confirm that tech. specs. will include surveillance requirements for the RTD bypass loop flow alarms.

69. Identify all remotely controlled valves in the Engineered Safety Features Systems which require power lockout during a certain mode of operation, and discuss how the design meets Branch Technical Position EICSB-18.

70. Table 1.3.1-1 states that there are no significant differences between the ESFAS for Catawba and those of McGuire and Watts Bar. Our review shows that several parameters associated with the safety injection and/or containment and steam line isolation for McGuire and Watts Bar are not provided for Catawba. These parameters are:

    (1) High Differential Pressure Between Steam Lines
    (2) High Steam Flow
    (3) Pressurizer Water Level (Watts Bar only)

    Although the above parameters are not included in Table 7.3.1-1 and are not discussed in 7.3.1.2.6, credit appears to be taken for monitoring pressurizer water level in 7.3.2.4.1. Clarify the apparent discrepancies and amend Table 1.3.1-1 as necessary.

71. Using detailed system schematics, discuss the bypass, bypass interlock, and test provisions for containment ventilation isolation and control room ventilation isolation. The discussion should indicate those design features which insure that the safety function is not defeated during system test and that portions of the system are not inadvertently left in a bypassed condition after test.

72 Describe the interface between the radiation monitoring system (RMS) and ESFAS for containment ventilation and fuel building isolation to include the use of non-safety grade equipment in RMS and ESFAS.

73. Describe the method of providing redundancy for equipment in certain ventilation systems such as Cable Spreading Room, Exhaust Isolation, Control Room Exhaust, Isolation and Control Building Outside Air. Prepare a list of Ventilation Isolation Control System actuated equipment and indicate number of actuation channels for each.

74. Describe automatic and manual design feature permitting switchover from injection to recirculation mode for emergency core cooling including protection logic, component bypasses and overrides, parameters monitored and controlled, and test capabilities. Discuss design features which insure that a single failure will neither cause premature switchover nor prevent switchover when required. Discuss the reset of Safety Injection actuation prior to automatic switchover from injection to recirculation and the potential for defeat of the automatic switchover function. Confirm whether the low-low level refueling water storage tank alarms which determine the time at which the containment spray is switched to recirculation mode are safety grade.

75. Using detailed system schematics, describe the sequence for automatic initiation, operation, reset, and control of the auxiliary feedwater system. The following should be included in the discussion:

   a) the effects of all switch positions on system operation.
   b) the effects of single power supply failures including the effect of a power supply failure on auxiliary feedwater control after automatic initiation circuits have been reset in a post accident sequence.

c) any bypasses within the system including the means by which it is insured that the bypasses are removed.

d) initiation and annunciation of any interlocks or automatic isolations that could degrade system capability.

e) the safety classification and design criteria for any air systems required by the auxiliary feedwater system. This should include the design bases for the capacity of air reservoirs required for system operation.

f) design features provided to terminate auxiliary feedwater flow to a steam generator affected by either a steam line or feed line break.

g) system features associated with shutdown from outside the control room.

h) logic circuits used to transfer pump suction from the Condensate Storage Tank to the Nuclear Service Water System including verification that all equipment used for this function is seismically qualified and protected from failure of near-by structures which may not be seismically qualified.

i) design features to insure that no single failure can result in an open flow path from the Nuclear Service Water System to the Condensate Storage Tank.

76. The information in Section 10.4.9 on the auxiliary feedwater system does not specify the criteria applied in the design of the control and instrumentation systems for the modulating level control valves. Describe the instrumentation and controls, identify the power sources used for each of the valves, and provide an analysis to show that no single failure can prevent supplying auxiliary feedwater when required.

77. Describe the instrumentation provided for monitoring the loss of both main feedwater pumps and how the design meets the requirements of IEEE 279-1971.

78. Revise Table 7.3.1-3 to include, under P-11, the automatic resetting of the "Auto-Start Defeat" logic for auxiliary feedwater pumps as discussed on page 7.4.3.

79. For main steam and feedwater line valve actuation, describe control circuits for isolation valves and include automatic, manual and test features. Indicate whether any valve can be manually operated and whether each valve actuation level is alarmed in the control room. Indicate specific interfaces with the safety system electrical circuits.

80. Describe the operation of the interlocks used for isolation of the seismic qualified portion of the CCW system. This discussion should include reference to the fluid system schematics indicating which specific valves are used for the isolation function. Discuss whether redundancy of instrumentation is within each CCW train or is accomplished by having one interlock per train.

81. Regarding CCW pumps; discuss (a) circuits which automatically start a pump in a CCW train on low pressure in the pump discharge and (b) circuits which automatically start a pump in the operating train on an SIS. Include subsequent operator control of the pumps.

82. Using detailed schematics, describe the sequence of operation for the RHR isolation valves. The discussion should include the effects of various single failures in power supplies for the valves and their controls. Indicate any single instrument bus failure which could cause inadvertent closure of RHR suction valves in both trains when the system is in use for decay heat removal. Describe how proper operation of the RHR isolation

valves will be insured in the event of interlock or power failure. Identify testing planned for this area.

83. Identify and describe features of the RHR system motor operated isolation valve interlocks designed to prevent overpressurization of the RHR system to include separation and independence measures. Discuss compliance with ICSB BTP No. 3.

84. As described in Section 10.4.9.2, the Nuclear Service Water System is used in emergencies to supply water to the auxiliary feedwater motor-driven pumps. This supply is initiated automatically by a two-out-of-three low pressure signal in the condensate suction line. Describe the type and qualification of the pressure monitoring equipment and circuits.

85. Table 7.5.1-1 shows that the Post-Accident Monitoring System includes two Wide-Range T_{hot} channels, and two T_{cold} channels. It is stated that "The T_{hot} channels are on separate power supply from the T_{cold} channels." From this statement we conclude that both T_{hot} channels are on one power supply, and both T_{cold} channels are also on one power supply. This would imply that a loss of a power supply would result in a loss of data from either both T_{hot} or T_{cold} channels whichever is applicable. Discuss what consideration was given to the loss of T_{hot} or T_{cold} data.

86. Using detailed schematics, describe the operation of the UHI system. From the description of upper head injection (UHI) interlocks in Section 7.6.3, it appears that the requirements of Branch Technical Position ICSB 4 in providing automatic opening of the valves whenever either primary coolant system pressure exceeds the preselected value, or a safety injection signal is present, are not followed. If so, justify the approach taken. Also confirm that the ac control power supply used for

the valve position indicating lights is independent of the power supply used for the annunciators that alarm if the valve is not fully closed above a set pressure, as required by ICSB 4.

87. Using detailed schematics, describe the operation of the containment spray system. Discuss the redundancy of the spray additive tank isolation valves which are closed on low additive tank level. Describe how the transmitter used to close these valves are monitored. Indicate periodic test requirements for the instrumentation and controls used.

88. Describe the design features used in the rod control system which

    1) Limit reactivity insertion rates resulting from single failures within the system.

    2) Limit incorrect sequencing or positioning of control rods.

    The discussion should cover the assumptions for determining the maximum control rod withdrawal speed used in the analyses of reactivity insertion transients.

89. List the basis, assumptions and results from the FMEA (WCAP-8976) for the rod control systems.

90. Section 7.8.2, Monitor Light Panels, includes a statement that "An energized light on a monitor panel normally indicates that the monitored equipment is in its safety position or mode. Exceptions to this convention are identified to the operator".

    Discuss the reasons why the exceptions were taken, list the equipment involved, and describe how the exceptions are identified to the operator.

91. Using detailed schematics describe the design of boric acid addition control and the volume control tank level control. Discuss the recent Westinghouse generic deficiency regarding volume control tank level and its applicability to Catawba.