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DENTON,H.R. Office of Nuclear Reactor Regulation, Director

SUBJECT: Forwards response to request for addl info on SER Open
Items 31,47,365 & 139/369.

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Carolina Power & Light Company

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NOV 04 1983

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
United States Nuclear Regulatory Commission
Washington, DC 20555

SHEARON HARRIS NUCLEAR POWER PLANT
UNIT NOS. 1 AND 2
DOCKET NOS. 50-400 AND 50-401
RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION

Dear Mr. Denton:

Carolina Power & Light Company hereby transmits one original and forty copies of additional information requested by the NRC as part of the safety review of the Shearon Harris Nuclear Power Plant. The cover sheet of the attachment summarizes the related Open Items addressed in the attachment along with the corresponding review branch and reviewer for each response.

We will be providing responses to other requests for additional information shortly.

Yours very truly,

M A McDuffie

M. A. McDuffie
Senior Vice President
Nuclear Generation

JHE/mf (8419COM)
Enclosures

cc: Mr. B.C. Buckley (NRC)
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ATTACHMENT

LIST OF OPEN ITEMS/NEW ISSUES, REVIEW BRANCH AND REVIEWER

Auxiliary Systems Branch/N. Wagner
Open Items 365 and 139/369

Core Performance Branch./T. Huang
Open Item 31

Reactor Systems Branch/E. Marinos
Open Item 47

Shearon Harris Nuclear Power Plant
Draft SER Open Item 365
ASB Question 9.2.2(1)
Revised Response

NRC Question:

Show that all non safety-related heat loads in the Component Cooling Water (CCW) System are isolated from safety-related loads in the event of suitable emergency initiating signals.

Clarification:

Identify how cooling would be provided to the spent fuel pool cooling system subsequent to a LOCA. The previous response states that this item is isolated from the CCW system at this time.

Response:

The following information supplements the response submitted October 26, 1983.

The maximum heat load which would exist in the Unit 1 spent fuel pool concurrent with a LOCA would be 18.20 MBTU/hr. The value of 38.34 MBTU/hr given in FSAR Table 9.1.3-1A is reduced by 20.12 MBTU/hr. This reduction exists because a LOCA on Unit 1 would not be concurrent with a complete Unit 1 core unload to the spent fuel pool.

With this load, the amount of CCW flow required to maintain the fuel pool temperature less than 150°F is less than 3500 gpm. One train of CCW has sufficient capacity to carry the heat loads from the applicable RHR pump (5 gpm) and RHR heat exchanger (5600 gpm). This leaves 3545 gpm available to the spent fuel pool heat exchanger. The time available to manually reconnect the spent fuel pool heat exchanger prior to reaching 150°F in the spent fuel pool is approximately 4.5 hours; this assumes a postulated LOCA with loss of one train of CCW as the single failure. The necessary manual actions can be accomplished in this time frame. In addition, the time available from the initial isolation of the spent fuel pool cooling heat exchangers to boiling in the spent fuel pool is approximately 12 hours.

It should also be noted that the SHNPP Spent Fuel Pool heat exchanger can be serviced from either Unit 1 or Unit 2 CCW, therefore, the above described measures are applicable only until Unit 2 is operational.

Shearon Harris Nuclear Power Plant
Draft SER Open Item 139/369
ASB Question 9.3.1
Revised Response

NRC Question:

Commit to periodic testing of instrument air quality in accordance with the requirements of ANSI MC 11.1-1976 (ISA-57.3)

NRC Clarifications:

CP&L's previous response stated that the 3 micron particle size acceptance criteria in ANSI MC 11.1-1976 would not be used at SHNPP. CP&L must provide additional justification for this deviation from the recommendations from the SRP. First CP&L must identify the specific acceptance criteria which would be used to detect introduction of contaminants into the instrument air system via breakdown of the instrument air filters or the desiccant in the air dryers.

Second, CP&L must identify and justify the acceptance criteria for preoperational flushing of the instrument air system. This response must address how air operated valves which are assumed to fail to safe condition will be able to perform in this manner.

Response:

CP&L's commitment to perform testing of the instrument air system was documented in a response dated October 11, 1983. The information provided below is supplementary to that response.

The first NRC staff concern is with regard to the effectiveness of the air filters immediately downstream of the air receivers and the instrument air dryer (refer to Figure 9.3.1-3). The instrument air will be tested downstream of the after filter for particulates at every refueling outage. The acceptance criteria for particulates at this point will be 3 microns. This acceptance criteria is consistent with the recommendations of the SRP 9.3.1. This test will assure that the filters are not degraded.

The second NRC staff concern is with regard to the operability of components which are served by the instrument air system. The components served by the instrument air system fall into the following categories:

a. Components which are assumed to fail-safe on loss of instrument air.

b. Components which are supplied from accumulators in the event of failure of the instrument air system.

c. Components where the consequences of non-random failures (multiple failures) would be bounded by FSAR Chapter 15.0 analysis.

Each of these categories is addressed below.

a. The instrument air system provides air to air-operated valves which are assumed to fail-safe on loss of instrument air; single failures of such valves are considered in accident analyses. These safety-related valves will be provided with filters immediately upstream of the component. The filter size will be in accordance with manufacturers' recommendations. This filtration will provide protection from particulates which may develop in the supply piping.

   The filters will be subject to surveillance during plant operation and preventive maintenance changeout of the filters in accordance with vendor recommendations and plant experience. Since the devices on the air system do not require high volumes of flow, little if any, filter degradation is expected. If surveillance of the filters indicates degradation surveillance will be expanded. In addition, valves in ASME Code Class 1, 2, & 3 systems would be subject to the quarterly testing requirements of Section XI of the ASME Code and/or response testing in accordance with Appendix J to 10 CFR Part 50. These measures will be adequate to assure the proper functioning of air operated valves.

b. Components which are supplied with accumulators include the pressurizer PORVs, containment hydrogen purge valves, and containment vacuum relief valves, which require a motive force to actuate to a safety function. These valves will be provided with filtration described above.

c. The instrument air system serves many items in the plant whose operability has no direct impact on accident analyses or where, non-random, multiple failures are bounded by accident analysis. These items include components in waste processing systems, water treatment systems, and portions of the main steam, condensate, and feedwater systems which are isolated by the main steam isolation valves or the main feedwater isolation valves. CP&L finds that a regulatory commitment with regard to instrument air quality for these items is not required.

CP&L finds that a commitment to test instrument air immediately upstream of component level filters would be of no real benefit because (1) testing of air entering the instrument air header is a representative test point for air quality in the entire system since this point is downstream of the major air quality control components (refer to (a) above); (2) provisions have been made for filtration at the component level; and (3) surveillance on valves required to function or failsafe provide assurance that components will function as assumed in the FSAR analyses.

Shearon Harris Nuclear Power Plant
Draft SER Open Item No. 31

Provide the itemized documentation required by Item II.F.2 of NUREG-0737.

Response:

The following information describes the instrumentation utilized for monitoring ICC and is organized per NUREG-0737, Item II.F.2, "Documentation Required."

(1) Information utilized to give the operator an advance warning of the approach to ICC and to monitor the recovery from ICC, if it occurs, is obtained via a qualified instrumentation package. The information is obtained by the use of the Reactor Vessel Level Indicating System (RVLIS) and incore exit thermocouples.

(a) The Westinghouse RVLIS being installed at SHNPP represents the most recent Westinghouse design. It is a fully qualified and redundant system for monitoring water inventory in the reactor vessel. Each of the two channels provide differential pressure cells and transmitters for narrow and wide range monitoring over the full length of the vessel, with the reactor coolant pumps off (natural circulation) and on, respectively. Additionally, narrow range monitoring is provided for each channel of the upper plenum during natural circulation. Each channel's microprocessor utilizes these D/P signals in conjunction with other inputs such as RCS pressure, RCS temperature, (loop RTDs or incore thermocouples), RVLIS reference leg temperature sensors, to compensate for density changes in the system reference legs so as to provide direct water level readings available for operator use.

Qualified incore thermocouples are utilized to determine core exit temperature. These 51 thermocouples (26 channel A, 25 channel B) are inputs to and processed by the RVLIS microprocessors. Both RVLIS water level readings and incore exit thermocouple data will be data-linked to the ERFIS computer for primary display on the SPDS CRT which is located on the MCB. The data link is supplied from an isolated non-Class 1E output from the qualified RVLIS microprocessors. Although ERFIS is non-class 1E, it is powered from a high reliability power source. The isolation device cabinets and ERFIS are readily accessible and adjacent to the Main Control Room.

Additionally, qualified microprocessor outputs (RVLIS water level and thermocouple data) will be transmitted to dedicated redundant backup displays. These backup displays are alpha-numeric and qualified (class 1E), and are located in the control room. The primary and backup displays have a selective capability for providing RVLIS water level, thermocouple data, and temperature mapping functions.

The input to the ERFIS computer will also be used to determine the margin of saturation which can be displayed on demand (at operator request) on the SPDS CRT or continuously on a strip chart recorder. The plant computer (ERFIS) processes and calculates subcooling data using temperature and pressure signals from the reactor coolant system. Displayed information includes margin of subcooling data both graphically and in engineering units.

In accordance with the provision of Regulatory Guide 1.97 Rev. 3 operator confirmation of subcooling data is provided through the use of qualified pressure and temperature signals and ASME steam tables.

(b) Existing instrumentation which provides operating information pertinent to ICC considerations consist of the non-safety incore thermocouple system and a digital list of thermocouple temperatures readout. This is being replaced by the system as described in Item (1)(a) above.

(c) Modifications to the instrumentation systems described in Item 1(b) above include upgrading the incore thermocouples, connectors, reference junction boxes RTDs and cables in order to be qualified in accordance with the IEEE 344 (1975) and IEEE 323 (1974), the procurement of a qualified RVLIS, and the procurement of redundant integrated plant process/emergency response computers.

(2) Design analysis and an evaluation of instruments to monitor water level, and available test data to support the design described in Item (1), above may be found in NUREG CR-2628 regarding the Westinghouse RVLIS design and will be available later for the incore exit thermocouple instrumentation.

(3) A description of test programs conducted for evaluation and qualification of the RVLIS was provided in NUREG CR-2628. For qualification of the thermocouples, see Item (4) below.

Although the system sensors and microprocessors are not directly testable at power for calibration, the calculated parameter of margin to saturation can be readily verified at power through use of the steam tables and observation of the independent indications of pressure and temperature. These observations should show higher margin to saturation since the system uses conservatively auctioneered values.

(4) An evaluation on the conformance of ICC instrumentation to Item II.F.2, Attachment 1, and NUREG-0737, Appendix B, is provided in NUREG CR-2628 for the RVLIS. RVLIS meets the intent of Regulatory Guide 1.97.

Technical specifications will be prepared for the instrumentation specifically installed for the detection of inadequate core cooling. The technical specifications will be prepared considering the recommendations of NRC's Standard Technical Specifications (STS) for Westinghouse PWRs (Rev. 4). CP&L is currently reviewing the technical specifications in Chapter 16.0 of the FSAR in view of the recommendations of Revision 4 to the Westinghouse STS; a revision to the technical specifications will be submitted to the NRC in the second quarter of 1984.

The thermocouples meet the intent of design and qualification criteria outlined in II.F.2, Attachment 1, as indicated below:

A.1 Thermocouples utilized for the core exit for each core quadrant (in conjunction with core inlet temperature data) are sufficient to provide indication of radial distribution of the coolant enthalpy (temperature) rise across representative regions of the core.

A.2 The primary display has the following capabilities:

(a) A spatially oriented core map indicting the temperature or temperature difference across the core (at each thermocouple location) is displayed on the CRT.

(b) A selective reading of core exit temperature, which is consistent with parameters pertinent to operator actions in connection with plant-specific inadequate core cooling procedures, will be continuous on demand.

(c) Direct readout and hard copy capability is available for all thermocouple temperatures. The range extends from $200^{o}F$ to $2300^{o}F$.

Hard copy will be provided by computer printout.

(d) Trend capability showing the temperature-time history of representative core exit temperature values is available on demand.

(e) Alarms are provided in the control room. These alarms will be set to be consistent with the decision points in the emergency operating procedures (refer to Items A.4 and (2) below).

(f) The operator display device (CRT) interface will be located in accordance with human-factor design in order to provide rapid access to requested displays. CP&L's human factors methodology for the main control board has been provided to the NRC in a submittal dated June 1, 1983. This document identified the methodologies and human engineering requirement specifications which apply to items such as ICC instrumentation, which were not defined when the Detailed Control Room Design Review was performed.

A.3 A backup display is provided with the capability for selective reading of each of the operable thermocouples. The range extends from $200^{o}F$ to $2300^{o}F$.

The backup display provided, which is in the control room, is described in Item 1 above.

A.4 The types and locations of displays and alarms will take into account the following:

(a) The use of this information by an operator during both normal and abnormal plant conditions

(b) Integration into emergency procedures

(c) Integration into operator training

(d) Other alarms during an emergency and need for prioritization of alarms.

Normal operating and emergency operating procedures are currently being developed and will be available for onsite review six months prior to fuel load (January 1985).

A.5 The instrumentation meets the requirements of Appendix B, "Design and Qualification Criteria for Accident Monitoring Instrumentation," as modified by the provisions of Items (6) through (9) below.

A.6 The primary and backup display channels are electrically independent, energized from independent station Class 1E power sources, and physically separated in accordance with Regulatory Guide 1.75 up to and including the isolation devices. The primary display and associated hardware beyond the isolation device are energized from a high reliability power source. The backup display and associated hardware is Class 1E. Refer to Item 1 above.

A.7 Primary and backup display are located in the control room envelope. Backup display will be completely qualified in accordance with IEEE 323 (1974) and 344 (1975) as defined in WCAP 8587, "Methodology for Qualifying Westinghouse WRD Supplied Safety Related Electrical Equipment" and WCAP 8687, "Equipment Qualification Test Reports." The isolation device is located in an area which is accessible for maintenance following an accident.

A.8 The primary and backup display channels are designed to provide 99% availability for each channel with respect to functional capability to display a minimum of four thermocouples per core quadrant. This can be accomplished since each quadrant will contain a minimum of four thermocouples for each of Train A and Train B. ICC systems will be addressed in the technical specifications.

A.9 Quality assurance meets the requirements of 10 CFR 50 as applicable. This is further addressed in the applicants response to Supplement 1 to NUREG-0737 (Reg. Guide 1.97) dated September 6, 1983.

(5) For a description of the computer functions associated with ICC monitoring, refer to Item (1) above.

(6) ICC instrumentation will be installed and preoperational tests will be completed before fuel load. Startup tests and calibrations which require the core to be in place will be completed prior to operation above 10 percent of full power.

(7) SHNPP Emergency Operating Procedures (EOPs) and Functional Restoration

Procedures (FRPs) will incorporate the Westinghouse Owners' Group Emergency Response Guidelines and Functional Restoration Guidelines. These procedures employ inadequate core cooling (ICC) instrumentation (RVLIS, the core exit thermocouples, and the subcooling data) along with other post-accident monitoring capabilities (i.e., reactor coolant system pressure, reactor coolant pump status, and safety injection flow). Therefore, SHNPP instrumentation for monitoring ICC will be used in accordance with the emergency response guidelines developed by the Westinghouse Owners' Group. The emergency response guidelines were accompanied by extensive analysis of the setpoints used in the critical safety function status tree and the functional restoration guidelines. These analyses are referenced in WOG Revision 1 (High Pressure Plant) Emergency Response Guidelines. SHNPP EOPs and FRPs will be completed by January 1985. The development of these procedures will include details such as the specification of SHNPP-specific setpoints; these setpoints will account for instrumentation uncertainties which are specific to SHNPP equipment. A draft copy of the EOP for ICC will be provided for NRC's information by January 1985.

(8) The SHNPP EOP for ICC will refer the operator to functional restoration procedures based on the readings on the ICC instrumentation. The SHNPP functional restoration procedures will incorporate the Westinghouse Owners' Group Functional Restoration Guidelines C.1, C.2, and C.3. The actions specified for the operator are fully addressed in WOG submittals and are briefly described below.

FRG C.1 This guideline will be used when indicated by the core cooling critical safety function status tree (refer to Attachment 1). The operator actions specified include:

(a) Verify safety injection actuation and flowpath alignment.

(b) Align and actuate systems required to support reactor coolant pump operation.

(c) Monitor containment hydrogen concentration.

(d) Operate pressurizer PORV, if necessary.

(e) Operate steam system PORVs.

(f) Actuate reactor coolant pumps.

The sequence, priority, and action levels for the listed actions are based on ICC instrumentation and other post-accident monitoring capability.

FRG C.2 This guideline will be used when indicated by the core cooling critical safety function status tree (see Attachment 1).

The operator actions specified include:

(a) Verify safety injection actuation and flowpath alignment.

(b) Align and actuate systems required to support reactor coolant pump operation.

(c) Observe trend in inadequate core cooling instrumentation.

(d) Operate steam system PORVs.

The sequence, priority, and action levels for the listed actions are based on ICC instrumentation and other post-accident monitoring capability.

FRG C.3 This guideline will be used when indicated by the core cooling critical safety function status tree (refer to Attachment 1).

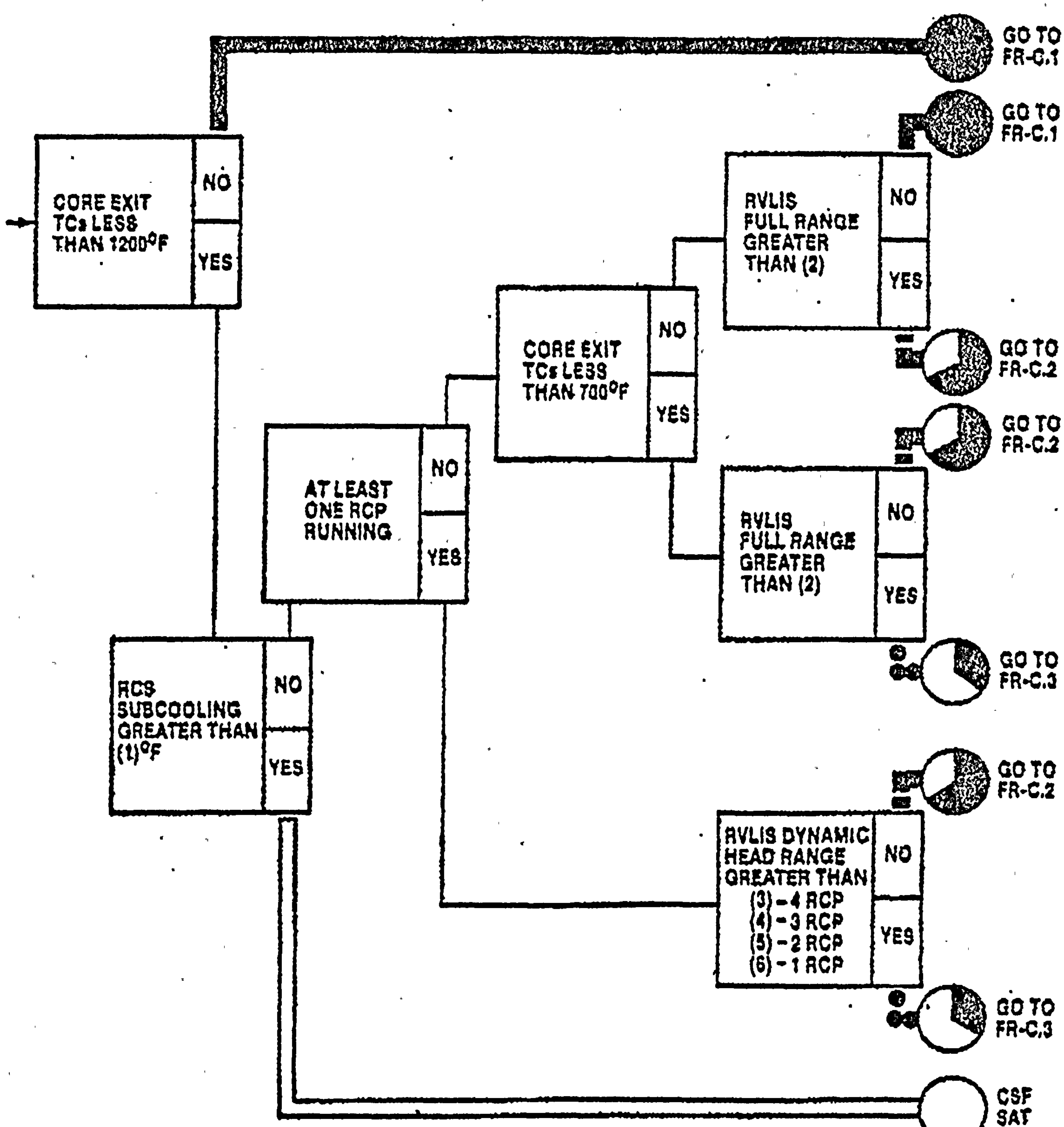
The operator actions for this guideline include:

(a) Verify safety injection actuation and flowpath alignment.

(b) Verify that pressurizer PORVs and reactor vessel head vents are closed.

(9) Additional information to support the acceptability of the ICC monitoring system was provided in the applicant's response to Supplement 1 of NUREG-0737 (Reg. Guide 1.97), dated September 6, 1983. Further information regarding test data for the incore exit thermocouples will be available by March 31, 1984. A draft emergency operating procedure for ICC will be submitted by January 4, 1985. Changes subsequent to the design and operation of the ICC instrumentation as described in the FSAR will be reported to the NRC in accordance with 10 CFR50.59.

Number:	Title:	Rev. Issue/Date:
F-0.2	CORE COOLING	HP/LP, REV. 1 1 Sept., 1983



Number:	Title:	Rev. Issue/Date:
F-0	CRITICAL SAFETY FUNCTION STATUS TREES	HP/LP, REV. 1 1 Sept., 1983

FOOTNOTES

F-0.2 CORE COOLING

(1) Enter sum of temperature and pressure measurement system errors, including allowances for normal channel accuracies and post accident transmitter errors, translated into temperature using saturation tables.

(2) Enter plant specific value which is 3-1/2 feet above the bottom of active fuel in core with zero void fraction, plus uncertainties.

(3) Enter plant specific value corresponding to an average system void fraction of 50 percent with 4 RCPs running, plus uncertainties.

(4) Enter plant specific value corresponding to an average system void fraction of 50 percent with 3 RCPs running, plus uncertainties.

(5) Enter plant specific value corresponding to an average system void fraction of 50 percent with 2 RCPs running, plus uncertainties.

(6) Enter plant specific value corresponding to an average system void fraction of 50 percent with 1 RCP running, plus uncertainties.

F-0.3 HEAT SINK

(1) Enter plant specific value showing SG level just in the narrow range, including allowances for normal channel accuracy, post-accident transmitter errors, and reference leg process errors, not to exceed 50%.

(2) Enter the minimum safeguards AFW flow requirement for heat removal, plus allowances for normal channel accuracy (typically one MD AFW pump capacity at SG design pressure).

(3) Enter plant specific pressure for highest steamline safety valve setpoint.

(4) Enter plant specific value for SG high-high level feedwater isolation setpoint.

(5) Enter plant specific pressure for lowest steamline safety valve setpoint.

Shearon Harris Nuclear Power Plant
DSER Open Item 47
Additional Information

The draft safety evaluation report response to Open Item 47 does not address the following listed concerns. In order for PSB to complete its review, responses to the following information must be provided by the licensee.

1. Describe whether or not positive indication(s) of valve positions are provided in the Main Control Room for Reactor Vessel Head Vent Valves 2RC-V280SB-1 and 2RC-V281SA-1 and Pressurizer Vent Valves 2RC-V282SB-1 and 2RC-V283SA-1 for both open and closed positions.

2. Each vent must have its power supplied from an emergency bus. A single failure within the power and control aspects of the reactor coolant vent system should not prevent isolation of the entire vent system when required. Provide a sketch of the RCS vent system showing power train arrangement and briefly describe how the above requirement is met.

3. A degree of redundancy should be provided by powering different vents from different emergency buses in order to ensure that reactor cooling system (RCS) venting capability from each hot leg high point is maintained after a single failure of an emergency power train. Described your method of accomplishing this requirement.

4. Describe the various measures used to minimize the probability of inadvertent actuation of the RCS vent system valves (e.g. key-lock switches, removing of control power to valves during normal operation, annunciators, administrative procedures, etc.).

5. If control power is removed from the RCS vent valves during normal operation, identify whether or not this action would cause interruption of power to position indication circuitry and thereby defeating the positive indication requirement.

Response:

1) Positive indication of valve positions is provided in the Main Control Room for valves 2RC-V280SB-1, 2RC-V281SA-1, 2RC-V282SB-1 and 2RC-V283SA-1 for both open and closed positions. These valves are solenoid valves. The positive valve position indication is provided by reed switches on the solenoid.

2. The vent system valves are designed such that if power is lost to a valve (or train of valves) the valve fails closed. The valve nomenclature used on Harris indicates the safety train and train power supply associated with safety class valves. As an example (ref. figure submitted in CP&L's August 11, 1983 submittal), valve number 2RC-V282SB-1 indicates the valve is powered from Safety Train B power, likewise valve number 2RC-V281SA-1 indicates the valve is powered from Safety Train A power supply. As shown in the RCS Vent

System Failure Modes and Effects Analysis (ref. CP&L's August 11, 1983 submittal Table 1), a single failure within either power or control would neither prevent the system from performing on demand or prevent isolation of the entire system when required.

3. Upon loss of emergency Power Train A the RV head can be vented via opening valves 2RC-V280SB-1 and 2RC-V285SB-1 to the PRT and the Pressurizer can be vented via opening valves 2RC-V282SB-1 and 2RC-V285SB-1 to the PRT. Upon loss of emergency Power Train B the RV head can be vented via opening valves 2RC-V281SA-1 and 2RC-V284SA-1 and the pressurizer can be vented via opening valves 2RC-V283SA-1 and 2RC-V284SA-1. Separation of Safety Trains is discussed in FSAR Section 8.3.1.2.30.

4. The Vent System is designed such that inadvertent actuation of any single vent system valve will not degrade the system. To vent the RCS through either the Pressurizer or Reactor Vessel Head requires actuation of two separate and independent valves. Inadvertent actuation of two valves is not a credible event. Additionally, the vent system utilizes a 3/8 inch diameter orifice. This orifice size is sufficient to limit flow to less than the make-up capacity of one charging pump.

5. Should control power be removed from the RCS vent valves via a Motor Control Center, indication of valve position would be lost. However control power is removed from the valves during normal operation via a pull to lock switch in the Main Control Room. Valve indication circuitry and thereby, valve indication is not interrupted during normal operation.