UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
CAROLINA POWER & LIGHT COMPANY and NORTH CAROLINA EASTERN	)	Docket Nos. 50-400 OL 50-401 OL
MUNICIPAL POWER AGENCY	)	
(Shearon Harris Nuclear Power Plant, Units 1 and 2)	)	

AFFIDAVIT OF LEONARD I. LOFLIN

County of Wake)
) ss.
State of North Carolina)

LEONARD I. LOFLIN, being duly sworn according to law, deposes and says as follows:

1. I am Manager-Engineering, Shearon Harris Nuclear Power Plant, Carolina Power & Light Company. A summary of my professional qualifications and experience is attached hereto as Exhibit "A". I have personal knowledge of the matters stated herein and believe them to be true and correct. I make this Affidavit in support of Applicants' Motion for Summary Disposition of CHANGE Contention 44 and Eddleman Contention 132.

2. In 1983, CP&L decided that the Westinghouse-designed Reactor Vessel Level Instrumentation System (RVLIS) would be installed at the Harris plant as a part of the enhanced inadequate core cooling instrumentation system responsive to Item II.F.2 of NUREG-0737. The RVLIS being installed at the Harris plant represents the most recent Westinghouse design. It is a fully qualified and redundant system for monitoring water inventory in the reactor vessel.

3. Exhibit "B" to this Affidavit is the documentation

CP&L provided to the NRC Staff in response to Item II.F.2 of NUREG-0737. It describes the Harris plant's instrumentation system, including RLVIS, for inadequate core cooling. Exhibit "C" clarifies and supersedes the information provided in Exhibit B.

Leonard I. Loflin
Leonard I. Loflin

Subscribed and sworn to before me
this 5th day of December, 1983

Deborah R. Hood
Notary Public

My Commission expires: 4-20-86

EXHIBIT A

LEONARD IRA LOFLIN

Manager - Harris Plant Engineering Section

BIRTH DATE: October 16, 1941

I. EDUCATION

A. B.S. Degree in Electrical Engineering from Clemson University - February 1964

B. Professional Degree in Nuclear Engineering from North Carolina State University - June 1969

C. Reactor Operator Training Programs

1. Westinghouse Corporation, Saxton Plant: AEC Reactor Operator License - February 1970

2. Virginia Electric & Power Company, Surry Plant: AEC Senior Reactor Operator License - April 1972

II. EXPERIENCE

A. 1960 to 1963

1. Duke Power Company

a. Three summer work periods at Buck Steam Plant, Spencer, N.C.

b. One summer work period at Greenville, S.C., Distribution Engineering Office

B. February 1964 to June 1973

1. Virginia Electric & Power Company

a. Assistant Engineer, Yorktown Power Plant (two 165 MWe fossil fired units): February 1964 to November 1964

Participated in maintenance and modifications of plant control systems. Responsible for plant performance testing and monitoring.

b. Assistant Engineer: November 1964 to May 1965

Associate Engineer: May 1965 to January 1967

Engineer: January 1967 to May 1967

Mt. Storm Power Plant (two 565 MWe fossil fired units)

Assignment to Mt. Storm was made prior to initial phases of first unit startup. As the only nonsupervisory utility engineer assigned to the plant during startup of both units, was integrally involved in all engineering, operations, and maintenance facets of startup on both units.

c. Engineering Supervisor, Mt. Storm Power Plant: May 1967 to September 1968

Supervisory and technical responsibility for all station engineers, chemists, instrument technicians, laboratory technicians, coal handling foremen, and coal handling union personnel. Handled contract interface and execution on station level between fuel vendors and VEPCO.

d. Staff Engineer, Richmond, VA

September 1968 to June 1969

Assigned to N. C. State University

e. Assistant Operating Supervisor, Surry Nuclear Power Plant (two 2441 MWt Pressurized Water Reactors):

June 1969 to September 1972

Responsible involvement: Core loading; initial criticality; escalation to power: preoperational startup of all plant systems, both secondary and primary; scheduling and organization of operations department; interface relations with Stone & Webster (A/E), Westinghouse, and Atomic Energy Commission; organization and coordination of nuclear training.

f. Operating Supervisor, Surry Nuclear Power Plant (two 2441 MWt Pressurized Water Reactors): September 1972 to June 1973

Responsible for all plant operational functions. Conducted escalation to rated power of Unit 1. Directly supervised core loading, initial criticality, and escalation to power of Unit 2. Personnel responsibility for forty-five (45) operators and eleven (11) first-line supervisors.

C. June 1973 to Present

1. Carolina Power & Light Company

a. Employed as Principal Engineer in the Power Plant Engineering Department. Assigned to Nuclear Plant Engineering Section II as Principal Engineer on the South River Project. Supervised first year of project activity including nuclear steam supply (NSS) negotiation;

creating functioning relationships between CP&L, NSS vendors, and the architect/engineer (A/E); generating plant conceptual design.

b. July 1974 — Assigned as Principal Engineer to the Brunswick Steam Electric Plant as Engineering Start-Up Coordinator. Responsible for identifying engineering areas which were not properly supporting start-up effort and rectifying the situation. Emphasis was given to interfacing with CP&L construction and operation organizations to optimize feedback and execution.

c. August 1975 - Promoted to Manager - Corporate Nuclear Safety Section of the Special Services Department. Transferred from Power Plant Engineering Department.

d. June 1976 - Assigned as Manager - Corporate Nuclear Safety Section in the Technical Services Department.

e. December 1976 - Manager - Corporate Nuclear Safety Section, System Planning & Coordination Department.

f. December 14, 1976 - Transferred to the Power Plant Engineering Department as Manager - Nuclear Plant Engineering Section.

g. January 13, 1977 - Assigned as Manager of Engineering Pool Section of the Power Plant Engineering Department, accountable for designing and engineering safe, economical, constructible, and operable power plants that have as little impact on the natural environment as possible by interacting with necessary project entities and by managing, supervising, developing, training, and motivating and organization of Company employees to carry out such activities.

h. December 1, 1979 - Assigned as Manager - Harris Plant Engineering Section of the Nuclear Power Plant Engineering Department, Shearon Harris Nuclear Power Plant, New Hill, N. C.

III. PROFESSIONAL SOCIETIES

A. ANS

B. P.E. - California - 1976

EXHIBIT B

SERIAL: LAP-83-354

AUG 11 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
DRAFT SAFETY EVALUATION REPORT RESPONSES

Dear Mr. Denton:

Carolina Power & Light Company (CP&L) hereby transmits one original and forty copies of responses to Shearon Harris Nuclear Power Plant Draft Safety Evaluation Report Open Items. The response numbers are listed on the cover page of the attachment along with the corresponding review branch and reviewer for each response.

We will be providing responses to other Open Items in the Draft Safety Evaluation Report shortly.

Yours very truly,

M. A. McDuffie
Senior Vice President
Engineering & Construction

JDK/ccc (7117JDK)

Attachment

cc: Mr. E. A. Licitra (NRC)
Mr. G. F. Maxwell (NRC-SHNPP)
Mr. J. P. O'Reilly (NRC-RII)
Mr. Travis Payne (KUDZU)
Mr. Daniel F. Read (CHANGE/ELP)
Chapel Hill Public Library
Wake County Public Library
Mr. Wells Eddleman
Dr. Phyllis Lotchin
Mr. John D. Runkle
Dr. Richard D. Wilson
Mr. G. O. Bright (ASLB)
Dr. J. H. Carpenter (ASLB)
Mr. J. L. Kelley (ASLB)

Shearon Harris Nuclear Power Plant
Draft Safety Evaluation Report (DSER)
Core Performance Branch
Open Item 31 (DSER Section 4.4.8, pages 4-47, 4-49, & 4-50)

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 core exit thermocouples.

The RVLIS 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 RTD's or incore thermocouples), and 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. These water level readings will be displayed by redundant, qualified, alpha/numeric displays in the control room.

Incore thermocouples are utilized to determine core exit temperature. These 51 thermocouples outputs will be data linked to the ERFIS computer system for primary display on the SPDS CRT located on the MCB. Additionally, the thermocouple outputs are transmitted to a back up display in the control room. The display has a number of switches for selection based upon thermocouple identification and location core map.

The input to the ERFIS computer will also be used to determine the margin of saturation which can be displayed on demand on the SPDS CRT or continuously on a strip chart recorder.

The operator can also use Main Control Room display information in conjunction with steam tables to determine the margin to saturation.

Response (Cont'd)

(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 design (RVLIS).

(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 recommendation of NRC's Standard Technical Specifications (STS's) for Westinghouse PWR's (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 of the Westinghouse STS; a revision of the Technical Specifications will be submitted to the NRC in the second quarter of 1984.

The thermocouples meet the intent of design 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 indicating the temperature or temperature difference across the core (at each thermocouple location) is displayed on the CRT.

Response (Cont'd)

(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 displayed on demand.

(c) Direct readout and hard copy capability is available for all thermocouple temperatures. The range extends from 200°F to 2300°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 consistent with operator procedure requirements.

(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.

A.3 A backup display is provided with the capability for selective reading of each of the operable thermocouples.

Backup display is provided in the control room.

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

(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; and

(d) other alarms during emergencies and the need for prioritization of alarms

Normal operating and emergency operating procedures are currently being developed and will be submitted at a later date. They will be available for onsite review six months prior to fuel load.

Response (Cont'd)

A.5 The RVLIS 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.

A.7 Primary and Backup displays 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 8587/8687. 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 minumum of four thermocouples per core quadrant.

A.9 Quality assurance meets the requirements of 10CFR50 as applicable.

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

(6) ICC Instrumentation, including associated testing, and calibration, will be installed before plant operation.

(7) Guidelines for the use of instrumentation for monitoring ICC and the basis for these procedures will be provided after their completion.

(8) A description of key operator action instructions in the emergency procedures for ICC will be provided when available.

(9) Additional information to support the acceptability of the ICC monitoring system will be provided by August 30, 1983 in the applicants response to Supplement 1 of NUREG 0737 (item R.G.1.97 Rev. 2)

(7292PSAtda)

EXHIBIT C

CP&L

Carolina Power & Light Company

SERIAL: LAP-83-521

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)
Mr. G.F. Maxwell (NRC-SHNPP)
Mr. J. P. O'Reilly (NRC-RII)
Mr. Travis Payne (KUDZU)
Mr. Daniel F. Read (CHANGE/ELP)
Mr. R. P. Gruber (NCUC)
Chapel Hill Public Library
Wake County Public Library
Mr. Wells Eddleman
Dr. Phyllis Lotchin
Mr. John D. Runkle
Dr. Richard D. Wilson
Mr. G. O. Bright (ASLB)
Dr. J. H. Carpenter (ASLB)
Mr. J. L. Kelley (ASLB)

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°F to 2300°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°F to 2300°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:

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

IMAGE EVALUATION
TEST TARGET (MT-3)

150mm

6″

(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:

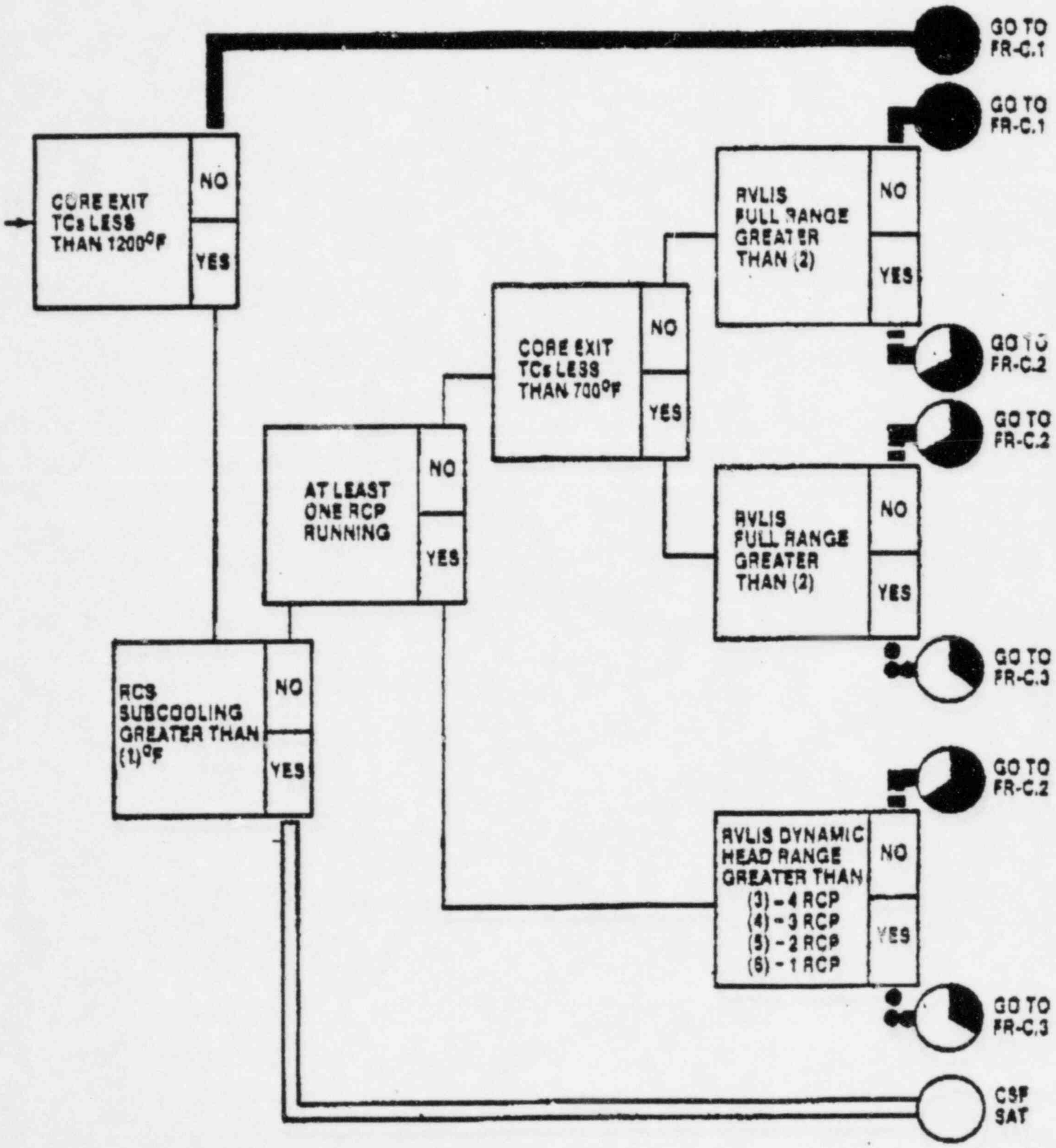
(a) Verify safety injection actuation and flowpath alignment.

(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.

ATTACHMENT 1
PAGE 1

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



Number: F-0	Title: CRITICAL SAFETY FUNCTION STATUS TREES	Rev. Issue/Date: 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.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
CAROLINA POWER & LIGHT COMPANY and NORTH CAROLINA EASTERN MUNICIPAL POWER AGENCY	)	Docket Nos. 50-400 OL 50-401 OL
	)	
(Shearon Harris Nuclear Power Plant, Units 1 and 2)	)	

CERTIFICATE OF SERVICE

I hereby certify that copies of "Applicants' Motion for Summary Disposition of CHANGE Contention 44 and Eddleman Contention 132," "Applicants' Statement of Material Facts as to Which There is No Genuine Issue to be Heard (CHANGE Contention 44 and Eddleman Contention 132)," "Affidavit of Clarence G. Draughon" and "Affidavit of Leonard I. Loflin" were served this 7th day of December, 1983, by deposit in the U.S. mail, first class, postage prepaid, to the parties on the attached Service List.

Thomas A. Baxter
Thomas A. Baxter, P.C.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
CAROLINA POWER & LIGHT COMPANY and NORTH CAROLINA EASTERN MUNICIPAL POWER AGENCY	)	Docket Nos. 50-400 OL 50-401 OL
	)	
(Shearon Harris Nuclear Power Plant, Units 1 and 2)	)	

SERVIC LIST

James L. Kelley, Esquire
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Mr. Glenn O. Bright
Atomic Safety and Licensing Board
U.S. Nucle.. Regulatory Commission
Washington, D.C. 20555

Dr. James H. Carpenter
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Charles A. Barth, Esquire
Janice E. Moore, Esquire
Office of Executive Legal Director
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Docketing and Service Section
Office of the Secretary
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Mr. Daniel F. Read, President
CHANGE/ELP
5707 Waycross Street
Raleigh, North Carolina 27606

John D. Runkle, Esquire
Conservation Council of North Carolina
307 Granville Road
Chapel Hill, North Carolina 27514

M. Travis Payne, Esquire
Edelstein and Payne
P.O. Box 12607
Raleigh, North Carolina 27605

Dr. Richard D. Wilson
729 Hunter Street
Apex, North Carolina 27502

Mr. Wells Eddleman
718-A Iredell Street
Durham, North Carolina 27705

Richard E. Jones, Esquire
Vice President and Senior Counsel
Carolina Power & Light Company
P.O. Box 1551
Raleigh, North Carolina 27602

Dr. Phyllis Lotchin
108 Bridle Run
Chapel Hill, North Carolina 27514

Dr. Linda W. Little
Governor's Waste Management Board
513 Albemarle Building
325 North Salisbury Street
Raleigh, North Carolina 27611

Bradley W. Jones, Esquire
U.S. Nuclear Regulatory Commission
Region II
101 Marrietta Street
Atlanta, Georgia 30303

Ruthanne G. Miller, Esquire
Atomic Safety and Licensing Board Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Mr. Robert P. Gruber
Executive Director
Public Staff - NCUC
P.O. Box 991
Raleigh, North Carolina 27602