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AUTH.NAME AUTHOR AFFILIATION
CUTTER,A.B. Carolina Power & Light Co.
RECIP.NAME RECIPIENT AFFILIATION
EISENHUT,D.G. Division of Licensing

SUBJECT: Forwards response to Generic Ltr 83-28 re required actions based on generic implications of Salem ATWS events.Util participating in NUTAC re Section 2.2.2 on vendor interference.

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Carolina Power & Light Company

SERIAL: LAP-83-517

NOV 7 1983

Mr. Darrell G. Eisenhut, Director
Division of Licensing
United States Nuclear Regulatory Commission
Washington, DC 20555

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261
LICENSE NO. DPR-23
GENERIC IMPLICATIONS OF
SALEM ATWS EVENTS

Dear Mr. Eisenhut:

Carolina Power & Light Company (CP&L) has received your letter, Generic Letter 83-28, Required Actions Based on Generic Implications of Salem ATWS Events, dated July 8, 1983, and hereby provides its response for the H. B. Robinson Steam Electric Plant (HBR2). Your letter requested CP&L to furnish "the status of current conformance with the positions contained" in your letter, "and plans and schedules for any needed improvements for conformance with the positions." The enclosed response is prepared in a format which addresses each of the NRC positions by the sequential number provided. In August 1983, a project engineer was assigned from CP&L's corporate office to coordinate the responses of all of our nuclear plants.

Representatives of CP&L are participating in the Nuclear Utility Task Action Committee (NUTAC), sponsored by the Institute of Nuclear Power Operations (INPO), concerning Section 2.2.2 of your letter, Vendor Interface; and in the Westinghouse Owners' Group's (WOG) generic evaluation of the positions stated in your letter. As stated in our response to Section 2.2.2, CP&L will review the results of the NUTAC and incorporate the recommendations as appropriate.

The plant procedures discussed in our response are those currently in effect. Future revisions to these procedures may be necessary and will be incorporated as appropriate.

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Pursuant to Generic Letter 82-14, CP&L hereby transmits one original and forty copies of our response to Generic Letter 83-28.

If you have any questions regarding our response, please do not hesitate to call a member of our Licensing staff.

Yours very truly,

A. B. Cutter
Vice President
Nuclear Engineering & Licensing

DCW/ccc (8371DCWb)
Attachment

cc: Mr. J. P. O'Reilly (NRC-RII)
Mr. G. Requa (NRC)
Mr. Steve Weise (NRC-HBR)

A. B. Cutter, having been first duly sworn, did depose and say that the information contained herein is true and correct to the best of his information, knowledge and belief; and the sources of his information are officers, employees, contractors, and agents of Carolina Power & Light Company.

Franklin Murray
Notary (Seal)

FRANKLIN MURRAY
NOTARY PUBLIC
WAKE COUNTY, N.C.

My commission expires: October 4, 1986

ROBINSON NUCLEAR PROJECT

RESPONSE TO GENERIC LETTER 83-28

"REQUIRED ACTIONS BASED ON GENERIC IMPLICATIONS OF SALEM ATWS EVENTS"

NOVEMBER 4, 1983

(8434DCWccc)

1.1 POST-TRIP REVIEW (PROGRAM DESCRIPTION AND PROCEDURE)

NRC POSITION

Licensees and applicants shall describe their program for ensuring that unscheduled reactor shutdowns are analyzed and that a determination is made that the plant can be restarted safely. A report describing the program for review and analysis of such unscheduled reactor shutdowns should include, as a minimum.

CP&L RESPONSE

The procedure governing the criteria for Post-Trip Review for the H. B. Robinson Plant Unit 2 (HBR 2) is incorporated in the Plant Operating Manual (POM): Post Trip/Safeguards Review. This document establishes a formal plant instruction whereby plant incidents are properly investigated prior to restart to ensure 1) causes of the event are understood and properly documented; 2) incidents have not resulted in an unsafe or unacceptable condition for continued operation; 3) appropriate corrective action is provided to prevent recurrence, and 4) continued unanalyzed conditions do not exist. Included in this procedure are the criteria for determining the acceptability for restart, responsibilities of personnel involved in the review, definitions of plant incidents, and a sequence of events instruction. The following is a description of the program and procedure pertinent to the items contained in NRC Generic Letter 83-28.

NRC POSITION

1.1.1. The criteria for determining the acceptability of restart.

CP&L RESPONSE

The Post Trip/Safeguards Authorization is written as such: "The Unit 2 Shift Foreman may recommend restart of the Unit if the cause of the event has been clearly identified and corrected and all of the protection/safeguards equipment or systems functioned as designed. The approval of the Unit 2 Operating Supervisor, or Manager—Operations and Maintenance, or the Plant General Manager (or the individual who is designated Acting Plant General Manager) is required prior to criticality. If the cause of the event has not been clearly identified or there are questions concerning the proper performance of protection/safeguards equipment or systems during the event, the Post Trip/Safeguards Review Report will be reviewed by the Plant Nuclear Safety Committee (PNSC). The PNSC shall make recommendations on the restart of the Unit. Upon completion of any additional corrective actions and the review thereof, the PNSC Chairman may authorize the restart of the Unit." In the event of a disagreement, the recommendation of the PNSC and the actions contemplated by the General Manager, the course determined by the General Manager to be more conservative will be followed.

NRC POSITION

1.1.2. The responsibilities and authorities of personnel who will perform the review and analysis of these events.

CP&L Response

Responsibilities and authorities of personnel who will perform the review and analysis of these events at HBR2 are as follows:

1. Unit 2 Shift Foreman

   (a) ensures the Post Trip/Safeguards Review Report is complete
   (b) provides central direction for the investigation of the event
   (c) recommends restart or forwards Post Trip/Safeguards Review report to the PNSC

2. Shift Technical Advisor

   (a) assist and advise the Unit 2 Shift Foreman in the completion of the Post Trip/Safeguards Review Report

3. Unit 2 Operating Supervisor (or Designated Alternate)

   (a) reviews the Post Trip/Safeguards Review Report
   (b) approves restart, if appropriate

4. Manager - Operations and Maintenance (or Designated Alternate)

   (a) reviews Post Trip/Safeguards Review Report
   (b) approves restart, if appropriate

5. Plant General Manager (or Designated Alternate)

   (a) reviews Post Trip/Safeguards Review Report
   (b) approves restart, if appropriate

6. Plant Nuclear Safety Committee (PNSC)(1)

   *(a) reviews the Post Trip/Safeguards Review Report
   (b) reviews the Post Trip/Safeguards Report prior to restart, if applicable
   (c) recommends restart, if appropriate

   *The Post Trip/Safeguards Review Report is to be reviewed by the PNSC at its next monthly meeting or sooner if deemed necessary by the Plant General Manager.

   NOTE: (1) The PNSC is currently composed of the following:

   Chairman - General Manager or designated alternate
   Member - Manager - Operations & Maintenance or designated alternate
   Member - Assistant to the General Manager
   Member - Manager - Technical Support or designated alternate
   Member - Manager - Environmental & Radiation Control or designated alternate
   Member - Director - QA/QC or designated alternate

7. PNSC Chairman

(a) approves the Post Trip/Safeguards Review Report
(b) approves restart, if appropriate

NRC POSITION

1.1.3. The necessary qualifications and training for the responsible personnel.

CP&L Response

The necessary qualifications for the responsible personnel who prepare and perform the Post Trip/Safeguards Review are addressed in the Plant Operating Manual (POM). The training of the licensed personnel is covered by HBR2 Training Instructions. The Administrative Procedures cover qualifications of non-licensed personnel. The detailed qualifications and training of the Shift Technical Advisor (STA) was addressed in our responses dated December 15, 1980 and December 31, 1980 to NUREG-0737; Item I.A.1.1 entitled, "STA Long Term Requirements". The training and requalification programs for licensed operators were addressed in our responses dated July 11, 1980, May 18, 1982 and March 17, 1983, to NUREG-0737, Item I.A.2.1.4, entitled, "Immediate Upgrading of Reactor Operator and Senior Reactor Operator Training and Qualifications - Modified Training". These submittals have been reviewed and approved by your staff on I.A.1.1 (SER dated April 18, 1983) and I.A.2.1.4 (IE Approved - January 19, 1981). All members of the PNSC have extensive plant experience as required by the H. B. Robinson Technical Specifications for PNSC members. The procedure and training for use of the Post Trip/Safeguards Review Procedure have been implemented.

NRC POSITION

1.1.4. The sources of plant information necessary to conduct the review and analysis. The sources of information should include the measures and equipment that provide the necessary detail and type of information to reconstruct the event accurately and in sufficient detail for proper understanding. (See Action 1.2)

CP&L RESPONSE

The sources of plant information necessary to conduct the review and analysis are specifically delineated in the Post Trip/Safeguards Review Procedure. The Data Collection Section (Plant information) of the procedure is provided to quickly obtain plant data necessary for evaluation of the event. In addition, the Data Collection Section is subdivided into four (4) areas as follows:

1) Reactor Trip Automatic Action Verification

This section provides a verification for certain automatic trip actions. The event is reviewed to determine which of the following occurred: the reactor trip function occurred automatically, or by operator action (manual), or the reactor trip did not function as designed. These reactor trip verifications are conducted on the following items:

a. Reactor Trip Breaker A Trip
b. Reactor Trip Breaker B Trip
c. Reactor Trip Bypass Breaker A Trip
d. Reactor Trip Bypass Breaker B Trip
e. Turbine Trip (all required turbine valves shut)
f. Feedwater Regulator Valves close when Tavg decreases to 533.5°F.

In addition, the First Out Annunciator indications received during the event are recorded, as an aid to review and evaluate the event.

2) Safeguards Automatic Action Verification

This section provides verification for safeguards equipment actuation. The event is reviewed to determine the following: if the safeguards actuation function occurred automatically, or by operator action (manual), or if the safeguards equipment did not function as designed. This verification of the safeguards equipment is completed only in the event a safeguards actuation occurs. The safeguards automatic action verification is conducted on the following equipment:

a. Safety Injection Pump A
b. Safety Injection Pump B
c. Safety Injection Pump C
d. RHR Pump A
e. RHR Pump B
f. Service Water Pump A
g. Service Water Pump B
h. Service Water Pump C
i. Service Water Pump D
j. Service Water Booster Pump A
k. Service Water Booster Pump B
l. Containment Fan HVH-1
m. Containment Fan HVH-2
n. Containment Fan HVH-3
o. Containment Fan HVH-4
p. Auxiliary Feedwater Pump A
q. Auxiliary Feedwater Pump B
r. Emergency Diesel Generator A
s. Emergency Diesel Generator B
t. Isolation Valve Seal Water System
u. Control Room Ventilization Isolation System
v. Containment Spray Pump A
x. Containment Spray Pump B
y. Steam Line Isolation
   (A, B, C Main Steam Isolation Valves Closed)

In addition, the Safeguards Status Panel (Valve Status) is reviewed. This status panel provides verification that the necessary safeguard valves actuated to the correct position for the signals received during an event (i.e., valves in Safety Injection, Containment Spray, Phase A Isolation, etc.)

3) Strip Chart Data

The Strip Chart Data to be recorded includes the value of the variable immediately prior to the event, the maximum and mimimum value of the variable during the event, and the value of the variable after the event has stabilized. Each strip chart is to be identified with the start and stop time of the event, date, and initials. In addition, any unexpected strip chart response is recorded on the data form. Strip Chart Data are obtained on the following items:

a. S/G A Narrow Range Level
b. S/G B Narrow Range Level
c. S/G C Narrow Range Level
d. Pressurizer Level
e. Pressurizer Pressure
f. T - Avg.
g. Loop 1 Hot Leg Temp.
h. Loop 2 Hot Leg Temp.
i. Loop 3 Hot Leg Temp.
j. Loop 1 Cold Leg Temp.
k. Loop 2 Cold Leg Temp.
l. Loop 3 Cold Leg Temp.
m. Nuclear Instrumentation System (N.I.S.) Recorder N-45
n. S/G A Wide Range Level
o. S/G B Wide Range Level
p. S/G C Wide Range Level
q. RCS Wide Range Pressure
r. Others - relevent to the event

4) Sequence of Events Summary

The Sequence of Events Summary form is to be completed only if the sequence of events printout from the plant computer is available. This form is used to verify the proper operation of the reactor trip breakers and the turbine trip function using the sequence of events data.

5) Event Summary Attachment

Event summaries are to be obtained from individuals who were present when the event occurred and who were actively involved in the cause and mitigation of the event. These summaries may be obtained through interviews conducted by the individual(s) investigating the event.

CP&L believes that the Post Trip/Safeguards Review procedure identifies sufficient information to reconstruct the event accurately during subsequent reviews.

NRC POSITION

1.1.5 The methods and criteria for comparing the event information with known or expected plant behavior (e.g., that safety-related equipment operates as required by the Technical Specifications or other performance specifications related to the safety function).

CP&L RESPONSE

The guidelines for comparing the event information with known or expected plant behavior are incorporated in the Post Trip/Safeguards Review procedure. The section in the procedure entitled "Evaluation of Trip/Safeguards Activation" provides the method to analyze the information obtained during data collections to ensure the reactor protection/safeguards equipment and First Out Annunciator panel operated properly. The following guidance is provided on how to use the data gathered during the event:

a. the Sequence of Events Section provides for the comparison of the First Out Annunicator and Sequence of Events Printout (if available) to identify any discrepancies,
b. additional First Out Annunicators section provides for the evaluation of proper Operation of the First Out Annunicator Panel and for the identification of other potential equipment problems,
c. Strip Chart Data section provides for evaluation of Strip Chart data against an expected response based on operator experience and training,
d. detail Summary of the Event section provides a summary of the event based upon all of the available information,
e. Cause of the Event section provides for establishing the cause of the Event and written justification for that cause,
f. the section for Malfunction of Equipment provides for establishing the cause of any protection/safeguards equipment malfunction or First Out Annunciator malfunctions and,
g. the Section for Others provides for listing of other equipment malfunctions which, if it had functioned properly, would have helped mitigate the event.

This information, obtained from the Post Trip/Safeguards Review Procedure, provides the necessary information (cause of the event, event summary, etc.) to evaluate and analyze the event. This event information can be evaluated against the initial plant conditions obtained prior to the event, and/or start up test data, and/or past transient occurrences, and/or other pertinent information that may pertain to the event. This evaluation and analysis is conducted by the appropriate personnel, as addressed in the response to Section 1.1.2.

CP&L believes this system provides sufficient methods and criteria for comparing event information with known or expected plant behavior.

NRC POSITION

1.1.6 The criteria for determining the need for independent assessment of an event (e.g., a case in which the cause of the event cannot be positively identified, a competent group such as the Plant Operations Review Committee, will be consulted prior to authorizing restart) and guidelines on the preservation of physical evidence (both hardware and software) to support independent analysis of the event.

CP&L Response

The criteria for determining the need for independent assessment of an event was discussed in response to Section 1.1.1. All information which is documented in the Post Trip/Safeguards Review procedure is maintained as a permanent plant QA record.

NRC POSITION

1.1.7 Items 1 through 6 above are considered to be the basis for the establishment of a systematic method to assess unscheduled reactor shutdowns. The systematic safety assessment procedures compiled from the above items, which are to be used in conducting the evaluation, should be in the report.

CP&L Response

The systematic safety assessment methodology described in the previous sections comprise the Post Trip/Safeguards Review Procedure. CP&L believes this procedure provides sufficient detail, documentation and accountability for determining the cause of an event and fully meets the criteria specified in Section 1.1, Post Trip Review.

1.2 POST-TRIP REVIEW - DATA AND INFORMATION CAPABILITY

NRC POSITION

Licensees and applicants shall have or have planned a capability to record, recall and display data and information to permit diagnosing the causes of unscheduled reactor shutdowns prior to restart and for ascertaining the proper functioning of safety-related equipment.

Adequate data and information shall be provided to correctly diagnose the cause of unscheduled reactor shutdowns and the proper functioning of safety-related equipment during these events using systematic safety assessment procedures (Action 1.1). The data and information shall be displayed in a form that permits ease of assimilation and analysis by persons trained in the use of systematic safety assessment procedures.

A report shall be prepared which describes and justifies the adequacy of equipment for diagnosing an unscheduled reactor shutdown. The report shall describe as a minimum:

1.2.1. Capability for assessing sequence of events (on-off indications)

1.2.1.1. Brief description of equipment (e.g., plant computer, dedicated computer, strip chart).

CP&L Response

The equipment available at HBR2 for assessing the Sequence of Events (SOE) are:

1. Plant Process Computer (Westinghouse P-250)

   The purpose of the plant computer SOE program is to record the order of contact status changes for selected reactor trip parameters in the plant. The operator can use this program, (when the computer is available), to verify the First Out Annuciator and separate in time a series of events.

2. Reactor Turbine Generator Board (RTGB) Bistable Status Panels

   The Bistable Status Panels are located on the RTGB in the HBR2 Control Room. These bistables can be observed by the control operators during the course of an event to determine the status of reactor protection and safeguards systems bistables.

3. RTGB Safeguards Status Panels

   The Safeguards Status Panels are located on the RTGB, in the HBR2 Control Room. These status panels may be observed by the control operators during the course of an event to verify the primary valve positions and selected damper positions associated with the parameters described in 1.2.1.2

4. RTGB Equipment ON/OFF Indicators

Major Equipment ON/OFF Indicators are located on the RTGB, in the HBR2 Control Room. These equipment indicators are observed by the control operators during normal plant conditions and during the course of an event. The operating conditions of pump/motors, valves/actuators and breakers (open/close) may be established using the ON/OFF indication.

NRC POSITION

1.2.1.2. Parameters Monitored

CP&L Response

1. Plant Process Computer

The SOE Program monitors the following parameters that initiate a reactor trip.

1. Overtemperature ΔT Reactor Trip.
2. Overpower ΔT Reactor Trip.
3. Power Range High Neutron Flux Reactor Trip (High Setting).
4. Pressurizer High Pressure Reactor Trip.
5. Stm/FW Flow Mismatch with Low S.G. Level.
6. Low-Low Steam Generator Water Level Reactor Trip.
7. a. Low Primary Coolant Flow Reactor Trip.
   b. Reactor Coolant Pump Breakers Open Reactor Trip.
   c. Reactor Coolant Pump Buss Undervoltage Reactor Trip.
   d. Reactor Coolant Pump Buss Underfrequency Reactor Trip.
8. Pressurizer Low Pressure Reactor Trip.
9. Pressurizer High Water Level Reactor Trip.
10. Turbine Trip Reactor Trip.
11. Safety Injection System Actuation Reactor Trip.
12. Manual Reactor Trip.
13. Source Range High Neutron Flux Reactor Trip.
14. Intermediate Range High Neutron Flux Reactor Trip.
15. Power Range High Neutron Flux Reactor Trip (Low Setting).

2. RTGB Bistable Status Panels

The Bistable Status Panels monitor the trip/not tripped condition of each reactor protection system bistable. The parameters that initiate the reactor trip (as indicated on plant process computer) are monitored on the bistables. These bistables may be visually checked during and after the event to verify the cause of the reactor trip and, to help ensure that the proper equipment has activated for the given condition. These bistables are also monitored by the plant process computer.

3. RTGB Safeguards Status Panels

The Safeguards Status Panels monitor the various valve and damper positions of the following safeguards system:.

1. Safety Injection
2. Containment Isolation (Phase A and B)
3. Isolation Valve Seal Water System
4. Containment Ventilation Isolation
5. Control Room Intake Duct Isolation

4. RTGB Equipment ON/OFF Indicators

The Equipment ON/OFF Indicators monitors various reactor protection and safeguards equipment on the RTGB. These indicators may be used to verify auto start of safeguards equipment. Some ON/OFF indication has breaker indication which may be helpful in verifying proper equipment operation during a safeguards actuation.

NRC POSITION

1.2.1.3. Time discrimination between events.

CP&L RESPONSE

1. Plant Process Computer - 20 milliseconds resolution
2. RTGB Bistable Status Panels - Monitors event on real time bases.
3. Safeguards Status Panels - Monitors event on real time bases.
4. RTGB Equipment ON/OFF Indicators - Monitors event on real time bases.

NRC POSITION

1.2.1.4. Format for displaying data and information.

CP&L RESPONSE

1. Plant Process Computer

The printout occurs automatically on the Alarm Typewriter and includes the time (hours, minutes, seconds) of the first event, identity of the first event, all subsequent occurrences of selected plant parameter changes, and an indication of the computer time in cycles (one cycle = 0.02 sec) between the events.

2. RTGB Bistable Status Panels

The bistable conditions are indicated on the bistable window and in an event the window is lit when the bistable trips. The control operator reviews these panels during the course of an event to determine the status of inputs to the reactor protection and safeguards systems.

3. RTGB Safeguards Status Panels

In an event, the colored lights (blue window, turned pink indicates valve and/or damper is in the safeguards position) indicate if the equipment has initiated to the safeguards position. The control operator reviews these panels during normal plant conditions (since some are normally in the safeguards position) and during the course of an event to determine the status of safeguards system valves and dampers.

4. RTGB Equipment ON/OFF Indicators

The ON/OFF indication is in the form of red (for on or open) and green (for off or closed) lights associated with each valve or piece of equipment. Some breaker indication is provided by white "breaker trip" lights with the respective ON/OFF light. In other cases, breaker trip is indicated by the ON/OFF lights being extinguished.

NRC POSITION

1.2.1.5. Capability for retention of data and information.

CP&L RESPONSE

1. Plant Process Computer

The SOE program has the capability to store event sequences for three (3) separate events without the loss of SOE information. The oldest SOE event is then overwritten when a fourth event occurs. As noted previously, each SOE event log is automatically printed out on the Alarm Typewriter after each event. The printout is retained and may be used as an aid to evaluate the Post Trip/Safeguards Review.

2. RTGB Bistable Status Panel

The bistable information is not retained except when the control operator logs the information in the Operator Log Book. However, it should be noted that all bistable trip signals which result in trips, are shown on the SOE print-out in terms of the trip functions and/or safeguards function generated.

3. RTGB Safeguards Status Panels

The safeguard status information is not retained except when the control operator logs the information in the operator log book. The appropriate emergency instruction requires this panel be checked to verify the valves and dampers went to the correct position. In addition, the Post Trip/Safeguards Review Report requires a review of this panel and documents valves/dampers which did not function properly.

4. RTGB Equipment ON/OFF Indicators

The Equipment ON/OFF Indicators information is not retained except when the control operator logs the information in the operator log book. However, it should be noted, the ON/OFF indications may be retained in logs in the form of statements such as "all safeguards equipment started" or "the following safeguards equipment failed to start", etc. In addition, the reactor protection/ safeguards equipment operation is documented by the Post Trip/Safeguards Review Report as described in response to Section 1.1

NRC POSITION

1.2.1.6. Power source(s) (e.g., Class-IE, non-Class IE, non-interruptable)

CP&L RESPONSE

1. Plant Process Computer

Power for the plant process computer is supplied by vital DC power. Detailed information and an electrical one line diagram is provided in our updated FSAR, Section 8.1.2.5.

2. RTGB Bistable Status Panels

Power for the Bistable Status Panels is supplied from various instrument busses. Detailed information pertaining to the instrument busses is provided in our Updated FSAR Section 8.1.2.6 and Table 8.3.1-2.

3. RTGB Safeguards Status Panels

Power for the Safeguards Status Panels is supplied from vital AC and DC power. Detailed information pertaining to these power supplies is provided in our Updated FSAR Section 8.1.

4. RTGB Equipment ON/OFF Indicators

Power for the Equipment Indicators is supplied by vital DC power. Detailed information and an electrical line diagram is provided in our Updated FSAR, Section 8.1.2.5.

NRC POSITION

1.2.2. Capability for assessing the time history of analog variables needed to determine the cause of unscheduled reactor shutdown, and the functioning of safety-related equipment.

1.2.2.1. Brief description of equipment (e.g., plant computer, dedicated computer, strip charts).

CP&L RESPONSE

The equipment available at HBR2 for assessing the time history of analog variables needed to determine the cause of unscheduled reactor shutdown and the functioning of safety-related equipment are:

1. Plant Process Computer (Westinghouse P-250)

   The Post Trip Review Program, available from plant process computer, provides a log of the value of many plant variables prior to the event, during the event and after the event has stabilized. The operator thus, can assess the values of variables leading up to a reactor trip and the transient response of parameters after the trip.

2. RTGB Strip Chart Recorders

   The purpose of the Strip Chart recorders provide continous data (except when chart paper is being changed) of selected variables during normal plant operation and during an event.

NRC POSITION

1.2.2.2. Parameters monitored, sampling rate, and basis for selecting parameters and sampling rate.

CP&L RESPONSE

1. Plant Process Computer

   The Post Trip Review program provides a log of the values of many plant equipment parameters including various reactor protection and safeguards initiating parameters. During the original design of this program by Westinghouse, parameters which provide necessary information to evaluate and record the plant condition during an event were selected for use in the program. The sampling rate was also pre-selected by Westinghouse when the computer system was purchased.

2. RTGB Strip Chart Recorder

   The Strip Chart recorder records the value of selected variables on continuous bases as described above. The parameters for the Post Trip/Safeguards Review Program were selected during the original plant design to provide operators with necessary information to evaluate and record the plant conditions. Some strip chart records which are relevant to evaluate an event,

were added after the original plant design to better aid the operators and to improve the data retention capabilities. Each of the strip charts are labeled with the start and stop time of the event, date, and initialed by the appropriate personnel. The parameters which are used to evaluate the plant condition and Post Trip/Safeguards Review are as follows:

	Parameters Monitored Strip Chart Recorder	Scale
1)	S/G A Narrow Range Level	(0 - 100%)
2)	S/G B Narrow Range Level	(0 - 100%)
3)	S/G C Narrow Range Level	(0 - 100%)
4)	Pressurizer Level	(0 - 100%)
5)	Pressurizer Pressure	(1700 - 2500 PSI)
6)	TAVG	(510 - 585^{o}F)
7)	Loop 1 Hot Leg Temp	(0 - 700^{o}F)
8)	Loop 2 Hot Leg Temp	(0 - 700^{o}F)
9)	Loop 3 Hot Leg Temp	(0 - 700^{o}F)
10)	Loop 1 Cold Leg Temp	(0 - 700^{o}F)
11)	Loop 2 Cold Leg Temp	(0 - 700^{o}F)
12)	Loop 3 Cold Leg Temp	(0 - 700^{o}F)
13)	N.I.S. Recorder N-45	(0 - 120%)
	a. Power Range	(0 - 120%)
	b. Intermediate Range	(10^{-11} - 10^{-3} amps)
	c. Source Range	(10^{0} - 10^{6}cps)
14)	S/G A Wide Range Level	(0 - 100%)
15)	S/G B Wide Range Level	(0 - 100%)
16)	S/G C Wide Range Level	(0 - 100%)
17)	RCS Wide Range Pressure	(0 - 3000 psi)
18)	Others - relevent to the event	

NRC POSITION

1.2.2.3. Duration of time history (minutes before trip and minutes after trip)

CP&L RESPONSE

1. Plant Process Computer

The Post Trip Review Program logs the plant parameters, for two (2) minutes before a reactor trip and for three (3) minutes after the reactor trip, at approximately eight (8) second intervals. In addition, there is a special record of several parameters logged for eight (8) seconds before and after the trip at two (2) second intervals.

2. RTGB Strip Chart Recorder

The parameters are recorded on continuous bases. All recorders listed above are single speed recorders (chart speed cannot be changed) except N.I.S. Recorder N-45.

NRC POSITION

1.2.2.4. Format for displaying data including scale (readability) of time histories.

CP&L RESPONSE

1. Plant Process Computer

The output occurs automatically on the Trend Typewriter beginning approximately five (5) minutes after the reactor trip. The time of the reactor trip is indicated in hours, minutes, and seconds. The Post Trip Review Program prints groups of parameters with a data collection time for each group of parameters in minutes, seconds, and tenths of a second.

2. RTGB Strip Chart Recorder

The output occurs continously on the chart paper provided on the recorder. The scale for each recorder is pre-set and is not changed. Chart speed can only be changed for the N.I.S. Recorder N-45.

NRC POSITION

1.2.2.5. Capability for retention of data, information, and physical evidence (both hardware and software)

CP&L RESPONSE

1. Plant Process Computer

The Post Trip Review data from the plant computer is continuously updated to provide information at the time of the event. The output information occurs automatically on the Trend Typewriter (printout) beginning approximately five (5) minutes after the reactor trip. This information is retained and may be used as an aid to evaluate the Post Trip/Safeguards Review.

2. RTGB Strip Chart Recorder

The strip chart data is evaluated per the Post Trip/Safeguards Review Report and is retained in the HBR 2 vault as a permanent record.

NRC POSITION

1.2.2.6. Power source(s) (e.g., Class IE, non-Class IE, non-interruptable)

CP&L RESPONSE

1. Plant Process Computer

Power for the plant process computer is supplied by vital DC power. Detailed information and electrical one line diagram is provided in the HBR2 updated FSAR, Section 8.1.2.5.

2. RTGB Strip Chart Recorder

Power for the strip chart recorders is supplied by various instrument busses. Detailed information pertaining to the instrument busses is provided in the HBR2 updated FSAR, Section 8.1.2.6 and Table 8.3.1-2.

NRC POSITION

1.2.3. Other data and information provided to assess the cause of unscheduled reactor shutdowns.

CP&L Response

Other data and information which can be used to assess the cause of unscheduled reactor shutdowns are as follows:

1. Core Cooling Instrumentation which includes Core Cooling Monitor and Core Exit Thermocouples. (Discussed in our April 26, 1983, response to Generic Letter 82-28)

2. Operator/Shift Foreman/STA logs of past plant conditions or events which may show trends leading up to an unscheduled shutdown. Plant tests may also be used to establish abnormal trends that may have preceded the unscheduled shutdown.

NRC POSITION

1.2.4. Schedule for any planned changes to existing data and information capability.

CP&L RESPONSE

Some or all of the following activities should result in improved post trip review capabilities.

1. Requirements for Emergency Response Capability (Generic Letter 82-33)
   The Emergency Response Capabilities at HBR2 will include, Safety Parameter Display System (SPDS), Detailed Control Room Design Review (DCRDR), Regulatory Guide 1.97 (RG 1.97), Emergency Operating Procedure (EOPs), and Emergency Response Facilities. (This is discussed in our April 15, 1983 and August 24, 1983 response to Generic Letter 82-33).

2. Installation of Westinghouse Reactor Vessel Level Instrumentation System (RVLIS). (This is discussed in our March 31, 1981 and April 26, 1983 letter).

3. The information from the Post Trip/Safeguards Review Program on the plant process computer is obtained only when the plant computer is available and functioning per its capabilities. The HBR Plant process computer does not uniquely perform any required safety functions and therefore the availability of the system is not required at all times. CP&L is in process of evaluating the need to replace the current computer with a more reliable system.

2.1 EQUIPMENT CLASSIFICATION AND VENDOR INTERFACE (REACTOR TRIP SYSTEM COMPONENTS)

NRC POSITION

Licensees and applicants shall confirm that all components whose functioning is required to trip the reactor are identified as safety-related on documents, procedures, and information handling systems used in the plant to control safety-related activities, including maintenance, work orders, and parts replacement. In addition, for these components, licensees and applicants shall establish, implement and maintain a continuing program to ensure that vendor information is complete, current and controlled throughout the life of the plant, and appropriately referenced or incorporated in plant instructions and procedures. Vendors of these components should be contacted and an interface established. Where vendors can not be identified, have gone out of business, or will not supply the information, the licensee or applicant shall assure that sufficient attention is paid to equipment maintenance, replacement, and repair, to compensate for the lack of vendor backup, to assure reactor trip system reliability. The vendor interface program shall include periodic communication with vendors to assure that all applicable information has been received. The program should use a system of positive feedback with vendors for mailings containing technical information. This could be accomplished by licensee acknowledgement for receipt of technical mailings. The program shall also define the interface and division of responsibilities among the licensees and the nuclear and nonnuclear divisions of their vendors that provide service on reactor trip system components to assure that requisite control of and applicable instructions for maintenance work are provided.

CP&L RESPONSE

The classification criteria, for safety related equipment is incorporated in the HBR2 Plant Operating Manual (POM), "Q-List Control Procedure." This equipment classification criteria is discussed further in our response to Section 2.2. The reactor trip breakers (RTB), bypass breakers, are included in the HBR2 Q-List. The vendor interface program for all safety related equipment (including the reactor trip breakers and by-pass breakers) is described in Section 2.2.2.

2.2 EQUIPMENT CLASSIFICATION AND VENDOR INTERFACE (PROGRAMS FOR ALL SAFETY-RELATED COMPONENTS)

NRC POSITION

Licensees and applicants shall submit, for staff review, a description of their programs for safety-related equipment classification and vendor interface as described below:

2.2.1. For equipment classification, licensees and applicants shall describe their program for ensuring that all components of safety-related systems necessary for accomplishing required safety functions are identified as safety-related on documents, procedures, and information handling systems used in the plant to control safety-related activities, including maintenance, work orders and replacement parts. This description shall include:

2.2.1.1 The criteria for identifying components as safety-related within systems currently classified as safety-related. This shall not be interpreted to require changes in safety classification at the systems level.

CP&L RESPONSE

The classification criteria for safety-related equipment is incorporated into the HBR2 Plant Operating Manual (POM) Q-List Procedure and is defined as follows:

> The Q-list plant items for a nuclear power station are those structures, systems and components (of both safety and non-safety systems) that are designed primarily to prevent or mitigate the consequences of postulated accidents (i.e., limiting plant process conditions) that could cause undue risk to the health and safety of the public. For qualifying as a Q-list plant item, the plant item in question must be necessary to:
>
> a. Provide for structural integrity (or be a part) of the reactor coolant pressure boundary (RCPB), or
> b. Provide over pressure protection for the RCPB, or
> c. Make and hold the reactor sub-critical during the occurrence of frequent, infrequent, and limiting plant process conditions, or
> d. Cool the core following normal reactor shutdown and during the occurrence of frequent, infrequent, and limiting plant Process conditions, or
> e. Cool another safety system component during the occurrence of frequent, infrequent and limiting plant process conditions, such that its failure would inhibit the cperation of the safety system, or
> f. Provide essential services to the reactor containment during the occurrence of infrequent and limiting plant Process conditions, or

g. Contain the radioactivity, control or reduce the radioactivity release to the environment during the occurrence of frequent, infrequent and limiting plant process conditions.
h. Meet other regulatory requirements involving specific seismic and/or environmental qualifications as determined by the Engineering Supervisor.

The POM defines how the Q-list is developed and will be controlled, covering all nuclear safety-related structures, systems, and components for HBR2. Plant Engineering is responsible for determining whether a system or component should be on the Q-list. This decision is based on the previous definition and using the following Regulatory Guides and Standards for guidance:

a. NRC Regulatory Guide 1.26, "Quality Group Classification and Standards for Water, Steam and Radioactive-Waste Containing Components of Nuclear Power Plants."
b. NRC Regulatory Guide 1.29, "Seismic Design Classification".
c. NRC Regulatory Guide 1.32, "Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants".
d. NRC Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident".
e. ANSI N18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Plants" and supplement, ANSI 18.2a.

In case the criteria referenced above are not directly applicable, an engineering judgement based on the functional need of the component in question is utilized to make the Q-list determination. The functional need of the component in question is determined from information contained in the Final Safety Analysis Report, Plant Operating Manual, Technical Specifications and vendor manuals, if necessary.

NRC POSITION

2.2.1.2. A description of the information system used to identify safety-related components (e.g., computerized equipment list) and the methods used for its development and validation.

CP&L RESPONSE

The Q-List is the plant information system used to identify safety-related components. As discussed in response to Section 2.2.1.1, the Q-list Control Procedure defines how the Q-list is developed and will be controlled covering all nuclear safety-related structures, systems, and components for HBR2. In addition, the Plant Design Control Procedure sets forth the requirements for processing and controlling modifications and design changes of plant structures, systems and components (Q-list, Fire-Q, Radwaste-Q, and others) in order to assure that they continue to meet their performance/functional objectives. This procedure includes guidance on updating the Q-List when new equipment is added.

Each addition and/or change to the Q-List is reviewed in detail during the procedure change process. This process includes, as a minimum a review by the Engineering Supervisor, a detailed safety analysis, a nuclear safety review by two (2) qualified safety reviewers, a review by Corporate Quality Assurance and approval by the Manager - Technical Support or the Plant General Manager. The review may also include a review by Fire Protection personnel and ALARA personnel.

NRC POSITION

2.2.1.3. A description of the process by which station personnel use this information handling system to determine that an activity is safety-related and what procedures for maintenance, surveillance, parts replacement and other activities defined in the introduction to 10 CFR 50, Appendix B, apply to safety-related components.

CP&L RESPONSE

The maintenance, surveillance, and procurement activities at HBR2 all utilize the Q-List to determine which components require additional controls. Each system has certain required reviews and approvals based on the component being identified as Q-List. In the maintenance area, Q-List Work Requests require reviews by maintenance, operations and quality assurance personnel. In addition, a Local Clearance/Test Request Form is prepared for Q-List Work Requests to ensure that appropriate post-maintenance testing is performed. In the surveillance area, Technical Specification testing is done by approved plant procedures. These procedures require the same level of detailed reviews and approvals as the Q-List. In the procurement area, safety-related Q-List materials are ordered on requisition forms which differentiate them from non safety related items. The requisition is then subjected to the required reviews and approvals, including Quality Assurance, to ensure appropriate design and inspection specifications are referenced. In addition, the review ensures that the material is procured from a qualified vendor established by the corporate Quality Assurance Manual and ANSI N45.2.13-1974, entitled, "Control of Procurement of Items and Services for Nuclear Power Plants."

NRC POSITION

2.2.1.4. A description of the management controls utilized to verify that the procedures for preparation, validation and routine utilization of the information handling system have been followed.

CP&L RESPONSE

The controls described in Section 2.2.1.2 ensure that additions and changes to the Q-List are appropriately controlled. Corporate Quality Assurance performs surveillances and audits of the maintenance, surveillance, and procurement systems described in Section 2.2.1.3. These surveillances and audits verify continued compliance with the requirements for handling Q-List components in all three (3) systems.

NRC POSITION

2.2.1.5. A demonstration that appropriate design verification and qualification testing is specified for procurement of safety-related components. The specifications shall include qualification testing for expected safety service conditions and provide support for the licensees' receipt of testing documentation to support the limits of life recommended by the supplier.

CP&L RESPONSE

The procurement of safety related components is governed by POM, procedure: "Procurement of Plant Material and Equipment", as discussed. The Purchase Requisition is reviewed by plant engineering and QA personnel, prior to approval by appropriate management, to ensure that all requirements of the component specification and the CP&L Quality Assurance Program are included. Appropriate service conditions, material requirements, testing requirements, qualifications testing data, packaging, shipping, and storage requirements, etc., are included in the purchase requisition. Vendors are also requested to provide shelf life information for applicable components.

In addition to these controls on the front end of the purchasing process, the Corporate Quality Assurance Department performs surveillance of vendors and contractors to ensure that they are meeting all applicable quality assurance requirements. Based on all of the above requirements and surveillances, an approved suppliers list is maintained, listing acceptable sources for the purchase of Q-list items or services.

Off-site procurement of safety-related material used at HBR is controlled by the Standard Procedure Nos. NPCD-P-0066 for "Construction Procurement," and NPCD-P-0075 for "Final Approval of Requisition for Site-Procured Items." These documents contain the guidelines used by Carolina Power & Light Company Nuclear Power Plant Construction Department; Robinson Nuclear Project Department, and Nuclear Engineering and Licensing Department for final approval of requisitions for construction-procured items. Directives are provided to ensure requisitions are prepared from approved drawings and specifications and that the preparation meets the project administrative procedures, particularly those related to specifications and quality assurance requirements. All safety-related purchase orders must be reviewed and approved by the Corporate Quality Assurance Department (CQAD). CQAD also approves the vendor selection. Items must include certification requirements and provisions of 10 CFR Part 21 on the purchase order when required.

Plant modifications which are determined to affect Q-list components receive the following reviews in addition to the routine management reviews to assure the appropriate procurement requirements are specified:

- Technical review
- Safety analysis
- QA review
- Two-party independent nuclear safety review

- o Manager of Technical Support approval
- o Corporate Nuclear Safety review of safety analysis

Carolina Power & Light Company believes these procedural and administrative controls with technical reviews imposed on procurement documents for safety related equipment provides reasonable assurances of the equipment spare parts and engineering services acquired from the vendors.

NRC POSITION

2.2.1.6. Licensees and applicants need only to submit for staff review the equipment classification program for safety-related components. Although not required to be submitted for staff review, your equipment classification program should also include the broader class of structures, systems, and components important to safety required by GDC-1 (defined in 10 CFR Part 50, Appendix A, "General Design Criteria, Introduction").

CP&L RESPONSE

As discussed in our response, the equipment classification program at HBR2 is encompassed within the plant Q-list. This program is addressed in detail in our response to Section 2.2.1 through 2.2.5.

NRC POSITION

2.2.2. For vendor interface, licensees and applicants shall establish, implement and maintain a continuing program to ensure that vendor information for safety-related components is complete, current and controlled throughout the life of their plants, and appropriately referenced or incorporated in plant instructions and procedures. Vendors of safety-related equipment should be contacted and an interface established. Where vendors cannot be identified, have gone out of business, or will not supply information, the licensee or applicant shall assure that sufficient attention is paid to equipment maintenance, replacement, and repair, to compensate for the lack of vendor backup, to assure reliability commensurate with its safety function (GDC-1). The program shall be closely coupled with action 2.2.1 above (equipment qualification). The program shall include periodic communication with vendors to assure that all applicable information has been received. The program should use a system of positive feedback with vendors for mailings containing technical information. This could be accomplished by licensee acknowledgement for receipt of technical mailings. It shall also define the interface and division of responsibilities among the licensee and the nuclear and nonnuclear divisions of their vendors that provide service on safety-related equipment to assure that requisite control of and applicable instructions for maintenance work on safety-related equipment are provided.

CP&L RESPONSE

Currently, HBR2 is phasing in a Vendor Interface Program to provide adequate assurance that vendor information for safety-related equipment is appropriately incorporated in the plant instructions and procedures. The existing vendor interface program includes:

1. Procedural processing of vendor recommendations.

   Vendor recommendations which are received on-site are processed in accordance with an Engineering procedure: Assurance of Operating Equipment Parameters and Limits. Vendor recommendations concerning plant equipment are handled by the Performance Engineering Subunit and then routed to the On-site Nuclear Safety (ONS) Unit for inclusion in the Operating Experience Feedback Program. This program ensures that the information is provided to Operations, or Maintenance or other appropriate organizations for their use.

2. Procedural control of technical manuals.

   Maintenance activities are performed in accordance with approved plant procedures. The vendor technical manuals are used as a source of reference material in preparing these procedures. This process provides for review and incorporation of technical manual information into the plant maintenance procedures. Thus, the vendor input is evaluated before use and integrated with site specific conditions and experience. In some cases where the equipment is complex and the vendor manuals are suitable for direct use, the particular section of the manual that provides instructions for accomplishing the desired activity is referenced or incorporated as part of the approved procedure. In such cases, the maintenance procedure requires that the technical manual be plant approved.

3. Control of the vendors manuals.

This is accomplished by incorporating the vendor technical manuals in the document control process as controlled documents. Control and accountability of the master copies of technical manuals are maintained in the HBR2 vault.

If it is felt that a particular vendor recommendation is inappropriate at HBR2 the H. B. Robinson Department Manager can approve changes to Plant Procedures which are not in accordance with the specific manufacturers recommendation. When old equipment is modified or new equipment is installed, vendor recommendations are processed as described previously.

Westinghouse, the HBR2 Nuclear Steam Supply System (NSSS) vendor issues vendor recommendations as Westinghouse Technical Bulletins. It has been verified that all Westinghouse Technical Bulletins applicable to HBR2 have been received by CP&L. In addition, the recommendations of the Westinghouse Technical Bulletins on Reactor Trip Breakers have been reviewed by CP&L and acted upon. CP&L is currently reviewing all other Westinghouse Technical Bulletins to verify that the recommendations have been reviewed and implemented as appropriate.

Westinghouse has now incorporated return receipt letters into their Technical Bulletins. These return receipt letters are followed-up by Westinghouse if the receipt is not returned within a reasonable time. Westinghouse is planning to periodically provide an updated list of Technical Bulletins which CP&L will use as an additional tool in ensuring that all Technical Bulletins have been received.

With respect to vendors other than the NSSS Supplier, CP&L is supporting the INPO Nuclear Utility Task Action Committee (NUTAC) on Generic Letter 83-28. This NUTAC is currently formulating recommendations for an industry wide vendor information program for safety-related equipment. CP&L believes this program will provide a practical industry wide approach to assuring safety related equipment reliability. CP&L will review the results of this NUTAC and make revisions to the HBR2 vendor interface program as appropriate.

The HBR2 plant staff both formally and informally utilizes vendor information from other industry sources; INPO, NPRDS, SEE-IN and Notepad. NPRDS contains data on safety-related equipment failures. SEE-IN contains safety-related equipment failure information and equipment failure trending. Significant industry-wide events are identified and recommended action provided in INPO Significant Operating Event Reports (SOER's). Significant events (without recommended action) are provided in INPO Significant Event Reports (SER's). In addition, operating and technical information is provided in INPO Operating and Maintenance Reports (O&MR's) and Operating Event Reports (OER's).

Another check and balance exists in the NRC notifications of safety-related concerns and regulatory requirements. IE Bulletins, Notices, and Circulars provides current information concerning component or design discrepancies that affect plant equipment and operating conditions.

In summary, the current vendor interface program in conjunction with established plant procedures for maintenance, surveillance testing, equipment repair and replacement and the quality assurance program provide a comprehensive equipment reliability program. The vendor interface program will continue to be reviewed and upgraded based on the INPO NUTAC and experience gained with the current program.

3.1 POST-MAINTENANCE TESTING (REACTOR TRIP SYSTEM COMPONENTS)

NRC POSITION

The following actions are applicable to post-maintenance testing:

3.1.1. Licensees and applicants shall submit the results of their review of test and maintenance procedures and Technical Specifications to assure that post-maintenance operability testing of safety-related components in the reactor trip system is required to be conducted and that the testing demonstrates that the equipment is capable of performing its safety functions before being returned to service.

CP&L RESPONSE

HBR2 has had a documented periodic test procedure on the Reactor Trip Breakers (RTB's) since November 1974. Prior to November 1974 the RTB's were inspected and tested every refueling outage on a Work Request, using the Technical Manual.

Upon receipt of IEB-83-01 and Westinghouse Technical Bulletins 83-02 and 83-03 a review of existing maintenance and test procedures on the RTB's was conducted. This resulted in issuing a new preventative maintenance procedure, corrective maintenance procedure, and revisions to the periodic test procedure and operations work procedures.

The monthly and annual periodic tests include testing of the logic cabinets circuits through the reactor trip breakers and bypass breakers. Prior to the next refueling outage a procedure will be issued to test the circuitry from the manual reactor trip push buttons on the Reactor Turbine Generator Board (RTGB) through the RTB's. This procedure will be performed on a refueling basis.

Carolina Power & Light Company is a member of the Westinghouse Owners' Group (WOG). Westinghouse under contract to WOG is conducting a compilation of all existing maintenance information regarding Westinghouse switchgear. Any new or improved maintenance requirements resulting from this effort as well as any new Technical Bulletins will be reviewed and incorporated into appropriate HBR2 maintenance and period test procedures.

In addition, the post maintenance operability testing of safety-related components in the reactor trip system are performed as described in our response to Section 2.1 and 3.2.

NRC POSITION

3.1.2. Licensees and applicants shall submit the results of their check of vendor and engineering recommendations to ensure that any appropriate test guidance is included in the test and maintenance procedures or the Technical Specifications, where required.

CP&L RESPONSE

Upon receipt of IEB-83-01, CP&L conducted a review of past history on reactor trip breakers. This review included the first failures of Undervoltage Trip Attachments in 1971 (Incident Reports 25, 28, and 29), Work Request and Authorization Forms, Manufacturers Technical Representative Trip Reports, Correspondence between Westinghouse and HBR-2, Technical Bulletins, Data letters, Periodic Test Procedures, and LER's.

Results of this review were as follow:

1. Continue to perform preventative maintenance on a yearly basis and periodic tests on a monthly basis.

2. Include appropriate information from Westinghouse Technical Bulletin TB-83-02 and TB-83-03 in maintenance and test procedures.

3. Renew the Undervoltage Trip Attachments (UVTAs) on a five (5) year schedule.

4. Specifically include the reactor trip breakers in the HBR2 "Q" List Procedure.

All of the above requirements have been completed except for incorporation of the five (5) year Preventative Maintenance Procedure for replacement of UVTA's into the Preventative Maintenance Program.

Upon receipt of results from the WOG program, CP&L will review the results and make appropriate changes to HBR2 Procedures. Upon completion by Westinghouse of the Test Program per Sections 4.2 and 4.3, the results will be reviewed and Technical Specifications Changes as appropriate will be prepared.

NRC POSITION

3.1.3. Licensees and applicants shall identify, if applicable, any post-maintenance test requirements in existing Technical Specifications which can be demonstrated to degrade rather than enhance safety. Appropriate changes to these test requirements, with supporting justification, shall be submitted for staff approval. (Note that action 4.5 discusses on-line system functional testing).

CP&L RESPONSE

The HBR2 Technical Specifications do not contain any known requirements which degrade plant safety.

3.2 POST-MAINTENANCE TESTING (ALL OTHER SAFETY-RELATED COMPONENTS)

NRC POSITION

The following actions are applicable to post-maintenance testing:

3.2.1. Licensees and applicants shall submit a report documenting the extending of test and maintenance procedures and Technical Specifications review to assure that post-maintenance operability testing of all safety-related equipment is required to be conducted and that the testing demonstrates that the equipment is capable of performing its safety functions before being returned to service.

CP&L RESPONSE

Prior to maintenance on equipment, a Work Request and Authorization Form must be completed by appropriate personnel. This form provides a description of the work to be done and identifies if the equipment is Q-list or non Q-list. If the equipment is Q-list, the form is reviewed by Quality Assurance (QA) personnel, prior to being issued to maintenance, to ensure appropriate requirements are specified, QA hold points are identified, and other reviews within the QA program are met. In addition, the Work Request is reviewed by QA after the work is completed. Once the Work Request is approved, operations issues a Local Clearance/Test Request Form (LCTR) to ensure that appropriate actions are taken to remove and reinstate equipment to service for all plant equipment with a few exeptions for secondary plant equipment.

Prior to equipment whose operability is described, in T. S. Section 3, being returned to service, the LCTR requires an Operation Work Procedure (OWP) to be completed. The OWP provides for the appropriate post maintenance testing to assure that the equipment meets its safety function prior to its return to service.

Carolina Power & Light Company believes that the above procedural controls provides reasonable assurance that safety related equipment is capable of performing its safety function before being returned to service.

NRC POSITION

3.2.2. Licensees and applicants shall submit the results of their check of vendor and engineering recommendations to ensure that any appropriate test guidance is included in the test and maintenance procedures or the Technical Specifications where required.

CP&L RESPONSE

Review of vendor or engineering recommendations for this item is the same as described in Section 2.2.2 and 3.1.2 since all safety related components are part of the "Q" List.

NRC POSITION

3.2.3. Licensees and applicants shall identify, if applicable, any post-maintenance test requirements in existing Technical Specification which are perceived to degrade rather than enhance safety. Appropriate changes to these test requirements, with supporting justification, shall be submitted for staff approval.

CP&L RESPONSE

At the present time no post testing requirements have been identified which degrade plant safety.

4.1 REACTOR TRIP SYSTEM RELIABILITY (VENDOR-RELATED MODIFICATIONS)

NRC POSITION

All vendor-recommended reactor trip breaker modifications shall be reviewed to verify that either: (1) each modification has, in fact, been implemented; or (2) a written evaluation of the technical reasons for not implementing a modifications exists.

For example, the modifications recommended by Westinghouse in NCD-Elec-18 for the DB-50 Breakers and a March 31, 1983 letter for the DS-416 breakers shall be implemented or a justification for not implementing shall be made available. Modifications not previously made shall be incorporated or a written evaluation shall be provided.

CP&L RESPONSE

HBR2 has Westinghouse DB-50 Breakers as both the main reactor trip breakers and bypass breakers.

A review of past and present recommendations by Westinghouse was completed in July, 1983. This review included Technical Bulletin 74-1, Data Letter 74-2, Technical Bulletin 83-02, and Technical Bulletin 83-03.

In response to operational problems at HBR2 in 1971 (Incident Reports 25, 28 and 29), Westinghouse made several design changes to the under voltage trip attachments (UVTA's). In October, 1971 the new design UVTA's were installed on the HBR2 trip and bypass breakers. In December, 1971 due to a Westinghouse commitment to provide the modified UVTAs to all sites NCD-Elec-18 was issued as an internal Westinghouse document to implement the change-out program. Since H. B. Robinson had already received and installed the redesigned UVTA's under Westinghouse Technical Representative Guidance, NCD-Elec-18 has been implemented at HBR2.

The recommendations per Data Letter 74-1 which superceded Technical Bulletin 74-1 were implemented in 1974. The maintenance and lubrication of reactor trip breakers were included beginning in 1974 on an annual basis as part of a periodic test procedure.

As a result of the Salem Event, HBR2 issued separate procedures for the monthly test, annual maintenance and lubrication, and corrective maintenance. The maintenance and lubrication procedure includes appropriate information from Technical Bulletin 83-02 and 83-03. (See response to Item 3.1.1)

4.2 REACTOR TRIP SYSTEM RELIABILITY (PREVENTATIVE MAINTENANCE AND SURVEILLANCE PROGRAM FOR REACTOR TRIP BREAKERS)

NRC POSITION

Licensees and applicants shall describe their preventative maintenance and surveillance program to ensure reliable reactor trip breaker operation. The program shall include the following:

4.2.1. A planned program of periodic maintenance, including lubrication, housekeeping, and other items recommended by the equipment supplier.

CP&L RESPONSE

HBR2 has an annual periodic test procedure that includes maintenance and inspection of reactor trip breakers. This procedure provides the instructions necessary to inspect and test the reactor trip and bypass breakers for both A and B trains of reactor protection. Also included is lubricating, contact cleaning, undervoltage device trip testing, shunt trip testing, and checking for proper breaker operation. In addition, this procedure incorporates Westinghouse Technical Bulletins TB-83-02 recommendations for cleaning and lubrication of the breakers. This procedure references the Westinghouse Technical Manual for DB-50 Switchgear, Technical Bulletin, TB-83-02, Technical Bulletin TB-83-03, and other maintenance procedures for additional guidance.

CP&L is a member of Westinghouse Owners' Group (WOG) which has contracted Westinghouse to compile a document containing all existing maintenance information regarding switchgear, including lessons learned in the post-Salem interval. Upon receipt of this document, it will be reviewed by plant personnel and appropriate information will be included in a future revision of the Plant Procedures.

NRC POSITION

4.2.2. Trending of parameters affecting operation and measured during testing to forecast degradation of operability.

CP&L RESPONSE

HBR2 will trend breaker opening response time for each Reactor Trip Breaker. This program is currently under development with estimated completion date of March 1984.

NRC POSITION

4.2.3. Life testing of the breakers (including the trip attachments) on an acceptable sample size.

CP&L RESPONSE

Life cycle testing of the shunt trip attachment and the undervoltage trip attachment (UVTA) of the reactor trip switchgear is being conducted by Westinghouse for the WOG. This program is aimed toward establishing the service life of these devices, and substantiating periodic test requirements with proper maintenance, replacement and qualification programs. The test program is scheduled for completion in 1984.

In addition, the WOG has currently undertaken an effort to review the plant specific histories of reactor trip switchgear performance and perform a reliability assessment of the data obtained.

HBR2 staff will review the results of the above study and testing and will make appropriate revisions to plant maintenance program as necessary.

NRC POSITION

4.2.4. Periodic replacement of breakers or components consistent with demonstrated life cycles.

CP&L RESPONSE

HBR2 staff's review of plant history has determined that the reactor trip breakers UVTAs should be replaced on a five (5) year schedule. If results of the WOG life cycle testing determines additional items or a different time frame for replacing UVTA's, this information will be evaluated and necessary changes incorporated in the plant maintenance program, as appropriate.

4.3 REACTOR TRIP SYSTEM RELIABILITY (AUTOMATIC ACTUATION OF SHUNT TRIP ATTACHMENT FOR WESTINGHOUSE AND B&W PLANTS)

NRC POSITION

Westinghouse and B&W reactors shall be modified by providing automatic reactor trip system actuation of the breaker shunt trip attachments. The shunt trip attachment shall be considered safety related (Class IE).

CP&L RESPONSE

Westinghouse Owner's Group is addressing several areas related to reactor trip system reliability. With respect to automatic actuation of the shunt trip, CP&L has received the generic design package for the automatic shunt trip modification that was submitted to NRC to June 14, 1983 by WOG. Carolina Power & Light Company has also received the NRC Safety Evaluation Report (SER) on the generic design.

The shunt trip modification is currently scheduled to be completed and operational prior to power operation after the Steam Generator Replacement Outage (scheduled to begin in July 1984).

In addition, the plant specific design information required in the SER will be addressed upon completion of the shunt trip modification which is scheduled for the Steam Generator Replacement Outage.

4.4 REACTOR TRIP SYSTEM RELIABILITY (IMPROVEMENTS IN MAINTENANCE AND TEST PROCEDURES FOR B&W PLANTS)

NRC POSITION

Licensees and applicants with B&W reactors shall apply safety-related maintenance and test procedures to the diverse reactor trip feature provided by interrupting power to control rods through the silicon controlled rectifiers.

This action shall not be interpreted to require hardware changes or additional environmental or seismic qualification of these components.

Applicability

This action applies to B&W licensees and OL applicants only.

CP&L RESPONSE

Carolina Power and Light Company's H. B. Robinson Unit 2 has a NSSS designed by Westinghouse Electric Corporation, therefore, Section 4.4 is not applicable.

4.5 REACTOR TRIP SYSTEM RELIABILITY (SYSTEM FUNCTIONAL TESTING)

NRC POSITION

On-line functional testing of the reactor trip system, including independent testing of the diverse trip features, shall be performed on all plants.

4.5.1. The diverse trip features to be tested include the breaker undervoltage and shunt trip features on Westinghouse, B&W (see Action 4.3 above) and CE plants; the circuitry used for power interruption with the silicon controlled rectifiers on B&W plants (see Action 4.4 above); and the scram pilot valve and backup scram valves (including all initiating circuitry) on GE plants.

CP&L RESPONSE

The present Design of the HBR2 reactor trip circuitry allows on line monthly testing of the UVTA's on the main reactor trip breakers. This periodic test includes the logic circuit from the reactor protection rack through the reactor trip breaker. During annual maintenance and lubrication, both the UVTA's and shunt trip are tested. This annual testing procedure independently verifies both the UVTA and the shunt trip mechanically and electrically. This procedure uses testing equipment which is not permanently installed in the reactor trip system. Upon completion of a plant modification per Section 4.3 both the UVTA and shunt trips will be tested using permanently installed test equipment. In addition, prior to the next refueling outage (currently scheduled to commence in December, 1983) procedures will be issued to test start the RTGB manual reactor trip button through the reactor trip breakers. This procedure will be performed on a refueling basis.

NRC POSITION

4.5.2. Plants not currently designed to permit periodic on-line testing shall justify not making modifications to permit such testing. Alternatives to on-line testing proposed by licensees will be considered where special circumstances exist and where the objective of high reliability can be met in another way.

CP&L RESPONSE

HBR2 performs a monthly and annual on line periodic test as discussed in response to 4.5.1. It is CP&L's position that the existing monthly periodic test plus the annual test assure high reliability of the reactor trip system. The addition of an RTGB push button test will give added assurance that both automatic and manual functions will result in tripping the reactor trip breakers when needed.

NRC POSITION

4.5.3. Existing intervals for on-line functional testing required by Technical Specifications shall be reviewed to determine that the intervals are consistent with acheiving high reactor trip system availability when accounting for considerations such as:

1. uncertainties in component failure rates
2. uncertainty in common mode failure rates
3. reduced redundancy during testing
4. operator errors during testing
5. component "wear-out" caused by the testing

Licensees currently not performing periodic on-line testing shall determine appropriate test intervals as described above. Changes to existing required intervals for on-line testing as well as the intervals to be determined by licensees currently not performing on-line testing shall be justified by information on the sensitivity of reactor trip system availability to parameters such as the test intervals, component failure rates, and common mode failure rates.

CP&L RESPONSE

HBR2 performs monthly testing of reactor trip breaker UVTA's. Based on specific plant data the monthly periodic testing has not identified any problems associated with the reactor trip breakers except some initial problems in 1971 (Incident Reports 25 and 28) which resulted in redesign of UVTA's by Westinghouse.

With respect to generic considerations the WOG in January, 1983, submitted WCAP-10271 to the NRC for review. WCAP-10271, "Evaluation of Surveillance Frequencies and out of Service Times for the Reactor Protection Instrumentation System" documents an evaluation of the impact on Reactor Protection System (RPS) unavailability of current and extended surveillance intervals.

The WCAP-10271 considers common mode failure, operator error, reduced redundancy during testing and equipment bypass. WCAP-10271 also considers correlative effects on plant operation and safety including the manpower expenditure associated with surveillance, the number of inadvertent trips which occur during testing and the distraction from plant monitoring on the part of the control room operator and shift supervisor associated with testing. Supplement 1 to WCAP-10271 submitted by WOG is an extension of the evaluation and provides a discussion of component wearout caused by testing. Additional information that will be submitted to the NRC by WOG will include an overall evaluation of the impact on plant safety of RPS unavailability. WCAP-10271, Supplement 1, and the information provided to the NRC in defense of WCAP-10271 provides in a comprehensive form the information requested by this item.

Upon Completion of the review by NRC of WCAP-10271 and Supplement 1, the findings will be reviewed for their impact on the present testing intervals and appropriate changes, will be implemented.