HRD/HBT/RLG
November 4, 1983

Duke Power Company
Oconee Nuclear Station

Response to Generic Letter 83-28
Required Actions Based on Generic
Implications of Salem ATWS Events

1.1 POST-TRIP REVIEW (PROGRAM DESCRIPTION AND PROCEDURE)

This report describes the Reactor Trip Investigation Program established at Duke Power Company's Oconee, McGuire, and Catawba Nuclear Stations. In particular, that portion of the Program that is performed immediately following a trip and prior to restart (e.g., post-trip review) is described. This report describes the Program's essential elements and basic requirements; details can be found in the implementing station directives and procedures.

Purpose

The purpose of the reactor trip investigation is to provide a mechanism whereby each reactor trip is systematically and thoroughly investigated in order to:

1. Determine the immediate and root causes of the trip.

2. Identify unexpected, abnormal responses to the trip by plant systems, equipment, and personnel.

3. Assess the impact of identified abnormalities on nuclear safety, equipment reliability, system performance, and availability.

4. Confirm the readiness of the unit to restart, or bring it to readiness.

5. Develop corrective actions to prevent the recurrence of the trip and mitigate abnormal responses commensurate with their significance.

6. Document observed plant behavior for use in subsequent evaluations.

7. Satisfy reporting requirements (10 CFR 50.73).

Scope

The Reactor Trip Investigation Program implemented at Duke nuclear stations applies to every reactor trip, planned and unplanned. However, planned reactor trips need not undergo all phases of the investigation if response is normal. The scope of the information reviewed under the program is sufficient to accomplish its objectives and includes data on plant system behavior, actuation and sequence of operation of equipment, records of operator actions, and plant activities affecting the event. The program prescribes activities that are performed immediately following a trip, prior to restart, and continues through a subsequent in-depth evaluation that supports preparation of internal and external reports.

Roles and Responsibilities

Several groups, including station and general office staff, participate in the reactor trip investigation program. The responsibilities and authority of each participant have been clearly defined.

At each station, the Operations group is responsible for operating the plant. Under the program, they notify station management and the Performance and Licensing groups of the occurrence of a reactor trip.

The reactor operators are responsible for diagnosing and controlling the event and thus will have firsthand knowledge of the particulars of the event. This information is to be promptly documented to help ensure that a complete record of the event is obtained. Operations is responsible for deciding when and how the unit is to be restarted.

At Duke nuclear stations, the Reactor Engineer, or an Engineer trained and qualified by the Reactor Engineer is responsible for performing the post-trip review.

The general office Reactor Safety Section staff provides analytical support to the Station staff as requested. The section provides expertise in event and trainsient analysis, safety analysis, and probabilistic risk analysis.

Program Phases

The Reactor Trip Investigation Program consists of four distinct phases:

1. Post-trip review
2. Restart decision
3. Independent review
4. Subsequent investigation

Every reactor trip will be subjected to a post-trip review and restart decision. Planned reactor trips, where no abnormalities have been identified, need not proceed to the subsequent investigation phase. The major elements of each of these phases is described below.

Post-trip Review

The post-trip review is performed immediately following the trip and completed prior to restart. The purpose of the post-trip review is to:

1. Determine the immediate cause of the trip.

2. Identify other-than-expected performance of operators, systems, and equipment.

3. Assess the impact of identified abnormal performance on safe operation.

4. Ensure continued availability of information and data pertaining to the event.

The scope of the post-trip review has been established to ensure that abnormal performance in important systems will be identified. Guidelines and criteria, which define the range of expected system response, are used in the process.

The major elements of the post-trip review are:

1. Determination of immediate cause of the trip.

2. Determination of the reactor trip sequence events.

3. Review of pre- and post-trip behavior of key parameters that reflect overall plant performance and will identify abnormal performance of important systems.

4. Review of the performance of important plant systems and equipment (safety and control) to identify other-than-expected response to the trip. These systems include ESF, AFW/EFW, RCS pressure relief and control, RCS inventory control, main steam relief and control, SG level control, and radiological control.

5. Determination of the need for an Independent Review prior to restart.

6. Approval for restart, or identification of corrective actions that must be completed prior to restart.

Restart Decision

Prior to restarting the unit, Operations must ensure that:

1. The immediate cause of the trip is known or has been investigated to the fullest extent possible while shutdown.

2. The plant's transient response either did not identify any problems that impact the ability of the unit to be safely restarted and operated, or that the problems have been corrected.

3. Any problems with equipment subject to Tech Spec LCO requirements are corrected as required.

4. The recommendations for corrective actions identified during the post-trip review are resolved.

Independent Review

Under certain conditions, an additional independent review must be performed prior to restart to ensure that all questions regarding the ability to safely restart and operate the plant are resolved. Criteria have been established as to when the additional independent review is required. They are as follows:

1. If the immediate cause of the reactor trip cannot be determined, or

2. If any unresolved safety issues exist, or

3. If compliance with licensing requirements is in question.

The need for an independent review will be identified by the individual performing the Post-Trip Review, concurred with by the Station Manager (or designee), and will be performed by a group of knowledgeable individuals designated by the station manager. They will report their results to Station Manager or his designee.

Subsequent Investigation

Every unplanned reactor trip will be subjected to a follow-up, in-depth investigation. In addition, planned reactor trips which show abnormalities in plant response will also be investigated. The purpose of the subsequent investigation is to ensure that all aspects of the events are fully investigated, evaluated, and documented. The subsequent investigation takes the knowledge gained from the post-trip review and expands upon it in areas of identified abnormal response. It ensures that the more subtle aspects of system performance, even though they did not significantly affect the plant response, are evaluated and needed corrective action identified. This report need not be completed before restart. The scope of the subsequent investigation is prescribed to ensure that all reporting requirements can be met.

Implementation

The Reactor Trip Investigation Program is implemented at each station via a station directive with supporting guidance contained in a manual to be used by personnel involved in the process.

Attachment A to this report is a copy of a generic station directive, upon which individual station directives are based.

Attachment B is a copy of the form used to document the post-trip review. The form is structured to provide guidance and direction, to help the reviewer "ask the right questions" in collecting and recording necessary information.

Training and Qualification Requirements

The investigation of reactor trips is not an exact science, it involves considerable amounts of interpretation and judgement. Consequently, a successful, thorough investigation is dependent upon the participation of knowledgeable individuals who understand plant design, operating characteristics, safety requirements and who are familiar with plant specific transient behavior.

The Reactor Trip Investigation Program will be conducted by sufficiently trained and experienced personnel in order to ensure high quality results. A list of individuals qualified to perform post-trip reviews will be maintained at each station. Only individuals on the list will perform post-trip reviews. As a minimum, these individuals will possess five years nuclear experience, of which four years may be satisfied by a bachelor's degree in physical science or engineering.

Data Acquisition

The essential element of the Reactor Trip Investigation Program is the acquisition of necessary plant data. At Oconee, McGuire, and Catawba these data are obtained from many sources. Post-trip reviewers are directed to alarm typer and events recorder printouts for data to be used in assemblying the sequence of events and documenting operator actions. The plant transient monitor is used to record analog and digital parameters for analysis of system responses. The parameters recorded by these devices include:

(a) RCS pressure, temperature, flow; pressurizer level; and power level;

(b) Steam generator pressure, level; condensate/feedwater flow, pressure, temperature; relief valve position;

(c) Protection system trip and actuation signals; reactor trip breaker status; generator breaker status; control system signals.

The sufficiency of parameters selected for recording has been proven by experience in performing reactor trip investigations.

Additional information on data acquisition capabilities can be found in the response to Item 1.2.

Schedule

The Reactor Trip Investigation Program described herein will be implemented...

at Oconee by January 1, 1984
at McGuire by January 1, 1984
at Catawba by initial criticality.

At that time the implementing station directive, operating procedures, and performance manual guidance will be in place.

ATTACHMENT A, RESPONSE TO GENERIC LETTER 83-28, ITEM 1.1

Station Directive ______________

DUKE POWER COMPANY
__________ NUCLEAR STATION
INVESTIGATION OF REACTOR TRIPS

1.0 PURPOSE

This directive defines the action to be taken in investigating reactor trips to ensure full understanding of the cause of the trip; the plant transient behavior before and after the trip; the trip's impact on nuclear safety, power production and performance; and to identify necessary corrective action. In addition, this directive prescribes the criteria that must be satisfied in order to restart the unit.

2.0 APPLICABILITY

Every reactor trip shall be reviewed as set forth in this directive, except as noted below. A reactor trip is defined as an event wherein the reactor trip breakers open and the (shutdown bank) control rods fall to their fully inserted position. Specifically excluded from this definition are control rod drop tests performed at power levels below 5% full power.

A reactor trip need not undergo Subsequent Investigation (Section 4.4) as prescribed herein if:

(a) The trip was initiated and specifically called for in an in-progress test procedure, or was part of a planned unit shutdown, and
(b) No anomalies in plant behavior were identified during the Post-Trip Review (Section 4.1).

3.0 RESPONSIBILITIES

Upon a reactor trip, the Superintendent of Operations, Shift Supervisor, or Operations Duty Engineer (hereinafter "Operations"), shall notify the Station Reactor Engineer or Performance Reactor Unit/Duty Engineer of the trip. Operations shall ensure that appropriate notification is made to the NRC and Nuclear Production Department Duty Engineer, and that appropriate notation of the event is entered into the SRO Logbook.

The Station Reactor Engineer, or an engineer trained and qualified by the Reactor Engineer, (hereafter Reactor Engineer) shall perform a Post-Trip Review. The Reactor Engineer shall make a recommendation to Operations on whether or not to restart the unit, including identification of any items that should be resolved prior to criticality.

Operations shall be responsible for deciding when and how the unit is to be restarted. The recommendation(s) from the Reactor Engineer shall be considered in this decision.

4.0 IMPLEMENTATION

An investigation as described herein shall be initiated immediately following a reactor trip. Prompt action shall be taken to ensure that relevant information and data are retained. In particular, necessary interviews of personnel knowledgeable of the reactor trip should be conducted promptly.

The investigation process described in this directive has four (4) distinct phases:

(a) Post-Trip Review
(b) Restart Decision
(c) Independent Review
(d) Subsequent Investigation

4.1 Post-Trip Review

A Post-Trip Review shall be performed immediately following a reactor trip and completed prior to restart of the unit.

The purpose of the Post-Trip Review is to:

(a) Determine the immediate cause of the reactor trip. It is not required that the root cause (e.g., the cause of a component failure leading to the trip) be determined at this time.
(b) Identify other-than-expected performance of operators, systems, and equipment and assess its impact on safe plant operation.

The actions to be performed as the Post-Trip Review are listed on Enclosure 4.1.

The Post-Trip Review shall be performed by the Reactor Engineer with assistance as needed from other personnel. The results of the review shall be adequately documented and provided to personnel performing a Subsequent Investigation (Section 4.4). The Reactor Engineer shall retain a record of the Post-Trip Review.

Written guidelines shall be prepared for use in performing the Post-Trip Review. These guidelines shall describe the various aspects of the trip event that should be considered in order to ensure that any pact on safe operation is identified and resolved and shall provide criteria and guidelines defining the range of expected plant responses.

4.2 Restart Decision

Prior to restart of the unit, Operations shall ensure that the following criteria are met.

1. The immediate cause of the reactor trip is known or has been investigated to the fullest extent possible while remaining in the shutdown condition.

2. The plant transient behavior, immediately preceding and until stabilization following the trip, does not identify any unresolved problems that impact the ability of the unit to be safely restarted and operated.
3. Any malfunctions or failures in equipment or components subject to Tech Spec LCO requirements are evaluated and corrected as required prior to restart.

Operations shall ensure that the Reactor Engineer's recommendation(s) are resolved prior to restart and shall obtain his/her concurrence with restart. Concurrence shall be indicated by the Reactor Engineer's signature on the trip recovery operating procedure. The Station Manager, or his designee, shall resolve disagreements between Operations and the Reactor Engineer regarding necessary corrective action.

4.3 Independent Review

The need for an Independent Review will be identified by the Reactor Engineer and concurred with by the Station Manager, or his designee.

An independent review shall be performed prior to restart if:

1. The immediate cause of the reactor trip cannot be determined, or
2. Any unresolved safety issues exist, or
3. Compliance with licensing requirements is in question.

The independent review shall be performed by a group of individuals identified by the Station Manager who are knowledgeable of the issue in question. This group shall make a recommendation on restart to the Station Manager, or his designee, who shall resolve the restart decision. Documentation of the results of this review, including the bases for conclusions reached, shall be included in the documentation of the event.

4.4 Subsequent Investigation

The investigation of each reactor trip shall be documented in an Incident Investigation Report. This report shall be prepared in accordance with Station Directive 2.8. ________.

As a minimum, the Incident Investigation Report for a reactor trip event shall include,

1. Information and data collected during the Post-Trip Review.
2. A sequence of Events describing all action taken and system/equipment performance during the event that materially affected the course of the event.
3. As assessment of the plant transient behavior that identifies any deviations from expected plant performance and that documents the behavior of key plant parameters.
4. An assessment of the performance of protection and Engineered Safety systems during the transient, identifying any malfunctions or failures to perform as expected.
5. As assessment of other equipment failures that contributed to the event.
6. As assessment of personnel performance and procedure adequacy relevant to the event.

7. A description of corrective actions taken or planned as a result of the event.

The Subsequent Investigation need not be completed prior to restart.

5.0 PERSONNEL QUALIFICATIONS

Personnel performing Post-Trip Reviews shall be qualified by training and/or experience to ensure that they are sufficiently knowledgeable of system design and operating characteristics and nuclear safety considerations to identify other-than-expected performance and to assess its significance. As a minimum, they shall have five (5) years experience, of which four (4) years may be satisfied by a bachelors degree in engineering or physical science. The Reactor Engineer shall maintain a list of personnel qualified to perform Post-Trip Reviews. Only personnel on the list shall perform Post-Trip Reviews.

Enclosure 4.1

Post-Trip Review Actions

(1) Verify unit stability following trip

(2) Service Transient Monitor

(3) Determine or verify immediate cause of trip; confirm reactor trip sequence of events through review of Events Recorder, Alarm Typer, and Transient Monitor.

The reactor trip sequence of events includes:

(a) type and time of initiating RPS/SSPS signal

(b) time of each Reactor Trip Breaker opening

(c) time of manual reactor trip signal

(d) time of turbine trip

(4) Review pre- and post-trip behavior for the following key parameters:

Primary	Secondary
RCS Tave, each loop	SG Pressure
Pressurizer Level	SG Level
RCS Pressure	
RCS Cooldown Limit	
Reactor Power	

Identify deviations from expected behavior and perform in-depth investigation as appropriate.

(5) Review performance of the following systems/equipment:

(a) Safety systems including Emergency/Auxiliary Feedwater; Emergency Core Cooling systems; Containment Isolation, cooling and spray; Reactor Protection System including trip breakers.

(b) RCS PORV and Code Relief valves

(c) Emergency power supply (source to 4160 v bus)

(d) SPDS "problem" indication (when installed)

(e) Main Steam (code) relief valves

(f) SG pressure control (TBV or steam dump)

(g) SG level control

(h) RCS inventory control

(i) RC Pumps including seal injection and cooling

(j) Sequence of operation for turbine runback, FW isolation, AFW start

(k) RIA/EMF (response as expected)

Identify other-than-expected performance. Abnormal behavior requires in-depth evaluation and resolution prior to restart. If performance in all areas was as expected, unit may be safely restarted.

(6) Ensure continued availability of data from the following sources: Transient Monitor, Alarm Typer, Events Recorder/Summary, Control Room Logbooks.

(7) Conduct interview of involved personnel as necessary.

(8) Identify the need for an (additional) Independent Review. If the criteria specified in Section 4.3 are not met, then an (additional) Independent Review is required. Indicate need on Post-Trip Review Report and notify Operations and Station Manager, or his designee.

(9) Identify corrective action that should be taken to resolve identified problems. Specifically identify those that should be completed prior to restart.

(10) Sign off operating procedure if in agreement with restart. If not in agreement, advise Operations of actions that should be completed prior to criticality to ensure continued safe operation and DO NOT sign operating procedure until completed.

Attachment B, Response to Generic Letter 83-28, Item 1.1

DUKE POWER COMPANY
Nuclear Production Department
Reactor Trip Investigation Program

OCONEE NUCLEAR STATION
POST-TRIP REVIEW REPORT

DATE: ______________

PREPARED BY: ______________

I. Identification of Transient

(a) Unit ______ Date of Occurrence ______________ Time ____________

(b) Cause of Transient: ____

____

____

(c) Description of Transient:

____

____

____

____

____

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____

____

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____

OCONEE NUCLEAR STATION
POST-TRIP REVIEW REPORT

II. Initial Conditions

(a) Reactor Power ____________

(b) No. of RC Pumps Operating ____________

(c) No. of MFW Pumps Operating ____________

(d) Turbine Control (Man/Auto) ____________ Turbine Load ____________

(e) Makeup Pumps Operating ____________

(f) Status of ICS Control Stations (Manual/Auto):

1. Unit Load Demand ____________

2. SG/Reactor Master ____________

3. Reactor Demand (Bailey) ____________
   Diamond Control ____________

4. Loop A FW Demand ____________
   Loop B FW Demand ____________

5. Delta Tc ____________

6. Loop A SU FW Valve ____________
   Loop B SU FW Valve ____________

7. Loop A Main FW Valve ____________
   Loop B Main FW Valve ____________

8. Loop A FW Pump Control ____________
   Loop B FW Pump Control ____________

9. Turbine Bypass Valves ____________

10. Pressurizer Level Control
    (Man/Auto) ____________

    Pressurizer Spray Control
    (Man/Auto) ____________

    Pressurizer Heater Control
    (Man/Auto) ____________

11. PORV Block Valve (Open/Closed) ____________

(g) Off-normal Status of Any Trains/Portions of any Safety Systems:

1. RPS ____________
2. ECCS ____________
3. Containment Cooling & Spray ____________
4. Emergency Feedwater ____________
5. Emergency Power ____________

(h) Testing in Progress: ____________

III. <u>Plant Response</u>

A. Safety System Actuation and Performance

1. Reactor Protection System

   Trip Initiating Signal ____________ Time ____________

2. Control Rod Drive Breakers

   CRD DC Bkr CB1 Open Time ____________

   CRD DC Bkr CB2 Open Time ____________

   CRD DC Bkr CB3 Open Time ____________

   CRD DC Bkr CB4 Open Time ____________

   CRD AC Bkr Unit 10 Open Time ____________

   CRD AC Bkr Unit 11 Open Time ____________

3. Manual Trip Signal Time ____________

4. Turbine Trip

   Trip Initiating Signal ____________ Time ____________

OCONEE NUCLEAR STATION
POST TRIP REVIEW REPORT

5. ECCS

   1. HPI — Actuation Time ______

      Actuation Signal ______ No. of Trains ______

   2. CFT — Actuation Time ______

   3. LPI — Actuation Time ______

      No. of Trains ______

6. Containment Isolation — Actuation Time ______

   Actuation Signal ______

7. Containment Cooling & Spray

   Actuation Time ______

   No. of Coolers ______

   No. of Building Spray Trains ______

8. Emergency FW Actuation Signal ______ Actuation Time ______

   MDEFWP #A Started at ______ Stopped at ______

   #B Started at ______ Stopped at ______

   TDEFWP Started at ______ Stopped at ______

9. Emergency Power ______ Actuation Time ______

   No. of Trains ______

10. PZR Code Safety Valves — Actuation Time ______

11. Main Steam Relief Valves (MSRV)

    Did MSRVs respond properly? (Yes/No) ______

B. Control Systems Actions

   1. Reactor Runback (Yes/No) ______ Runback Signal ______

      Power Level/Time Beg: _____/_____ End: _____/_____

2. Turbine Runback (Yes/No) __________ Runback Signal __________

   Load/Time Beg: _____/_____ End: _____/_____

3. RCS Pressure Control

   Did PORV lift? (Yes/No) __________ Time __________

   Was PZR heater response normal? (Yes/No) __________

   Was PZR spray response normal? (Yes/No) __________

4. RCS Inventory Control

   Was Pressurizer Level control normal? __________

   Was coolant released from the quench tank? __________

   Was letdown isolated? When? __________

   Were additional makeup (HPI) pump(s) started? (Y/N)____Which?____

   How:

   Auto Start __________ On What Signal? __________

   Manual Start __________

   Time Started __________ Time Stopped __________

5. RCS Flow Response

   Were any RC Pumps tripped)? __________

   Time ______________ Pumps ______________

6. RCS Temperature Response

   Was RCS Cooldown within the Technical

   Specification Limit? ______________

OCONEE NUCLEAR STATION
POST TRIP REVIEW REPORT

7. SG Pressure Control

Did the Turbine Bypass Valves control properly? ____________

Was steam pressure control normal? ____________

8. SG Level Control

Did SG level respond normally? ____________

C. Manual Actions

Were any ICS control stations taken from auto to manual? (Specify station and ∿ time/sequence). ____________________

____

____

____

Other manual actions: ______________________________

____

____

D. Radiological Response (Describe any unexpected behavior, such as unexpected RIA response) ____________________

____

____

____

____

____

____

E. Transient Data for Key Plant Parameters

	MAX				MIN			
Reactor Power	______				______			
RCS Pressure	______				______			
RCS Temp (Tave)	Loop A	______	B	______	Loop A	______	B	______
Pressurizer Level	______				______			
RCS Cooldown Rate	______				______			
SG Pressure	Loop A	______	B	______	Loop A	______	B	______
SG Level	Loop A	______	B	______	Loop A	______	B	______

(Attach T.M. plots of key plant parameters when available.)

IV. Summary of Unexpected Responses

A. Discussion of Any Unexpected Behavior of Key Parameters (Refer to III.E.)

____

____

____

____

____

B. Discussion of Unexpected Personnel Response (Refer to III.C)

____

____

____

____

____

C. Identification of Systems with Inadequate Performance (Discuss the Nature of the Deficiency) (Refer to III.A and B.)

V. Identification of Needed Action - Prior to Restart

Is an Independent Review Required? ( )Yes ( )No

VI. Identification of Recommended Action - Following Restart

VII. Recovery

Operating Procedure Signed Off, time ________

Reactor Critical, time ________

Turbine On-Line, time ________

HRD/HBT/RLG
November 4, 1983

1.2 POST-TRIP REVIEW - DATA AND INFORMATION CAPABILITY

Each unit has three primary sources for collecting data for analyzing unscheduled reactor shutdowns. These are the:

(1) Events Recorder,
(2) Alarm Typer, and
(3) Transient Monitor.

The first two are used to determine the sequence of events, while the third is used for analog data trending.

Events Recorder

The events recorder is a dedicated computer system. Inputs from electrical devices are wired to the events recorder cabinets. When an input alarms, this fact is printed out on a paper tape. The time of each event is recorded, with discrimination down to 1 millisecond, along with the parameter number. Some of the inputs also print out when the alarm clears. This tape is removed (torn off) daily and is stored in the station master file. A representative list of parameters monitored by this device is attached. As shown, the Events Recorder input sources include the equipment and components pertinent to reactor trip investigations.

This device is normally powered by a non-class IE AC source. This source is interruptible but will automatically transfer to the onsite AC backup on loss of power. In addition, the events recorder is supplied by a safety-related, non-interruptible DC supply as a backup.

Alarm Typer

The alarm typer is an output device associated with the plant computer. All digital alarms and safety-related pumps, valves, and motors change of status indications that are received in the control room are printed on the typer. Event times are printed out to the nearest second. The data are printed out giving the time, the alarm number, and a description of the alarm or device. The alarm typer printouts are removed daily and stored in the station master file. This device is powered by a non-class IE, non-interruptible AC source.

Transient Monitor

The transient monitor for Oconee is a dedicated computer system. Data are recorded once a second from inputs which are hardwired to the transient monitor device. A list of parameters that are routinely delogged following a trip is attached. The time history is recorded over a 120 minute window, with 60 minutes of pre-trip data and 60 minutes of post-trip data available. The data is stored on a hard disk away from the computer. This device is powered by a non-class IE AC source. This source is interruptible but will automatically transfer to the onsite AC backup on loss of power.

The transient monitor can output data either in graph or table form. For graphs, the timescale and parameter ranges can be specified. Time windows can be as long or as short as desired, within the bounds of collection capability. (See attached

examples.) Hard copies of graphs of major plant parameters are normally kept. Tabular data can be printed, at specified intervals, and stored as well.

The selection of parameters available on the transient monitor was made based on Duke Power's experience with transient analysis. We have several years experience with in-depth analysis of transients at Oconee, and McGuire transients are undergoing similar analysis. The parameters recorded by the transient monitor document the pre- and post-trip behavior of the primary system, secondary system, important BOP systems and equipment, and Safety Systems. The parameters listed have been proven sufficient to analyze plant response and to identify and resolve abnormal system performance.

In addition to the data available from the events recorder, alarm typer, and transient monitor, information is available from the control room strip chart recorders. Operator interviews are routinely conducted as part of the post-trip review process and the control room logbooks are available for additional information.

HRD/HBT/RLG
November 4, 1983

Attachment 1

Event Recorder Inputs

Representative List of Points Available for Post-Trip Review

Reactor Protection System Trips - each channel and parameter
Reactor Trip Breakers - each breaker
Reactor Trip Confirm
Turbine Trips - each signal
Feedwater Pump Trips - each signal and pump
Condensate Booster Pump Trips - each signal and pump
Generator Lockout - each bus or train
Station Battery Trouble Alarm - each battery
4KV Bus Transfer - each bus
7KV Bus Transfer - each bus
Generator Loss of Excitation
Reactor Coolant Pump Trip - each pump

In addition, some Condensate Booster Pump and Main Feedwater Pump status alarms (e.g. low suction pressure) are on the events recorder.

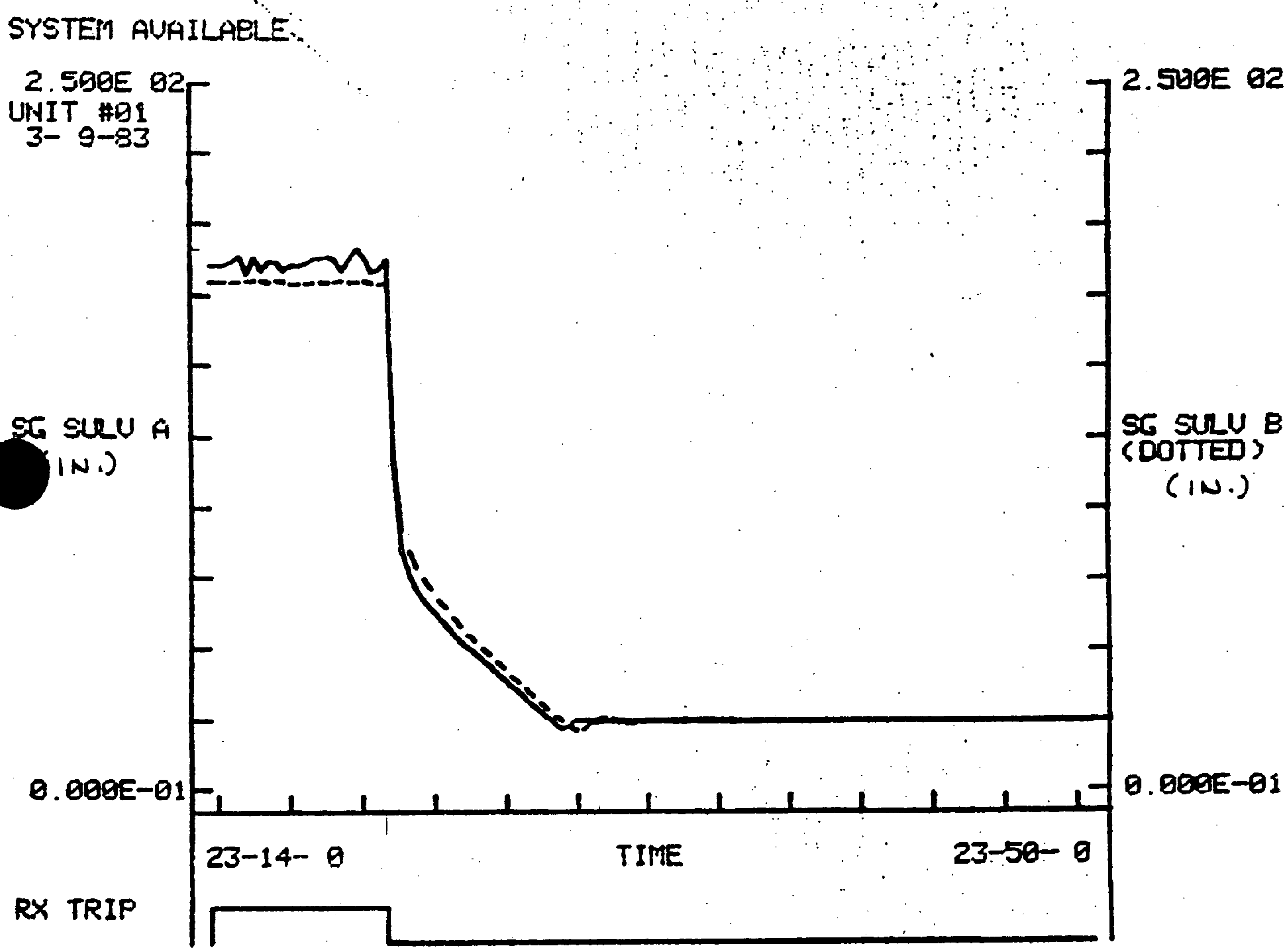
Attachment 2

Oconee Nuclear Station
Standard Transient Monitor Plot List

Parameter		Zero Value	Increment Value
1.	Rx Power	0	10
2.	PZR LVL C	0	40
3.	RC PR WR	1500	100
4.	RC TH A	525	10
	RC TC A	525	10
5.	RC TH B	525	10
	RC TC B1	525	10
6.	RC FLOW A	0	10
	RC FLOW B	0	10
7.	MS PRESS	700	50
8.	SG PRES A	700	50
	SG PRES B	700	50
9.	BP VLV A	0	10
	BP VLV B	0	10
10.	SG SULV A	0	25
	SG SULV B	0	25
11.	FW FLOW A	0	0.6
	FW FLOW B	0	0.6
12.	FW VLV A	0	10
	FW VLV B	0	10
13.	FW TEMP	0	50

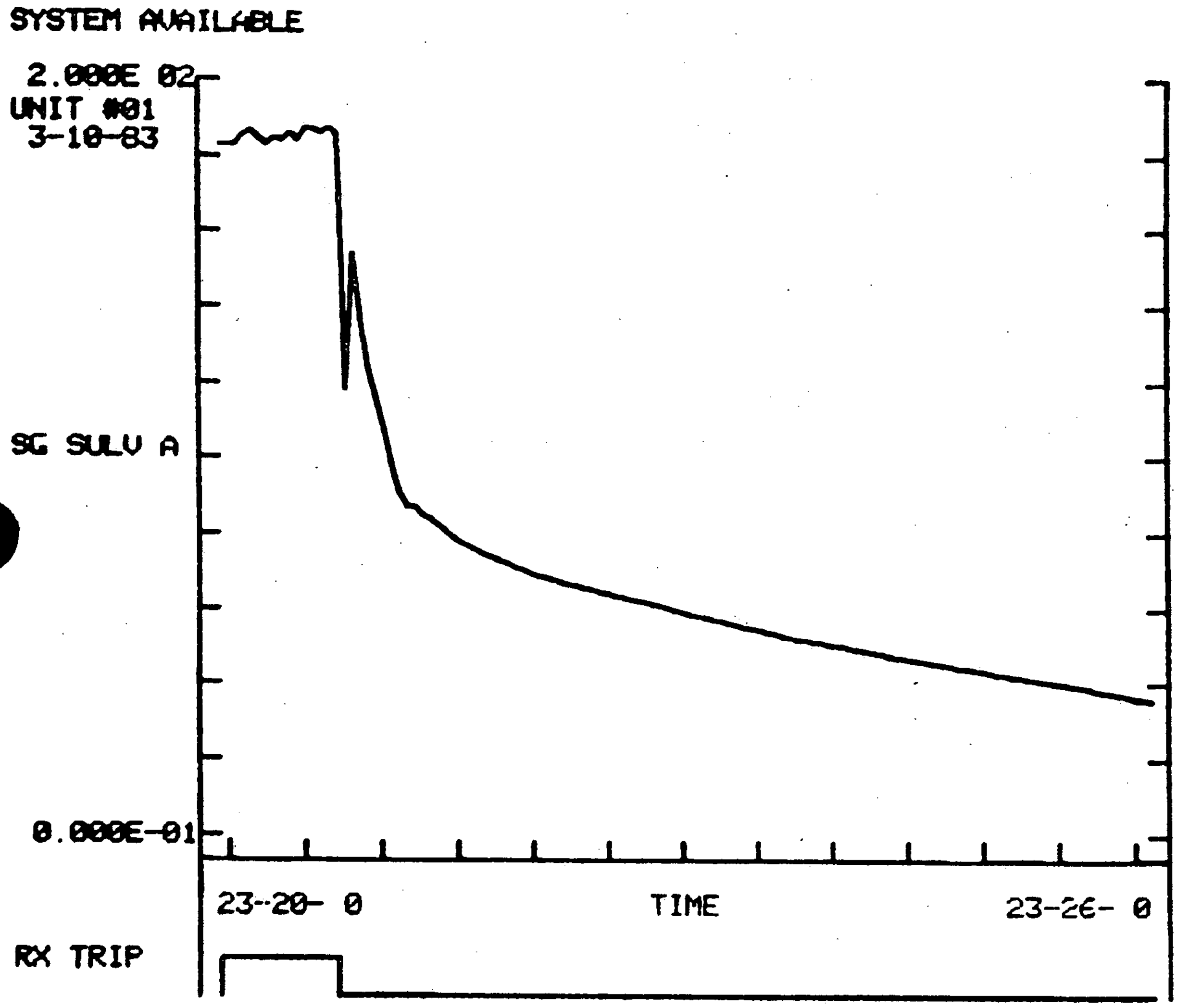
Attachment 3
Page 1

Oconee Unit 1 Reactor Trip
March 9, 1983



Sample Time History Plot

Attachment 3
Page 2



Same Trip, Same Parameter
Narrower "Window"

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

Equipment Classification

As discussed in more detail in response to Item 2.2, Duke has in place at Oconee a program to assure that components are properly identified as safety-related. With respect to components whose function is required to trip the reactor, Duke will evaluate the listing of components developed by the B&W Owners Group and expects to confirm that all items are properly identified on documents, procedures, and information handling systems used in the plant to control safety-related activities.

Vendor Interface

Duke supports the B&W Owners Group response to this item. In addition, Duke Power Company has a program in place for Oconee which ensures evaluation and implementation of vendor information transmitted by B&W. Safety concerns identified through this information are verified and documented in the course of this program. Periodically (not to exceed one year), program records of vendor transmittals are reviewed to ensure receipt of all applicable information.

Upon receiving the vendor information, the vendor information coordinator transmits the information to the appropriate organization within Duke Power to determine applicability and safety significance. If applicable, the information is transmitted to the station staff for implementation. The station staff would incorporate the information into plant procedures and instructions, as appropriate. The vendor information coordinator maintains records of the disposition of each item and provides follow-up to ensure timely resolution.

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

1. For equipment classification --

Station specific documents provide the mechanism for the determination of whether or not a given station structure, system or component is safety-related. This document is provided to appropriate station and General Office personnel and is entitled, "Oconee Nuclear Station Safety-Related Structures, Systems, and Components Manual". This document is approved and issued by the Vice President, Nuclear Production Department, with appropriate administrative procedures implemented to control distribution and future revisions. The manual listed above provides the criteria to be utilized in determining safety-related structures, systems, and components (both mechanical and electrical). In general, an item is considered to be safety-related if necessary to assure:

(a) the integrity of the reactor coolant pressure boundary,

(b) the capability to shut down the reactor and maintain it in a safe shutdown condition, or

(c) the capability to prevent or mitigate the consequences of accidents which could result in potential off-site exposures comparable to the guideline exposures of 10 CFR Part 100.

In addition to the general criteria, the following considerations have been established as guidance in the determination of safety-related mechanical systems and components and are contained in the station manual.

(a) Reactor coolant pressure boundary.

(b) Systems or portions of systems that are designed for emergency core cooling, post-accident containment heat removal or post-accident fission product removal.

(c) Systems or portions of systems required for reactor shutdown or residual heat removal.

(d) Portions of the main steam and feedwater systems from the steam generator to, and including, the first normally closed or automatic isolation valve.

(e) Cooling water and auxiliary feedwater systems required for emergency core cooling, post-accident containment heat removal, post-accident atmosphere cleanup or residual removal from the reactor and from the spent fuel storage pool.

(f) Radioactive waste treatment, handling, and disposal systems, except those portions of these systems whose postulated failure would not result in off-site doses that exceed 0.5 rem whole-body or equivalent.

(g) Instrumentation that is essential to the performance of reactor protection and/or engineered safeguards function.

(h) Instrument piping connected to a nuclear safety-related system. However, the connected instrument is not safety-related unless it performs a necessary safety function.

(i) Containers/packages utilized by a station for off-site shipment of fissile material and quantities of licensed materials in excess of type A quantities, as defined in 10 CFR §71.4(q), are considered safety-related.

The classification of mechanical systems and components primarily considers the functional aspects of a system. In some cases, however, a system or portion of a system is considered safety-related because it services as a pressure boundary. This would include such items as vessel shells, heads and nozzles, pipes, tubes and fittings, valve bodies, bonnets and discs, pump casings and covers, and bolting which joins pressure-retaining items. Specifically excluded are items not associated with the pressure-retaining function of a component such as shafts, stems, trims, bearings, bushings, wear plates or seals, packing, gaskets and valve seats.

Systems and components which may contain radioactive material but whose postulated failure would not result in off-site doses that exceed 0.5 rem whole-body or equivalent, and that are not otherwise related to safe shutdown or accident mitigation, are classified as non-safety-related.

In addition to the general criteria, the following considerations have been established as guidance in the determination of safety-related electrical systems and components and are contained in the station manual.

(a) Electric systems and components that are essential to shut down the reactor and limit the significant release of radioactive material following a design basis event.

(b) Controls and instrumentation systems and components that are utilized to develop a signal which is essential to initiate reactor protection and/or engineered safeguards functions.

(c) Instrumentation systems and components that are essential for post-accident monitoring.

(d) Cables and their support systems that provide power to or control safety-related components.

In addition to the general criteria, the following considerations have been established as guidance in the determination of safety-related structures and are contained in the station manual.

(a) The integrity of the reactor coolant pressure boundary.

(b) The capability to shut down the reactor and maintain it in a safe shutdown condition, or

(c) The capability to prevent or mitigate the consequences of accidents which could result in potential off-site exposures comparable to the guideline exposures of 10 CFR 100.

Based upon the general criteria, and the specific criteria for all systems, components, and structures listed above, each Duke nuclear station has predetermined the safety-related status of systems, structures, and major components. This information is provided in one or more of the following formats as part of the station manual.

(a) Tabulated listings of the plant systems showing individual components as necessary with the safety-related status of each item indicated.

(b) Tabulated listings of the plant structures showing individual areas as necessary with the safety-related status of each item indicated.

(c) Station specific drawings such as system flow diagrams or piping and instrumentation diagrams with the safety-related portions designated by color-coded highlighting.

(d) A criteria checklist for use when an item under consideration is not contained in either of the above formats (provided as Attachment A).

Station personnel use the applicable manual as the basis for determining the safety-related functions of structures, systems, or components. To expedite this process station management procedures (Station Directives) define the use of the manuals. The Station Directives provide guidance in the following areas:

(a) Applicability of the manual.

(b) Responsibilities of the station organizational units in determining safety-related classification of a particular structure, system, or component.

(c) The procedure for use in applying the specific information contained in manuals or in using the Nuclear Safety-Related System and Component checklist.

(d) Personnel qualifications.

(e) Review and Approval Process.

DUKE POWER COMPANY
NUCLEAR PRODUCTION DEPARTMENT
NUCLEAR SAFETY-RELATED SYSTEM AND COMPONENT CHECK LIST

(1) STATION: ______________________________

(2) CHECKLIST APPLICABLE TO: ____

(3) EVALUATION:

(3.1) Is the component part of the pressure boundary of any of the following?

Yes ____ No ____ a. Reactor Coolant System
Yes ____ No ____ b. Any system connected to the Reactor Coolant System out to the second isolation valve
Yes ____ No ____ c. An Engineered Safeguards-actuated system
Yes ____ No ____ d. The reactor containment
Yes ____ No ____ e. A post-accident containment atmosphere cleanup system
Yes ____ No ____ f. A system containing radioactive material, the failure of which could cause off-site doses greater than 0.5 rem whole body

(3.2) Is the component required for the functioning of or actuation of any of the following?

Yes ____ No ____ a. Emergency core cooling
Yes ____ No ____ b. Residual heat removal from the reactor or spent fuel pool (including auxiliary feedwater and secondary cooling systems)
Yes ____ No ____ c. Containment isolation
Yes ____ No ____ d. Post-accident containment heat removal or atmosphere cleanup
Yes ____ No ____ e. Reactor shutdown
Yes ____ No ____ f. A cooling or seal water system required for functioning of systems or components important to safety
Yes ____ No ____ g. Control room habitability
Yes ____ No ____ h. Post-accident monitoring

(3.3) Is the component required to assure any of the following?

Yes ____ No ____ a. The integrity of the reactor coolant pressure boundary
Yes ____ No ____ b. The capability to shutdown the reactor and maintain it in a safe shutdown condition
Yes ____ No ____ c. The capability to prevent or mitigate the consequences of accidents which could result in potential off-site exposures comparable to the guideline exposures of 10 CFR 100.

If the answer to any of the above questions is "yes", then the component must be treated as safety-related.

(4) PREPARED BY: ______________________________ DATE: ______________

(5) REVIEWED BY: ______________________________ DATE: ______________

If a structure, system, or component is identified as being safety-related, in accordance with the provisions of the Safety-Related Structures and Components Manual, then the activities affecting the quality of the identified item are controlled to the extent consistent with 10 CFR 50, Appendix B. The Duke Power Company Topical Report, "Quality Assurance Program Duke-1", describes the Duke program for conformance to 10 CFR 50, Appendix B.

During the development of the Oconee safety-related manual an inter-departmental review of the draft document is conducted to ensure completeness, technical accuracy, and convert application of the safety-related designation criteria.

Additionally, an established procedure exists in the Licensing Section Manual concerning revisions to the safety-related or quality standards documents. This procedure defines the following aspects of the revision process.

(a) Organizational Responsibilites

(b) Determination of Necessary Revisions

(c) Frequency of Revision

(d) Development of the Document Revision

(e) Determination of Safety-Related Designation

(f) Approval Process

(g) Distribution and Documentation

Duke Power Company has developed management procedures and controls which cover the utilization of the safety-related or quality standards documents at each nuclear station. The Administrative Policy Manual for Nuclear Stations contains requirements for determining the safety-related status of an affected item in each of the following activities.

(a) Station Modifications

(b) Station Maintenance Activities

(c) Procedure Development and Revision

(d) Procurement

Each activity requires the determination and documentation of the safety status of the affected item. Appropriate management approval is received in the documentation package. If an item cannot be shown to be non-safety-related, based on the established criteria, it is considered to be safety-related with all applicable QA requirements enforced.

All documentation packages (procedures, work requests, etc.) for activities affecting a safety-related structure, system, or component receive an interdisciplinary station review including the Quality Assurance Department.

Item 2.2.1.5 requires a demonstration of appropriate design verification, qualification testing, procurement control, and receipt of limits-of-life test documentation for safety-related components. The response for this item will be common for all three of Duke's nuclear stations and requires a thorough interdepartmental review of a draft position which is presently being developed. January 15, 1984 is proposed as the date for submitting Duke's response on this section to the NRC.

2. For vendor interface --

A Nuclear Utility Task Action Committee (NUTAC) has been established for the specific purpose of defining an appropriate industry-wide vendor interface program. Duke Power is participating in the activities of this NUTAC. Following completion of this activity, currently planned for February 1984, Duke Power will submit a description of our vendor interface program.

3.1 POST-MAINTENANCE TESTING (REACTOR TRIP SYSTEM COMPONENTS)

1. Post-maintenance operability testing is performed on the GE AK-2 breakers in the reactor trip system. The breakers are periodically bench tested and a functional test is performed to verify that the breaker is capable of performing its intended function prior to return to service. The general requirements for assuring equipment is capable of performing its safety function before return to service is provided in response to item 3.2.1. In addition to the above, Duke will review this item with respect to the specific components developed by the generic B&W Owners Group effort in this area.

2. Initial testing and maintenance procedures for the above breakers were developed with the assistance of General Electric and B&W. All known recommendations from General Electric and B&W have been incorporated into these procedures. The latest GE Service Advice for the AK-2 breakers will be incorporated into the maintenance procedure by December 31, 1983.

   Duke Power is reviewing vendor recommendations to ensure that any appropriate test guidance for the reactor trip system is incorporated in the station procedures at Oconee. In the course of this review, Duke Power will verify that any resulting procedure changes are in compliance with Technical Specifications. A statement of work completion will be provided to the NRC on or before December 31, 1984.

3. To the best of our knowledge, Oconee Technical Specifications do not contain test requirements which can be demonstrated to degrade rather than enhance safety.

HRD/HBT/RLG
November 4, 1983

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

1. Oconee has in place a program to assure that post-maintenance testing of all safety-related equipment is conducted and that such testing demonstrates that the equipment is capable of performing its safety functions. (The criteria for safety-related equipment are described in response to Item 2.2.). The Oconee Technical Specifications define those systems whose operability must be controlled and the required surveillance necessary to determine operability. For example, the Reactor Protective System is defined and includes all rod drive protective trip breakers and activating relays or coils. The Technical Specifications also require that procedures be in place to govern operation of all systems and components involving nuclear safety; preventive or corrective maintenance which could affect nuclear safety; and nuclear safety-related periodic test procedures.

   Additionally, the Duke Nuclear Production Department Administrative Policy Manual for Nuclear Stations provides guidance in the areas of testing procedures, maintenance procedures, independent verification of return of systems to operation following test or maintenance, and periodic review of procedures.

   An Oconee station directive exists which defines the basic areas of responsibility and control for the maintenance of equipment, structures, or components that are directly related to the functioning of the Oconee Nuclear Station. This directive states that functional verification shall be indicated as required on all safety-related components following maintenance. Completion of functional verification shall be documented by Operations in operational logs or procedures. Documentation of the functional verification will be the responsibility of a Maintenance Supervisor responsible for the work and shall be documented on the work request by stating the method and results obtained. The group accepting the work is responsible for ensuring all necessary functional verification and/or retest, necessary to satisfy Technical Specifications, is complete and documented prior to returning the component to service. Work requests involving safety-related components require correct component verification and independent verification which enhances safety by confirming independently that the correct system/component on the correct unit is properly being removed from or returned to service.

   Maintenance or testing performed on safety-related components requires an approved written procedure.

   Another station directive provides guidelines for determining when a retest or functional verification is required. This directive also lists the components which require performance testing after replacement, repair, modification, and/or significant maintenance and provides administrative controls to ensure proper identification, scheduling and performance of such required testing. These procedures are periodically reviewed to assure conformance with existing requirements.

Based on this program, Duke considers that the Oconee procedures are effective in assuring that safety-related equipment is operable upon return to service.

2. Existing administrative procedures are established to control the distribution of vendor instruction manuals and to ensure their guidance is appropriately incorporated into plant procedures. However, changes to these administrative procedures are planned to resolve weaknesses identified during internal audits. In addition, Duke Power is implementing a substantial review to verify that (1) all vendor instruction manuals are complete and consistent, (2) the information in the vendor manuals is appropriately incorporated into plant procedures. This review will involve comparing all in-house copies of vendor manuals to ensure that all copies of a given manual are complete and consistent. If differences are identified between the copies of a given vendor manual, the differences will be resolved -- the vendor may be contacted, if necessary, to resolve differences. Any changes made as a result of this review will be appropriately incorporated into plant procedures.

3. To the best of our knowledge, Oconee Technical Specifications do not contain test requirements which can be demonstrated to degrade rather than enhance safety.

4.1 REACTOR TRIP SYSTEM RELIABILITY (VENDOR-RELATED MODIFICATIONS)

For the General Electric Type AK-2-25 Reactor Trip Breakers in service at Oconee, no vendor recommendations for breaker modifications have been issued. The only modification planned is discussed in response to Item 4.3.

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

The two AC and four DC Reactor Trip Breakers on each Oconee unit are maintained in accordance with standard practices for items designated as nuclear safety-related. Since the spring of 1979 the maintenance program for these breakers has required that preventive maintenance be performed at six month intervals.

On-line performance of preventive maintenance requires the use of spare breakers as replacements. The spare breakers utilized for this temporary service have also had preventive maintenance performed within the preceeding six months.

The Oconee AK-2 breakers did experience a period of less-then-desirable trip reliability during late 1978 and early 1979, as documented in LERs describing on-line RPS surveillance failures. A timely on-site review of the failures was conducted utilizing a manufacturer's senior engineer. Station procedures were revised immediately and thereby complied in advance with GE Service Advice 175 (CPDD) 9.3 dated April 2, 1979. Thus, critical dimensional and trip torque valves have been periodically checked, with appropriate periodic lubrication performed.

The adequacy of the 1979 maintenance revisions and pick-up voltage setpoint change on the undervoltage trip device has been demonstrated by the numerous successful surveillance trip tests since that time. The handling of that issue demonstrates Duke Power's commitment to identify and promptly resolve adverse equipment performance trends, utilizing the manufacturer when appropriate.

General Electric issued a supplement to the 1979 Service advice on April 15, 1983 (175-9.3.S). The intent of the instruction supplement is believed to be met by the present maintenance program, with the following clarifications:

1) By December 31, 1983 the Oconee AK-2 reactor trip breaker maintenance procedure will be revised to include the new manufacturer's tolerance limits for: (a) the undervoltage trip attachment pickup voltage at room temperature and (b) the clearance check for the undervoltage armature to trip latch (positive trip overtravel).

2) The existing six month maintenance interval will be retained without tracking the number of breaker operations. The supplement suggests a twelve month interval or 1750 cycles, whichever occurs first. Conservative estimates indicate that each Oconee breaker has been tripped via RPS/undervoltage device circuits less than one hundred times per year since commercial operation. No adverse environmental effects have been noted on the six month intervals (i.e., lubricant aging, cleanliness, etc.).

3) The supplement recommends removal of the undervoltage device to check a factory clearance between a rivet and armature, a check which cannot be performed with the undervoltage device in place. This removal/replacement activity could create an additional physical interference through improper reinstallation of the device, in our opinion. The Oconee procedure will continue to require a check to ensure no drag exists between the rivet and armature, a technique which has been very effective since early 1979.

In summary, the Oconee trip breaker maintenance program has been proven to be effective and adequate to ensure reliability of the RPS/undervoltage device trip function. Manufacturer's recommendations have been followed to a great extent. The present maintenance interval has been demonstrated to be adequate. Duke plans to support the B&W Owners Group effort in response to this item.

HRD/HBT/RLG
November 4, 1983

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

Duke will submit to the NRC by December 31, 1983 a description and schedule for adding an automatic RPS shunt trip feature for the Oconee reactor trip breakers. This submittal will address the plant-specific design issues noted in the generic SER provided by NRC letter dated September 12, 1983 (H. R. Denton to H. B. Tucker) which addressed the generic aspects of the design.

HRD/HBT/RLG
November 4, 1983

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

Surveillance Testing

Oconee performs on-line functional surveillance testing of individual Reactor Protective System channels, utilizing procedures classified as safety-related. The diverse SCR trip feature is actuated by RPS channels C and D which are tested once per month during on-line conditions. These particular tests are to be revised to include adequate documented verification that the associated SCRs have appropriately responded to a RPS signal. Vendor guidelines for performing these tests have been received. All necessary procedure revisions will be completed by December 31, 1983.

Maintenance Procedures

Requirements for the content of safety-related procedures are primarily derived from ANSI N18.7. Typically, these include the use of calibrated test instruments, independent verification, functional verification following maintenance, documented instructions and independent review of procedure content by qualified reviewers.

The maintenance of the SCRs will be performed at Oconee under the control of a procedure which complies with all requirements of the safety-related procedures utilized at the affected stations. This procedure will be developed and approved by December 31, 1983. Any future revisions to these procedures would be controlled in the manner typical for safety-related procedures.

Since the Generic Letter 83-28 Section 4.4 is not to be interpreted as requiring hardware changes or additional equipment qualification, maintenance revisions should not require any revision in material procurement handling activities. Replacement materials utilized during repairs are selected based upon their durability for industrial use and meet established manufacturer's criteria which ensures adequate quality for various end users. No unusual material storage or handling methods are necessary for typical industrial grade electronic components such as those utilized in the SCR circuitry. Revisions to replacement parts procurement and storage methods do not appear warranted for SCR circuitry and future treatment typical for "balance plant" components is expected to yield continued satisfactory results. Reclassification of the Control Rod Drive electronic controls to a "safety-related" component would therefore produce no added assurance of functional integrity and would in fact contradict the Generic Letter notification that no hardware changes or additional environmental or seismic qualification of these components are required.

Technical Specifications

Existing specifications are being reviewed to determine what, if any, changes should be proposed. Presently, no changes are required.

HRD/HBT/RLG
November 4, 1983

4.5 REACTOR TRIP SYSTEM RELIABILITY (SYSTEM FUNCTIONAL TESTING)

1. The diverse shunt trip and undervoltage trip features will be tested periodically to verify independent operability per the procedures to be provided in the submittal for the shunt trip modification pre-implementation NRR review.

   The diverse SCR trip feature at Oconee will be tested per the response to Section 4.4 of this Generic Letter.

2. Periodic on-line testing features are designed into existing Oconee trip circuitry and will be designed into the shunt trip modification.

3. Duke intends to support the B&W Owners Group in response to this item. Our present position is that the monthly testing is adequate.