Byzro

November 9, 1982

Docket No. 50-155
LS05-82-11-022

Mr. David J. VandeWalle
Nuclear Licensing Administrator
Consumers Power Company
1945 West Parnall Road
Jackson, Michigan 49201

Dear Mr. VandeWalle:

SUBJECT: SEP TOPIC VI-10.A, TESTING OF REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURES, INCLUDING RESPONSE TIME, FINAL SAFETY EVALUATION REPORT (BIG ROCK POINT)

Enclosure 1 is our contractor's final technical evaluation of this topic for your plant. The report has been modified to reflect the additional information and comments provided in a letter from R. A. Vincent to D. M. Crutchfield dated March 29, 1981.

Enclosure 2 is the staff's final safety evaluation report (SER) on this topic. Our evaluation, which is based on Enclosure 1, proposes modifications to the Technical Specifications and some equipment to implement additional response time testing.

The need to actually implement these changes will be determined during the integrated safety assessment. This topic assessment may be revised in the future if your facility design is changed or if NRC criteria relating to this topic are modified before the integrated assessment is completed.

Sincerely,

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

Enclosure:
As stated

cc w/enclosure:
See next page

SEO4
DSU USE EX (18)

JVW

OFFICE	SEPB:DL	SEPB:DL	SEPB:DL	ORB#5:PM	ORB#5:BC	AD:SA:DL	
SURNAME	RScholl:dk	RHermann	WRussell	REmch	DCrutchfield	FMiraglia	
DATE	10/21/82	10/25/82	11/3/82	11/5/82	11/8/82	10/ /82	

Mr. David J. VandeWalle

cc
Mr. Paul A. Perry, Secretary
Consumers Power Company
212 West Michigan Avenue
Jackson, Michigan 49201

Judd L. Bacon, Esquire
Consumers Power Company
212 West Michigan Avenue
Jackson, Michigan 49201

Joseph Gallo, Esquire
Isham, Lincoln & Beale
1120 Connecticut Avenue
Room 325
Washington, D. C. 20036

Peter W. Steketee, Esquire
505 Peoples Building
Grand Rapids, Michigan 49503

Alan S. Rosenthal, Esq., Chairman
Atomic Safety & Licensing Appeal Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Mr. John O'Neill, II
Route 2, Box 44
Maple City, Michigan 49664

Mr. Jim E. Mills
Route 2, Box 108C
Charlevoix, Michigan 49720

Chairman
County Board of Supervisors
Charlevoix County
Charlevoix, Michigan 49720

Office of the Governor (2)
Room 1 - Capitol Building
Lansing, Michigan 48913

Herbert Semmel
Counsel for Christa Maria, et al.
Urban Law Institute
Antioch School of Law
2633 16th Street, NW
Washington, D. C. 20460

U. S. Environmental Protection
Agency
Federal Activities Branch
Region V Office
ATTN: Regional Radiation Representative
230 South Dearborn Street
Chicago, Illinois 60604

Peter B. Bloch, Chairman
Atomic Safety and Licensing Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

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

Mr. Frederick J. Shon
Atomic Safety and Licensing Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Big Rock Point Nuclear Power Plant
ATTN: Dave Hoffman
Plant Superintendent
Charlevoix, Michigan 49720

Christa-Maria
Route 2, Box 108C
Charlevoix, Michigan 49720

William J. Scanlon, Esquire
2034 Pauline Boulevard
Ann Arbor, Michigan 48103

Resident Inspector
Big Rock Point Plant
c/o U.S. NRC
RR #3, Box 600
Charlevoix, Michigan 49720

Hurst & Hanson
311 1/2 E. Mitchell
Petoskey, Michigan 49770

Mr. David J. VandeWalle

cc
Dr. John H. Buck
Atomic Safety and Licensing Appeal Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Ms. JoAnn Bier
204 Clinton Street
Charlevoix, Michigan 49720

Thomas S. Moore
Atomic Safety and Licensing Appeal Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

James G. Keppler, Regional Administrator
Nuclear Regulatory Commission, Region III
799 Roosevelt Road
Glen Ellyn, Illinois 60137

EGG-EA-6029

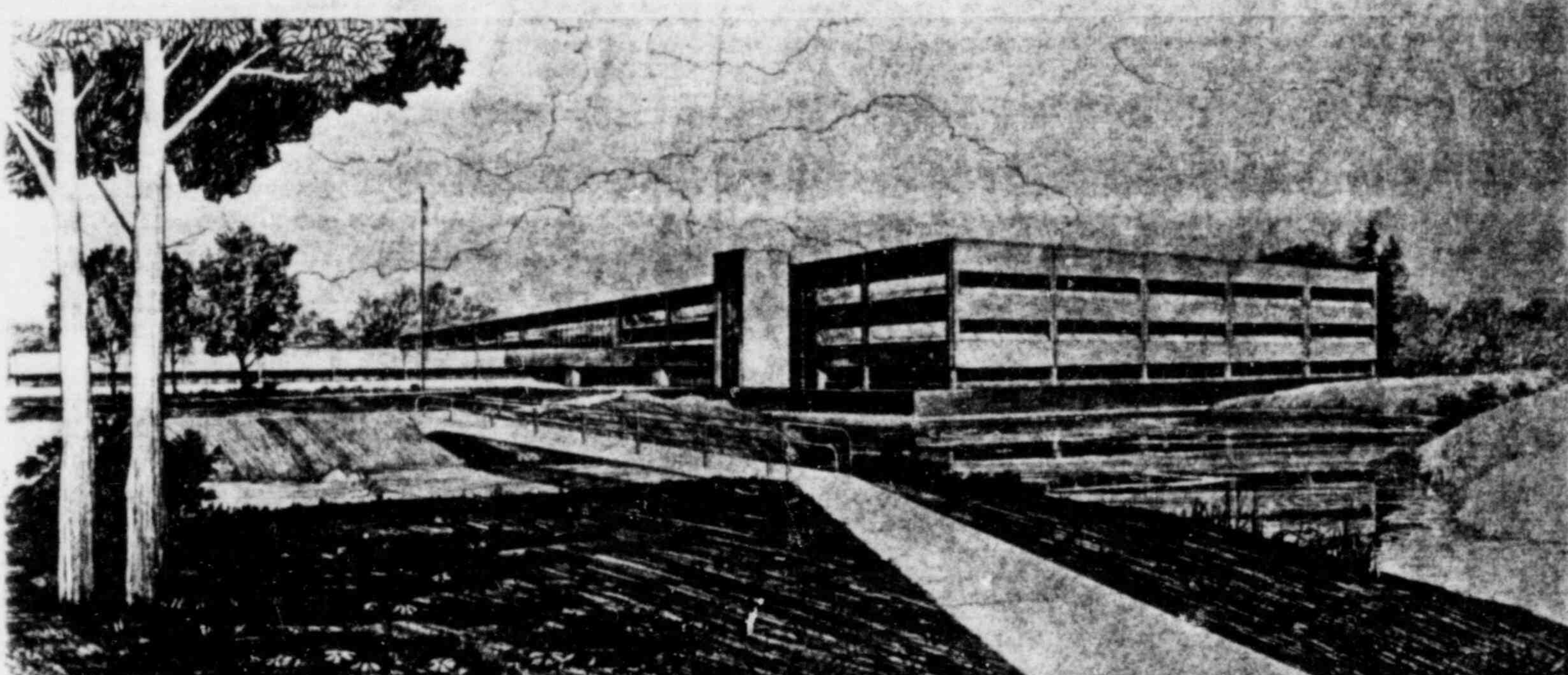
SEPTEMBER 1982

SYSTEMATIC EVALUATION PROGRAM, TOPIC VI-10.A

TESTING OF REACTOR PROTECTION SYSTEM AND ENGINEERED SAFETY FEATURES, BIG ROCK POINT PLANT

R. VanderBeek
D. J. Morken

Idaho National Engineering Laboratory
Operated by the U.S. Department of Energy



This is an informal report intended for use as a preliminary or working document

8211120319

Prepared for the
U.S. NUCLEAR REGULATORY COMMISSION
Under DOE Contract No. DE-AC07-76ID01570
FIN No. A6425

EG&G Idaho

EG&G Idaho Inc

FORM EG&G-398
(Rev 03-82)

Accession No. ____________

Report No. EGG-EA-6029

Contract Program or Project Title:

Electrical, Instrumentation, and Control Systems
Support for the Systematic Evaluation Program (II)

Subject of this Document:

Systematic Evaluation Program, Topic VI-10.A, Testing of
Reactor Protection System and Engineered Safety Features,
Big Rock Point Plant

Type of Document:

Letter Report

Author(s):

R. VanderBeek
D. J. Morken

Date of Document:

September 1982

Responsible NRC Individual and NRC Office or Division:

R. F. Scholl, Jr., Division of Licensing

EG&G Idaho, Inc.
Idaho Falls, Idaho 83415

Prepared for the
U.S. Nuclear Regulatory Commission
Washington, D.C.
Under DOE Contract No. DE-AC07-76ID01570
NRC FIN No. A6425

SYSTEMATIC EVALUATION PROGRAM

TOPIC VI-10.A
TESTING OF REACTOR PROTECTION SYSTEM AND
ENGINEERED SAFETY FEATURES

BIG ROCK POINT PLANT

Docket No. 50-155

September 1982

R. VanderBeek
D. J. Morken

Reliability and Statistics Branch
Engineering Analysis Division
EG&G Idaho, Inc.

9/21/82

ABSTRACT

This SEP Technical Evaluation, for the Big Rock Point Plant, reviews the scope and frequency of periodic testing of the Reactor Protective System and the Engineered Safety Features and compares the required testing against current licensing criteria.

FOREWORD

This report is supplied as part of the "Electrical, Instrumentation, and Control Systems Support for the Systematic Evaluation Program (II)" being conducted for the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Division of Licensing, by EG&G Idaho, Inc., Reliability and Statistics Branch.

The U.S. Nuclear Regulatory Commission funded the work under the authorization B&R 20-10-02-05, FIN A6425.

NRC FIN No. A6425--Electrical, Instrumentation, and Control Systems Support for the Systematic Evaluation Program (II)

CONTENTS

TABLES

SYSTEMATIC EVALUATION PROGRAM

TOPIC VI-10.A
TESTING OF REACTOR PROTECTION SYSTEM AND
ENGINEERED SAFETY FEATURES

BIG ROCK POINT PLANT

1.0 INTRODUCTION

The objective of this review is to determine if all Big Rock Point (BRP) Reactor Protection System (RPS) components, including pumps and valves, are included in component and system tests, if the scope and frequency of periodic testing is adequate, based on comparison with current Standard Technical Specifications (STS)[7] and if the test program meets current licensing criteria. The review will also address these same matters with respect to the engineered safety features (ESF) systems.

2.0 CRITERIA

General Design Criterion 21 (GDC 21), "Protection System Reliability and Testability,"[1] states, in part, that:

> The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failure and losses of redundancy that may have occurred.

Regulatory Guide 1.22, "Periodic Testing of the Protection System Actuation Functions,"[2] states, in Section D.1.a, that:

The periodic tests should duplicate, as closely as practicable, the performance that is required of the actuation devices in the event of an accident;

and further, in Section D.4, it states that:

When actuated equipment is not tested during reactor operation, it should be shown that:

a. There is no practicable system design that would permit operation of the actuated equipment without adversely affecting the safety or operability of the plant,

b. The probability that the protection system will fail to initiate the operation of the actuated equipment is, and can be maintained, acceptably low without testing the actuated equipment during reactor operation, and

c. The actuated equipment can be routinely tested when the reactor is shut down.

IEEE Standard 338-1977, "Periodic Testing of Nuclear Power Generating Station Class 1E Power and Protection Systems,"[3] states, in part, in Section 3:

Overlap testing consists of channel, train, or load-group verification by performing individual tests on the various components and subsystems of the channel, train or load group. The individual component and sub-system tests shall check parts of adjacent subsystems, such that the entire channel, train or load group will be verified by testing of individual components or subsystems.

and, in part, in Section 6.3.4:

Response time testing shall be required only on safety systems or subsystems to verify that the response times are within the limits of the overall response times given in the Safety Analysis Report.

Sufficient overlap shall be provided to verify overall system response.

The response time shall include as much of each safety system, from sensor input to actuated equipment, as is practicable in a single test. Where the entire set of equipment from sensor to actuated equipment cannot be tested at once, verification of system response time shall be accomplished by measuring the response times of discrete portions of the system and showing that the sum of the response times of all is within the limits of the overall system requirement.

In addition, the following criteria are applicable to the ESF: General Design Criterion 40 (GDC 40), "Testing of Containment Heat Removal System,"[4] states that:

The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure:

a. The structural and leaktight integrity of its components.

b. The operability and performance of the active components of the system.

c. The operability of the system as a whole and under conditions as close to the design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection systems, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

GDC 37, "Testing of Emergency Core Cooling System," GDC 43, "Testing of Containment Atmosphere Cleanup Systems," and GDC 46, "Testing of Cooling Water System," are similar.

Standard Review Plan, Section 7.1, Appendix B, "Guidance for Evaluation of Conformance to IEEE STD 279,"[5] states, in Section 11, that:

> Periodic testing should duplicate, as closely as practical, the overall performance required of the protection system. The test should confirm operability of both the automatic and manual circuitry. The capability should be provided to permit testing during power operation. When this capability can only be achieved by overlapping tests, the test scheme must be such that the tests do, in fact, overlap from one test segment to another.

Also applicable are Standard Technical Specifications for General Electric Boiling Water Reactors, Tables 4.3.1.1-1, 4.3.2.1-1, and 4.3.3.1-1.[7]

3.0 REACTOR PROTECTION SYSTEM

3.1 Description

The Reactor Protection System (RPS) includes the sensors, amplifiers, logic and other equipment essential to the monitoring of selected nuclear power plant conditions. It must reliably effect a rapid shutdown of the reactor if any one or a combination of parameters deviates beyond pre-selected values to mitigate the consequences of a postulated design basis event.

The RPS consists of two parallel channels, each having separate power supplies and separate trains of sensor trip contacts. Each channel is designed on a fail safe principle, i.e., de-energization will cause a scram. The RPS is designed so that both channels must be de-energized to cause a scram and must be reset manually subsequent to a scram and prior to startup. The RPS parameters and channel logic as identified in the Big Rock Point Technical Specifications[6] are:

Sensor and Trip Device	Trip Contacts in Each Channel	Coincidence in Each Channel
High Reactor Pressure	2	1 out of 2
Low Reactor Water Leve	2	1 out of 2
Low Steam Drum Water Level	2	1 out of 2
Main Steam Line Backup Isolation Valve Closure	2	1 out of 2
High Condenser Pressure	2	1 out of 2
High Enclosure Pressure	2	1 out of 2
High Scram Dump Tank Level	2	1 out of 2
Recirculation Line Valves Closed	12	1 set out of 2
High Neutron Flux	3	2 out of 3
Short Period	2	1 out of 2
Manual Scram	1	1 out of 1
Loss of Auxiliary Power Supply	1	1 out of 1
Protection Against Picoammeter Failure	3	1 downscale 1 upscale

3.2 Evaluation

The Standard Technical Specifications (STS) for General Electric Boiling Water Reactors incorporate all the criteria identified in Section 2 of this report. The evaluation of the RPS is made by comparing BRP Technical Specifications testing and surveillance requirements, insofar as possible, with the requirements for similar functions in the STS.

Table 1 provides a comparison of the testing and surveillance requirements of the BRP RPS as specified in the BRP Technical Specifications, supplemented by plant test procedures, and as specified in the STS. Since the STS was prepared for use with BWR Class 5 and BRP is a BWR Class 1, the RPS functions are not necessarily the same for the two technical specifications. This is taken into account in the evaluation.

Table 1 was originally prepared by BRP based on the surveillance and testing requirements in Section 6.1.1 and 6.1.2 of the BRP Technical Specifications and supplemented by BRP Test Procedures. The table has been

TABLE I. REACTOR PROTECTION SYSTEM SURVEILLANCE

RPS Input Trip Function[a]	Channel Check		Channel Functional Test		Channel Calibration		Modes For Which Surveillance Is Required	
	BRP	STS	BRP	STS[b]	BRP	STS	BRP	STS
High Reactor Pressure	NP[c,d]	S	R[e]	M	R[e]	R	PO[f]	PO,SU
Low Reactor Water Level	NP[c,d]	S	R[g]	M	R[g]	R	PO[f]	PO,SU
Low Steam Drum Water Level	NP[c,d]	NA	R[h]	NA	R[h]	NA	PO[f]	NA
Main Steam Line Backup Isol. Valve Closure	NP[c,d]	NA	R[i]	M	R[i]	R	PO[f]	PO
High Condenser Pressure	NP[c,d]	NA	R[j]	NA	R[j]	NA	PO[f]	NA
High Enclosure Pressure	NP[c,d]	NA	R[k]	M	R[k]	Q	PO[f]	PO,SU
High Scram Dump Tank Level	NP[c,d]	NA	R[l]	M	R[l]	R	PO[f]	PO,SU, R
Recirculation Line Valves Closed	NP[c]	NA	R[m]	NA	R[m]	NA	PO[f]	NA
High Neutron Flux	H[n]	SU,S	M,R[o]	SU,W	W[o],V[p]	SA, W	PO[f]	PO
Short Period	H[n]	NA	M,R[q]	NA	V[r]	NA	PO[f]	NA
Loss Auxiliary Power Supply[s]	NP[d]	NA	R[t]	NA	R[t]	NA	PO[f]	NA
Protection Against Picoammeter Circuit Failure	NP[u]	NA	R[o]	NA	R[o]	NA	PO[f]	NA

TABLE 1. (continued)

RPS Input Trip Function[a]	Channel Check		Channel Functional Test		Channel Calibration		Modes For Which Surveillance Is Required	
	BRP	STS	BRP	STS[b]	BRP	STS	BRP	STS
Manual Scram	NP[d]	NA	R[v]	NA	NA	NA	PO[f]	PO,SU, HS,CS, R
RPS Logic and Power Switch Operation	W[w]	NA	M,R[w]	NA	NA	NA	PO[f]	R

a. A description of each function is given in letter: R. A. Vincent to D. M. Crutchfield dated 3/29/82 regarding SEP Topic VI-10.A.

b. Also referred to as "High Reactor Building Pressure."

c. Continuously monitored by RPS logic and operational event recorder, control room alarm and RPS channel scram occurs if parameter exceeds RPS trip setpoint.

d. Channel checks, as defined by the Standard Technical Specifications, are not used to make a qualitative assessment of channel behavior.

e. Calibrated per procedure IRPS-1, functionally tested per procedure TR-32.

f. Refer to paragraphs 1.2.1, 6.1.1 and 6.1.2 of Big Rock Point Technical Specifications.

g. Calibrated per procedure IRPS-2, functionally tested per procedure TR-32.

h. Calibrated per procedure IRPS-4, functionally tested per procedure TR-32.

TABLE 1. (continued)

Footnotes (continued)

i. Switch set per procedure TR-32, switch tested per procedure TR-32.

j. Calibrated per procedure IRPS-5, functionally tested per procedure TR-32.

k. Calibrated per procedure IRPS-3, functionally tested per procedure TR-32.

l. Calibrated per procedure IRPS-11, functionally tested per procedure TR-32.

m. Calibrated per procedure IRPS-8, functionally tested per procedure TR-32.

n. Power range channels #1, 2 and 3 (for High Neutron Flux) and intermediate range channels #4 and 5 (for Short Period) meter indication logged by operator.

o. Calibration checked against heat balance weekly (if not more frequently) per T7-06, functionally tested monthly per T30-01 and at refueling per TR-32.

p. Bench calibrated when required per procedure INMS-6.

q. Functionally tested per procedures T30-01 and TR-32.

r. Bench calibrated when required per procedure INMS-8.

s. Also referred to as "RPS Bus Undervoltage."

t. Calibrated per procedure IRPS-7, functionally tested per procedure TR-32.

u. Continuously monitored by RPS logic, channel scram occurs as described in note 7.

v. Functionally tested per procedure TR-32.

TABLE 1. (continued)

Footnotes (continued)

w. Checked weekly per procedure T7-04, tested monthly per T30-01, tested at refueling per procedure TR-32 and TR-33 (timing).

PO - Power Operation

HS - Hot Shutdown

CS - Cold Shutdown

SU - Start Up

H - Hourly

M - Monthly

NA - Not applicable

NP - Not performed

R - Major refueling (not less frequent than once every 12 months)

SA - Semiannually

V - Variable

W - Weekly

modified to include the surveillance and test requirements specified in the STS for the RPS systems. Because of the differences between a BWR Class 1 and a BWR Class 5, six of the BRP RPS are not included in the Standard Technical Specifications. This difference does not effect the evaluation presented below.

3.2.1 Channel Checks

The STS defines Channel Checks as "the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status derived from independent instrument channels measuring the same parameter." The BRP Technical Specifications do not require RPS channel checks. Test Procedure T7-04 does require weekly channel checks of the RPS logic by simulating a High Reactor Building Pressure trip. Operating procedures also require the reactor operator to log the power and intermediate range neutron flux channels each hour.

3.2.2 Channel Calibrations

The STS requires RPS channel calibrations quarterly, semi-annually, or at refueling intervals. BRP Technical Specifications do not require RPS channel calibrations. BRP's RPS channels are calibrated from test procedures on a refueling time interval, but not less frequent than once every 12 months. (See Appendix A for a partial list of test procedures.) With the exception of the Main Steam Line Valve Closure, BRP's calibration intervals are greater than those specified in the STS.

3.2.3 Channel Functional Tes

The BRP Technical Specifications, Section 7.6, establish the requirements for monthly channel operational checks of the RPS and Section 6.1.5 requires functional tests be performed at each refueling interval (but not to exceed 12 months). As can be seen from Table 1, this is less frequent than required by the STS.

3.2.4 P S Response Tests

The STS requires that the RPS response time of each trip function shall be demonstrated to be within its limit at least once every 18 months. The BRP Technical Specifications require only that the RPS response time shall be less than 100 milliseconds. There are no requirements in the BRP Tech-''al Specifications for RPS response time testing. Response time testing of the RPS is accomplished during each refueling interval under the control of test procedures.[9]

The procedures do not include response testing of the RPS sensors. The RPS logic response test consists of measuring the time from the opening of High Reactor Building Pressure (HRBP) sensor trip circuit breaker until the RPS output relays and solenoids are de-energized. This test is repeated for each of the HRBP sensors. Since all sensors with the exception of the Neutron Flux Monitors are bistables, this test is adequate for the RPS logic respons^ test. The response times of the Neutron Flux Monitor channels, which can be critical under certain acc'dent conditions, and have a different input path to the RPS logic, are not measured.

The response time test procedures do not include measuring the time response of the Turbine Trip signal initiation to the Emergency Stop Valve closure.

4.0 ENGINEERED SAFETY FEATURES SYSTEM

4.1 Description

The Big Rock Point Final Hazards Report[8] does not differentiate between the Reactor Protection System (RPS) and the Engineered Safety Features (ESF) systems. Using the definition in the Standard Review Plan, Section 7.3, the ESF is comprised of those systems which are required to function automatically to mitigate the consequences of a postulated design basis accident. Based on this definition and the Big Rock Point Technical Specifications, the following safety systems will be classified as comprising the Big Rock Point ESF for this report:

Emergency Core Cooling System

Containment Spray System

Emergency Condenser System

Containment Isolation System

Reactor Depressurization System

4.2 Evaluation

Table 2 provides a comparison of the requirements for testing and surveillance of the ESF as established by the STS and those established by the BRP Technical Specifications. The information was obtained from the same sources as those identified in Section 3.2 of this report. The following summarizes the results of this evaluation:

4.2.1 Emergency Core Cooling System

The BRP Technical Specifications, Table 11.4.1.4a, require calibration of the core spray initiating sensors at each major refueling, and an instrumentation functional trip test at quarterly intervals. Test procedures are used for performing these calibrations and functional tests. There are no requirements for channel checks or response time measurements in the BRP Technical Specifications. Test procedures do provide for monitoring the actuation times of the various core spray valves. The primary emergency core cooling valves are tested and timed on a monthly basis using test procedures. All other valves in the Core Spray System are tested and timed at each refueling interval. BRP has no requirements or procedures for performing a response time test of the Core Spray System from the sensor input to the completion of the output function.

4.2.2 Enclosure (Containment) Spray

Table 11.4.3.4 and Section 11.4.3.4 of the BRP Technical Specifications require channel functional testing and instrument calibration at each refueling outage but not to exceed 18 months. The BRP Technical Specifications do not require channel checks or response time testing of the system; however, test procedures do require valve response time tests be made and

TABLE 2. ENGINEERED SAFETY FEATURES SYSTEMS SURVEILLANCE

ESF System	Channel Check		Channel Functional Test		Channel Calibration		Modes For Which Surveillance For Is Required	
	BRP	STS	BRP	STS	BRP	STS	BRP	STS
Emergency Core Cooling (Core Spray System)[a]	NP[b]	S	M[c]Q[d] R[g]	M	R[e]	R	PO,R[f]	PO,SU,HS CS,R
Containment (Enclosure) Spray[h]	NP[b]	NA	R[g]	NA	R[i]	NA	PO[j]	NA
Emergency Condenser System	NP[b,k]	S	R[l]	M	R[l]	R	PO[x]	PO,SU
Containment Isolation System	NP	S	M[n]R[o]	M	M[n]R[o]	Q	PO,[n]R[m] SD,CS[p]	PO,SU,HS
Reactor Depressurization System	W[q]	S	M[r]Q[s] R[v]	M	R[t]	R	PO,HS[u]	PO SU,HS

a. Emergency Core Cooling System (both Core Spray System and Core Spray Recirculation System; the latter of which is a strictly manual system) testing is fully described in letter: R. A. Vincent to D. M. Crutchfield dated 6/4/82 regarding SEP Topic VI-7.A.3.

b. Channel checks, as defined by the Standard Technical Specifications, are not performed since initiation channels do not feature indicators which can be used to make a qualitative assessment of channel behavior.

c. Core spray valve stroke is verified using the hand controller per procedure T30-22 (see note 1 above).

d. Pressure switch operation is verified per procedure T90-09 (see note 1 above).

e. Calibration is performed per procedures IPIS-1 and IRPS-2.

f. Refer to BRP Technical Specification 11.3.1.4A.

Footnotes (continued)

g. Functional testing is performed per procedure T180-15.

h. Containment Spray System testing is fully described in letter: R. A. Vincent to D. M. Crutchfield dated 3/29/82 regarding SEP topic VI-10.A.

i. Calibration is performed per procedure IPIS-2.

j. Refer to BRP Technical Specification 11.3.3.4A.

k. Pressure sensors that effect automatic emergency condenser operation are those that initiate RPS scram (see Table 1, Trip Function #1). As described in Table 1, these sensors are continuously monitored by RPS logic and the RPS operational events recorder.

l. Calibration is performed per procedure IRPS-1. Functional testing is performed per procedure TR-34.

m. Functional test of containment isolation as a result of RPS scram performed per procedure TR-52.

n. Calibration of area monitors, which provide containment ventilation isolation from high radiation, is performed by procedure RIP-15. RIP-15 also verifies automatic ventilation valve closure.

o. RPS scram results in full containment isolation. RPS input sensors are calibrated as described in Table 1.

p. Refer to BRP Technical Specification 3.6.

q. Channel checks and verification of automatic operation verified per procedure T7-22.

r. Functional tests to verify proper operation of RDS cabinet electronics performed per procedure T30-28.

s. Functional tests of the RDS depressurization valves are performed per procedure T90-12 to verify valve operability. Functional tests of the RDS isolation valves are performed per procedure T90-07 to verify valve operability.

TABLE 2. (continued)

Footnotes (continued)

t. Calibration is performed using procedures IRDS-1 (reactor water level transmitters), IRDS-2 (steam drum water level transmitters), IRDS-3 (electric fire pump discharge pressure switches) and IRDS-4 (diesel fire pump discharge pressure switches).

u. Refer to BRP Technical Specification 3.1.5.

v. RDS's full actuation capability verified per TR-48.

x. Refer to BRP Technical Specification 4.1.2b.

CS - Cold Shutdown

HS - Hot Shutdown

M - Monthly

NA - Not Applicable

PO - Power Operation

NP - Not Performed

PO - Power Operation

Q - Quarterly

R - Refueling

S - Once every 12 hours

SD - Shutdown

W - Weekly

data taken once each refueling period. There are no requirements or procedures for response testing the Enclosure Spray System sensors or logic circuitry. Consumers Power Company considers the above valve response time testing adequate.[9]

4.2.3 Emergency Condenser System

Section 6.1.5 of the Big Rock Technical Specifications requires functional testing of the Emergency Condenser System not less frequent than once every 12 months. It does not require channel checks, instrument calibration or response testing. Reference 9 implies that test procedures are available for surveillance testing of all safety related systems and that the Containment Spray System is typical. To the extent the Emergency Condenser System is typified by the Containment Spray System surveillance require- ments, as implied in Reference 9, test procedures will require instrumenta- tion sensors to be calibrated and the Emergency Condensor valves closing response time to be measured during each refueling cycle. The system would not be response tested from sensor to valve operation (end to end) as required by IEEE Standard 338.[3] Reference 10 states that test procedure IRPS-1 calibrates the reactor pressure sensors that initiate the Emergency Condenser actuation and IRPS-1 is required by Test Procedure TR-34.

4.2.4 Containment Isolation System

Section 3.7 of the BRP Technical Specifications provides the requirements for testing the Containment Isolation System. Section 3.7a requires that the containment sphere be leak checked at least once every six months. Section 3.7b requires functional testing of the main isolation valves in both the manual and automatic modes at least once every 12 months. It also requires that the automatic controls and instrumentation associated with the valves be tested at quarterly intervals. Reference 10 indicates test procedures are used to perform response time testing of the isolation valves at each refueling but not for the total system. Since Containment Isolation is initiated by the RPS sensors "Reactor Low Water Level" and "Reactor High

Containment Pressure," these sensors are calibrated every refueling cycle as described in Section 3.2.2 of this report. The system response time is not measured from the sensor actuation to the valve closures as required in IEEE Standard 338.[3] System response time testing and calibrations are not specified in the BRP Technical Specification.

4.2.5 Reactor Depressurization System (RDS)

Sections 4.1.5 and Table 4.5.2.h of the BRP Technical Specifications set forth the surveillence requirements for the RDS. They specify that the RDS isolation valves shall be test operated at least once every three months, the depressurization valves will be test operated during cold shutdown, but not more often than once every three months, the instrumentation and system logic will be functionally tested once a month, and the instruments will be calibrated at each major refueling.

The BRP Technical Specifications do not require channel checks or response time testing. To the extent that the RDS is typified by the Containment Spray System, as implied in Reference 9, the valves will be response tested at each refueling interval. The procedures for these tests were not included in Appendix A. There are no requirements for end-to-end response time testing of the RDS.

5.0 SUMMARY

The review of the reference material has determined that parts of the testing and surveillance of the Big Rock Point Plant RPS and ESF do not meet the criteria established in Section 2 of this report. The areas of noncompliance are as follows:

1. The Big Rock Point Technical Specifications do not require calibration of the initiation channels for the RPS, the Emergency Condenser System, and the Containment Isolation System. Calibration of these systems is controlled by plant test procedures;

2. The Big Rock Point Technical Specifications specify response times but do not require response time testing of the RPS and ESF systems. Response tests are controlled by plant test procedures; RPS response test intervals are greater than that specified in the STS;

3. Response testing of the RPS does not include the sensors which initiate RPS action or ESF action. Response testing of the ESF systems does not include the system logic which actuates the valves. It includes only the opening and/or closing time of the valves when actuated from the hand switch in the control room;

4. Neither the BRP Technical Specifications nor the information in Reference 9 requires channel checks be performed at any time as required in the Standard Technical Specifications.[7]

5. Channel functional tests of the RPS are performed at each refueling interval which is not in agreement with the monthly tests required by the STS.

6.0 REFERENCES

1. General Design Criterion 21, "Protection System Reliability and Testability," of Appendix A, "General Design Criteria for Nuclear Power Plants," 10 CFR Part 50, Domestic Licensing of Production and Utilization Facilities, January 1, 1981.

2. Regulatory Guide 1.22, Periodic Testing of the Protection System Actuation Functions, February 17, 1972.

3. IEEE Standard 338-1975, Periodic Testing of Nuclear Power Generating Station Class 1E Power and Protection Systems.

4. General Design Criterion 40, "Testing of Containment Heat Removal Systems," of Appendix A, "General Design Criteria for Nuclear Power Plants," 10 CFR Part 50, Domestic Licensing of Production and Utilization Facilities, January 1, 1981.

5. Nuclear Regulatory Commission Standard Review Plan, Section 7.1, Appendix B, "Guidance for Evaluation of Conformance to IEEE STD 279."

6. Appendix "A" Consumers Power Company Big Rock Point Plant Technical Specifications as amended through February 25, 1981.

7. NUREG-0123, Rev. 3, Standard Technical Specifications for General Electric Boiling Water Reactors, Fall 1980.

8. Final Hazards Summary Report, Vol. 1, "Plant Technical Description and Safeguards Evaluation," Revised March 1962.

9. Consumers Power Company letter, Robert A. Vincent to NRC, D. M. Crutchfield, "Big Rock Point Plant--SEP Topic VI-10.A, Electrical, Instrumentation and Control Portions of the Testing of RTS and ESF," dated March 29, 1981.

10. Consumer Power Company letter, Robert A. Vincent to NRC, D. M. Crutchfield, "Big Rock Point Plant--SEP Topic VI-10.A. Testing of Reactor Trip System and Engineered Safety Features Including Response Time," dated August 5, 1982.

APPENDIX A

A PARTIAL LISTING OF TEST PROCEDURES FOR BIG ROCK POINT RPS AND ESF TESTING AND SURVEILLANCE

APPENDIX A

A PARTIAL LISTING OF TEST PROCEDURES FOR BIG ROCK POINT RPS AND ESF TESTING AND SURVEILLANCE

Number	Title
IRPS-1, Rev. 4	"Calibration and Testing of Reactor Pressure Sensors."
IRPS-2, Rev. 6	"Calibration and Testing of Reactor Water Level Sensors."
IRPS-3, Rev. 4	"Calibration and Testing of Reactor Enclosure Hi-Pressure Scram Sensors."
IRPS-4, Rev. 5	"Calibration and Testing of the Reactor Steam Drum Water Level Sensors."
IRPS-5, Rev. 5	"Calibration and Testing of High Condenser Pressure Scram Sensors."
IRPS-6, Rev. 4	"Calibration and Testing of High Condenser Pressure Scram Bypass Pressure Switches."
IRPS-7, Rev. 3	"Reactor Protection System Undervoltage Breaker Check."
IRPS-8, Rev. 3	"Calibration and Testing of the Reactor Recirculation Pump Valve Position Scram Switches."
IRPS-11, Rev. 3	"Cleaning and Inspection of Scram Dump Tank High Level Sensors."
TR-01, Rev. 8	"Control Rod Drive Performance Testing."
TR-32, Rev. 11	"Reactor Protection System Scram Sensor Test."
TR-33, Rev. 5	"Reactor Protection System Response Time."
SOP 31, Rev. 2	"Nuclear Instrumentation System."
TR-37, Rev. 6	"Control Rod Drive Scram Dump Tank Vent Delay."

TR-52, Rev. 11	"Sphere Isolation Test."
TR-80, Rev. 1	"Master Scram Valve Operability Test."
TR-04, Rev. 7	"Weekly Reactor Protection Logic System Check."
TR-01, Rev. 5	"Monthly Reactor Protection System Test at Power."

SYSTEMATIC EVALUATION PROGRAM
TOPIC VI-10.A

BIG ROCK POINT

TOPIC: VI-1 ., TESTING OF REACTOR TRIP SYSTEM AND ENGINEERED SAFETY ES, INCLUDING RESPONSE TIME TESTING

I. INTRODUCTION

The purpose of this topic is to review the reactor trip system (RTS) and engineered safety features (ESF) test program for verification of RTS and ESF operability on a periodic basis and to verify RTS and ESF response time in order to assure the operability of the RTS and ESF. Response times should not exceed those assumed in the plant accident analyses. Accordingly, the test program of the RTS and ESF was reviewed in accordance with the Standard Review Plan, including applicable Branch Technical Positions.

II. REVIEW CRITERIA

The review criteria are presented in Section 2 of EG&G Report EGG-EA-6029, "Testing of Reactor Trip System and Engineered Safety Features."

III. RELATED SAFETY TOPICS AND INTERFACES

Topic VI-7.A.3 discusses the question of testing protection systems under conditions as close to design condition as practical. There are no topics that are dependent on the present topic information for their completion.

IV. REVIEW GUIDELINES

Review guidelines are presented in Sections 3 and 4 of Report EGG-EA-6029.

V. EVALUATION

As noted in Sections 3 and 4 of Report EGG-EA-6029, Big Rock Point does not satisfy current licensing criteria because:

1. "The Big Rock Point Technical Specifications do not require calibration of the initiation channels for the RPS, the Emergency Condenser System, and the Containment Isolation System. Calibration of these systems is controlled by plant test procedures."

   Our contractor's audit of these procedures is described in Section 3.2, 3.2.1, 3.2.2, and 3.2.3. The only area where the procedures do not conform to the current technical specification requirements is calibration frequency. The licensee has noted that operating experience has demonstrated acceptable failure rates and little instrumentation drift. The staff agrees.

2. "The Big Rock Point Technical Specifications specify response times but do not require response time testing of the RPS and ESF systems. Response tests are controlled by plant test procedures; RPS response test intervals are greater than that specified in the STS."

   As noted previously in this report, the use of procedures is acceptable to the staff. For the Big Rock Point, the staff agrees with the licensee position that operating experience justifies a test interval that is larger than that specified in the Standard Technical Specification (STS).

3. "Response testing of the RPS does not include the sensors which initiate RPS action or ESF action. Response testing of the ESF systems does not include the system logic which actuates the valves. It includes only the opening and/or closing time of the valves when actuated from the hand switch in the control room."

   With regard to the testing of RPS and ESF sensors, the staff noted that neither IEEE Standard 338-1977 nor Regulatory Guide 1.118 requires response time testing of neutron detectors. However, Regulatory Guide 1.118 does recommend the testing of cable capacitance or other suitable test. The remainder of the sensors that provide an input to the protection system logic are snap action, blind sensors. Such sensors are not suitable candidates for response time testing in the field. However, the neutron monitoring cables and signal processing equipment should be response time tested.

   With regard to the ESF valve actuation logic, the staff noted that it is composed of relays that are similar to those found in the RPS and the valve controls. The RPS and valve control relays are response time tested.

   Based on past experience at Big Rock Point, response time testing of the ESF valve logic is unnecessary (assuming that degradation due to wear out will be detected by response time testing of similar components).

4. "Neither the BRP Technical Specifications nor the information in the licensee's March 29, 1981, (letter from R. A. Vincent to D. M. Crutchfield on this topic) requires channel checks be performed at any time as required in the Standard Technical Specifications."

   The staff has noted that the protection system input sensors are blind, bistable devices for which channel checks are not suitable. Procedures require hourly checks of the power range and the intermediate range channels and weekly functional test of the high reactor building pressure trip. The neutron monitoring system is the only analog system that provides an input to the protection logic.

5. "Channel functional tests of the RPS are performed at each refueling interval which is not in agreement with the monthly tests required by the STS."

VI. CONCLUSION

Our contractor has conducted an audit of typical instrumentation that provides input signals to reactor protection system (as defined in General Design Criterion 21) and the protection logic. From the staff's review of our contractor's report, we have determined that some sensors and their signal processing and logic elements are not tested in a manner that satisfies current licensing criteria.

It is the staff's position that the design of systems which are required for safety shall include provisions for periodic verification that the minimum performance of instruments and control is not less than that which was assumed in the safety analyses. The bases for this position are General Design Criterion 21, Section 3.9 of IEEE Standard 279-1971, and IEEE Standard 338-1971.

The need to implement a neutron monitoring system response time test will be determined during the integrated assessment.

The staff has also determined that required instrument calibration is performed in accordance with plant procedures, however, some of these tests are not included in the plant technical specifications. The need to revise the plant technical specifications will be determined during the integrated assessment.