Enclosure 1

EGG-EA-5961

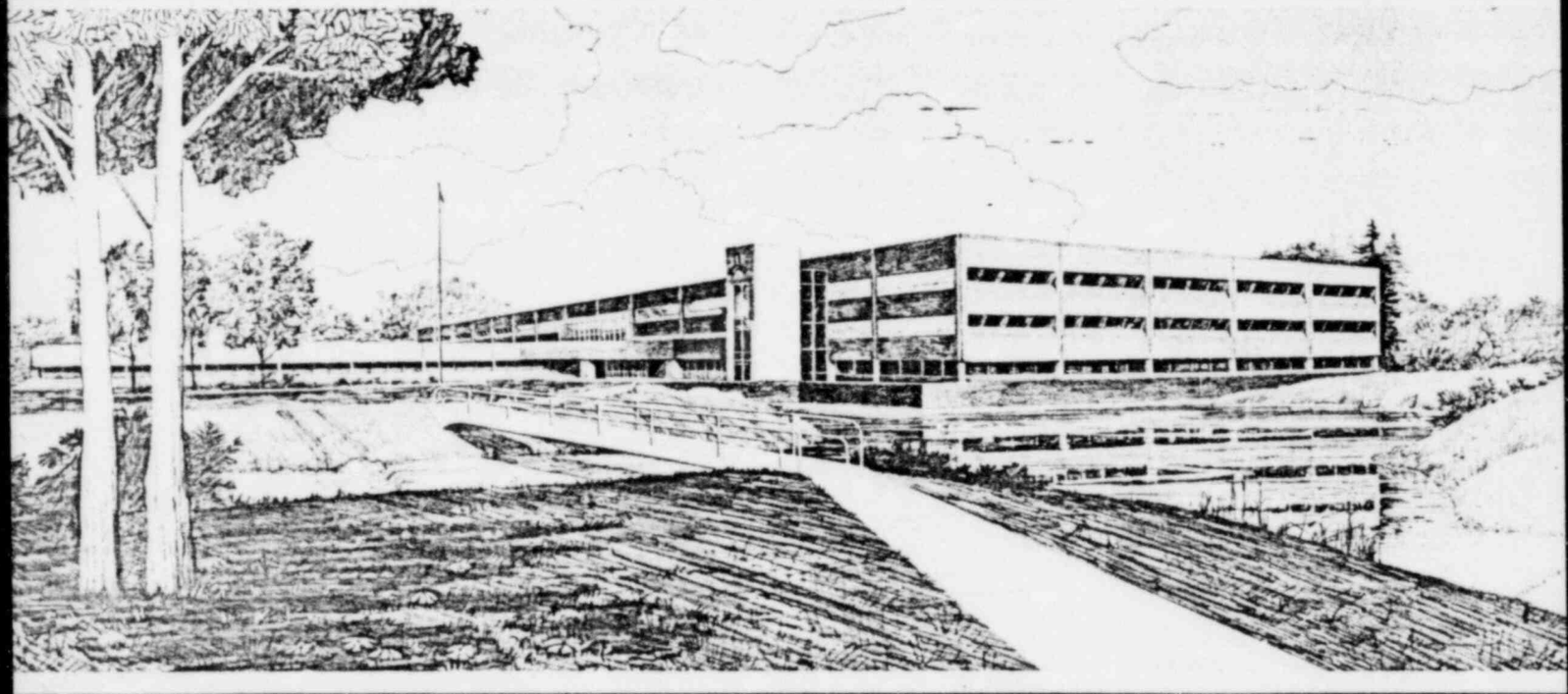
July 1982

SYSTEMATIC EVALUATION PROGRAM, TOPIC VI-7.A.3

ECCS ACTUATION SYSTEM, BIG ROCK POINT

R. VanderBeek

U.S. Department of Energy
Idaho Operations Office • Idaho National Engineering Laboratory



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)

INTERIM REPORT

Accession No. _____

Report No. EGG-EA-5961

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-7.A.3, ECCS Actuation System, Big Rock Point

Type of Document:

Informal Report

Author(s):

R. VanderBeek

Date of Document:

July 1982

Responsible NRC Individual and NRC Office or Division:

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

This document was prepared primarily for preliminary or internal use. It has not received full review and approval. Since there may be substantive changes, this document should not be considered final.

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

INTERIM REPORT

SYSTEMATIC EVALUATION PROGRAM

TOPIC VI-7.A.3
ECCS ACTUATION SYSTEM

BIG ROCK POINT

Docket No. 50-155

July 1982

R. VanderBeek

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

7-19-82

ABSTRACT

This SEP Technical Evaluation, for the Big Rock Point Plant, reviews the scope and frequency of periodic testing of the Emergency Core Cooling System 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.

CONTENTS

SYSTEMATIC EVALUATION PROGRAM

TOPIC VI-7.A.3
ECCS ACTUATION SYSTEM

BIG ROCK POINT

1.0 INTRODUCTION

The objective of this review is to determine if all Emergency Core Cooling System (ECCS) components, including pumps and valves, are included in component and system tests, if the scope and frequency of periodic testing are identified, and if the test program meets current licensing criteria. The systems included in the ECCS are the Core Spray and Core Spray Recirculation System.[1]

2.0 CRITERIA

General Design Criterion 37 (GDC 37), "Testing of Emergency Core Cooling Systems," requires that:

> The ECCS be designed to permit appropriate periodic pressure and functional testing to assure the operability of the system as a whole and to verify, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.[2]

Branch Technical Position ICSB 25, "Guidance for the Interpretation of GDC 37 for Testing the Operability of the Emergency Core Cooling System as a Whole," states that:

> All ECCS pumps should be included in the system test.[3]

Regulatory Guide 1.22, "Periodic Testing of the Protection System Actuation Functions," 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.[4]

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

Regulatory Guide 1.22 states, in Section D.4, that:

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

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

2. 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.

3. The actuated equipment can be routinely tested when the reactor is shut down.[4]

3.0 CORE SPRAY SYSTEM

3.1 Description

The Core Spray System (CSS) is designed to provide reactor cooling in the event of a loss of coolant accident (LOCA). The Core Spray System consists of two parallel lines which run from the station fire header in the turbine building through the reactor containment and into the reactor vessel. A spray nozzle is provided for one of the lines and a spray ring is utilized for the other line, inside the vessel and about six feet above the core, to spray water directly onto the fuel assemblies in a pattern which provides adequate cooling. Water is provided to the Core Spray System by the two station fire pumps (one electric- and one diesel-driven) and by the Fire Water System (FWS). In the initial stages of a LOCA, the pumps will automatically start upon receiving a start signal from the steam drum level instrumentation when a low level condition is reached. Both fire pumps will have started prior to the automatic opening of the four CSS injection valves, since the automatic opening of these valves is, in part, a function of reactor low water level (a condition which should occur some time after a low steam drum level). The fire pumps also receive an automatic start signal resulting from decaying FWS header pressure. The water supply from the FWS supplying the CSS is normally maintained in a pressurized state by the fire jockey pump. FWS flow to the core would commence, even if the fire pumps did not start, as soon as the CSS injection valves automatically open and reactor pressure decreases below the FWS pressure to unseat the CSS check valves.

3.2 Evaluation

The Big Rock Point Plant Technical Specification requires on a monthly bases that (a) the CSS injection valves and the CSS shell side inlet valve are verified to be operable by remote manual actuation, (b) the CSS core spray heat exchanger is leak tested, (c) both fire pumps are automatically actuated by the pump actuating circuitry, (d) valve MO-7069 is verified to be locked or sealed in the open position, and (e) the closure of the deluge system automatic isolation valve is verified when the core spray injection

valves receive an open signal. During each major refueling outage, the Big Rock Point Plant Technical Specification requires that (a) the CSS actuation instrumentation, the pressure instrumentation, and the flow instrumentation is calibrated, (b) the two CSS containment isolation check valves are verified not to be stuck shut, (c) the fire system basket strainer differential pressure switches are calibrated, and (d) the CSS injection valves with water flow normally blocked are verified to be operational upon manual and automatic actuation. The Big Rock Point Plant Technical Specification has also established that the instrumentation for the CSS be checked, tested, and calibrated on a periodic basis.

There is no established requirement in the technical specifications for a periodic systems integrated test to determine the operability of the system as a whole as required by General Design Criterion 37. However, the licensee, through the use of procedures[6] performs system level testing (i.e., the automatic actuation from sensor input to verification of proper actuated component response) which includes test overlap and time response verifications. These procedures are performed at every refueling and specific precautions are taken to assure that no FWS water or water from Lake Michigan flows into the reactor core.

4.0 CORE SPRAY RECIRCULATION SYSTEM

4.1 Description

The Core Spray Recirculation System (CSRS) is designed to prevent excessive water buildup in the containment sphere and to provide long-term, post-accident cooling. The system consists of two 400 gpm pumps and a heat exchanger. The pumps take suction from the lower levels of containment and discharge to the core spray headers. The CSRS recirculation flow path consists of only non-electrically operated valves which are locked open or non-electrically operated check valves. Therefore, during power operation, the CSRS is normally prealigned for service such that recirculation flow will commence whenever one of the core spray pumps is started. The system is actuated manually when the water level in the containment rises to an elevation of 587 feet. This elevation level will be achieved between 6 to 24 hours operation of one core spray and one containment spray channel.

4.2 Evaluation

The Big Rock Point Plant Technical Specifications specify that a test tank and appropriate valving is provided in the Core Spray Recirculation System so the pump suction conditions and the flow characteristics of the sytem can be periodically tested. The technical specification also requires, on a monthly bases, that the hose required for backup cooling water to the core spray recirculation heat exchanger is verified to be installed on a designated rack in the screen house. During each major refueling outage, the technical specification requires that (a) the CSRS be operationally checked through the test flow tank flow path, (b) the valve MO-7066 is verified to be operable upon manual actuation, (c) the hose used for backup cooling water to the core spray recirculation heat exchanger is verified to be operable and free of obvious defects, and (d) a leak and flow check of the backup cooling water hose when connected between the screen house fire water connection and the core spray recirculation heat exchanger

is performed. The technical specification requires that the instrumentation for the CSRS be checked, tested and calibrated on a periodic basis.

The Core Spray Recirculation System is not tested from the manual actuation through to the establishment of flow during reactor operation as specified by Standard Review Plan Section 7.2, Appendix B, Section 11. Testing of the Core Spray Recirculation System during reactor operation is not practical during plant operation or shutdown because full operation of the CSRS requires that the CSS be in operation (i.e, the CSS injection valves must be open) and a water supply be available in the bottom of the containment sphere.[6] The technical specifications do not establish testing or position verification requirements for the valves in the flow path for the Core Spray Recirculation System, nor is an integrated systems test required during refueling to determine system operability. However, the licensee through the use of two refueling outage test procedures[6] verifies the capability of the CSRS to serve in the recirculation mode.

5.0 SUMMARY

The following has been determined pertaining to the testing and testability of the Core Spray and Core Spray Recirculation Systems.

1. The design of both the Core Spray and the Core Spray Recirculation system make testing of the systems impractical during reactor operation.

2. The Big Rock Point Technical Specifications do not require a systems integrated test to determine system operability as a whole. However, the licensee performs this testing by plant test procedures,[6] using test overlap and time response verification. Therefore, the CSS does comply to the current reactor licensing criteria.

3. The Big Rock Point Technical Specifications do not establish testing or position verification requirements for the valves in the flow path for the CSRS nor an integrated system test. However, the licensee through the use of plant test procedures[6] verifies the capability of the CSRS to serve in the recirculation mode. Therefore, the CSRS does comply to the current reactor licensing criteria.

6.0 REFERENCES

1. Appendix "A" Consumers Power Company Big Rock Point Plant Technical Specifications, Section 3.1.4/4.1.4, Emergency Core Cooling System, as amended through February 25, 1981.

2. General Design Criterion 37, "Testing of Emergency Core Cooling System," of Appendix A, "General Design Criteria for Nuclear Power Plants," 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," January 1, 1981.

3. Branch Technical Position ICSB 25, "Guidance for the Interpretation of GDC 37 for Testing the Operability of the Emergency Core Cooling System as a Whole."

4. Regulatory Guide 1.22, "Periodic Testing of the Protection System Actuation Functions."

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

6. Letter from Robert A. Vincent, Staff Licensing Engineer, to Dennis M. Crutchfield, Chief of Operating Reactors Branch No. 5, "Docket 50-155-Licensee DPR-6-Big Rock Point Plant-SEP Topic VI-7.A.3, ECCS Actuation," dated June 4, 1982.

SYSTEMATIC EVALUATION PROGRAM
TOPIC VI-7.A.3

BIG ROCK POINT

TOPIC: VI-7.A.3, ECCS ACTUATION SYSTEM

I. INTRODUCTION

The ECCS actuation system was reviewed with respect to the testability of operability and performance of individual active components of the system and of the entire system as a whole under conditions as close to the design condition as practical. The purpose of the reviews was to assure that all ECCS components (e.g., valves and pumps) are included in the component and system test and to assure that the scope of the periodic testing is adequate and meets the requirements of GDC 37. The technical specifications were also audited for large differences between the present test requirements and those in the Standard Technical Specifications.

II. REVIEW CRITERIA

The current licensing criteria are identified in Section 2 of EG&G Report EGG-EA-5961, "ECCS Actuation System."

III. RELATED SAFETY TOPICS AND INTERFACES

The scope of review for this topic was limited to avoid duplication of effort since some aspects of the review were performed under related topics. The related topics and the subject matter are identified below. Each of the related topic reports contain the acceptance criteria and review guidance for its subject matter.

Topic VI-3,	Containment Pressure and Heat Removal Capability
Topic VI-4,	Containment Isolation System
Topic VI-7,	Emergency Core Cooling System
Topic VI-7.C	ECCS Single Failure Criterion and Requirements for Locking Out Power to Valves Including Independence of Interlocks on ECCS Valves
Topic VI-9	Main Steam Isolation
Topic VI-10.A	Testing of Reactor Trip System and Engineered Safety Features, Including Response Time Testing

Only Topic VI-10.A is dependent on the present topic information for completion. Response time testing is addressed in Topic VI-10.A.

VI. REVIEW GUIDELINES

The review guidelines are presented in Sections 3 and 4 of Report EGG-EA-5961.

V. EVALUATION

Report EGG-EA-5961 describes the extent to which the ECCS actuation system is tested, except for the question of response time testing. In summary, the safety systems meet the review criteria.

VI. CONCLUSION

Based upon our review of our contractor's evaluation, the staff concludes that Big Rock Point conforms to current licensing criteria and is, therefore, acceptable.