June 10, 1981

Docket No. 50-219
LS05-81-06-027

Mr. I. R. Finfrock, Jr.
Vice President - Jersey Central
Power & Light Company
Post Office Box 388
Forked River, New Jersey 08731

Dear Mr. Finfrock:

SUBJECT: SEP TOPIC VI-7.A.3 - ECCS ACTUATION SYSTEM (OYSTER CREEK)

A copy of our contractor's draft evaluation of Systematic Evaluation Program Topic VI-7.A.3 is enclosed. This assessment compares your facility, as described in Docket No. 50-219, with the criteria currently used by the regulatory staff for licensing new facilities. Please inform us if your as-built facility differs from the licensing basis assumed in our assessment within 30 days of receipt of this letter.

This evaluation will be a basic input to the staff's safety evaluation report for this topic for your facility unless you identify changes needed to reflect the as-built conditions at your facility. 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 No. 5
Division of Licensing

Enclosures:
As stated

cc w/enclosures:
See next page

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OFFICE	SEPB	SEPB	SEPB	ORB#5	ORB#5	AD:SA:DL
SURNAME	RScholl:bl	RHermann	WRussell	JLombardo	DCrutchfield	GLainas
DATE	6/5/81	6/8/81	6/9/81	6/9/81	6/10/81	6/10/81

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

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cc w/enclosures:
See next page

cc
G. F. Trowbridge, Esquire
Shaw, Pittman, Potts and Trowbridge
1800 M Street, N. W.
Washington, D. C. 20036

GPU Service Corporation
ATTN: Mr. E. G. Wallace
Licensing Manager
260 Cherry Hill Road
Parsippany, New Jersey 07054

Natural Resources Defense Council
917 15th Street, N. W.
Washington, D. C. 20006

Steven P. Russo, Esquire
248 Washington Street
P. O. Box 1060
Toms River, New Jersey 08753

Joseph W. Ferraro, Jr., Esquire
Deputy Attorney General
State of New Jersey
Department of Law and Public Safety
1100 Raymond Boulevard
Newark, New Jersey 07012

Ocean County Library
Brick Township Branch
401 Chambers Bridge Road
Brick Town, New Jersey 08723

Mayor
Lacey Township
P. O. Box 475
Forked River, New Jersey 08731

Commissioner
Department of Public Utilities
State of New Jersey
101 Commerce Street
Newark, New Jersey 07102

Gene Fisher
Bureau Chief
Bureau of Radiation Protection
380 Scotts Road
Trenton, New Jersey 08628

Commissioner
New Jersey Department of Energy
101 Commerce Street
Newark, New Jersey 07102

Plant Superintendent
Oyster Creek Nuclear Generating Station
P. O. Box 388
Forked River, New Jersey 08731

Resident Inspector
c/o U. S. NRC
P. O. Box 445
Forked River, New Jersey 08731

Director, Criteria and Standards Division
Office of Radiation Programs (ANR-460)
U. S. Environmental Protection Agency
Washington, D. C. 20460

U. S. Environmental Protection Agency
Region II Office
ATTN: EIS COORDINATOR
26 Federal Plaza
New York, New York 10007

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SEP TECHNICAL EVALUATION

TOPIC VI-7.A.3
ECCS ACTUATION SYSTEM

OYSTER CREEK

Docket No. 50-219

May 1981

5-12-81

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SEP TECHNICAL EVALUATION

TOPIC VI-7.A.3
ECCS ACTUATION SYSTEM

OYSTER CREEK

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 Low Pressure Core Spray (LPCS), the Automatic Depressurization System (ADS) and the Emergency Condenser System.

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.[1]

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.[2]

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.[3]

Standard Review Plan, Section 7.3, Appendix A, "Use of IEEE Standard 279 in the Review of the ESFAS and Instrumentation and Controls of Essential Auxiliary Supporting Systems," states, in Section 11.b, that:

> Periodic testing should duplicate, as closely as practical, the integrated performance from the ESFAS systems and their essential auxiliary supporting systems. If such a "system level" test can be performed only during shutdown, the testing done during power operation must be reviewed in detail. Check that "overlapping" tests do, in fact, overlap from one test segment to another. For example, closing a circuit breaker with the manual breaker control switch may not be adequate to test the ability of the ESFAS to close the breaker.[4]

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.[3]

3.0 CORE SPRAY

3.1 Description. The LPCS consists of two independent subsystems which are designed to start automatically and pump water from the torus into the reactor vessel to be dispersed directly above the fuel region. Each subsystem can be started manually in addition to the automatic initiation by the control logic. The starting sequence is:

1) The priority core spray pump in each subsystem starts. Should either core spray pump fail to start, the alternate pump in that subsystem will receive a start signal within 10 seconds of the initial signal.

2) Both diesel generators start immediately.

3) Upon sensing both a system start signal and a discharge pressure of its associated core spray pump the priority core spray booster pump will start. If the priority booster pump in either loop fails to start within 5-seconds of sensing both conditions a start signal is sent to the alternate core spray booster pump in that subsystem.

4) All motor-operated valves in each subsystem are open with the exception of two parallel core spray/booster pump output valves. The two valves for each subsystem will not open after the core spray initiation until the reactor pressure sensors detect the reactor pressure is below 285 psig.

Each subsystem has two independent control logic channels. All of the four low-low reactor level or four drywell high pressure sensors will initiate the logic channels in both subsystems. Each of the two logic channels will complete the above described equipment control functions for its respective subsystem. A keylock switch permits testing of each logic channel without the normal interaction (start) of the other core spray system. The following control instrument parameters are used in each logic channel:

1) Low-low reactor water level

2) High drywell pressure

3) Low reactor pressure

4) Manual start switch

5) Bus undervoltage relays

6) Low core spray pump discharge

7) Time relay (core spray pumps)

8) Time relay (booster pumps)

9) Test interlock switch.

3.2 Evaluation. The design of the LPCS allows complete testing from sensor to final actuated devices, including motors and valves, for each subsystem during reactor operation by opening a test valve to torus and closing a block valve to the reactor. By use of the test interlock switch the other subsystem will be available for use if required.

The Oyster Creek Technical Specifications do not specifically indentify the instrument parameters (see 3.1 above) for checks, functional testing and calibration. Table 4.1.2 and Paragraph 4.4.A of the Oyster Creek Technical Specifications do require system automatic actuation tests at least every three months and during refueling. The motor-operated valves are tested monthly and the pump operability monthly, after major maintenance and during a refueling outage.

4.0 EMERGENCY CONDENSER SYSTEM

4.1 Description. The emergency condenser system consists of two independent high pressure isolation condensers subsystems. Each condenser operates by natural circulation, with steam flowing from the reactor pressure vessel through the heat exchangers, with condensate returning by gravity to the pressure vessel. Each condenser is placed into operation by opening a normally closed D.C. motor-operated valve in its condensate return line to the vessel. In addition to this valve there are normally open AC and DC motor operated valves on the steam (inlet) lines and a normally open AC motor-operated valve on the condensate return (outlet) line to each condenser. Both manual and automatic actuation of each condenser is possible.

During operation of the system the water in the shell side of the condensers will boil, and the steam will vent to the atmosphere. Makeup water can be supplied by either the condensate transfer system or the fire protection system. There is no automatic water makeup for the condensers.

There are two channels of instrumentation which can generate an automatic initiation signal to the isolation condenser system. A start signal from either of the two channels will cause both condensers to initiate. These channels of instrumentation monitor reactor pressure and reactor water level. Should both high pressure sensors in one channel, or both low reactor water level sensors in one channel, or a combination of a high pressure and a low water level in one channel exceed the trip settings then an initiate signal to both emergency condensers is generated. A 15 second time delay prevents an initiation due to a spurious signal.

There is, also, a potential for a loss of reactor primary cooling water should a pipe in this system rupture. Four sensors are installed in the inlet and outlet piping to each emergency condenser to detect line breaks. Two sensors are for the detection of high flow in the steam line and two are used for the detection of high flow in the condensate line to each condenser. Should one of these four sensors detect a high flow condition, lasting as long as 5 seconds, the three motor-operated isolation valves and the operational valve to that condenser are immediately shut.

The following instrument parameter are utilized in each logic channel:

1) Reactor high pressure

2) Reactor low-low water level

3) High flow, condenser inlet

4) High flow, steam

5) Time delay relays

6) Reset

7) Manual start.

4.2 Evaluation. The design of the emergency condensors does not allow testing during operation since both systems are operated by either logic channel and there is no test switch to allow one condenser to be tested with the other system available for use.

The Oyster Creek Technical Specifications do not require channel checks, functional tests and calibration for the emergency condenser instrument parameters listed in Section 4.1 above. Table 4.1.2 of the Oyster Creek Technical Specifications does require condenser actuation and isolation by each trip circuit independently (one valve at a time) at each refueling outage.

5.0 AUTO DEPRESSURIZATION SYSTEM (ADS)

5.1 Description. The five electromatic relief valves (EMRV) for the ADS enable the core spray system to provide protection against a small break in the event that the feedwater system is not active or in the event of high reactor pressure.

There are two automatic methods of operation for the EMRVs in addition to the manual mode of actuation. A brief description of these two methods of operation of the EMRVs follow:

1) High Pressure Actuation--Each of the five EMRVs has an associated pressure switch which constantly monitors the reactor vessel pressure. Upon sensing an over-pressure condition the pressure

switch completes an eletrical circuit to the solenoid actuator of its associated EMRV, causing the valve to open. When the over-pressure conditions terminates, the electrical signal to the valve is removed, and the valve shuts automatically.

2) Auto-Depressurization Actuation--There are two redundant channels of instrumentation which are capable of generating the auto-depressurization actuation signal. A signal from either of the two instrument channels is sufficient to open all five valves sequentially after a 100 second time delay. The following conditions are required in one of the two logic channels for actuation.

   a) High drywell pressure--Two out of two logic from core spray logic relays.

   b) Low-low-low reactor water level as detected by two of two level sensors in one of the two instrument channels.

   c) The core spray pumps are operating as verified by core spray booster pump logic relays, which are activated from booster pump discharge line sensors.

The following control instrument parameters are utilized in the ADS:

1) Reactor high pressure (each valve)

2) High drywell pressure

3) Low-low-low reactor water level

4) High core spray booster pump discharge pressure

5) Time delay relays

6) Manual start/stop

7) Reset.

5.2 Evaluation. The ADS system cannot be tested during operation without effecting the operability of the plant. However, there is a very low probability that the system will fail to initiate when required, without testing during operation.

The Oyster Creek Technical Specifications do not indentify ADS instrument parameters for checks, functional testing and calibration. Table 4.1.2 and Paragraph 4.4.B of the Oyster Creek Technical Specifications do require valve operability tests and an automatic actuation test during each refueling outage.

6.0 SUMMARY

The review of the referenced material has determined the following in reguard to the Oyster Creek ECCS testing and testability:

1) The design of the low pressure core spray system meets all current requirements for testability during operation.

2) The ADS, although not testable during operation of the reactor, has sufficient probability of operation to meet the requirements.

3) The design of the emergency condenser logic does not allow testing of one system during operation without disabling both systems.

The ECCS surveillance requirements as required by the plant technical specifications do not meet current licensing requirements, since specific periodic channel checks, functional tests and calibrations are not specifically required for each instrument parameter in each logic channel for each ECCS system. Checks, tests and calibrations should be required during reactor operation and during refueling.

The testing of pumps and valves are included in the systems tests and meet current licensing requirements.

7.0 REFERENCES

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

2. BTP ICSB 25, "Guidance for the Interpretation of GDC 37 for Testing the Operability of the Emergency Core Cooling System as a Whole."

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

4. Nuclear Regulatory Commission Standard Review Plan, Section 7.3, Appendix A, "Use of IEEE Standard 279 in the Review of the ESFAS and Instrumentation and Controls of Essential Auxiliary Supporting Systems."

5. Nuclear Regulatory Commission Standard Review Plan, Section 7.3, Appendix A, "Use of IEEE Standard 279 in the Review of the ESFAS and Instrumentation and Controls of Essential Auxiliary Supporting Systems."

6. Standard Technical Specifications for General Electric Boiling Water Reactors (BWRs), NUREG-0123, Revision 2, Fall 1980.

7. Oyster Creek Nuclear Power Plant No. 1, "Facility Description and Safety ANalysis Report," Amendment 3, dated January 25, 1967.

8. Technical Specifications and Bases for Oyster Creek Nuclear Power Plant Unit 1, Appendix A, to Provisional Operating License DPR-16, Amendments 1 through 15, dated February 24, 1976.

9. JPC&L letter to NRC, "Supplement #6 to Amendment #68," dated November 1, 1973.

10. NRC (Lear) letter to JPC&L (Finfrock), Amendment 8, dated May 24, 1975.

11. JPC&L (Finfrock) to NRC (Lear), dated June 24, 1975.

12. NRC (Lear) letter to JPC&L (Finfrock), Amendment 15, dated February 24, 1976.

13. JPC&L letter EAGB-80-?83 (Finfrock) to NRC, dated August 11, 1980.

14. General Electric Drawing No. 885D781, Rev. No. 10.

15. General Electric Drawing No. 718E644, Rev. No. 11.

16. NUS Drawing No. 5060F032, Rev. No. 1.

17. MPR Associates, Inc., Drawing No. 1083-55-09, Sht. 1 (Rev. A) and Sht. 2 (Rev. B).

18. General Electric Drawing No. 148F262, Rev. 5.