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

March 10, 1980

Docket No. 50-219

MEMORANDUM FOR: D. L. Ziemann, Chief
Operating Reactors Branch #2, DOR

FROM: D. M. Crutchfield, Chief
Systematic Evaluation Program Branch, DOR

SUBJECT: SEP SAFETY TOPIC ASSESSMENT INPUT - OYSTER CREEK NUCLEAR STATION

Attached is the revised SEP technical evaluation report on Topic VIII-4, "Electrical Penetration of Reactor Containment", for Oyster Creek Nuclear Station incorporating the comments made by the SEP project managers. This report supersedes the previous report dated December 17, 1979. Please transmit this report to the licensee.

The staff is presently developing a position for SEP plants on the protection of containment electrical penetrations from fault currents. This position will be forwarded to the licensee at a future date depending upon the review and approval of the position.

C. H. Trammell for

Dennis M. Crutchfield, Chief
Systematic Evaluation Program Branch
Division of Operating Reactors

Attachment:
As stated

cc:
D. Eisenhut
R. Vollmer
J. Shapaker
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8003280355

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SEP TECHNICAL EVALUATION

TOPIC VIII-4
ELECTRICAL PENETRATIONS OF REACTOR CONTAINMENT

OYSTER CREEK NUCLEAR STATION

Jersey Central Power and Light

Docket No. 50-219

January 1980

1-15-80

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SEP TECHNICAL EVALUATION

TOPIC VIII-4
ELECTRICAL PENETRATIONS OF REACTOR CONTAINMENT

OYSTER CREEK NUCLEAR STATION

1.0 INTRODUCTION

This review is part of the Systematic Evaluation Program (SEP), Topic VIII-4. The objective of this review is to determine the capability of the electrical penetrations of the reactor containment to withstand short circuit conditions of the worst expected transient fault current resulting from single random failures of circuit overload protection devices.

General Design Criterion 50, "Containment Design Basis" of Appendix A, "General Design Criteria for Nuclear Power Plants" to 10 CFR Part 50 requires that penetrations be designed so that the containment structure can, without exceeding the design leakage rate, accommodate the calculated pressure, temperature, and other environmental conditions resulting from any loss-of-coolant accident (LOCA).

IEEE Standard 317, "Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations", as augmented by Regulatory Guide 1.63, provides a basis of electrical penetrations acceptable to the staff.

Specifically, this review will examine the protection of typical electrical penetrations in the containment structure to determine the ability of the protective devices to clear faults prior to exceeding the penetration design rating under LOCA temperatures.

2.0 CRITERIA

IEEE Standard 317, "Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations" as supplemented by

Nuclear Regulatory Commission Regulatory Guide 1.63, "Electric Penetration Assemblies in Containment Structures for Light-Water-Cooled Nuclear Power Plants" provides the basis acceptable to the NRC staff. The following criteria are used in this report to determine compliance with current licensing requirements:

(1) IEEE Standard 317, Paragraph 4.2.4 -- "The rated short circuit current and duration shall be the maximum short circuit current in amperes that the conductors of a circuit can carry for a specified duration (based on the operating time of the primary overcurrent protective device or apparatus of the circuit) following continuous operation at rated continuous current without the temperature of the conductors exceeding their short circuit design limit with all other conductors in the assembly carrying their rated continuous current under the specified normal environmental conditions."

This paragraph is augmented by Regulatory Guide 1.63, Paragraph C-1 -- "The electric penetration assembly should be designed to withstand, without loss of mechanical integrity, the maximum possible fault current versus time conditions that could occur given single random failures of circuit overload protection devices."

(2) IEEE Standard 317, Paragraph 4.2.5 -- "The rated maximum duration of rated short circuit current shall be the maximum time that the conductors of a circuit can carry rated short circuit current based on the operating time of the backup protective device or apparatus, during which the electrical integrity may be lost, but for which the penetration assembly shall maintain containment integrity."

3.0 DISCUSSION AND EVALUATION

In this evaluation, the results of typical containment penetrations being at LOCA temperature initially concurrent with a random failure of the circuit protective devices will be analyzed.

Jersey Central Power and Light provided information (Reference 1) on typical penetrations. No evaluation of the data was provided. Jersey Central Power and Light has established a temperature limit of 350°F (177°C) before seal failure for the two penetrations based on testing. Maximum short circuit current available (I_{sc}) was provided by Jersey Central Power and Light for a three-phase bolted fault. Rated current (I_r) for each penetration was also provided.

The following formula (Reference 3) was used to determine the time allowed before a short circuit would cause the penetration to heat up to the temperature limit.

$$t = \frac{A^2}{I^2} \; .0297 \log \frac{T_2+234}{T_1+234} \qquad \text{(Formula 1)}$$

where

- t = time in seconds
- I = current in amperes
- A = conductor area in circular mils
- T_1 = initial temperature (138°C, LOCA condition)
- T_2 = maximum penetration temperature before failure.

This is based on the heating effect of the short circuit current on the conductor and does not take into account heat losses of the conductor. For times less than several seconds, this heat loss is negligible.

In evaluating the capability of the penetration to withstand a LOCA temperature with a short circuit current, Formula 1 was used to calculate the time required to heat the conductor from the LOCA temperature to penetration failure temperature for currents from rated current to maximum short circuit current in 20% increments. Times for the primary and secondary overcurrent devices to interupt these fault currents were calculated. Where breaker ratings provided by the licensee indicated minimum and maximum fault clearing times, the maximum time was used for conservatism.

3.1 Typical Low Voltage (0-1000 V) Penetration. Jersey Central Power and Light has identified penetration #11 (GE type NS04) as being typical of low voltage penetrations. This penetration provides 460 V ac power to Drywell Recirculation Fan RF-1-1.

This penetration uses #2 AWG cable and has a continuous current rating of 66 amps. The maximum short circuit current has been determined by JCP to be 4200 amps. Jersey Central Power and Light has established 352°F (177°C) as the limiting temperature before seal failure based on testing (Reference 2). At the maximum short circuit current (4200 amps), overtemperature will be reached in 0.31 second from LOCA temperature initially.

From LOCA temperature initially, the primary breaker will operate to clear any fault current prior to attaining the penetration seal limiting temperature.

From LOCA temperature initially, the secondary breaker will not operate to clear any fault currents prior to exceeding the 352°F (177°C) penetration seal temperature limit.

3.1.1 Low Voltage Penetration Evaluation. With the initial penetration temperature at 138°C (LOCA), penetration #11 does not meet current licensing requirements of RG 1.63 and IEEE Std. 317 with a random failure of the primary breaker.

See Attachment for Details

3.2 Typical Medium Voltage (>1000 V) Penetration. Jersey Central Power and Light has identified penetration # 0 (GE type NS03) as being typical of medium voltage penetrations. This penetration provides 4160 V ac power to Reactor Recirculation Pump Motor NG01-B.

This penetration uses 500 MCM cable with a continuous current rating of 475 amps. The maximum available short circuit current has been determined by JCP to be 1800 amps. Jersey Central Power and Light has established 352°F (177°C) as the limiting temperature before seal failure based on testing (Reference 2). At the maximum short circuit current (1800 amps), overtemperature would be reached in 98 seconds from LOCA temperature initially.

There are no circuit protective devices located between the motor generator output and the Reactor Recirculation Pump Motor. Overcurrent protection is provided by a differential current sensing relay and a line overcurrent sensing relay, each of which will operate to trip the motor generator by securing power to the motor generator motor and opening the generator field windings. At ≥90 amps of current difference between phases, the differential relay will cause a trip of the motor generator in 0.7 second or less. At line currents in excess of 360 amps, the overcurrent relay will cause a trip of the motor generator in 0.17 second or less.

For a three-phase short circuit condition, it cannot be assumed that sufficient current differences will exist to cause the differential relay to operate and trip the motor generator. Therefore, operation of this relay cannot be expected to clear fault currents prior to exceeding the penetration seal temperature limit of 177°C. For fault currents producing current differences between phases in excess of 90 amps, this relay will operate to trip the motor generator prior to reaching the penetration seal temperature limit.

The line overcurrent relay will operate to clear all fault currents prior to reaching the penetration seal temperature limit of 177°C.

GPU has analyzed to show there is no problem with medium voltage penetration. — GPU internal memo — Jan 21, 1982 - letter from P.J. Chisholm to D.G. Holloway.

3.2.1 Medium Voltage Penetration Evaluation. From LOCA temperature initially, penetration #20 does not meet current requirements with a failure of the line overcurrent relay since the differential relay cannot be assumed to operate for a three-phase short circuit. With a failure of the differential relay, the penetration will not exceed its design limits for any fault currents.

4.0 SUMMARY

From LOCA temperature, neither penetration #11 nor #20 meet the current licensing requirements of RG 1.63 and IEEE Std. 317 for a short circuit fault and failure of the primary protective device.

The review of Topic III-12, "Environmental Qualification," may result in changes to the electrical penetration design and therefore, the resolution of the subject SEP topic will be deferred to the integrated assessment, at which time, any requirements imposed as a result of this review will take into consideration design changes resulting from other topics.

5.0 REFERENCES

1. Jersey Central Power and Light letter (Finfrock) to NRC (Ziemann) dated April 24, 1979.

2. Final Description and Safety Analysis Report, Oyster Creek Nuclear Station, Ammendment 62 (Docket No. 50-219-102).

3. IPC&A Publication P-32-382, "Short Circuit Characteristics of Insulated Cable."

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

July 8, 1981

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Docket No. 50-219
LS05-81- 07-013

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 V-11.A, REQUIREMENTS FOR ISOLATION OF HIGH AND LOW PRESSURE SYSTEMS, SAFETY EVALUATION FOR OYSTER CREEK

The enclosed staff safety evaluation supplements our contractor's evaluation that has been made available to you previously. This evaluation is consistent with the findings in our safety evaluation on Topic V-11.A which proposes modifications to the RWCU valve indication and control circuits.

The need to actually implement these changes will be determined during the integrated plant 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 No. 5
Division of Licensing

Enclosure:
As stated

PDR Dupe

cc w/enclosure:
See next page

TOPIC: V-11.A REQUIREMENTS FOR ISOLATION OF HIGH AND LOW PRESSURE SYSTEMS

I. INTRODUCTION

Several systems that have a relatively low design pressure are connected to the reactor coolant pressure boundary. The valves that form the interface between the high and low pressure systems must have sufficient redundancy and interlocks to assure that the low pressure systems are not subjected to coolant pressures that exceed design limits. The problem is complicated since under certain operating modes (e.g., shutdown cooling and ECCS injection) these valves must open to assure adequate reactor safety.

II. REVIEW CRITERIA

The review criteri are presented in Section 2 of EG&G Report 1309F, "Isolation of High and Low Pressure Systems."

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 criteria and review guidance for its subject matter.

V-10.B RHR Reliability
VI-4 Containment Isolation

Topic V-11.B is dependent on the present topic information for completion.

IV. REVIEW GUIDELINES

The review guidelines are presented is Section 7.3 of the Standard Review Plan.

V. EVALUATION

As noted in EG&G Report 1309F, "Isolation of High and Low Pressure Systems," the Oyster Creek Nuclear Station has two systems with a lower design pressure rating than the RCS that are directly connected to the RCS. These systems are the Core Spray (CS) and the Re :tor Water Cleanup (RWCU) Systems. The RWCU system does not satisfy the staff's requirements because the redundant pressure interlocks are not provided and the check valves do not have position indication in the control room.

VI. CONCLUSIONS

Because of the severe consequences of a LOCA outside of containment the staff proposes that: 1) redundant interlocks should be installed on the RWCU suction valve; 2) the indication and control of the RWCU discharge valves should be modified to satisfy the interlock provisions of SRP Section 6.3 and BTP RSB 5-1; and/or 3) RWCU discharge check valve position indication circuits as specified in BTP-ICSB 3 should be provided.

V-11-A

1309F

SEP TECHNICAL EVALUATION REPORT

ELECTRICAL, INSTRUMENTATION, AND CONTROL FEATURES FOR ISOLATION OF HIGH AND LOW PRESSURE SYSTEMS

OYSTER CREEK NUCLEAR STATION

Jersey Central Power and Light

Docket No. 50-219

December 1979

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SEP TECHNICAL EVALUATION REPORT

ELECTRICAL, INSTRUMENTATION, AND CONTROL FEATURES FOR ISOLATION OF HIGH AND LOW PRESSURE SYSTEMS

OYSTER CREEK NUCLEAR STATION

1.0 INTRODUCTION

The purpose of this review is to determine if the electrical, instrumentation, and control (EI&C) features used to isolate systems with a lower pressure rating than the reactor coolant primary system are in compliance with current licensing requirements as outlined in SEP Topic V-11A. Current guidance for isolation of high and low pressure systems is contained in Branch Technical Position (BTP) EICSB-3, BTP RSB-5-1, and the Standard Review Plant (SRP), Section 6.3.

2.0 CRITERIA

2.1 Residual Heat Removal (RHR) Systems. Isolation requirements for RHR systems contained in BTP RSB-5-1 are:

(1) The suction side must be provided with the following isolation features:

(a) Two power-operated valves in series with position indicated in the control room.

(b) The valves must have independent and diverse interlocks to prevent opening if the reactor coolant system (RCS) pressure is above the design pressure of the RHR system.

(c) The valves must have independent and diverse interlocks to ensure at least one valve closes upon an increase in RCS pressure above the design pressure of the RHR system.

(2) The discharge side must be provided with one of the following features:

(a) The valves, position indicators, and interlocks described in (1)(a) through (1)(c) above.

(b) One or more check valves in series with a normally-closed power-operated valve which has its position indicated in the control room. If this valve is used for an Emergency Core Cooling System (ECCS) function, the valve must open upon receipt of a safety injection signal (SIS) when RCS pressure has decreased below RHR system design pressure.

(c) Three check valves in series.

(d) Two check valves in series, provided that both may be periodically checked for leak tightness and are checked at least annually.

2.2 Emergency Core Cooling System. Isolation requirements for ECCS are contained in SRP 6.3. Isolation of ECCS to prevent overpressurization must meet one of the following features:

(1) One or more check valves in series with a normally-closed motor-operated valve (MOV) which is to be opened upon receipt of a SIS when RCS pressure is less than the ECCS design pressure

(2) Three check valves in series

(3) Two check valves in series, provided that both may be periodically checked for leak tightness and are checked at least annually.

2.3 Other Systems. All other low pressure systems interfacing with the RCS must meet the following isolation requirements from BTP EICSB-3:

(1) At least two valves in series must be provided to isolate the system when RCS pressure is above the system design pressure and valve position should be provided in the control room

(2) For systems with two MOVs, each MOV should have independent and diverse interlocks to prevent opening until RCS pressure is below the system design pressure and should automatically close when RCS pressure increases above system design pressure

(3) For systems with one check valve and a MOV, the MOV should be interlocked to prevent opening if RCS pressure is above system design pressure and should automatically close whenever RCS pressure exceeds system design pressure.

3.0 DISCUSSION AND EVALUATION

There are two systems at the Oyster Creek Nuclear Station which have a direct interface with the RCS pressure boundary and have a design pressure rating for all or part of the system that is lower than the RCS design pressure. These systems are the Core Spray (CS) system and the Reactor Water Clean-Up (RWCU) system.

3.1 Core Spray System. The CS system consists of two loops which take a suction on the suppression pool and discharge into the reactor vessel through a set of parallel MOVs in each loop. Isolation is provided by a set of parallel testable check valves in series with the set of parallel MOVs. Each of these valves has position indication in the control room. The MOVs open upon receipt of a safety injection signal after RCS pressure has decreased below CS system design pressure. Therefore, the CS system is in compliance with the requirements for isolation of high and low pressure systems contained in SRP 6.3.

3.2 Reactor Water Clean-Up System. The RWCU system takes suction on the RCS, cools the water by circulation through a regenerative and non-regenerative heat exhanger, and lowers the water pressure by the use of a pressure control valve. After passing through the low pressure filtering and cleaning portions of the system, the water is pumped at high pressure through the regenerative heat exchanger and back to the reactor via the feed line.

Isolation on the suction side of the system is provided by three MOVs, an inboard valve (closest to RCS), a pump suction valve, and a pump bypass valve. Isolation on the discharge side is provided by a MOV and two check valves. None of the MOVs will open if pressure in the low pressure portions of the system is higher than its design pressure. All the MOVs will close on high RWCU system temperature, low flow, or high RWCU system pressure. However, the interlocks for these valves use the same sensors and relays. All the MOVs have position indication in the control room.

The RWCU system is not in compliance with requirements for isolation of high and low pressure systems contained in BTP EICSB-3 since the interlocks for the isolation valves are not independent.

4.0 SUMMARY

The Oyster Creek Nuclear Station has two systems directly connected to the RCS which have lower design pressure ratings than the RCS. The CS system meets the current licensing requirements for isolation of high and low pressure systems contained in SRP 6.3. The RWCU system is not in compliance with BTP EICSB-3 since the isolation valve interlocks are not independent.

5.0 REFERENCES

1. NUREG-075/087, Branch Technical Positions EICSB-3, RSB-5-1; Standard Review Plan 6.3.

2. Final Facility Description and Safety Analysis Report, Oyster Creek Nuclear Station.

3. GE Drawings 148F444, 237E566, and 858D781.

4. Oyster Creek Drawings BR 3020 and BR 3019.

5. JCP&L letter (Finfrock) to NRC (Ziemann) dated February 5, 1979.

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

SEP File

July 3, 1980

Docket No. 50-219

Mr. I. R. Finfrock, Jr.
Vice President - Generation
Jersey Central Power & Light Company
Madison Avenue at Punch Bowl Road
Morristown, New Jersey 07960

Dear Mr. Finfrock:

RE: SEP TOPICS III-10.A, V-11.A, VI-7.C.1, VIII-3.B, VIII-4
(Oyster Creek Nuclear Generating Station)

Enclosed is a copy of our current evaluation of Systematic Evaluation Program Topics III-10.A, Thermal-Overload Protection for Motors of Motor-Operated Valves; V-11.A, Electrical, Instrumentation, and Control Features for Isolation of High and Low Pressure Systems; VI-7.C.1, Independence of Redundant Onsite Power Systems; VIII-3.B, D C Power System Bus Voltage Monitoring and Annunciation; VIII-4, Electrical Penetration of Reactor Containment. 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 60 days of receipt of this letter.

This evaluation will be a basic input to the integrated safety assessment 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

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

Enclosure:
SEP Topics III-10.A, V-11.A,
VI-7.C.1, VIII-3.B, VIII-4

cc w/enclosure:
See next page

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