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

TERA

July 10, 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 TOPIC III-5.B, Pipe Break Outside Containment
(Oyster Creek Nuclear Generating Station)

Enclosed is a copy of our current evaluation of Systematic Evaluation Program Topic III-5.B, Pipe Break Outside Containment (Enclosure 1). 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.

We are also enclosing a request for additional information (Enclosure 2) to enable us to complete the review of issues identified in the conclusions section of the above referenced evaluation. Please submit your response to this request within 60 days of receipt of this letter.

There are also two staff positions enclosed (Enclosure 3) that requests that you submit schedules by September 1, 1980 for the completion of certain modifications to your facility.

Sincerely,

Dennis M. Crutchfield

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

Enclosure:
1. Completed SEP Topic III-5.B.
2. Request for Additional Information
3. Staff Positions

cc w/enclosure:
See next page

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ENCLOSURE 1

SEP EVALUATION

OF

PIPE BREAK OUTSIDE CONTAINMENT

TOPIC III-5.B

FOR THE

OYSTER CREEK NUCLEAR POWER PLANT

TABLE OF CONTENTS

1.0 INTRODUCTION

The safety objective of Systematic Evaluation Program (SEP) Topic III-5.B, "Pipe Break Outside Containment" is to assure that pipe breaks would not cause the loss of needed functions of "safety-related" systems, structures and components and to assure that the plant can be safely shut down in the event of such breaks. The needed functions of "safety-related" systems are those functions required to mitigate the effects of the pipe break and safely shutdown the reactor plant. The current criteria for review of pipe breaks outside containment are contained in Standard Review Plan 3.6.1 and 3.6.2 including their attached Branch Technical Positions.

2.0 BACKGROUND

In December 1972, the staff sent letters (Reference 1) to all power reactor licensees requesting an analysis of the effects of postulated failures of high energy lines outside of containment. A summary of the criteria and requirements in this letter is set forth below:

a. Protection of equipment and structures necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming a concurrent and unrelated single active failure of protected equipment, should be provided from all effects resulting from ruptures in pipes carrying high energy fluid, where the temperature and pressure conditions of the fluid exceed 200°F and 275 psig, respectively, up to and including a double-ended rupture of such pipes. Breaks should be assumed to occur in those locations specified in the "pipe whip criteria." The rupture effects to be considered include pipe whip, structural (including the effects of jet impingement), and environmental.

b. In addition, protection of equipment and structures necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming a concurrent and unrelated single active failure of protected equipment, should be provided from the environmental and structural effects (including the effects of jet impingement) resulting from a single open crack at the most adverse location in pipes carrying fluid routed in the vicinity of this equipment. The size of the cracks should be assumed to be 1/2 the pipe diameter in length and 1/2 the wall thickness in width.

In response to our letter, Jersey Central Power and Light Company (JCP&L), the licensee) submitted Amendment 75 to the Facility Description and Safety Analysis Report (FDSAR) dated July 1, 1974 (Ref. 2) and Revisions 1 through 4 to Amendment 75 dated December 24, 1974 (Ref. 3), March 24, 1975 (Ref. 4), April 25, 1975 (Ref. 5), and June 1, 1976 (Ref. 6). Additional information was provided in an analysis of jet impingement loads on the torus dated September 21, 1976 (Ref. 7). The staff's review of these documents is contained in a Safety Evaluation Report dated December 27, 1976 (Ref. 8).

Additional information regarding moderate energy line breaks (MELB) has been provided by the licensee in Supplement 6 (Addendum 1) to the Application for a Full Term Operating License for Oyster Creek (Ref. 9) and by letters dated March 13, 1974 (Ref. 10) and July 3, 1974 (Ref. 11).

The licensee has proposed an additional analysis concerning the acceptability of pipe breaks outside containment associated with the isolation condenser steam and condensate lines. The analysis addresses the same concerns identified in this evaluation.

The NRC staff reevaluation of the effects of pipe breaks outside containment under SEP Topic III-5.B includes the comparison of Oyster Creek with current criteria for pipe breaks outside containment. The staff used an "effects oriented" approach to determine the acceptability of plant response to pipe breaks, i.e., each structure, system, component, and power supply which must function to mitigate the effects of the pipe break and to safely shut down the plant was examined to determine its susceptibility to the effects of the postulated break. Break effects considered were compartment pressurization, pipe whip, jet impingement, spray, flooding, and environmental conditions of temperature, pressure, and humidity. This review complements that of SEP Topic III-12, "Environmental Qualification of Safety-Related Equipment."

(The effects of potential missiles generated by fluid system ruptures and rotating machinery were also considered and are evaluated under SEP Topic III-4.C, "Internally Generated Missiles.")

The previous evaluation of pipe breaks outside containment for Oyster Creek was performed using some methods and criteria which are no longer used by the staff in the review of current plants. For example the current definition of a high energy fluid system as one that is maintained under conditions where either or both the maximum operating temperature and pressure exceeds 200°F and 275 psig is different from the definition applied in the previous review where a high energy fluid system was one in which both temperature and pressure exceeds 200°F and 275 psig. The SEP reevaluation of this topic was performed using the current criteria in Standard Review Plans 3.6.1 and 3.6.2 and their attached Branch Technical Positions.

Data for this assessment was gathered during a visit to the Oyster Creek plant on January 15-17, 1980.

3.0 EVALUATION

The results of the SEP reevaluation of pipe breaks outside containment for Oyster Creek are provided in Table 1. The following paragraphs provide additional information used to evaluate certain pipe breaks listed in Table 1.

The safe shutdown systems which were examined from the standpoint of protection from pipe break effects are identified in the SEP Safe Shutdown Review for Oyster Creek (Reference 12). These systems are:

(a) Reactor Control and Protection System,

(b) Emergency Condensers,

(c) Condensate Transfer System,

(d) Automatic Depressurization System,

(e) Core Spray System,

(f) Emergency Service Water System,

(g) Instrumentation for Shutdown and Cooldown,

(h) Emergency Power (AC and DC) and control power for the above systems and components.

3.1 Emergency Condensers

The two emergency condensers are located in the reactor building, 95' elevation, east side. The steam supply and condensate return lines for the condensers are routed from containment penetrations on the 75' elevation to the condensers and back. These lines are maintained at reactor system pressure because the steam supply valves, all of which are outside containment, are open during normal plant operations. The condensate return valve inside containment is also normally open, while the condensate return valve outside containment is shut. Each condenser's steam and condensate valves close (in approx. 55 sec.)* upon receipt of a high flow signal from sensors in its own steam supply and/or condensate return lines.

The shell side of the emergency condensers is supplied by the condensate transfer system through air operated fill valves which are controlled by the control room operator.

Emergency condenser high energy line breaks on the 95' level of the reactor building were analyzed in detail in Reference 4. Based on Reference 4, the licensee concluded that there were no pipe break locations in either emergency condenser which interact with the emergency condenser of the redundant system, cable tray 45, or conduit containing or supplying safety related equipment. Based on our reevaluation of HELBs on the 95' elevation, taking into account Reference 4, we have determined that interactions are possible between the two emergency condenser systems. These interactions are:

*The 55 second valve closure time consists of 20 seconds for the valve to shut following a 35 second time delay after the shut signal.

1. Jet impingement on cable tray 45 from a longitudinal break in the A condenser steam line. (Tray 45 contains level and control cables for both emergency condensers.)

2. Pipe whip damage to condu- containing level indication and control signals for the B condenser from a break in the A condenser steam line.

(The remaining potential targets of a HELB, the emergency condenser shells, condensate fill lines and fill valve air supply, are adequately protected by the geometry of piping layout and shielding provided by structures and equipment.)

The two potential interactions above could result in (1) the immediate loss of function of the condenser system suffering the break and (2) the eventual loss, in approximately 40 minutes, of the other condenser when its shell side water is boiled away by core decay heat. The steam line break would be isolated automatically by the high flow sensors. A reactor trip would occur because of either high power, low reactor water level, or main steam isolation valve closure on low steam pressure caused by the steam line break.

In accordance with current criteria, a reactor trip causes an assumed loss of offsite power; therefore, the main reactor feed system and main condenser are inoperable.

After the second emergency condenser has boiled dry, reactor system pressure would increase to the safety/relief valve setpoints. Pressure would be limited by the relief valves, but reactor system coolant inventory would continue to be lost through the relief valves. To put the plant in a safe condition, the operator must manually initiate the Automatic Depressurization System (ADS) and ensure that at least one train of the Core Spray (CS) System is operating. Adequate long-term core cooling is accomplished, even assuming the single failure of one of the two emergency diesel generators, with the ADS, CS, containment spray, and emergency service water systems (Ref. 12). These actions are included in the plant emergency procedures. The availability of these emergency systems to provide safe shutdown capability and sufficient time for operator action to initiate these sytems, even with the loss of both emergency condensers, provide adequate mitigation of the effects of these postulated HELBs.

Emergency condenser HELBs on the 75' elevation of the reactor building could result in damage to the emergency condenser isolation valves and controls, and cable trays V22, V2? 41, 42, and 43. The motor operators are susceptible to jet impingement damage from both steam and condensate line cracks and breaks and from pipe whip of the A condenser condensate lines. The conduit containing isolation valve control and power cable are susceptible to pipe whip and impingement effects from both steam and condensate line breaks. The steam supply line motor operated isolation valves for both emergency condensers are outside containment and are normally open. The steam line isolation valves could be prevented from automatically closing by the effects of a break in the isolation condenser piping. Considering the single failure assumption, only one valve needs to be damaged by the break effects to result in an unisolable break.

Cable trays V22, V23, 41, 42 and 43 carry electrical cables for one train of the core spray (CS) system, standby liquid control system (SLCS) and the emergency condenser system. As indicated previously in the discussion of HELBs on the 95' elevation, the CS is needed to cope with a loss of both emergency condensers. The damage to cable trays could prevent the opening of the emergency condenser condensate return valves which must open to initiate emergency condenser operation. In this case, the operator would not have the 40 minutes emergency condenser boil-dry time to initiate ADS and CS for core cooling. Recall that a loss of offsite power has been assumed because of a reactor trip. In addition the postulated break has damaged one CS train.

Based on this discussion, we have determined that there is inadequate protection from the effects of postulated emergency condenser line breaks on the 75' elevation at Oyster Creek. Our position on this is provided in the CONCLUSIONS section of this report.

3.2 Reactor Water Cleanup System

The Reactor Water Cleanup (RWCU) system high energy piping is located on the 51' elevation of the reactor building, south side. The isolation valves for the system include (1) inside the containment drywell: a check valve on the RWCU system return line and a MOV on the letdown line, and (2) outside containment: a MOV on the return line and a MOV on each leg of the letdown line which branches into two lines just outside of the containment penetration. The isolation MOVs automatically shut on a containment isolation signal on low-low reactor vessel level indication.

HELBs in the RWCU system could affect the motor operated isolation valve electrical power or the operator itself. This damage in combination with a single failure of the isolation valve inside containment could result in an unisolable break path from the reactor system. This is addressed further in the CONCLUSIONS section of this report. The HELB could also damage cables in trays 13A and 14A. Damage to cable tray 14A was assumed and analyzed in Reference 2, however, potential damage and effects of damage to cable tray 13A were not addressed. Additional information regarding the cables in tray 13A is required to assess the effects of this damage. The licensee will be requested to supply this information.

3.3 Control Rod Drive Hydraulic System

Pipe breaks associated with the CRD hydraulic control units, on the 23' elevation of the Reactor Building, could involve (1) the drive insert and withdraw lines which lead through containment penetrations to the drives, (2) the CRD hydraulic drive and cooling water lines, (3) the CRD charging water line, and (4) the CRD return line. A break in the drive withdraw line would cause its associated control rod to insert (scram). A break in any of the other lines would cause the rod to remain in position but the rod could still be inserted by the operator or by a reactor protection system scram signal. Loss of electrical power to the CRD hydraulic control unit would also result in a rod insertion. Therefore, potential pipe break damage to the CRD hydraulic control units would not prevent control rod trip (scram) by the operator or the reactor protection system.

3.4 Main Steam and Main Feed Systems

The effects of main steam (MS) and main feed (MF) HELBs in the Turbine Building Mezzanine area on the control room, cable spreading room, main steam isolation valves (MSIV), main feedwater piping and isolation valves, and cable trays 12, 13, 14, 15, 30, 31, and 32 were evaluated in Reference 2. Information regarding the MS line break detection system, MS and MF isolation valve supports and jet impingement effects on the torus is provided in Reference 3.

The SEP reevaluation of this area of the turbine building has determined that interactions between postulated MS and MF HELBs and one train of emergency service water system (ESWS) piping (loop II) is possible. The interactions could result in the loss of function of this loop. Since MS and MF breaks result in turbine and reactor trips with concurrent assumed loss of offsite power, the single failure of diesel generator #1 would result in the loss of function of ESWS loop I. Thus, the HELB could result in total loss of ESWS function.

The ESWS is used, as described in Reference 12, for long term cooling of the reactor by cooling the containment spray system which cools the torus water which is circulated through the reactor by the core spray and automatic depressurization systems.

For the above described scenario in which the ESWS system is lost, the shutdown cooling system is available for long term core cooling after reactor system temperature is reduced by the emergency condensers and/or the automatic depressurization system. Therefore, the ESWS is not essential for safe shutdown following a MS or MF line break which disables ESWS in the turbine

building mezzanine area; and the plant is adequately protected from these potential breaks.

As described in Ref. 2, the MS and MF breaks in the mezzanine area can also damage electrical control cables used for the control of core spray (CS), containment spray, ESWS, diesel generators, MS line break detection, automatic depressurization system (ADS) and control rod drive (CRD) hydraulic pumps. Damage to these cables would not prevent the functioning of the diesel generators, MS break detection, ADS, and CRD hydraulic pump. However, with the assumed loss of offsite power and a single failure, damage to even one train of the CS, containment spray, and ESW control systems could result in the complete loss of these system functions. Again, the shutdown cooling system could be used for long term core cooling and so the containment spray and ESW functions are not essential. Loss of the CS system function, however, would severely restrict the ability of the plant operator to keep the reactor core covered with coolant during plant recovery from the postulated break. In Ref. 2, the licensee stated that the CRD hydraulic system was required to cope with the postulated breaks. The implied requirement of the CRD hydraulic pumps is to maintain reactor vessel coolant level during the plant cooldown following initiation of the emergency condensers. However, the CRD hydraulic system was not designed as a safety system, and no credit is given for its capability to inject water into the reactor coolant system. This evaluation depends on the availability of the CS system for reactor system makeup during cooldown.

4.0 CONCLUSIONS

Based on the information submitted by the licensee and obtained during our site visit to Oyster Creek, we have determined that the following review areas have not been addressed adequately in previous staff safety evaluations and should be resolved with the SEP:

1. Inadequate protection exists for postulated HELBs in the emergency condenser steam and condensate lines on the 75' elevation of the reactor building. The licensee is currently preparing a report to address HELBs in this area of the plant. The NRC staff position is that the licensee should submit a schedule by September 1, 1980, for modifications to be effected in this area of the plant to provide adequate protection from the effects of these postulated HELBs. The modifications to be installed must be in accordance with the acceptance criteria of Standard Review Plan 3.6.1 and provide protection for the emergency condenser isolation valves and controls and cable trays V22, V23, 41, 42, and 43. Justification for continued operation of the facility while the modifications are developed and implemented is based on the extremely low probability that (1) the HELB will occur in the time required to effect the modifications and (2) the postulated HELB would have the proper orientation to cause the worst case damage described above.

2. Postulated pipe breaks outside of the primary containment between the containment penetration and the first containment isolation valve have not been evaluated for the main steam lines, emergency condenser steam and condensate lines, and reactor water cleanup suction and discharge lines.

Currently the staff applies the provisions of Branch Technical Position MEB 3-1 (Reference 10) section B.1.b., to the review of the postulated break areas. The licensee will be required to compare the design of the Oyster Creek plant systems with these current regulatory provisions.

3. The effects of postulated pipe breaks in certain systems could result in damage to the containment isolation valves or power supply and control cables to the containment isolation valves for those systems. The combination of the single active failure provision and damage to the containment isolation valve could result in an unisolable break flow path. The systems of concern are the emergency condenser system (steam lines only), and reactor water cleanup system letdown and return lines. The staff currently applies the provisions of Branch Technical Position ASB 3-1 (Reference 11), Section B.2.c., to the review of these break areas. The licensee will be required to compare the design of the Oyster Creek systems with these current regulatory provisions.

4. The postulated break of certain high energy reactor water cleanup lines could damage cable tray 13A on the 51' elevation of the reactor building. The effects of damage to this tray have not been previously evaluated. The licensee will be required to provide an analysis of the effects of HELB damage to this cable tray.

5. A MS or MF HELB in the Turbine Building Mezzanine area could damage control cables for the CS system (and for other systems as previously described). The CS is needed to provide makeup water to the reactor system during a plant cooldown following the postulated HELB. The licensee will be

required to move or protect all CS control cables from the effects of these potential breaks.

6. A MELB in the cable spreading room could flood the room to some level before the floor drains could accommodate the flow. The TBCCW system leakage flow rate of 118 gpm is the largest potential MELB for this room. The licensee will be required to determine the depth of the flooding and what equipment would be affected by the flooding.

The staff is continuing this reevaluation of pipe breaks outside containment and will update this report as additional information is provided and conclusions are reached.

TABLE 1. EFFECTS OF PIPE BREAK OUTSIDE CONTAINMENT

Zone	Pipe Break	Affected Mitigating System	Affected Safe Shutdown System	Adequacy of Protection Remarks
Intake Structure	SWS, Screen Wash System, New Radwaste SWS, CW (MELB)*	None	ESWS	Adequate. Spray from MELB systems would not affect the ESWS pumps which are designed for outdoor use. The open-air intake structure precludes flooding of ESWS. However, if the structure is enclosed in the future, flood warning and protection for the ESWS must be considered.
Condensate Transfer Pump Area	Fire system (MELB)	None	Condensate Transfer System	Adequate. A fire system MELB in the cond. transfer pump enclosure (267 gpm) could result in flooding the pumps. Loss of the transfer pumps would not cause any plant transients or LOP. The pumps are designed for outdoor use. Loss of ether pump would result in a "pump" tripped" alarm in the control room to warn the operator of the flooding condition.
Reactor Build. (119')	Fire System (MELB)	None	None	Adequate. Hatches and floor drains are adequate to remove this leakage. Fire system MELB envelopes other MELBs in this zone, e.g., demin. water.

*See last page of Table 1 for list of abbreviations.

TABLE 1. (Continued)

Zone	Pipe Break	Affected Mitigating System	Affected Safe Shutdown System	Adequacy of Protection Remarks
Reactor Build. (95')	Emergency Condenser Steam Line (HELB)	None	Emergency Condensers	Adequate. Potential targets of HELB effects are the Emerg. Cond. shells, condensate supply lines, fill valve air supply line, level instruments, and cable tray 45. These interactions are discussed in the EVALUATION section.
	Fire System (MELB)	None	None	Adequate. Hatches and floor drains are adequate to remove leakage from a fire system MELB which envelopes other MELBs on this elevation.
	SLCS (HELB)	None	None	Adequate. A SLCS HELB outside containment would result in the containment isolation check valve inside containment seating with reactor system pressure. This would isolate the flow path from the reactor recirculation system to the pipe break.
Reactor Build. (75')	SLCS (HELB)	None	None	Adequate. See above remarks.
	Emergency Condenser Steam & Condensate Lines (HELB)	Emergency Condenser Isolation Valves	Emergency Condensers	Inadequate. Emerg. cond. steam or condensate HELBs may result in damage to the emerg. cond. containment isolation valves, and these HELBs could damage electrical cables in trays V22, V23, [illegible], 42 and 43. These breaks are discussed in the EVALUATION section.

TABLE I. (Continued)

Zone	Pipe Break	Affected Mitigating System	Affected Safe Shutdown System	Adequacy of Protection Remarks
Reactor Build. (75' cont.)	RBCCW, RWCU, Fire System (MELB)	None	None	Adequate. RBCCW MELB envelopes all other MELB's on 75' elev. (approx. 200 gpm). Sufficient drainage via hatches and floor drains exists to prevent flooding.
Reactor Build. (51')	RWCU (HELB)	RWCU Isolation Valves	Cable Trays 13A, 14A	Potentially inadequate. HELB may damage the RWCU system isolation valve motor operator or electrical power. This could prevent operation of the normally open valves. Effects of break in RWCU system could damage electrical cables in trays 13A and 14A. Trays 13A and 14A carry cables for the CS system, the ADS system, and the RPS. These effects are discussed in the EVALUATION section.
	SWS, RBCCW, Fire System (MELB)	None	None	Adequate. Adequate drainage exists via hatches, stairs, etc. to prevent flooding of equipment by the largest MELB on this elevation (a 20" SWS line break of approximately 550 gpm).

TABLE 1. (Continued)

Zone	Pipe Break	Affected Mitigating System	Affected Safe Shutdown System	Adequacy of Protection Remarks
Reactor Build. (23')	CRD Hydraulic Control Units (HELB)	None	Control Rods (RPS)	Adequate. Ruptures of high energy portions of CRD control units or damage to units resulting from pipe whip would result in either a tripped rod (scram) or loss of CRD supply to the affected control rod. In the latter case, the control rod could still be scrammed manually or automatically by the RPS.
	CRD (HELB)	None	CRD Modules (RPS) and Cable Trays 15, 16, 17, 18, 19, 20, 21, 22, and 23.	Adequate. See remarks above for CRD modules. The effects of CRD HELBs on this elevation were previously analyzed in Reference 2. These effects are discussed further in the EVALUATION section.
	Fire System, SWS, (MELB)	None	None	Adequate. Floor drains and hatches provide adequate drainage to prevent equipment flooding.
Reactor Build. (-19')	CRD Supply Line (HELB)	None	Torus	Adequate. Although, the torus may be damaged by pipe whip, no reactor trip (and assumed concurrent LOP) would result; and the plant could be shut down in an orderly manner. Damage would be restricted to the upper portions of the torus and none of the torus water volume would be lost.

TABLE 1. (Continued)

Zone	Pipe Break	Affected Mitigating System	Affected Safe Shutdown System	Adequacy of Protection Remarks
Reactor Build. (-19' cont.)	Torus (MELB)	None	Torus	Adequate. Flooding from the torus would not affect any safe shutdown equipment other than the torus itself. The reactor build. corner rooms are separated from the torus area by watertight doors. The torus area is designed to contain the leaked water volume of the torus without loss of the ability of the torus to function as a shutdown heat removal system.
	ESWS, RBEDT, MELB's from levels above -19' elev.	None	CS, Containment Spray	Adequate. Flooding of individual reactor build. corner rooms could disable pumps in that room: Room — Pumps NW — CS - B, D NE — CS - A, C SW — Contain. spray - 3, 4 SE — Contain. spray - 1, 2 Loss of a corner room would not result in a reactor trip or plant transient event, and redundant pumps are available in other corner rooms. SE and NE rooms have sump levels alarmed in control room. RBEDT is in NW corner room and has high and low level alarms in the control room. Flooding conditions in the torus area are indicated in the control room by high sump level alarms.

TABLE 1. (Continued)

Zone	Pipe Break	Affected Mitigating System	Affected Safe Shutdown System	Adequacy of Protection Remarks
Turbine Build. (Mezannine, 23')	MS, MF (HELB)	MSIV, MS Break Detection System	Control Room, Cable Spreading Room, Torus, Cable Trays 30, 31, 32, 12, 13, 14, 15, and ESW line.	Potentially inadequate. Potential targets of HELB effects are the control room and cable spreading room structures, torus shell, ESW piping (one loop) and cable trays which contain control and instrument cables for CS, contain. spray, ESW, RPS, and emergency diesel systems. These interactions are discussed in the EVALUATION section.
	Fire System, Demin. Water System (MELB)	None	4160 V Switchgear Panels 1C, 1D, Battery C Switchgear	Adequate. Spray from MELB could impact Bettery C switchgear and both 4160 V switchgear panels 1C, 1D. Spray would not enter the DC switchgear panel; and an enclosure is being constructed around the 4160 V switchgear (as a result of the fire protection review) and will prevent spray from MELB's from impacting the switchgear.
Cable Spreading Room	TBCCW, Fire System, Demin. Water (MELB)	None	AC and DC Emergency Power, RPS	Potentially inadequate. A TBCCW MELB 118 gpm could flood the room to a depth of (to be determined). The effects of flooding on redundant emergency power and RPS motor generator sets must be determined.

TABLE 1. (Continued)

Zone	Pipe Break	Affected Mitigating System	Affected Safe Shutdown System	Adequacy of Protection Remarks
Turbine Build. (Basement)	CW, ESW, TBCCW (MELB)	None	None	Adequate. CW MELB have been previously analyzed in Refs. 9 (question #10) and 10. Flooding from smaller MELB in the condenser room is enveloped by the CW break. MELBs outside the condenser room would be alarmed in the control room by drain sump alarms, and no safe shutdown or break mitigating systems are affected. Operator action is require to stop MELB's in this area. ESW function, if required, would not be lost in either train from a MELB.
	MF, Condensate (HELB)	None	None	Adequate. HELB causes reactor trip, but no mitigating or safe shutdown systems are affected by the break.

TABLE 1. (Continued)

List of Abbreviations

ADS - Automatic Depressurization System (part of the emergency core cooling systems)

CRD - Control Rod Drive

CS - Core Spray System (part of the emergency core cooling systems)

CW - Circulating Water System

demin. - demineralized

ESWS - Emergency Service Water System

HELB - High Energy Line Break

LOCA - Loss of Coolant Accident

LOP - Loss of Offsite Power

MELB - Moderate Energy Line Break

MF - Main Feed

MS - Main Steam

RBCCW - Reactor Building Closed Cooling Water System

RBEDT - Reactor Building Equipment Drain Tank

RPS - Reactor Protection System

RWCU - Reactor Water Clean-Up System

SLCS - Standby Liquid Control System

SWS - Service Water System

TBCCW - Turbine Building Closed Cooling Water System

REFERENCES

1. NRC letter, A. Biambusso to CE Co., dated December 18, 1972.

2. Oyster Creek FDSAR Amendment 75, dated July 1, 1974.

3. Revision 1 to Oyster Creek FDSAR Amendment 75, dated December 24, 1974.

4. Revision 2 to Oyster Creek FDSAR Amendment 75, dated March 24, 1975.

5. Revision 3 to Oyster Creek FDSAR Amendment 75, dated April 25, 1975.

6. Revision 4 to Oyster Creek FDSAR Ame dment 75, dated June 1, 1976.

7. Revision 5 to Oyster Creek FDSAR Amendment 75, dated September 21, 1976.

8. NRC letter, G. Lear to I. Finfrock, Jr., dated December 27, 1976.

9. Oyster Creek Supplement No. 6 (Addendum 1) to the Application for a Full Term License, Docket No. 50-219, dated November 21, 1973.

10. JCP&L, I. Finfrock, Jr. to D. Ziemann, dated March 13, 1974.

11. JCP&L letter, I. Finfrock, Jr. to D. Ziemann, dated July 3, 1974.

12. SEP Review of Safe Shutdown Systems for the Oyster Creek Nuclear Power Plant. (SEP Topics VII-3, V-10.3, V-11.A, V-11.B, X).

REQUEST FOR ADDITIONAL INFORMATION
SEP TOPIC III-5.B, PIPE BREAK OUTSIDE CONTAINMENT
OYSTER CREEK

1. Provide a comparison of the design of the containment penetration piping outside containment between the containment and the outermost containment isolation valves for the main steam lines, emergency condenser steam and condensate lines, and reactor water cleanup lines with the provisions of section B.1.b of Branch Technical Position MEB 3-1 (appended to Standard Review Plan 3.6.2) in sufficient detail to identify the degree of conformance with and deviations from these provisions.

2. Provide a comparison of the design of the containment penetration piping outside containment for the emergency condenser steam lines and reactor water cleanup lines with the provisions of section B.2.C of Branch Technical Position ASB 3-1 (appended to Standard Review Plan 3.6.1) in sufficient detail to identify the degree of conformance with and deviations from these provisions.

3. Provide an evaluation of the potential effects of damage to cable tray 13A on the 51' elevation of the reactor building from a postulated break in the reactor water cleanup system. Consider the effects of pipe whip, jet impingement, and high temperature on the electrical cables.

4. Provide an evaluation of potential flooding in the cable spreading room from a postulated break in the fire water system or turbine building closed cooling water system. Determine the depth of flooding, what equipment could be flooded, and the effects of loss of that equipment.

ENCLOSURE 3

STAFF POSITIONS ON SEP TOPIC III-5.B
PIPE BREAKS OUTSIDE CONTAINMENT
OYSTER CREEK NUCLEAR PLANT

1. Because inadequate protection exists from the effects of postulated breaks in the emergency condenser steam and condensate lines on the 75' elevation of the reactor building, the licensee should submit, by September 1, 1980, a schedule for modifications to be installed to provide adequate protection from these postulated breaks. The modifications must be in accordance with the acceptance criteria of Standard Review Plan 3.6.1 and provide protection for the emergency condenser isolation valves and controls and for cable trays V22, V23, 41, 42, and 43.

2. To provide adequate protection from the effects of postulated main steam and main feed line breaks in the turbine building mezzanine area, the licensee should move or provide protection for all core spray system control cables in that area. By September 1, 1980, the licensee should provide a schedule for resolution of this issue.