JCP&L / GPU

Jersey Central Power & Light Company
Madison Avenue at Punch Bowl Road
Morristown, New Jersey 07960
(201) 455-8200

October 6, 1980

Mr. Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing
Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Crutchfield:

Subject: Oyster Creek Nuclear Generating Station
Docket No. 50-219
SEP Topic III-5.B, Pipe Break
Outside Containment

The NRC evaluation of SEP Topic III-5.B, Pipe Breaks Outside Containment for the Oyster Creek Nuclear Generating Station is contained in NRC Letter dated July 10, 1980.

Enclosed are 10 copies of JCP&L responses to Enclosure 2, Request for Additional Information - SEP Topic III-5.B, Pipe Break Outside Containment, and Enclosure 3, Staff Positions on SEP Topic III-5.B, Pipe Break Outside Containment, of the NRC letter dated July 10, 1980. These responses were discussed with NRC staff representatives during a meeting at the NRC on September 9, 1980. If you should have any further questions, please call Mr. J. Knubel of my staff at (201) 455-8753.

Very truly yours,

Ivan R. Finfrock, Jr.

Ivan R. Finfrock, Jr.
Vice President

la

Enclosure

8010090415

P

ATTACHMENT 1

JCP&L RESPONSES TO NRC STAFF POSITIONS ON SEP TOPIC III-5.B PIPE BREAKS OUTSIDE CONTAINMENT OYSTER CREEK NUCLEAR PLANT

October 1, 1980

1. INTRODUCTION

The NRC regulatory staff evaluation of pipe breaks outside containment (SEP Topic III-5.B) for the Oyster Creek Nuclear Generating Station is contained in Reference 1. The NRC evaluation concludes that modifications should be made to provide increased protection from the effects of postulated high energy line breaks (HELB) in two areas of the plant. These two areas are:

- Postulated HELBs in the emergency condenser piping on the 75 foot elevation of the reactor building, and
- Postulated HELBs in the main steam and feedwater piping in the turbine building mezzanine area.

The licensee, Jersey Central Power and Light Company (JCP&L) was requested to submit by September 1, 1980 1/ a schedule for completion of plant modifications to provide increased protection from the effects of HELBs in these two areas. The planned plant modifications and the schedule for their completion are described in this report.

In addition, the NRC evaluation included a request for additional information to be submitted by September 14, 1980 1/ (i.e., 60 days of JCP&L's receipt of the NRC evaluation). This information is provided in a separate report. If additional plant modifications are determined to be necessary, a schedule for their completion will also be provided.

1/ On August 6, 1980, the due date for the submittal was extended 30 days.

2. PLANT MODIFICATIONS

2.1 REACTOR BUILDING-EMERGENCY CONDENSER PIPING

2.1.1 DESCRIPTION

The emergency condenser system is a standby, high-pressure system for removal of fission-product decay heat after reactor isolation when the main turbine condenser is not available as a heat sink. A schematic of the emergency condenser system is shown in Figure 1. The two emergency condensers (designated A and B) are located on the 95 foot elevation of the reactor building. Each emergency condenser has two loops. Normally, both loops are placed into operation simultaneously, and either loop can be activated or shut down separately.

The steam supply and condensate return lines are routed from containment penetrations on the 75 foot elevation to the condensers on the 95 foot elevation and back. Each line contains two motor-operated isolation valves, one a-c and one d-c, as follows:

a. Steam Supply Lines

Both isolation valves on the steam supply lines are located outside the drywell. The valve bodies are welded into one assembly with no intermediate pipe between them. An 18-inch guard pipe surrounds the steam supply line outside the drywell penetration, and is welded to the first isolation valve on one end and to the 24-inch containment penetration on the other end, as shown in Figure 2, so that the penetration sleeve up to the valve body is, in effect, part of the drywell. The isolation valves on the steam supply lines are normally open so that the tube bundles in the condensers are at reactor pressure during standby conditions.

b. Condensate Return Lines

The condensate return lines contain one isolation valve outside and one isolation valve inside the drywell. The drywell penetration up to the isolation valve outside the drywell is protected with a guard pipe as described above for the steam supply lines. The isolation valves outside the drywell are normally closed whereas the isolation valves inside the drywell are normally open. The emergency condenser system is placed into operation by opening the normally closed isolation valves outside the drywell.

Each of the steam supply lines and condensate return lines have pipe break detectors, (i.e., differential pressure taps) located on elbows on the piping inside the drywell. The steam supply and condensate return isolation valves close automatically in the affected loop in the event of a line flow greater than three times rated flow.

2.1.2 EFFECTS OF HELBs

A plan view of the emergency condenser piping on the 75 foot elevation of the reactor building is shown in Figure 3. The effects of postulated HELBs in the emergency condenser piping are discussed in References 1 and 2. In essence, they conclude that protection from the effects of HELBs in the emergency condenser piping on the 75 foot elevation should be increased for the following reasons:

- Postulated pipe breaks in the steam supply and condensate return lines in the vicinity of the containment isolation valves could result in damage to both isolation valves, including power and control cables, on the steam supply line containing the break. Such damage could prevent the closing of these normally open valves on demand.

- Postulated pipe breaks at other locations in the condensate return lines could result in damage to Cable Trays V22, V23, 41, 42 and 43 and conduits containing power and control cables to both isolation valves on the steam supply lines containing the break, preventing their closure on demand.

2.1.3 PLANNED PLANT MODIFICATIONS

JCP&L has evaluated possible plant modifications for mitigating the effects of HELBs in the emergency condenser piping on the 75 foot elevation of the reactor building. Based on our evaluation, we have concluded that the following plant modification will provide increased protection for a) the isolation valves on the steam supply lines, and b) Cable Trays V22, V23, 41, 42 and 43 and conduits containing power and control cables to the isolation valves on the steam supply lines. The modification consists of installing a leak detection system on the emergency condenser piping as described below.

Leak Detection System

A leak detection system will be installed on the emergency condenser piping on the 75 foot elevation of the reactor building. The leak detection system will be capable of detecting and locating small leaks in the pipe before the break size becomes large enough to cause damage to the isolation valves, cable trays, and conduits due to jet impingement or pipe whip. This will allow plant operators sufficient time to take appropriate actions such as to visually inspect the area, isolate the line containing the leak, or shut down the plant.

At the present time, the following two types of leak detection systems are being evaluated and one will be selected for installation.

a. Acoustic Monitoring System

This system, provided by Westinghouse Electric Corporation, uses acoustic sensors mounted at selected locations on the pipe to detect and locate leakage. The system is expected to have a leak detection sensitivity of 0.1 gpm in actual plant piping systems. (Laboratory data indicate that leaks as small as 0.02 gpm can be detected in a few seconds.) This leak detection capability exceeds the requirements of NRC Regulation Guide 1.45 that a 1 gpm leak in the primary coolant system be detected within one hour.

The principle of operation is as follows. As pressurized fluid escapes through a crack in a metal boundary, the turbulent flow generates metalborne acoustic waves which are detected by acoustic transducers (piezoelectric elements) mounted at selected locations on the piping system. The signal passes through a small pre-amplifier and then is transmitted to an electronics package where the signal is further amplified and the true rms level is extracted and transmitted as an analog signal to a microprocessor based leak detection system. The microprocessor contains analog-to-digital converters and a microcomputer with memory. The microprocessor analyzes the data from each of the installed acoustic channels and provides continuous on-line monitoring of the piping system. Control room alarms are set at predetermined rms levels above the background noise level of the system. In the event of an alarm, the control room display provides the leak location and rms magnitude.

Since each transducer can be used as a sender as well as a receiver of acoustic signals, the system can be verified on-line by transmitting a signal to one transducer and checking to see if the signal is picked-up by the adjacent transducers.

b. <u>Moisture Sensing Unit</u>

This system, provided by Nutech, Inc., consists of a sensing element, shown in Figure 4, which provides an electrical signal when activated by moisture, and an indicating device, which converts the signal to a visual and/or audible alarm to alert the plant operators to the presence of moisture at a given location.

The principle of operation is as follows. The sensing element is placed over a small weep hole in the pipe insulation. The sensing element contains a pair of electrical conductors separated by a small gap. A strip of polyester webbing is placed over the pair of conductors so that there is a large resistance between the conductors. The exterior surface of the element is impervious to moisture so that the only moisture which can affect the element is that which comes through the weep holes. When the polyester web between the two conductors is wetted by leaking fluid, chemicals impregnated in the polyester web are activated by the water and the web becomes electrically conducting. The indicating device provides a continuous low d-c voltage to the sensor through insulated connecting wires. When the sensor is activated by moisture, the decrease in resistance across the gap between conductors actuates an indicating light or alarm in the control room to alert the plant operators. The operators can then determine the location of the leak and take appropriate action.

In addition, temperature and/or radiation monitors will be placed in the general area of the emergency condenser piping on the 75 foot elevation of the reactor building. These area alarms will provide redundant leak detection capability by sensing the increase in temperature and radiation level of the building environment in the event of a small leak.

2.2 TURBINE BUILDING MEZZANINE AREA - MAIN STEAM AND FEEDWATER PIPING

2.2.1 DESCRIPTION

The main steam and feedwater piping penetrate the containment at the 26'-6" and 33'-0" elevations, respectively. The pipes are routed to the turbine building through a pipe tunnel between the reactor and turbine buildings. There are two isolation valves on each line, one inside and one outside the containment.

2.2.2 EFFECTS OF HELBs

Plan views of the main steam and feedwater piping in the pipe tunnel and turbine building mezzanine area are shown in Figures 5 and 6, respectively. The effects of postulated HELBs in the main steam and feedwater lines in the pipe tunnel and turbine building mezzanine area are discussed in Reference 3. In summary, HELBs in the main steam and feedwater piping in the pipe tunnel and turbine building mezzanine area could cause damage to the following electrical cables.[1/]

- System II control cables for the emergency service water (ESW) system.
- System II control cables for the core spray (CS) system.
- System II control cables for the containment spray system.

The NRC evaluation of the effects of HELBs in the main steam and feedwater piping in the pipe tunnel and turbine building mezzanine area with regard to core cooling is given in Reference 1, as follows.

Since a main steam or feedwater break would result in a turbine and reactor trip as a direct consequence of the break, a further loss of site-power together with a single active component failure must be assumed in accordance with NRC regulations. If the single active failure is assumed to occur in Diesel Generator #1, this would result in the

---

1/ In addition, HELBs in the main steam feedwater piping also produce acceptable interactions with reactor protection system, d-c power supply, main steam isolation valves, and feedwater piping.

additional loss of function of System I of the ESW, CS, and containment spray systems. Thus, an HELB of the main steam or feedwater piping in the turbine building mezzanine area could result in the total loss of the ESW, CS, and containment spray systems.

The ESW system is used for long term cooling of the reactor core following a LOCA by cooling the containment spray system which cools the torus water which is circulated through the reactor by the CS system. For the above postulated HELB in which the ESW, CS, and containment spray systems are lost, long term cooling of the reactor core could be performed using the shutdown cooling system.

Thus the complete loss of the ESW, CS, and containment spray systems is considered acceptable from a safe shutdown standpoint. However, loss of the CS system would severely restrict the ability of the plant operators to keep the core covered with water during recovery from the postulated accident. For such an incident, water could be added to the reactor vessel by the control rod drive (CRD) hydraulic system. However, since the CRD hydraulic system was not designed as a safe shutdown system, no credit has been taken for it in safety analyses. Therefore, additional protection is desirable to assure the availability of at least one train of the CS system to supply water to the reactor coolant system for postulated HELBs in the main steam and feedwater systems in the pipe tunnel and turbine building mezzanine area.

The NRC evaluation is not correct with regard to CS availability for the above scenerio, in that each of the two redundant CS systems is completely single active failure proof. That is, each CS system contains sufficient power supplies, valves, pumps, etc., so that no single active failure would prevent the system from operating. For the above postulated HELB which included a) pipe break in main system or feedwater piping in the turbine building mezzanine area with subsequent damage to CS System II control cables as a direct consequence of the break, b) turbine and reactor trips with assumed loss of off-site power, and c) assume single active failure of Diesel Generator #1, the undamaged System I CS could be run from Diesel Generator #2. Therefore, at least one train of the CS system is available to supply water to the reactor coolant system. Accordingly, no plant modifications are required in the turbine building mezzanine area.

3. SCHEDULE

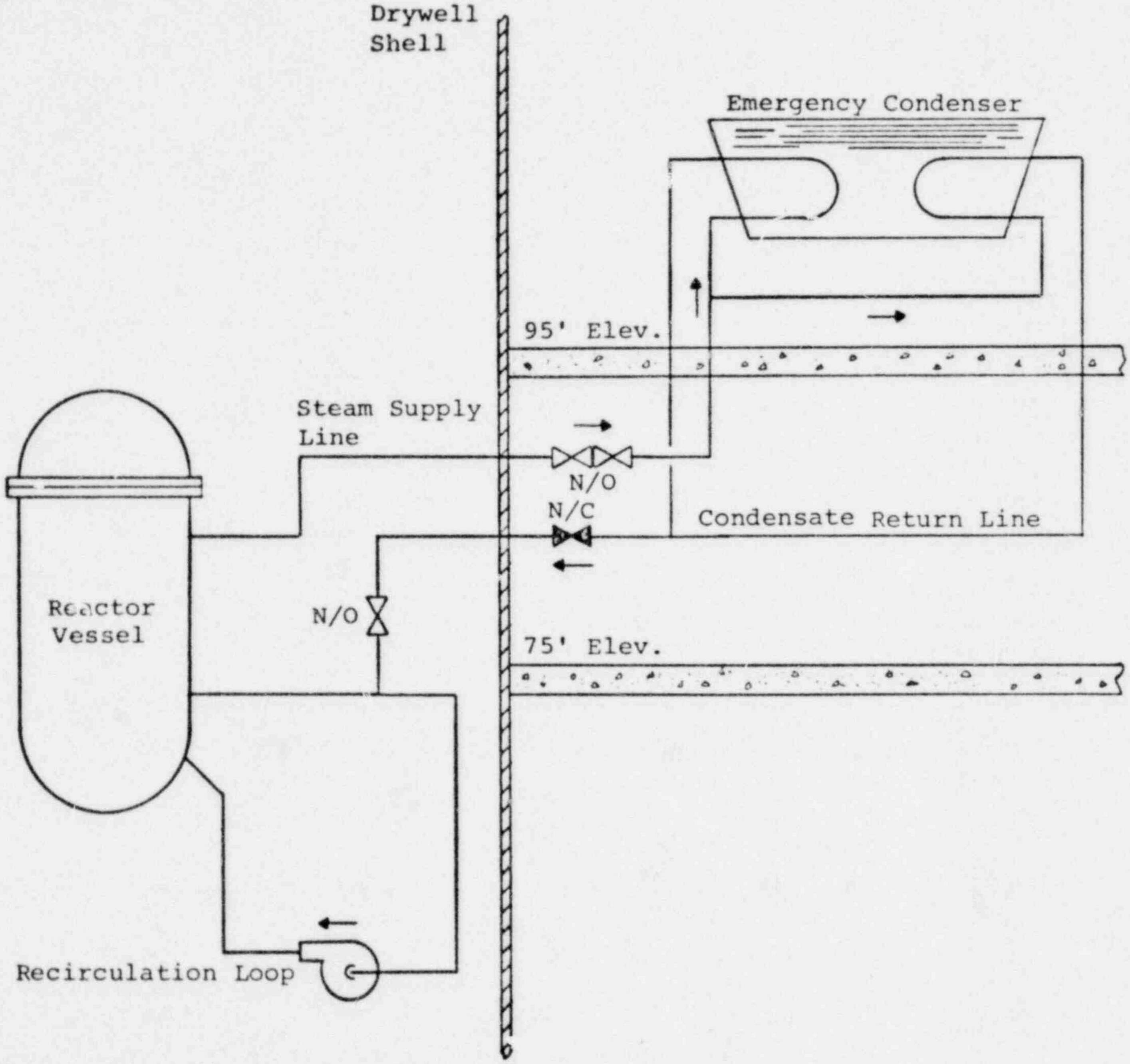
The plant modifications described in the previous section of this report to provide additional protection from the effects of pipe breaks outside containment in the emergency condenser piping will be completed by the end of the next scheduled refueling shutdown. Operation of the plant until the next scheduled refueling shutdown is justified because of a) the extremely low probability of an HELB during the next cycle of operation and b) the extremely low probability that the orientation of the break would be of the direction that would cause the worst case damage described in this report.

4. REFERENCES

1. NRC Letter, D.M. Crutchfield to I.R. Finfrock, Jr., (JCP&L) dated July 10, 1980.

2. JCP&L Letter, D.A. Ross to NRC (Region I) dated February 1980.

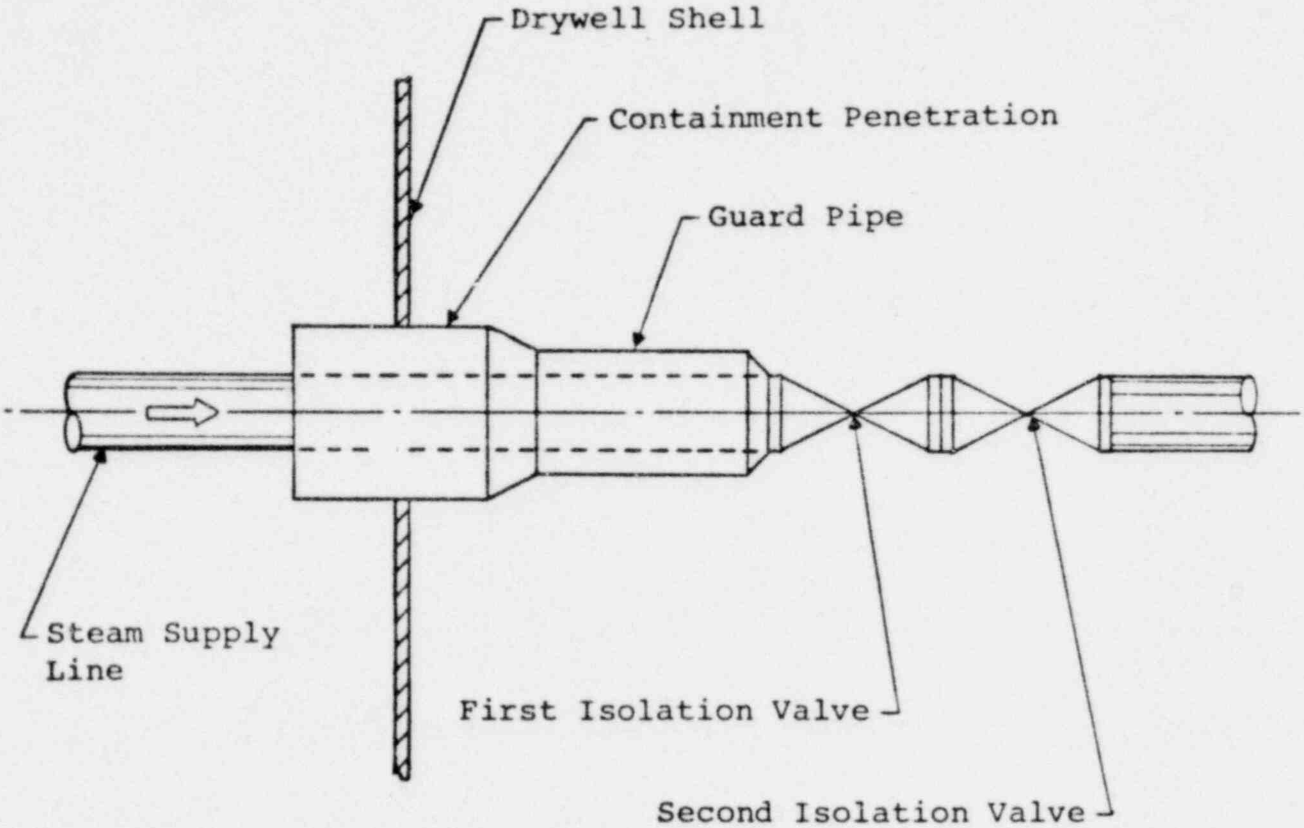
3. Oyster Creek FDAR Amendment 75, dated July 1, 1974.



NOTE: Only One of Two Systems Shown.

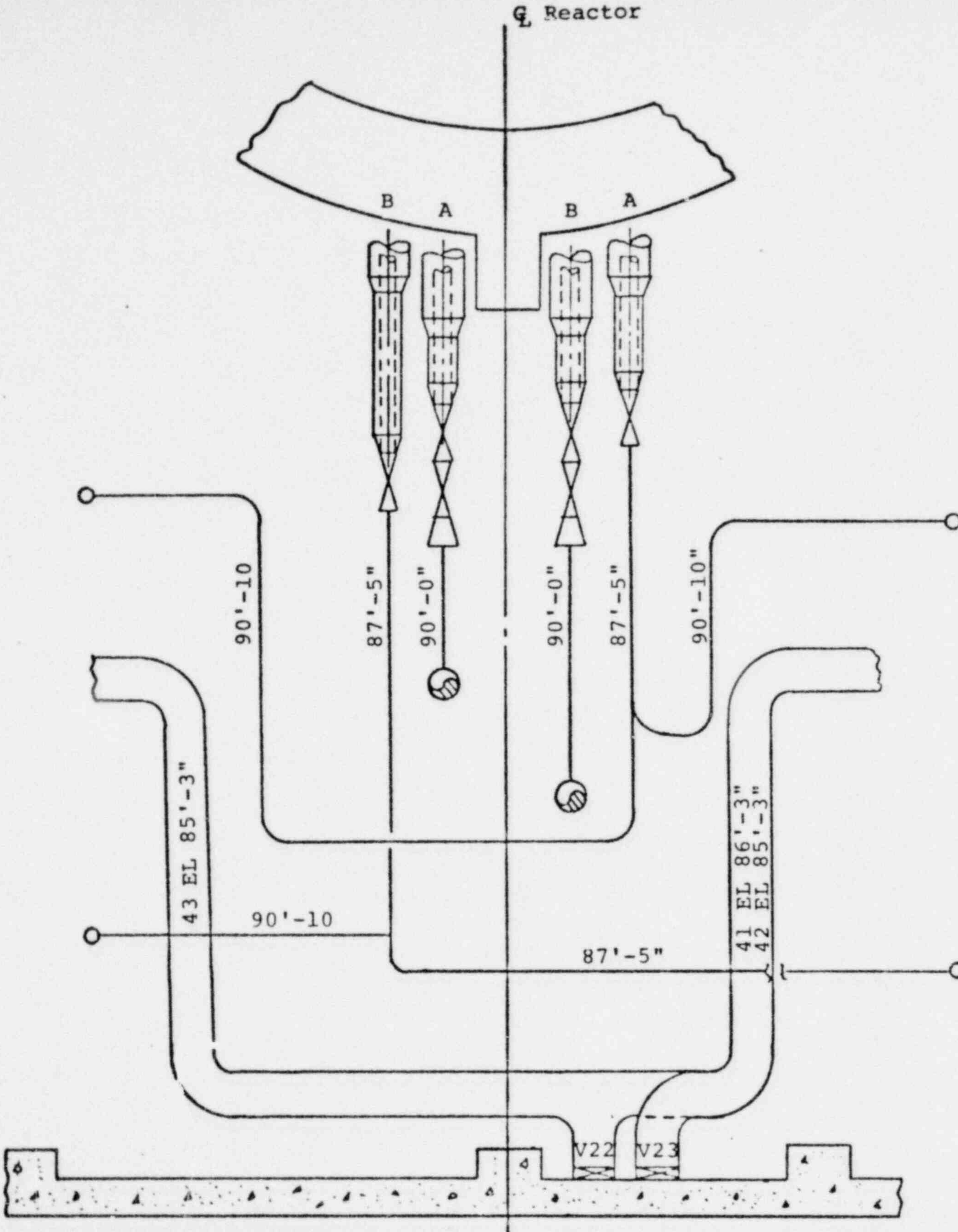
SCHEMATIC DIAGRAM
EMERGENCY CONDENSER SYSTEM

FIGURE 1



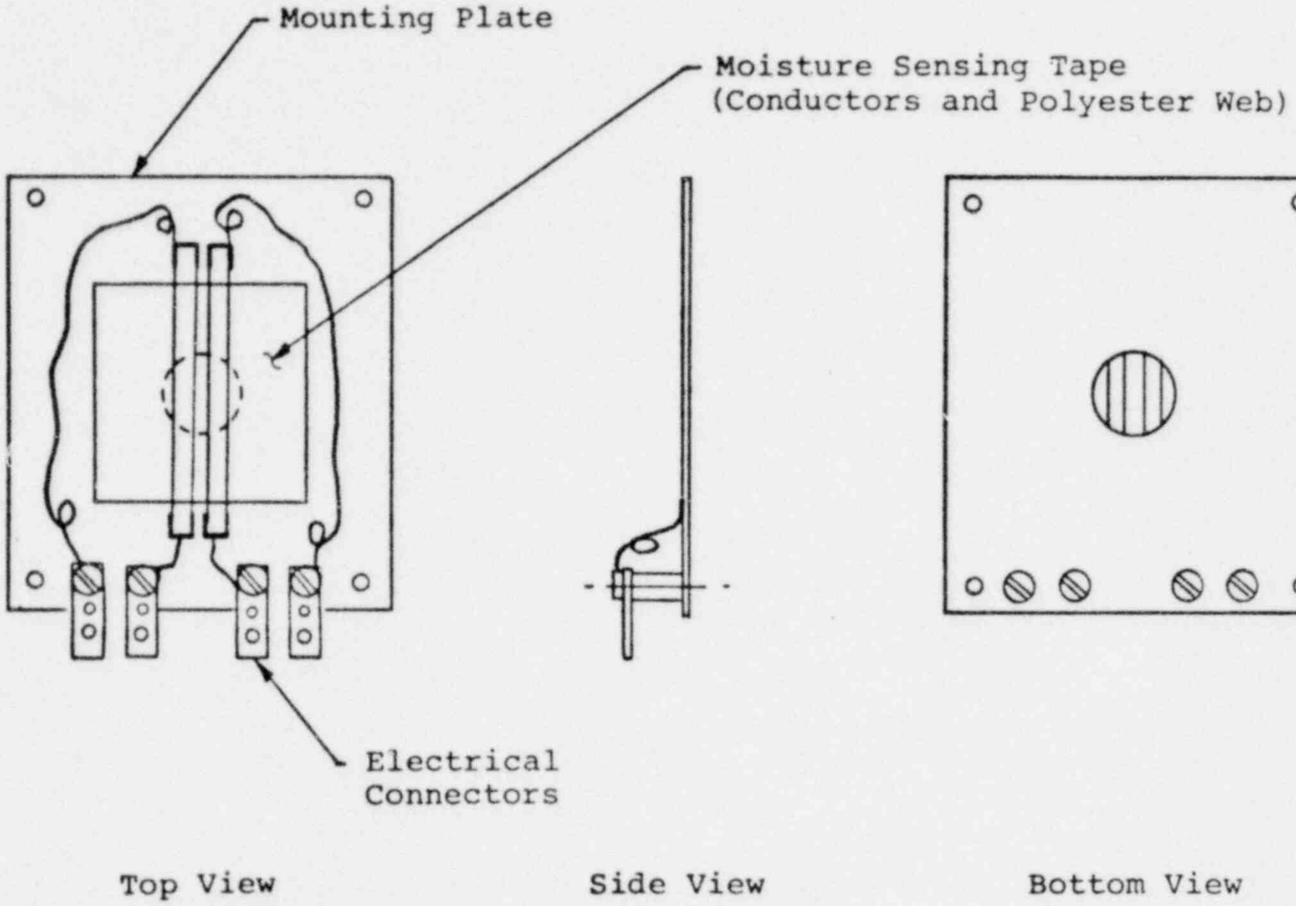
CONTAINMENT PENETRATION
EMERGENCY CONDENSER SYSTEM

FIGURE 2



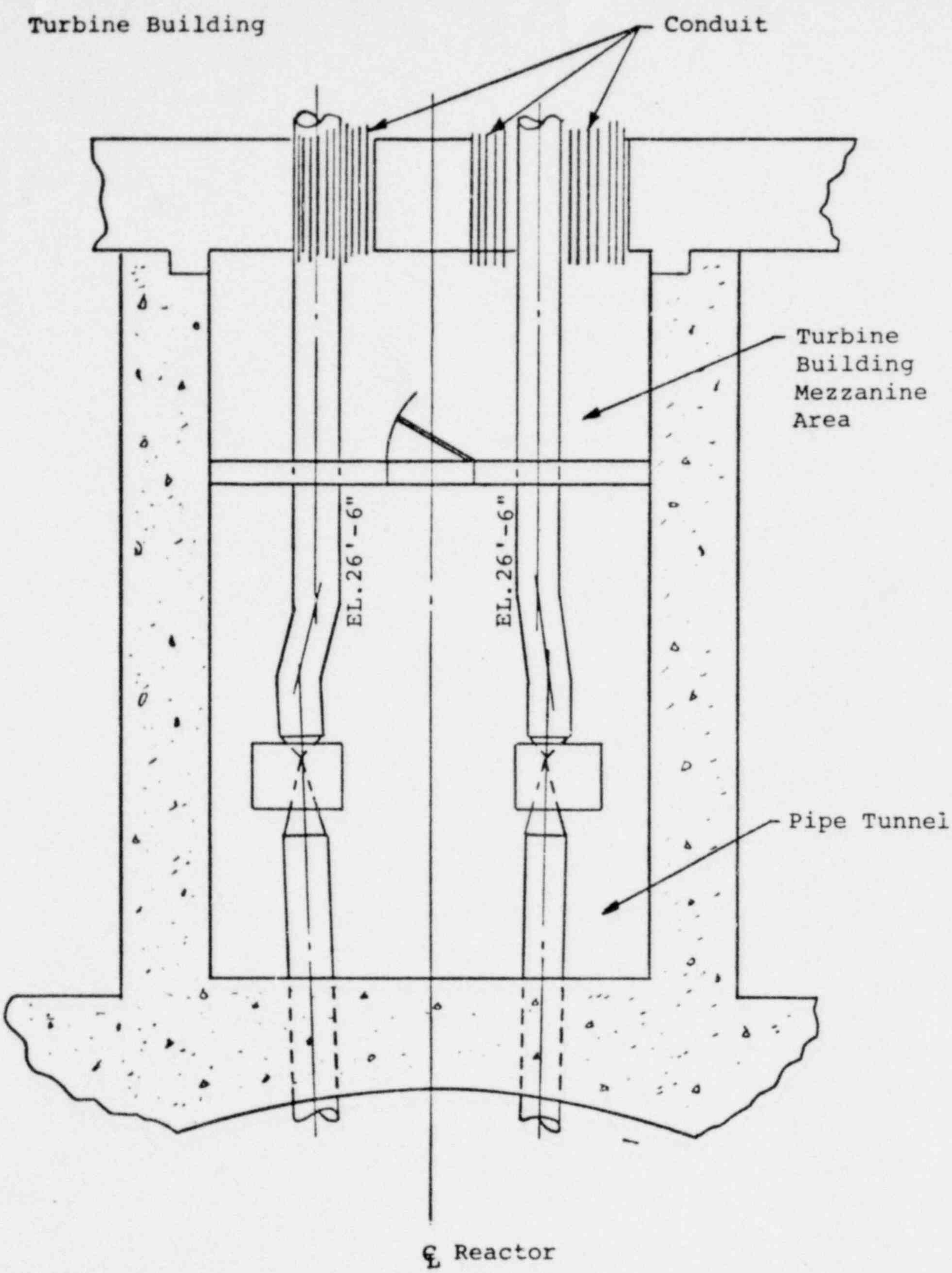
PLAN VIEW
EMERGENCY CONDENSER PIPING

FIGURE 3



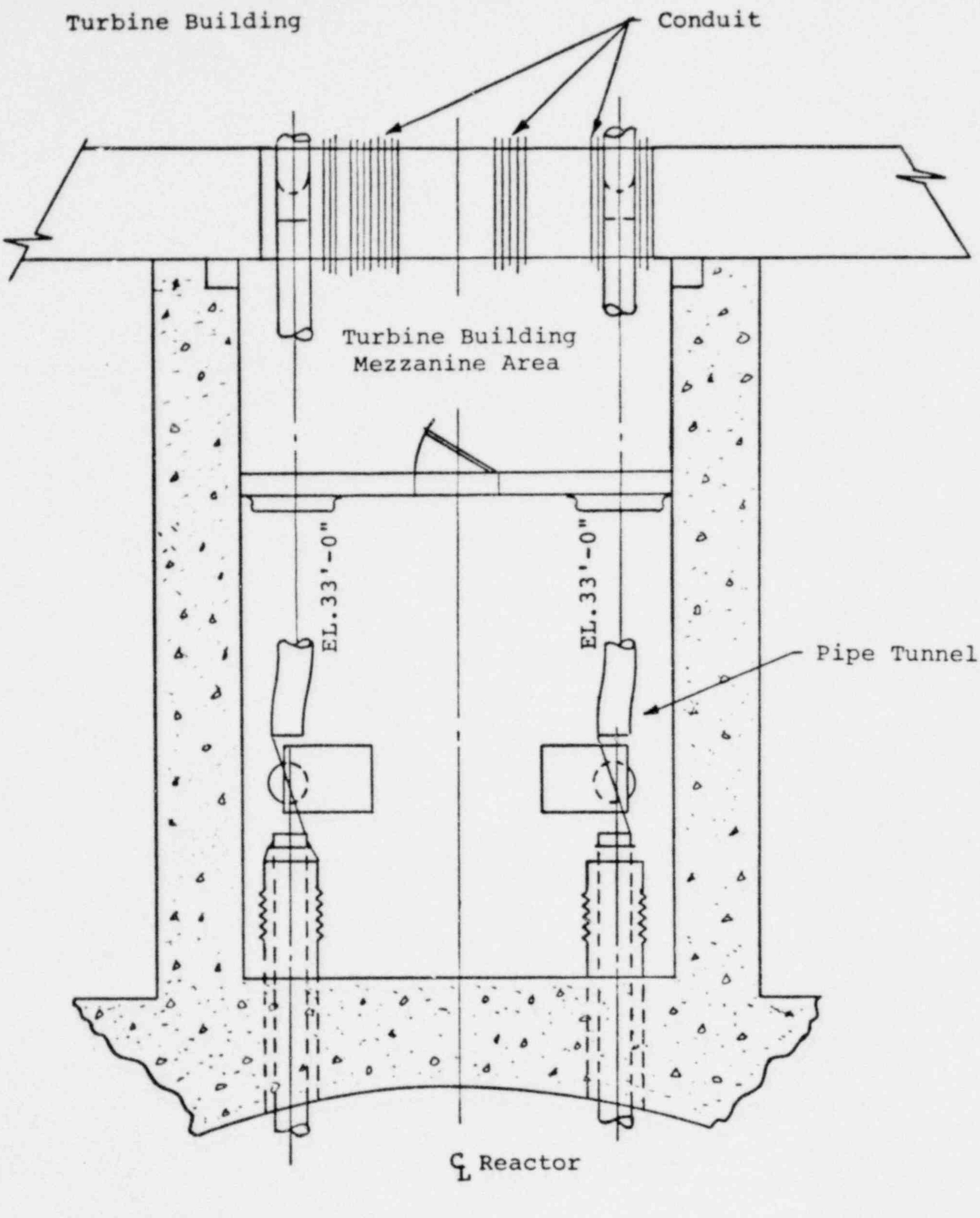
MOISTURE SENSING ELEMENT

FIGURE 4



PLAN VIEW
MAIN STEAM PIPING

FIGURE 5



PLAN VIEW
FEEDWATER PIPING

FIGURE 6

ATTACHMENT 2

JCP&L RESPONSES TO NRC REQUEST FOR ADDITIONAL INFORMATION SEP TOPIC III-5.B, PIPE BREAK OUTSIDE CONTAINMENT OYSTER CREEK

October 1, 1980

1. INTRODUCTION

The NRC regulatory staff evaluation of pipe breaks outside containment (SEP Topic III - 5.B) for the Oyster Creek Nuclear Generating Station is contained in Reference 1. The NRC evaluation included a request for additional information to be submitted by Jersey Central Power and Light Company (JCP&L) by September 14, 1980[1/] (i.e., 60 days of JCP&L's receipt of the NRC evaluation).

The additional information requested by NRC is provided in the following section of this report.

1/ On August 6, 1980, the due date for the submittal was extended 30 days.

2. RESPONSES TO NRC REQUEST FOR ADDITIONAL INFORMATION

The additional information requested by the NRC regulatory staff to enable them to complete their review of pipe breaks outside containment for the Oyster Creek Nuclear Generating Station is identified in Enclosure 2 of Reference 1. Each NRC request from Enclosure 2 is reproduced below along with the response.

REQUEST

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 the deviations from these provisions.

RESPONSE

Branch Technical Position (BTP) MEB 3-1 states that pipe breaks need not be postulated in fluid system piping in containment penetration areas that meet the requirements of Section B.1.b. Typical containment penetrations for the main steam, emergency condenser, and reactor water cleanup lines are shown in Figures 1, 2 and 3, respectively. A comparison of the design of the Oyster Creek containment penetrations and piping outside containment for the main steam, emergency condenser, and reactor water cleanup lines with the provisions of Section B.1.b of BTP MEB 3-1 is given in Table 1. In summary, it is concluded that the Oyster Creek containment penetrations and piping outside containment meet the intent of the provisions of Section B.1.b of BTP MEB 3-1 except as noted below.

a. BTP MEB 3-1 requires that fluid system piping in containment penetration areas meet the requirements of NE-1120 of the ASME Code Section III. NE-1120 requires that pipes and valves which are part of the containment or are attached to the containment shall be classified as Class 1 or Class 2 and meet the requirements of the applicable subsection of the Code. The main steam, emergency condenser, and reactor water cleanup piping systems for Oyster Creek were designed and constructed in accordance with the Code for Pressure Piping ASA B31.1 (1955). However, the margin of safety provided by ASA B31.1 is considered to be comparable to that provided by ASME Code Section III. Therefore, the Oyster Creek piping is considered to meet the intent of BTP MEB 3-1 in this regard.

b. The design stress limits for Oyster Creek piping for normal and upset plant conditions plus seismic loads were different from those specified in BTP MEB 3-1, as indicated in Table 1. However, the margin of safety provided by the Oyster Creek limits is considered comparable to that provided by the limits specified in BTP MEB 3-1.

c. BTP MEB 3-1 requires that piping between the containment penetration and pipe whip restraint located outboard of the outside isolation valve be analyzed for postulated pipe breaks beyond the restraint and meet certain stress limits. As indicated in Table 1, there are no pipe whip restraints on the Oyster Creek main steam, emergency condenser, and reactor water cleanup lines outborad of the outside isolation valves. Because of the lack of pipe whip restraints, the Oyster Creek piping systems between the containment penetration and isolation valve outside containment do not meet the stress requirements of BTP MEB 3-1 with regard to pipe breaks beyond the outside isolation valve.

It should be noted that the Oyster Creek containment penetrations have been analyzed for postulated pipe breaks outside containment. The basis for design of the containment penetrations for high pressure piping systems is given in Reference 2. The design objectives were as follows:

- The fundamental basis for penetration protection must be preserved, namely: "preclude any cause for failure of the drywell penetrations and one of its associated isolation valves."
- The penetration design analysis is to take into account all the accident conditions and combinations of loads as outlined in FDSAR Amendment No. 11, Section III-15 (Reference 3).
- The design is to be based on analytical methods and stress limits established by recognized design codes, preferably ASME Code for setting limits on allowable pressure vessel stresses.
- Established engineering practices for calculating reaction forces from pipe ruptures are to be used.

Based on the design objectives given in Reference 2, restraints are provided on the main steam, emergency condenser, and reactor water cleanup penetrations as follows:

PENETRATION	INSIDE DRYWELL	OUTSIDE DRYWELL
Main Steam	Restraint not Required	Restraint Installed
Emergency Condenser	Restraint not Required	Restraint Installed
Reactor Water Cleanup	Restraint not Required	Restraint not Required

Since the main steam, emergency condenser, and reactor water cleanup lines between the containment penetrations and isolation valve outside containment do not meet requirements of BTP MEB 3-1 in full, pipe breaks are assumed between the containment penetration and outside isolation valve. Effects of these postulated breaks are summarized below.

a. Main Steam

The effects on safe shutdown systems of HELBs in the main steam line between the containment penetration and outside isolation valve are the same as for HELBs beyond the isolation valve. As discussed in References 4 and 5, such breaks produce acceptable interactions with the emergency service water, core spray, containment spray, reactor protection, d-c power, and feedwater systems. However, since there would be only one isolation valve (i.e., the valve inside containment) between the reactor vessel and the break location, an assumed single active failure of this valve would result in an unisolated break. Continued operation of the plant is justified on the basis of the extremely low probability of a break in the short section of pipe between the containment penetration and isolation valve outside containment along with an assumed non-mechanistic failure of the isolation valve inside containment.

b. Emergency Condenser

The emergency condenser piping is protected with a guard pipe from the containment penetration up to the isolation valves outside containment. Thus, postulated HELBs in the emergency condenser piping between the containment penetration and outside isolation valve are considered to be the sam. as pipe breaks inside containment. Accordingly, such brea)s do not have any effect on safe shutdown systems located outside containment.

c. Reactor Water Cleanup

The effects on safe shutdown systems of HELBs in the reactor water cleanup lines between the containment penetration and isolation valve outside containment are the same as for HELBs beyond the isolation valve. As discussed in Reference 5 and the following sections of this report, such breaks produce acceptable interactions with the core spray and automatic depressurization systems. However, since there would be only one isolation valve (i.e., the valve inside containment) between the reactor vessel and the break location, an assumed single active failure of this valve would result in an unisolated break. Continued operation of the plant is justified on the basis of the extremely low probability of a break in the section of pipe between the containment penetration and isolation valve outside containment along with an assumed non-mechanistic failure of the isolation valve inside containment.

REQUEST

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 the deviations from these provisions.

RESPONSE

BTP ASB 3-1 requires that fluid system piping in containment penetration areas meet the provisions of B.2.c. A comparison of the containment penetrations and piping outside containment for the emergency condenser steam lines and reactor water cleanup lines with the provisions of Section B.2.c of BTP ASB 3-1 is given in Table 2. In summary, it is concluded that the Oyster Creek containment penetrations and piping meet the intent of the provisions of Section B.2.c of BTP ASB 3-1 except as noted below:

a. BTP ASB 3-1 requires that high pressure piping in containment penetration areas meet the stress requirements of Section B.1.b of BTP MEB 3-1. As discussed in the response to the previous request for additional information, the emergency condenser and reactor water cleanup lines do not meet the stress requirements of BTP MEB 3-1 for pipe break loads beyond the outside isolation valve. It is concluded that the emergency condenser and reactor water cleanup lines meet the intent of the stress requirements of

BTP MEB 3-1 for normal and upset conditions plus seismic loads, and that the containment penetrations meet the intent of the stress requirements of BTP MEB 3-1 for normal and upset conditions plus seismic loads and for pipe rupture loads.

b. BTP ASB 3-1 requires that pipes be provided with pipe whip restraints that are capable of resisting the bending and torsional moments produced by postulated pipe breaks beyond the containment isolation valves. The restraints should be located reasonably close to the containment isolation valves and should be designed so that neither isolation valve operability nor the leak-tight integrity of the containment will be impaired.

As discussed in the response to the previous request for additional information, there are no pipe whip restraints on the emergency condenser and reactor water cleanup lines outboard of the containment isolation valves.

The intent of BTP ASB 3-1 is that nuclear plants should be able to withstand postulated pipe breaks outside containment (i.e., beyond the outside isolation valve) taking into account the direct results of such breaks and the further failure of any single active component, with acceptable off-site consequences. The Oyster Creek emergency condenser and reactor water cleanup piping systems do not meet the above criteria. These systems and JCP&L proposed corrective actions are discussed below.

a. Emergency Condenser System

As discussed in Reference 4, postulated pipe breaks in the emergency condenser piping on the 75 foot elevation of the reactor building could result in damage to both isolation valves and/or electrical cables to the valves on the steam supply lines which are located outside containment. This could result in the failure of these valves to close on demand, even without an assumed single active failure. Accordingly, as discussed in Reference 4, JCP&L plans to install a leak detection system on these pipes. The leak detection system will detect small leaks (less than 1 gpm) before the break size becomes large enough to cause damage to the isolation valves or electrical cables. The system will be installed during the next scheduled refueling shutdown.

b. Reactor Water Cleanup System

The reactor water cleanup system contains one isolation valve inside and one isolation valve outside containment. As discussed in Reference 1, and the following sections of this report, postulated pipe breaks in the reactor water cleanup piping near the outside isolation valve on the 51 foot elevation of the reactor building could result in damage to the isolation valve and/or electrical cables to the valve. A further assumed single active failure of the isolation valve inside containment could result in the failure of both isolation valves to close on demand. Continued operation of the plant is justified because a) the extremely low probability that an HELB in the reactor water cleanup system would be in an orientation that would cause the worst case damage described above, and b) the extremely low probability of a non-mechanistic failure of the isolation valve inside containment in combination with the HELB.

In-service examinations of the emergency condenser and reactor water cleanup piping and other high pressure piping systems are in accordance with the ASME Boiler and Pressure Vessel Code, Section XI, 1974 through 1975 Addenda.

REQUEST

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.

RESPONSE

The effects of postulated pipe breaks in the reactor water cleanup (RWC) system on the 51 foot elevation of the reactor building are discussed in FDSAR Amendment No. 75 (Reference 5). The evaluation considered the effects of damage to cable trays 13A, 14A, 13, and 14. In summary, postulated pipe breaks in the RWC system were found to produce acceptable interactions with the core spray and auto depressurization systems.

The effect of damage to Cable Tray 13A on Oyster Creek safe shutdown systems has been re-evaluated by JCP&L. (Note: Oyster Creek safe shutdown systems are listed in Section 3.0 of Reference 1). A review of electrical circuits contained in Cable Tray 13A indicates that there are no circuits for safe shutdown systems routed in Cable Tray 13A, except for a control cable to one safety relief valve on the auto depressurization system. Since the auto depressurization system is not required to cope with a break in the reactor water cleanup system, this interaction is considered acceptable.

During this review it was noted that the power cables to the RWC isolation valves located outside containment are contained in Cable Trays 13A and 14A. As discussed in the above response, damage to these electrical cables in combination with an assumed single active failure to the isolation valve inside containment could result in failure of both isolation valves to close on demand. As stated previously, continued operation of the plant is justified on the basis of the extremely low probability that a break in the reactor water cleanup system would be in an orientation that would cause the worst case damage described above, along with the extremely low probability of a non-mechanistic failure of the isolation valve outside containment.

REQUEST

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.

RESPONSE

The floor of the cable spreading room (Elevation 36'-0") contains two 4-inch diameter drains. These drains have a total capacity of about 850 gpm (425 gpm per drain) at a water height that is below the elevation of any vital equipment or cables in the cable spreading room.

The largest potential flow rate into the cable spreading room is about 730 gpm which is the maximum discharge capacity of the fire water system. This is less than the total capacity of the drains. Furthermore, the drains in the cable spreading room discharge to the basement floor of the turbine building (Elevation 0'-0"). Thus, there is no way for water to back-up in the drains and flood the cable spreading room.

3. REFERENCES

1. NRC Letter, D. M. Crutchfield to I.R. Finfrock, Jr., (JCP&L) dated July 10, 1980.

2. FDSAR Amendment No. 50, Primary Containment Penetration Design.

3. FDSAR Amendment No. 11, Answers to 109 AEC Questions Regarding Additional Plant Information.

4. JCP&L Responses to Staff Positions on SEP Topic III - 5.B Pipe Breaks Outside Containment Oyster Creek Nuclear Plant, dated October 1, 1980.

5. FDSAR Amendment No. 75, Effects of Postulated Pipe Failures Outside Containment.

TABLE 1

COMPARISON OF OYSTER CREEK MAIN STEAM, EMERGENCY CONDENSER AND REACTOR WATER CLEANUP PIPING WITH REQUIREMENTS OF BTP MEB 3-1 SECTION B.1.b

REQUIREMENTS - BTP MEB 3-1 SECTION B.1.b	OYSTER CREEK PIPING
1. Piping should be designed to the stress limits of the ASME Code, Section III - Class II piping plus the following additional requirements: a. The maximum stress range, sum of Eq. (9) and Eq. (10) of NC-3652, ASME Code, Section III, considering normal and upset plant conditions and OBE, shall be less than 0.8 $(1.2S_h + S_A)$ b. The maximum stress, Eq. (9) of NC-3652, considering loads from postulated pipe break, shall be less than 1.8 S_h. Higher stresses are allowed provided: 1) A plastic hinge is not formed, 2) The operability of the isolation valve with the higher stresses is assured in accordance with SRP 3.9.3,	1. The Oyster Creek piping was designed to the general stress limits of the Code for Pressure Piping ASA B31.1 (1955), plus the following additional requirements for seismic loads: a. Dynamic analyses were required for piping 10" and larger (e.g., emergency condenser and main steam piping). b. Equivalent static analyses were permitted for piping less than 10" (e.g., reactor water clean-up piping). c. Pipe supports were located to limit support loads to less than 10,000 lb per support. The piping between the containment penetration and outside isolation valve has not been analyzed for postulated pipe breaks beyond the isolation valve.

TABLE 1 (continued)

COMPARISON OF OYSTER CREEK MAIN STEAM, EMERGENCY CONDENSER AND REACTOR WATER CLEANUP PIPING WITH REQUIREMENTS OF BTP MEB 3-1 SECTION B.1.b

REQUIREMENTS - BTP MEB 3-1 SECTION B.1.b	OYSTER CREEK PIPING
3) If the piping is constructed in accordance with Power Piping Code ANSI B31.1, then the piping shall be seamless with full RT of circumferential welds, or full RT of all longitudinal and circumferential welds.	The containment penetrations were designed to the following stress limits in accordance with Reference 2. a. Accident design condition-Include all live loads, dead loads, seismic loads, thermal growth, and pressure conditions resulting from LOCA inside drywell. 1) General membrane stress less than Sm above. 2) Local membrane stress less than 1.5 Sm 3) Bending stress less than 1.5 Sm b. Pipe rupture design condition-Include jet load resulting from rupture of a pipe. 1) Local membrane stress less than 1.5 Sm

TABLE 1 (continued)

COMPARISON OF OYSTER CREEK MAIN STEAM, EMERGENCY CONDENSER AND REACTOR WATER CLEANUP PIPING WITH REQUIREMENTS OF BTP MEB 3-1 SECTION B.1.b

REQUIREMENTS - BTP MEB 3-1 SECTION B.1.b	OYSTER CREEK PIPING
	2) Local membrane + secondary membrane + secondary bending less than 3 Sm
2. Welded attachments, for pipe supports or other purposes, shall be avoided except where detail stress analyses, or tests, are performed to demonstrate compliance with the stress requirements above.	2. Welded attachments on the main steam, emergency condenser, and reactor water cleanup lines are in accordance with Chapter 1 of Section 6 of ASA B31.1.
3. The length of pipe from the containment penetration to the pipe whip restraint outboard of the isolation valve should be reduced to the minimum length practical.	3. There are no pipe whip restraints outboard of the isolation valves on the main steam, emergency condenser and reactor water cleanup lines. Restraints are provided on the emergency condenser and main steam piping penetrations as discussed in Reference 2.
4. The design of pipe anchors or restraints should not require welding directly to the OD of the pipe except where such welds can be 100 percent volumentrically examined in service and detail stress analyses are performed to demonstrate compliance with the stress requirements above.	4. Welded attachments on the main steam, emergency condenser, and reactor water cleanup lines are in accordance with Chapter 1 of Section 6 of ASA B31.1.

TABLE 1 (continued)

COMPARISON OF OYSTER CREEK MAIN STEAM, EMERGENCY CONDENSER AND REACTOR WATER CLEANUP PIPING WITH REQUIREMENTS OF BTP MEB 3-1 SECTION B.1.b

REQUIREMENTS - BTP MEB 3-1 SECTION B. 1.b	OYSTER CREEK PIPING
5. Guard pipes should be constructed in accordance with ASME Code, Section III Subsection NE (Class MC Vessels) and the following requirements. a. The design pressure and temperature of the guard pipe shall not be less than the maximum operating pressure and temperature of the enclosed pipe under normal condition. b. Stresses due to containment design pressure and temperature and safe shutdown earthquake shall not exceed limits NE-3131 (c) of ASME Code, Section III. c. Guard pipes should be pressure tested at a pressure not less than the design pressure.	5. The guard pipes on the main steam, emergency condenser, and reactor water cleanup piping were designed to the same requirements as the drywell. (ASME Code, Section VIII, 1962 plus Code Cases 1272N5, 1271N, 1272N5). a. The design pressure and temperature for the guard pipes is greater than maximum operating temperature and pressure of enclosed pipe. b. Stress limits for containment penetrations are given in Item 2, above. c. The guard pipes were tested in accordance with N-711 and N-714 through N-716 of the ASME Code Section III (1965) at the design pressure at ambient temperature for the enclosed pipe.

TABLE 2

COMPARISON OF OYSTER CREEK EMERGENCY CONDENSER
AND REACTOR WATER CLEAN-UP PIPING
WITH REQUIREMENTS OF BTP ASB 3-1 SECTION B.2.c

REQUIREMENTS - BTP ASB 3-1 SECTION B.2.c	OYSTER CREEK PIPING
1. Piping should be be designed to the stress limits of BTP MEB 3-1, Section B.2.b (ASME Code, Section III - Class 2 piping) plus the following additional requirements: a. The maximum stress range, sum of Eq. (9) and Eq. (10) of NC-3652, ASME Code, Section III, considering normal and upset plant conditions and OBE, shall be less than $0.8\ (1.2S_h + S_A)$ b. The maximum stress, Eq. (9) of NC-3652, considering loads from postulated pipe break, shall be less than 1.8 S_h. Higher stresses are allowed provided: 1) A plastic hinge is not formed, 2) The operability of the isolation valve with the higher stresses is assured in accordance with SRP 3.9.3	1. The Oyster Creek piping was designed to the general stress limits of the Code for Pressure Piping ASA B31.1 (1955), plus the following additional requirements for seismic loads: a. Dynamic analyses were required for piping 10" and larger (e.g., emergency condenser piping). b. Equivalent static analyses were permitted for piping less than 10" (e.g., reactor water cleanup piping). c. Pipe supports were located to limit support loads to less than 10,100 lb per support. The piping between the containment penetration and outside isolation valve has not been analyzed for postulated pipe breaks beyond the isolation valve.

TABLE 2 (continued)

COMPARISON OF OYSTER CREEK EMERGENCY CONDENSER AND REACTOR WATER CLEAN-UP PIPING WITH REQUIREMENTS OF BTP ASB 3-1 SECTION B.2.c

REQUIREMENTS - BTP ASB 3-1 SECTION B.2.c	OYSTER CREEK PIPING
3) If the piping is constructed in accordance with Power Piping Code ANSI B31.1, then the piping shall be seamless with full RT of circumferential welds, or full RT of all longitudinal and circumferential welds.	The containment penetrations were designed to the following stress limits in accordance with Reference 2. a. Accident design condition- Includes all live loads, and thermal, growth and pressure contitions resulting from LOCA inside drywell. 1) General membrane stress less than Sm 2) Local membrane stress less than 1.5 Sm 3) Bending stress less than 1.5 Sm b. Pipe rupture design condition- Include jet load resulting from rupture of a pipe 1) Local membrane stress less than 1.5 Sm

TABLE 2 (continued)

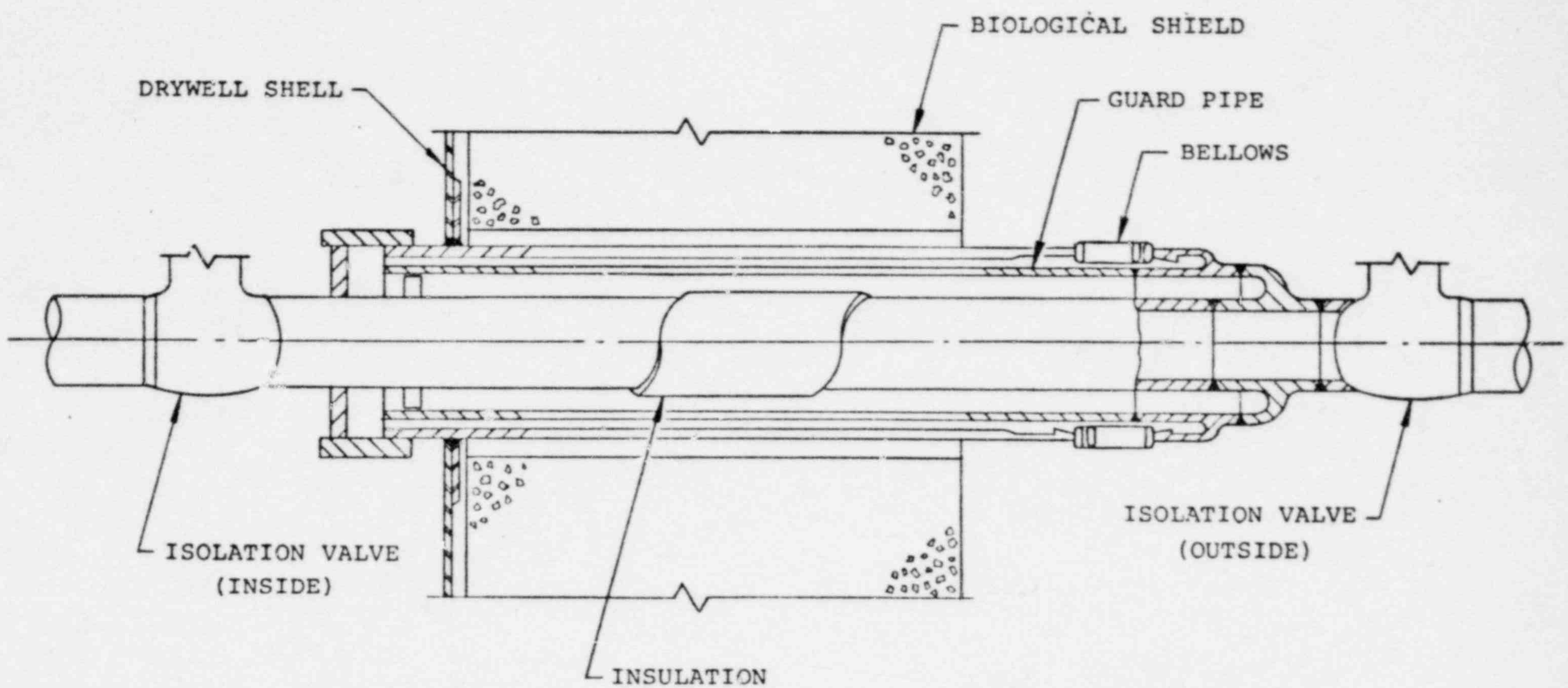
COMPARISON OF OYSTER CREEK EMERGENCY CONDENSER AND REACTOR WATER CLEAN-UP PIPING WITH REQUIREMENTS OF BTP ASB 3-1 SECTION B.2.c

REQUIREMENTS - BTP ASB 3-1 SECTION B.2.c	OYSTER CREEK PIPING
	2) Local membrane + secondary membrane + secondary bending less than 3 Sm
2. Piping should be provided with pipe whip restraints that are capable of resisting bending and torsional moments produced by postulated pipe breaks upstream or downstream of the isolation valves. The restraints should be located reasonably close to the isolation valves and should be designed to withstand the loads from a postulated pipe break so that neither isolation valve operability nor the leaktight integrity of the containment is impaired.	2. Pipe whip restraints outboard of the isolation valves are not provided on the emergency condenser and reactor water cleanup lines. Restraints are provided on the emergency condenser and reactor water cleanup containment penetrations in accordance with Reference 2 as follows: a. Emergency Condenser 1) Inside drywell-Restraint not needed. 2) Outside drywell-Restraint installed. b. Reactor Water Cleanup 1) Inside drywell-Restraint not needed.

TABLE 2 (continued)

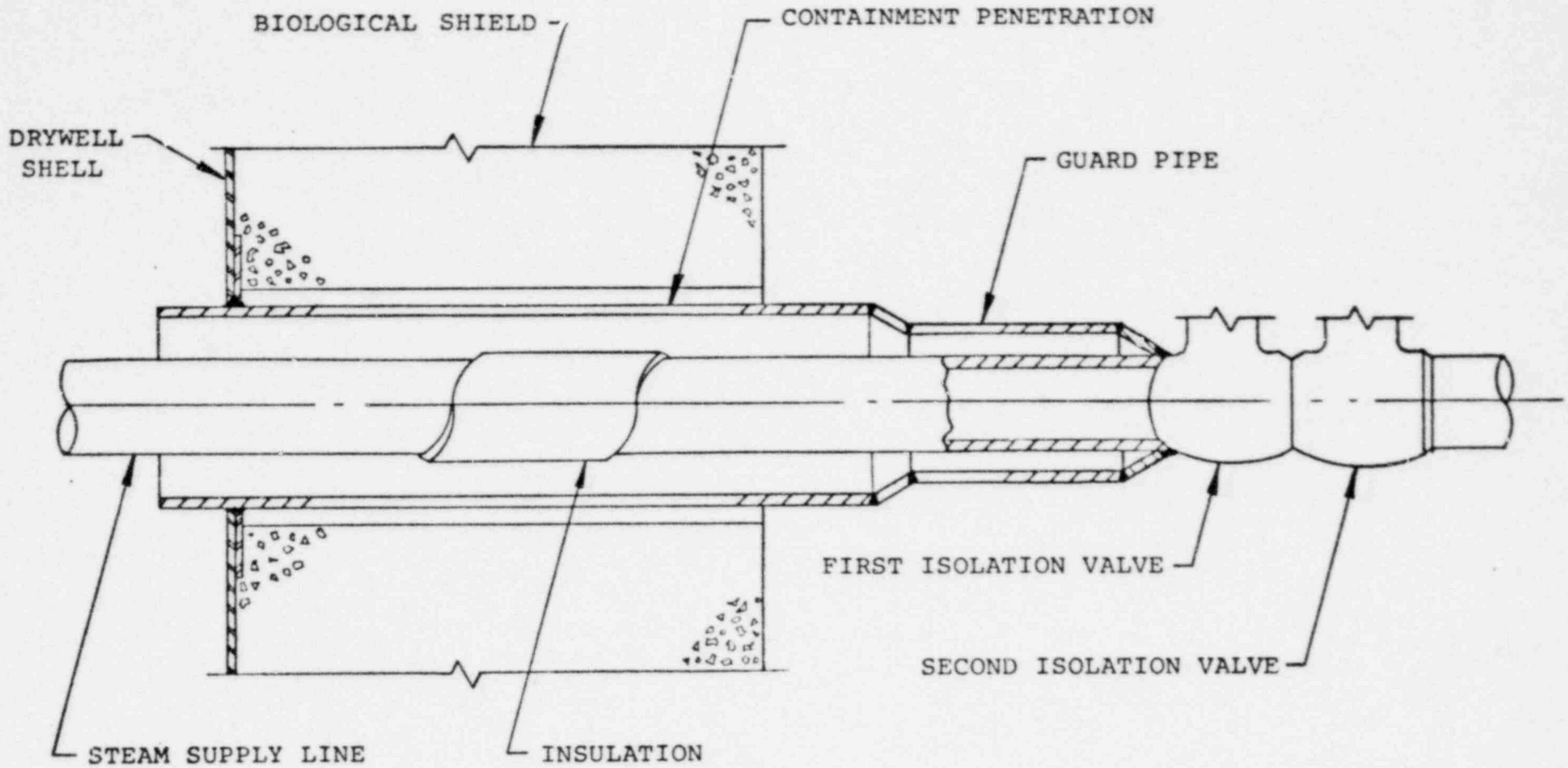
COMPARISON OF OYSTER CREEK EMERGENCY CONDENSER
AND REACTOR WATER CLEAN-UP PIPING
WITH REQUIREMENTS OF BTP ASB 3-1 SECTION B.2.c

REQUIREMENTS - BTP ASB 3-1 SECTION B.2.c	OYSTER CREEK PIPING
	2) Outside drywell-Restraint not needed.
3. Terminal ends should be considered to originate at a point adjacent to the required pipe whip restraints.	3. This requirement is not applicable for the emergency condenser and reactor water cleanup lines where there are no pipe whip restraints outboard of the isolation valve. Terminal ends are assumed to originate at the isolation valve for this case.
4. Piping classification as required by Regulatory Guide 1.26 should be maintained without change until beyond the restraint. If the restraint is located at the isolation valve, a classification change at the valve interface is acceptable.	4. This requirement is not applicable for the emergency condenser and reactor water cleanup lines where there are no pipe whip restraints outboard of the isolation valve.



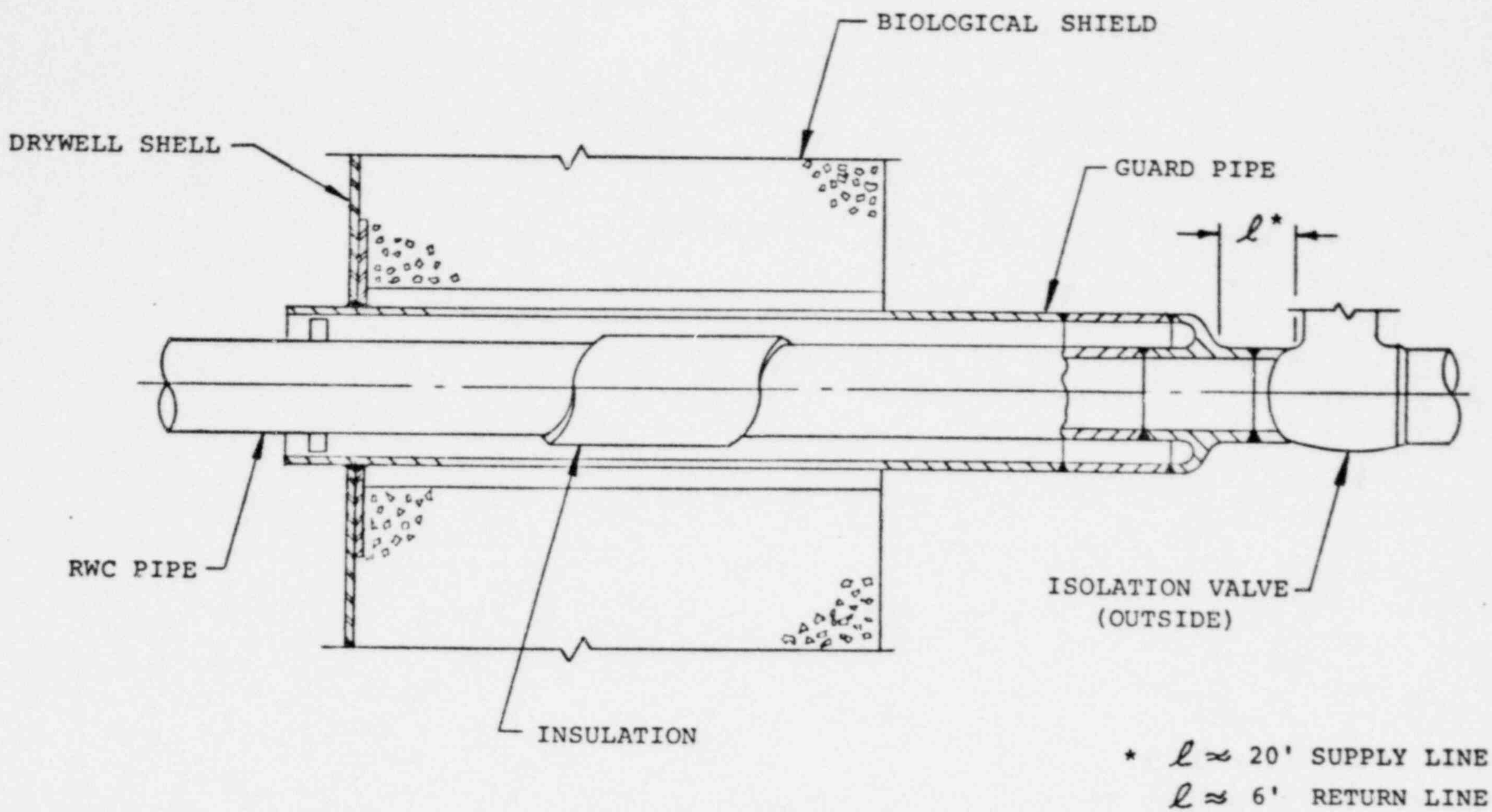
CONTAINMENT PENETRATION MAIN STEAM

FIGURE 1



CONTAINMENT PENETRATION EMERGENCY CONDENSER

FIGURE 2



CONTAINMENT PENETRATION (TYP)
REACTOR WATER CLEANUP

FIGURE 3