June 29, 1984

MEMORANDUM FOR: Commissioner Gilinsky

FROM: William J. Dircks
Executive Director for Operations

SUBJECT: SHOREHAM ECCS

Your June 28, 1984 memorandum asked if the ECCS pumps at Shoreham meet the regulations.

Enclosed are several staff evaluations which provide our basis for concluding that these pumps will satisfy the applicable General Design Criteria.

Enclosures 1 and 2 are excerpts from the fire protection evaluations contained in SSER 1 and 2. Note particularly Sections 9.5.3 and 9.5.6 on pages 9-9 and 9-11 of Enclosure 1 and the final paragraphs on page 9-1 of Enclosure 2. For clarity, elevation 8' is the bottom floor of the reactor building where these pumps are located and where the applicant's additional fire protection measures described in Enclosure 1 are located.

With regard to flooding of the ECCS pump level of the reactor building, we have determined that plant indications/alarms provide adequate information for plant operations to isolate any design basis pipe breaks in the reactor building. We conclude in Enclosure 3 that the pipe break protection in the reactor building meets GDC 3. Compliance with GDC 4 ensures that the equipment in the reactor building, including ECCS pumps, are not subject to common mode failure due to flooding. That is, no single pipe break in the reactor building can disable the ECCS functions. Therefore, GDC 35 is met for the worst case design basis internal flooding event in the reactor building. Enclosure 4 discusses another potential concern, namely flooding due to procedural errors during maintence. Enclosure 5, incorporates a BNL PRA type evaluation of the potential for flooding due to maintenance procedure errors, a beyond-design-basis event, and shows that the probability of such an event is acceptably low.

In addition to the design basis calculations, and evaluations of maintenance induced flooding, the staff has performed inspections and calculations to determine available time for operators to secure and mitigate the flooding from postulated breaks in the various piping systems inside the reactor building. Protection of vital instrumentation, junction boxes, and components of safety grade systems, were examined as to their susceptibility for flooding or spray. In most cases, several hours were available for the operators to identify and secure the flooding. In order to bound the problem, two

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coincident errors were assured in the limiting system to obtain a conservatively large flooding rate; the results of this scenario indicate in excess of two hours for the operators to take action. In summary, we have considered the man-machine interface as well as the design-basis and PRA aspects and find both are acceptably resolved.

(Signed) William J. Dircks

William J. Dircks
Executive Director for Operations

Enclosures:
1. Pages 9-1 - 9-12 of Shoreham SSER-1
2. Page 9-1 of Shoreham SSER-2
3. Pages 3-2 - 3-4 of Shoreham SSER-4
4. Memo (Mattson to Denton) dtd 12/29/83
5. Memo (Eisenhut to Starostecki) dtd 5/4/84

cc: Chairman Palladino
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Commissioner Asselstine
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*See previous concurrences

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William J. Dircks
Executive Director for Operations

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cc: Chariman Palladino
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	FINAL REPLY	s/Dircks 6/29/84	☐ CHAIRMAN
	FILE LOCATION		☒ EXECUTIVE DIRECTOR
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Question re ECCS pumps at Shoreham

SPECIAL INSTRUCTIONS OR REMARKS

PRIORITY

ASSIGNED TO	DATE	INFORMATION ROUTING
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NRC FORM 232
(6-80)

EXECUTIVE DIRECTOR FOR OPERATIONS
PRINCIPAL CORRESPONDENCE CONTROL

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

OFFICE OF THE COMMISSIONER

June 28, 1984

MEMORANDUM FOR: WILLIAM J. DIRCKS
EXECUTIVE DIRECTOR FOR OPERATIONS

SUBJECT: SHOREHAM ECCS

Bill,

It has come to my attention that essentially all of the ECCS pumps at Shoreham are located next to one another on the bottom floor of the reactor building. Since there are no walls or dams to isolate the pumps, I am curious as to how the plant meets the Appendix R fire barrier requirements or General Design Criterion 35, 34, and 35 with respect to flooding induced by a single pipe failure. Has the staff made a finding that the ECCS pumps meet the regulations? I would appreciate a response by tomorrow evening if possible.

Victor

Victor Gilinsky

cc: Chairman Palladino
Commissioner Roberts
Commissioner Asselstine
Commissioner Bernthal
SECY

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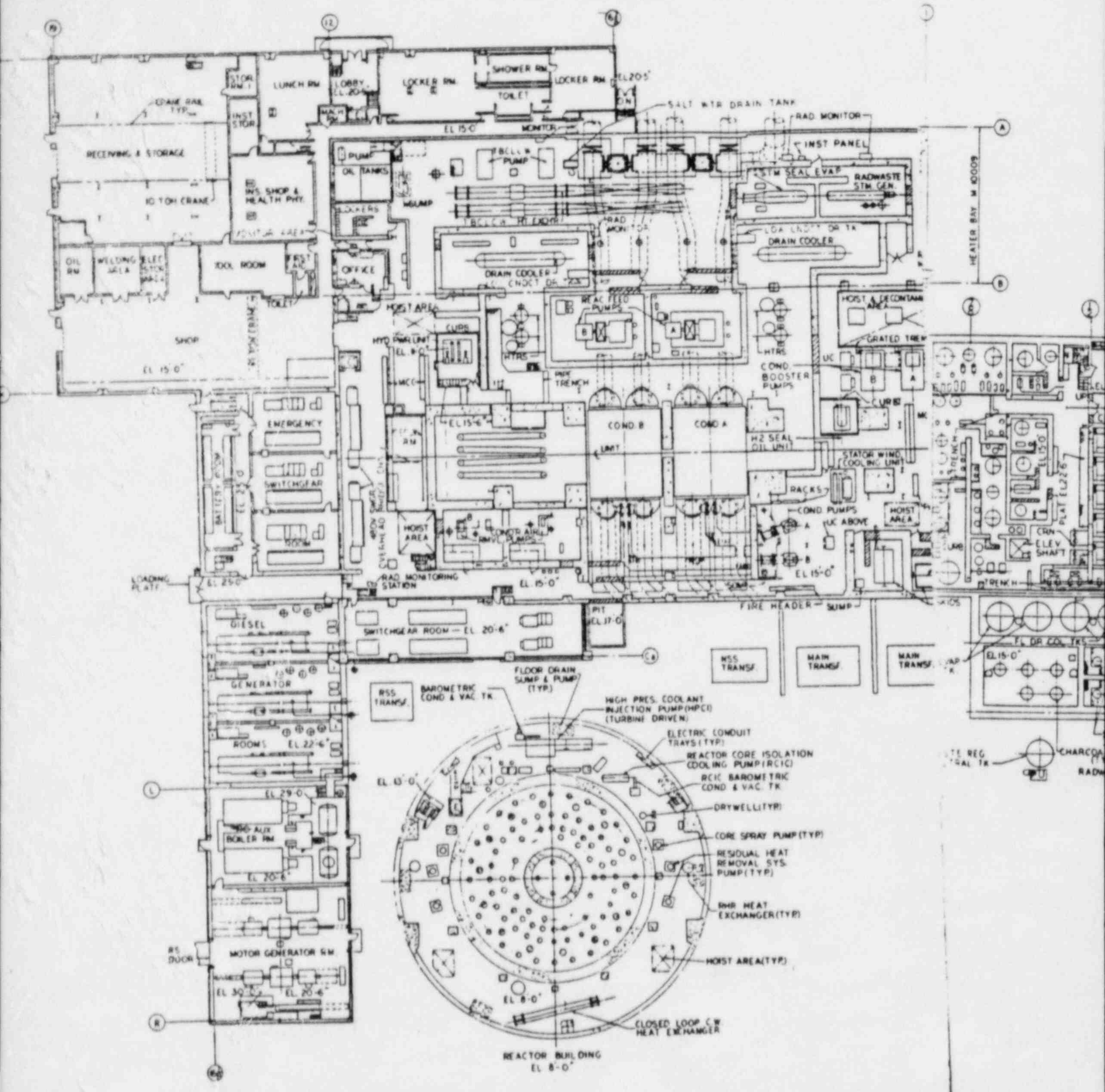

LUNCH RM
LOBBY
LOCKER RM
SHOWER RM
TOILET
LOCKER RM
RECEIVING & STORAGE
10 TON CRANE
INST STOR
TOOL ROOM
OIL RM
WELDING AREA
OFFICE
SHOP
EL 15'-0"
SALT WTR DRAIN TANK
RAD MONITOR
INST PANEL
STM SEAL EVAP
RADWASTE STM GEN
DRAIN COOLER
HEATER BAY
REAC FEED PUMPS
HTRS
COND BOOSTER PUMPS
HOIST & DECONTAM AREA
GRATED TRENCH
PIPE TRENCH
COND B
COND A
H2 SEAL OIL UNIT
STATOR WIND COOLING UNIT
RACKS
COND PUMPS
UC ABOVE
HOIST AREA
EMERGENCY
SWITCHGEAR
ROOM
BATTERY RM
RAD MONITORING STATION
LOADING PLATF
EL 25'-0"
DIESEL
GENERATOR
ROOMS
EL 22'-6"
SWITCHGEAR ROOM - EL 20'-6"
PIT EL 17'-0"
FIRE HEADER
SUMP
RSS TRANSF
MAIN TRANSF
MAIN TRANSF
ELEV SHAFT
EL 15'-0"
BAROMETRIC COND & VAC TK
FLOOR DRAIN SUMP & PUMP (TYP)
HIGH PRES. COOLANT INJECTION PUMP (HPCI) (TURBINE DRIVEN)
ELECTRIC CONDUIT TRAYS (TYP)
REACTOR CORE ISOLATION COOLING PUMP (RCIC)
RCIC BAROMETRIC COND & VAC TK
DRYWELL (TYP)
CORE SPRAY PUMP (TYP)
RESIDUAL HEAT REMOVAL SYS. PUMP (TYP)
RHR HEAT EXCHANGER (TYP)
HOIST AREA (TYP)
CLOSED LOOP C.W. HEAT EXCHANGER
REACTOR BUILDING EL 8'-0"
EL 8'-0"
EL 13'-0"
EL 29'-0"
AUX BOILER RM
EL 20'-6"
MOTOR GENERATOR RM
EL 30'-0"
RS DOOR

NUREG-0420
Supplement No. 1

Safety Evaluation Report
related to the operation of Shoreham Nuclear Power Station, Unit No. 1

Docket No. 50-322

Long Island Lighting Company

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

September 1981

9 AUXILIARY SYSTEMS

9.1.4 Fuel Handing System

In the Safety Evaluation Report, we concluded that the fuel handling system met the intent of Branch Technical Position ASB 9-1, "Overhead Handling Systems for Nuclear Power Plants." We also concluded that it will perform its safety function and is, therefore, acceptable.

In letters dated December 22 1980 and February 3, 1981, the applicant was requested to establish the extent to which their heavy load handling operations satisfy the guidelines of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." This NUREG addressed unresolved safety issue A-36. Further, the applicant was requested to identify the changes and modifications which would be required to fully satisfy these guidelines. The applicant will submit the results of their review against NUREG-0612 guidelines at a later date. Since this effort will extend over some period of time, certain measures that could be readily implemented such as identifying safe load paths, the development of procedures, operation training and crane inspections, testing and maintenance, were separately identified in Enclosure 2 to the December 22, 1980 letter. We require the applicant to implement these interim actions prior to the final implementation of the NUREG-0612 guidelines and prior to reccipt of their operating license.

Based on our review of the Final Safety Analysis Report and the applicant's July 31, 1981 commitment to the interim position, we continue to believe that the fuel handling system meets the intent of Branch Technical Position ASB 9-1 and is in conformance with the requirements of General Design Criteria 2 and 61 relating to its protection against natural phenomena and safe fuel handling and the guidelines of Regulatory Guides 1.12 and 1.29 with respect to overhead crane interlocks and maintaining plant safety in a seismic event. The fuel handling system is, therefore, acceptable. We further conclude that implementation of the interim actions of NUREG-0612 prior to final implementation of NUREG-0612 guidelines and prior to receipt of the operating license provides reasonable assurance of safe handling of heavy loads until NUREG-0612 can be fully implemented and is, therefore, acceptable.

9.5 Fire Protection System

9.5.1 Introduction

We have reviewed the Shoreham fire protection program reevaluation and fire hazards analysis submitted by the applicant by letter dated June 1977. The Shoreham reevaluation was in response to our request to review their fire protection program against the guidelines of Appendix A to Branch Technical Postion (BTP) APCSB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants." As part of our review, we visited the plant site to examine the relationship of safety-related components, systems, and structures in specific plant areas to both combustible materials and to fire detection and suppression systems. The overall objective of our review was to ensure that, in the event of a fire, Shoreham personnel and the plant equipment would be adequate to safely shutdown the reactor, to maintain the plant in a safe shutdown conditon, and to minimize the release of radioactivity to the environment.

Our review included an evaluation of the automatic and manually operated water and gas fire suppression systems, the fire detection systems, fire barriers, fire doors and dampers, fire protection administrative controls, and the fire brigade size and training.

On October 27, 1980 the Commission approved for publication in the Federal Register a new rule 10 CFR 50.48 and its Appendix R to 10 CFR Part 50, delineating certain fire protection provisions for nuclear power plants licensed to operate prior to January 1, 1979. Although this fire protection rule does not apply to Shoreham, we used the contents of this rule also in the evaluation of the fire protection program.

The applicant has been informed that all fire protection modifications have to be implemented prior to fuel load. Portions of our evaluation are based upon verbal commitments by the applicant. Our final evaluation is dependent upon adequate documentation of these commitments.

9.5.2 Fire Protection Systems Description and Evaluation

9.5.2.1 Water Supply Systems

The fire water supply system consists of two fire pumps separately connected to a 12-inch cement lined cast iron underground fire water loop. The fire pumps are rated at 2,500 gpm at 125 psig head. One of the fire pumps is electric deiven and is fed from two sources of offsite power. The other fire pump is deiven by a diesel engine which has its own batteries and battery chargers for starting power. The fire pump installation conforms to the guidelines of NFPA 20, "Standard for the Installation of Centrifugal Fire Pumps." The diesel driven and motor driven fire pumps and their associated controls are separated by a 3-hour fire rated wall equipped with 3-hour fire rated doors.

Two 350,000-gallon water storage tanks provide water for fire protection. Three hundred thousand gallons of the 350,000 gallons of water in each tank are reserved for fire protection by locating connections to the tanks for other services at the 300,000-gallon level. The two tanks will be filled automatically by the station well water pumps which can refill one tank within 8 hours. Each tank is heated to maintain the temperature above 42°F by an electric immersion tank heater. Each fire pump takes suction from a separate tank. A valved, normally open cross-connection between the two suction lines is provided within the fire pump house. Each tank is equipped with instrumentation to sourd an alarm in the main control room when a tank reaches the 300,000-gallon mark. If a leak is detected in one tank, the piping can be manually aligned to isolate the leaking tank and have both fire pumps take suction from the other tank.

A 30-gpm pressure maintenance pump (jockey pump) maintains the system pressure. The fire pumps start automatically on low header pressure. If the fire water supply system pressure falls to 100 psi, the electric driven fire pump starts automatically. As the pressure falls to 85 psi, the diesel-driven fire pump starts automatically. The fire pumps can also be started manually from the control room and at the pumps. Separate alarms are provided in the control room to monitor pump operation, diesel-driven fire pump fuel oil day tank level, electric motor-driven fire pump breaker tripped, and diesel-driven fire pump relief valve high flow.

The water flow requirement for the fire suppression system requiring the greatest water demand for areas containing or exposing safety-related equipment is 500 gpm and, coupled with 750 gpm for hose streams, totals a water demand of 1250 gpm. Since the system can deliver 2500 gpm at rated pressure with one pump out of service, the fire water supply system is adequate. Based on our review, we conclude that the fire water supply system meets the guidelines of Section C.2 of Appendix A to BTP ASB 9.5-1 and, therefore, is acceptable.

9.5.2.2 Sprinkler and Standpipe Systems

The automatic and manual sprinkler (Spray) systems and the manual hose station standpipe system are connected to the outside fire protection underground main as follows:

1. Turbine Building - independent connections to the interior fire header.

2. Automatic systems in the reactor building and radwaste building connected into hose station riser pipes - one connection to underground.

3. Control Building - one connection to underground.

All control and isolation valves for the sprinkler and standpipe systems are electrically supervised. All other major valves are locked open. Also, actuation of any water fire-suppression system will cause a fire pump to start on a low head pressure signal.

The automatic sprinkler systems, e.g., wet pipe sprinkler system, preaction sprinkler systems, and deluge spray systems, are designed to the requirements of National Fire Protection Association (NFPA) Standard No. 13, "Standard for Installation of Sprinkler Systems," and NFPA Standard No. 15, "Standard for Water Spray Fixed Systems."

The areas that are equipped with water suppression systems include the following:

1. Personnel tunnel

2. Office and service building, receiving and storage, print, record and file, oil, and store rooms.

3. Warehouse

4. Auxiliary Boiler Room

5. Diesel engine driven fire pump fuel tank area

6. Low level storage room and boiler area in the radwaste building

7. Portions of the turbine building

Preaction sprinkler systems are provided for the following areas:

1. Hydrogen seal oil unit

2. Lube oil room

Automatic deluge systems are provided for the following areas:

1. Main transformers

2. Normal station transformer

3. Reserve station transformer

Manually actuated deluge systems with open nozzles and separate detection systems are provided for the following areas:

1. Charcoal filter beds in each of the two reactor building standby ventilation filter trains

2. HPCI and RCIC turbine lube oil area (will be modified to be automatic)

3. Ventilation exhaust charcoal filter train radwaste building

4. Turbine driven reactor feed pumps

5. Primary containment purge filter train - reactor building

We were concerned that a rapidly developing fire at the HPCI turbine-driven pump would affect the adjacent RCIC pump and would threaten vertical safety-related cable trays in the area before the manual suppression system would be actuated. At our request, by letter dated July 10, 1981, the applicant agreed to modify the existing manually actuated deluge system for the HPCI/RCIC equipment area to an automatic pre-action actuation system.

Manual hose stations are located throughout the plant to ensure that an effective hose stream can be directed to any safety related area in the plant. The standpipe systems are consistent with the requirements of NFPA 14, "Standpipe and Hose Systems for Sizing, Spacing, and Pipe Support Requirements."

Based on our review, we conclude that the water suppression systems meet the guidelines of Appendix A to BTP ASB 9.5-1 and are, therefore, acceptable.

9.5.2.3 Gas Fire Suppression Systems

Total flooding low pressure carbon dioxide suppression systems are provided for the following areas:

1. Diesel generator rooms

2. MG room

3. Battery room

4. Emergency switchgear room

5. Relay room (cable spreading room)

6. Normal switchgear room

7. Cable tunnel

8. Turbine generator bearings

CO_2 hose reel stations are located at the entrances to the main control room and at various locations throughout the turbine and control building.

Carbon dioxide protected rooms and areas are provided with local warning alarms, delayed operation,a nd lockouts for personnel protection.

The low pressure carbon dioxide CO_2 system consists of fire detection, storage tank distribution piping and valves, discharge devices, and associated instrumentation control room. A system of rate compensated thermal detectors is provided for automatic actuation of the carbon dioxide extinguishing systems. A time delay of sufficient time to enable personnel to leave the area is provided for each system. Activation of the system may also be accomplished manually at local points.

At our request the applicant verbally agreed to provide a 30 minute soak time of the CO_2 system for the relay room (cable spreading room) to insure extinguishment of a postulated fire. The present system is designed for a 50% concentration for 20 minutes.

We have reviewed the design criteria and bases for the CO_2 fire suppression systems. We conclude that these systems are in accordance with the applicable portions of NFPA 12, satisfy the provisions of Appendix A to BTP ASB 9.5-1 and are, therefore, acceptable.

9.5.2.4 Fire Detection Systems

The fire alarm system consists of both high and low voltage ionization detectors, low voltage duct detectors, heat and photoelectric detectors and various control and power supply panels which feed the data to a control room security console. The fire alarm system through the security console gives both audible and visual alarms in the plant control room. Standby power is provided by either an emergency AC bus or the battery system for the security system. Fire detection systems will be installed in all areas.

The fire detection systems will be installed according to HFPA 72D, "Standard for the Installation, Maintenance, and Use of Proprietary Protective Signaling Systems."

The control room security console provides audible and visual indication of an alarm or trouble condition in the fire alarm system. The console supplies an output to a common audible alarm in the main control room upon indication of a fire detection or system trouble condition. The security console also provides visual output of alarm conditions via a CRT screen in the control room and one of two automatic line printers. All alarm signals are fed to a data processor computer at the control room security console for information, storage, and indexing. Instant recall and line printer printout of previous trouble or alarm signals are possible.

We were concerned that an open or break in the non-Class A circuit of the fire alarm system from the relay room to the control room would eliminate all fire alarm signals. At our request, by letter dated July 10, 1981, the applicant provided further description of circuits from the interface panel in the relay room to the CRT console and printer in the control room. The interface panel in the relay room is connected by a four-conductor cable to the computer for the fire detection and station security system console, located in the control room. A second parallel four-conductor cable is provided for this interface wiring to provide a second path should a conductor break. The alarms appear on the CRT in this console and the printer associated with it. Simultaneously, they also appear on the CRT and printer located in the Security Building. The entire wiring for the detection system is supervised and any broken or shorted wire will alarm at both locations.

Based on the applicant's description of the redundant circuits between the relay room and control room, we conclude that the fire alarm system meets the guidelines of Appendix A to BTP 9.5-1 and is, therefore, acceptable.

9.5.3 Other Items Related to Fire Protection Programs

9.5.3.1 Fire Barriers and Fire Barrier Penetrations

Walls that separate safety-related buildings are three-hour fire rated. The floor/ceiling assemblies separating areas in buildings containing safe shutdown systems are also 3-hour fire rated barriers. Fire rated barriers are constructed of reinforced concrete or masonry block construction. Concrete fire barriers have been evaluated by comparison with designs which have been tested and rated in accordance with NFPA-251 and ASTM E-119. For fire areas not having a 3-hour fire rated assembly, we evaluated each individually with respect to its fuel load, fire suppression and detection systems, and proximity to safe shutdown equipment and conclude that the fire rated assemblies provded are adequate for the areas affected, meet the guidelines in Section D.i.j of Appendix A to BTP ASB 9.5-1 and are, therefore, acceptable.

The design of penetration seals used has not been documented. The applicant has verbally agreed to provide specific 3-hour U.L. designs for all fire penetration seals used in the penetration cable trays, conduits, and piping which pass the penetration qualification tests including the time-temperature exposure fire curve specified by ASTM E-119, "Fire Test of Building Construction and Materials." We have concluded that the fire seals meet the guidelines of Appendix A to BTP ASB 9.5-1 and are, therefore, acceptable.

9.5.3.2 Fire Doors and Dampers

The applicant has indicated that the following areas contain 3-hour fire rated dampers where ventilation ducts penetrate the 3-hour fire wall:

1. Relay Room

2. Control Room

3. Computer Room

4. Emergency Switchgear Rooms

5. Battery Rooms

In ducts penetrating the fire barrier walls surrounding the safety-related equipment, a fire damper of 1½-hour rating is used. Some areas also contain motorized 1½-hour fire dampers in which the motorized assembly, including cables, are not U.L. listed. We are concerned that the unlisted assemblies will prevent the fire dampers from performing its function. We require that all such operators be replaced with approved listed operators or a surveillance program be developed and included in the plant Technical Specifications to assure an adequate level of reliability.

9.5.4 Emergency Lighting

Eight-hour battery pack emergency lights are required for areas of the plant necessary for safe shutdown. The applicant will install self-contained eight-hour battery pack emergency lighting in all areas of the plant which could be manned to bring the plant to a safe cold shutdown and in access and egress routes to and from all fire areas.

We conclude that the emergency lighting meets the requirements of Appendix to BTP ASB 9.5-1, and, also, the provisions of Section III.J of Appendix R to 10 CFR Part 50 and is, therefore, acceptable.

9.5.5 Fire Protection for Specific Areas

9.5.5.1 Control Room

The control room complex is separated from all other areas of the plant by 3-hour fire rated walls, ceiling/floors assemblies, floors and doors. All ventilation ducts penetrating these barriers have 3-hour fire rated dampers. The control room complex peripheral rooms, except the visitors gallery which has bullet-resistant noncombustible materials, are constructed to provide a minimum fire rating of 1 hour. The ventilation openings in the peripheral rooms are protected with 1½-hour rated fire dampers. At our request the applicant has agreed to install additional smoke detectors in these rooms which will alarm and annunciate in the control room.

All cabinets, consoles, and the ventilation exhaust system within the control room have ionization fire detectors installed. The main control room ventilation system can be remote manual isolated from the main control room as it has capability of being used as a smoke removal system.

Manual fire fighting is provided through the use of portable extinguishers and CO_2 hose reels (supplied from the station hose pressure CO_2 storage tank) which are located outside the main control panel at the access door. At our request the applicant has agreed to provide and increase hose length of 100 feet for the 1½-inch water hose stations also located just outside the control room for additional protection inside the control room. This increased hose length will be provided for FHR Nos. 1 and 71.

The suspended ceiling is of the aluminum egg crate type design.

The applicant has agreed to install the emergency shutdown panel so that alternate shutdown capability exists independent of the control room.

Based on our review, we conclude that the control room fire protection meets the guidelines of Appendix A to BTP ASB 9.5-1 and is, therefore, acceptable.

9.5.5.2 Cable Spreading Room (Relay Room)

The single cable spreading room is separated from the balance of the plant by 3-hour fire-rated walls and floor/ceiling assemblies. Three-hour fire rated fire dampers are provided for all ventilation ducts that pierce the walls. At our request, the applicant has verbally agreed to upgrade to 2 hours a shaft in the southeast corner of the room, elevation 44, as well as provide 3-hour fire damps for any ducting penetrating this shaft. Exits are provided at each end of the room.

Automatic fire detection by Class 1E seismic Category I heat detectors will actuate a total flooding CO_2 suppression system, isolate ventilation, initiate local predischarge warning, and annunciate in the main control room. At our request the applicant verbally agreed to increase the 50% concentration to 30 minutes soak time. Area, duct, and panel-mounted smoke detectors are also provided for the room.

A CO_2 hose reel is located at the south end of the room as backup in addition to portable fire extinguishers. Standpipe water hose stations are provided on the outside of the main exits from the room.

We were initially concerned that a fire would affect redundant shutdown systems located in the cable spreading room. However, the applicant has installed an alternate shutdown capability independent of the cable spreading room (refer to Section 7.4.3 of this report). The fire protection for the cable spreading room meets the guidelines of Appendix A to BTP ASB 9.5-1 and is, therefore, acceptable.

9.5.5.3 Containment and Reactor Building

The primary containment will be inerted during normal operation hence eliminating any fire hazard associated with the lubricating oil of the recirculation pumps.

The containment and reactor building fire protection features include hose stations, fire detectors, fire extinguishers, automatic sprinklers, manual deluge and fire control barriers. Fire detectors are distributed throughout the areas with alarm and annunication in the control room.

In the reactor building at elevation 8', we were concerned that a fire at the HPCI turbine driven pump could affect the adjacent RCIC turbine pump as well as vertical safety related cable trays running up the walls in the area. At our request, the applicant by letter dated May 21, 1981, agreed to provide a 9-foot high 3-hour fire rated barrier between the vertical cable trays and the adjacent RCIC pump. Also, by letter dated July 10, 1981, the applicant agreed to modify the existing manually actuated deluge system for the HPCI/RCIC equipment area to an automatic pre-action actuation system.

We were concerned that if the preset single feed to the reactor building should fail, both the primary and secondary fire protection would be lost. At our request, the applicant agreed to provide a secondary feed from the underground to the reactor building, as well as necessary valves such that primary or secondary water fire protection will always be provided. The fire protection for the containment and reactor buildings meets the guidelines of Appendix A to BTP ASB 9.5-1 and is, therefore, acceptable.

9.5.5.4 Emergency Diesel Generator Rooms

Each of the emergency diesel generators is in its own protected room separated by walls, floor, ceiling/floor assembly and doors having a minimum fire rating of 2 hours. Automatic fire detection by Class 1F, seismic Category I heat detectors actuate a total flooding CO_2 suppression system, isolate ventilation (with the exception of the diesel air intakes), shut down the diesel fuel oil transfer pumps for the diesel, actuate the local predischarge warning and annunciate in the main control room. Area smoke detectors are also installed for these rooms. Backup fire protection is provided by protable extinguishers plus manual hose stations located in the room. Smoke purging is provided for through the normal ventilation system.

The diesel fuel oil storage tanks are buried and located at a distance of more than 50 feet from the diesels.

Based on our evaluation, we conclude that the fire protection for the diesel generator rooms meets the guidelines of Appendix A to BTP ASB 9.5-1 and is, therefore, acceptable.

9.5.5.5 Other Plant Areas

The applicant's Fire Hazards Analysis addressed other plant areas not specifically discussed in this report. The applicant has committed to install additional detectors, portable extinguishers, and fire barriers prior to fuel load. We find the fire protection for these areas, with the commitment made by the applicant, to be in accordance with the guidelines of Appendix A to BTP ASB 9.5-1 and is, therefore, acceptable.

9.5.6 Fire Protection of Safe Shutdown Capability

The applicant's post-fire safe shutdown analysis of the fire protection of safe shutdown is presented in three letters.

By letter dated May 21, 1981, the applicant provided a comparison of the plant design with the requirements of Appendix R. The applicant also provided a separation analysis of cables within the reactor building by letter dated February 10, 1981 and analysis of shutdown circuits outside the reactor building by letter dated July 10, 1981.

The applicant's post-fire safe shutdown analysis demonstrated that systems needed for hot shutdown and cold shutdown are redundant and that one of the redundant systems needed for safe shutdown would be free of fire damage, by providing separation, fire barriers, and/or alternative shutdown capability.

The safe shutdown analysis considered components, cabling, and support equipment for systems needed to shut down. Thus, in the event of a fire, at least one train of systems free of fire damage would be available to achieve and maintain hot shutdown or to proceed to cold shutdown. For hot shutdown, at least one of the following shutdown systems would be available: (1) the Reactor Core Isolation Cooling System, (2) the High Pressure Coolant Injection System, and (3) a combination of the pressure relief system, the core spray system and residual heat removal (RHR) system. For cold shutdown, an appropriate portion of the RHR system would be available.

For equipment located in the primary containment, no fire protection features are provided because the containment atmosphere will be inert.

For equipment located in the reactor building (secondary containment), the applicant provided a cable separation analysis which divided the reactor building into overlapping 45 degree segments. The applicant assumed that all components, the cables and raceways, in a given segment were lost due to a fire; yet demonstrated the capability to shut down still existed. We have reviewed the cable separation analysis and conclude that it is an acceptable method of demonstrating that adequate separation exists between the redundant trains. Additionally, the applicant has committed (by July 10, 1981 letter) to verify that the "as-built" design has a minimum 20 ft separation between redundant safety-related components.

The secondary containment is a cylindrical structure with a 135-foot outside diameter and 240 ft high with 2-foot concrete walls. There are six complete elevations with each elevation containing large open penetrations. The area between the primary and secondary containment is one fire area.

Throughout the reactor building both smoke and temperature detectors are installed with alarm and annunciation in the control room. All cable trays have solid bottoms with covers or ladder type with solid covers attached to both sides. All vertical trays incorporate fire stops within the tray and external to all trays where they penetrate floor levels. Fire stops are provided at the midpoints when the elevation is more than 25 ft.

The two main vertical safety-related cable risers are located at 138° and 223° azimuth, extend from elevation 8' to elevation 40' and are separated by 85 ft. The applicant's analysis demonstrated that a 45° segment in which a fire caused the disability of all cables and raceways in that segment, a separation distance of 20 ft on the inside of secondary containment existed and 35 ft existed on the outside. The applicant then rotated this segment 22.5° for additional verification and overlapping.

The applicant provided fire detection, alarm, annunciation, water spray systems for the RBCV's charcoal filters, hose stations, automatic pre-action sprinkler systems, fire barriers, two fire main feeds and portable equipment for secondary containment. Due to the preceding separation distances and protection provided, an automatic suppression system is not needed for protection against a transient exposure fire.

For equipment in areas outside the reactor building, the applicant has identified seven areas which contain cable for redundant shutdown equipment: the relay room, the control room, the diesel-generator rooms, the emergency switchgear room, the fuel oil pumphouse rooms, the screenwell, and the HVAC room.

In the diesel-generator rooms, the emergency switchgear room, the fuel oil pumphouse rooms, and the screenwell, redundant equipment is separated by a 3-hour fire-rated barrier. Cabling to this equipment is contained in underground ducts. In the event that fire disables redundant equipment in the HVAC room, control room, or relay room, a remote shutdown panel is provided in the reactor building (refer to section 7.4.3 of this report).

Sections 7.4.1.4, 7.5.1.4, and 7.5.1.5 of the FSAR describe the remote shutdown panel's design and capability. By letter dated May 21, 1981, the applicant addressed Section III.L of Appendix R. The design objective of the remote shutdown panel is to achieve and maintain cold shutdown in event of a fire disabling the relay room or the control room. The reactor core isolation cooling (RCIC) system, safety/relief valves and one division of the residual heat removal (RHR) system can be controlled from the remote shutdown panel to achieve cold shutdown.

The design of the remote shutdown panel complies with the performance goals outlined in Section III.L. Reactivity control will be accomplished by a manual scram before the operator leaves the control room. The RCIC system will provide reactor coolant makeup and the RHR system and the safety relief valves will be used for reactor heat removal. Reactor water level, reactor pressure, suppression pool water level and temperature, and drywell pressure and temperature are among instrumentation available at the remote shutdown panel to provide direct reading of process variables. The remote shutdown panel will also include instrumentation and control of support functions needed for the shutdown equipment. Procedures for use of the remote shutdown panel include sequencing of equipment and operator actions.

Based on the above, we conclude that the fire protection of safe shutdown capability meets the guidelines of Appendix A to BTP 9.5-1 and is, therefore, acceptable.

9.5.7 Administrative Controls and Fire Brigade

The administrative controls for fire protection consist of the fire protection organization, the fire brigade training, the controls over combustibles and ignition source, the prefire plans and procedures for fighting fires and quality assurance. The fire brigade will be composed of five members per shift. To have proper coverage during all phases of operation, members of each shift crew will be trained in fire protection in accordance with our guidance including Regulatory Guide 1.101, "Emergency Planning for Nuclear

Power Plants." The applicant has agreed to implement the fire protection program contained in the staff supplemental guidance "Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls and Quality Assurance," dated August 29, 1977, including (1) fire brigade training, (2) control of combustibles (3) control of ignition sources, (4) fire-fighting procedures, and (5) quality assurance.

The applicant will implement the plant administrative controls and procedures before fuel loading.

We conclude that, with these commitments, the size of the fire brigade, the necessary equipment, and the adequacy of the training, training will conform to the recommendations of the National Fire Protection Association, to Appendix A to BTP ASB 9.5-1, and to our supplemental staff guidelines and are, therefore, acceptable.

9.5.8 Technical Specification

The applicant has committed to follow our Standard Technical Specifications. We find this acceptable.

9.5.9 Appendix R Statement

On October 27, 1980, the Commission approved for publication in the Federal Register a new rule §50.48 and Appendix R to 10 CFR Part 50, delineating certain fire protection provisions for nuclear power plants licensed to operate prior to January 1, 1979. Although this fire protection rule does not apply to Shoreham, we used the technical requirements of this rule as guidance in our evaluation of the fire protection program.

By letter, dated May 21, 1981, the applicant provided a comparison of its fire protection program with the NRC guidelines given in the technical requirements of Appendix R. The applicant's program is in conformance with these guidelines.

9.5.10 Conclusion

There is one unresolved fire protection item to be reviewed. This item involves the fire dampers (Section 9.5.3.2). We will report our review of this item in a supplement to the Safety Evaluation Report. The applicant has been informed that all fire protection items need to be resolved prior to fuel loading.

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Supplement No. 2

Safety Evaluation Report
related to the operation of Shoreham Nuclear Power Station, Unit No. 1

Docket No. 50-322

Long Island Lighting Company

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

February 1982

9 AUXILIARY SYSTEMS

9.4.1 Control Room Air Conditioning System

In our Safety Evaluation Report, we stated that all control room air conditioning system outside air intakes and exhausts are tornado missile protected. By letter dated November 13, 1981, we were informed by the Shoreham Resident Inspector that the control room air conditioning system east air intake is not tornado protected and that all the piping from the east air intake is not in a tornado protected structure.

The east air intake is a remote intake located in the radwaste building and therefore penetration of missiles via this air intake will not affect safe plant shutdown and will not prevent operation of the control room air conditioning system since the air intake located in the control building is tornado missile protected. Since only one air intake is necessary for operation of the control room air conditioning system, protection of the remote air intake is not required. Many plants have only one air intake. Therefore, our previous conclusion that the control room air conditioning system is acceptable remains unchanged.

9.5 Fire Protection System

9.5.3.2 Fire Doors and Dampers

In Supplement No. 1 to the Safety Evaluation Report, we stated that certain areas of the plant contained motorized 1½-hour fire dampers in which the motorized assembly, including cables, are not U.L. listed. We were concerned that the unlisted assemblies would prevent the fire dampers from performing their function.

By letters dated September 25, 1981 and October 13, 1981, the applicant provided additional information. The installation has been modified to include solenoid and motor circuits approved by U.L. As a result, we now conclude that the fire dampers, as modified, meet the design guidelines of Section D.1.j of Appendix A to BTP ASB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants," and are, therefore, acceptable.

Based on our review, we conclude that the Shoreham fire protection program will meet the technical requirements of Appendix R to 10 CFR Part 50, when committed modifications have been completed, meets the guidelines of Appendix A to BTP ASB 9.5-1, meets the requirements of General Design Criterion 3, and is, therefore, acceptable.

NUREG-0420
Supplement No. 4

Safety Evaluation Report
related to the operation of Shoreham Nuclear Power Station, Unit No. 1

Docket No. 50-322

Long Island Lighting Company

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

September 1983

in Revision 5 to the Design Assessment Report (DAR). The 30 representative piping systems were located throughout the reactor building and contained the least design margin available for accepting potential increases in dynamic loads. It was shown that the pipe stresses and support loads for these 30 piping subsystems were within design allowables when reassessed to the NUREG-0808 loads. However, as a result of a meeting on August 9, 1982, with the applicant, the staff concluded that in addition to reassessing the 30 piping systems with the least design margin, the applicant should also reevaluate all piping systems affected by the high frequency exceedances associated with the NUREG-0808 loads. The applicant committed to perform this reevaluation of high frequency exceedance in a letter from J. L. Smith to H. R. Denton dated August 20, 1982.

In a letter from J. L. Smith to H. R. Denton dated March 17, 1983, the applicant submitted the results of the high frequency reevaluation. The results included a reevaluation of an additional 67 piping subsystems.

In all cases, it was found that the piping and supports that are affected by the NUREG-0808 high frequency load increases were designed with sufficient design margin to accommodate the increase.

Therefore, based on the results of the assessment performed by the applicant in Revision 5 of the DAR and on the results of the reevaluation reported in the March 17, 1983, letter, the staff concludes that the applicant has satisfactorily demonstrated that the piping and supports on the Shoreham facility have been adequately designed to withstand the suppression pool hydrodynamic loads associated with the BWR Mark II containment. Thus, the confirmatory item associated with the ability of the Shoreham piping systems to accommodate steam condensation oscillation and chugging loads is considered to be resolved.

3.12 Reactor Building Internal Flooding

3.12.1 Background

The NRC staff had expressed concern about the potential for flooding safety-related equipment in the event of a pipe break in the Shoreham reactor building (memorandum from R. W. Starostecki, NRC Region I, to D. G. Eisenhut dated June 8, 1982).

Both core spray pumps, all four RHR pumps, the high pressure coolant injection (HPCI) turbine and pump, and the reactor core isolation cooling (RCIC) turbine and pump are located on the lowest level (8-foot elevation) of the reactor building. There are no flood walls or barriers separating redundant trains of this equipment. The applicant has provided nonsafety-related pumps and alarms, safety-related flooding alarms, and one 100-gpm safety-related pump to return water from the 8-foot elevation sump to the suppression pool. The small safety-related pump-back system pump has been provided to deal with postulated post-LOCA leakage in the reactor building; the adequacy of the pump for this purpose has been addressed separately.

The applicant proposes to rely on the safety-related flooding alarms, fluid system instrumentation, and operator actions to prevent flood damage to essential reactor building equipment as a result of high- and moderate-energy pipe breaks during normal operation. The adequacy of the applicant's proposals to protect against internal flood damage at the 8-foot elevation of the reactor building during normal operation is addressed below.

3.12.2 Evaluation

The applicant provided an analysis of the effects of pipe breaks in the reactor building in Appendix 3C to the Shoreham Final Safety Analysis Report (FSAR). The applicant noted in Appendix 3C that the maximum flooding rate at the 8-foot elevation of the reactor building would result from an RHR pump discharge line break with a leakage rate of about 2900 gpm at 350 psig. The applicant determined that other moderate-energy line cracks as well as high-energy line breaks in the reactor building would have a lower leakage rate; thus, an evaluation of the RHR leak was used as the limiting condition for which flood protection is provided. Plant alarms followed by operator entry into the reactor building would be used to determine the existence and location of pipe breaks in the reactor building. The applicant estimates that the limiting RHR system leak could be detected, identified, and isolated within 30 minutes. The existence of a leak would be alarmed almost immediately by redundant, safety-related reactor building flooding alarms, which alarm at a water level of 1/2 inch at the 8-foot elevation of the reactor building. For the limiting condition--an RHR discharge line crack during shutdown cooling, or refueling operations--the applicant calculates that the water level at the 8-foot elevation would approach a depth of 22 inches in the 30 minutes allowed for operator action. Because the postulated line crack would not affect the availability of offsite power, the nonsafety-related sump pumps would be available to reduce the 30-minute flooding depth to 20 inches. The applicant states that shutdown cooling capability would be maintained for this maximum leakage if isolation takes place within 30 minutes.

On August 24, 1982, a meeting was held at the Shoreham site between the applicant and members of the NRC staff. After a tour of the reactor building 8-foot elevation, the staff expressed a concern that identification of a specific leak location and isolation of that leak within 30 minutes may not be possible for all break locations. By letter dated September 9, 1982, the applicant was asked to demonstrate that plant procedures and instrumentation would be adequate to ensure leak detection, identification, and isolation within 30 minutes for all postulated pipe breaks in the reactor building. The applicant was also asked to demonstrate that access to the 8-foot elevation for the purpose of break location identification would be possible, considering the accumulation of potentially radioactive and/or thermally hot water on the elevation, and that the accumulation of water could submerge the leak.

By letter dated December 3, 1982, the applicant provided the additional information requested above. The applicant noted that the analysis in Appendix C to the FSAR was based on preventing flood damage to RHR flow indication instrumentation located approximately 2 feet above the 8-foot elevation floor. The applicant stated, however, that this instrumentation is not required for safe shutdown. The applicant further stated that flooding depths of up to 4 feet above the 8-foot elevation floor could be postulated before damage to essential safe shutdown equipment would be incurred. However, the applicant provided the information in the following paragraphs to demonstrate that postulated leaks could be isolated before a flooding depth of 2 feet is attained.

The applicant's submittal of December 3, 1983 addressed the limiting RHR system pipe crack as well as other postulated leaks in the reactor building. In all cases, redundant safety-related instrumentation would alarm a flooding depth of 1/2 inch on the 8-foot elevation floor. The applicant demonstrated that safety-related instrumentation is available for the operator to identify and isolate a postulated 2900-gpm RHR system leak from the control room in less than 30 minutes. The next largest leakage flow (650 gpm) would occur from an HPCI system leak. At 650 gpm, approximately 2 hours would be available to isolate the leak before the flooding depth reached 2 feet. This postulated leak would not result in a harsh thermal or radioactive environment in the reactor building and would not prevent operator access for identification of the leak location.

In addition to RHR system leaks, the applicant also addressed reactor building pipe breaks that could result in radioactive or thermally hot leakage. A break in the hot water heating (HWH) system would flood the reactor building with hot water, but the flooding depth would be limited to 3 inches initially because of the limited system water inventory. With no operator action to isolate the leak, continued makeup to the HWH system at 25 gpm would leak to the reactor building, and the flood depth would approach 2 feet after several days. However, various indications would alert the operator to the system leakage, and the leak can be isolated from the control room. Aside from RHR system pipe cracks, only a break in the reactor water cleanup system could introduce radioactive leakage into the reactor building. The maximum leakage of 180 gpm from the system could be identified and terminated in the control room.

The applicant noted that although an RHR system crack would be most likely to be hidden by submergence as a result of flooding, the leak location can be identified and isolated from the control room. Other leaks would be less likely to be submerged because of the lower leakage rates versus height of the piping from the floor of the 8-foot elevation. Alarm response procedures and operating procedures are being modified to address both post-LOCA leaks and moderate-energy line cracks postulated to occur during normal operation. These procedures will direct the operator to start leak location identification walk-throughs on the 8-foot elevation to ensure leak detection before the leak is submerged. The applicant is also participating in the Boiling Water Reactor (BWR) Owners Group program to develop a secondary containment control procedure that will provide additional specific guidance for operator response to postulated flooding events.

3.12.3 Conclusions

The NRC staff has determined, on the basis of its review, that the applicant has adequately identified and provided internal flooding protection for systems and components at the 8-foot elevation of the reactor building required for safe shutdown in the event of pipe failures. The reactor building design meets the criteria set forth in Branch Technical Position (BTP) ASB 3-1 regarding protection of safety-related systems and components from postulated piping system failures. The design, therefore, meets the requirements of General Design Criterion (GDC) 4, "Environmental and Missile Design Bases," regarding flooding protection for pipe breaks. The NRC staff therefore, concludes that the reactor building design for protection against internal flooding is acceptable.