Amendment 39
October 22, 1979

Pilgrim Station Unit 2

Preliminary Safety Analysis Report

1273 344

BOSTON
Edison
COMPANY

7911010172

PILGRIM STATION UNIT 2
PRELIMINARY SAFETY ANALYSIS REPORT
Amendment 39, October 22, 1979

The change pages included in this Amendment comprise general update information and pages added in response to NRC commitments.

All pages supplied with this Amendment are identified by the Amendment number and date in the upper outside corner of each page.

All insert material is collated in the order in which it will be inserted in its respective chapter.

The following change page instructions should be used as a guide for the removal of old pages and insertion of change pages for this Amendment. These instructions will serve as a permanent record of the affected pages of this Amendment and should be placed at the end of each chapter following the yellow AEC Question tab page.

The new title page supplied should replace the old title page in Volume I. These general instruction pages should also be placed following the new title page, after the instruction pages for the previous amendment.

1273 345

BOSTON Edison COMPANY

CHANGE PAGE INSTRUCTIONS

ALL VOLUMES

Table of Contents

Remove	Insert
i/ii	i/ii

VOLUME I

Volume I 1-i/1-ii	Volume I 1-i/1-ii
Volume I 1-iii/1-iv	Volume I 1-iii/1-iv

CHAPTER 1

1.2-21/Blank and 1.2-21A/1.2-22	1.2-21/1.2-22
----	1C-i thru 1C-48
----	AM39.1-i

1273 346

TABLE OF CONTENTS

CHAPTER 1 - VOLUME I

INTRODUCTION AND GENERAL

DESCRIPTION OF PLANT

1273 347

CHAPTER 2 - VOLUME I

SITE CHARACTERISTICS

1273 348

TABLE OF CONTENTS

CHAPTER 1 - VOLUME I

INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

CHAPTER 2 - VOLUME I

SITE CHARACTERISTICS

1273 350

TABLE OF CONTENTS

CHAPTER 1 - VOLUME I

INTRODUCTION AND GENERAL

DESCRIPTION OF PLANT

1273 351

CHAPTER 2 - VOLUME I

SITE CHARACTERISTICS

1273 352

TABLE OF CONTENTS

CHAPTER 1 - VOLUME I

INTRODUCTION AND GENERAL

DESCRIPTION OF PLANT

1273 353

CHAPTER 2 - VOLUME I

SITE CHARACTERISTICS

TABLE OF CONTENTS

CHAPTER 1 - VOLUME I

INTRODUCTION AND GENERAL

DESCRIPTION OF PLANT

CHAPTER 2 - VOLUME I

SITE CHARACTERISTICS

TABLE OF CONTENTS

CHAPTER 1 - VOLUME I

INTRODUCTION AND GENERAL

DESCRIPTION OF PLANT

1273 357

CHAPTER 2 - VOLUME I

SITE CHARACTERISTICS

TABLE OF CONTENTS

CHAPTER 1 - VOLUME I

INTRODUCTION AND GENERAL

DESCRIPTION OF PLANT

1273 359

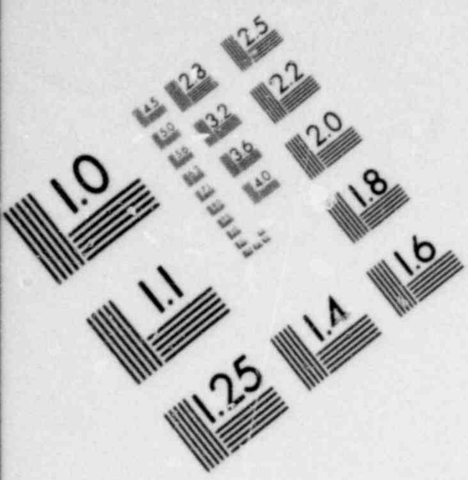
CHAPTER 2 - VOLUME I

SITE CHARACTERISTICS

1273 360

IMAGE EVALUATION
TEST TARGET (MT-3)

6″



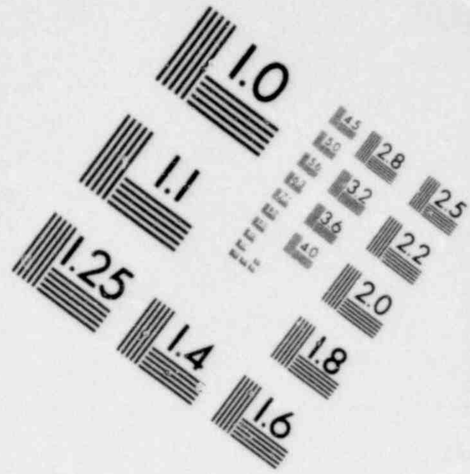
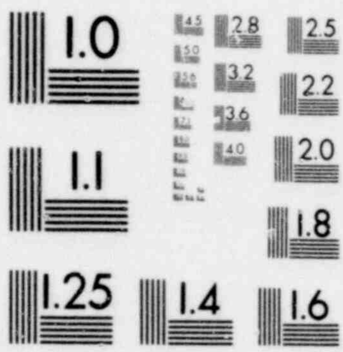
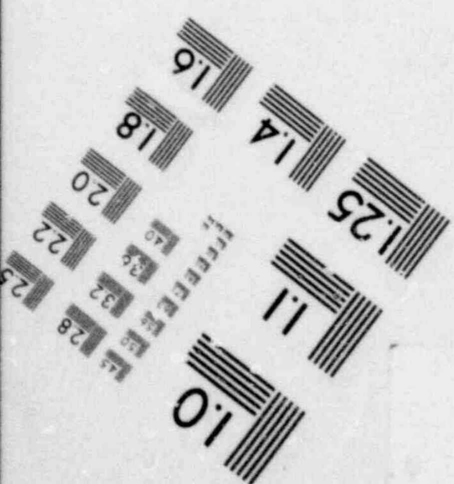


IMAGE EVALUATION TEST TARGET (MT-3)



6″



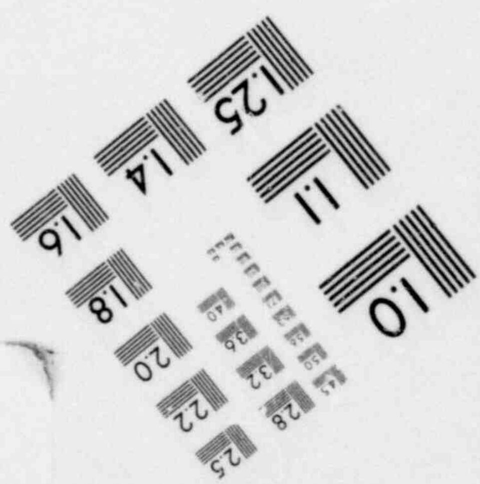


TABLE OF CONTENTS

CHAPTER 1 - VOLUME I

INTRODUCTION AND GENERAL

DESCRIPTION OF PLANT

CHAPTER 2 - VOLUME I

SITE CHARACTERISTICS

TABLE OF CONTENTS

CHAPTER 1 - VOLUME I

INTRODUCTION AND GENERAL

DESCRIPTION OF PLANT

1274 003

CHAPTER 2 - VOLUME I

SITE CHARACTERISTICS

1274 004

TABLE OF CONTENTS

CHAPTER 1 - VOLUME I

INTRODUCTION AND GENERAL

DESCRIPTION OF PLANT

1274 005

CHAPTER 2 - VOLUME I

SITE CHARACTERISTICS

TABLE OF CONTENTS

CHAPTER 1 - VOLUME I

INTRODUCTION AND GENERAL

DESCRIPTION OF PLANT

CHAPTER 2 - VOLUME I

SITE CHARACTERISTICS

1274 008

TABLE OF CONTENTS

CHAPTER 1 - VOLUME I

INTRODUCTION AND GENERAL

DESCRIPTION OF PLANT

CHAPTER 2 - VOLUME I

SITE CHARACTERISTICS

1274 010

TABLE OF CONTENTS

CHAPTER 1 - VOLUME I

INTRODUCTION AND GENERAL

DESCRIPTION OF PLANT

1274 011

1274 012

1274 013

LIST OF TABLES

CHAPTER 1 - VOLUME I

INTRODUCTION AND GENERAL

DESCRIPTION OF PLANT

1274 014

The main condenser transfers rejected heat in the turbine exhaust steam directly to sea water being circulated through the condenser tubes by the circulating water pumps. Circulating water is extracted from and discharged to Cape Cod Bay.

Parallel strings of regenerative feedwater heaters utilize turbine extraction steam to heat the condensate and feedwater after it is pumped from the main condenser hotwell and before it is pumped into the economizer section of each steam generator.

A more detailed discussion of the steam and power conversion system is presented in Chapter 10.

1.2.11 RADIOACTIVE WASTE MANAGEMENT SYSTEMS

The waste management systems provide the means for controlled handling, storage and disposal of liquid, gaseous and solid wastes. The waste release rates will be within the guidelines and limits for waste release established by applicable regulations.

Radioactive liquid wastes are processed in the liquid waste management system. Some of the waste management system liquids are recycled for use in the Facility. Liquid wastes will be sampled prior to release from the Facility.

All solid wastes will be placed in containers for offsite disposal.

Radioactive waste gases evolve from several plant components as a result of routine plant operations. These waste gases are collected and processed through the gaseous waste management system (GWMS). The GWMS employs moisture removal, followed by adsorption on activated charcoal to reduce the waste gas radioactivity content. After passage through the GWMS, the gases are discharged to the environment. 26

1.2.12 EFFECTS OF FACILITY CONSTRUCTION ON UNIT 1

Facility construction will take place during Unit 1 operation; therefore, special precautions as outlined below are utilized to eliminate any interference with the normal operation of Unit 1.

The basic principles used to ensure safe operation of Unit 1 during Facility construction are as follows:

A. The critical areas of the units are physically separated.

B. Barriers will be constructed as necessary to prevent unauthorized personnel or equipment transit into the Unit 1 area.

1274 015

C. Construction procedures that would disrupt the operation of Unit 1, or defeat the functions of the critical structures, systems or components of Unit 1 will be conducted during scheduled shutdown periods for Unit 1.

Barrier structures, such as a continuous chain fence or secured gates and doors enclosing Unit 1, will prohibit unauthorized travel between the two areas. Construction offices, parking and equipment laydown areas are outside the Unit 1 areas. Sufficient access roads are provided, so that deliveries are made without having to cross the Unit 1 area. In addition, access to the Facility construction area is controlled.

Construction personnel will be instructed in emergency evacuation procedures. Evacuation routes will be marked and emergency procedures posted at key construction areas. Film badges are not normally issued to construction personnel. Area environmental monitors will be placed at strategic locations in the construction area.

39 1.2.13 Actions Resulting from Review of the Three Mile Island Incident

The Pilgrim 2 facility will be designed, constructed, staffed, and operated in accordance with lessons learned from review of the incident in March, 1979 at the Three Mile Island Unit 2 Nuclear Power Plant. Appendix 1C presents the NRC positions on various issues regarding these lessons learned, and the Applicant's responses related to Pilgrim 2. If there are instances in which the statements in Appendix 1C conflict with statements elsewhere in this PSAR and made prior to the issuance of Appendix 1C, the statement in Appendix 1C applies.

1274 016

1274 017

39

APPENDIX 1C

COMMITMENTS RELATED TO
REVIEW OF THE INCIDENT AT
THREE MILE ISLAND UNIT 2

1274 018

The following pages identify the Applicant's commitments regarding the design and operation of Pilgrim 2 in response to the review of the incident at Three Mile Island Unit 2.

Commitments in this Appendix supersede any conflicting statements elsewhere in the PSAR where such conflicting statements were made earlier than the date of the current revision of this appendix.

The following text consists of NRC positions and Boston Edison responses on each one. The NRC positions are those from NUREG-0578 dated July 1979, as modified by a letter dated October 10, 1979 from Domenic B. Vassallo, USNRC, to all Construction Permit Applicants; and from IE Bulletin 79-06B dated April 14, 1979, as modified by IE Bulletin 79-06C dated July 26, 1979.

1274 019

1C.1 Emergency Power Supply Requirements for the Pressurizer Heaters, Power-Operated Relief Valves and Block Valves, and Pressurizer Level Indicators in PWRs (Section 2.1.1)

POSITION

Consistent with satisfying the requirements of General Design Criteria 10, 14, 15, 17, and 20 of Appendix A to 10 CFR Part 50 for the event of loss of offsite power, the following positions shall be implemented:

Pressurizer Heater Power Supply

1. The pressurizer heater power supply design shall provide the capability to supply, from either the offsite power source or the emergency power source (when offsite power is not available), a predetermined number of pressurizer heaters and associated controls necessary to establish and maintain natural circulation at hot standby conditions. The required heaters and their controls shall be connected to the emergency buses in a manner that will provide redundant power supply capability.

2. Procedures and training shall be established to make the operator aware of when and how the required pressurizer heaters shall be connected to the emergency buses. If required, the procedures shall identify under what conditions selected emergency loads can be shed from the emergency power source to provide sufficient capacity for the connection of the pressurizer heaters.

3. The time required to accomplish the connection of the preselected pressurizer heater to the emergency buses shall be consistent with the timely initiation and maintenance of natural circulation conditions.

4. Pressurizer heater motive and control power interfaces with the emergency buses shall be accomplished through devices that have been qualified in accordance with safety-grade requirements.

Power Supply for Pressurizer Relief and Block Valves and Pressurizer Level Indicators

1. Motive and control components of the power-operated relief valves (PORVs) shall be capable of being supplied from either the offsite power source or the emergency power source when the offsite power is not available.

2. Motive and control components associated with the PORV block valves shall be capable of being supplied from either the offsite power source or the emergency power source when the offsite power is not available.

3. Motive and control power connections to the emergency buses for the PORVs and their associated block valves shall be through devices that have been qualified in accordance with safety-grade requirements.

4. The pressurizer level indication instrument channels shall be powered from the vital instrument buses. These buses shall have the capability of being supplied from either the offsite power source or the emergency power source when offsite power is not available.

1274 020

RESPONSE

Pressurizer Heater Power Supply

1. The pressurizer heater power supply design will have the capability to supply, from either the offsite power source or the emergency power source (when offsite power is not available), a predetermined number of pressurizer heaters and associated controls necessary to establish and maintain natural circulation at hot standby conditions. The required heaters and their controls will be connected to the emergency buses in a manner that will provide redundant power supply capability.

2. Procedures and training will be established to make the operator aware of when and how the required pressurizer heaters shall be connected to the emergency buses. If required, the procedures will identify under what conditions selected emergency loads can be shed from the emergency power source to provide sufficient capacity for the connection of the pressurizer heaters.

3. The time required to accomplish the connection of the preselected pressurizer heater to the emergency buses will be consistent with the timely initiation and maintenance of natural circulation conditions.

4. Pressurizer heater motive and control power interfaces with the emergency buses will be accomplished through devices that have been qualified in accordance with safety-grade requirements.

Power Supply for Pressurizer Relief and Block Valves and Pressurizer Level Indicators

1. Motive and control components of the power-operated relief valves (PORV's) will be capable of being supplied from either the offsite power source or the emergency power source when the offsite power is not available.

2. Motive and control components associated with the PORV block valves will be capable of being supplied from either the offsite power source or the emergency power source when the offsite power is not available.

3. Motive and control power connections to the emergency buses for the PORV's and their associated block valves will be through devices that have been qualified in accordance with safety-grade requirements.

4. The pressurizer level indication instrument channels will be powered from the vital instrument buses. These buses will have the capability of being supplied from either the offsite power source or the emergency power source when offsite power is not available.

1274 021

1C.2 Performance Testing for BWR and PWR Relief and Safety Valves (Section 2.1.2)

POSITION

Pressurized water reactor and boiling water reactor licensees and applicants shall conduct testing to qualify the reactor coolant system relief and safety valves under expected operating conditions for design basis transients and accidents. The licensees and applicants shall determine the expected valve operating conditions through the use of analyses of accidents and anticipated operational occurrences referenced in Regulatory Guide 1.70, Revision 2. The single failures applied to these analyses shall be chosen so that the dynamic forces on the safety and relief valves are maximized. Test pressures shall be the highest predicted by conventional safety analysis procedures. Reactor coolant system relief and safety valve qualification shall include qualification of associated control circuitry piping and supports as well as the valves themselves.

RESPONSE

Pilgrim Unit 2 will support the industry efforts to conduct testing necessary to qualify the reactor coolant system relief and safety valves under expected operating conditions for design basis transients and accidents. The expected valve operating conditions will be determined through the use of analyses of accidents and anticipated operational occurrences referenced in Regulatory Guide 1.70, Rev. 2. The single failures applied to these analyses will be chosen so that the dynamic forces on the safety and relief valves are maximized. Test pressures will be the highest predicted by conventional safety analysis. Reactor coolant system relief and safety valve qualification will include qualification of associated control circuitry, piping and supports necessary for proper valve performance, as well as the valves themselves.

1274 022

1C.3 Direct Indication of Power-Operated Relief Valve and Safety Valve Position for PWRs and BWRs (Section 2.1.3.a)

POSITION

Reactor system relief and safety valves shall be provided with a positive indication in the control room derived from a reliable valve position detection device or a reliable indication of flow in the discharge pipe.

RESPONSE:

Reactor system relief and safety valves will be provided with a positive indication in the control room derived from a reliable valve position detection device or a reliable indication of flow in the discharge pipe.

1274 023

1C.4 Instrumentation for Detection of Inadequate Core Cooling in PWRs and BWRs (Section 2.1.3.b)

POSITION

1. **Licensees shall develop procedures to be used by the operator to recognize inadequate core cooling with currently available instrumentation. The licensee shall provide a description of the existing instrumentation for the operators to use to recognize these conditions. A detailed description of the analyses needed to form the basis for operator training and procedure development shall be provided pursuant to another short-term requirement, "Analysis of Off-Normal Conditions, Including Natural Circulation" (see Section 2.1.9 of this appendix).**

   In addition, each PWR shall install a primary coolant saturation meter to provide on-line indication of coolant saturation condition. Operator instruction as to use of this meter shall include consideration that it not to be used exclusive of other related plant parameters.

2. **Licensees shall provide a description of any additional instrumentation or controls (primary or backup) proposed for the plant to supplement those devices cited in the preceding section giving an unambiguous, easy-to-interpret indication of inadequate core cooling. A description of the functional design requirements for the system shall also be included. A description of the procedures to be used with the proposed equipment, the analysis used in developing these procedures, and a schedule for installing the equipment shall be provided.**

RESPONSE

1. Procedures to be used by the operator will be developed to recognize inadequate core cooling. A detailed description of the analyses which form the basis for operator training and procedure development and a description of the instrumentation for the operators to use to recognize these conditions will be provided in the FSAR.

   Instrumentation will be installed in the control room which will indicate the approach of the reactor coolant system to saturation conditions. Operator instructions as to the use of this instrumentation will stress that it not be used exclusive of other related plant parameters.

2. Since the final design is not complete, any aspect of position 2 above will be appropriately addressed by our commitment in response to position 1 above.

1274 024

1C.5 Containment Isolation Provisions for PWRs and BWRs (Section 2.1.4)

POSITION

1. **All containment isolation system designs shall comply with the recommendations of SRP 6.2.4; i.e., that there be diversity in the parameters sensed for the initiation of containment isolation.**

2. **All plants shall give careful reconsideration to the definition of essential and non-essential systems, shall identify each system determined to be essential, shall identify each system determined to be non-essential, shall describe the basis for selection of each essential system, shall modify their containment isolation designs accordingly, and shall report the results of the re-evaluation to the NRC.**

3. **All non-essential systems shall be automatically isolated by the containment isolation signal.**

4. **The design of control systems for automatic containment isolation valves shall be such that resetting the isolation signal will not result in the automatic reopening of containment isolation valves. Reopening of containment isolation valves shall require deliberate operator action.**

RESPONSE

1. The Pilgrim Unit 2 containment isolation system design will comply with the recommendations of SRP 6.2.4; i.e., that there be diversity in the parameters sensed for the initiation of containment isolation.

2. Careful consideration to the definition of essential and non-essential systems shall be given, and each system shall be identified as to whether it is essential or non-essential. The bases for the selection of each essential system shall be provided.

3. All non-essential systems will be automatically isolated by the containment isolation signal.

4. The design of control systems for automatic containment isolation valves will be such that resetting the isolation signals will not result in the automatic reopening of containment isolation valves. Reopening of containment isolation valves will require deliberate operator action.

1274 025

1C.6 **Dedicated Penetrations for External Recombiners or Post-Accident Purge Systems (Section 2.1.5.a)**

POSITION

Plants using external recombiners or purge systems for post-accident combustible gas control of the containment atmosphere should/provide containment isolation systems for external recombiner or purge systems that are dedicated to that service only, that meet the redundancy and single failure requirements of General Design Criteria 54 and 56 of Appendix A to 10 CFR Part 50, and that are sized to satisfy the flow requirements of the recombiner or purge system.

RESPONSE

Post-accident combustible gas control of the containment atmosphere for Pilgrim Unit 2 will be performed by redundant internal recombiners. Capability for post-accident containment purging will also be provided. The containment isolation system for this containment purge will be dedicated only to combustible gas control, will meet the redundancy and single failure requirements of General Design Criteria 54 and 56 of Appendix A to 10 CFR Part 50, and will be sized to satisfy the flow requirements of the purge system.

1274 026

1C.7 Inerting BWR Containments (Section 2.1.5.b)

POSITION

It shall be required that the Vermont Yankee and Hatch 2 Mark I BWR containments be inerted in a manner similar to other operating BWR plants. Inerting shall also be required for near term OL licensing of Mark I and Mark II BWRs.

RESPONSE

This is not applicable to Pilgrim Unit 2.

1274 027

1C.8 **Capability to Install Hydrogen Recombiner at Each Light Water Nuclear Power Plant (Section 2.1.5.c)**

POSITION (Minority View)

1. **All licensees of light water reactor plants shall have the capability to obtain and install recombiners in their plants within a few days following an accident if containment access is impaired and if such a system is needed for long-term post-accident combustible gas control.**

2. **The procedures and bases upon which the recombiners would be used on** all plants should be the subject of a review by the licensees in considering sheilding requirements and personnel exposure limitations as demonstrated to be necessary in the case of TMI-2.

RESPONSE

This is not applicable to Pilgrim Unit 2. Redundant internal recombiners are provided.

1274 028

1C.9 Integrity of Systems Outside Containment Likely to Contain Radioactive Materials (Engineered Safety Systems and Auxiliary Systems) for PWRs and BWRs (Section 2.1.6.a)

POSITION

Applicants and licensees shall immediately implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as-low-as-practical levels. This program shall include the following:

1. Immediate Leak Reduction

a. Implement all practical leak reduction measures for all systems that could carry radioactive fluid outside of containment.

b. Measure actual leakage rates with system in operation and report them to the NRC.

2. Continuing Leak Reduction

Establish and implement a program of preventive maintenance to reduce leakage to as-low-as-practical levels. This program shall include periodic integrated leak tests at a frequency not to exceed refueling cycle intervals.

RESPONSES

1. The design of systems outside containment that would or could contain highly radioactive fluids during a serious accident or transient will be carefully reviewed. This review will include implementing all practical leak reduction measures for all systems that could carry radioactive fluid outside of containment, to reduce leakage to as-low-as-practical levels. Actual leakage rates with the systems in operation will be measured and results reported to the NRC.

2. A program of preventive maintenance to reduce leakage to as-low-as-practical levels will be established and implemented. This program shall include periodic integrated leak tests at a frequency which will be established at the time Pilgrim Unit 2 receives an operating license.

1274 029

1C.10 Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems Which May Be Used in Post-Accident Operations (Section 2.1.6.b)

POSITION

With the assumption of a post-accident release of radioactivity equivalent to that described in Regulatory Guides 1.3 and 1.4 (i.e., the equivalent of 50% of the core radioiodine and 100% of the core noble gas inventory are contained in the primary coolant), each licensee shall perform a radiation and shielding design review of the spaces around systems that may, as a result of an accident, contain highly radioactive materials. The design review should identify the location of vital areas and equipment, such as the control room, radwaste control stations, emergency power supplies, motor control centers, and instrument areas, in which personnel occupancy may be unduly limited or safety equipment may be unduly degraded by the radiation fields during post-accident operations of these systems.

Each licensee shall provide for adequate access to vital areas and protection of safety equipment by design changes, increased permanent or temporary shielding, or post-accident procedural controls. The design review shall determine which types of corrective actions are needed for vital areas throughout the facility.

RESPONSE

With the assumption of a post-accident release of radioactivity equivalent to that described in Regulatory Guide 1.4 (i.e., the equivalent of 50% of the core radioiodine and 100% of the core noble gas inventory are contained in the primary coolant) a radiation and shielding design review will be performed of the spaces around systems that may, as a result of an accident, contain highly radioactive materials. The design review will identify the location of vital areas and equipment in which personnel occupancy may be unduly limited or safety equipment may be unduly degraded by the radiation fields during post-accident operation of these systems.

Adequate access to vital areas and protection of safety equipment by design changes, increased permanent or temporary shielding, or post-accident procedural controls will be provided. The design review will determine which types of corrective actions are needed for vital areas throughout the facility.

1274 030

1C.11 Automatic Initiation of the Auxiliary Feedwater System for PWRs (Section 2.1.7.a)

POSITION

Consistent with satisfying the requirements of General Design Criterion 20 of Appendix A to 10 CFR Part 50 with respect to the timely initiation of the auxiliary feedwater system, the following requirements shall be implemented in the short term:

1. The design shall provide for the automatic initiation of the auxiliary feedwater system.

2. The automatic initiation signals and circuits shall be designed so that a single failure will not result in the loss of auxiliary feedwater system function.

3. Testability of the initiating signals and circuits shall be a feature of the design.

4. The initiating signals and circuits shall be powered from the emergency buses.

5. Manual capability to initiate the auxiliary feedwater system from the control room shall be retained and shall be implemented so that a single failure in the manual circuits will not result in the loss of system function.

6. The a-c motor-driven pumps and valves in the auxiliary feedwater system shall be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.

7. The automatic initiating signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the ..S from the control room.

In the long term, the automatic initiation signals and circuits shall be upgraded in accordance with safety-grade requirements.

1274 031

RESPONSE

1. The design provides for the automatic initiation of the emergency feedwater system.

2. The automatic initiation signals and circuits are designed so that a single failure will not result in the loss of emergency feedwater system function.

3. Testability of the initiating signals and circuits will be a feature of the design.

4. The initiating signals and circuits will be powered from the emergency buses.

5. Manual capability to initiate the emergency feedwater system from the control room is included in the design in such a manner that a single failure in the manual circuits will not result in the loss of system function.

6. The a-c motor driven pumps and valves in the emergency feedwater system are included in the automatic actuation (simultaneous or sequential) of the loads to the emergency buses.

7. The automatic initiating signals and circuits are designed so that their failure will not result in the loss of manual capability to initiate the emergency feedwater system from the control room.

8. The automatic initiation signals and circuits for the emergency feedwater system are in accordance with safety-grade requirements.

1274 032

1C.12 Auxiliary Feedwater Flow Indication to Steam Generators for PWRs (Section 2.1.7.b)

POSITION

Consistent with satisfying the requirements set forth in GDC 13 to provide the capability in the control room to ascertain the actual performance of the AFWS when it is called to perform its intended function, the following requirements shall be implemented:

1. **Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.**

2. **The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements of the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the *Standard Review Plan, Section 10.4.9*.**

RESPONSE

1. Safety-grade indication of emergency feedwater flow to each steam generator will be provided in the control room.

2. The emergency feedwater flow instrument channels will be powered from the emergency buses consistent with satisfying the emergency power diversity requirements of the emergency feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9.

1274 033

1C.13 Improved Post-Accident Sampling Capability (Section 2.1.8.a)

POSITION

A design and operational review of the reactor coolant and containment atmosphere sampling systems shall be performed to determine the capability of personnel to promptly obtain (less than 1 hour) a sample under accident conditions without incurring a radiation exposure to any individual in excess of 3 and 18 3/4 Rems to the whole body or extremities, respectively. Accident conditions should assume a Regulatory Guide 1.3 or 1.4 release of fission products. If the review indicates that personnel could not promptly and safely obtain the samples, additional design features or shielding should be provided to meet the criteria.

A design and operational review of the radiological spectrum analysis facilities shall be performed to determine the capability to promptly quantify (less than 2 hours) quantify certain radioisotopes that are indicators of the degree of core damage. Such radionuclides are noble gases (which indicate cladding failure), iodines and cesiums (which indicate high fuel temperatures), and non-volatile isotopes (which indicate fuel melting). The initial reactor coolant spectrum should correspond to a Regulatory Guide 1.3 or 1.4 release. The review should also consider the effects of direct radiation from piping and components in the auxiliary building and possible contamination and direct radiation from airborne effluents. If the review indicates that the analyses required cannot be performed in a prompt manner with existing equipment, then design modifications or equipment procurement shall be undertaken to meet the criteria.

In addition to the radiological analyses, certain chemical analyses are necessary for monitoring reactor conditions. Procedures shall be provided to perform boron and chloride chemical analyses assuming a highly radioactive initial sample (Regulatory Guide 1.3 or 1.4 source term). Both analyses shall be capable of being completed promptly; i.e., the boron sample analysis within an hour and the chloride sample analysis within a shift.

1274 034

(Section 2.1.8.a)

RESPONSE

A design and operational review of the reactor coolant and containment atmosphere sampling systems will be performed to determine the capability of personnel to promptly obtain a sample under accident conditions without incurring a radiation exposure to any individual in excess of 3 and 18 3/4 Rems to the whole body or extremities, respectively. Accident conditions will assume a Regulatory Guide 1.4 release of fission products. If the review indicates that personnel could not promptly and safely obtain the samples, additional design features or shielding will be provided to meet the criteria.

A design and operational review of the radiological spectrum analysis facilities will be performed to determine the capability to promptly quantify certain radioisotopes that are indicators of the degree of core damage. Such radionuclides are noble gases (which indicate cladding failure), iodines and cesiums (which indicate high fuel temperatures), and non-volatile isotopes (which indicate fuel melting). Analysis of the initial reactor coolant spectrum will correspond to a Regulatory Guide 1.4 release. The review will also consider the effects of direct radiation from piping and components in the auxiliary building and possible contamination and direct radiation from airborne effluents. If the review indicates that the analyses required cannot be performed in a prompt manner with existing equipment, then design modifications or equipment procurement will be undertaken to meet the criteria.

In addition to the radiological analyses, certain chemical analyses are necessary for monitoring reactor conditions. Procedures will be provided to perform boron and chloride chemical analyses assuming a highly radioactive initial sample (Regulatory Guide 1.4 source term). Both analyses will be capable of being completed promptly.

1274 035

1C.14 Increased Range of Radiation Monitors (Section 2.1.8.b)

POSITION

The requirements associated with this recommendation should be considered as advanced implementation of certain requirements to be included in a revision to Regulatory Guide 1.97, "Instrumentation to Follow the Course of an Accident," which has already been initiated, and in other Regulatory Guides, which will be promulgated in the near-term.

1. Noble gas effluent monitors shall be installed with an extended range designed to function during accident conditions as well as during normal operating conditions; multiple monitors are considered to be necessary to cover the ranges of interest.

   a. Noble gas effluent monitors with an upper range capacity of 10^5 μCi/cc (Xe-133) are considered to be practical and should be installed in all operating plants.

   b. Noble gas effluent monitoring shall be provided for the total range of concentration extending from normal condition (ALARA) concentrations to a maximum of 10^5 uCi/cc (Xe-133). Multiple monitors are considered to be necessary to cover the ranges of interest. The range capacity of individual monitors should overlap by a factor of ten.

2. Since iodine gaseous effluent monitors for the accident condition are not considered to be practical at this time, capability for effluent monitoring of radioiodines for the accident condition shall be provided with sampling conducted by adsorption on charcoal or other media, followed by onsite laboratory analysis.

3. In-containment radiation level monitors with a maximum range of 10^8 rad/hr shall be installed. A minimum of two such monitors that are physically separated shall be provided. Monitors shall be designed and qualified to function in an accident environment.

1274 036

(Section 2.1.8.b)

RESPONSE

1. Noble gas effluent monitors will be installed with an extended range designed to function during accident conditions as well as during normal operating conditions; multiple monitors will be provided to cover the ranges of interest.

   a. Noble gas effluent monitors with an upper range capacity of 10^5 uCi/cc (Xe-133) will be installed.

   b. Noble gas effluent monitoring will be provided for the total range of concentration extending from normal conditions (ALARA) concentrations to a maximum of 10^5 uCi/cc (Xe-133). Multiple monitors will be provided to cover the ranges of interest. The range capacity of individual monitors will overlap by a factor of ten.

2. Since iodine gaseous effluent monitors for the accident condition are not considered to be practical at this time, capability for effluent monitoring of radioiodines for the accident condition will be provided with sampling conducted by adsorption on charcoal or other media, followed by onsite laboratory analysis.

3. Monitors suitable for detection of in-containment radiation levels up to 10^8 rad/hour will be provided. Such monitors will be redundant, physically separated, and will be qualified to function in the accident environment to which they will be exposed.

1274 037

1C.1 **Improved Plant Iodine Instrumentation (Section 2.1.8.c)**

POSITION

Each licensee shall provide equipment and associated training and procedures for accurately determining the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident.

RESPONSE

Equipment and associated training and procedures for accurately determining the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident will be provided using state-of-the-art techniques.

1274 038

1C.16 Analysis of Design and Off-Normal Transients and Accidents (Section 2.1.9)

POSITION

Analyses, procedures, and training addressing the following are required:

1. Small break loss-of-coolant accidents;

2. Inadequate core cooling; and

3. Transients and accidents.

Some analysis requirements for small breaks have already been specified by the Bulletins and Orders Task Force. These should be completed. In addition, pretest calculations of some of the Loss of Fluid Test (LOFT) small break tests (scheduled to start in September 1979) shall be performed as means to verify the analyses performed in support of the small break emergency procedures and in support of an eventual long term verification of compliance with Appendix K of 10 CFR Part 50.

In the analysis of inadequate core cooling, the following conditions shall be analyzed using realistic (best-estimate) methods:

1. Low reactor coolant system inventory (two examples will be required - LOCA with forced flow, LOCA without forced flow).

2. Loss of natural circulation (due to loss of heat sink).

These calculations shall include the period of time during which inadequate core cooling is approached as well as the period of time during which inadequate core cooling exists. The calculations shall be carried out in real time far enough that all important phenomena and instrument indications are included. Each case should then be repeated taking credit for correct operator action. These additional cases will provide the basis for developing appropriate emergency procedures. These calculations should also provide the analytical basis for the design of any additional instrumentation needed to provide operators with an unambiguous indication of vessel water level and core cooling adequacy (see Section 2.1.3.b in this appendix).

The analyses of transients and accidents shall include the design basis events specified in Section 15 of each FSAR. The analyses shall include a single active failure for each system called upon to function for a particular event. Consequential failures shall also be considered. Failures of the operators to perform required control manipulations shall be given consideration for permutations of the analyses. Operator actions that could cause the complete loss of function of a safety system shall also be considered. At present, these analyses need not address passive failures or multiple system failures in the short term. In the recent analysis of small break LOCAs, complete loss of auxiliary feedwater was considered. The complete loss of auxiliary feedwater may be added to the failures being considered in the analysis of transients and accidents if it is concluded that more is needed in operator training beyond the short-term actions to upgrade auxiliary feedwater system reliability. Similarly, in the long term, multiple failures and passive failures may be considered depending in part on staff review of the results of the short-term analyses.

1274 039

The transient and accident analyses shall include event tree analyses, which are supplemented by computer calculations for those cases in which the system response to operator actions is unclear or these calculations could be used to provide important quantitative information not available from an event tree. For example, failure to initiate high-pressure injection could lead to core uncovery for some transients, and a computer calculation could provide information on the amount of time available for corrective action. Reactor simulators may provide some information in defining the event trees and would be useful in studying the information available to the operators. The transient and accident analyses are to be performed for the purpose of identifying appropriate and inappropriate operator actions relating to important safety considerations such as natural circulation, prevention of core uncovery, and prevention of more serious accidents.

The information derived from the preceding analyses shall be included in the plant emergency procedures and operator training. It is expected that analyses performed by the NSSS vendors will be put in the form of emergency procedure guidelines and that the changes in the procedures will be implemented by each licensee or applicant.

In addition to the analyses performed by the reactor vendors, analyses of selected transients should be performed by the NRC Office of Research, using the best available computer codes, to provide the basis for comparisons with the analytical methods being used by the reactor vendors. These comparisons together with comparisons to data, including LOFT small break test data, will constitute the short-term verification effort to assure the adequacy of the analytical methods being used to generate emergency procedures.

1274 040

(Section 2.1.9)

RESPONSE

Analyses, procedures, and training will address the following:

1. Small break loss-of-coolant accidents;

2. Inadequate core cooling, and

3. Transients and accidents.

In the analysis of inadequate core cooling, the following conditions will be analyzed using realistic (best-estimate) methods:

1. Low reactor coolant system inventory (two examples will be provided - LOCA with forced flow, LOCA without forced flow).

2. Loss of natural circulation (due to loss of heat sink).

These calculations will include the period of time during which inadequate core cooling is approached as well as the period of time during wh ch inadequate core cooling exists. The calculations will be carried out far enough that all important phenomena and instrument indications are included. Each c se will then be repeated taking credit for correct operator action. Thes additional cases will provide the basis for developing appropriate emergency procedures. These calculations will also provide the analytical basis for the design of any additional instrumentation needed to provide operators with an unambiguous indication of vessel water level and core cooling adequacy.

The analyses of transients and accidents will include the design basis events specified in Section 15 of the FSAR including off-normal conditions and events. The analyses will include the most limiting single active failure for a particular event. Consequential failures will also be considered. Failures of the operators to perform required control manipulations will be considered. Operator actions that could cause the complete loss of function of a safety system will also be considered. These analyses will not address passive failures or multiple system failures in the short term.

The transient and accident event tree methodology used by Boston Edison is the safety sequence diagram method. This technique will be continued to support operator training. Best-estimate transient and accident analyses will be performed to the extent required for the purpose of identifying appropriate and inappropriate operator actions relating to important safety considerations such as natural circulation, prevention of core uncovery, and prevention of more serious accidents.

The information derived from the preceding analyses will be included in the plant emergency procedures and operating training.

1274 041

1C.17 Shift Supervisor's Responsibilities (Section 2.2.1.a)

POSITION

1. The highest level of corporate management of each licensee shall issue and periodically reissue a management directive that emphasizes the primary management responsibility of the shift supervisor for safe operation of the plant under all conditions on his shift and that clearly establishes his command duties.

2. Plant procedures shall be reviewed to assure that the duties, responsibilities, and authority of the shift supervisor and control room operators are properly defined to effect the establishment of a definite line of command and clear delineation of the command decision authority of the shift supervisor in the control room relative to other plant management personnel. Particular emphasis shall be placed on the following:

   a. The responsibility and authority of the shift supervisor shall be to maintain the broadest perspective of operational conditions affecting the safety of the plant as a matter of highest priority at all times when on duty in the control room. The idea shall be reinforced that the shift supervisor should not become totally involved in any single operation in times of emergency when multiple operations are required in the control room.

   b. The shift supervisor, until properly relieved, shall remain in the control room at all times during accident situations to direct the activities of control room operators. Persons authorized to relieve the shift supervisor shall be specified.

   c. If the shift supervisor is temporarily absent from the control room during routine operations, a lead control room operator shall be designated to assume the control room command function. These temporary duties, responsibilities, and authority shall be clearly specified.

3. Training programs for shift supervisors shall emphasize and reinforce the responsibility for safe operation and the management function the shift supervisor is to provide for assuring safety.

4. The administrative duties of the shift supervisor shall be reviewed by the senior officer of each utility responsible for plant operations. Administrative functions that detract from or are subordinate to the management responsibility for assuring the safe operation of the plant shall be delegated to other operations personnel not on duty in the control room.

1274 042

(Section 2.2.1.a)

RESPONSE

1. The Vice President - Nuclear of Boston Edison will issue and periodically reissue a management directive that emphasizes the primary management responsibility of the Nuclear Watch Engineer for safe operation of the plant under all conditions on his shift and that clearly establishes his command duties.

2. Plant procedures will be developed and reviewed to assure that the duties, responsibilities, and authority of the Nuclear Watch Engineer, Operations Supervisor and control room operators are properly defined to affect the establishment of a definite line of command and clear definition of the command decision authority of the Nuclear Watch Engineer in the control room relative to other plant management personnel. Particular emphasis will be placed on the following:

   a. The responsibility and authority of the Nuclear Watch Engineer will be to maintain the broadest perspective of operational conditions affecting the safety of the plant as a matter of highest priority at all times when on duty in the control room. The idea will be reinforced that the Nuclear Watch Engineer should not become totally involved in any single operation in times of emergency when multiple operations are required in the control room.

   b. The Nuclear Watch Engineer, until properly relieved, will remain in the control room at all times during accident situations to direct the activities of control room operators. Persons authorized to relieve the Nuclear Watch Engineer will be specified.

   c. If the Nuclear Watch Engineer is temporarily absent from the control room during operations, an Operations Supervisor will be designated to assume the control room command function. These temporary duties, responsibilities, and authority will be clearly specified.

3. Training programs for Nuclear Watch Engineers and Operations Supervisors will emphasize and reinforce the responsibility for safe operation and that the management function of the Nuclear Watch Engineer is to provide for assuring safety.

4. The administrative duties of the Nuclear Watch Engineer will be reviewed by the VP-Nuclear to assure that administrative functions that detract from or are subordinate to the Nuclear Watch Engineer's management responsibility for assuring the safe and reliable operation of the plant are delegated to personnel who are not directly responsible for control room operations.

1274 043

1C.18 **Shift Technical Advisor (Section 2.2.1.b)**

POSITION

Each licensee shall provide an on-shift technical advisor to the shift supervisor. The shift technical advisor may serve more than one unit at a multi-unit site if qualified to perform the advisor function for the various units.

The shift technical advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline and have received specific training in the response and analysis of the plant for transients and accidents. The shift technical advisor shall also receive training in plant design and layout, including the capabilities of instrumentation and controls in the control room. The licensee shall assign normal duties to the shift technical advisors that pertain to the engineering aspects of assuring safe operations of the plant, including the review and evaluation of operating experience.

RESPONSE

Pilgrim Unit 2 will comply with the necessary staffing requirements at the time of operating license issuance.

The information submitted in the FSAR will include details of how the following will be achieved.

1. Personnel capable of accident assessment based on comprehensive education in engineering and science subjects related to nuclear power plant design and on training and experience in the dynamic response of Pilgrim Unit 2 will be rapidly available in the control room.

2. An engineering group with diverse technical knowledge, experience and perspective in relevant areas such as electrical, mechanical, fluid systems and human factors will be available to maintain and upgrade safe plant operations through the cognizance and evaluation of applicable operating experience.

1274 044

1C.19 Shift and Relief Turnover Procedures (Section 2.2.1.c)

POSITION

The licensees shall review a revise as necessary the plant procedure for shift and relief turnover to sure the following:

1. A checklist shall be provided for the oncoming and offgoing control room operators and the oncoming shift supervisor to complete and sign. The following items, as a minimum, shall be included in the checklist:

   a. Assurance that critical plant parameters are within allowable limits (parameters and allowable limits shall be listed on the checklist).

   b. Assurance of the availability and proper alignment of all systems essential to the prevention and mitigation of operational transients and accidents by a check of the control console (what to check and criteria for acceptable status shall be included on the checklist);

   c. Identification of systems and components that are in a degraded mode of operation permitted by the Technical Specifications. For such systems and components, the length of time in the degraded mode shall be compared with the Technical Specifications action statement (this shall be recorded as a separate entry on the checklist).

2. Checklists or logs shall be provided for completion by the offgoing and oncoming auxiliary operators and technicians. Such checklists or logs shall include any equipment under maintenance of test that by themselves could degrade a system critical to the prevention and mitigation of operational transients and accidents or initiate an operational transients (what to check and criteria for acceptable status shall be included on the checklist); and

3. A system shall be established to evaluate the effectiveness of the shift and relief turnover procedure (for example, periodic independent verification of system alignments).

1274 045

(Section 2.2.1.c)

RESPONSE

Plant procedures for shift and relief turnover will include the following:

1. A checklist will be provided for the oncoming and offgoing control room operators and the oncoming shift supervisor to complete and sign. The following items will be included in the checklist:

    a. Assurance that critical plant parameters are within allowable limits (parameters and allowable limits will be listed on the checklist);

    b. Assurance of the availability and proper alignment of all systems essential to the prevention and mitigation of operational transients and accidents by a check of the control console (what to check and criteria for acceptable status will be included on the checklist);

    c. Identification of systems and components that are in a degraded mode of operation permitted by the Technical Specifications. For such systems and components, the length of time in the degraded mode will be compared with the Technical Specification action statement (this will be recorded as a separate entry of the checklist).

2. Checklists or logs will be provided for completion by the offgoing and oncoming auxiliary operators and technicians. Such checklists or logs will include any equipment under maintenance or test that by themselves could degrade a system critical to the prevention and mitigation of operational transients and accidents or initiate operational transients (what to check and criteria for acceptable status will be included on the checklist); and

3. A system will be established to evaluate the effectiveness of the shift and relief turnover procedure (for example, periodic independent verification of system alignments).

1274 046

1C.20 Control Room Access (Section 2.2.2.a)

POSITION

The licensee shall make provisions for limiting access to the control room to those individuals responsible for the direct operation of the nuclear power plant (e.g., operations supervisor, shift supervisor, and control room operators), to technical advisors who may be requested or required to support the operation, and to predesignated NRC personnel. Provisions shall include the following:

1. **Develop and implement an administrative procedure that establishes the authority and responsibility of the person in charge of the control room to limit access.**

2. **Develop and implement procedures that establish a clear line of authority and responsibility in the control room in the event of an emergency. The line of succession for the person in charge of the control room shall be established and limited to persons possessing a current senior reactor operator's license. The plan shall clearly define the lines of communication and authority for plant management personnel not in direct command of operations, including those who report to stations outside of the control room.**

RESPONSE

Provisions will be made for limiting access to the control room to those individuals responsible and necessary for the direct operation of Pilgrim Unit 2, including the Chief Operating Engineer, Nuclear Watch Engineer, Operations Supervisor, control room operators, and technical advisors or other specialists who may be requested or required to support the operation, and to predesignated NRC personnel. Provisions will include the following:

1. An administrative procedure that establishes the authority and responsibility of the person in charge of the control room to limit access; and

2. Procedures that establish a clear line of authority and responsibility in the control room in the event of an emergency. The line of succession for the person in charge of the control room will be established and limited to persons possessing a current senior reactor operator's license. The plan will clearly define the lines of communication and authority for plant management personnel not in direct command of operations, including those who report to stations outside of the control room.

1274 047

1C.21 Onsite Technical Support Center (Section 2.2.2.b)

POSITION

Each operating nuclear power plant shall maintain an onsite technical support center separate from and in close proximity to the control room that has the capability to display and transmit plant status to those individuals who are knowlegeable of and responsible for engineering and management support of reactor operations in the event of an accident. The center shall be habitable to the same degree as the control room for postulated accident conditions. The licensee shall revise his emergency plans as necessary to incorporate the role and location of the technical support center.

Records that pertain to the as-built conditions and layout of structures, systems and components shall be stored and filed at the site and accessible to the technical support center under emergency conditions. Examples of such records include system descriptions, general arrangement drawings, piping and instrument diagrams, piping system isometrics, electrical schematics, wire and cable lists, and single line electrical diagrams. It is not the intent that all records described in ANSI N45.2.9-1974 be stored and filed at the site and accessible to the technical support center under emergency conditions; however, as stated in that standard, storage systems shall provide for accurate retrieval of all pertinent information without undue delay.

RESPONSE

An onsite technical support center for Pilgrim Unit 2 will be provided, separate from and in close proximity to the control room that has the capability to display and transmit plant status to those individuals who are knowlegeable of and responsible for engineering and management support of reactor operations in the event of an accident. The center will be habitable to the same degree as the control room for postulated accident conditions to enable the on-site technical support function to be accomplished. Emergency plans will describe the role and location of the technical support center.

Records that pertain to the as-built conditions and layout of structures, systems and components will be stored and filed at the site and accessibile to the technical support center under emergency conditions. Examples of such records include system descriptions, general arrangement drawings, piping and instrument diagrams, piping system isometrics, electrical schematics, wire and cable lists, and single line electrical diagrams. It is not the intent that all records described in ANSI N45.2.9-1974 be stored and filed at the site and accessible to the technical support center under emergency conditions; however, storage systems will provide for accurate retrieval of all pertinent information without undue delay.

. 1274 048

1C.22 Onsite Operational Support Center (Section 2.2.2.c)

POSITION

An area to be designated as the onsite operational support center shall be established. It shall be separate from the control room and shall be the place to which the operations support personnel will report in an emergency situation. Communications with the control room shall be provided. The emergency plan shall be revised to reflect the existence of the center and to establish the methods and lines of communication and management.

Response:

An area to be designated as the onsite operational support center will be established. It will be separate from the control room and will be the place to which the operations support personnel will report in an emergency situation. Communications with the control room will be provided. The emergency plan will reflect the existance of the center and will establish the methods and lines of communication and management.

1274 049

1C.23 Revised Limiting Conditions for Operation of Nuclear Power Plants Based Upon Safety System Availability (Section 2.2.3)

POSITION

All NRC nuclear power plant licensees shall provide information to define a limiting operational condition based on a threshold of complete loss of safety function. Identification of a human or operational error that prevents or could prevent the accomplishment of a safety function required by NRC regulations and analyzed in the license application shall require placement of the plant in a hot shutdown condition within 8 hours and in a cold shutdown condition within 24 hours.

The loss of operability of a safety function shall include consideration of the necessary instrumentation, controls, emergency electrical power sources, cooling or seal water, lubrication, operating procedures, maintenance procedures, test procedures and operator interface with the system, which must also be capable of performing their auxiliary or supporting functions. The limiting conditions for operation shall define the minimum safety functions for modes 1, 2, 3, 4, and 5 of operation.

The limiting conditions of operation shall require the following:

1. If the plant is critical, restore the safety function (if possible) and place the plant in a hot shutdown condition within 8 hours.

2. Within 24 hours, bring the plant to cold shutdown.

3. Determine the cause of the loss of operability of the safety function. Organizational accountability for the loss of operability of the safety system shall be established.

4. Determine corrective actions and measures to prevent recurrence of the specific loss of operability for the particular safety function and generally for any safety function.

5. Report the event within 24 hours by telephone and confirm by telegraph, mailgram, or facsimile transmission to the Director of the Regional Office, or his designee.

6. Prepare and deliver a Special Report to the NRC's Director of Nuclear Reactor Regulation and to the Director of the appropriate regional office of the Office of Inspection and Enforcement. The report shall contain the results of steps 3 and 4, above, along with a basis for allowing the plant to return to power operation. The senior corporate executive of the licensee responsible and accountable for safe plant operation shall deliver and discuss the contents of the report in a public meeting with the Office of Nuclear Reactor Regulation and the Office of Inspection and Enforcement at a location to be chosen by the Director of Nuclear Reactor Regulation.

7. A finding of adequacy of the licensee's Special Report by the Director of Nuclear Reactor Regulation will be required before the licensee returns the plant to power.

1274 050

(Section 2.2.3)

RESPONSE

Boston Edison will participate in the rulemaking procedures and will comply with the results of that process as applicable to Pilgrim Unit 2.

1274 051

1C.24 Instrumentation to Monitor Containment Conditions During the Course of an Accident

POSITION

Consistent with satisfying the requirements set forth in General Design Criterion 13 to provide the capability in the control room to ascertain containment conditions during the course of an accident, the following requirements shall be implemented:

1. A continuous indication of containment pressure shall be provided in the control room. Measurement and indication capability shall include three times the design pressure of the containment for concrete, four times the design pressure for steel, and minus five psig for all containments.

2. A continuous indication of hydrogen concentration in the containment atmosphere shall be provided in the control room. Measurement capability shall be provided over the range of 0 to 10% hydrogen concentration under both positive and negative ambient pressure.

3. A continuous indication of containment water level shall be provided in the control room for all plants. A narrow range instrument shall be provided for PWRs and cover the range from the bottom to the top of the containment sump. Also for PWRs, a wide range instrument shall be provided and cover the range from the bottom of the containment to the elevation equivalent to a 500,000 gallon capacity. For BWRs, a wide range instrument shall be provided and cover the range from the bottom to 5 feet above the normal water level of the suppression pool.

The containment pressure, hydrogen concentration and wide range containment water level measurements shall meet the design and qualification provisions of Regulatory Guide 1.97, including qualification, redundancy, and testability. The narrow range containment water level measurement instrumentation shall be qualified to meet the requirements of Regulatory Guide 1.89 and shall be capable of being periodically tested.

RESPONSE:

Consistent with satisfying the requirements set forth in General Design Criterion 13 to provide the capability in the control room to ascertain containment conditions during the course of an accident, the following requirements will be implemented:

1274 052

1. A continuous indication of containment pressure will be provided in the control room. Measurement and indication capability will include three times the design pressure of the containment and minus five psig.

2. A continuous indication of hydrogen concentration in the containment atmosphere will be provided in the control room. Measurement capability will be provided over the range of 0 to 10% hydrogen concentration under both positive and negative ambient pressure.

3. A continuous indication of containment water level will be provided in the control room. A narrow range instrument will be provided and cover the range from the bottom to the top of the containment liquid waste collection sumps. A wide range instrument will be provided and cover the range from the bottom of the containment to the elevation equivalent to at least a 500,000 gallon capacity.

The containment pressure, hydrogen concentration and wide range containment water level measurements will meet the design and qualification provisions of Regulatory Guide 1.97 (Rev. 1), including qualification, redundancy, and feasibility. The narrow range containment water level measurement instrumentation will be qualified to meet the requirements of Regulatory Guide 1.89 (Rev. 0) and will be capable of being periodically tested.

1274 053

1C.25 Installation of Remotely Operated High Point Vents in the Reactor Coolant System

POSITION

Each applicant and licensee shall install reactor coolant system and reactor vessel head high point vents remotely operated from the control room. Since these vents form a part of the reactor coolant pressure boundary, the design of the vents shall conform to the requirements of Appendix A to 10 CFR Part 50 General Design Criteria. In particular, these vents shall be safety grade, and shall satisfy the single failure criterion and the requirements of IEEE-279 in order to ensure a low probability of inadvertent actuation.

Each applicant and licensee shall provide the following information concerning the design and operation of these high point vents:

1. A description of the construction, location, size, and power supply for the vents along with results of analyses of loss-of-coolant accidents initiated by a break in the vent pipe. The results of the analyses should be demonstrated to be acceptable in accordance with the accetpance criteria of 10 CFR 50.46.

2. Analyses demonstrating that the direct venting of noncondensable gases with perhaps high hydrogen concentrations does not result in violation of combustible gas concentration limits in containment as described in 10 CFR Part 50.44, Regulatory Guide 1.7 (Rev. 1) and Standard Review Plan Section 6.2.5.

3. Procedural guidelines for the operators' use of the vents. The information available to the operator for initiating or terminating vent usage shall be discussed.

RESPONSE

The Pilgrim Unit 2 design will provide reactor coolant system(pressurizer and reactor vessel head)high point vents remotely operated from the control room. The design of the vents will conform to the requirements of Appendix A to 10 CFR Part 50 General Design Criteria. In particular, these vents will be safety grade, and will satisfy the single failure criterion and the requirements of IEEE Std. 279-1971 in order to ensure a low probability of inadvertent actuation.

The following information concerning the design and operation of the high point vents will be provided in the FSAR.

1. A description of the construction, location, size, and power supply for the vents along with results of analyses of loss-of-coolant accidents initiated by a break in the vent pipe. The results of the analyses will be demonstrated to be acceptable in accordance with the acceptance criteria of 10 CFR 50.46.

1274 054

2. Analyses demonstrating that the direct venting of noncondensable gases with perhaps high hydrogen concentrations does not result in violation of combustible gas concentration limits in containment as described in 10 CFR Part 50.44, Regulatory Guide 1.7 (Rev. 1) and Standard Review Plan Section 6.2.5 (Rev. 1).

3. Procedural guidelines for the operators' use of the vents. The information available to the operator for initiating or terminating vent usage will be discussed.

1274 055

1C.26 Near Term Requirements for Improving Emergency Preparedness

(1) Element:

Upgrade licensee emergency plans to satisfy Regulatory Guide 1.101, with special attention to the development of uniform action level criteria based on plant parameters.

(1) Response:

The Pilgrim Unit 2 emergency plan will be based on and will satisfy Regulatory Guide 1.101. Special attention will be given to the development of uniform action level criteria based on plant parameters.

(2) Element:

Assure the implementation of the related recommendations of the NRR Lessons Learned Task Force involving instrumentation to follow the course of an accident and relate the information provided by this instrumentation to the emergency plan action levels. This will include instrumentation for post-accident sampling, high range radioactivity monitors, and improved in-plant radioiodine instrumentation. The implementation of the Lessons Learned recommendation on instrumentation for detection of inadequate core cooling will also be factored into the emergency plan action level criteria.

(2) Response:

The Lessons Learned Task Force recommendations in NUREG-0578 involving instrumentation to follow the course of an accident will be implemented as described elsewhere in this Appendix. The information provided by this instrumentation will be related to the emergency plan action levels. This will include instrumentation for post accident sampling, high range radioactivity monitors, and improved in-plant radioiodine instrumentation. The implementation of the Lessons Learned recommendation on instrumentation for detection of inadequate core cooling will also be factored into the emergency plan action level criteria.

1274 056

(3) Element:

Determine that an Emergency Operations Center for Federal, State and local personnel has been established with suitable communications to the plant, and that upgrading of the facility in accordance with the Lessons Learned recommendation for an in-plant technical support center is underway.

(3) Response:

An Emergency Operations Center will be established for Federal, State and local personnel with suitable communications between the plant and the Emergency Operations Center. As indicated in Section 1C.21, an in-plant technical support center will be provided.

(4) Element:

Assure that improved licensee offsite monitoring capabilities (including additional TLD's or equivalent) have been provided for all sites.

(4) Response:

Offsite monitoring capabilities for Pilgrim Unit 2 will comply with the requirements in effect at the operating license stage.

(5) Element:

Assess the relationship of State/local plans to the licensee's and Federal plans so as to assure the capability to take appropriate emergency actions. Assure that this capability will be extended to a distance of 10 miles as soon as practical but not later than January 1, 1981. This item will be performed in conjunction with the Office of State Programs and the Office of Inspection and Enforcement.

(5) Response:

Boston Edison is currently cooperating with the Commonwealth of Massachusetts in its development of an emergency action plan out to a radius of 10 miles from Pilgrim Station. It is our understanding that the Commonwealth of Massachusetts will submit documentation of this plan to the NRC to enable implementation prior to 1/1/81.

(6) Element:

Require test exercises of approved Emergency Plans (Federal, State, local, licensees), review plans for such exercises, and participate in a limited number of joint exercises. Tests of licensee plans will be required to be conducted as soon as practical for all facilities and before reactor startup for new licensees. Exercises of State plans will be performed in conjunction with the concurrence reviews of the Office of State Programs. Joint text exercises involving Federal, State local and licensees will be conducted at the rate of about 10 per year, which would result in all sites being exercised once each five years.

1274 057

(6) Response:

The applicants will participate in test exercises of approved Emergency Plans (Federal, State, local, licensees). We will participate in reviews of plans for such exercises, and participate in joint exercises. We will cooperate in any tests conducted before reactor startup. We will participate in exercises of State plans to be performed in conjunction with the concurrence reviews of the Office of State Programs.

1C.27 Response to IE Bulletin 79-06B (As Modified by Bulletin 79-06C)

The following additional information provides the NRC with the necessary assurance that Pilgrim Unit 2 will meet the intent of IE Bulletin 79-06B, "Review of Operational Errors and System Misalignments Identified During the Three Mile Island Incident," April 14, 1979. While the commitments to each applicable position are addressed below, the implementation details will be described more fully in the FSAR. IE Bulletin 79-06B requires action to be taken by licensees of operating Combustion Engineering designed light water reactors. A review of this bulletin has indicated that, although some areas of concern are not immediately applicable to Pilgrim Unit 2 at the Construction Permit stage, it is appropriate to identify and commit to necessary future actions. The following responses address each item of Bulletin 79-06B (as modified by Bulletin 79-06C in Item #6C):

ITEM:

1. Review the description of circumstances described in Enclosure 1 of IE Bulletin 79-05 and the preliminary chronology of the TMI-2 3/28/79 accident included in Enclosure 1 to IE Bulletin 79-05A.

   a. This review should be directed toward understanding: (1) the extreme seriousness and consequences of the simultaneous blocking of both auxiliary feedwater trains at the Three Mile Island Unit 2 plant and other actions taken during the early phases of the accident; (2) the apparent operational errors which led to the eventual core damage; (3) that the potential exists, under certain accident or transient conditions, to have a water level in the pressurizer simultaneously with the reactor vessel not full of water; and (4) the necessity to systematically analyze plant conditions and parameters and take appropriate corrective action.

   b. Operational personnel should be instructed to: (1) not override automatic action of engineered safety features unless continued operation of engineered safety features will result in unsafe plant conditions (see Section 6a); and (2) not make operational decisions based solely on a single plant parameter indication when one or more confirmatory indications are available.

   c. All licensed operators and plant management and supervisors with operational responsibilities shall participate in this review and such participation shall be documented in plant records.

1274 059

RESPONSE:

1a. These issues and concerns will be incorporated into the training program for the Pilgrim Unit 2 operating staff.

1b. The operator training program for Pilgrim Unit 2 will emphasize the key role of the operator in reactor safety and will implement the requirements of item 1b.

1c. Licensed operators and plant management and supervisors with operational responsibilities will participate in the efforts described in 1a and 1b.

ITEM:

2. Review the actions required by your operating procedures for coping with transients and accidents, with particular attention to:

   a. Recognition of the possibility of forming voids in the primary coolant system large enough to compromise the core cooling capability, especially natural circulation capability.

   b. Operator action required to prevent the formation of such voids.

   c. Operator action required to enhance core cooling in the event such voids are formed. (e.g., remote venting)

RESPONSE:

2. Operating procedures for coping with transients and accidents will be developed with particular attention to:

   a. Recognition of the possibility of forming voids in the primary coolant system large enough to compromise the core cooling capability, especially natural circulation capability.

   b. Operator action required to prevent the formation of such voids.

   c. Operator action required to enhance core cooling in the event such voids are formed. (e.g., remote venting)

   This is also addressed in our responses to Recommendations 2.1.3b and 2.1.9b of NUREG-0578 in Sections 1C.4 and 1C.16 of this Appendix.

1274 060

ITEM:

3. Review the containment isolation initiation design and procedures, and prepare and implement all changes necessary to permit containment isolation whether manual or automatic, of all lines whose isolation does not degrade needed safety features or cooling capability, upon automatic initiation of safety injection.

RESPONSE:

3. As addressed in our response to Recommendation 2.1.4 of NUREG-0578 in Section 1C.5 the Pilgrim Unit 2 design will conform to the containment isolation initiation design requirements. Implementing procedures and operator training will be provided accordingly.

ITEM:

4. For facilities for which the auxiliary feedwater system is not automatically initiated, prepare and implement immediately procedures which require the stationing of an individual (with no other assigned concurrent duties and in direct and continuous communication with the control room) to promptly initiate adequate auxiliary feedwater to the steam generator(s) for those transients or accidents the consequences of which can be limited by such action.

RESPONSE:

4. Pilgrim Unit 2 will provide automatic initiation of emergency feedwater as described in our response to Recommendation 2.1.7 of NUREG-0578 in Section 1C.11, therefore this item is not applicable.

ITEM:

5. For your facilities, prepare and implement immediately procedures which;

   a. Identify those plant indications (such as valve discharge piping temperature, valve position indication, or valve discharge relief tank temperature or pressure indication) which plant operators may utilize to determine that pressurizer power operated relief valve(s) are open, and

   b. Direct the plant operators to manually close the power operated relief block valve(s) when reactor coolant system pressure is reduced to below the set point for normal automatic closure of the power operated relief valve(s) and the valve(s) remain stuck open.

1274 061

RESPONSE:

5. Procedures will be developed identifying the indications of an open pressurizer power operated relief valve (PORV) including the use of instrumentation described in our response to Recommendation 2.1.3.a or NUREG-0578 in Section 1C.3. Procedures will be developed to cope with the event of a PORV stuck open below the normal automatic closure setpoint.

ITEM:

6. Review the action directed by the operating procedures and training instructions to ensure that:

   a. Operators do not override automatic actions of engineered safety features, unless continued operation of engineered safety features will result in unsafe plant conditions. For example, if continued operation of engineered safety features would threaten reactor vessel integrity then the HPI should be secured (as noted in b(2) below).

   b. Operating procedures currently, or are revised to, specify that if the high perssure injection (HPC) system has been automatically actuated because of low pressure condition, it must remain in operation until either:

      (1) Both low pressure injection (LPI) pumps are in operation and flowing for 20 minutes or longer; at a rate which would assure stable plant behavior; or

      (2) The HPI system has been in operation for 20 minutes, and all hot and cold leg temperatures are at least 50 degrees below the saturation temperature for the existing RCS pressure. If 50 degrees subcooling cannot be maintained after HPI cutoff, the HPI shall be reactivated. The degree of subcooling beyond 50 degrees F and the length of time HPI is in operation shall be limited by the pressure/temperature considerations for the vessel integrity.

   c. Operating procedures currently, or are revised to, specify that in the event of HPI initiation with reactor coolant pumps (RCP) operating, at least one RCP shall remain operating in each loop as long as the pump(s) is providing forced flow.

   d. Operators are provided additional information and instructions to not rely upon pressurizer level indication alone, but to also examine pressurizer pressure and other plant parameter indications in evaluating plant conditions, e.g., water, inventory in the reactor primary system.

1274 062

RESPONSE:

6. a) Procedures will be developed to insure that operators do not override automatic actions of engineered safety features unless continued operation will result in unsafe plant conditions.

   b) The HPSI System will not be secured unless hot and cold leg temperature indications are at least 50 degrees below the saturation temperature for the existing reactor coolant system pressure. If 50 degrees subcooling cannot be maintained after the HPSI System is secured the HPSI System will be reactivated.

   c) Reactor coolant pump operation in the event of safety injection will be based upon maximizing reactor safety. (This response is consistent with IE Bulletin 79-06C dated July 26, 1979).

   d) The operator training program will emphasize use of confirmatory indications in the implementation of our response to Item 1.f(2) above.

ITEM:

7. Review all safety-related valve positions, positioning requirements and positive controls to assure that valves remain positioned (open or closed) in a manner to ensure the proper operation of engineered safety features. Also review related procedures, such as those for maintenance, testing, plant and system startup, and supervisory periodic (e.g., daily/shift checks,) surveillance to ensure that such valves are returned to their correct positions following necessary manipulations and are maintained in their proper positions during all operational modes.

RESPONSE:

7. Safety-related valve positions,positioning requirements and positive controls will be established such that assurance will be provided that valves (including locked valves) remain positioned (open or closed) in a manner to ensure the proper operation of engineered safety features. Related procedures, such as those for maintenance, testing, plant and system startup, and supervisory period (e.g., daily/shift checks) surveillance will be provided to insure that such valves are returned to their correct positions following necessary manipulations and are maintained in their proper positions during all operational modes.

ITEM:

8. Review your operating modes and procedures for all systems designed to transfer potentially radioactive gases and liquids out of the primary containment to assure that undesired pumping, venting or other releases of radioactive liquids and gases will not occur inadvertently.

1274 063

In particular, ensure that such an occurrence would not be caused by the resetting of engineered safety features instrumentation. List all such systems and indicate:

a. Whether interlocks exist to prevent transfer when high radiation indication exists, and

b. Whether such systems are isolated by the containment isolation signal.

c. The basis on which continued operability of the above features is assured.

RESPONSE:

8. The Pilgrim Unit 2 design and procedures will provide the necessary control to assure that the transfer of radioactive fluids or gases out of containment will not occur inadvertently.

ITEM:

9. Review and modify as necessary your maintenance and test procedures to ensure that they require:

a. Verification, by test or inspection, of the operability of redundant safety-related systems prior to the removal of any safety-related system from service.

b. Verification of the operability of all safety-related systems when they are returned to service following maintenance or testing.

c. Explicit notification of involved reactor operational personnal whenever a safety-related system is removed from and returned to service.

RESPONSE:

9. Maintenance and test procedures will require verification, by test or inspection, of the operability of redundant safety-related systems prior to their removal from service, verification of the operability of all safety-related systems when they are returned to service following maintenance or testing, and the explicit notification of appropriate personnel of the change in status of those safety-related systems.

ITEM:

10. Review your prompt reporting procedures for NRC notification to assure that NRC is notified within one hour of the time the reactor is not in a controlled or expected condition of operation. Further, at that time an open continuous communication channel shall be established and maintained with NRC.

1274 064

RESPONSE:

10. Pilgrim Unit 2 prompt reporting procedures will provide for NRC notification within one hour and for establishing and maintaining an open communication channel with the NRC.

ITEM:

11. Review operating modes and procedures to deal with significant amounts of hydrogen gas that may be generated during a transient or other accident that would either remain inside the primary system or be released to the containment.

RESPONSE:

11. Operating modes and procedures for dealing with significant amounts of hydrogen gas either inside the primary system or the containment will be established.

ITEM:

12. Propose changes, as required, to those technical specifications which must be modified as a result of your implementing the above items.

RESPONSE:

12. Technical specifications will be prepared and submitted with the FSAR considering the responses given above.

1274 065

1C.28 Additional Commitment

The capability to monitor core conditions using in-core thermocouples will not be precluded from the design.

1274 066