12.3 SHIELDING AND RADIATION PROTECTION

12.3.1 Design Basis

12.3.1.1 Radiation Exposure of Individuals

The basis for the radiation shielding design for normal operation is 10 CFR 20.

Design objective dose rates for all normally accessible areas of the station are a maximum whole body exposure of 5 rem per calendar year. For all areas outside the controlled area boundary, the allowable dose rate is not more than 0.1 rem to any individual for one calendar year. All areas of the station are subject to these design objectives. The areas are zoned according to their expected radiation exposure level under normal operating conditions.

For compliance to 10 CFR 20, the controlled area is defined as that part of the PNPS property that is outside the restricted areas and between Rocky Hill Road and the shoreline. Thus, the controlled area boundary east and west correspond to the property lines and the controlled area boundary south runs along Rocky Hill Road. These boundaries are shown in Figure 1.6-3.

For the design basis accident, the station design is based on the guideline values of 10 CFR 100. For continued control room occupancy during the design basis accident, the shielding design is based on a whole body exposure of less than 0.5 rem in any 8 hour period.

12.3.1.2 Radiation Exposure of Materials and Components

Materials and components are selected on the basis that radiation exposure as a result of the shielding design will not cause significant changes in their physical properties which adversely affect operation of equipment during their design life. Materials for equipment required to operate under accident conditions are selected on the basis of the additional exposure received in the event of a design basis accident. The following general radiation exposure limits were considered in the selection of materials:

Material	Approximate Damage Threshold
Teflon	$\geq 1.0 \times 10^{4}$ Rads
Most thermoplastic resins and elastomers	$\geq 1.0 \times 10^{6}$ Rads
Some thermoplastics	$\geq 1.0 \times 10^{7}$ Rads
Ceramics	$\geq 1.0 \times 10^{10}$ Rads
Metals	$\geq 1.0 \times 10^{11}$ Rads

12.3.2 Radiation Zoning and Access Control

The areas inside the station structures as well as the general yard areas are all identified by different radiation zones.

Restricted Areas

In compliance with 10 CFR 20.1003 certain areas at the Pilgrim Nuclear Power Station are identified as restricted areas. All restricted areas are within property owned by the Licensee. The restricted area is any area to which access is limited by the Licensee for the purposes of protecting individuals against undue risk from exposure to radiation or radioactive materials. The PNPS restricted area includes:

- the structures and yard areas enclosed by the Protected Area Fence described in the PNPS Security Plan;
- the Trash Compaction Facility (TCF) and the yard area within the TCF fence;
- the Main Stack and Filter building.

The occupancy of the restricted areas is controlled by the plant access control plan. Visitors will be allowed to enter a restricted area only when properly escorted.

Radiation Areas

In compliance with 10 CFR 20.1003, certain areas at the Pilgrim Nuclear Power Station are identified, enforced, and controlled as radiation areas for the safe and efficient operation of the station. All radiation areas are located within the restricted area.

All accessible station areas that either experience or are expected to experience radiation fields at 30 cm from the radiation source or from any surface that the radiation penetrates in excess of 5 mrem in any hour, but less than 100 mrem in any hour are identified as radiation areas. Identification is by posting a sign, or signs, at strategic locations bearing the radiation caution symbol and the words "Caution, Radiation Area."

The general station layout and the use of personnel barriers such as doors, gates, chains, etc., enforce the controlled safe entry into these areas.

High Radiation Areas

In compliance with 10 CFR 20.1003, certain areas at the Pilgrim Nuclear Power Station are identified, enforced, and controlled as high radiation areas for the safe and efficient operation of the station. All high radiation areas are located within the restricted areas.

All station areas that either experience or are expected to experience radiation fields at 30 cm from the radiation source or from any surface that the radiation penetrates in excess of 100 mrem in any hour, but less than 1000 mrem in any hour, are identified as high radiation areas.

Identification is by posting a sign or signs, at strategic locations bearing the radiation caution symbol and the words, "Caution, High Radiation Area," or the words, "Danger, High Radiation Area."

The entry and occupancy of high radiation areas are enforced by doors, gates, chains, ropes, etc. Personnel entering high radiation areas must be wearing a self-indicating dosimeter and a dosimeter of legal record (e.g., TLD or OSLD) and must be signed on to an appropriate radiation work permit.

Locked High Radiation Areas

High radiation areas with a dose rate of 1000 mrem per hour or more at 30cm from the radiation source or from any surface that the radiation penetrates are locked.

The entry and occupancy of locked high radiation areas are enforced by locked doors or gates. Identification is by posting a sign or signs at strategic locations bearing the radiation symbol and the words "Danger Locked High Radiation Areas".

Very High Radiation Areas

In compliance with 10 CFR 20.1003, certain high radiation areas within the PNPS process building are identified, enforced, and controlled as very high radiation areas.

All accessible station areas that either experience or may experience radiation fields, at 1 meter from the radiation source or from any surface that radiation penetrates, in excess of 500 rad in any hour are identified as very high radiation areas. Identification is by posting a sign or signs at strategic locations bearing the radiation symbol and the words, "Grave Danger, Very High Radiation Area."

The entry and occupancy of very high radiation areas are enforced by enclosures and locked doors which prevent unauthorized access. Anyone entering very high radiation areas is required escort by a qualified Radiation Protection Technician who is in possession of a radiation monitoring instrument.

Station Operation Versus Station Shutdown

The shielding and radiation protection criteria discussed apply to the condition of the station during normal full power operation as well as for the shut down station. Identification of high radiation areas is a continuing function of the station radiation protection staff. Access restrictions in certain areas may be relaxed after shutdown if radiation surveys indicate that such reduction is permissible. Where, due to shutdown activities, radiation exposures might increase, action by the radiation protection staff can result in more restrictive controls and enforcement that would be necessary under full power operation.

Access control and radiation zones based upon design analyses are shown on Figures 12.3-1 through 12.3-5 for the station in operation and Figures 12.3-6 through 12.3-10 for 24 hours after shutdown of the reactor.

12.3.3 Radiation Shielding Description

The shielding design considers three conditions:

1. Normal full power operation. This also includes shielding requirements for certain off normal conditions such as the release of fission products from leaking fuel elements
2. Shutdown. This condition deals mainly with the radioactivity from the subcritical reactor core, with radiation from spent fuel bundles during onsite transfer, and with the residual activity in the reactor coolant and neutron activated materials
3. Design basis accidents. Several postulated design basis accidents have been investigated which release fission products to the general free space of the Reactor Building

The materials used for most of the station shielding is ordinary concrete with a nominal bulk density of 144 lb/ft^3. Wherever cast in place concrete has been replaced by concrete blocks the design assures protection on an equivalent shielding basis. Only in a very few instances has steel or water been utilized as primary shielding materials.

For the power level of 1998 MWt design conditions assumed that the core was operating with a power density of 40.5 watts/cc. At this power level the N-16 coolant activity leaving the pressure vessel is 1.78 X 10^7 MeV/gm-sec. The station shielding is based on an assumed stack release rate of 0.1 Ci/sec after a 30 min holdup time in the Offgas System. Reactor water fission product and activated corrosion product concentrations were assumed to be the maximum values expected: 2.3m μCi/cc, and 0.7m μCi/cc, respectively. These conditions yield maximum shielding conditions in the demineralizers, cleanup systems, and other associated radiation handling facilities.

The shutdown condition assumes that the reactor core has been operating at 1,998 MWt for approximately 1,000 hours. At 1,000 hours, the fission product inventory approaches an equilibrium condition.

The change in core activity inventory resulting from the 10 CFR 50 Appendix K uprate is expected to increase post-accident radiation levels by no more than approximately the percentage increase in power level of 1.5%. The slight increase in the post-accident radiation levels has no significant effect on the plant or the habitability of the

Technical Support Center or the Emergency Operations Facility.
The different areas of radiation shielding are described and listed by specific location or building for convenience.

12.3.3.1 Main Control Room

The design basis accidents define the protection required for the main control room. These accident conditions are described in Section 14. The main control room design was based on the airborne fission product inventory in the Reactor Building following the design basis accidents. For continued control room occupancy during the design basis accident, shielding design is based on a whole body exposure of less than 0.5 rem in any 8 hour period.

12.3.3.2 Reactor Building

The Reactor Building contains four major shielding structures; the reactor sacrificial shield, the drywell biological shield, the main steam pipe chase, and the spent fuel pool.

The sacrificial shield has several shielding functions. It protects certain major portions of the drywell space from excessive nuclear radiation exposures during operation. After shutdown it provides shielding from reactor vessel radiation for personnel engaged in inspection, maintenance, and repair of drywell equipment and components. Also, together with the drywell biological shield, it protects the general Reactor Building work areas. The sacrificial shield concrete is approximately 2 feet 3 inches thick. Several shield blocks in the sacrificial shield were permanently removed to facilitate ISI nozzle inspections.

The drywell biological shield together with the reactor sacrificial shield provide the main protection for the areas surrounding the reactor vessel, and the Primary Coolant and Recirculation Systems.

More than 8 feet of concrete thickness is used to keep the radiation dose rates in the fully accessible Reactor Building work areas to less than 2.5 mrem/hour due to primary radiation from the reactor core.

The main steam line pipe chase, with up to 3 feet 6 inches thick concrete walls, is the connecting shield structure between the Reactor and Turbine Buildings. The chase shielding protects against the very penetrating N-16 gamma radiation which is radiated from the passing steam.

The spent fuel pool contains the highly radioactive spent fuel assemblies, control, and instrument rods. A maximum of 6 feet of concrete thickness is used for radiation protection at the sides and bottom of the storage pool. A minimum cover of 10 feet of water above the fuel assemblies will be

maintained for shielding of plant personnel during fuel storage.

The Reactor Cleanup System, the incore flux monitoring equipment, radwaste equipment, and the reactor internals storage are housed in numerous concrete shielded rooms surrounding the drywell concrete structure. Enclosing these secondary sources of radiation in shielded rooms permits the adjacent areas to be accessible to personnel.

The entrances into the drywell space are well shielded with up to 5 feet 0 inch thick shield plugs at the equipment lock and at the equipment and personnel access lock, both located at elevation 23 feet 0 inch.

12.3.3.3 Turbine Building

Radioactive steam enters the Turbine Building from the reactor. Besides N-16, fission product gases and some radioisotopes are carried over from the reactor water. At least 80 percent of the activity goes to the Offgas System, with the remainder following the condensate and being treated by the condensate demineralizers.

For this reason, radiation shielding is provided around the following areas:

1. Main Steam Lines
2. Primary and Extraction Steam Piping
3. High Pressure and Low Pressure Turbines
4. Moisture Separators
5. Reactor Feedwater System Heaters
6. Main Condenser and Hotwell
7. Air Ejectors and Steam Packing Exhauster
8. Condensate Demineralizer

The front end standard of the main turbine above the turbine operating floor is accessible to personnel when the turbine is in operation. A 3 inch thick steel shield provides protection for limited access for instrument check.

12.3.3.4 Radwaste Building and Trash Compaction Facility (TCF)

All areas for preparing, handling, storing, or shipping the radwaste are shielded to permit controlled access as required for operation of the Radwaste Systems.

The individual radwaste systems have been separated from each other and shielded as much as practical in order to minimize personnel exposure during maintenance and repair of any of the equipment.

12.3.3.5 Administration Buildings

All areas of the Executive Building, the Operations and Maintenance Building, and the Support Building are fully

accessible at all times. Turbine Building shielding will prevent excessive N-16 shine into these Administration Buildings.

12.3.3.6 Main Stack and Offgas Piping

The shielding design for the main stack provides for controlled access at ground level to maintain the filters and instrumentation. A security fence surrounding the stack and the buried offgas piping will keep dose rates at uncontrolled areas to less than 1.0 mrem/hour. See Figure 12.3-11.

12.3.3.7 General Station Yard Areas

Station yard areas which are frequently occupied by plant personnel receive an average radiation field of less than 1.0 mrem/hour. These areas are also surrounded by a security fence and closed off from areas accessible to the general public.

12.3.4 Inspection and Performance Analysis

The normal construction quality control program assures that there are no major defects in the shielding. The control program includes onsite inspection surveys by the shielding designer prior to startup.

After startup, the adequacy of the shielding and the efficiency of the access control are periodically checked by radiation and contamination surveys. Installation of new radiation shielding, replacement of existing shielding with a new shield design or material, or the removal of existing radiation shielding, while in a shutdown condition, may require verification of the design change for adequacy via testing (i.e., radiation measurement mapping) while at various power levels at the time of restart, if deemed necessary by either the Radiation Department (for ALARA and access control purposes), or by the Systems and Safety Analysis Department of Engineering (for radiological engineering design purposes).

The following FSAR figures have been removed from the FSAR. Please refer to the corresponding BECo controlled drawing.

FSAR FIGURE	BECO CONTROLLED DRAWING
12.3-1	A 100
12.3-2	A 101
12.3-3	A 102
12.3-4	A 103
12.3-5	A 104
12.3-6	A 106
12.3-7	A 107
12.3-8	A 108
12.3-9	A 109
12.3-10	A 110
12.3-11	A 111

12.4 RADIOACTIVE MATERIALS SAFETY

12.4.1 Materials Safety Program

Use, Handling, and Storage of Licensed Radioactive Sources

When not in use, all licensed radioactive sources shall be stored in a locked container, cabinet, or room. The sources will normally be stored in the radiochemistry lab and the radiation protection lab.

Other controls for the storage, possession, and use of these sources are presented in PNPS Radiation Protection Operating Procedures and are as follows:

1. Each source and/or source container shall be labeled with a radiation sign and a control number. For each source there will be a Radioactive Source Record Sheet with the following information:

   (a) source control number

   (b) source type

   (c) quantity

   (d) date quantity measurement was made

2. Sources will be controlled by the Radiation Protection Manager (or designee) and will be in a locked container, cabinet, or room when not in use.

3. Sign out logs on which to record the removal of various sources from the assigned storage area and to authorize such removals are provided to the radiation protection lab and radiochemistry lab. The user's signature is required on this record.

4. Sources are to be used, transported, and stored in such a way as to minimize personnel exposure to them. Shielded sources shall be kept in their shielded containers except when they are in use.

5. Each sealed source containing radioactive material either in excess of 100 micro curies of beta and/or gamma emitting material or 5 micro curies of alpha emitting material shall be free of > 0.005 micro curies of removable contamination at all times.

   Each sealed source with removable contamination in excess of the above limit shall be immediately withdrawn from use and:

A. Either decontaminated and repaired, or

B. Disposed of in accordance with Commission Regulations.

6. Each sealed source shall be tested for leakage and/or contamination by:

   A. The licensee, or

   B. Other persons specifically authorized by the Commission or an Agreement State.

   The test method shall have a detection sensitivity of at least 0.005 micro curies per test sample.

7. Each category of sealed sources, excluding startup sources and fission detectors previously subjected to core flux, shall be tested at the frequency described below.

   A. Sources in use - At least once per six months for all sealed sources containing radioactive material:

      1. With a half-life greater than 30 days, excluding Hydrogen 3, and

      2. In any form other than gas.

   B. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous six months. Sealed sources transferred without a certificate indicating the last test date shall be tested prior to being placed into use.

   C. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source.

NOTE - If a vendor or source supplier furnishes a certificate indicating that a test has been made within 6 months, the source need not be tested for 6 months and may be made available for immediate use.

The Radiation Protection Department shall assure compliance with provisions of 10CFR20, 10CFR30 and applicable conditions of the Facility Operating License.

A complete inventory of radioactive materials in possession shall be maintained current at all times by the Radiation Protection Department. All such sources shall be inventoried at intervals not

to exceed 6 months. Reactor Engineering shall maintain a complete inventory of all special nuclear material maintained on site. A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of $\geq$0.005 micro curies of removable contamination.

Records will be kept of all receipts, transfers, disposal, leak test, and other information pertinent to by-product licensed material.

Records required to be maintained for two years:

A. Test results, in units of micro curies, for leak test performed pursuant to Section 12.4.1

B. Record of annual physical inventory verifying accountability of sources on record.

Unsealed sources will be stored in a locked location in the radiochemistry lab.

Liquid sources will be used in accordance with the PNPS radiation protection procedures and general rules of good practice in radioactive material handling when they are unsealed.

Use, Handling and Storage of Nuclear Fuel

The new and used fuel storage facilities are described in FSAR Sections 10.2 and 10.3. The multiplication constants for fuel in both the new and spent fuel racks are specified in the Pilgrim I Technical Specifications.

The new fuel storage vault, the spent fuel storage pool, and other locations on the refueling floor where fuel assemblies will be handled or stored are monitored by a Radiation Monitoring System complying with the requirements of 10 CFR 70.24.

All spent fuel handling shall be with cranes and hoists designed specifically for that purpose. These consist of the 5 ton auxiliary hook of the Reactor Building crane, the refueling platform grapple, the refueling platform auxiliary hoists, and the socket mounted jib cranes.

The probability of fire in the inspection and preparation area will be minimized by restricting the allowable quantities of flammable materials in the area. Solvents, such as acetone, are required for cleaning and will be handled in small quantities.

Access to the refueling floor and to the overhead bridge crane shall be permitted only to authorized personnel. Fuel loading and unloading operations will be performed by qualified personnel (including contractors) under the direct supervision of a licensed Senior Reactor Operator or Senior Reactor Operator restricted to fuel handling. "Qualified" within the context of the training rule (10 CFR 50.120) means training in, and testing of, site specific refueling procedures to include demonstration of equipment operation (reference NRC Information Notice 94-13).

New fuel inspection shall be performed by Pilgrim Station or contractor personnel. Entry to the refueling floor will be restricted by locked gates or doors any time that full time surveillance by guards or authorized personnel is not in effect and fuel handling operations are in progress.

Fuel shall be brought to the refueling floor in metal criticality proof shipping containers. Only one metal container will be opened and placed in an upright position at any one time.

A minimum of nine assemblies in a flooded square arrangement is necessary for a minimum critical array. Therefore, handling only two assemblies and inspecting no more than two additional assemblies at any one time precludes the possibility of criticality during the handling and inspection sequence.

12.4.2 Facilities and Equipment

Laboratory instruments will be provided for measuring alpha, beta, and gamma radiation, and for the analysis of radioactive gaseous, liquid, and solid samples. These will include an instrument for gross beta gamma counting of smear samples used for contamination control, and a multi-channel gamma analyzer for gaseous and liquid samples used for effluent release control.

Portable radiation survey instruments will be available as required for measurement of alpha, beta, gamma, and neutron radiation expected during normal operation, and in emergencies.

12.4.2.1 Method, Frequency, and Standards Used in Calibrating Instruments

All beta-gamma instruments will be calibrated in accordance with PNPS Radiation Protection Procedures. These calibrations will employ a calibration unit with an appropriate calibration source.

Alpha instruments will be calibrated in accordance with PNPS Radiation Protection Procedures using an appropriate alpha source set.

The neutron instruments will be calibrated in accordance with PNPS Radiation Protection Procedures using an appropriate calibration source. In addition, neutron instruments are response checked prior to use.

12.4.2.2 Dosimeters and Bio-Assay Procedures Used

Personal monitoring devices, i.e., TLDs, OSLDs, electronic dosimeters, and pocket dosimeters, will be furnished to and worn by personnel who require radiation dose monitoring in accordance with PNPS station procedures.

Bio-Assay: Whole body counting is normally done onsite by the Licensee. A licensed off-site facility is available for contingency in whole body counting or analysis of in vitro bio-essay materials.

12.4.3 Personnel and Procedures

The Reactor Engineering Superintendent is the custodian of special nuclear materials received, possessed, used, or transferred under authorization of Operating License DPR-35.

The Radiation Protection Manager is the custodian of by-product and source materials received, possessed, used, or transferred under the authorization of Operating License DPR-35 and NRC Materials License 20-07626-04, and Commonwealth of Massachusetts Materials License 07-6262. The health physics aspects of the handling, storage, and use of these materials will be administered by the Radiation Protection Manager as defined by ANSI N18.1, 1971.

Radiation protection procedures assure compliance with applicable regulations and appropriate sections of the Operating License, Technical Specifications, and FSAR.

12.4.4 Required Materials

The Licensee is authorized to receive, possess, use, and transfer materials as required for operation of the facility by License No. DPR-35.

12.4.5 Offsite Materials Safety Program

All radioactive materials fixed or contained within reactor system components and shipped to temporary field locations such as vendor facilities will remain in the custody of PNPS, and will be under direct supervision of a qualified PNPS representative normally on the Radiation Protection staff.

Radiation protection activities shall be conducted at the temporary field locations in order to assure that all Pilgrim radioactive reactor components are appropriately packaged, surveyed, and labeled in accordance with applicable NRC/Massachusetts DOT regulations and PNPS radiation protection procedures.

PNPS shall assume responsibility for all radiation protection activities incident to inspection, repair, and testing of Pilgrim equipment containing radioactive material while such equipment is at temporary field locations. These activities shall be conducted in accordance with the requirements of 105 CMR 120, Massachusetts Regulations for the Control of Radiation, at temporary field locations within the borders of Massachusetts, or the requirements of 10 CFR 20, Standards for Protection Against Radiation for temporary field locations outside the borders of Massachusetts but within the borders of the continental United States, as applicable. Radiation monitoring instrumentation and personnel monitoring devices such as those used at Pilgrim Station will be utilized at the offsite location.

The maximum total activity of mixed corrosion products contained within and/or fixed upon the surface of the reactor system components shipped to temporary field locations within the borders of Massachusetts or temporary field locations outside the borders of Massachusetts but within the borders of the continental United States, provided that the state has a reciprocity agreement with Massachusetts shall be limited to the values indicated in Massachusetts Materials License No. 07-6262.

All handling of Pilgrim equipment containing radioactive material at vendor's facilities shall be conducted in such a manner as to preclude the onsite release or disposal of any radioactive materials. All radioactive waste from temporary field stations shall be appropriately packaged, surveyed, and labeled and either returned to Pilgrim Station for ultimate disposal through a licensed contractor, or directly transferred to a licensed waste disposal contractor at the field location.

All vendor company employees shall receive radiation protection orientation prior to their assignment of work on radioactive reactor components in any area controlled by PNPS. The orientation will cover all pertinent radiation protection practices, and procedures to the degree sufficient to allow an employee to perform his assignment without incurring unnecessary radiation exposure.

PNPS shall maintain records of all licensed activities conducted at temporary field locations including records showing the transfer of radioactive materials to and from the location, records of radiation surveys, and records of personnel radiation exposure. A report showing individual radiation exposures shall be furnished to the vendor company upon the completion of licensed activities at the temporary location.

TABLE 12.4-1 deleted

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SECTION 13

CONDUCT OF OPERATIONS

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CONDUCT OF OPERATIONS

13.1 INTRODUCTION AND SUMMARY

The Conduct of Operations was developed to comply with the requirements of the applicable portions of the Code of Federal Regulations.

The Boston Edison Company (BECo) , the original owner and operator of Pilgrim Nuclear Power Station (Pilgrim Station), was responsible for all station operations from the start of preoperational testing and was responsible for providing properly licensed personnel to operate the station. General Electric personnel provided technical guidance for preoperational testing, initial core loading, startup, and precommerical operation. Prior to initial fuel loading, General Electric had a sufficient number of their startup personnel obtain AEC licenses to properly assist BECo licensed personnel during the period preceding commercial operation.

A training program was established and is maintained to provide a properly trained staff of technical, maintenance, and licensed operating personnel to accomplish all of the various station functions within their respective disciplines.

The conduct of operations, as set forth in Sections 13.2 to 13.9 of the original FSAR represented the procedures used during the construction, initial training, and pre-startup phases of Pilgrim Station. Sections 13.4 and 13.5 remained intact in the updated FSAR in order that the NRC may locate previously submitted information related to the pre-operational test and initial startup programs. Where necessary, these sections have been annotated to reflect subsequent changes or additions.

The operation of Pilgrim Station is conducted in accordance with station procedures which are kept current to reflect system design changes, regulatory requirements and improvements based on experience at Pilgrim Station and elsewhere. Of these procedures, those in the following categories which particularly describe methods of operation in normal and emergency conditions are considered to be FSAR related:

Emergency Operating Procedures
Standard Operating Procedures for Safety-Related Equipment/ Systems
Refueling Procedures
Radiation Protection Overview Procedures

A list of the specific procedures in these categories was and is maintained.

13.2 ORGANIZATION AND RESPONSIBILITIES

The Boston Edison Company (BECo) Nuclear Division was originally responsible for the engineering, maintenance, and operation of Pilgrim Nuclear Power Station (Pilgrim Station).

BECo established a project type organization to direct all activities of Pilgrim Station. The organization for operation of the Pilgrim Station was shown on Figure 13.2-1 (deleted). The nuclear division was responsible for the quality assurance, engineering, maintenance, and operations activities. This organization directed the activities of all Boston Edison personnel who worked on the Pilgrim effort and was also responsible for coordinating and supervising the activities of Boston Edison's principal contractors for Pilgrim Station. This organization was responsible for coordinating the testing programs and for approval of test results and acceptance of station operating performance.

The quality assurance function reported to the responsible BECo executive. The nuclear engineering services and regulatory relations organizations reported to the general manager - technical. The plant, nuclear training & management services, and nuclear services organizations reported to the station director. The operation review committee reported to the station director. The nuclear safety review and audit committee reported to the senior vice president - nuclear.

On July 13, 1999, the ownership and authority for the operation of Pilgrim Station was transferred from BECo to the Entergy Nuclear Generation Company (ENGC). On May 5, 2002, the authority for the operation of Pilgrim Station was transferred from ENGC to Entergy Nuclear Operations, Inc. (ENO), with ENGC remaining the owner of Pilgrim Station.

The following are the plant specific titles for personnel fulfilling responsibilities of positions delineated in Technical Specifications.

a. The specified corporate officer at Pilgrim responsible for overall plant nuclear safety is the Site Vice President.

b. The plant manager is the General Manager, Plant Operations.

c. The operations manager is the Manager, Operations.

d. The assistant operations manager is the Assistant Manager, Operations.

e. The control room supervisor is the Control Room Supervisor.

f. The qualified individual that provides advisory technical support to the unit operations shift crew in the areas of engineering and accident analysis is the Shift Control Room Engineer.

g. The radiation protection manager is the Manager, Radiation Protection.

13.2.1 EXECUTIVE MANAGEMENT

Refer to the Quality Assurance Program Manual (QAPM) for discussion of the Entergy Nuclear Operations Incorporated (ENO) Management Structure.

13.2.1.1 Deleted

13.2.1.2 Site Vice President

The Pilgrim Station site vice president reports to the Senior Vice President of Entergy Nuclear Operations.

13.2.1.3 Deleted

13.2.2 MANAGEMENT (GENERAL)

General management is achieved by directors, senior managers, and managers.

13.2.2.1 Directors

Directors head functional areas and report to the site vice president or the executive responsible for the functional area.

13.2.2.2 Deleted

13.2.2.3 Managers

Managers of functional areas report to the director or executive responsible for the functional area.

13.2.2.4 Assistant or Deputy Managers

As necessary, an assistant or deputy manager is assigned to assist the manager.

13.2.2.5 Deleted

13.2.2.6 Senior Management Staff Position

A person selected or appointed for a specific task or range of tasks typically fulfills this function.

13.2.3 PLANT ORGANIZATION

13.2.3.1 Plant Organization

For administrative purposes, the plant organization consists of operations, maintenance, chemistry, and radiation protection. These functions report to the general manager of plant operations.

13.2.3.1.1 General Manager of Plant Operations

The general manager functions as a single point of responsibility for achieving standards of performance for overall plant operation.

13.2.3.1.2 Operations

The operations function is responsible for operating Pilgrim Station, manipulating all process systems, identification of system performance problems, and technical diagnosis of plant events.

The Pilgrim Station licensed operators and non-licensed operators are part of the operations function.

13.2.3.1.3 Shift Manning

The minimum shift crew composition will be as defined in Pilgrim Station Technical Specifications and in accordance with 10 CFR 50.54 (k),(l) and (m).

13.2.3.1.4 Deleted

13.2.3.1.5 Maintenance

The maintenance function is responsible for the management of maintenance activities including corrective maintenance, planned maintenance, and surveillance testing. This function supports safe and reliable plant operations.

The principal responsibilities include:

- Performing plant maintenance activities in a safe and quality manner, ensuring plant safety in accordance with the applicable local, state, and federal regulations and requirements, and in conformance with good Industry practices and the applicable corporate requirements.

- Enforcing radiological controls in accordance with the radiological control program ensuring radiation exposures at a level as-low-as-reasonably-achievable (ALARA).

- Coordinating activities to implement modifications and construction projects.

13.2.3.1.6 Outage

The outage function is responsible for outage administration and planning.

13.2.3.1.7 Instrumentation and Control

The instrumentation and control function is part of the maintenance function and is responsible for the management and coordination of all instrumentation and control activities for the station. This function implements programs which provide administrative and technical controls for the maintenance of the station.

The principle responsibilities of this function are to implement on-line and off-line instrumentation and control activities in a safe manner ensuring plant safety in accordance with the applicable local, state and federal regulations and in accordance with good industry practices.

13.2.3.1.8 Work Control

The work control function is part of the maintenance function and is responsible for all on-line and outage maintenance activities. Implementation of the work control function occurs when the unit is on-line, during planned outages and forced outages that require a progression of events to restore electrical generation in a timely fashion including planning and scheduling of corrective maintenance, preventive maintenance, modifications, surveillances and post work testing.

This function primarily interfaces with the operations and other maintenance organizations in allocating manpower, task prioritization, obtaining necessary support resources and assessing schedule progress to achieve outage milestones or on-line goals.

13.2.3.2 Deleted

13.2.3.2.1 Chemistry

The chemistry function is responsible for the chemistry programs and assuring compliance with applicable regulatory requirements, station operational requirements, and corporate policy.

The chemistry function plans, develops, coordinates and directs:

- a plant chemistry program for surveillances, required by the operating license, monitoring chemical parameters or plant process streams;

- technical guidance to ensure systems are operated efficiently to prevent unnecessary system degradation;

- operational chemical engineering services to develop and operate systems and components with plant chemistry impact such as the hydrogen water chemistry system; and

- all onsite chemical programs to insure compliance with federal, state, and industry standards to include chemical surveillance testing, chemical control, and the chemical preventive maintenance and quality assurance programs.

13.2.3.2.2 Radiation Protection

The radiation protection function deals with all aspects of the control and monitoring of radioactive material and monitoring, documenting, and control of personnel radiation dose.

The radiation protection department manager, is responsible for effective management of all radiological programs at Pilgrim Station.

The radiation protection manager approves, directs, and oversees the development and implementation of policies for:

- management of the Pilgrim Station Radiological Program, including implementation of the ALARA Program.

- input to facility design and operational planning.

- data collection and trend analysis in radiation work performance of station personnel, contamination and dose control, and job doses.

- initiation of action to correct adverse radiological trends.

- investigation of incidents associated with radiation protection controls, and assignment and follow-up of corrective actions.

The radiation protection manager is the senior advisor to management for radiological affairs concerning both the plant work force and the general public affected by the site. As such, the manager has direct access to the site vice president, when required, to ensure timely action on matters of significant radiological consequence.

The radiation protection manager interfaces with managers on a routine basis for the provision of support services, including aid in goal setting to control radiation dose to the work force and the general population.

This position acts as the spokesperson for Radiation Protection. The manager or designee fulfills requirements for radiation protection manager as specified in Regulatory Guide 1.8. In this regard, the incumbent exercises the authority to initiate and lift "Stop Work" orders imposed for inadequate radiation protection practices.

The Radioactive Material Control function is part of the radiation protection function and is responsible for:

- those activities associated with the generation, collection, transfer, packaging, shipment, storage, and disposal of low level radioactive waste at Pilgrim Station;
- maintaining compliance with applicable federal and state regulations and licenses on all radwaste shipments.

13.2.3.2.3 Protective Services

The protective services function directs onsite security activities in support of the safe operation of Pilgrim Station and ensures that provisions of the Site Security Plan, Contingency Plan, Training and Qualification Plan, and applicable regulations are met.

The Manager Security provides a single point of responsibility for standards, development, performance, and needs of security personnel. The manager ensures personnel are in a state of readiness to allow prompt and effective response to security incidents.

This function ensures security personnel interface effectively with the General Employee Training, Health Physics Records and Medical, plus local and state law enforcement agencies and other utility security groups.

13.2.3.2.4 Facilities

The facilities function is part of the maintenance function and performs general housekeeping and related activities including site painting, maintenance and cleaning.

13.2.3.3 Training

The training function is responsible for the Pilgrim Station training program. The training function develops and implements training programs to support line management including operations, technical and professional staff, and simulator training.

13.2.3.4 Quality Assurance

The quality assurance function is responsible for:

- assuring the implementation of the Quality Assurance Program.
- providing feedback to responsible management on compliance to and effectiveness of the Quality Assurance Program
- ensuring the establishment, maintenance, and implementation of an effective quality control function. This may be performed by other organizations, but remains the responsibility of the quality assurance function.

- assuring that all operational phase activities falling within the scope of the Quality Assurance Program are prescribed by and implemented according to approved procedures, and that these procedures provide effective management controls.

In order to implement the quality assurance functional responsibilities, this function is provided with “Stop Work” authority whereby the manager can suspend any quality related activity or process, which may adversely affect the safe operation of Pilgrim Station.

13.2.4 Deleted

13.2.4.1 Nuclear Engineering

The nuclear engineering function is responsible for the following:

- establishing plant design requirements.

- design engineering for design changes and plant modifications and selective maintenance.

- design verification and evaluation of changes to the base configuration of Pilgrim Station, which could affect the design and licensing basis.

- design configuration control, including establishment and maintenance of a system to incorporate approved design changes into engineering documents.

- engineering review of the current design in response to requests from the other members of the Nuclear Organization.

- preparation and review of 10 CFR 50.59 and 10 CFR 72.48 (note: this is added as "plant" is taken to include the ISFSI) evaluations for plant modifications and design changes.

- systems engineering review and analysis of performance data to determine potential changes to improve plant safety and operations reliability.

- make component design changes to support plant operations, including design changes required to support procurement of equivalent replacement items.

- fire protection

- ASME Section XI Pump and Valve Program

- 10 CFR 50 Appendix J Test Program

- ASME Section XI Inservice Inspection Program

13.2.4.2 Nuclear Assessment

The nuclear assessment function manages the interface between Pilgrim Station and federal, state, and local regulating agencies and insurers. The principal interface is between Pilgrim Station and the U. S. Nuclear Regulatory Commission (NRC) and includes the Office of Nuclear Reactor Regulation and Region I.

This function has the continuing responsibility for the following:

- preparing and processing changes to the Facility Operating License and Updated Final Safety Analysis Report.
- preparing reports to the NRC, FEMA and other agencies on emergency preparedness matters,
- technical evaluations and management of regulatory requirements and licensing issues.
- reporting requirements under 10 CFR 50.73.
- coordinating inspections and responses.
- environmental permits and monitoring programs.
- maintaining the Emergency Preparedness Program.
- maintaining the Pilgrim Station and Corporate Radiological Emergency Plan and implementing procedures in accordance with applicable regulations and industry standards.
- assuring adequate support is provided to ensure the maintenance of offsite emergency response plans and procedures for the Commonwealth of Massachusetts and the local communities involved in a response to an incident at Pilgrim Station.
- the Emergency Response Organization training program conducted to ensure the existence of an adequate level of knowledge, and that adequate records are maintained to track individual qualifications.
- the training program for offsite response personnel.
- preparing for and conducting the drill and exercise program.
- assuring that Emergency Response Facilities (onsite and offsite) are maintained in a constant state of readiness.
- developing and implementing the Emergency Preparedness public information program.

13.2.4.3 Purchasing, Materials and Contracts

The purchasing, materials and contracts function provides procurement and related tasks such as planning, requisitioning, purchasing, inspecting, expediting, manufacturing oversight, commercial grade item evaluation, logistics management, and warehousing in support of Pilgrim Station.

13.2.5 QUALIFICATIONS AND TRAINING

The operations manager, operations shift managers, and operations shift supervisors shall hold a Senior Reactor Operator License. The licensed nuclear plant reactor operators shall hold a Reactor Operator License.

13.2.5.1 Unit Staff Qualifications

The qualifications with regard to educational and experience backgrounds of the unit staff at the time of appointment to the active position shall meet the minimum requirements as described in the American National Standards Institute N18.1-1971, "Selection and Training of Personnel for Nuclear Power Plants." In addition, the individual performing the function of radiation protection manager shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

13.2.5.2 Training Program

A retraining and replacement training program for the unit staff is maintained under the direction of the training function.
The training programs for the licensed personnel shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and 10 CFR Part 55.

The training programs for the Fire Brigade shall meet or exceed the requirements of NFPA Standard No. 27-1975 "Private Fire Brigade."

For further information regarding Pilgrim Station Training Programs, see FSAR Section 13.3.

13.3 TRAINING

13.3.1 INTRODUCTION

The major operator and technical training programs at Pilgrim Nuclear Power Station are accredited programs of the Institute of Nuclear Power Operations (INPO). Entergy Nuclear is also a member of the National Academy for Nuclear Training in recognition of the quality and accreditation of its training programs.

Accredited programs use an Instructional System Design (ISD) model to develop performance-based training. The process includes five phases: analysis, design, development, implementation and evaluation. The process ensures that training is based on the measurable or observable performance of knowledge and skills required to safely and effectively perform the tasks of the job. The system uses continuous feedback on performance to help identify problems and improve training. Approximately every sixEvery four years INPO reexamines accredited programs to ensure that they continue to meet accreditation standards.

13.3.2 GENERAL TRAINING

13.3.2.1 Plant Access and Basic Radiation Worker Training

The goal of this INPO certified program is to ensure workers possess the knowledge and skills necessary to work efficiently and safely in a radiological environment with minimum exposure to themselves and other workers.

The scope of the program is designed to address the subject areas common to all employees requiring access to Pilgrim Nuclear Power Station. This program is not applicable to escorted visitors.

13.3.2.1.1 Initial Training

This training course is designed to introduce the new employee to Pilgrim Station and its Boiling Water Reactor operation. Emphasized throughout the course are those personal and plant safety concepts of radiological protection, industrial safety, security, quality assurance, and fitness for duty.

The course is subdivided into two (2) levels of training. Plant Access Training is designed for personnel requiring access to the protected/restricted area. Basic Radiation Worker Training is designed for personnel requiring access to contaminated areas or areas requiring radiation work permits for entry.

13.3.2.1.2 Requalification Training

This training course is required annually for those employees who have held and are required to maintain their unescorted access status at Pilgrim Station. The course provides refresher training in the subject areas included in Initial Training.

13.3.2.1.3 Respiratory Protection Training

This course is designed to qualify and requalify those individuals who perform work using respiratory protection equipment at Pilgrim Station. The purpose of the course is to provide the trainee with the skills and knowledge necessary to work safely and efficiently while utilizing respiratory protection equipment. This course is offered as needed for those employees requiring respiratory protection qualification.

13.3.2.1.4 Priority Access Training

Priority Access Training is a course designed for personnel requiring temporary access to Pilgrim Station to fulfill specific functions.

United States Nuclear Regulatory Commission (USNRC) inspectors, Institute of Nuclear Power Operations (INPO), and contractor escorted radiation workers are provided an overview of general procedures, policies and practices used at PNPS in order to ensure that these individuals can perform required tasks without jeopardizing his/her health and safety and the safety of co-workers.

13.3.2.2 Management and Supervision

The management and supervision training program assists the professional growth of personnel from the time they are newly hired or appointed to exempt status. PNPS department representatives and vendors from the management training field teach various segments of the overall program. The program consists of initial training and continuing training.

13.3.2.2.1 Initial Management and Supervisory Training

Initial management and supervisory training is designed to provide new management/supervisory personnel with an overview of the nuclear organization and its administrative policies and practices, regulatory and safety issues. It further focuses on developing the skills and knowledge required by supervisors to effectively communicate, direct, and evaluate the activities of subordinates.

13.3.2.2.2 Continuing Management and Supervisory Training

Continuing management and supervisory training is designed to provide management/supervisory personnel with refresher training in management and professional development skills. It focuses on the current changes within the organization to ensure that PNPS maintains a viable, effective management work force. Additionally this program provides management/supervisory personnel with specialty training in order to enhance the skills and knowledge required to effectively perform their duties. The duration and sequence of training is based on the operational requirements of the nuclear organization.

13.3.2.2.3 Maintenance Supervisor Initial Training Program

The maintenance supervisor initial training program is designed to provide maintenance supervisors with the skills necessary to perform their duties in a manner which promotes safe and reliable plant operations. The program focuses on the supervisory, leadership and judgment skills as well as the technical and administrative skills necessary to effectively supervise the work of maintenance personnel.

13.3.2.2.4 Shift Manager Initial Training

The Shift Manager program is designed for senior reactor operators with on-shift experience who have been selected as candidates for assuming the duties of the Shift Manager. This program focuses on the higher-level management skills and behaviors and strives for higher levels of understanding built upon the training and experience of the senior reactor operator.

13.3.2.3 Industrial Safety Training

This program is designed to provide station personnel with the skills and knowledge necessary to safely perform their assigned tasks. The program focuses on increasing the trainees awareness of the hazards associated with the industrial environment and with the precautions to be observed and practiced in the performance of daily activities.

13.3.2.3.1 First Aid and CPR Training

This training course is designed to provide the trainee with the skills and knowledge required to provide immediate care to victims of accident and injury situations.

13.3.2.3.2 Safety Awareness

Safety awareness concepts are provided to ensure that line management employees are knowledgeable in the process of identifying and rectifying hazardous conditions.

13.3.2.3.3 Asbestos Associated Worker

This unit is designed to provide the trainee with a working knowledge of asbestos hazards, implementation of special work procedures, and federal, state and local requirements associated with the health effects of asbestos exposure. A refresher training course is available on a non-mandatory basis for qualified asbestos workers who wish to attend.

13.3.3 OPERATOR TRAINING PROGRAMS

13.3.3.1 NRC License Training Programs

13.3.3.1.1 RO/SRO Initial License Training Program

The NRC initial license training program is an INPO accredited program and is designed from a job-task-analysis of the duties and responsibilities of both NRC licensed reactor operators (ROs) and NRC Licensed Senior Reactor Operators (SROs) at Pilgrim Station.

Satisfactory completion of the RO portions of this training program prepares an individual to safely, competently, and efficiently perform all tasks and duties which are the responsibility of an on-watch NRC licensed reactor operator at Pilgrim Station.

Satisfactory completion of the SRO portions of this training program prepares an individual to safely, competently, and efficiently perform all tasks and duties which are the responsibility of an on-watch NRC Licensed Senior Reactor Operator at Pilgrim Station.

Additional details of the RO and SRO Initial License Training Programs are contained within the individual program description.

13.3.3.1.2 Shift Control Room Engineer (SCRE) Qualification Training Program

The SCRE Qualification Training Program is designed from a job-task-analysis of the duties and responsibilities of a qualified SCRE at Pilgrim Station.

The SCRE training program augments the initial SRO training for selected individuals to prepare personnel for the duties and responsibilities formerly performed by Shift Technical Advisor.

Satisfactory completion prepares an individual to safely, competently, and efficiently perform all tasks and duties which are the responsibility of the on-watch SCRE at Pilgrim Station.

13.3.3.1.3 Licensed Operator/Shift Control Room Engineer Requalification Training Program

The Licensed Operator (RO/SRO)/Shift Control Room Engineer (SCRE) Requalification Training Program is an INPO accredited program and was developed in order to maintain the qualifications and competency of NRC licensed operators and qualified SCREs. This program is conducted in accordance with 10CFR55.59 using a systematic approach to training as described in section 55.59(c).

The program reinforces the knowledge and skills acquired in the Initial Training Programs and keeps knowledge and skills up-to-date/current regarding pertinent:

Station equipment/system changes
Station procedure changes
Station instruction/policy changes
Current industry events/lessons learned

Additional details of the Licensed Operator/Shift Control Room Engineer Requalification Training Program are contained within the individual program description.

13.3.3.2 SRO Certification

13.3.3.2.1 Pilgrim Station SRO Certification Training Program

The Pilgrim Station SRO Certification Training Program has been developed in order to provide a senior reactor operator level of knowledge and abilities to selected Nuclear Organization personnel. These personnel require an advanced level of technical understanding of integrated plant operations, but do not require an NRC License to perform their assigned jobs. (For example: Selected Nuclear Training Department staff.)

Additional details of the SRO Certification Training Program are contained within the individual program description.

13.3.3.2.2 SRO Certification Requalification Training Program

The SRO Certification Requalification Training Program was developed to maintain the competency of selected Nuclear Organization personnel who have previously completed initial SRO Certification training. Those individuals that require current SRO level knowledge and abilities due to their position within the organization participate in this program.

This program refreshes and updates those skills and knowledges acquired in initial training. Additional details of the SRO Certification Training Program are contained within the individual program description.

13.3.3.3 Non-Licensed Nuclear Plant Reactor Operator

13.3.3.3.1 Non-Licensed Nuclear Plant Reactor Operator Qualification Training Program

The Non-Licensed Nuclear Plant Operator (NLNPRO) Qualification Training Program is INPO accredited and is designed from a job-task-analysis of the duties and responsibilities of a qualified NLNPRO at Pilgrim Station.

Satisfactory completion prepares an individual to safely, competently, and efficiently perform all tasks and duties which are the responsibility of the on-watch NLNPRO at Pilgrim Station.

Additional information on Pilgrim Station's NLNPRO Qualification Training Program is contained within the individual program description.

13.3.3.3.2 Non-Licensed Nuclear Plant Reactor Operator Requalification Training Program

The NLNPRO Requalification Training Program, an INPO accredited program, was developed in order to maintain the competency of qualified NLNPROs by:

Re-enforcing knowledge and skills acquired in the Initial NLNPRO Qualification Training Program.

Keeping the NLNPRO's knowledge and skills up-to-date/current regarding pertinent:

Station equipment/system changes
Station procedure changes
Station instruction/policy changes
Current industry events/lessons learned

Additional details of the NLNPRO Requalification Training Program are contained within the individual program description.

13.3.3.4 Plant Specific Simulator

Pilgrim Station Operator Training programs extensively utilize the PNPS plant-specific control room simulator. Simulator time is integrated into the programs to emphasize operations techniques, procedural adherence, control room operations, Emergency Operations Procedure performance, communications techniques, and other specialized training functions. The simulator meets ANS/ANSI 3.5, 1985 standards and incorporates advanced system modeling software allowing accurate simulator fidelity with the actual plant.

The simulator is capable of responding, in real time, to almost all functions an operator would perform in the PNPS Control Room under any condition. It also has the capability to be used during Emergency Plan Drills and has all required communications equipment installed for full drill performance.

13.3.3.5 (deleted)

13.3.4 TECHNICAL TRAINING PROGRAM

13.3.4.1 Nuclear Maintenance Mechanic

13.3.4.1.1 Initial Training

Initial Training for Nuclear Maintenance Mechanics is conducted to ensure that newly assigned Nuclear Maintenance Mechanics can perform at the desired competency level without detailed supervision. This training ensures maintenance of equipment is performed in accordance with station procedures, regulatory and industry standards.

Training is provided for both an orientation and duty area qualification.

A description of the training program can be found within the individual program description.

13.3.4.1.2 Continuing Training

Continuing Training for the nuclear maintenance mechanics provides for the continuous updating of skills and knowledge via a planned training program conducted on a periodic bases. Operating experience, procedure change, changes in plant configuration and manufacturer’s recommendations are included in this program. In addition, advanced skills training is provided to selected personnel, as necessary, to meet the needs of the station.

Continuing Training uses the systematic approach to training and is evaluated and scheduled on periodic bases. Periodic training requirements are incorporated into this schedule.

A description of the program can be found within the individual program description.

13.3.4.2 Nuclear Maintenance Electrician

13.3.4.2.1 Initial Training

This program is applicable to newly assigned Entergy Nuclear Maintenance Electricians. Initial Training for the Nuclear Maintenance Electrician is conducted to ensure performance at the desired competency level without detailed supervision. This training ensures maintenance of equipment is performed in accordance with station procedures, and regulatory and industry standards.

This training is provided for both an orientation and duty area qualification

A description of the training program can be found within the individual program description.

13.3.4.2.2 Continuing Training

Continuing Training for the Nuclear Maintenance Electrician provides for continuous updating of skills and knowledge via a planned training program conducted on a periodic basis. Operating experience, procedure changes, changes in plant configuration and manufacturer's recommendations are all included in this program. In addition, advanced skills training is provided to selected personnel, as necessary, to meet the needs of the station.

The description of the training program can be found within the individual program description.

13.3.4.3 Nuclear Control Technician

13.3.4.3.1 Initial Training

Initial training for Nuclear Control Technicians is conducted to ensure that newly assigned technicians can perform at the desired competency level without detailed supervision. This training ensures maintenance of equipment is performed in accordance with station procedures, and regulatory and industry standards.

The description of the training program can be found within the individual program description.

13.3.4.3.2 Continuing Training

Continuing training for Nuclear Control Technicians provides for the continuous updating of skills and knowledge via a planned training program conducted on a periodic basis. Operating experience, updated procedures, changes in plant configuration and manufacturer’s recommendations are all included in this program. In addition, advanced skills training is provided to selected personnel to meet the needs of the station.

Continuing training uses the systematic approach to training and is evaluated and scheduled on a periodic bases. Periodic training requirements are incorporated into this schedule. A description of the training program can be found within the individual program description.

13.3.4.4 Radiation Protection Technician

13.3.4.4.1 RP - Technician Initial Training

The purpose of this program is to train newly assigned Entergy radiological protection (RP) technicians who meet the following entry level requirements:

Job description requirements for RP Technician
ANSI N18.1

The program, developed using a systems approach to training, is designed to produce highly competent RP technicians who are able to respond to actual or simulated plant situations, performing all tasks required in accordance with the guidelines and procedures established at the station.

The initial RP Technician Training Program is made up of three qualification phases:

PHASE I - Job Coverage Qualification
PHASE II - Watch Stander Qualification
PHASE III -Full Qualification

Successful completion of the program is demonstrated by passing course quizzes, milestone exams, practical exams.

13.3.4.4.2 RP Technician - Requalification Training

This program, developed using a systems approach to training, is designed to assure the availability of qualified RP technicians, who have maintained their skills through a program of continuous training and evaluation, the content of which is regularly based upon an on-going analysis.

Requalification Training for RP Technicians is conducted to ensure that the detailed and plant-specific knowledge of RP Technicians will be such that they perform at the desired competence level. This program was designed to:

Maintain and enhance the skills and knowledge necessary to accomplish routine, abnormal and emergency duties.

Emphasize lessons learned from industry and plant operating experience to prevent repetition of errors.

Correct performance deficiencies related to training, while enhancing professionalism.

Improve knowledge and skills when changes in responsibilities are identified.

Systematically evaluate individual and team performance to identify areas for improvement.

Ensure that technicians receive information on plant modifications, industry events and procedural changes in a timely manner.

Increase the level of understanding of fundamental principles obtained in initial training, including an emphasis in areas of demonstrated weakness.

Maintain awareness of responsibilities for safe operation of the plant including consequences of incorrect job performance.

Provide input to station management for improvements in operation, practices and procedures.

This program is applicable to all Entergy RP technicians and supervisory personnel with RP responsibilities.

13.3.4.4.3 Radiation Protection Technician - Contractor

Contractor RP Technicians are assigned to PNPS to supplement the Entergy staff during periods of peak workloads. Contractor RP Technicians are often well trained and have extensive experience in Radiation Protection at several facilities.

The purpose of this program is to train contractor RP technicians in PNPS specific procedures and radiological protection practices. The Program is applicable to all experienced contractor RP Technicians assigned to PNPS. The program is developed using the systems approach to training.

13.3.4.5 Chemistry Technician

13.3.4.5.1 Initial Training

Initial training for Chemistry Technicians is conducted to ensure that the detailed and plant-specific knowledge of Chemistry Technicians will be such that they can perform at the required level of competence.

The purpose of the program is to train newly-assigned Entergy Chemistry Technicians who meet the following entry-level job description requirements:

Job description requirements for Chemistry Technicians
ANSI N18.1

The program, developed using a systems approach to training, is designed to produce highly competent Chemistry Technicians who are able to perform their assigned responsibilities during routine and emergency conditions in accordance with guidelines and procedures established at PNPS.

Successful completion of the program is demonstrated by passing quizzes, milestone exams, and performance exams.

13.3.4.5.2 Requalification Training

This program, developed using a systems approach to training, is designed to ensure the availability of qualified Chemistry Technicians who have maintained their skills through a program of continuous training and evaluation, the content of which is regularly updated based upon an on-going analysis.

Requalification Training for Chemistry Technicians is conducted to ensure that the detailed and plant-specific knowledge of Chemistry Technicians will be such that they perform at the desired competence level. This program was designed to:

Maintain and enhance the skills and knowledge necessary to accomplish routine, abnormal, and emergency duties;

Emphasize “lessons learned” from industry and plant operating experience to prevent repetition of errors;

Correct performance deficiencies related to training, while enhancing professionalism;

Improve knowledge and skills when changes in responsibilities are identified;

Systematically evaluate individual and team performance to identify areas for improvement;

Ensure that Technicians receive information on plant modifications, industry events, and procedural changes in a timely manner.

Increase the level of understanding of fundamental principles obtained in initial training, including an emphasis in areas of demonstrated weakness;

Maintain awareness of responsibilities for safe operation of the plant, including consequences of incorrect job performance;

Provide input to station management for improvements in operation, practices and procedures.

This program is applicable to all Entergy Chemistry Technicians and Supervisory personnel with Chemistry responsibilities. Requalification Training for Chemistry personnel will be scheduled to meet various regulatory requirements on the basis of an assessment of training needs and/or the need to impart knowledge and/or skills caused by plant configuration of procedure changes.

13.3.4.5.3 Chemistry Technician - Contractor

Contractor Chemistry Technicians are assigned to PNPS to supplement the Entergy staff during periods of peak work loads. Contractor Chemistry Technicians are often well trained and generally have had extensive experience in Chemistry at several facilities.

The purpose of this Program, developed using a systems approach to training, is to train the Contractor Technicians on site-specific procedures and practices. This program is applicable to all Contractor Chemistry Technicians assigned to PNPS.

13.3.4.5.4 Chemistry Supervisory Training

Chemistry Supervisory Training is designed to ensure that a sufficiently trained staff is available to manage and supervise the PNPS Chemistry Division at the desired competence level.

The program is designed to meet requirements for updating and retraining of Chemistry supervisory personnel to assure safe operations at PNPS. This ensures that the individual Chemistry Supervisor is able to respond to actual or simulated plant situations, performing all tasks required in accordance with the guidelines and procedures established at the station.

This program is applicable to all Chemistry Supervisors assigned to PNPS. Retraining for Chemistry Supervisors is scheduled to meet regulatory requirements to assess training needs, and/or to impart knowledge or skills needed due to plant configuration and procedure changes. As required, retraining includes results of industry experience and "lessons learned" as a result of continuing operation.

The Chemistry Supervisory Training Program consists of two parts.

Phase One is the Supervisory Orientation program which is intended as an initial orientation to various "administrative" practices for which the Chemistry Supervisor will have some responsibility. The program is delivered using a combination of monitored, self-study, and formal courses, as well as examiner review of various program elements.

Phase Two is intended to address the ongoing training needs of Chemistry Supervisors and Managers who have already satisfied the requirements of Phase One (through either completion of the orientation program or on-the-job experience). Lessons are available for numerous supervisory and management skill areas. Training for individual supervisors is based on an assessment of their needs.

13.3.5 OTHER TECHNICAL TRAINING

13.3.5.1 Fire Brigade Training

The Fire Brigade Training Program is comprised of Initial and Requalification Training. The Initial Training consists of approximately 80 hours. The Requalification Training consists of one session (training session) per quarter. The session length varies depending on subject(s) to be taught. The training program meets or exceeds the requirements of NFPA standard No. 27-1975 "Private Fire Brigade" (Reference 14).

The program is designed for those personnel assigned to perform Fire Brigade duties at Pilgrim Station.

The goal of the Fire Brigade Training program is to provide education and training in order to ensure that workers have the knowledge and skills necessary to work efficiently and safely in a fire or smoke filled environment.

The training is divided into eight areas:

Fire Protection Systems
Self-Contained Breathing Apparatus
Fire Water Supply
Fire Behavior
Plant Fire Protection
Fire Brigade Equipment
Advanced Training
Recruit Training

13.3.5.2 Engineering Support Personnel

The Engineering Support Personnel Training Program is INPO accredited and is designed to ensure safe and reliable plant operation by providing a training and qualification process for plant personnel responsible for engineering and support functions. The program is comprised of both initial and continuing training components.

13.3.5.3 Emergency Preparedness

This program is designed to qualify specified individuals of the PNPS Emergency Response Organization to properly perform their assigned functions of protecting health and safety of the general public in the event of an accident at PNPS.

13.3.6 REFERENCES

1. Regulatory Guide 1.8 (ANSI N18.1, Section 5.4)
2. Regulatory Guide 1.70.38, Section 13.2.2.3
3. 10CFR50, Appendix B, Criteria II (ANSI N18.7, Section 3.3)
4. 10 CFR 19.12
5. 10 CFR 20.101, 20.102
6. 10CFR50.73
7. INPO Guideline 82-004
8. Regulatory Guide 1.8 (ANSI N18.1, Section 5.3.4)
9. 10CFR50, Appendix B, Criteria II (ANSI N18.7, Section 5.2.10; ANSI N45.2.3, Section 2.4)
10. 10CFR50, Appendix B, Criteria II
11. 10CFR50, Appendix B, Criteria VI
12. 10CFR50, Appendix B, Criteria III
13. 10CFR50, Appendix B, Criteria VI (ANSI N18.7, Section 5.2.15)
14. NFPA Standard No. 27-1975 "Private Fire Brigade"
15. 10CFR26 (54FR24468), Published June 7, 1989.

13.4 PREOPERATIONAL TEST PROGRAM

13.4.1 Objectives

The purpose of the preoperational test program in the 1971-1972 timeframe was three fold: confirm that construction was complete to the extent that equipment and systems can be put into use during completion of other construction; adjust and calibrate the equipment to the extent possible in the "cold" plant; assure that all process and safety equipment were operational and in compliance with license requirements, to the extent necessary to proceed into initial fuel loading and the startup program. The foregoing was achieved by construction tests, formal written preoperational tests on systems related to nuclear safety, and acceptance tests on systems not related to nuclear safety.

In order to present the most efficient startup program, some tests not related to safety were included in the formal preoperational category. An example of this were the tests conducted on the 4,160V buses used in the balance of plant. In some cases, tests not related to safety and itemized as acceptance tests were similar in nature to the formal preoperational tests. Examples of this were the tests conducted on the Condensate and Feedwater System.

The formal preoperational and acceptance tests were an important phase in the training of operating personnel. Experience and understanding of station systems and components was gained with a minimum of risk to the equipment or personnel. This gave maximum opportunity to evaluate and train individual operators and to troubleshoot systems. In addition, equipment and systems were operated for a sufficient period of time to discover and correct any design, manufacturing, or installation errors, and to adjust and calibrate the equipment.

The remaining sub-sections of this section are historical and no changes were made to reflect the historical context of the information presented.

13.4.2 Preoperational Test Considerations

The following key points were considered in developing the sequence and schedule of preoperational tests:

1. Systems are sequenced for early testing and placed in routine operation to provide necessary auxiliary services for other systems. Examples are electrical systems, instrument air, and makeup water supply systems.

2. Preoperational testing is coordinated with construction to permit fuel loading as early as possible, without compromising nuclear safety or impeding construction work. As a result, fuel loading may occur while construction work is still in progress on unrelated systems and areas.

3. Temporary construction power is sometimes required for initial tests at the beginning of the preoperational test program. However, unnecessary use of temporary power and improvised setups was avoided because of the possibility of errors and inconsistency, with the ultimate objective of proving the final installation.

4. Electrical jumpers are used to facilitate preoperational testing in some instances, but their use is minimized and controlled by proper identification of such jumpers, by tags on the equipment jumpered, and by log book records. All jumpers are removed before fuel loading.

5. When the station is ready for fuel loading, strict control is enforced over personnel access to the main control room, electrical equipment rooms, and Reactor Building.

6. The Applicant's operating personnel will operate the station and equipment during formal preoperational and acceptance testing. Construction tests are performed by subcontractors and are under the surveillance of the Applicant. These construction tests utilize procedures and data sheets and required reporting and acceptance. Typical construction tests are listed in Section 13.4.3.1.

7. Specialized electronic equipment and nuclear instrumentation manufactured by General Electric is checked and preoperationally tested by the station staff under the technical direction of General Electric representatives.

8. Formal preoperational and acceptance test procedures are prepared by the Applicant's personnel under the technical direction of General Electric and/or Bechtel, depending upon system design responsibility.

9. In general, tests are performed using permanently installed instrumentation for the required data. Where it was not possible to use permanently installed instrumentation during the system tests, it will be necessary to install test thermometers, vibrometers, stroboscopes, or other test instrumentation to ensure safe operation of the equipment. Any test requiring artificial simulation of a station parameter has the method detailed in the procedure as well as the means for assuring that the system was returned to normal.

13.4.3 Summary of Preoperational Test Content

13.4.3.1 Construction Tests

Construction tests, where required, include but are not limited to the following examples:

1. Containment overpressure test

2. System hydrostatic tests

3. Cleaning, flushing, and leak testing piping and equipment.

4. Nondestructive testing of field welds

5. Wiring continuity checks

6. Megger tests

7. Electrical system tests, e.g., checking grounding, checking circuit breaker operation and controls, continuity checks, and megger tests.

8. Initial adjustment and bumping of motors.

9. Check control and interlock functions to verify wiring and/or pneumatic tubing per design.

10. Pneumatic test of Instrument and Service Air System and blow out of lines.

11. Adjustments such as alignment, lubrication of equipment, and tightening of bolts.

12. Tests of motor operated valves including adjusting limitorque switches and limit switches, measuring operating speed, and checking leaktightness of stem packing and valve seat during hydrotests.

13. Tests of air operated valves including pilot solenoids, adjusting limit switches, measuring operating speed, checking leaktightness of stem packing and valve seat during hydrotests, and checking leaktightness of pneumatic operators.

14. Verify installation of main steam flow restrictors.

15. Verify installation of control rod drive housing support.

16. Verify installation of reactor vessel stabilizers.

17. Verify installation of pipe hangers, component restraints, and expansion joints.

13.4.3.2 Formal Preoperational Tests

13.4.3.2.1 DC Power Systems

Check and/or calibrate relays, instruments, breakers, interlocks, and other electrical components. Verify battery charger capacity and battery discharge rate.

13.4.3.2.2 AC Power Systems

Check and/or calibrate all protective devices, interlocks, followup schemes, and other electrical components. Verify insulation quality and/or circuit continuity, phase rotation, and functional operation.

13.4.3.2.3 Control Rod Drive Hydraulic and Manual Control Systems

These tests will include performance checks and/or calibration of pumps, instrumentation, flow control valves, interlocks, alarms, and controls. Control rod selection relays and valves were tested as well as scram valves and their pilot solenoids, backup scram solenoids, and scram discharge volume vent and drain valves.

13.4.3.2.4 Salt Service Water System

The objective of the preoperational test will be to verify the functional capability of the Service Water System to provide cooling water for the Reactor and Turbine Building Closed Cooling Water Systems. The tests will verify system flows, instrumentation, and controls in the cold conditions.

13.4.3.2.5 Reactor Building Closed Cooling Water System

The objective of the preoperational tests will be to verify the functional capability, flows, temperature controller response, and instrumentation in the cold conditions.

13.4.3.2.6 Core Spray System

The objective of the preoperational tests will be to verify the automatic initiation and interlocks, spray function, flow rate, and system operation using the Standby AC Power System.

13.4.3.2.7 Reactor Heat Removal Shutdown Cooling Mode

The objective of the preoperational tests will be to verify the permissives, interlocks, flow capacity, and valve operation of the system.

13.4.3.2.8 Reactor Heat Removal Low Pressure Coolant Injection Mode

The objective of the preoperational tests will be to verify the automatic initiation and interlocks, valve control, flow rate, and system operation using the Standby AC Power System.

13.4.3.2.9 Reactor Heat Removal Containment Spray Mode

The objective of the preoperational tests will be to check the containment spray nozzles by blowing air into the spray headers after the headers have been flushed with water.

13.4.3.2.10 Reactor Heat Removal Suppression Pool Cooling Mode

The objective of this preoperational test is to verify the flow capacity, valve control, and system operation using the Standby AC Power System.

13.4.3.2.11 Standby Liquid Control

This test will include operation of pumps and valves using demineralized water. The explosive valves will be tested and the discharge rate to the reactor vessel will be measured. The liquid control solution will then be prepared in the liquid control tank prior to fuel loading. The boron content will be verified by laboratory testing.

13.4.3.2.12 Reactor Core Isolation Cooling

The objective of this test will be to check and/or calibrate instrumentation, relays, interlocks, protective devices, and equipment that can be operated. Flow capacity testing of this system will be deferred until nuclear steam is available.

13.4.3.2.13 High Pressure Coolant Injection

The objective of this test will be to check and/or calibrate instrumentation, relays, interlocks, protective devices and equipment that can be operated. Flow capacity testing of this system was deferred until adequate nuclear steam is available.

13.4.3.2.14 Reactor Recirculation System

The objective of this test will be to check and/or calibrate instrumentation, interlocks, protective devices, and flow control equipment to the degree possible with cold water conditions. Jet pump instrumentation will be checked during these tests.

13.4.3.2.15 Liquid Radwaste System

Testing will assure proper instrument operation and capacity of sumps, tanks, filters, pumps, and demineralizers. Required portions of the Radwaste Disposal System will be available to receive wastes from the Reactor Building drains, fuel pool, and interconnecting auxiliary systems required for fuel loading.

13.4.3.2.16 Primary Containment Leak Rate

This test will be performed after completing the installation of all penetrations. Penetrations and isolation valves which were testable, including the main steam isolation valves will be checked for leakage. Integrated leak rate measurement tests will be performed on the containment at two different pressures.

13.4.3.2.17 Control Rod Drives

These tests will include control rod stroking (both continuous withdrawal and notch by notch withdrawal) and stroke timing, drive speed setting, scram time measurements, proper position indication, control rod scrams to verify safety circuit sensors, and rod withdrawal interlocks exclusive of the rod worth minimizer.

13.4.3.2.18 Reactor Protection System, PC and RV Isolation and Control System, and Core Standby Cooling Systems

The Reactor Protection System will be tested after safety system sensors are installed and calibrated and wiring is installed and checked for continuity. The motor generator sets will be operated with a resistance load to check capacity and regulation. Each safety sensor will be checked for operation of the proper relay. Using test signals, scram set points will be verified for each sensor. All positions of the reactor mode switch and control rod permissives will be checked to verify interlocks function properly. Automatic closing of primary containment and reactor vessel isolation valves will be checked and closing times measured. Automatic initiation of core spray, high pressure coolant injection, reactor heat removal, low pressure coolant injection, reactor core isolation cooling, and diesel generator start will be verified using their proper initiation signals.

13.4.3.2.19 Neutron Monitoring System

A comprehensive check of each neutron monitor will be made from the chamber to the indicators, recorders, and safety interlocks and trips. This includes the source range monitor, intermediate range monitors, local power range monitor, average power range monitor, and traveling incore probe systems. The proper installation of the chambers, chamber drive, and other components will be verified. Continuity, ground resistance, and noise level of cables will be checked. Chamber installation and removal procedures will be verified.

13.4.3.2.20 Fuel Handling System

The equipment will be tested with dummy fuel or blade guide assemblies through dry run simulations of the required operations. This test consists of many separate operations to check the different pieces of equipment which are used for service and refueling operations. The tests include checking fuel preparation and inspection equipment, operation of the refueling platform, operation of the service platform and fuel grapple, and operation of equipment used for the installation and removal of internals such as guide tubes and fuel support castings. Refueling interlocks will be included as part of this preoperational test.

13.4.3.2.21 Reactor Level Control

The reactor level control will be tested after sensors are installed and calibrated by simulating flow, level, and reactor feed pump signals to verify the operation of the Control System.

13.4.3.2.22 Standby Gas Treatment and Reactor Building Leak Rate

Instrumentation and controls will be calibrated and interlocks checked. Blowers will be operated to check flow capacity; automatic isolation of the Reactor Building and initiation of the Standby Gas Treatment System (SGTS) will be verified. With the normal Reactor Building Ventilation System isolated, the SGTS will be operated to measure Reactor Building leak rate under a negative pressure. The absolute and charcoal filter collection efficiency will be measured.

13.4.3.2.23 Rodworth Minimizer and Position Indication

Simulate control rod movement to verify the response of the rod position information control to position data from all control rod locations and check computer logged position data. Run computer diagnostic programs to verify proper program operation. After control rod drive system is operational, verify actual rod identification against output readings. In addition, verify that improper rod withdrawal at selected points in the withdrawal sequence result in correct interlock action.

13.4.3.2.24 Nuclear System Pressure Relief

Safety and relief valves will be tested at the vendor's facility, where set points will be adjusted, verified, and indicated on the valve. The relief valve controls will be functionally tested during the preoperational phase. The Automatic Depressurization System control and trip signals will be functionally tested. During the subsequent power test program, the relief valves will be actuated to prove their operability.

13.4.3.2.25 Nuclear System Leak Detection

Check and/or calibrate pressure and temperature sensors for monitoring primary containment, flow elements for measuring flow rates from drywell equipment and floor sumps, temperature sensors for main steamline leak detection, temperature sensors on drywell coolers, and the Vessel Head Seal Leakoff System.

13.4.3.2.26 Standby AC Power System Tests

After instrumentation and controls are installed and calibrated, and wiring is checked, the capability of each diesel generator to pick up core spray, reactor heat removal, cooling water pumps and associated emergency loads in sequence will be demonstrated. Each diesel generator will be tested for load carrying capability. All interlocks, automatic initiation, and followup schemes on the auxiliary and shutdown transformers will be tested as part of these tests.

13.4.3.2.27 Main Steam, Offgas, Main Stack, and Reactor Building Ventilation Radiation Monitoring Systems

Check and/or calibrate relays, sampling pumps, recorders, trips, interlocks, valve operations, and logic associated with these systems.

13.4.3.2.28 Area Cooling for Safeguards Equipment and Main Control Room Environmental Control

Check and/or calibrate relays, temperature sensors, pressure sensors, trips, interlocks, and logic associated with these systems.

13.4.3.3 Acceptance Tests

13.4.3.3.1 Electrical System Test, Normal Auxiliaries

The objectives of this acceptance test will be accomplished as part of preoperational test 13.4.3.2.2, AC Power Systems.

13.4.3.3.2 Domestic and Makeup Water System

Check and/or calibrate all instrumentation, valves and pumps operability, interlocks, and protective devices. Perform functional operation to verify time cycle. Verify water quality.

13.4.3.3.3 Instrument and Service Air System

Check and/or calibrate all instrumentation, interlocks, and followup schemes, and perform operational check of system. Verify dew point of instrument air to system.

13.4.3.3.4 Fire Protection

Check and/or calibrate all instrumentation, pumps, engines, piping, followup schemes, and perform operational check of the system. Verify spray patterns of automatic initiated deluge.

13.4.3.3.5 Turbine Building Closed Cooling Water System

The objective of the acceptance tests will be to verify the functional capability, flows, temperature controller response, and instrumentation in the cold conditions.

13.4.3.3.6 Chlorination System

The objective of the acceptance tests will be to verify the functional capability of the system and to operate system in order to verify the salt water residual levels during different seasons of the year.

13.4.3.3.7 Screen Wash System

The objective of the acceptance tests will be to verify the operation of the pumps, valves, and controls to properly develop a spray pattern to minimize screen fouling.

13.4.3.3.8 Circulating Water System

Check and/or calibrate pumps, valves, instrumentation, interlocks, and check system operation.

13.4.3.3.9 Condensate Demineralizer Regeneration System

Check and/or calibrate pumps, valves, instrumentation, interlocks, resin transfer, and resin interface, and verify system operations.

13.4.3.3.10 Condensate and Feedwater System

Check and/or calibrate pumps, valves, piping, instrumentation, controls, interlocks, trips, water quality, and condensate demineralizer operation.

13.4.3.3.11 Reactor Vessel Instrumentation

The objective of the acceptance tests is to verify all instrumentation associated with the reactor vessel that is not related to nuclear safety or contained within other control systems.

13.4.3.3.12 Reactor Cleanup System

Check and/or calibrate instrumentation, valve and pumps operability, interlocks and protective devices; perform functional operation to verify time cycle. Verify water quality and flow capacity of filter demineralizers.

13.4.3.3.13 Drywell Cooling System

Verify operability of fans and coolers in drywell. Check and/or calibrate instrumentation associated with containment pressure, temperature, and suppression pool level.

13.4.3.3.14 Area and Process Radiation Monitoring Systems

Check and/or calibrate radiation monitoring instrumentation not related to nuclear safety. These monitors include liquid and process radiation monitors, area monitors, and personnel monitors.

13.4.3.3.15 Solid Radwaste System

The objective of the acceptance test will be to gain experience in operating the equipment before it becomes significantly contaminated.

13.4.3.3.16 Fuel Pool Cooling and Filtering

Check and/or calibrate instrumentation, valves, pumps, heat exchangers, filters, demineralizers, and verify operability of system. Spent fuel pool will be filled with demineralized water, checked for leakage, and pool and surge test instrumentation checked.

13.4.3.3.17 Heating, Ventilating, and Air Conditioning

Check and/or calibrate instrumentation, valves, pumps, fans, heaters, coolers, dampers, louvers, and other equipment associated with these systems. Verify operability of systems.

13.4.3.3.18 Turbine Extraction Steam

Check and/or calibrate instrumentation, valves, piping, controls, trips, interlocks, and other equipment associated with this system. Verify operability of system. Included with these tests were the Feedwater Heaters' Level Control Systems.

13.4.3.3.19 Lube Oil Purification

Check and/or calibrate instrumentation, valves, piping, pumps, filters, separators, and other equipment associated with this system. Verify operability of system.

13.4.3.3.20 Gland Steam System

Check and/or calibrate instrumentation, valves, regulators, blower, loop seals, and other equipment associated with this system. Verify operability of system.

13.4.3.3.21 Steam Air Ejector

Check and/or calibrate instrumentation, valves, regulators, loop seals, protective devices, interlocks, and other equipment associated with this system. Verify operability of system.

13.4.3.3.22 Generator Cooling Systems

The objective of the acceptance test will be to check the functional operation of the hydrogen (including CO purge) gas and stator liquid cooling systems, including checking and/or calibration of instrumentation, pumps, regulators, and other equipment associated with these systems.

13.4.3.3.23 Isolated Phase Bus Cooling System

The objective of the acceptance test will be to check the functional operation of the system, including checking and/or calibration of instrumentation, fans, dryers, and other equipment.

13.4.3.3.24 Turbine Generator Protection System

Check and/or calibrate all relays, limit switches, pressure switches, interlocks, overspeed trips, position indicators, turbine valves, supervisory instrumentation, and functional check tripping sequence. Verify all trip inputs.

13.4.3.3.25 Station Heating Boilers

Check and/or calibrate all protective devices. Verify trips and operability of system.

13.4.3.3.26 Process Computer

Check all hardware, software, and periphery equipment. Simulate calculation for balance of plant and Nuclear Steam Supply System.

13.5 REACTOR STARTUP AND POWER TEST PROGRAM

13.5.1 General Objectives

The initial reactor startup and power test program was performed to demonstrate that the station was capable of operating safely and satisfactorily. Systems and components which could not be fully checked out in the preoperation tests were tested at power during this phase of the station startup. The nuclear characteristics of fuel, control rods, and control curtains were compared with calculations throughout the startup program to confirm the design values. The detailed startup procedures were completed and the schedule of tests was carefully supervised to achieve the planned objectives. A typical test sequence for the startup test phase is presented on Figure 13.5-1.

Except for sub-section 13.5.7, the remaining sub-sections are historical and no changes were made to reflect the historical context of the information presented.

13.5.2 Fuel Loading and Tests at Atmospheric Pressure

The initial fuel loading and critical testing are performed at near zero power and at atmospheric pressure with the reactor pressure vessel open. The following tests are performed during this phase of the startup program:

1. Chemical and radiochemical tests were conducted to establish water conditions prior to initial operation and to maintain these throughout the test program. Chemical and radiochemical checks are made at primary coolant, offgas exhaust, waste, and auxiliary system sample locations. Base or background radioactivity levels are determined at that time for use in fuel assembly failure detection and long range activity buildup studies

2. Control rod drive system tests are performed on all drives prior to fuel loading to assure proper operability and to measure and adjust operating speeds. Functional testing of each drive is performed just prior to and then following the fuel loading in each cell. Drive line friction and scram times were determined for each drive at zero reactor pressure with fuel loaded

3. Vibration measurements at cold flow conditions are performed as necessary to determine the vibrational characteristics of reactor vessel internals of modified designs. The results of extensive vibration measurements made at other BWR installations will be considered in selecting the components to be tested at Pilgrim Station

4. Fuel loading is performed according to detailed, step-by-step written procedures. Control curtains are in place and the test and operational neutron sources are installed as required. Loading proceeded to the full sized core

5. Shutdown margin - It is demonstrated periodically during fuel loading that the reactor is subcritical by more than a

specified amount with the single highest worth control rod withdrawn. The magnitude of the margin is chosen with consideration for expected reactivity changes during the first operating cycle, and for the accuracy of measurement. The test has three parts: (a) the analytical determination of the control rod having the greatest reactivity worth, (b) the calibration of an adjacent control rod, determined analytically, and (c) the demonstration of subcriticality with the highest worth rod fully withdrawn and the second at the position necessary to insert the required margin. This demonstration is made for the fully loaded core and for selected smaller core loadings

6. Control rod sequences were evaluated to verify that the stated criteria of safety, simplicity, and operating requirements are met during routine cold startup. The reactor is brought critical by withdrawing control rods in a specified sequence and reactivity addition rates are measured near critical. The preselected sequence is modified if necessary to meet criteria. A few nonstandard arrays are utilized to check out the operation of the rod worth minimizer

7. Source Range Monitor (SRM) Performance - Adequate performance of the source range monitors is established from data taken with the operational neutron sources in place. During reactor operations, the SRM subsystem is calibrated and its performance is compared with criteria on noise, signal-to-noise ratio, and response to change incore reactivity

8. Intermediate Range Monitor (IRM) Calibration - The intermediate range monitors are initially calibrated to give useful readings and to supply protection for this phase of the test program. This initial calibration is made by comparing to SRM readings in the overlap region

9. Process Computer - As process variable signals become available to the computer, verification is made of these signals and of the computerized systems performance calculations

13.5.3 Heatup from Ambient to Rated Temperature and Pressure

Following satisfactory completion of the core loading and low power test program, the core components are visually verified for proper installation, and the additional in-vessel hardware is installed. This includes the steam separator and dryer assemblies and any special instrumentation such as vibration sensors or instrumented fuel assemblies. The reactor head is installed, followed by a hydrostatic test to assure satisfactory sealing of the vessel head. The drywell head is installed and shield plugs placed over it. Recirculation pumps are started at reduced speeds. A sequence of tests is performed to confirm a number of the nuclear steam supply

system characteristics as the temperature and pressure are increased. The following tests are conducted during this phase of the startup:

1. IRM Calibration - The IRM subsystem is recalibrated during heatup by making the IRM readings proportional to a known heat input to the reactor coolant from a nonnuclear heat source, such as the main recirculation pumps. The proportionality is determined by measuring the reactor coolant temperature rise produced by pump heating and by nuclear heating

2. SRM performance is determined by checking for proper overlap with the IRM subsystem

3. Reactor vessel temperatures are monitored during heatup and cooldown to determine that specified temperature differences are not excessive

4. System expansion checks are made during heatup to verify freedom of motion of major equipment and piping

5. Control rod drive system tests are made by measuring scram times on a selected number of drives at two intermediate pressures, scram times, and drive line friction tests on a representative set of drives at rated reactor pressure, and in-out driving times of selected rods during heatup

6. Control rod sequence to be used during the heatup is checked periodically for satisfactory performance

7. Radiation measurements are made periodically during nuclear heating and at rated temperatures

8. Chemical and radiochemical checks are made during heatup

9. Main steamline isolation valve functional tests are made at rated pressure

10. Core performance evaluations are made near or at rated temperature and pressure. This includes a reactor heat balance at rated temperature

11. Reactor pressure control is instituted using the turbine initial pressure regulator.

13.5.4 From Rated Temperature to 100 Percent Power

Reactor power is increased to 100 percent in increments of 25 percent by withdrawing control rods in the standard sequence. Major testing is performed at the 25 percent, 50 percent, 75 percent, and 100 percent power levels.

1. Chemical and radiochemical tests are continued.

2. Radiation measurements of limited extent are made at 25 percent of rated power and thorough surveys are made at 50 percent, 75 percent, and 100 percent power.

3. Vibration measurements are performed, if necessary, to determine the vibrational characteristics of reactor vessel internals of modified design, as reactor pressure is increased.

4. System expansion tests are continued on a limited basis as reactor power is increased.

5. Main steam isolation valve functional and operational tests are made as reactor power is increased.

6. RCIC system is actuated when the reactor is shut down but hot and pressurized, to demonstrate full capacity operation of the steam turbine driven pump.

7. HPCI system is tested to demonstrate proper performance of the system including the steam turbine driven pump.

8. Recirculation pump trips and their effects on the jet pumps and the reactor are tested periodically during power increase.

9. Flow control capabilities are determined at specified power levels.

10. Turbine trip tests are performed to determine speed and reactor response.

11. Generator trip tests are performed to determine speed and reactor response.

12. Pressure regulator tests are made to determine the response of the reactor and the turbine governor system. Regulator settings are optimized using data from this test.

13. Bypass valve measurements are performed by opening a turbine bypass valve and recording the resulting reactor transients. Final adjustments to the pressure regulators are made.

14. Feedwater pump trip tests are made to demonstrate reactor water level and station response to loss of part of feedwater supply. Reactor water level changes are made to determine reactor response and to optimize level controller settings.

15. Flux response to control rod movements is determined in both equilibrium and transient conditions. Steady state noise may be measured. Power void loop stability is verified from this data.

16. LPRM calibrations, which include use of the traversing incore probe (TIP) subsystem, are made at 50 percent, 75 percent, and 100 percent of rated power. Each local power range monitor is calibrated to read in terms of local fuel rod surface heat flux.

17. APRM calibrations are performed after making significant power level changes. Reactor heat balances form the bases of the calibrations of these average power range monitors.

18. Core performance evaluations are made periodically to demonstrate that the core is operating within allowable limits on maximum local surface heat flux and minimum critical heat flux ratio. That test included reactor heat balance determinations.

19. Calibrations of rods are performed to obtain reference relationships between control rod motion, reactor power, and steam flow in the specified control rod sequence.

20. Axial power distribution measurements are made with the traversing incore probe system after significant changes in power, control rod pattern, or flow rate. The TIP data are used for core performance evaluations and LPRM calibrations.

21. Rod pattern exchanges are demonstrated from one specified control rod sequence to the other at the highest practical reactor power.

22. IRM Calibration. The final calibration of the IRM system will be made in the APRM-IRM power overlap region subsequent to the calibration of the APRM system.

23. Auxiliary power loss tests shall be made to verify acceptable performance of the reactor and the electrical equipment and auxiliary systems during the resulting transients.

24. Electrical output and heat rate tests are performed to verify that the station will deliver the specified electrical output at an acceptable heat rate.

13.5.5 Restart Testing Following Recirculation Piping Replacement

The restart test program is performed following the replacement of the recirculation piping to verify that system performance is satisfactory for safe operation of the station at all expected operating conditions. During the restart program calibration of the affected systems based on the new recirculation system configuration is conducted or verified to be correct and piping expansion and vibration is monitored to confirm to design values. Plant specific

restart procedures are completed and a detailed schedule of testing is followed.

1. Heatup from Ambient to Rated Temperature and Pressure

   Following satisfactory installation of the replacement recirculation pipe, special instruments are installed in the drywell at preselected locations on the recirculation and the RHR piping to monitor piping vibration, expansion and strain. In the main control room, signal taps are installed to monitor selected process signals. The following tests are conducted during this phase of the restart program:

   a. Reactor vessel process temperatures are monitored during heatup to determine that specified temperature limits are not exceeded.

   b. System expansion checks are made during heatup to verify freedom of motion of major recirculation system equipment and piping.

   c. Strain measurements are taken at rated temperature and pressure to confirm design values.

   d. Nuclear instrumentation is monitored to verify proper operation following cable replacement after the outage.

   e. Safety Relief Valves are tested to verify proper discharge and that no blockage exists in the Safety Relief Valve Discharge lines.

2. From Rated Temperature to 100% Power

   Reactor power is increased to 100% in a controlled fashion with two major testing plateaus (1) along the 75% rod line, (2) along the 100% rod line. The following tests are conducted:

   a. Vibration measurements are performed to determine the vibration characteristics of the recirculation piping & RHR piping as reactor pressure is increased.

   b. Control system stability is demonstrated for both the feedwater and recirculation system controllers and controller tuneup is conducted if necessary.

   c. The 100% load line is reverified and new jet pump base line data is obtained.

   d. The jet pumps are calibrated based on the current jet pump riser flow distribution and current recirculation pipe design.

e. Recirculation pump trips are conducted to verify that vibration and system performance is within design values.

f. A cavitation search is conducted to verify that the low feedwater flow interlock is adequate to prevent jet pump and recirculation pump cavitation.

g. The recirculation pump high speed stops are reset based on the current recirculation system configuration.

13.5.6 Startup Test Restrictions

All operations and tests must comply with the safety limitations and limiting conditions for operations specified in the test procedures and remain within the limits of facility license. Restrictions are minimized because the prime objective of the startup program is to demonstrate that the station is capable of operating safely and satisfactorily up to full power. Any restrictions are detailed in the written startup procedures and in supporting instructions.

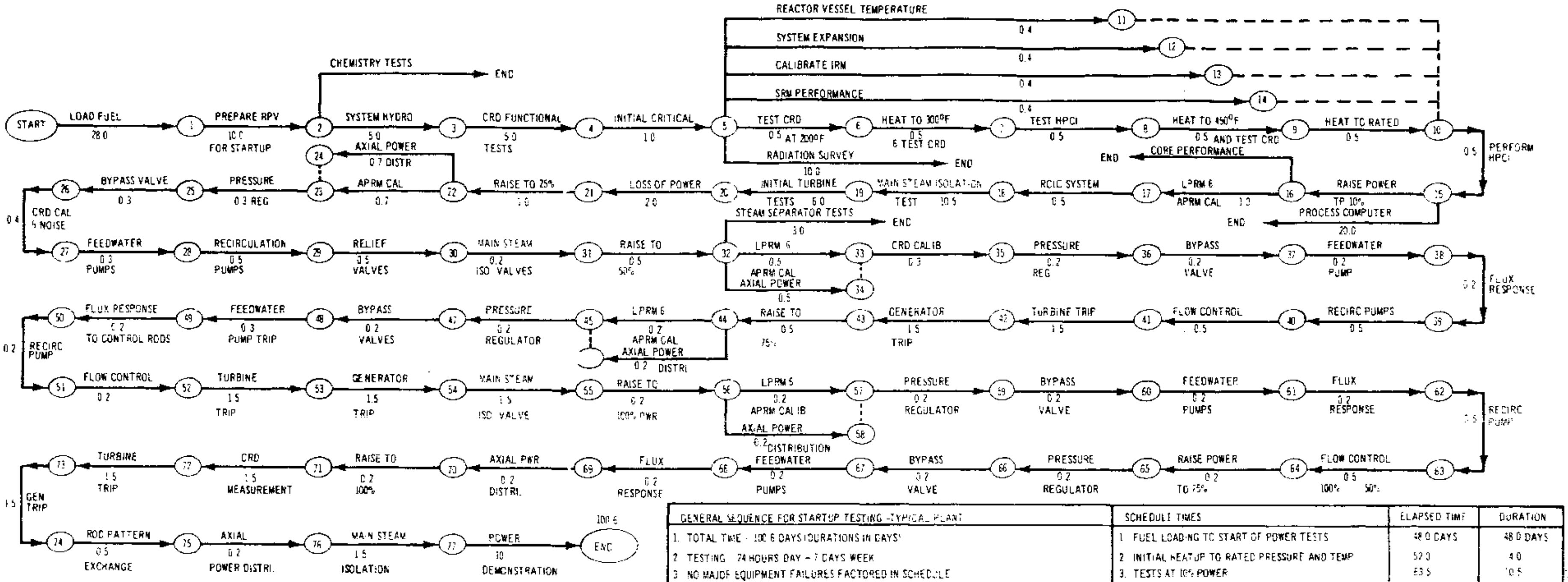
13.5.7 Startup Report [License Amendment 177]

The startup report is a summary of plant startup and power escalation testing submitted for NRC review only. A startup report is typically prepared as a result of a significant milestone action such as:

- an increase in licensed power level,
- the installation of nuclear fuel with a different design or manufacturer than the current fuel,
- modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the facility.

The report provides a mechanism for the NRC to review the appropriateness of Pilgrim activities.

The startup report shall be submitted to the NRC within 90 days following completion of the respective milestone.



GENERAL SEQUENCE FOR STARTUP TESTING -TYPICAL PLANT
1. TOTAL TIME - 100.6 DAYS (DURATIONS IN DAYS)
2. TESTING - 24 HOURS DAY - 7 DAYS WEEK
3. NO MAJOR EQUIPMENT FAILURES FACTORED IN SCHEDULE
4. NATURE OF TESTING LIMITS PARALLEL PATH TESTING
5. ASSUMED GENERATOR TURBINE ISOLATION VALVE TRIPS PERMITTED BY GRID WITHOUT EXTENSIVE DELAYS
6. SCHEDULE IS BASED ON GE EXPERIENCE GAINED FROM 5 TO 10 PREVIOUS STARTUPS
7. PRESENT AEC TESTING REQUIREMENTS

SCHEDULE TIMES	ELAPSED TIME	DURATION
1. FUEL LOADING TO START OF POWER TESTS	48.0 DAYS	48.0 DAYS
2. INITIAL HEATUP TO RATED PRESSURE AND TEMP	52.0	4.0
3. TESTS AT 10% POWER	63.5	10.5
4. TESTS AT 25% POWER	67.7	4.2
5. TESTS AT 50% POWER	73.8	6.1
6. TESTS AT 75% POWER	80.3	6.5
7. TESTS AT 100% POWER (INCLUDE FLOW VARIATION AND FINAL TRIPS)	90.6	10.3
8. POWER DEMONSTRATION	100.6	10.0

FIGURE 13.5-1

TYPICAL STARTUP TEST SEQUENCE

PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

13.6 STATION PROCEDURES

The operations manual, which includes all station procedures will include the following procedures:

- Administrative
- Power operations
- Maintenance and testing
- Fuel handling
- Emergency
- Radiation protection

Written procedures shall be established, implemented, and maintained covering the following activities (TS 5.4.1):

- The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;

- The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33;

- Quality assurance for effluent and environmental monitoring;

- Fire Protection Program implementation; and

- All programs specified in Section 5.5 of the Technical Specifications.

Temporary changes which clearly do not change the intent of the approved procedure, shall as a minimum be approved by two members of the plant staff knowledgeable in the areas affected by the procedures. At least one of these individuals shall be the supervisor in charge of the shift and hold a senior operator's license at PNPS (ANSI N18.7.1976, Section 5.2.2).

13.6.1 Administrative Procedures

The administrative procedures include control room procedure, station organization and responsibilities, operating criteria, records requirements, and routine work assignments.

13.6.2 Power Operations Procedures

Station operations will be conducted in accordance with power operations procedures. These procedures will cover all normal and foreseeable abnormal operating conditions.

These procedures will include the following:

1. Detailed check lists for all major systems, including safety and instrumentation systems, to ensure that all

necessary equipment is functioning and in the proper mode for startup

2. Detailed procedures for startup, normal operation, and shutdown of major pieces of equipment, systems, and integrated station operation

3. Alarm procedures to define the meaning of alarms and specify the action required

4. Procedures for abnormal operations

5. Chemical and radiochemical control procedures

13.6.3 Maintenance and Testing

Maintenance and testing is conducted in accordance with written maintenance and testing procedures.

These procedures include the following:

1. Routine testing of systems related to nuclear safety as required by the technical specifications.

2. Routine testing of systems not related to nuclear safety. The frequency of testing will follow normal steam plant practice.

3. Special testing. Special testing encompasses all testing not routinely done. For example:

    a. Operational testing of equipment after overhaul

    b. Testing of equipment for proposed changes to operational procedures

    c. Testing of equipment for proposed design changes in systems and/or equipment. Procedures will be written for all special testing and reviewed and approved by the Operations Review Committee.

4. Routine inspection and preventive maintenance.

5. Normal overhaul and repair. Written procedures will be provided where complexity dictates. All overhaul and repair work will be strictly performed within radiation protection procedures.

13.6.4 Fuel Handling Procedures

Fuel handling procedures are written by station personnel. These procedures are reviewed and approved by the Operations Review Committee. These procedures will be similar in nature to those used for initial fueling.

13.6.5 Emergency Procedures

There are two general categories of potential emergency situations that may arise at the Station: those that affect or have the potential impact on plant safety, and those with no potential impact on plant safety (e.g., medical emergencies). In each case, the situation may also have actual or potential radiological implications.

All emergencies that have the potential to affect plant safety, or which have actual or potential radiological implications are classified into one of the 10 CFR 50, Appendix E (IV.C) emergency classes:

Unusual Event;
Alert;
Site Area Emergency;
General Emergency.

Detailed information regarding the Pilgrim Station response to the declaration of any one of the four classes can be found in the Pilgrim Emergency Plan, which is included as Appendix N to this Final Safety Analysis Report.

Emergency situations which do not affect plant safety and which have negligible radiological implications are dealt with by implementing other approved Station procedures.

13.6.6 Radiation Protection Procedures

The radiation protection procedures are designed to provide for protection of station personnel and members of the public against exposure to radiation and radioactive materials in a manner consistent with 10 CFR 20.

These procedures include the following:

1. Measurement and evaluation of radiological conditions
2. Control of radioactive material
3. Access control for radiological areas
4. Monitoring of personnel for exposure to radiation
5. Monitoring of personnel for exposure to radioactive materials
6. Radiological controls for station operation, maintenance, and testing activities
7. Operation and calibration of instruments used for radiation detection
8. Control of radioactive sources
9. Handling, packaging, and shipping of radioactive materials
10. Implementation of the ALARA Program
11. Implementation of the respiratory protection program

13.7 RECORDS

13.7.1 Initial Tests and Operations

The records of the preoperational and startup test program will be compiled and retained with the station records to document initial station performance for reference and comparison with subsequent test results. These records will include the results of preoperational testing, initial fuel loading, low level core tests, and all startup and power tests prior to commercial operation.

13.7.2 Normal Operations

The primary records during normal operation will include the following:

1. Recorder charts
2. Written logs
3. Computer logs and/or operators' logs
4. Laboratory and exposure records
5. Administrative records and reports

Records of the pertinent operating conditions of the reactor, the auxiliary equipment, and the radwaste facility will be maintained in log books. The log books which will be kept are:

1. Reactor Log Book. This log book will contain information regarding current status of, and major changes in, control rod patterns, reactor power level, and other significant reactor conditions

2. Equipment and Station Status Log Book. This log book will contain information regarding the current status of, and major changes in, the station output and auxiliary equipment

3. Narrative Log Book. At the end of each shift, the nuclear watch engineer will summarize the overall station operation during that shift in this log book

4. Radwaste Log Book. This log book will contain records of the level and rate of release of radioactive materials, and any significant operations or changes in status of the associated equipment

Administrative records and reports will include weekly operating reports summarizing all pertinent operations, minutes of On-Site Safety Review Committee, and any safety review and audit Committee proceedings, plus all necessary reports required by the various sections of Title 10, Code of Federal Regulations.

The computer will print logs, consisting of pertinent operating parameters and calculated performance values, on an hourly, daily, and monthly basis. During periods when the computer is out of service, hourly readings from pertinent instrumentation will be logged by the operating staff.

13.7.3 Maintenance and Testing

Records of maintenance and testing performed within the station can be divided as specified in Section 13.6.3 and will be maintained.

13.7.4 Radiation Monitoring

Radiation Protection -- The following personnel exposure records will be maintained:

1. Daily records of dosimeter readings
2. Quarterly records of TLD data
3. Records of integrated exposure as required by applicable regulations

Radiation Conditions -- The following records of radiation conditions in accessible areas will be maintained:

1. Radiation levels
2. Contamination levels
3. Airborne activity

Radioactive Releases or Shipments -- Records of environmental radioactive releases and of radioactive shipments offsite will be maintained as required by applicable regulations.

13.7.5 Record Retention [License Amendment 177]

The following records shall be retained for at least five years:

1. Records of facility operation covering time interval at each power level.
2. Records of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.
3. Reportable Event Reports.
4. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.
5. Records of reactor tests and experiments.

6. Records of changes made to Operating Procedures.

7. Records of radioactive shipments.

8. Records of sealed source leak tests and results.

9. Records of annual physical inventory of all source material of record.

The following records shall be retained for the duration of the Operating License:

1. Record and drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.

2. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

3. Records of facility radiation and contamination surveys.

4. Records of radiation exposure for all individuals entering radiation control areas.

5. Records of the service lives of all hydraulic and mechanical snubbers listed in PNPS procedures including the date at which the service life commences and associated installation and maintenance records.

6. Records of gaseous and liquid radioactive material released to the environs.

7. Records of transient or operational cycles for those facility components designed for a limited number of transients or cycles.

8. Records of training and qualification for current members of the plant staff.

9. Records of in-service inspections performed pursuant to the Technical Specifications.

10. Records of Quality Assurance activities required by the QA Manual.

11. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10CFR50.59.

12. Records of meetings of the On-Site Safety Review Committee (OSRC) and the Off-Site Safety Review Committee (SRC).

13. Records for Environmental Qualification.

13.8 OPERATIONAL REVIEW AND AUDITS

13.8.1 Administrative Control

Under the ownership and operation of Boston Edison Company, the Nuclear Organization was responsible for the engineering, construction, and operation of Pilgrim Station.

The Boston Edison Company had established a project type organization reporting to the Senior Vice President, Nuclear, to direct all activities of Pilgrim Station. The Nuclear Organization was responsible for the quality assurance, engineering, construction, and operations activities. The organization directed the activities of all Boston Edison personnel who worked on the Pilgrim effort and was also responsible for coordinating and supervising the activities of Boston Edison's principal contractors for Pilgrim Station. The organization was responsible for coordinating the pre-operational and startup testing programs and for approval of test results and acceptance of station operating performance.

On July 13, 1999 the ownership and operation of Pilgrim Station was transferred from Boston Edison to the Entergy Nuclear Generation Company with the continuation of equivalent administrative controls.

13.8.2 Routine Reviews

Routine review of station operations and maintenance activities will be conducted by the supervisory and technical staff of the Pilgrim Nuclear Power Station. Support personnel provide additional support where necessary for the review of operating logs, recorded data, and performance records, and make recommendations for changes in station operating procedures if necessary. Proposed revisions in operating procedures will be referred to and reviewed by the On-Site Safety Review Committee. See Section 13.8.5.

The General Manger Plant Operation's organization is responsible for operating the station in strict compliance with the facility license and the Technical Specifications. The Design Engineering Organization recommends changes in the facility design or changes to the facility license based upon operating experience and station performance evaluations.

13.8.3 Deleted

13.8.4 Technical Assistance

The engineering group maintains several departments having personnel with expertise in generating station design, construction, and operational technology. The manager responsible for engineering is responsible for the development and maintenance of engineering programs, policies, and procedures, providing engineering services and for the evaluation of proposed changes to the facility license, major facility design changes, or major facility repairs. Different aspects of these responsibilities (e.g., fuel design) may be fulfilled by separate managers.

13.8.5 Safety Review Committees

Pilgrim Station has both On-Site and Off-Site Safety Review Committees that meet the operational requirements of ANSI-N18.7-1976 and the QAPM. The details of their operation are described in ANSI-N18.7, the QAPM and in implementing procedures.

SECTION 14

STATION SAFETY ANALYSIS

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STATION SAFETY ANALYSIS

14.1 INTRODUCTION

Previous sections of this report provide the objective design basis and description of each major system and component. Systems that have unique requirements arising from considerations of nuclear safety are evaluated in the safety evaluation portions of those sections of the report.

The objective of the station safety analysis is to evaluate the ability of the plant to operate without undue hazard to the health and safety of the public. In this section the effects of anticipated process disturbances, postulated component failures and operator errors are examined to determine their consequences and to evaluate the capability built into the plant to control or accommodate such failures.

Definitions for key terms used in this section are presented in Section 1, Introduction and Summary. A list of references is included at the end of this section.

14.2 REACTOR LIMITS

Limits on reactor operation are established to assure that the plant can be safely operated and not pose any undue risk to the health and safety of the public. This is accomplished by demonstrating that identified unacceptable results are avoided for normal operation, abnormal operation transients (special events) and postulated accidents.

14.2.1 Abnormal Operational Transients

14.2.1.1 Unacceptable Results

The following are considered to be unacceptable safety results for abnormal operational transients:

1. a release of radioactive material to the environs that exceeds the limits for 10CFR20;
2. reactor operation induced fuel cladding failure; and
3. nuclear system stresses in excess of that allowed for the transient classification by applicable industry codes.

14.2.1.2 Reactor Limits

To avoid the unacceptable safety results identified in Section 14.2.1.1, nucleate boiling (avoiding transition to film boiling) must be maintained and fuel cladding strain must be limited. Operating limits are specified to maintain adequate margin to the onset of the boiling transition and failure due to cladding strain. The figures of merit utilized for plant operation are the critical power ratio (CPR) and linear heat generation rate (LHGR).

The critical power ratio is defined as the ratio of the critical power (bundle power at which some point within the assembly experiences onset of boiling transition) to the operating bundle power. Thermal margin is stated in terms of the minimum value of the critical power ratio, MCPR, and corresponds to the most limiting bundle in the core. To ensure that adequate margin is maintained and an unacceptable result avoided, a design requirement based on a statistical analysis was selected as follows:

> Abnormal operational transients caused by a single operator error or equipment malfunction shall be limited such that, considering uncertainties in manufacturing and monitoring the core operating state, more than 99.9% of the fuel rods would be expected to avoid boiling transition.

This design requirement is called the Fuel Cladding MCPR Safety Limit. The MCPR Safety Limit is dependent upon the

fuel types loaded into the reactor core and are designated in Reference 1. The CPR for the limiting analyzed transient or fuel loading error event is added to the MCPRSL to obtain the operating limit MCPR (OLMCPR).

The LHGR limit is established such that the fuel rod cladding 1% plastic strain limit is not exceeded. The 1% plastic strain limit is discussed in Section 3.2 of this report.

Under transient conditions, Section III of the ASME Code requires that reactor pressures are not to exceed 110% of the design pressure (1.1 x 1250 = 1375 psig).

14.2.2 Postulated Design Basis Accidents

14.2.2.1 Unacceptable Results

The following are considered to be unacceptable results for postulated design basis accidents.

1. radioactive material release which results in dose consequences that exceeds the guideline values of 10CFR100;

2. failure of fuel cladding which would cause changes in core geometry such that core cooling would be inhibited;

3. nuclear system stress in excess of those allowed for the accident classification by applicable codes;

4. containment stresses in excess of those allowed for the accident classification by applicable industry codes when containment is required;

5. overexposure to radiation of station personnel in the control room.

14.2.2.2 Reactor Limits

To avoid the unacceptable results identified in section 14.2.2.1, limits are established such that:

1. fuel cladding temperature, oxidation fraction and hydrogen generation during loss of coolant accidents are maintained within the criteria of 10CFR50.46;

2. peak fuel rod enthalpy does not exceed the established safety limit;

3. MCPRSL is not exceeded for fuel bundle mislocation or misposition;

4. primary containment internal pressure does not exceed 62 psig; and

5. designated dose limits are not exceeded for any accident event.

The limits established in 10 CFR 50.46 for loss of coolant accidents are 2200°F peak cladding temperature, 17% of initial cladding thickness total oxidation, and 1% hydrogen generation from the chemical reaction of the cladding with the water. To assure that these limits are not exceeded, a maximum average planar LHGR (MAPLHGR) as a function of exposure is designated. Every fuel enrichment has a specific MAPLHGR limit which are designated in the supplemental reload licensing report (Appendix Q). During Single Loop Operation, the MAPLHGR thermal limit is adjusted by a reduction factor to account for the assumption of reduced core flow coastdown for recirculation line breaks.

The specific energy design limit for rod drop accident has been established as a peak fuel enthalpy of 280 cal/gm.

14.2.3 Special Events

14.2.3.1 Unacceptable Results

14.2.3.1.1 Shutdown from Outside the Control Room

The unacceptable safety results for this event are:

1. Inability to shut reactor down by local controls and equipment outside the control room.

2. Inability to bring the reactor to the cold shutdown condition from outside the control room.

14.2.3.1.2 Shutdown Without Control Rods

The unacceptable safety results for this event are:

1. Inability to shutdown the reactor independent of control rods.

2. The inability to automatically trip the recirculation pumps to reduce reactor power while shutting down the reactor independent of control rods.

14.2.3.2 Criteria

14.2.3.2.1 Shutdown from Outside the Control Room

1. It shall be possible to bring the reactor to a hot shutdown condition from any steady state normal condition by manipulating controls and equipment outside the control room.

2. The station design shall not preclude the ability to bring the reactor to a cold shutdown condition from a hot shutdown condition by using controls and equipment outside the control room.

14.2.3.2.2 Shutdown Without Control Rods

1. It shall be possible to shutdown the reactor by manual initiation of the Standby Liquid Control (SLC) System.

2. The station design shall include automatic tripping of the recirculation pumps (RPT) to promptly reduce power during abnormal transients prior to SLC operation.

14.3 METHOD OF APPROACH

A unique systematic approach to plant safety consistent with the technology base was developed. The key to this approach to plant safety was the station Nuclear Safety Operational Analysis (NSOA) document in Appendix G. Inputs into the NSOA were derived from the applicable regulations and industry codes and standards. The entire spectrum of events in the NSOA were evaluated to establish the limiting or design basis events. These events are then quantified in this section.

14.3.1 Nuclear Safety Operational Analysis

In the NSOA, all unacceptable safety results and all required safety actions were identified. The NSOA criteria were based on event probability. Thus, events more likely to occur were tested against more restrictive limits. This was consistent with industry practice and the applicable regulatory requirements.

The starting point for the nuclear safety operational analysis was the establishment of unacceptable safety results which are shown in Section 14.2. This concept enables the results of any safety analysis to be compared to acceptable criteria. Unacceptable safety results represents an extension of the general design criteria in 10CFR50 Appendix A for plant systems and components which are used as the basis for system design.

The focal point of the NSOA was the event analysis. In the event analysis, all essential protection sequences were evaluated until all required safety actions were successfully completed. The event analysis identified all required front line safety systems and their essential auxiliaries.

The full spectrum of initial conditions limited by the constraints placed on planned operation for transients, accidents, and plant capability demonstrations were evaluated. All events were analyzed until a stable condition is obtained. This assured that the event being evaluated did not have a characteristic for long-term consideration which is important.

In the event analysis all essential system, operator actions and limits to satisfy the required safety actions were identified. Limits were derived only for those parameters continuously available to the operator. Credit for operator action was taken only when an operator could be reasonably expected to perform the required action based on the information available to him.

In the NSOA, a complete and consistent set of safety actions was developed. These safety actions were those required to prevent unacceptable results. For transients and accidents, a single failure proof path to plant shutdown was shown. The application of a single failure criterion to these events was imposed as an additional measure of conservatism in the NSOA process.

14.3.2 Event Categorization

The spectrum of postulated initiating events developed from the nuclear safety operational analysis is divided into categories based upon the type of disturbance and the expected frequency of the initiating occurrence. The limiting events in each combination of category and frequency are quantitatively analyzed to demonstrate the capability of the plant to operate without the unacceptable safety results given in Section 14.2.

14.3.3 Event Categories

Each transient and accident event is assigned to one of the following applicable categories:

1. Decrease in Core Coolant Temperature - Reactor vessel water (moderator) temperature reduction results in an increase in core reactivity which could lead to fuel cladding damage.

2. Increase in Reactor Pressure - Nuclear system pressure increases threaten to rupture the reactor coolant pressure boundary. Increasing pressure also collapses the voids in the core-moderator, thereby increasing core reactivity and power level which threatened fuel cladding due to overheating.

3. Decrease in Reactor Core Coolant Flow Rate - A reduction in the core coolant flow rate threatens to overheat the cladding as the coolant becomes unable to adequately remove the heat generated by the fuel.

4. Reactivity and Power Distribution Anomalies - Transient events included in this category are those which cause rapid increases in power which are due to increased core flow disturbance events. Increased core flow reduces the void content of the moderator, thereby increasing core reactivity and power level.

5. Increase in Reactor Coolant Inventory - Increasing coolant inventory could result in excessive moisture carryover to the main turbine.

6. Decrease in Reactor Coolant Inventory - Reductions in coolant inventory could threaten the fuel as the coolant becomes less able to remove the heat generated in the core.

7. Radioactive Release from a Subsystem or Component - Loss of integrity of a radioactive containment component is postulated.

In addition a special event category was established. This category contains the shutdown from outside the control room and shutdown without control rod events. These events were analyzed for the initial core loading and are found in Appendix R.4. These events are not reanalyzed for reload cores.

14.4 ABNORMAL OPERATIONAL TRANSIENTS

The transients corresponding to the event categories in Section 14.3.3 are shown in Table 14.4-1. However, only a few of these transients would result in a significant reduction in MCPR or increase in LHGR.

To determine the limiting transient events, the relative dependency of CPR upon various thermal-hydraulic parameters was examined. A sensitivity study was performed to determine the effect of changes in bundle power, bundle flow, sub cooling, R-factor, and pressure on CPR for fuel designs. The R-factor is a weighted peaking factor used to characterize the local peaking pattern in the vicinity of the rod.

From this study it was determined that CPR is most responsive to fluctuations in the R-factor and bundle power. A slight sensitivity to pressure and flow changes and relative independence to changes in inlet sub cooling was also shown. The R-factor is a function of bundle geometry and local power distribution and is assumed to be constant throughout a transient. Therefore, transients which would be limiting because of MCPR would primarily involve significant changes in power. Based on this, the transients most likely to limit operation because of MCPR considerations are:

1. Generator load rejection without bypass or turbine trip without bypass

2. Loss of feedwater heating or inadvertent HPCI startup.

3. Feedwater controller failure (maximum demand).

4. Control rod withdrawal error.

Subsequent transient analyses verified the results of the above sensitivity study. Descriptions of the above limiting events are given below.

For reloads, the potentially limiting events are evaluated to determine the required operating limits. The analytical results for the limiting transients and the required operating limits are provided in the supplemental reload licensing report (Appendix Q).

14.4.1 Margin Improvement Options

The cycle specific transient analyses may be performed with the following margin improvement options:

1. ODYN Option B scram times and

2. Exposure dependent limits.

The ODYN Option B scram time improvement option uses generic, measured scram times that are more realistic than Technical Specification scram times to analyze core-wide pressurization transients. To use this option it must be demonstrated that the actual plant scram time distribution is consistent with the generic, measured scram time distribution. This is accomplished through an approved Technical Specification which consists of testing at the 5% significance level and allows adjustment of the MCPR operating limit if the actual scram speed distribution is outside the assumed distribution.

Exposure dependent limits may be established by repeating the transient analyses for selected mid-cycle exposures. The severity of any plant transient is typically worst at the end of the cycle primarily because the EOC all-rods-out scram curve gives the worst possible scram response. By analyzing the transients at other interim points in the cycle, a lower OLMCPR will be calculated and the plant may operate at these lower limits up until the analyzed exposure limit is reached. The plant then operates at the OLMCPR limit calculated for the next analyzed point. The transients which are most affected by the scram response are the increase in reactor pressure events and feedwater controller failure (maximum demand).

14.4.2 Operating Flexibility Options

The following operating flexibility options may be reflected in the cycle specific transient analyses:

1. Maximum Extended Operating Domain

2. ARTS

3. Feedwater Temperature Reduction.

4. Single Loop Operation (SLO)

The modified operating envelope termed Maximum Extended Operating Domain (MEOD) permits extension of operation into additional power/flow area, provides improved power ascension capability to full power and additional flow range at rated power, and includes an increased flow region to compensate for reactivity reduction due to exposure during an operating cycle. Overall, MEOD can be utilized to increase operating flexibility and plant capacity factor.

The extended load line region boundary of MEOD is typically limited to 75% of original core flow at 100% of original licensed thermal power and the corresponding power/flow constant rod line. The increased core flow region is limited by plant recirculation system capability, acceptable flow-induced vibration, fuel lift considerations, and force impact on the vessel internal components.

Evaluations performed for MEOD include normal and transient conditions, LOCA analysis, containment responses, stability, flow-induced vibration, and the effects of increased flow-induced loads on reactor internal components and fuel channels. The results of these analyses must be re-evaluated each cycle.

The ARTS improvement program is a comprehensive project involving the Average Power Range Monitor (APRM), the Rod Block Monitor (RBM), and Technical Specifications.

Implementing the ARTS improvement program provides for the following improvements which enhance the flexibility of the BWR during power level monitoring:

1. The APRM trip setdown requirement is replaced by a power-dependent MCPR operating limit similar to that used in the BWR6, and a flow-dependent MCPR operating limit to reduce the need for manual setpoint adjustments. Another set of LHGR power and flow dependent limits will be specified for more rigorous fuel thermal protection during postulated transients at off-rated conditions. In addition, another set of MAPLHGR power and flow dependent limits will be specified to provide protection of the fuel during postulated loss-of-coolant accidents at off-rated conditions. These power and flow dependent limits were verified for plant specific application during the initial ARTS licensing implementation and are applicable to subsequent cycles provided that there are no changes to the plant configuration as assumed in the licensing analyses. The power and flow correction factors applicable to the current spent fuel cycle are specified in the supplemental reload licensing report (Appendix Q.)

2. The RBM system is modified from flow-biased to power-dependent trips to allow the use of a new generic non-limiting analysis for the rod withdrawal error and to improve response predictability to reduce the frequency of nonessential alarms. The applicability of the RWE analysis for subsequent cycles must be verified as part of the general reload core design analysis.

The resulting improvements in the flexibility of the BWR provided by ARTS are designed to significantly minimize the time to achieve full power from startup conditions.

Analyses are performed in order to justify operations at a reduced feedwater temperature at rated thermal power. Usually, the analyses are performed for end-of-cycle operation with the last stage feedwater heaters out of service. However, throughout cycle operation, an appropriate feedwater temperature reduction can be justified by analyses at the appropriate operating conditions.

The limiting transients are reanalyzed for operation at a reduced feedwater temperature. In addition, the loss-of-coolant (LOCA), fuel loading error, rod drop accident, and rod withdrawal error are also re-evaluated for operation at a reduced feedwater temperature.

The increase in the feedwater nozzle fatigue usage factor must also be considered.

PNPS was licensed for continuous Single Loop Operation (SLO) beginning in Cycle 16 (Ref 19). The capability of operating at reduced power with a single recirculation loop is highly desirable in the event that maintenance of a recirculation pump or other components renders one loop inoperable. This operating flexibility offers a significant increase in plant availability. The SLO analysis evaluates the plant for continuous operation at a maximum expected power output during SLO, which is lower than that which is attainable for two-pump operation.

To justify SLO, safety analyses have to be reviewed for one-pump operation. The MCPR fuel cladding integrity safety limit. AOO analyses, operating limit MCPR, and non-LOCA accidents are evaluated. Increased uncertainties in the total core flow and TIP readings result in a small increase in the fuel cladding integrity safety limit MCPR.

SLO can also result in changes to plant response during a LOCA. These changes are accommodated by the application of reduction factors to the two-loop operation MPLHGR limit, MAPLHGR reductions factors are evaluated on a plant and fuel type dependent basis. In each subsequent reload, reduction factors are checked for validity and, if new fuel types are added, new reduction factors may be needed in order to maintain the validity of the SLO analysis. (Ref 20)

14.4.3 Generator Load Rejection without Bypass

Fast closure of the turbine control valves is initiated whenever electrical grid disturbances occur which result in significant loss of load on the generator. The turbine control valves are required to close as rapidly as possible to prevent overspeed of the turbine generator rotor. The closing causes a sudden reduction of steam flow which results in a nuclear system pressure increase. The reactor is scrammed by the fast closure of the turbine control valves.

14.4.3.1 Starting Conditions and Assumptions

The following plant operating conditions and assumptions form the principle bases for which reactor behavior are analyzed during a load rejection:

1. The reactor and turbine generator are initially operating at full power when the load rejection occurs.

2. All of the plant control systems continue normal operation.

3. Auxiliary power is continually supplied at rated frequency.

4. The reactor is operating in the manual flow control mode when load rejection occurs, although the results do not differ significantly for operation in the automatic flow control mode.

5. The turbine bypass valve system is failed in the closed position.

14.4.3.2 Event Description

Complete loss of the generator load produces the following sequence of events:

1. The power load unbalance device steps the load reference signal to zero and closes the turbine control valves at the earliest possible time. The turbine accelerates at a maximum rate until the valves start to close.

2. Reactor scram is initiated upon sensing control valve fast closure.

3. If the pressure rises to the safety/relief valve setpoint, these valves open and discharge to the suppression pool.

4. If pressure rises to approximately 1210 psig, the MG set drive motor breakers and generator field breakers trip.

5. If pressure rises to the spring safety valve setpoint, these valves open and discharge to the drywell.

14.4.4 Turbine Trip Without Bypass

A variety of turbine or nuclear system malfunctions will initiate a turbine trip. Some examples are: moisture separator and heater drain tank high levels, large vibrations, loss of control fluid pressure, low condenser vacuum and reactor high water level. The turbine stop valve closes causing a sudden reduction in steam flow which results in a nuclear system pressure increase and the shutdown of the reactor.

14.4.4.1 Starting Conditions and Assumptions

The plant operating conditions and assumptions are identical to those of the generator load rejection in Section 14.4.3.1.

14.4.4.2 Event Description

The sequence of events for a turbine trip is similar to those for a generator load rejection. Stop valve closure occurs over a period of 0.10 second.

Position switches at the stop valves sense the turbine trip and initiate reactor scram. If the pressure rises to the pressure relief setpoints, the relief function of the safety/relief valves open discharging steam to the suppression pool.

14.4.5 Loss of Feedwater Heating

Loss of feedwater heating results in a core power increase due to the increase in core inlet sub cooling.

14.4.5.1 Starting Conditions and Assumptions

The following plant operating conditions and assumptions form the principal basis for which reactor behavior is analyzed during the loss of feedwater heating transient:

1. The plant is operating at full power.

2. The plant is operating in the manual flow control mode. The transient is moderated by the runback in core flow if operation is in the automatic flow control mode.

14.4.5.2 Event Description

Feedwater heating can be lost in at least two ways:

1. Steam extraction line to heater is closed.

2. Feedwater is bypassed around heater.

The first case produces a gradual cooling of the feedwater. In the second case, the feedwater bypasses the heater and no heating of the feedwater is generated. In either case the reactor vessel receives cooler feedwater. The maximum number of feedwater heaters which can be tripped or bypassed by a single event represents the most severe transient for analysis considerations. The feedwater heaters are assumed to trip instantaneously. This event causes an increase in core inlet sub cooling, collapsing core coolant voids which increases core power due to the negative void reactivity coefficient. In automatic control some compensation of core power is realized by modulation of core flow.

In either case, power would increase at a very moderate rate. If power exceeded the normal full power flow control line, the operator would be expected to insert control rods to return the power and flow to their normal range. If this were not done, the neutron flux could exceed the scram set point where a scram would occur.

14.4.6 Inadvertent Start of HPCI Pump

The inadvertent HPCI startup event description was revised in the analysis of the Cycle to include the possibility that the Feedwater Control System would not respond to the effective increase in Feedwater flow (and reactor vessel level) associated with a HPCI startup prior to reaching the high level turbine trip set point.

Instead of a relatively simple power increase driven by the coldwater injection, the inadvertent HPCI startup is a power increase followed by a level turbine trip. Above bypass the turbine trip initiates a reactor scram.

This changes the HPCI event from a cold water injection to a more severe form of a Feedwater Controller Failure (FWCF) event. More severe because HPCI injection water is colder than excess Feedwater modeled in the FWCF resulting in a larger power excursion before the turbine trip. See App. Q for reference to the current analysis.
This event established the Operating Limit MCPR.

14.4.6.1 Starting Conditions and Assumptions

The HPCI system starts due to a malfunction or operator error. The Feedwater Control System does not respond sufficiently; resulting in rising water level in the reactor vessel.

14.4.6.2 Event Description

The relatively cold HPCI flow caused a slow rise in neutron flux and thermal power, as well as a slow rise in reactor vessel level. The power and level increases continue until the reactor vessel high level trip set point is reached causing a turbine trip. Above bypass, the turbine trip initiates a scram. The event is similar to the Feedwater Controller Failure – Maximum Demand.

14.4.7 Feedwater Controller Failure - Maximum Demand

This event is postulated on the basis of a single failure of a control device, specifically one which can directly cause an increase in coolant inventory by increasing the Feedwater flow. The most severe applicable event is a Feedwater controller failure during maximum flow demand. The Feedwater controller is forced to its upper limit at the beginning of the event.

14.4.7.1 Starting Conditions and Assumptions

The starting conditions and assumptions considered in this analysis are as follows:

1. Feedwater controller fails during maximum flow demand.
2. Maximum feedwater pump run out is assumed.
3. The reactor is operating in a manual flow control mode which provides for the most severe transient.
4. Since the high level reactor feed pump trip uses 2 out of 2 taken once logic, failure of one channel to indicate high, does not cause the reactor feed pump trip. For this reason, this trip is not used in transient analysis beginning with Cycle 21.

14.4.7.2 Event Description

A feedwater controller failure during maximum demand produces the following sequence of events:

1. The reactor vessel receives an excess of feedwater flow.
2. This excess flow results in an increase in core sub cooling, which reduces the void fraction and thus induces an increase in reactor power and in the reactor vessel water level.
3. The rise in the reactor vessel water level eventually leads to a high water level turbine trip and reactor scram trip.
4. If the pressure rises to the safety/relief valve setpoint, these valves open and discharge to the suppression pool
5. If pressure rises to approximately 1210 psig, the MG set drive motor breakers and generator field breakers trip.
6. If pressure rises to the spring safety valve setpoint, these valves open and discharge to the drywell.

14.4.8 Rod Withdrawal Error

Rod withdrawal error results in a core power increase due to positive reactivity insertion. The results reported in the plant

supplemental reload submittal are either plant/cycle specific or from the generic rod withdrawal error analyses described in Reference 1. If the generic analysis value is reported, it will be designated as generic in the plant supplemental reload submittal.

14.4.8.1 Starting Conditions and Assumptions

The reactor is operating at a power level above 75% of rated power at the time the control rod withdrawal error occurs. The reactor operator has followed procedures and up to the point of the withdrawal error is in a normal mode of operation (i.e., the control rod pattern, flow set points, etc., are all within normal operating limits). For these conditions, it is assumed that the withdrawal error occurs with the maximum worth control rod. Therefore, the maximum positive reactivity insertion will occur.

14.4.8.2 Event Description

While operating in the power range in a normal mode of operation, the reactor operator makes a procedural error and withdraws the maximum worth control rod to its fully withdrawn position. Due to the positive reactivity insertion, the core average power and local power increase causing a LPRM alarm. The event ends with a rod block by the RBM.

Table 14.4-1

LIST OF TRANSIENT EVENTS

Decrease in Core Coolant Temperature

1. Loss of Feedwater Heating
2. Inadvertent Start of HPCI Pump
3. Feedwater Controller Failure, Maximum Demand
4. Pressure Regulator Failure - Maximum Output
5. Inadvertent Opening of Safety or Relief Valve
6. RHR Shutdown Cooling Malfunction - Decreasing Temperature

Increase in Reactor Pressure

1. Pressure Regulator Failure - Zero Output
2. Generator Load Rejection with Bypass
3. Generator Load Rejection without Bypass
4. Turbine Trip with Bypass
5. Turbine Trip without Bypass
6. Inadvertent MSIV Closure
7. Loss of Condenser Vacuum
8. Loss of Auxiliary Power Transformer
9. Loss of All Feedwater Flow

Decrease in Reactor Coolant System Flow Rate

1. Trip of One Recirculation Pump Motor
2. Trip of Two Recirculation Pump Motors
3. Recirculation Flow Control Failure - Decreasing Flow

Table 14.4-1 (Cont)

Reactivity and Power Distribution Anomalies

1. Rod Withdrawal Error - Refueling
2. Rod Withdrawal Error - Startup
3. Rod Withdrawal Error - At Power
4. Abnormal Startup of Idle Recirculation Loop
5. Recirculation Control Failure - Increasing Flow

Increase in Reactor Coolant Inventory

1. Inadvertent Start of HPCI Pump (See Decrease in Core Coolant Temperature Events)

14.5 POSTULATED DESIGN BASIS ACCIDENTS

The abnormal operating transients documented in Section 14.4 are evaluated to determine the normal plant operating MCPR limit and compliance with the LHGR 1% plastic strain limit. In addition to these analyses, evaluations of less frequent postulated events are made to assure an even greater depth of safety. Accidents are events which have a projected frequency of occurrence of less than once in every one hundred years for every operating BWR. The broad spectrum of postulated accidents is covered by four categories of design basis events. These events are as follows:

1. Decrease in Reactor System Flow Rate - Recirculation Pump Seizure,

2. Reactivity and Power Distribution Anomalies - Control Rod Drop Accident and Loading Error Accident,

3. Decrease in Reactor Coolant Inventory - Steam Line Break and Loss of Coolant Accident, and

4. Radioactive Release from Subsystem or Component - Fuel Handling Accident.

The recirculation pump seizure and misplaced fuel bundle events were analyzed as abnormal operating transients in the initial core evaluation. Since that time, these events have been recategorized as accidents.

As documented in Reference 1, only some of the above accidents are reanalyzed for each reload cycle. These include control rod drop accident and misoriented fuel bundle. The loss of coolant accident analysis is performed only when a new bundle enrichment or new fuel is placed in the core. These three events, the steam line break accident, and the fuel handling accident are addressed below.

14.5.1 Control Rod Drop Accident

There are many ways of inserting reactivity into a boiling water reactor; however, most of them result in a relatively slow rate of reactivity insertion and therefore pose no threat to the system. It is possible, although unlikely, that a rapid removal of a high worth control rod could result in a potentially significant excursion. Therefore, the accident which encompasses the consequences of a reactivity excursion is the control rod drop accident.

The drop of the control rod results in a high local reactivity in a small region of the core and for large, loosely coupled cores like PNPS, significant shifts in the spatial power generation during the course of the excursion. Therefore, the method of analysis must be capable of accounting for any possible effects of the power distribution shifts.

Analysis of this accident is performed at various reactor operating states; the key reactivity feedback mechanism affecting the termination of the initial prompt power burst is the Doppler Reactivity Coefficient. Final shutdown is achieved by scramming all but the dropped rod. The methods utilized to evaluate the rod drop accident are documented in Reference 1. The limit for this event is 280 cal/gm.

14.5.1.1 Sequence of Events

For this accident, the reactor is assumed to be at a control rod pattern corresponding to the maximum incremental rod worth. The rod worth minimizer or operators are functioning within the constraints of the Banked Position Withdrawal Sequences (BPWS). The control rod that will result in the maximum incremental reactivity worth addition at any time in core life, under any operating condition while employing the BPWS, becomes decoupled from the control rod drive.

The operator selects and withdraws the drive of the decoupled rod along with the other control rods assigned to the Banked-Position group such that the proper core geometry for the maximum incremental rod worth exists. The decoupled control rod sticks in the fully inserted position.

The control rod then becomes unstuck and drops at the maximum velocity determined from experimental data (3.11 fps). The reactor goes on a positive period and the initial power burst is terminated by the Doppler Reactivity Feedback. The APRM 120% power signal scrams the reactor. The MSIV's, Steam Line Drain Isolation Valves, and Reactor Water Sample Valves remain open. The Mechanical Vacuum Pump receives an auto trip signal.

14.5.1.2 Analytical Methods and Results

Techniques and models used to analyze the control rod drop accident (CRDA) are documented in Reference 1. Analytical results from BPWS plants like PNPS have been statistically analyzed. The results of this statistical analysis show that, in all cases, the peak fuel enthalpy in a CRDA would be much less than the 280 cal/gm event limit even with a maximum incremental rod worth corresponding to 95% probability at the 95% confidence level. The details of this analysis are given in Reference 1.

14.5.1.3 Radiological Consequences

The analysis for removing the MSIV isolation function was performed by General Electric in NEDO-31400A, Safety Evaluation for Eliminating the Boiling Water Reactor Main Steam Line Isolation Valve Closure Function and Scram function of the Main Steam Line Radiation Monitor. Input values were collected from all participating utilities (includes PNPS) to consider the most bounding case of the effects of removing the MSIV isolation function. The analysis considered the offsite dose consequences for 2 release scenarios.

1) A CRDA where the source term is not reduced, even though the MSIVs close, and the radionuclides enter the condenser at atmospheric pressure to leak directly to the environment.

2) A CRDA where the MSIVs do not close and the activity is processed through the AOG and released via the main stack.

The consequences for the CRDA at PNPS were evaluated using PNPS-specific assumptions and parameter values. The source term is based on the failure of 1200 rods for GE14 fuel. Conservatively, the maximum radial peaking factor that is expected during the operating fuel cycle was applied to all affected rods. The approach is as used in NEDO 31400 and as outlined in Standard Review Plan (SRP) 15.4.9 "Spectrum of Rod Drop Accidents (BWR)."

For the first scenario in which the radioactivity leaks directly from the condenser to the environment, the estimated consequences are 3.7 rem to the thyroid and 0.03 rem to the whole body.

In the event of the second scenario, in which the MSIVs do not close, the offgas pre-treatment or post-treatment monitors would automatically isolate the main stack prior to any release. The off-gas monitors are required by Technical Specifications and the Offsite Dose Calculation Manual to be in continuous operation to isolate the main stack in the event of a noble gas release rate greater than the setpoint value used for normal plant effluent releases. In the event of a CRDA the activity release rate would be significantly greater than that allowed during normal plant operation. The monitors would, therefore, isolate the main stack. The accident activity released from the fuel would be contained in the condenser and the consequences would be as determined for the condenser release scenario.

The CRDA activity release via the AOG system and main stack is highly unlikely at PNPS. However, since the off-gas monitors are not safety-related, a conservative evaluation assuming such a scenario was performed. It was assumed that the AOG system is in service but that the monitors fail to isolate the main stack. Conservatively, the AOG charcoal delay bed hold-up times for noble gases was assumed to be zero. A conservative AOG system flow rate was also used. For this scenario the estimated consequences are still well below the limits established in SRP 15.4.9, which were used in the safety evaluation report as the bases for accepting NEDO 31400. Therefore, the AOG system is not required to mitigate the consequences of a CRDA.

NEDO-31400A recognized that early vintage BWRs like Pilgrim operate at power levels above the mechanical vacuum pump capability while AOG is bypassed. NEDO-31400A states that this operating mode is acceptable because the pretreatment radiation monitors' set points are established to automatically isolate the effluent pathway before Technical Specification dose rate limits are exceeded. If a CRDA occurred while operating in the AOG bypass mode, the resulting offsite dose is expected to be similar to the condenser leakage scenario discussed in this section (i.e., that dose would be a small fraction of 10 CFR 100 dose limits).

The NEDO-31400A safety evaluation did not address removing any other trip functions from the Main Steam Line Radiation Monitors. The other possible trip functions from the Main Steam Radiation monitoring are as follows:

1) Trip the Mechanical Vacuum Pump.
2) Close the Main Steam Line Drain Isolation Valves
3) Close the Reactor Sample Isolation Valves

The Mechanical Vacuum Pump trip function is operable at PNPS. The other trip functions are not operable at PNPS as explained below.

There is no effect on the off-site dose as a result of the main steam line drain valves remaining open during a CRDA since the piping is also routed to the condenser. The source term in the condenser is unaffected because no plate out or condensation of the source term from the reactor to the condenser is assumed in the NEDO-31400A analysis. The occurrence of these phenomena in the drain lines would tend to diminish the condenser source term.

The reactor water sample enters the Reactor Building as a 1-inch line in the Reactor Water Cleanup Heat Exchanger Room. The line splits off to a 1/2-inch line to the Crack Arrest Verification System (CAVS) and 1/2-inch line to the Reactor Water Sample Panel (C121). There is a 1/2-inch line from the CAVS that goes to a sample panel (C136) and drains to the Reactor Water sample panel drain.

The conservative assumptions used in calculating the dose contribution from the open reactor water sample valves are as follows:

1. The valves are assumed to be open for 2 hours before action is taken by the operator.

2. The same fraction of halogens get vented to the atmosphere from the sample line as from the condenser. This assumes no condensation occurs. However, the process stream that goes to the drain first goes to a sample panel cooler and has an outlet temperature of approximately 77°F. Therefore, the actual fraction of halogens that get vented to the atmosphere will be very small.

The estimated contributing doses from this drain are 20.1 rem thyroid and 7.0 E-03 rem whole body.

Another contribution to offsite dose during a CRDA is from the Turbine Gland Seal Condenser Exhausters. This source draws steam from the steam chest of the Main Steam System and supplies it to the gland seals of the turbine. There is a separate condenser for this steam that is mixed with air and then exhausted to the same effluent path as the mechanical vacuum pump. The amount of steam existing through this path depends on the clearances of the packing on the turbine seals. The maximum seal clearance was assumed when considering the offsite dose contribution during a CRDA. Also, no condensation of the steam is assumed after it leaves the exhausters and is released to the environment. The actual process flow is through piping that contains 2 loop seals to collect condensation before being released to the stack. The estimated dose contribution from this source was 0.64 Rem thyroid and 1.3E-02 Rem whole body.

The offsite dose from all sources during a CRDA is totaled below (reference 14.7.11):

Source	Thyroid	WB
Condenser	3.7	3.0E-02
RX Sample Line	20.1	7.0E-03
Gland Seals	0.64	1.3E-02
TOTAL	24.4	5.0E-02

At PNPS, the worst case CRDA most probably would occur during mechanical vacuum pump operation. The mechanical vacuum pump would trip and the radionuclides would be trapped in the condenser. Therefore the condenser leakage scenario is bounding for PNPS.

The CRDA whole body dose for PNPS is less than the NEDO-31400A of 0.31 Rem. The PNPS CRDA thyroid dose is greater than the NEDO-31400A value of 4.3 Rem because of the PNPS site specific added contributions from the reactor sample line and gland seals. However, the total offsite dose for a CRDA is less than the SRP 15.4.9 limits of 75 Rem thyroid and 6 Rem whole body.

NEDO-31400A considered the failure of 850 fuel rods of the 8x8 configuration. For 9x9 fuel, approximately 1000 fuel rods are expected to fail at the same level of deposited energy due to the postulated accident. However, the radiological consequences for the 9x9 fuel designs are the same as for 8x8 fuel designs due to an offsetting lower plenum activity (per rod).

14.5.2 Loading Error Accident

One of the events which is evaluated each cycle is the fuel bundle loading error. The probability of a significant fuel assembly loading error is much less than once in a plant lifetime and requires multiple operator errors to occur. A loading error in the core configuration is defined as one of the following:

1. A fuel bundle is inserted in an improper location (mislocated bundle accident); or

2. A fuel bundle is loaded in an improper orientation, i.e., rotated 90 or 180 degrees (misoriented bundle accident).

The results of this accident must not exceed the Fuel Cladding Integrity MCPR Safety Limit; therefore, there are no radiological consequences.

14.5.2.1 Mislocated Bundle Accident

Mislocated bundle analyses are not performed for reload cores because, based on an analysis of data available from past reloads, the probability that a mislocated fuel bundle loading error will result in a CPR less than the safety limit is sufficiently small that plant/cycle specific analyses are not necessary. Details of this analysis are provided in Reference 1.

14.5.2.2 Misoriented Bundle Accident

Proper orientation of fuel assemblies in the reactor core is readily verified by visual observation and assured by verification procedures during core loading. Five separate visual indications of proper fuel assembly orientation exist:

1. The channel fastener assemblies, including the spring and guard used to maintain clearances between channels, are located at one corner of each fuel assembly adjacent to the center of the control rod.

2. The identification boss on the fuel assembly handle points toward the adjacent control rod.

3. The channel spacing buttons are adjacent to the control rod passage area.

4. The assembly identification numbers which are located on the fuel assembly handles are all readable from the direction of the center of the cell.

5. There is cell-to-cell replication.

Experience has demonstrated that these design features are clearly visible so that any misoriented fuel assembly would be readily identifiable during core loading verification.

Analysis methods for the misoriented fuel assembly are given in Reference 1. A penalty of 0.02 CPR to account for tilting of the misoriented bundle is added to the calculated CPR used in determining the operating limit MCPR. The misoriented bundle accident is evaluated on a cycle specific basis.

14.5.3 Loss of Coolant Accident

Break of a large recirculation pipe represents the limiting pipe break inside the containment. This event has been analyzed quantitatively in Section 6.5. The following is a discussion of the containment analysis and radiological consequences. Assumptions used in these analyses are given in Appendix R.6 and below.

Two ultimate heat sink (UHS) temperatures (65°F and 75°F) are presented in Loss-of-Coolant-Accident, Primary Containment and ECCS Pump NPSH analysis contained Section 14.5.3. The 65°F analysis represents the original design and licensing value. The 65°F analysis information was retained in the FSAR for its historical value and because it was the original basis for the sizing and selection of the containment heat removal systems.

By license amendment (Ref. 12 & 13), the design and licensing basis maximum UHS temperature was raised from 65°F to 75°F. The current operational limit in the Technical Specifications is 75°F. Therefore, the design and licensing value for the maximum UHS temperature is defined at 75°F, to ensure plant operation is not limited below 75°F and that all safety functions directly or indirectly dependent on UHS temperature can be satisfied up to 75°F.

14.5.3.1 Primary Containment Response

14.5.3.1.1 Initial Conditions and Assumptions

The following assumptions and initial conditions were used in the calculation of the effects of a LOCA on the primary containment. The plant response to the accident can be separated into two distinct phases: the short term response and the long-term recirculation phase. The short-term response includes that period of time in the accident up to 600 seconds when initiation of containment cooling is assumed. The peak drywell and wetwell airspace temperatures occur in this period of time and are not influenced by the performance of containment cooling. The long-term recirculation phase of the accident response is defined to begin at 600 seconds with the initiation of containment cooling and continue past the peak suppression pool temperature to the point of minimum NPSH margin.

Historically, the primary containment response has been established using the design value of 65°F for the SSW inlet temperature to the RBCCW heat exchanger. The following discussion of containment response includes analysis performed for the DBA LOCA using a site maximum SSW injection temperature of 75°F. In the following discussion the analysis that used a 75°F SSW injection temperature is referred to as the 75°F SSW Case, likewise the analysis based on 65°F SSW injection temperature is referred to as the 65°F SSW Case.

1. The reactor is operating at full power with all valves in the recirculation system open. Initial power for the 75°F SSW Case was increased to 102% consistent with the current standard based on the requirements of Regulatory Guide 1.49.

65°F SSW Case	75°F SSW Case
1998 MWt	2038 MWt

2. The reactor is assumed to go subcritical at the time of accident initiation due to void formation in the core region. Scram also occurs in less than 1 sec from receipt of the high drywell pressure signal, but the difference in shut down time between 0 and 1 sec is negligible.

3. The sensible heat released in cooling the fuel to 545°F (the normal primary system operating temperature) and the core decay heat were included in the reactor vessel depressurization calculation. The rate of energy release was calculated using a conservatively high heat transfer coefficient throughout the depressurization. Because of this assumed high energy release rate, the vessel is maintained at near rated pressure for almost 6 sec. The high vessel pressure increases the calculated flow rates out of the break; this is conservative for containment analysis purposes. With the vessel fluid temperature remaining near 545°F, however, the release of sensible energy stored below 545°F is negligible during the first 6 sec. The later release of this sensible energy does not affect the peak drywell pressure. The small effect of this energy on the end of transient suppression pool temperature is included in the calculations.

4. The main steam line isolation valves were assumed to start closing at 0.5 sec after the accident, and the valves were assumed to be fully closed in the shortest possible time of 3 sec following closure initiation. Actually, the closures of the main steam line isolation valves are expected to be the result of reactor low-low water level, so these valves may not receive a signal to close for over 4 sec, and the closing time could be as high as 10 sec. By assuming rapid closure of these valves, the reactor vessel is maintained at a high pressure, which maximizes the discharge of high energy steam and water into the primary containment, which in turn maximizes the loading on the containment.

5. For both the short and long-term analysis in the 65°F SSW Case, the feedwater flow was assumed to stop instantaneously at time zero. This conservatism is used because the relatively cold feedwater flow, if considered to continue, tends to depressurize the reactor vessel, thereby reducing the discharge of steam and water into the primary containment.

   Short-term containment response in the 75°F SSW Case is consistent with the 65°F SSW Case. For the 75°F SSW Case long-term analysis, feedwater flow into the RPV continues until the high-energy feedwater (above feedwater enthalpy of 201 BTU/lbm) is injected into the reactor vessel. This assumption is conservative for the long-term suppression pool temperature analysis because additional energy is added to the reactor vessel and containment.

6. The vessel depressurization flow rates were calculated using Moody's critical flow model[2] assuming "liquid only" outflow because this maximizes the energy release to the containment. "Liquid only" outflow means that all vapor formed in the vessel due to bulk flashing rises to the surface rather than being entrained in the exiting flow. Some entrainment of the vapor would occur and would significantly reduce the reactor vessel discharge flow rates. Moody's critical flow model, which assumes annular, isentropic flow, thermodynamic phase equilibrium, and maximized slip ratio, accurately predicts vessel outflows through small diameter orifices. However, actual flow rates through larger flow areas are less than the model indicates due to the effects of a near homogeneous two phase flow pattern and phase nonequilibrium. These effects are in addition to the reduction due to vapor entrainment discussed above.

   For the 75°F SSW Case, the vessel depressurization rates were calculated using the homogeneous equilibrium critical flow model described in NEDO-21052, "Maximum Discharge of Liquid-Vapor Mixtures from Vessels." (Reference 6).

7. The pressure response of the containment is calculated assuming:

   a. Thermodynamic equilibrium in the drywell and suppression chamber. Because complete mixing is nearly achieved, the error introduced by assuming complete mixing is negligible and in the conservative direction.

   b. The constituents of the fluid flowing in the drywell to suppression chamber vents are based on a homogeneous mixture of the fluid in the drywell. The consequences of this assumption result in complete liquid carryover into the drywell vents. Actually, some of the liquid will remain behind in a pool on the drywell floor so that the calculated drywell pressure is conservatively high.

c. The flow in the drywell suppression pool vents is compressible except for the liquid phase. In the development of the drywell flow model, it is noted that the mass fraction of liquid in the drywell is on the order of 0.60, while the volumetric fraction is only about 0.005. This fact resulted in the following interpretation of the flow pattern. The liquid is in the form of a fine mist that is carried along by the predominately steam air flow and does not affect the flow except to add inertia to the flowing fluid. Except for the corrections to account for the liquid inertia, flow is treated as compressible flow of an ideal gas in a duct with friction. The loss coefficients of the Vent/Header/Downcomer System are lumped as an equivalent length of pipe.

   The accuracy of this interpretation of the effects of liquid carryover is supported primarily by a series of tests performed as part of the Humboldt Bay series of pressure suppression tests.(3) In this series of tests, changes in the drywell geometry resulted in variation in the amount of liquid carryover achieved. The liquid remaining in the drywell at the end of the test was measured and recorded. These tests were performed with a relatively small diameter orifice in the reactor vessel so that the reactor vessel outflow can be calculated accurately using Moody's critical flow model.(2) On Figure 14.5-1 the calculated and measured pressure responses for these tests are shown. Note that with 100 percent carryover the agreement is excellent. In this test, the drywell was preheated to 184°F before the steam water mixture was introduced to the drywell; the preheating prevented any condensation on the drywell walls. A calculated response assuming condensation but no carryover is also shown on Figure 14.5-1, and the agreement with the measured response with no carryover is excellent.

d. No heat loss from the gases inside the primary containment is assumed. The model is compared against the Bodega Bay test data for two of the smaller orifices tested on Figures 14.5-2 and 14.5-3. As can be seen in the figures, the reactor vessel depressurization model accurately predicted the results of these tests. However, the predicted drywell pressure response is slightly higher than the test results. The over prediction is believed to be due to a combination of:

   No condensation assumed in calculated response,

   Slight over prediction of reactor vessel discharge flow rates, and

Incomplete liquid carryover into the drywell vents.

As the chosen size of the vessel orifice increases, the vessel depressurization rate is over-predicted and the over prediction of drywell pressure increases. This trend is illustrated on Figure 14.5-4, where calculated and measured drywell peak pressure is compared. In no case did the model underpredict the test data.

14.5.3.1.2 Containment Response

65°F SSW Case

The calculated pressure and temperature responses of the containment are shown on Figures 14.5-5, 14.5-6, and 14.5-7. Figure 4.5-5 shows that the calculated drywell peak pressure is 45 psig, which is well below the maximum allowable pressure of 62 psig. After the discharge of the primary coolant from the reactor vessel into the drywell, the temperature of the suppression chamber water approaches 130°F (Figure 14.5-7), and the primary containment pressure stabilizes at about 27 psig, as shown on Figure 14.5-5. Most of the noncondensible gases are forced into the suppression chamber during the vessel depressurization phase. However, the noncondensibles soon redistribute between the drywell and the suppression chamber via the vacuum-breaker system as the drywell pressure decreases due to steam condensation.

The core spray system removes decay heat and stored heat from the core, thereby controlling core heatup and limiting metal water reaction to less than 0.1 percent. The core spray water transports the core heat out of the reactor vessel through the broken recirculation line in the form of hot water. This hot water flows into the suppression chamber via the drywell to suppression chamber vent pipes. Steam flow is negligible. The energy transported to the suppression chamber water is then removed from the primary containment system by the residual heat removal system (RHR) heat exchangers.

Prior to activation of the RHR containment cooling mode (arbitrarily assumed at 600 sec after the accident), the RHR pumps (low pressure coolant injection (LPCI) mode) have been adding liquid to the reactor vessel along with the core spray. After the reactor vessel is flooded to the height of the jet pump nozzles, the excess flow discharges through the recirculation line break into the drywell. This flow, in addition to cooling the fuel, offers considerable cooling to the drywell and causes a depressurization of the containment as the steam in the drywell is condensed. At 600 sec, the RHR pumps are assumed to be switched from the LPCI mode to the containment cooling mode. The containment spray would normally not be activated at all and the changeover to the containment cooling mode need not be made for several hours. There is considerable time available to place the containment cooling system in operation because about 8 hr will pass before the containment design pressure is reached, assuming no containment cooling.

To access the primary containment long term response after the accident, an analysis was made of the effects of various containment spray and containment cooling combinations. For all cases, one of the core spray loops is assumed to be in operation. The long term pressure and temperature response of the primary containment was analyzed for the following containment spray and cooling conditions:

Case A - Operation of both RHR cooling loops with two residual heat removal (RHR) pumps and two RHR heat exchangers in suppression pool cooling mode. No containment spray.

Case B - Operation of one RHR cooling loop with one RHR pump and one RHR heat exchanger in suppression pool cooling mode. No containment spray.

Case C - Operation of one RHR cooling loop with one RHR pump and one RHR heat exchanger in containment spray mode.

The initial pressure response of the containment (the first 30 sec after break) is the same for each of the conditions. During the long term containment response (after depressurization of the reactor vessel is complete) the suppression pool is assumed to be the heat sink in the containment system. The effects of decay energy, stored energy, and energy from the metal water reaction on the suppression pool temperature are considered.

Case A

This case assumes that both RHR loops are operating at design heat removal capacity. This includes two RHR heat exchangers, two RHR pumps, and design values of cooling water flow to both RHR loops operating in the suppression pool cooling mode. The RHR pumps draw suction from the suppression pool and pump water through the RHR heat exchangers and back into the suppression pool. This forms a closed cooling loop with the suppression pool. This suppression pool cooling condition is arbitrarily assumed to start at 600 sec after the accident. Prior to this time the RHR pumps are used to flood the core (LPCI mode).

The containment pressure response to this set of conditions is shown as curve "a" on Figure 14.5-5. The corresponding drywell and suppression pool temperature responses are shown as curves "a" on Figures 14.5-6 and 14.5-7. After the initial rapid depressurization, energy addition due to core decay heat results in a gradual pressure and temperature rise in the containment. When the energy removal rate of the RHR exceeds the energy addition rate from the decay heat, the containment pressure and temperature begin to decrease. Table 14.5-1 summarizes the peak containment pressure following the initial blowdown peak, the peak suppression pool temperature, and a summary of the equipment capability assumed in the analysis.

Case B

This case assumes that one RHR loop is operating at design heat removal capacity (one RHR heat exchanger, one RHR pump, and design value of cooling water flow to one RHR loop operating in the suppression pool cooling mode). As in the previous case, there is no containment spray operation and the suppression pool cooling mode is assumed to be activated at 600 sec after the accident. The containment pressure response to this set of conditions is shown as curve "b" on Figure 14.5-5. The corresponding drywell and suppression pool temperature responses are shown as curves "b" on Figures 14.5-6 and 14.5-7. A summary of this case is shown on Table 14.5-1, including a summary of the equipment capability assumed in the analysis.

Case C

This original case assumes the same equipment operability as Case B except that the entire discharge from the RHR heat exchanger is routed to the containment spray headers in the drywell and wetwell. It assumed that the containment spray is established at 600 sec after the accident.

The containment response to this set of conditions is shown as curve "c" on Figure 14.5-5. The corresponding drywell and suppression pool temperatures are shown as curves "c" on Figures 14.5-6 and 14.5-7. A summary of this case is shown on Table 14.5-1, including a summary of the equipment capability assumed in the analysis.

Comparing the "containment spray" Case C with the "no spray" Case B, it is seen that the suppression pool temperature response is the same because the same amount of energy is removed from the pool via the RHR heat exchanger. The total flow rate through the RHR heat exchanger is the same for Case B & C. However, the post blowdown containment pressure is higher for the "no spray" case, as shown by Figure 14.5-5. This, however, is of no consequence since the pressure is still much less than the containment design pressure of 56 psig. Figure 14.5-8 illustrates the slight effect on calculated containment leakage rate, due to the higher pressure.

The containment spray flowrate used in the original FSAR containment analysis (based on a 65°F SSW inlet temperature) was substantially reduced from its design value of 5,000 gpm down to 1,100 gpm by capping the majority of the drywell spray nozzles. The Case C, containment pressure and temperature response curves shown on Figure 14.5-5 and Figure 14.5-6 were not recalculated using the current containment spray flow rate of approximately 1,100 gpm. The spray reduction will increase the drywell temperature and pressure between 600 seconds when spray is initiated and 1×10^6 seconds, the time at which the analysis is terminated. Although, drywell temperature and pressure increase, that increase is bounded by the results for Case B which is based on one loop of RHR in the suppression pool cooling mode and no spray.

To assure the suppression pool temperature response is the same as that shown on Figure 14.5-7, the RHR heat exchanger flowrate must be maintained consistent with values in Table 14.5-1. Operating procedures require that a portion of the discharge from the RHR heat exchanger be routed to the containment spray headers and the remaining portion return to the suppression pool via the suppression pool bypass line.

75°F SSW Case

For the 75°F SSW Case, the calculated pressure and temperature responses of the containment are shown on Figures 14.5-16, and 14.5-17. The short-term response of the drywell, wetwell, and suppression pool is the same as for the Case B from the 65°F SSW Case. The containment response prior to 600 seconds is unaffected by containment cooling and remains the same for both cases. The 65°F SSW Case provides an additional description of the short-term response.

Prior to the activation of containment cooling, the LPCI and core spray pumps have been adding liquid to the reactor vessel. After the vessel is flooded to the height of the jet pump nozzles, the excess flow discharges through the break into the drywell. This flow cools the fuel and flushes sensible heat from the reactor vessel into the drywell. The flow of sub cooled liquid into the drywell causes depressurization of the containment as the steam in the drywell is condensed.

For the 75°F SSW Case, the long-term analysis assumes one RHR loop is available for containment cooling. At 600 seconds, the necessary valves are opened admitting cooling water flow to the RHR heat exchanger. The RHR heat exchanger bypass valve is assumed to remain in its full open normal position and the RHR system is assumed to remain in the LPCI mode with containment cooling by heat rejection through the RHR heat exchanger. No disruption of LPCI flow is required to enter this mode of cooling. This configuration will provide maximum core cooling, but does not provide rated heat removal because more than half of the two pump LPCI flow rate goes through the heat exchanger bypass line and not the heat exchanger.

At two hours after the start of the accident, a transition is made from the two pump LPCI with Heat Rejection mode to the one pump LPCI with Heat Rejection mode to maximize the heat removal function of the RHR System. Rated heat removal from the containment is obtained using the LPCI with Heat Rejection mode by removal of one RHR pump from LPCI service and closure of the RHR heat exchanger bypass valve while maintaining maximum LPCI injection flow from the single RHR pump. One pump LPCI with Heat Rejection mode is assumed to run continuously throughout the remainder of the accident response.

For the design basis LOCA analysis, it is assumed that there is only one loop of containment heat removal (RHR, RBCCW, and SSW) operable. The containment heat removal assumed in the design basis LOCA analysis is that which can be obtained with one loop of the RHR, RBCCW, and SSW Systems operating at the limiting conditions for pump and heat exchanger performance. Suppression pool temperature will continue increasing from the transfer of sensible and decay heat from the reactor core to the suppression pool until reaching the peak approximately 5 to 6 hours after the accident. The design peak suppression pool temperature for the DBA-LOCA is 185°F, which is well below the primary containment design temperature of 281°F. Subsequently, the decreasing decay heat results in a steady cooldown and depressurization of primary containment. The RHR heat transfer parameters at the peak suppression pool temperature are given in Table 14.5-6 and the resulting containment and suppression pool temperature profiles are given in Figure 14.5-17.

14.5.3.1.3 Core Standby Cooling System Pump Net Positive Suction Head

To assure proper operation of the CSCS pumps following a design basis LOCA, the Primary Containment and CSCS system design is such that Net Positive Suction Head (NPSH) margin is available to the pumps at all times.

The NPSH available (NPSHA) at the suction to the CSCS pumps is equal to the total absolute pressure minus the vapor pressure of water at the suppression pool temperature. The NPSH required at the pump suction (NPSHR) is the minimum pressure over and above the vapor pressure that must be present in order to prevent pump cavitation.

NPSH design margin is based on calculations that include the effect from the increase in wetwell vapor pressure and air/nitrogen partial pressure in equilibrium with increasing suppression pool temperature with an accounting for containment initial conditions and leakage.

The design margin for NPSH available to the RHR and core spray pumps is determined using the following assumptions:

1. The primary containment is assumed to contain the minimum credible mass of noncondensible gas (air/nitrogen) prior to the design basis LOCA. The drywell initial condition is 150°F, 80% RH, 1.3 psig, and the wetwell is 85°F, 100% RH, 0 psig.

2. The water vapor pressure in containment increases to be in equilibrium with the suppression pool temperature.

3. The partial pressure of the containment air/nitrogen increases with the pool temperature per the ideal gas laws after the initial mixing of the drywell and wetwell air has occurred.

4. Where stated on the figures, containment leakage has been calculated based on a leak rate of 1% per day for design basis conditions and 5% per day to demonstrate conservative design margin with impaired containment integrity.
   The leakage values represent percent mass per day at a reference pressure of 45 psig using the mass leakage formulation described in Appendix R.5.4.2 "Long Term Containment Response."

5. The suppression pool temperature profile is based on minimum primary containment system cooling, i.e., one RHR loop in containment cooling is assumed, with an initial suppression pool temperature of 85°F and a salt service water heat sink temperature of 75°F.

6. Minimum initial water volume in the suppression pool is assumed (84,000 ft^3).

7. Drywell free volume temperature is equal to wetwell temperature following the accident. This is based on the redistribution of noncondensible gases between the drywell and wetwell via the vacuum-breaker system following the vessel depressurization phase.

8. Maximum flow rates are used for the CSCS pumps to maximize the suction line losses and NPSH required by the pumps. The NPSHR is 27 ft at 5670 gpm for the RHR pumps and 29 ft at 4950 gpm for the core spray pumps.

Based on the above conservative assumptions, the margin for NPSH available was evaluated for the limiting accident event which is the design basis LOCA. The NPSH available and NPSH margin for the RHR and core spray pumps were evaluated for both a 75°F SSW injection temperature and 65°F SSW injection temperature. In the following discussion the analysis that used a 75°F SSW injection temperature is referred to as the 75°F SSW Case, likewise the analysis based on a 65°F SSW injection temperature is referred to as the 65°F SSW Case.

The 65°F SSW Case is based on the suppression pool temperature profile in Figure 14.5-7 from the original design basis LOCA analysis as described earlier in this section. The assumed flow rates, head losses, initial containment mass of nitrogen, and NPSH required for the RHR and core spray pumps have been revised from the original 65°F NPSH analysis so that the same values are used for both the original 65°F and updated 75°F NPSH analyses presented in this Section. The 75°F SSW Case uses the suppression pool temperature profile from Figure 14.5-17 that is from the updated design basis LOCA analysis described earlier.

Figure 14.5-9 shows the NPSH available as a function of pool temperature with zero containment leakage which makes this curve independent of time. Since no leakage effect is included, Figure 14.5-9 represents the highest NPSH margin that can be obtained using the above assumptions and as can be seen, a large margin exists for all pool temperatures. NPSH margin for the 65°F SSW Case, with leakage effects included, is present in a different format on Figure 14.5-10 and Figure 14.5-13. Here, the suppression pool temperature and containment pressure are shown as a function of time. Also shown is the primary containment pressure required to provide the required NPSH to the RHR and core spray pumps at their maximum required flow rates. As can be seen, substantial margin exists throughout the duration of the event. Therefore, it can be concluded that adequate NPSH will be available at all times following a design basis LOCA for the 65°F SSW Case.

NPSH margin for the 75°F SSW Case, with leakage effects included, is presented on Figure 14.5-18 and Figure 14.5-19. Here, the suppression pool temperature and containment pressure are shown as a function of time. Also shown is the primary containment pressure required to provide the required NPSH to the RHR and core spray pumps at their maximum required flow rates. As can be seen, substantial margin exists throughout the duration of the event. Therefore, it can be concluded that adequate NPSH will be available at all times following a design basis LOCA for the 75°F SSW Case.

The RHR and Core Spray System design analysis shows that substantial NPSH margin is available at all times following the bounding design basis LOCA. The design margin for NPSH available is that which exists between the minimum containment pressure that provides the required NPSH and the containment pressure that exists due to equilibrium conditions for the gas/vapor mixture with an accounting for containment initial conditions and leakage. This method of analysis for determining NPSH margin is in accordance with the original design basis for Pilgrim and other similar BWRs. The NRC has chosen to impose limits on the amount of containment pressure that may be included in the NPSH margin for CSCS pump suction strainer evaluations. These time-dependent containment pressure limits were selected based on NRC review of plant-specific accident analysis and are considerably less than the calculated equilibrium pressure.

In accordance with the NRC Safety Evaluation Report for License Amendment 185, the amount of containment positive pressure that may be included in a CSCS pump NPSH analysis has been limited to the following:

Time After Accident (sec)	(hour)	Containment Pressure (psig)	(psia)
0 to 1,200	0.00 to 0.33	0.0	14.7
1,200 to 1,800	0.33 to 0.50	1.9	16.6
1,800 to 3,600	0.50 to 1.0	3.0	17.7
3,600 to 57,600	1.0 to 16.0	5.0	19.7
57,600 to 108,000	16.0 to 30.0	2.5	17.2
108,000 to 172,800	30.0 to 48.0	1.0	15.7
172,800 to 864,000	48.0 to 240.0	0.0	14.7

These limits on containment pressure are included in the evaluation of LOCA debris head losses for the RHR and core spray pumps and the resulting NPSH available for long-term containment heat removal. There remains sufficient NPSH margin within these containment pressure limits to accommodate the postulated LOCA debris without affecting pump performance. The limits listed above are included in Figure 14.5-18 along with the calculated amount of containment pressure available. Figure 14.5-18 also includes a curve showing the amount of containment pressure required to provide adequate NPSH to the most limiting core spray pump operating with the maximum suction strainer head loss from the bounding analysis for strainer debris described in Section 6.4.3.

Evaluations of NPSH for reactor isolation events are bounded by the design basis LOCA. Analysis for isolation scenarios such as a fire event, where the high pressure makeup systems are assumed unavailable, are included in the updated containment analysis with a 75°F SSW heat sink. It is assumed that reactor depressurization occurs at 1450 seconds (24 minutes) due to low reactor water level and there is no suppression pool cooling for two hours. The peak pool temperature is less than 185°F while the equilibrium mechanism for containment pressure and NPSH available are the same as for the LOCA. The resulting NPSH available exceeds that for the design basis LOCA due to the lower pool temperature.

During Reactor Core Isolation Cooling System (RCIC) operation, the drywell free air volume cooler will normally remain operational. Due to the reduced heat load on the air coolers caused by the shutdown of the two reactor coolant recirculation system pumps, the drywell temperature could actually be less than the normal operating value in spite of the fact that some of the air cooler capacity may also be shut down. The lower drywell temperature would tend to reduce the primary containment pressure which would reduce the NPSH available. In order to arrive at a conservative lower bound on the total NPSH available, the following model was assumed:

1. No leakage from the primary containment (even at 5 percent free volume per day, leakage would be negligible during the short time period being considered).

2. Drywell and wetwell pressure equal (maintained equal by the vacuum breakers between the wetwell and drywell).

3. Torus air temperature equal to pool water temperature.

4. Drywell temperature during reactor core isolation cooling equal to 110°F, 20 percent rh (very conservative estimates).

5. Initial drywell conditions: 150°F, 0 psig, 100 percent rh

Actually, Assumption 4 and 5 are contradictory. If the heat load during normal operation is large enough to cause a drywell temperature of 150°F and a relative humidity of 100 percent, the air coolers would not be capable of reducing the drywell temperature to 110°F during RCIC operation. Such a heat load implies a small steam leak from the primary system.

This RCIC NPSH evaluation is based on very conservative assumptions for drywell and wetwell conditions during RCIC operation. The drywell atmosphere is assumed to be cooled and dehumidified down to 110°F at 20% rh by operation of the drywell coolers. The wetwell is in thermal equilibrium with the suppression pool, but the drywell and wetwell pressures are equalized due to the drywell vacuum breakers. The reactor is assumed to be scrammed at a suppression pool temperature of 110°F and depressurized when the pool reaches 120°F per the Technical Specifications. Due to the fixed drywell temperature at low humidity and the pressure equalization between the drywell and wetwell, the resulting containment pressure is minimized to a level only slightly above atmospheric pressure.

Figure 14.5-11 plots the NPSH available versus suppression pool temperature and show that there is NPSH margin at pool temperatures up to at least 175°F. The RCIC System is specified for continuous operation up to a suppression pool temperature of 140°F; however, short term operation at up to 170°F is also considered for system design since this represents the temperature at the end of a reactor depressurization that begins at 120°F. Figure 14.5-12 plots the containment pressure and a suppression pool temperature profile for a postulated controlled cooldown and depressurization of the reactor. The peak suppression pool temperature at the end of the reactor depressurization is 163°F, which is well within the range for which sufficient NPSH is available.

Assumptions regarding initial pool temperature, heat sink temperature, and decay heat have a minor effect on this peak pool temperature. The predominant effect on pool temperature is the fixed assumptions of reactor shutdown at 110°F and depressurization at 120°F which ensure the suppression pool temperature will not exceed 170°F during the depressurization.

The conservative assumption that pump NPSH required is 28 feet makes this RCIC analysis bounding for both the RCIC and HPCI pumps for operation through vessel depressurization to less than 200 psig.

Vessel depressurization to 200 psig allows the LPCI and/or Core Spray System to maintain core cooling. The NPSH evaluations for RCIC operation is inherently more limiting than for HPCI since during RCIC operation, there is no assumption of a steam leak to heatup and pressurize the drywell. As can be seen in Figure 14.5-11, there is significantly more NPSH available than required for suppression pool temperatures up to 170°F. Therefore, it can be concluded that adequate NPSH will be available during RCIC and HPCI operation.

14.5.3.1.4 Metal Water Reaction Effects on the Primary Containment

If Zircaloy in the reactor core is heated to temperatures above 2,000°F in the presence of steam, a chemical reaction occurs in which zirconium oxide and hydrogen are formed. This is accompanied by an energy release of about 2,800 Btu/lb of zirconium reacted. The energy produced is accommodated in the suppression chamber pool. The hydrogen formed, however, will result in an increased drywell pressure due simply to the added volume of gas in the fixed containment volume. Although very small quantities of hydrogen are produced during the accident, the containment has the inherent ability to accommodate a much larger amount as discussed.

The basic approach to evaluating the capability of a Containment System with a given Containment System spray design is to assume that the energy and gas are liberated from the reactor vessel over some time period. The rate of energy release over the entire duration of the release is arbitrarily taken as uniform, since the capability curve serves as a capability index only, and is not based on any given set of accident conditions as an accident performance evaluation might be.

It is conservatively assumed that the suppression pool is the only body in the system which is capable of storing energy. The considerable amount of energy storage which would take place in the various structures of the containment is neglected. Hence, as energy is released from the core region, it is absorbed by the suppression pool. Energy is removed from the pool by heat exchangers which reject heat to the station cooling systems. Because the energy release is taken as uniform and the service water temperature and system flow rates are constant, the temperature responses of the pool can be determined. It is assumed that the suppression chamber gases are at the suppression chamber water temperature.

The metal water reaction during core heatup is calculated by the core heatup model described in Appendix R.5. The extent of the metal water reaction thus calculated is less than 0.1 percent of all the zirconium in the core. As an index of the containment's ability to tolerate postulated metal water reactions, the concept of "Containment Capability" is used. Since this capability depends on the time domain, the duration over which the metal water reaction is postulated to occur is one of the parameters used.

Containment capability is defined as the maximum percent of fuel channels and fuel cladding material which can enter into a metal water reaction during a specified duration without exceeding the maximum allowable pressure of the containment. To evaluate the containment capability, various percentages of metal-water reaction are assumed to take place over certain time periods. This analysis presents a method of measuring system capability without requiring prediction of the detailed events in a particular accident condition.

Since the percent metal water reaction capability varies with the duration of the uniform energy and gas release, the percent metal water reaction capability is plotted against the duration of release. This constitutes the containment capability curves as shown on Figure 14.5-14. All points below the curves represent a given metal water reaction and a given duration which will result in a containment peak pressure which is below the maximum allowable pressure. The calculations are made at the end of the energy release duration because the number of moles of gases in the system is then at a maximum, and the suppression pool temperature is higher at this time than at any other time during the energy release.

It should be noted that the curves are actually derived from separate calculations of two conditions: the steaming and the non-steaming situations. The minimum amount of metal water reaction which the containment can tolerate for a given duration is given to the condition where all of the noncondensible gases are stored in the suppression chamber. This condition assumes that steaming from the drywell to the suppression chamber results in washing all of the noncondensible gases into the suppression chamber. This is shown as the flat portion of the containment capability characteristic curve. Activation of containment sprays condense the drywell steam so that no steaming occurs, thus allowing noncondensibles to also be stored in the drywell. This is denoted by the rising spray curve. The intersection between the no spray curve and the spray curve represents the duration and metal water reaction energy release which just raises all the spray water to the saturation temperature at the maximum allowable containment pressures.

For durations to the left of the intersection some steam is generated and all the gases are stored in the suppression chamber. For durations to the right of the intersection, the spray flow is subcooled as it exits from the drywell by increasing amounts as the duration is increased.

The energy release rate to the containment is calculated as follows:

$$qin = \frac{Qo + Qmw + Qs}{T_D}$$

Where:

qin = Arbitrary energy release rate to the containment, Btu/sec

Qo = Integral of decay power over selected duration of energy gas release, Btu

Qmw = Total chemical energy released exothermically from selected metal-water reaction, Btu

Qs = Initial internal sensible energy of core fuel and cladding, Btu

T_D = Selected duration of energy and gas release, sec

The total chemical energy released from the metal water reaction is proportional to the percent metal water reaction. The initial internal sensible energy of the core is taken as the difference between the energy in the core after the blowdown and the energy in the core at a datum temperature of 250°F.

The temperature of the drywell gas is found by considering an energy balance on the spray flows through the drywell as described in Appendix R.5.

Based upon the drywell gas temperature, suppression chamber gas temperature, and the total number of moles in the system, as calculated above, the containment pressure is determined. The containment capability curves on Figure 14.5-14 present the results of the parametric investigation.

14.5.3.2 Radiological Consequences

14.5.3.2.1 Loss of Coolant Accident Assumption

1. The reactor has operated for an extended period at 1,998 MWt. To account for power measurement uncertainty, a 2% allowance was added.

2. One hundred percent of the noble gases and 25 percent of the iodine in the core instantaneously become available for leakage from the primary containment.

3. The primary containment leak rate is a constant 1.25 percent/day for 30 days.

4. For radiological dose considerations, release to the atmosphere was assumed to occur via drywell leakage, main steam isolation valve (MSIV) leakage, and emergency core cooling system (ECCS) leakage. Drywell and ECCS Leakage, flows through the Standby Gas Treatment System without the inherent benefit of mixing in the Secondary Containment Building, and is released to the environment via the Main Stack. Main steam isolation valve leakage is a ground level, unfiltered release through the condenser and high pressure turbine.

5. Ninety-nine percent of the iodine entering the Standby Gas Treatment System is retained by the charcoal filters.

6. Atmosphere dispersion factors were based on Regulatory Guide 1.145 models.

7. The breathing rate is 347 cm^3/sec for the first 8 hr, 175 cm^3/sec for the next 16 hr, and 232 cm^3/sec thereafter.

14.5.3.2.2 Analytical Results

Radiological consequences for the loss of coolant accident based on the above assumptions are given in Table 14.5-2.

14.5.4 Main Steam Line Break Accident

The analysis of the main steam line break accident depends on the operating thermal-hydraulic parameters of the overall reactor (such as pressure) and overall factors affecting the consequences (such as primary coolant activity). Insertion of reload fuel does not change any of these parameters. Therefore the analyses presented for the initial core remains applicable. The results of this analysis based on the initial core thermal-hydraulic basis are given in Appendix R.3.5. The results of the coolant loss and radiological consequences are given below. The loss of coolant due to a main steam line break accident evaluation was not performed for the Pilgrim 10 CFR 50 Appendix K power uprate (1.5% of 1,998 MWt at constant pressure). All safety and operational aspects of MSIV and steam flow restrictors performance are within pre-power uprate evaluations.

14.5.4.1 Coolant Loss Analysis

The steam flow rate through the upstream side of the break increases from the initial value of 550 lb/sec in the line to 1,100 lb/sec (about 200 percent of rated flow for one steam line) with critical flow initially occurring at the flow restrictor. The steam flow rate was calculated using an ideal nozzle model. That the flow model predicts the behavior of the flow limiter has been substantiated by tests conducted on a scale model over a variety of pressure, temperature, and moisture conditions.

The steam flow rate through the downstream side of the break consists of equal flow components from each of the unbroken lines. In each of the unbroken lines, the steam flow rate increases from an initial value of 550 lb/sec to 1,100 lb/sec. Critical flow would be occurring at the flow limiters in these lines.

The total steam flow rate leaving the vessel is thus approximately 4,500 lb/sec, which is in excess of the steam generation rate of 2,200 lb/sec. The steam flow steam generation mismatch causes an initial depressurization of the reactor vessel at a rate of 45 psi/sec. The formation of bubbles in the reactor vessel water causes a rapid rise in the water level. The analytical model used to calculate level rise predicts a rate of rise of about 6 ft/sec. Thus, the water level reaches the vessel steam nozzles at 2 to 3 sec after the break. From that time on a two phase mixture is discharged from the break as shown on Figure 14.5-15. The two phase flow rates are determined by vessel pressure and mixture enthalpy.[(4)] The vessel depressurization is calculated using a digital computer model in which the reactor vessel is divided into nine major nodes. The model includes the flow resistance between nodes, as well as heat addition from the core.

As shown on Figure 14.5-15, two phase flow is discharged through the break at an almost constant rate until late in the transient. This is the result of not taking credit for the effect of valve closure on flow rate until isolation valves are far enough closed to establish critical flow at the valve locations. The linear decrease in discharge flow rate at the end of the transient is the result of the assumption regarding the effect of valve closure on flow rate after critical flow is established at the valve location.

The following total masses of steam and liquid are discharged through the break prior to isolation valve closure:

Steam	25,000 lb
Liquid	60,000 lb

14.5.4.2 Radiological Consequences

14.5.4.2.1 Steam Line Break Accident Assumptions (Ground Level Release)

1. The reactor has operated for an extended period at 1,998 MWt. To account for power measurement uncertainty, a 2% allowance was added.

2. The concentrations of radionuclides in the reactor water are those corresponding to the maximum reactor coolant iodine concentration permitted by plant Technical Specifications.

3. The total mass of steam and water released from the steam line contains concentrations of radionuclides identical with those in the reactor water.

4. All of the radionuclides contained in the steam and water mass released from the steam line are released to the atmosphere from the top of the Turbine Building. All the radioactivity was assumed to be released to the environment as a puff release.

5. It is assumed that there is no thermal rise of the steam cloud.

6. Atmospheric relative concentration values were based on Regulatory Guide 1.145 models.

7. The breathing rate is 347 cm^3/sec.

14.5.4.2.2 Analytical Results

Radiological consequences for the steam line break accident based on the above assumptions are given in Table 14.5-2.

14.5.5 Fuel Handling Accident

Accidents that result in the release of radioactive materials directly to the containment can occur when the drywell is open. A survey of the various conditions that could exist when the drywell is open reveals that the greatest potential for the release of radioactive material occurs when the drywell head and reactor vessel head have been removed. In this case, radioactive material released as a result of fuel failure is available for transport directly to the containment.

Various mechanisms for fuel failure under this condition have been investigated. With the current fuel design the refueling interlocks, which impose restrictions on the movement of refueling equipment and control rods, prevent an inadvertent criticality during refueling operations. In addition, the reactor protection system can initiate a reactor scram in time to prevent fuel damage for errors or malfunctions occurring during planned criticality tests with the reactor vessel head off. It is concluded that the only accident that could result in the release of significant quantities of fission products to the containment during this mode of operation is one resulting from the accidental dropping of a fuel bundle onto the top of the core.

This event occurs under non-operating conditions for the fuel. The key assumption of this postulated occurrence is the inadvertent mechanical damage to the fuel rod cladding as a consequence of the fuel bundle being dropped on the core in the cold condition. Therefore, fuel densification considerations do not enter into or affect the accident results.

14.5.5.1 Sequence of Events

The assumptions and analyses applicable to this type of fuel handling accident are described below.

(1) The fuel assembly is dropped from 32.95 feet (the maximum height allowed by the fuel handling equipment).

(2) The entire amount of potential energy, including the energy of the entire assemblage falling to its side from a vertical position (referenced to the top of the reactor core), is available for application to the fuel assemblies involved in the accident. This assumption neglects the dissipation of some of the mechanical energy of the falling fuel assembly in the water above the core and requires that the grapple cable break, allowing the grapple head and three sections of the telescoping mast to remain attached to the falling assembly.

(3) None of the energy associated with the dropped fuel assembly is absorbed by the fuel material (uranium dioxide).

(4) All fuel rods, including tie rods, were assumed to fail by 1% strain in compression, the same mode as ordinary fuel rods. For the fuel designs considered here, there is no propensity for preferential failure of tie rods.

14.5.5.2 Fuel Damage

Because of the complex nature of the impact and the resulting damage to fuel assembly components, a rigorous prediction of the number of failed rods is not possible. For this reason, a simplified energy approach was taken and numerous conservative assumptions were made to assure a conservative estimate of the number of failed rods.

The number of failed fuel rods was determined by balancing the energy of the dropped assemblage against the energy required to fail a rod. The wet weight of the dropped bundle is 617 pounds and the wet weight of the grapple component is 350 pounds. The drop distance is 32.95 feet. The total energy to be dissipated by the first impact is

$$E = (617 + 350)\ (32.95) = 31{,}870 \text{ ft-lb}$$

One half of the energy was considered to be absorbed by the falling assembly and one half by the four impacted assemblies.

No energy was considered to be absorbed by the fuel pellets (i.e., the energy was absorbed entirely by the non-fuel components of the assemblies). The energy available for clad deformation was considered to be proportional to the mass ratio:

$$\frac{\text{mass of cladding}}{(\text{mass of assembly - mass of fuel pellets})}$$

and is equal to a maximum of 0.519 for the fuel designs considered here.

The energy absorbed by the cladding of the four impacted assemblies is

$$(15{,}935 \text{ ft-lb})\ (0.519) = 8270 \text{ ft-lb}$$

Each 7x7 or 8x8 assembly rod that fails is expected to absorb approximately 250 ft-lb before cladding failure, based on uniform 1% plastic deformation of the cladding. The energy required to fail a fuel rod depends on the cladding thickness and is smaller for 9x9 and 10x10 designs.

The number of rods failed in the impacted assemblies is

$$N_F = \frac{(8270 \text{ ft-lb})}{(250 \text{ ft-lb})} = 33 \text{ rods}$$

The dropped assembly was considered to impact at a small angle, subjecting all the fuel rods in the dropped assembly to bending moments. The fuel rods are expected to absorb little energy prior to failure as a result of bending. For this reason, it was assumed that all the rods in the dropped assembly fail. The total number of failed rods on initial impact was 62 + 33 = 95.

The assembly was assumed to tip over and impact horizontally on the top of the core. The remaining available energy was used to predict the number of additional rod failures. The available energy was calculated by assuming a linear weight distribution in the assembly with a point load at the top of the assembly to represent the fuel grapple weight.

$$E = W_G H_G + \int_0^{H_B} \frac{W_B}{H_B} y \, dy = W_G H_G + \frac{1}{2} W_B H_B$$

$$= (350 \text{ lb}) \frac{160}{12} + \frac{1}{2} (617) \frac{160}{12} = 8780 \text{ ft-lb}$$

As before, the energy was considered to be absorbed equally by the falling assembly and the impacted assemblies. The fraction available for clad deformation was 0.519. The energy available to deform clad in the impacted assemblies was

$$E_c = (0.50)\ (8780 \text{ ft-lb})\ (0.519) = 2278 \text{ ft-lb}$$

and the number of failures in the impacted assemblies was

$$N_F = \frac{(2278 \text{ ft-lb})}{(250 \text{ ft-lb})} = 9 \text{ rods}$$

Since the rods in the dropped assembly were considered to have failed in the initial impact, the total failed rods in both impacts are 95 + 9 = 104.

Both the GE8x8EB and the GE8x8NB fuel designs contain 2 fewer fuel rods than the 62 fuel rods assumed in the preceding analysis. Hence, this analysis is conservatively bounding for these fuel types.

The number of rods assumed failed in a Fuel Handling Accident for GE11, GE14, and GNF2 are obtained from the GESTAR Amendment 22 Reports. For GE11, 123 rods are calculated to fail per Ref. 25. For GE14, 151 rods are calculated to fail per Ref. 26. For GNF2 fuel, 150 rods are calculated to fail per Ref. 27.

14.5.5.3 Radiological Consequences

The Fuel Handling Accident (FHA) could occur inside the open reactor vessel or, inside the spent fuel pool, both of which are located inside the reactor building, during shutdown refueling operations.

Of the two possible FHA's, it is the FHA occurring inside the open reactor vessel that would be expected to release more radioactive gaseous material from the gap spaces of fuel bundles containing damaged cladding of spent fuel rods.

14.5.5.3.1 Method and Assumptions

The DBA FHA analysis uses the AST guidelines outlined in NUREG-1465 (Reference 21), Regulatory Guide 1.183 (Reference 22), and Regulatory Guide 1.194 (Reference 23).

The following assumptions and initial conditions are used in calculating the fission product release to the environment:

(a) The accident is assumed to occur 24 hours after shutdown.

(b) The FHA results in 151 rods failing, and the release to the environment from the refueling floor occurs within 2 hours.

(c) A decontamination factor (DF) of 200 was assumed for the scrubbing effects of water on halogen activity release. The DF was based on a minimum of 23 feet of water over the dropped assembly. No DF was applied to noble gases and the DF for other radionuclides were assumed to be infinite.

(d) The core inventory was based on a thermal power level of 2028 MW_t, plus a measurement uncertainty of 0.5% (2038 MW_t). A radial peaking factor of 2.1 was used. The bounding core and FHA inventories are given in Table 14.5-4.

(e) All activity within the gaps of the failed fuel rods is released to the refueling cavity water. The released activity corresponds to 8% of the entire inventory of I-131 in the rods, 10% of the Kr-85, 5% of the remaining halogens and noble gases, and 12% of the alkalis (Cs and Rb).

(f) The reactor building is assumed to be open during the refueling operations, with the normal ventilation on, such that all releases to the environment would be via the reactor building vent.

(g) 5 years of hourly meteorological data was used for atmospheric dispersion factors as shown in Table 14.5-4A.

(h) The control room ventilation system was assumed to remain in the normal operating mode during the entire exposure interval (30 days).

(i) Breathing rates, and control room occupancy factors are as given in Reg. Guide 1.183.

(j) The control room air intake rate was assumed to be 1000 cfm (a low value), and 9000 cfm (a high value).

14.5.5.3.2 Results

The dose evaluations of the postulated fuel handling accident are summarized in Table 14.5-5 and demonstrate that the calculated TEDE values to the control room, EAB, and LPZ are less than the limits set forth in 10CFR50.67 and Reg. Guide 1.183. Reference 24 contains the dose consequences for the Fuel Handling Accident.

14.5.6 Radwaste System Accidents

The reactor building, the radwaste building, and the turbine building contain systems which have significant amounts of radioactive materials. The reactor building, the radwaste building, and the turbine building, where they house or support Class I equipment, are Class I. The condenser hotwell, the offgas system piping, the monitor tanks, the treated water holdup tanks, and the condensate storage tanks contain significant amounts of liquid or gaseous radioactivity not enclosed in a Class I structure. This response analyzes the effects, which would result from the failure of the condenser hotwell, of the offgas system piping (rupture disk failure), or of any radwaste system tank.

14.5.6.1 Assumptions

The following assumptions are made in evaluating the potential effects resulting from condenser hotwell or Radwaste System tank failures inside the Radwaste and Turbine building. This analysis is not based on TID-14844 source terms.

1. The maximum activity concentrations in the reactor water and in the radwaste tanks are those expected, assuming offgas stack release rate of 100,000 microcuries/sec after 30 min holdup and, for the Radwaste System, normal daily liquid volume.

2. The activity concentrations in the condenser hotwell are based upon a reactor water steam separation factor of 10^{+4}.

3. The iodine activity concentration in a Radwaste System tank is equal to the ratio of the iodine activity concentration to total reactor water activity concentration after 8 hr decay, multiplied by the maximum activity concentration in the inlet stream to the particular tank.

4. The partition coefficient, the ratio of the iodine concentration in the liquid phase to the iodine concentration in the gas phase at equilibrium, equals $4.43 \times 10^{+5}$ based upon a pH of 7.0, a temperature of 25°C and a total iodine concentration of less than 1×10^{-9} moles /1.[5] Expected iodine concentrations are at least two orders of magnitude less.

5. The partition coefficient is constant and does not increase with decreasing concentration of iodine in the liquid phase.

6. Instantaneous dynamic equilibrium is maintained between the gaseous and liquid phases. The equilibrium exists between the release liquid and the net free volume of the Radwaste Turbine Building basement area; or between the condenser hotwell condensate and condenser compartment net free volume.

7. The area ventilation fans continue exhausting through the Reactor Building vent at full capacity, resulting in one air change every 6.5 min from the Radwaste Turbine Building basement area and one air change every 10 min from the condenser compartment.

8. An airborne iodine reduction factor of 2 results due to plateout of iodine in the gaseous phase.

9. All releases except from offgas piping failure occur during the following meteorological conditions:

    a. For the first 8 hr, Pasquill Type F, 1 m/s, nonvarying wind direction and a volumetric building wake correction factor of c = 1/2 used with the cross sectional area of the structure with a maximum building wake reduction factor of 1/3

    b. From 8 to 24 hr, Pasquill Type F, 1 m/s with plume meander in a 22 1/2 degree sector

    c. From 1 to 4 days, Pasquill Type F and 2 m/s with a frequency of 60 percent, Pasquill Type D 3 m/s with a frequency of 40 percent with a meander in the same 22 1/2 degree sector

    d. From 4 to 30 days, Pasquill Types C, D, and F each occurring 33 1/3 percent of the time with wind speeds of 3 m/s, 3 m/s, and 2 m/s, respectively, with meander in the same 22 1/2 degree sector 33 1/3 percent of the time

10. A breathing rate of 3.47×10^{-4} m3/s for the first 8 hr, $1.75 \times 10^{-4} m^3/s$ from 8 to 24 hr and $2.32 \times 10^{-4} m^3/s$ thereafter is assumed.

11. The effective release height is 30 m with downwash occurring.

12. No credit is taken for radioactive decay in the environment.

14.5.6.2 Radiological Effects

The above assumptions have been chosen to maximize the initial activities, and overestimate the release rates, total releases, and total doses.

Consider first the failure of the condenser hotwell, or a Radwaste System tank. Each set of tanks is surrounded by waterproof shield walls designed to contain the spillage from the failure of one tank in the immediate vicinity until the liquid can be pumped into another tank. The floors in the Radwaste Turbine Building basement area are sloped such that gross tankage failure in that area, if assumed, would result in fluid flow in direction of the Radwaste Building and the Class I portions of the Turbine Building. Thus, all released liquid from tank failure in that area could be assumed to be contained and enclosed by a Class I structure. Also, the release of radioactivity from the condenser hotwell or a tank failure directly into the environment through the ground water is prevented or minimized by the PVC, waterproof membrane which encloses and forms a continuous seal around the Turbine Building, Reactor Building and Radwaste Building footings below grade and by the hydrostatic head exerted on the foundations by the ground water. The primary release of radionuclides to the environment would be due to the release of gaseous iodine in chemical equilibrium with iodine in the liquid phase of the spillage.

During normal operation, Radwaste System tanks containing high activity liquids are vented directly into the Radwaste Exhaust system and tanks containing low activity liquids, such as the monitor tanks and the treated water holdup tanks are vented directly into their immediate areas which are vented into the Radwaste Ventilation Exhaust System. All Radwaste Building Ventilation System air passes through one of two parallel sets of high efficiency particulate air (HEPA) filter trains and is released through the Reactor Building exhaust vent stack.

The iodine activity released to the Ventilation Exhaust System during normal station operation or from spillage following an assumed tank or condenser hotwell failure is a function of the partition coefficient which is the ratio of the iodine concentration in the liquid phase to the iodine concentration in the gas phase at equilibrium. Because iodine undergoes a series of hydrolysis reactions, the partition coefficient is a function of solution temperature, pH, and concentration. The partition coefficient is not a function of the container, thus the coefficient existing prior to the hypothetical tank failure would continue to be appropriate after the failure.

The calculation model developed to determine the release of iodine from spillage to the air above after postulated tank failure assumes that an instantaneous equilibrium exists between the iodine in the liquid and gaseous phases. Solution of the simultaneous coupled differential equations resulting from the differential activity balances across the liquid gas interface, and between the Radwaste Turbine Building basement air and the environment, yields the total iodine activity released to the environment as a function of time.

Solutions to the differential equations show that the release rate to the environment increases with increasing activity and/or volume of the gaseous phase, and decreases with increasing liquid volume, and with time. Thus the maximum release rate results from the failure of one tank containing a maximum activity in a minimum liquid volume. Application of the assumption that an iodine equilibrium condition exists between the released liquid and the total free volume of the Radwaste Turbine Building basement area yields extremely conservative releases, since the released liquid would normally be contained by the waterproof shield walls, and thus would not have the large surface area with which to communicate and establish equilibrium with the total net free volume of the area.

The resulting maximum iodine release rates, total release, and total doses are shown on Table 14.5-3 for various tank failures and the condenser hotwell failures. The maximum iodine release rate is within the Technical Specification limits for releases from the building exhaust vent.

The dose consequences for the failure of the offgas piping is no more than 2.5 REM total body applied over a 2 hour period at the Exclusion Area Boundary. This limit was endorsed by the NRC in their acceptance of Pilgrim’s limit of 500,000 micro curies per second (referenced to a 30 minute hold-up) of noble gases at the steam jet air ejector contained in the Amendment 89 (Reference 15).

The Safety Evaluation Report contained in Reference 15 endorsed Pilgrim’s use of the NRC guidance contained in NUREG-0133 to meet the intent of NUREG-0473. In Pilgrim’s analysis (Reference 18) an air ejector discharge line break is assumed to release the discharge from the air ejectors for one hour, and from the hold up line and process piping downstream of the break for 2 hours.

The other source of low level activity not enclosed in a Class I structure is the water in the condensate storage tanks. As a worst case, simultaneous nonmechanistic failure of both condensate storage tanks was assumed. Station yard layout and design dictates that the direction of unrestrained surface flow will be towards the intake canal, through the armor stone, and onto the surface of the salt water in the intake canal. The rate and extent of vertical dilution is a function of the relative temperatures and densities of sea water and condensate storage water. The total volume of sea water into which the condensate storage tank would be diluted is a function of: 1) the prevailing wind direction and speed; 2) the wave action inside the intake canal; 3) the tidal cycle; 4) whether or not the circulating water pumps are operating; and 5), the relative temperatures of the ambient air, the sea water, and the condensate storage tank water. Calculations were performed to determine the whole body dose resulting from 30.5 cm, 7.63 cm, and 1 cm, layers of undiluted condensate storage tank water supported by a thermocline between it and the colder salt water below. The estimated doses at the surface of the water were 2.4 x 10^{-2} mrem/hr, 1.4 x 10^{-2} mrem/hr, and 0.3 x 10^{-2} mrem/hr, respectively. These dose rates would be reduced as mixing and dilution occur in the intake channel due to the effects noted previously.

Table 14.5-1

LOSS OF COOLANT ACCIDENT PRIMARY CONTAINMENT RESPONSE SUMMARY
ORIGINAL 65° F SSW CASES (6)

	Case A	Case B	Case C
Secondary Peak Pressure, psig	None (1)	13.1	8.0 (5)
Peak Pool Temperature, °F	142	166	166
Mode of RHR operation Pool Cooling	Suppression Pocl Cooling	Suppression\` Pool Cooling	Containment Spray
Number of RHR loops operating	2	1	1
Number of RHR pumps operating	2	1	1
Number of RHR heat exchangers operating	2	1	1
Total RHR flowrate, gpm	10,000	5,000	5,000
Core Spray System flowrate gpm	3,600 (2)	3,600 (2)	3,600 (2)
Containment Spray System flowrate, gpm	0	0	5,000 (5)
RHR heat exchanger flowrate, gpm	10,000	5,000	5,000
RHR heat transfer rate when suppression pool temperature = 165°F, Btu/hr	128×10^6	64×10^6	64×10^6
Number of RBCCW loops operating	2	1	1
Number of RBCCW pumps operating	4	2	2
Number of RBCCW heat exchangers operating	2	1	1
Total RBCCW flowrate to RHR, gpm	5,400	2,700	2,700
Number of SSW loops operating	2	1	1
Number of SSW pumps operating	4	2	2
Total SSW flowrate to RBCCW	10,000 (3)	5,000 (4)	5,000 (4)
SSW inlet water temperature, °F	65	65	65

Table 14.5-1 (Cont)

NOTES FOR LOSS OF COOLANT ACCIDENT PRIMARY CONTAINMENT RESPONSE SUMMARY

NOTES:

1. Pressure steadily decreases after containment cooling is established.
2. Use of Technical Specification value of 3240 gpm would have negligible impact on containment response.
3. The RBCCW Heat Exchangers are capable of removing 130 x 10^6 Btu/hr with a combined SSW Flow of 9,000 gpm (4,500 gpm for each Hx)
4. RBCCW Heat Exchanger is capable of removing 65 x 10^6 Btu/hr with a SSW flow rate of 4500 gpm.
5. Containment spray flowrate was reduced, see section 14.5.3.1.2, Case "C".
6. This table contains the original containment heatup analysis parameters for temperatures and flowrates, some of which were superseded by the 75° F SSW Case in Table 14.5-6.

Table 14.5-2

RADIOLOGICAL DOSES FOR LOSS OF COOLANT AND STEAM LINE BREAK ACCIDENTS BASED ON TID-14844 ASSUMPTIONS

Accident Evaluated		Dose Location	Radiological Exposure (rem) Cloud	Inhalation
Loss of Coolant	(2 hour)	EA	2.9×10^{0}	9.8×10^{1}
	(30 day)	LPZ	7.0×10^{-1}	1.3×10^{1}
Steamline Break	(2 hour)	EA	1.0×10^{-2}	7.6×10^{1}
	(30 day)	LPZ	3.6×10^{-3}	7.0×10^{-1}

NOTES:

EA - Exclusion Area (320 m)
LPZ - Low Population Zone (6,840 m)

Table 14.5-3

OFFSITE THYROID DOSES RESULTING FROM FAILURES OF CLASS II COMPONENTS CONTAINING LIQUID WITH SIGNIFICANT AMOUNTS OF RADIOACTIVE MATERIALS NOT ENCLOSED IN CLASS I STRUCTURES

Vessel Name	Maximum Iodine Release Rate through Building Exhaust Vent (microcuries/sec)	Total Iodine Released (curies)	2 Hour Thyroid Dose at 320 Meters (Mrem)	30 Day Thyroid Dose at 6,800 Meters (Mrem)
Condenser Hotwell	3.3×10^{-4}	4.0×10^{-5}	5.2×10^{-4}	6.5×10^{-5}
Chemical Waste Receiver Tank	2.6×10^{-2}	1.1×10^{-2}	1.2×10^{-1}	1.6×10^{-6}
Monitor Tank	2.6×10^{-2}	1.1×10^{-2}	1.2×10^{-1}	1.6×10^{-2}
Treated Water Holdup Tank	0.7×10^{-5}	2.6×10^{-5}	2.9×10^{-4}	4.0×10^{-5}
Clean Waste Receiver Tank	4.1×10^{-1}	1.6×10^{-1}	1.75	2.4×10^{-1}
Total Tankage Failure	3.0×10^{-2}	1.8×10^{-2}	1.3×10^{-1}	1.9×10^{-2}

Table 14.5-4

BOUNDING CORE AND FHA INVENTORIES

Radionuclide	Undecayed Inventory (CI)		Fuel Rod Gap Fraction	FHA Undecayed Source Term (CI)
	Full Core	Peak Assembly		
BR 82	6.872E+05	2.488E+03	0.05	2.042E+02
BR 82M	2.656E+05	9.617E+02	0.05	7.892E+01
BR 83	8.640E+06	3.128E+04	0.05	2.567E+03
BR 84	1.593E+07	5.768E+04	0.05	4.733E+03
BR 84M	4.468E+05	1.618E+03	0.05	1.328E+02
BR 85	1.957E+07	7.086E+04	0.05	5.815E+03
BR 86	1.466E+07	5.308E+04	0.05	4.356E+03
BR 87	3.339E+07	1.209E+05	0.05	9.921E+03
BR 88	3.803E+07	1.377E+05	0.05	1.130E+04
I128	1.919E+06	6.948E+03	0.05	5.702E+02
I129	6.033E+00	2.184E-02	0.05	1.793E-03
I130	4.655E+06	1.685E+04	0.05	1.383E+03
I130M	1.818E+06	6.582E+03	0.05	5.402E+02
I131	5.716E+07	2.070E+05	0.08	2.717E+04
I132	8.113E+07	2.937E+05	0.05	2.411E+04
I133	1.150E+08	4.164E+05	0.05	3.417+04
I134	1.284E+08	4.649E+05	0.05	3.815E+04
I134M	1.371E+07	4.964E+04	0.05	4.074E+03
I135	1.071E+08	3.878E+05	0.05	3.182E+04
I136	5.198E+07	1.882E+05	0.05	1.544E+04
I136M	3.179E+07	1.151E+05	0.05	9.446E+03
KR 83M	8.638E+06	3.128E+04	0.05	2.567E+03
KR 85	1.439E+06	5.210E+03	0.1	8.551E+02
KR 85M	1.979E+07	7.165E+04	0.05	5.880E+03
KR 87	3.956E+07	1.432E+05	0.05	1.175E+04
KR 88	5.592E+07	2.025E+05	0.05	1.662E+04
KR 89	7.054E+07	2.554E+05	0.05	2.096E+04
KR 90	7.004E+07	2.536E+05	0.05	2.081E+04
XE131M	6.4123+05	2.322E+03	0.05	1.905E+02
XE133	1.150E+08	4.164E+05	0.05	3.417E+04
XE133M	3.541E+06	1.282E+04	0.05	1.052E+03
XE135	5.869E+07	2.125E+05	0.05	1.744E+04
XE135M	2.297E+07	8.317E+04	0.05	6.825E+03
XE137	1.012E+08	3.664E+05	0.05	3.007E+04
XE138	1.022E+08	3.700E+05	0.05	3.037E+04
XE139	8.237E+07	2.982E+05	0.05	2.447E+04

Table 14.5-4A

Atmospheric Dispersion Factors (X/Q's)
For Control Room, EAB and LPZ

No	Release Point	Receptor Point	Interval	Concentration (χ/Q) (sec/m^3)	Gamma (χ/Q) (sec/m^3)
1	Reactor Building Vent	EAB (actual)[a]	0-2 hrs	7.479E-04	3.199E-04
2		LPZ(4.25 miles)	0-2 hrs	3.692E-05	3.551E-05
			2-8 hrs	1.915E-05	1.782E-05
			8-24 hrs	1.066E-05	9.627E-05
			24-96 hrs	4.339E-06	3.745E-05
			96-720 hrs	1.194E-06	9.656E-07
3		Control Room Fresh Air Intake	0-2 hrs	1.76E-03	N/A
			2-8 hrs	1.25E-03	
			8-24 hrs	4.26E-04	
			24-96 hrs	3.67E-04	
			96-720 hrs	3.15E-04	
4	Truck Lock Door	EAB (actual)	0-2 hrs	See RB vent[b]	
5		LPZ (4.25 miles)	0-720 hrS		
6		Control Room Fresh Air Intake	0-2 hrs	9.72E-04	N/A
			2-8 hrs	7.52E-04	
			8-24 hrs	2.80E-04	
			24-96 hrs	1.93E-04	
			96-720 hrs	1.61E-04	

(a) The EAB distances employed in the atmospheric dispersion analysis are from the closest point of the reactor building; as such, they conservatively apply to releases via the RB vent, which is a the plant SW corner. The critical receptor is in the true NE sector, at a distance of 486 m (at the over-water exclusion zone).

(b) The atmospheric dispersion factors for releases via the Truck Lock (which is in the plant W side, near the NW corner of the building site), are conservatively bounded by releases from the reactor building proper.

Table 14.5-5

RADIOLOGICAL CONSEQUENCES OF FUEL
HANDLING DESIGN BASIS ACCIDENT (REM)

LOCATION	Exposure interval	Unfiltered Outside Air Intake Rate	TEDE Dose (rem)	Regulatory Limit (rem)	Percent of Regulatory Limit
Control Room	30 days	1000 cfm	2.846	5	56.9
		9000 cfm	2.863	5	57.3
EAB	2 hrs	N/A	1.439	6.3	22.8
LPZ	30 days	N/A	0.0920	6.3	1.46

Table 14.5-6

LOSS OF COOLANT ACCIDENT PRIMARY
CONTAINMENT RESPONSE SUMMARY

	75°F SSW Case
Peak Wetwell Pressure, psia	22.7
Peak Pool Temperature, °F	185
Mode of RHR Operation	2 Pump LPCI-Heat Rejection Mode and 1 Pump LPCI-Heat Rejection Mode
Number of RHR Loops operating	1
Number of RHR Pumps operating	2 Pumps from 600 seconds to 7200 seconds 1 Pump from 7200 seconds to 30 days
Number of RHR Heat Exchangers	1
Total LPCI - Heat Rejection Mode Flowrate, gpm	9,500 at zero reactor pressure (2 pump) 5,100 at zero reactor pressure (1 pump)
Core Spray System Flowrate, gpm	4,100 at zero reactor pressure
RHR Heat Exchanger Flowrate, gpm	3,430 (2 pump LPCI-Heat Rejection Mode) 5,100 (1 pump LPCI-Heat Rejection Mode)
RHR Heat Transfer Rate at peak suppression pool temperature, Btu/hr	70.8×10^6
Number of RBCCW loops operating	1
Number of RBCCW pumps operating	2 each, 10 minutes after the accident
Number of RBCCW heat exchangers operating	1
Total RBCCW flow rate to RHR, gpm	3050
Number of SSW loops operating	1
Number of SSW pumps operating	2 each, 10 minutes after the accident
Total SSW flow rate to RBCCW	4,500
SSW inlet water temperature, °F	75

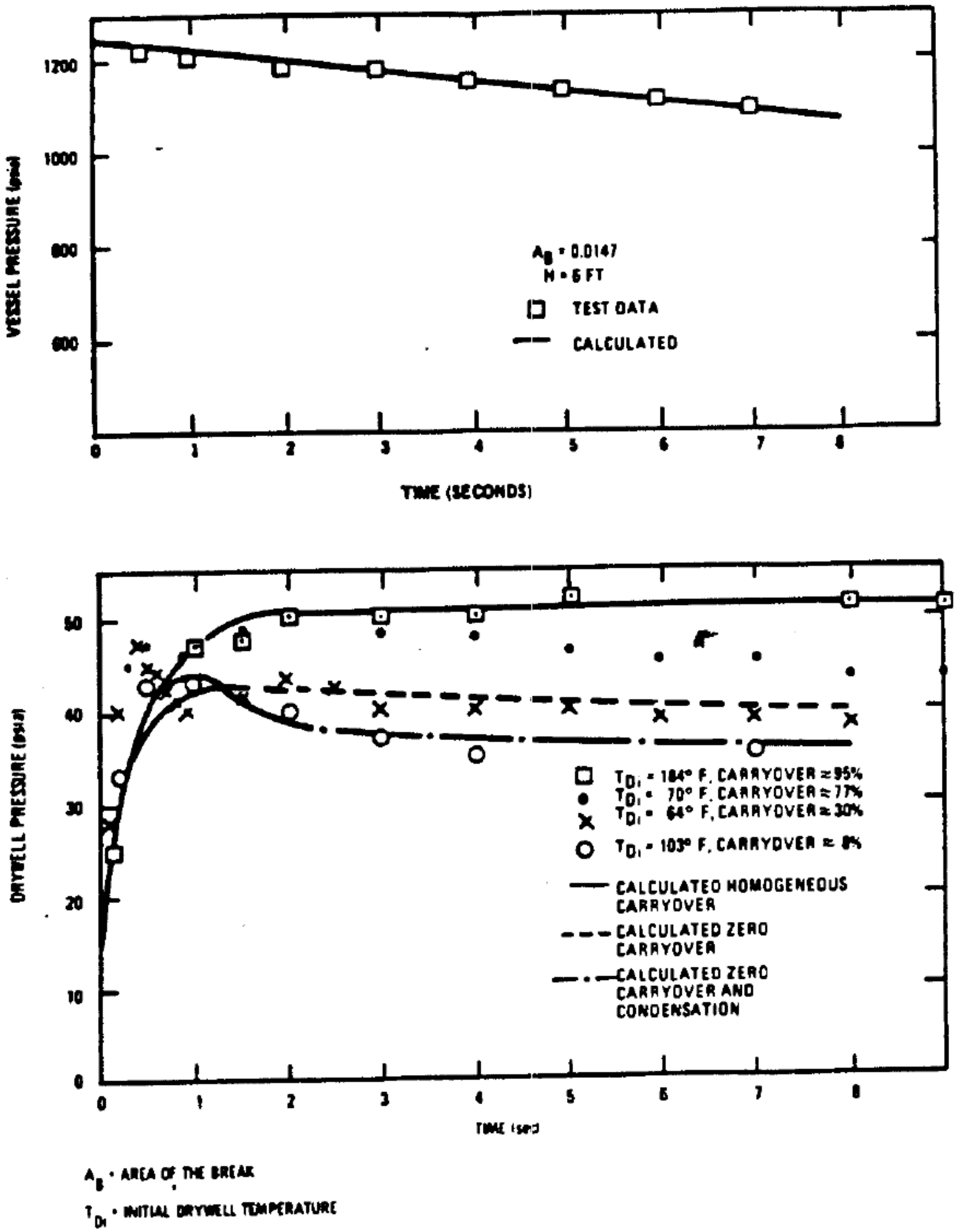


Figure 14.5-1. Loss of Coolant Accident Humboldt - Primary Containment Pressure Response

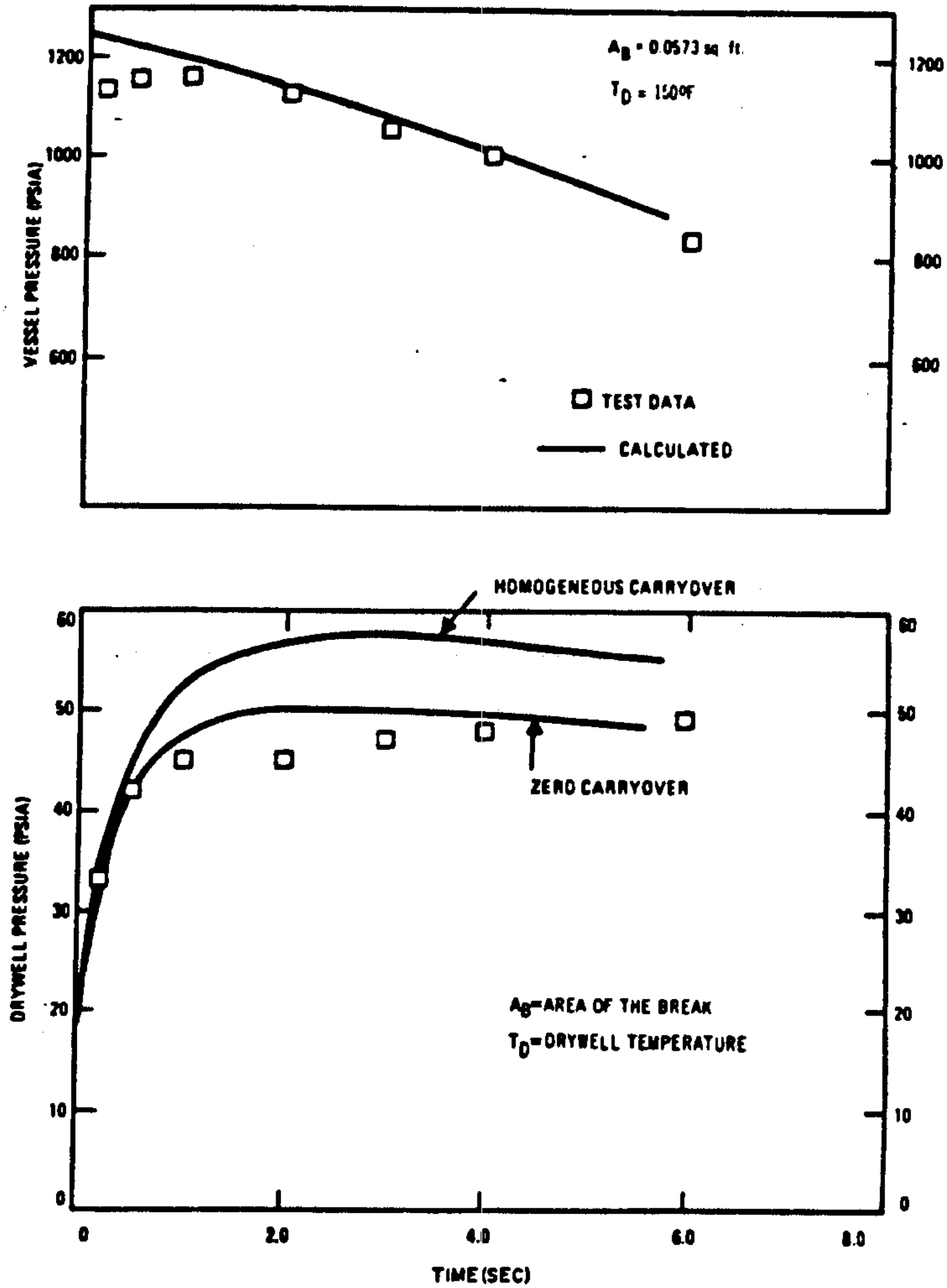


Figure 14.5-2. Loss of Coolant Accident Bodega Bay - Primary Containment Pressure Response

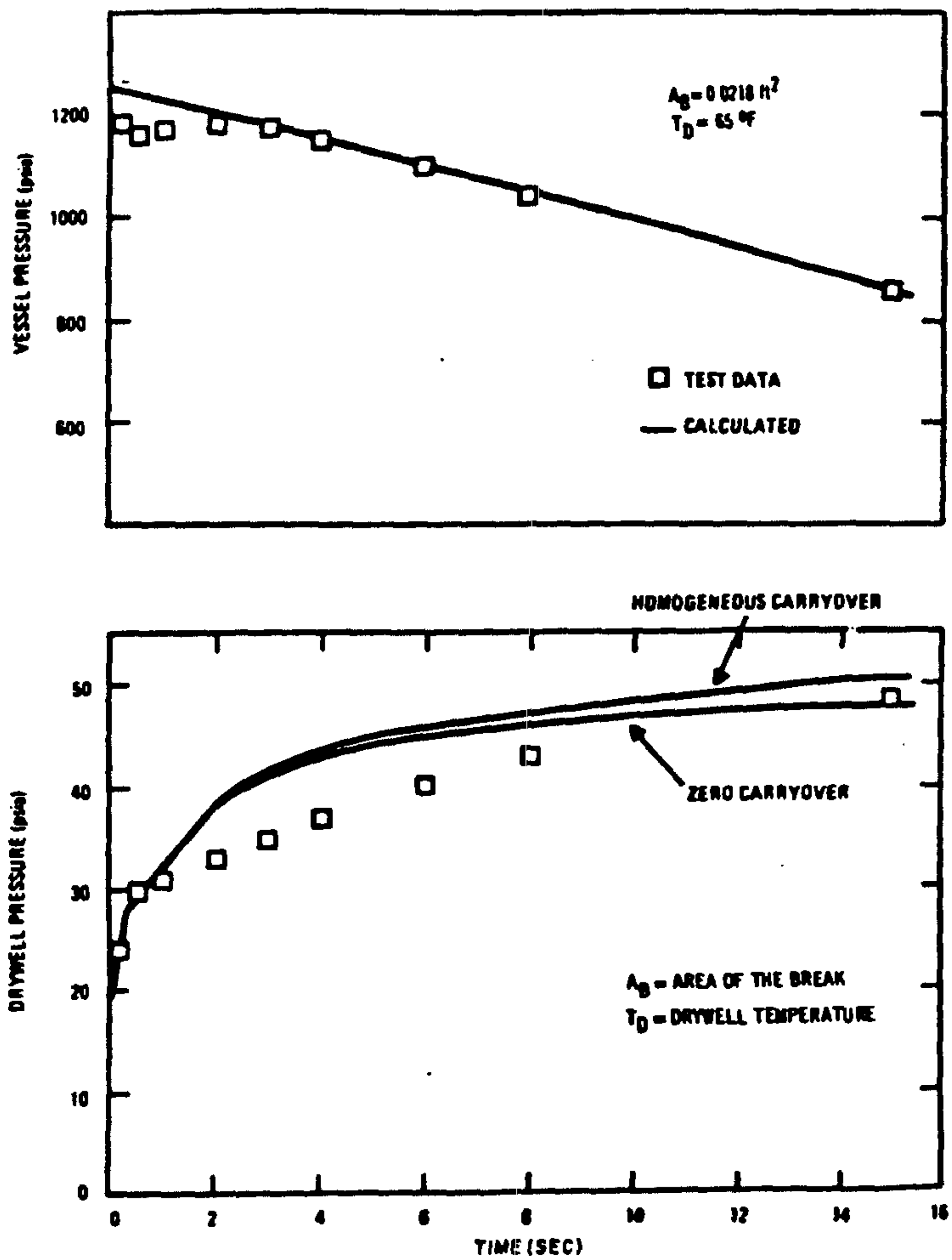


Figure 14.5-3. Loss of Coolant Accident Bodega Bay - Primary Containment Pressure Response

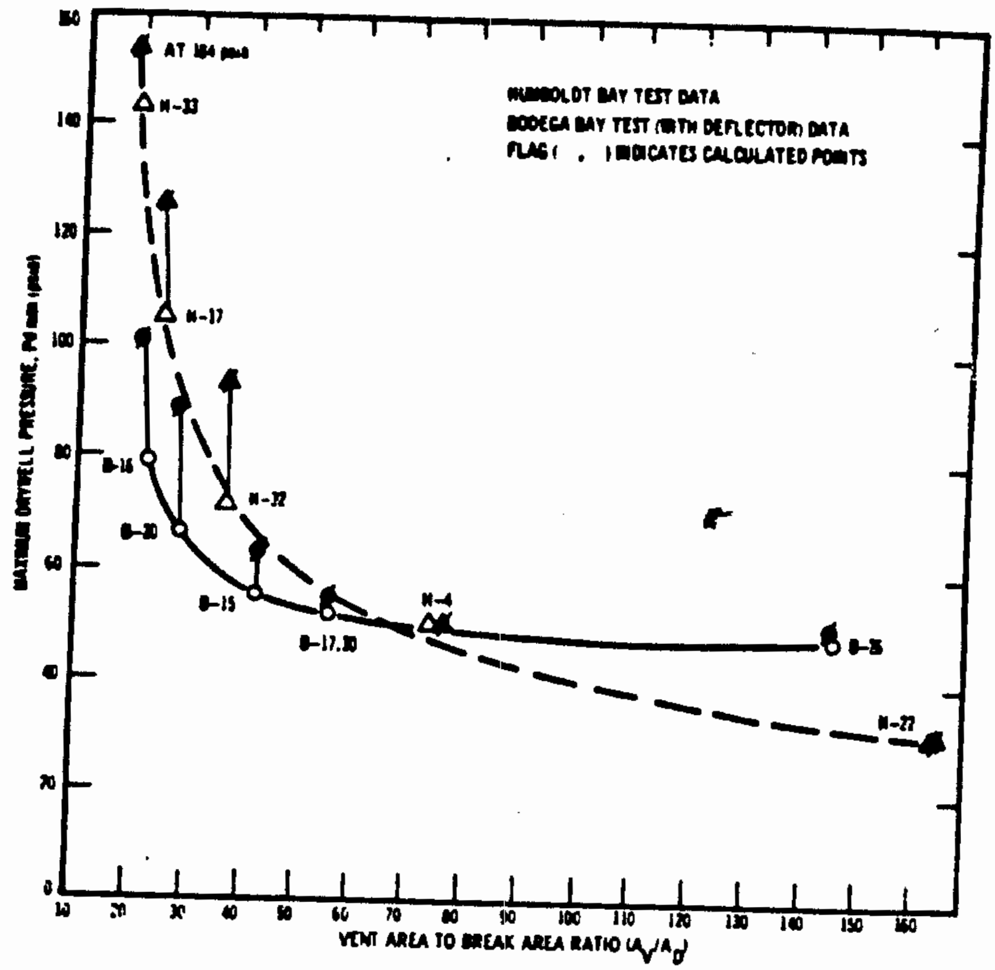


Figure 14.5-4. Loss of Coolant Accident, Comparison of Calculated and Measured Peak Drywell Pressured for Bodega Bay and Humboldt

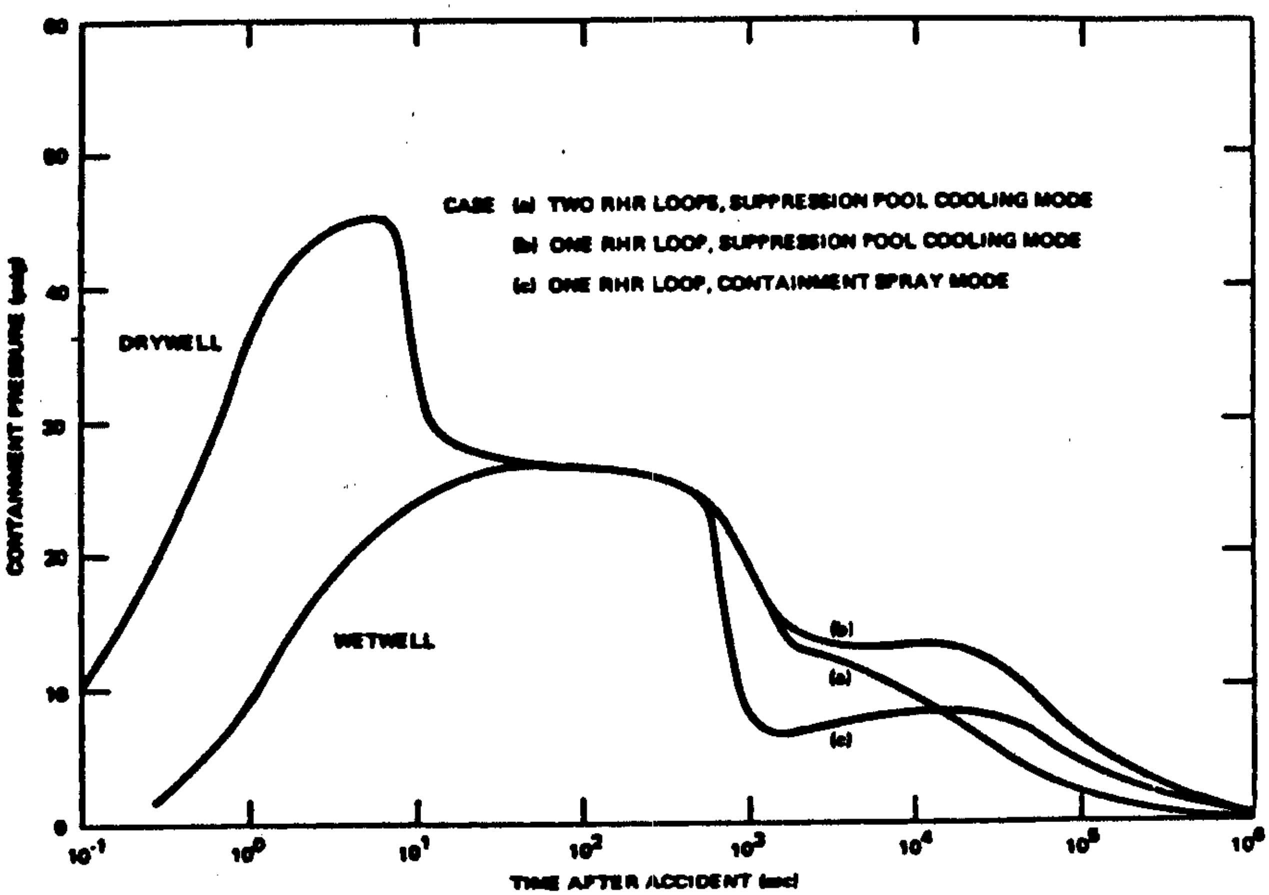


Figure 14.5-5. Loss of Coolant Accident, Primary Containment Pressure Response

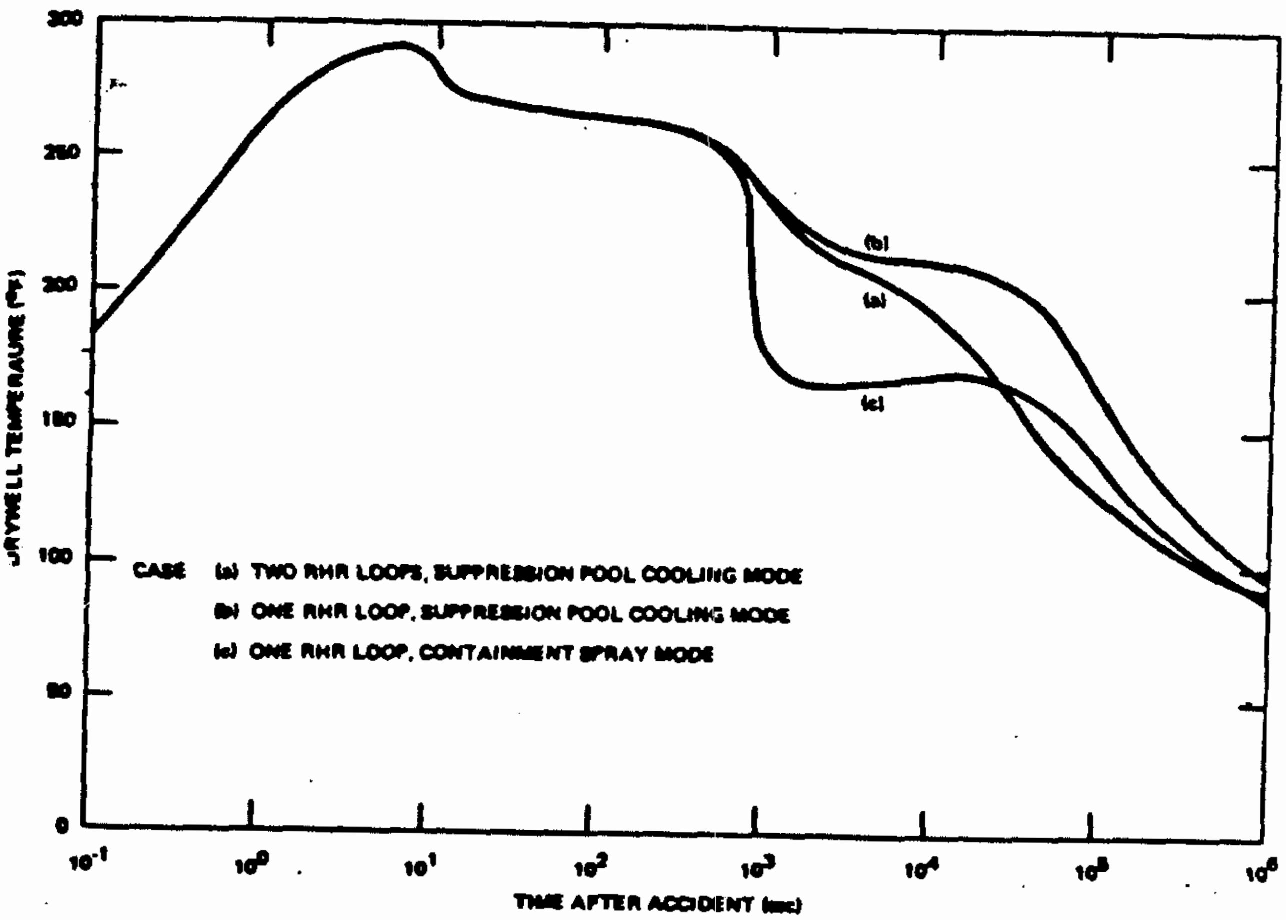


Figure 14.5-6. Loss of Coolant Accident Drywell Temperature Response

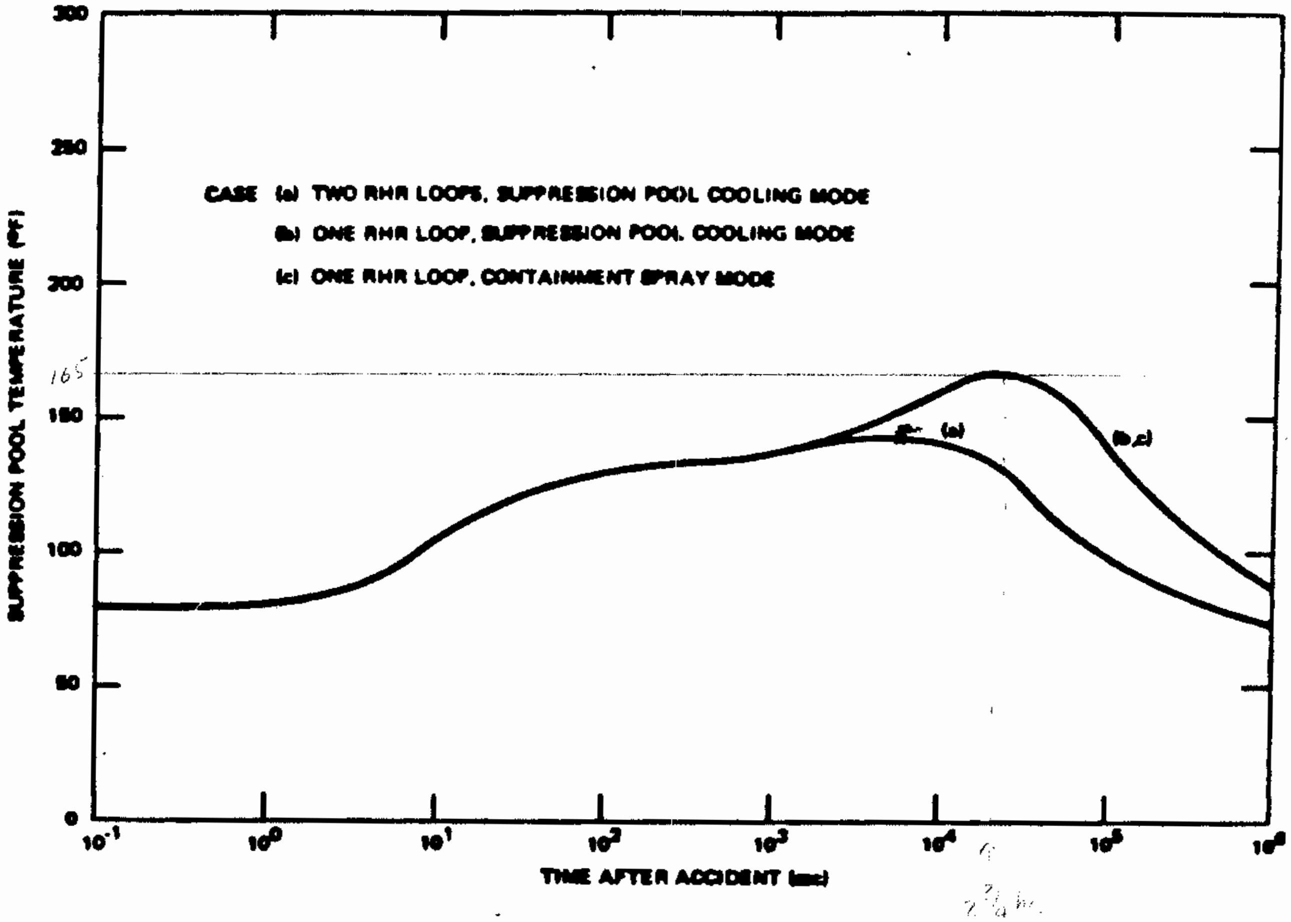


Figure 14.5-7. Loss of Coolant Accident Suppression Pool Temperature Response

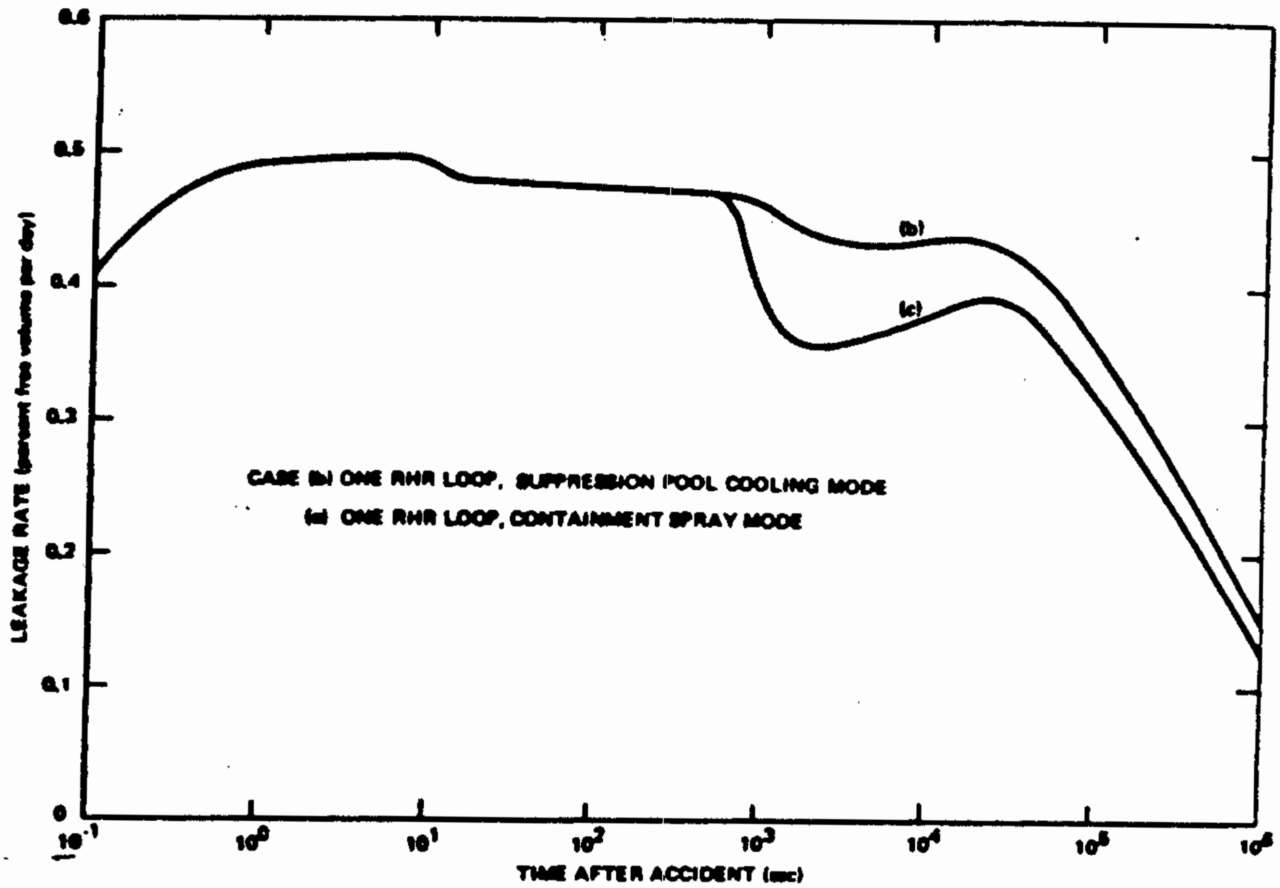
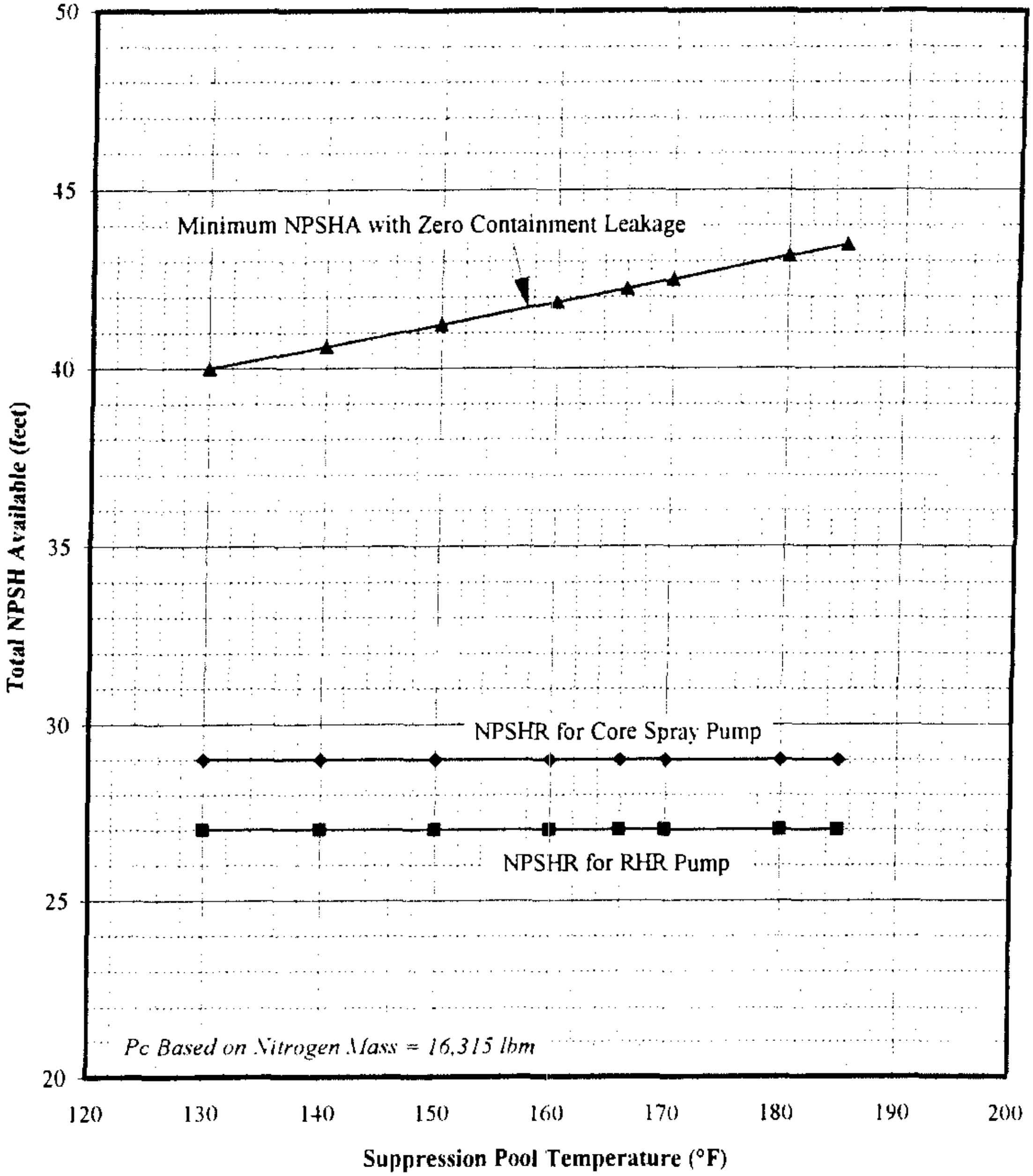


Figure 14.5-8. Loss of Coolant Accident Primary Containment Leak Rate



Initial Conditions:

Drywell 150°F, 80% RH, 1.3 psig, 132,000 ft^3

Wetwell 85°F, 100% RH, 0 psig, 124,500 ft^3

Figure 14.5-9

Total NPSH Available at Maximum Flow

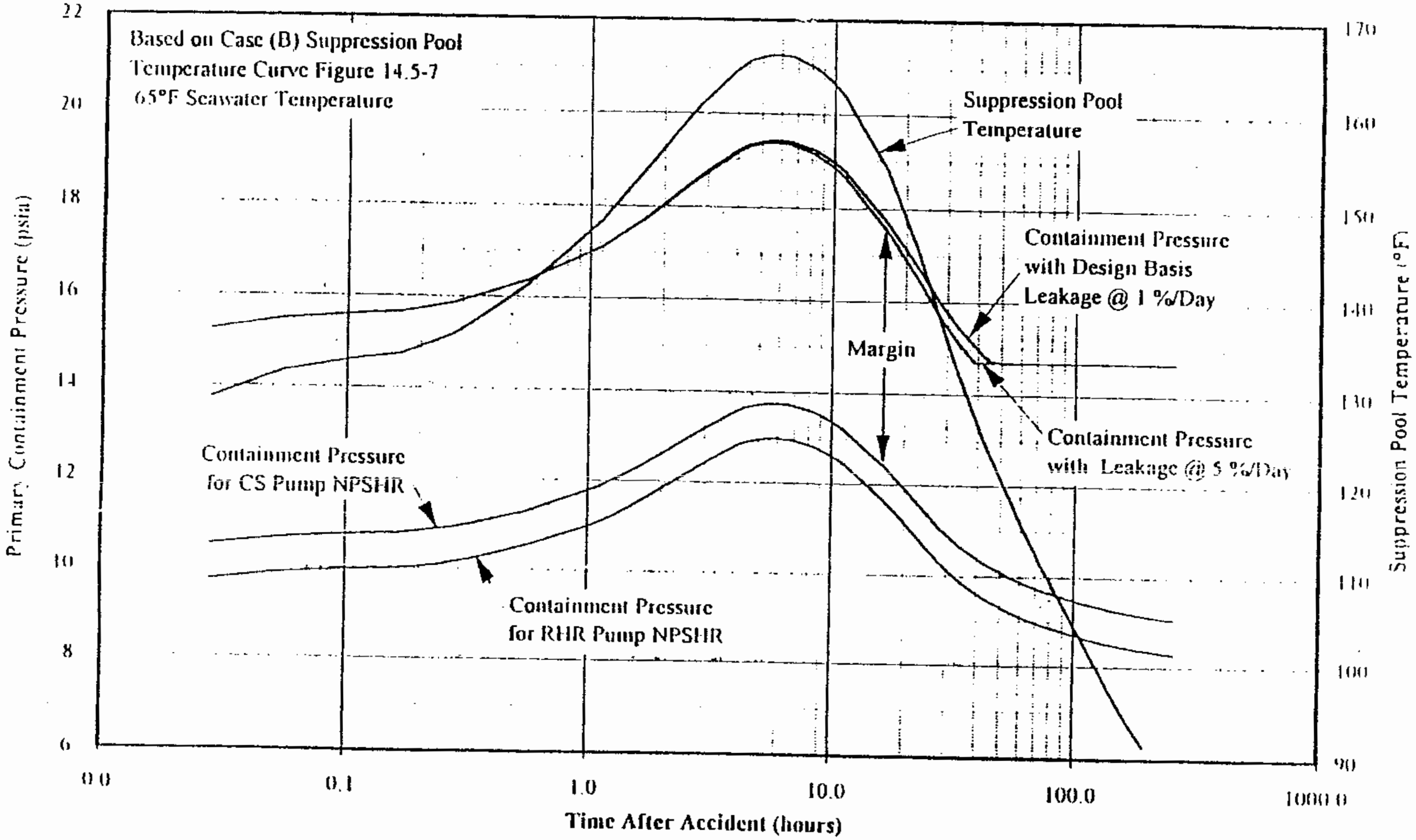
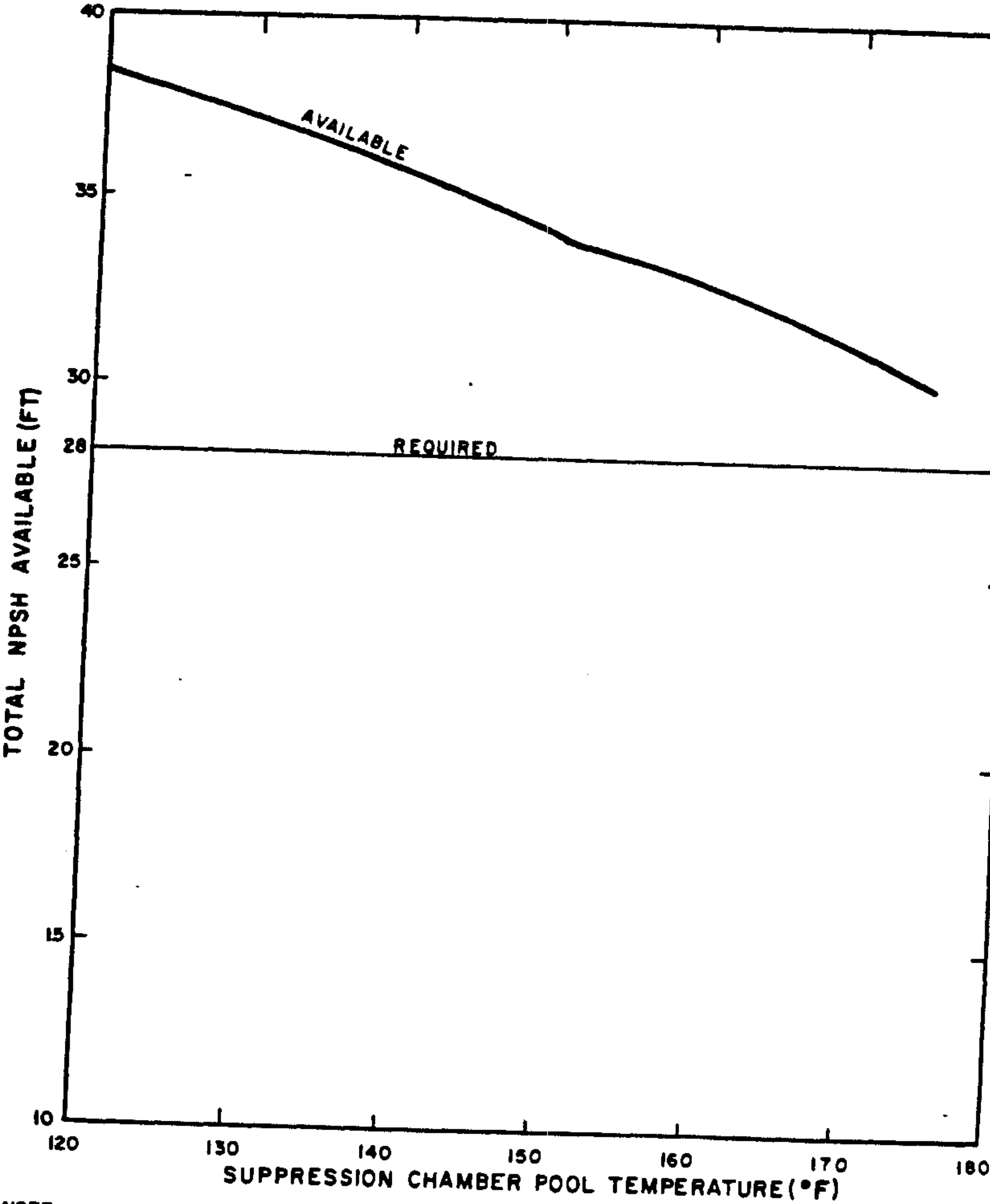


Figure 14.5-10

NPSH Availability for RHR and Core Spray Pumps after a DBA-LOCA



NOTE

DRYWELL INITIALLY SATURATED
AT 150°F 0.0 PSIG

DRYWELL AT 110°F 20% RELATIVE
HUMIDITY DURING RCIC OPERATION

Figure 14.5-11. Total NPSH Available During RCICS Operation - Rated Flow

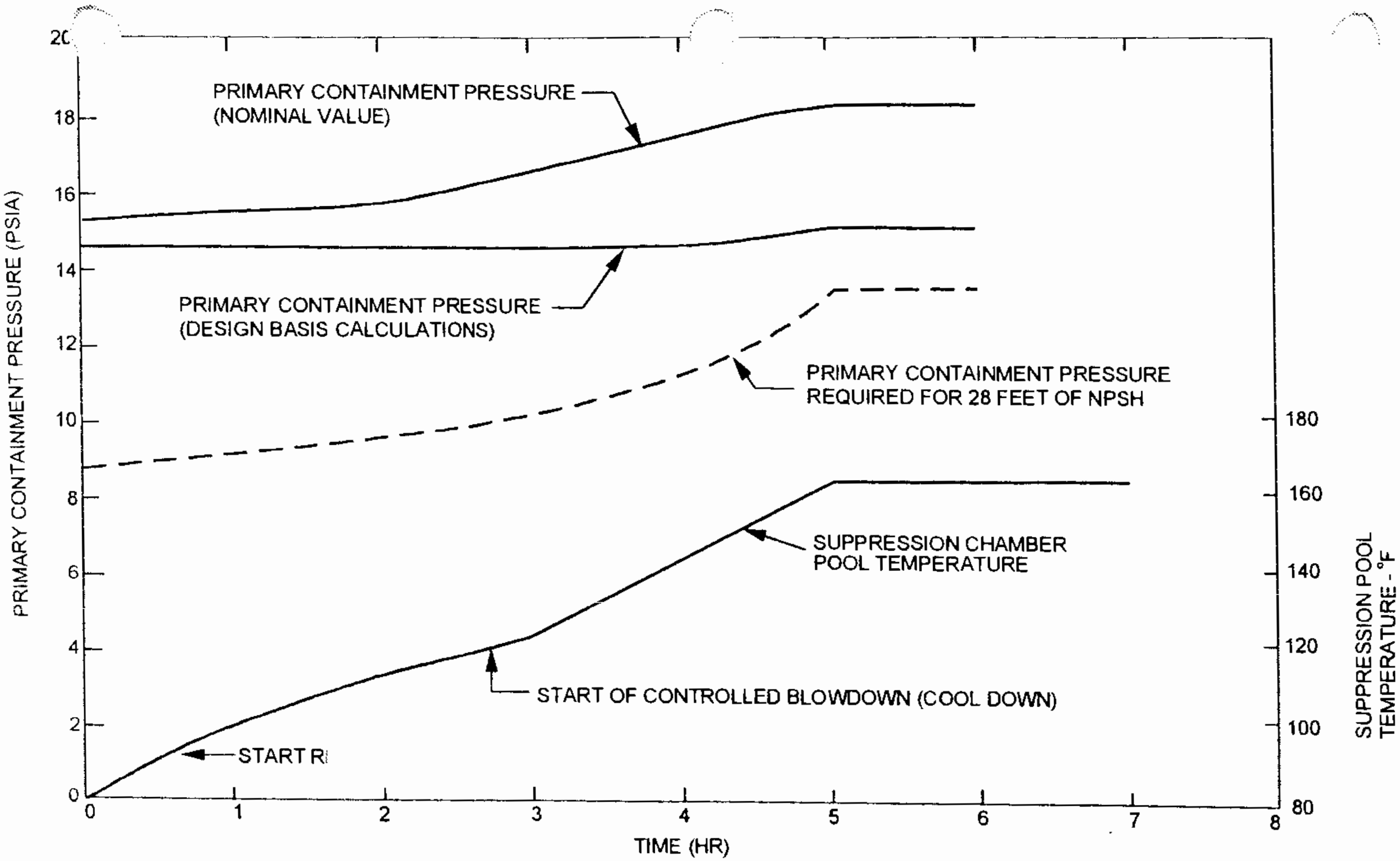


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Figure 14.5-12
NPSH Availability During RCIC Operation and Reactor Depressurization (Cool Down)

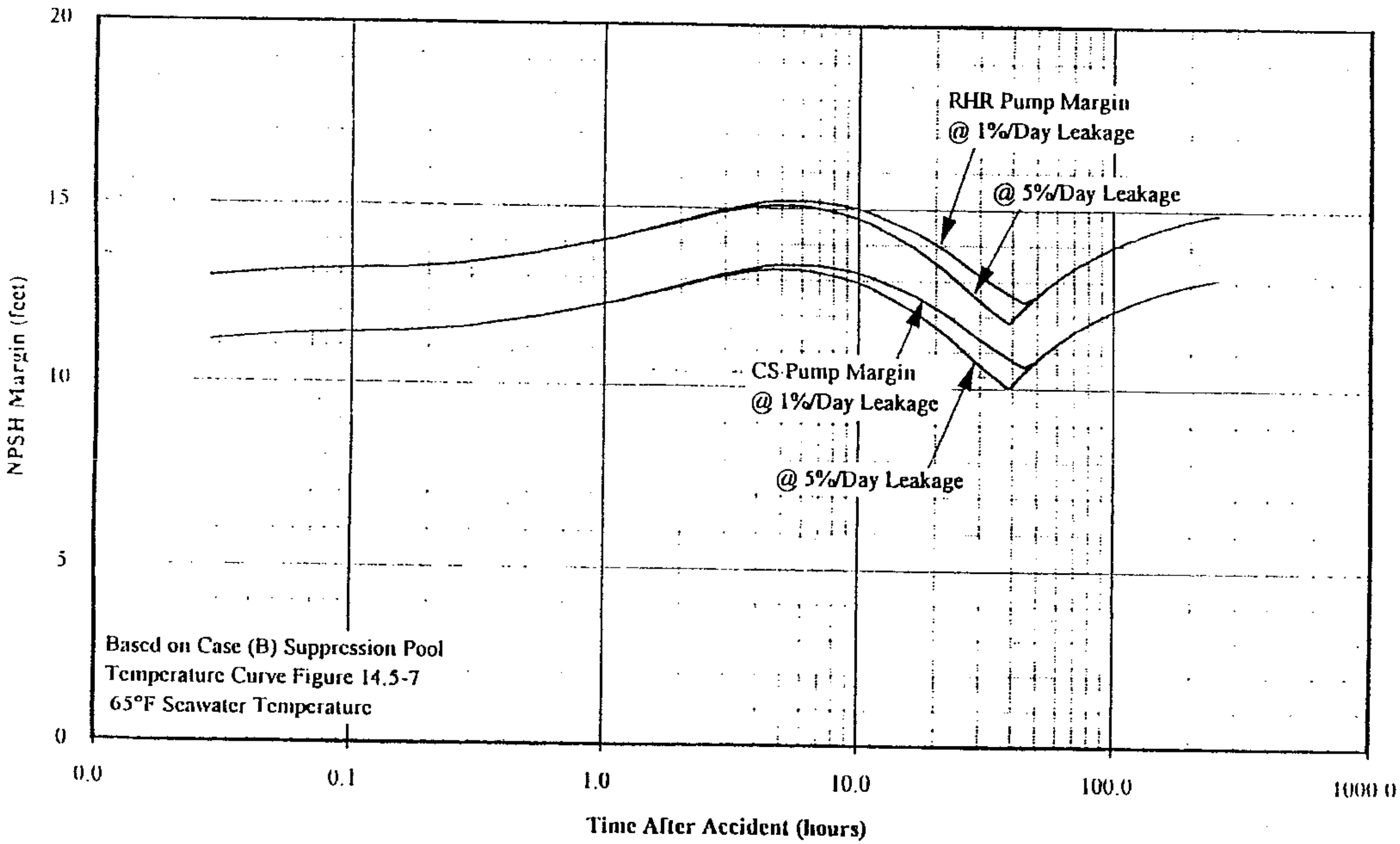

Figure 14.5-13
NPSH Margin for RHR and Core Spray Pumps after a DBA-LOCA

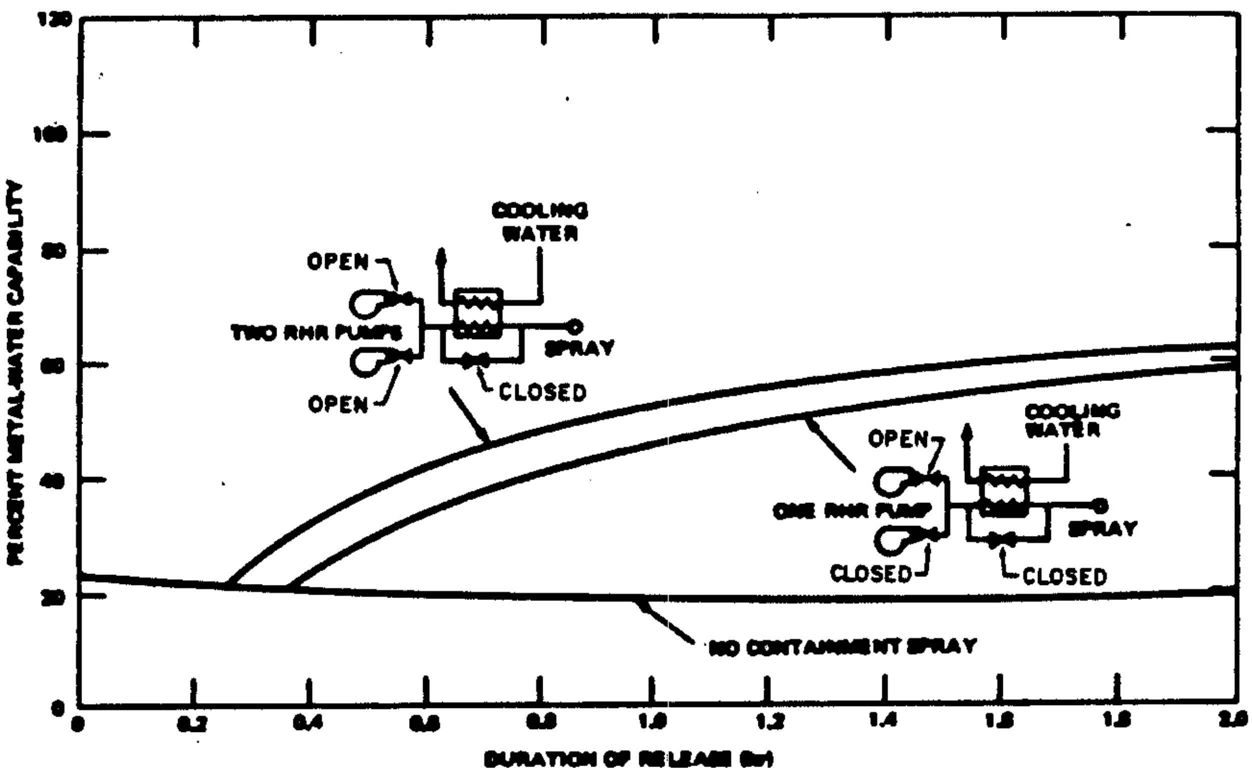

Figure 14.5-14. Initial Core Loss of Coolant Accident Primary Containment Capability Index for Metal - Water Reaction

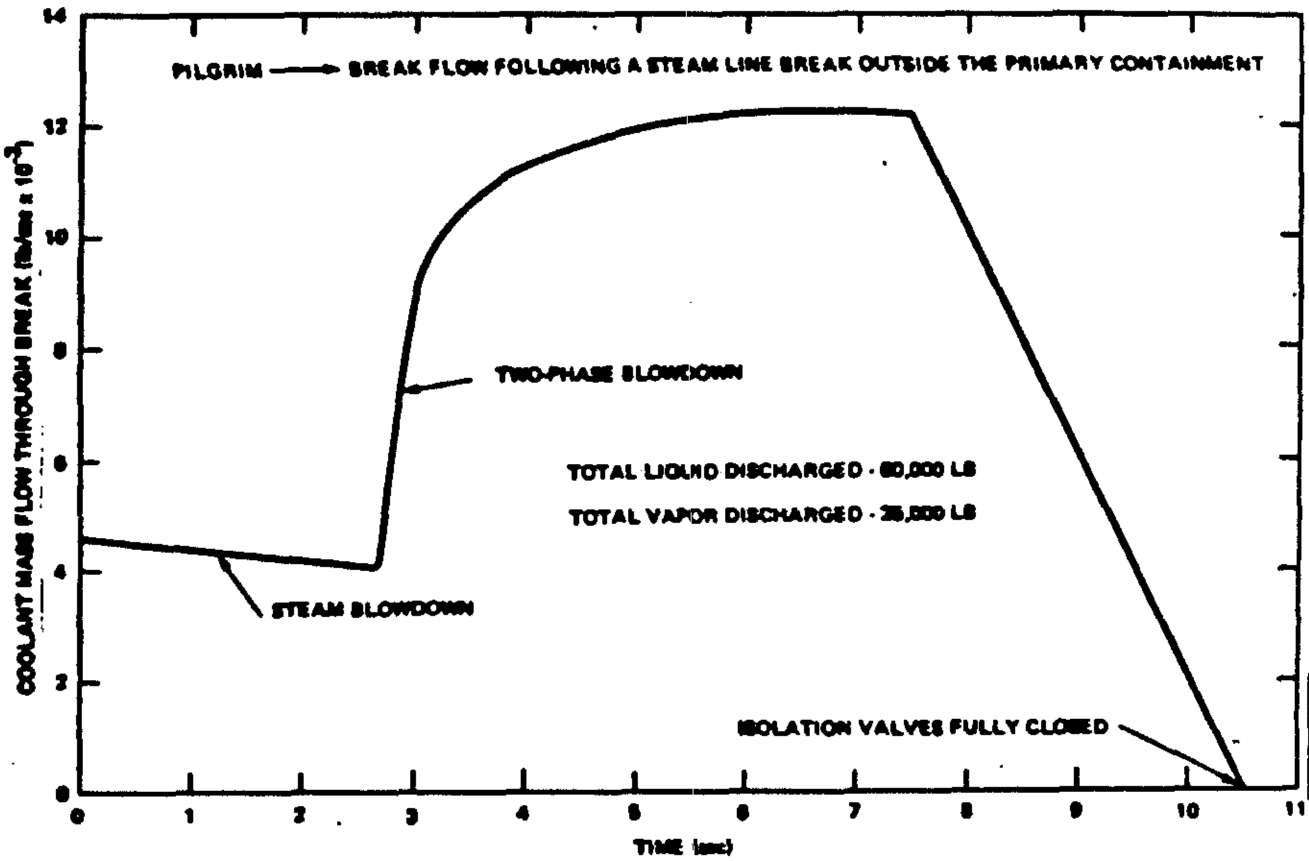

Figure 14.5-15. Main Steam Line Break Accident Mass of Coolant Lost Through Break

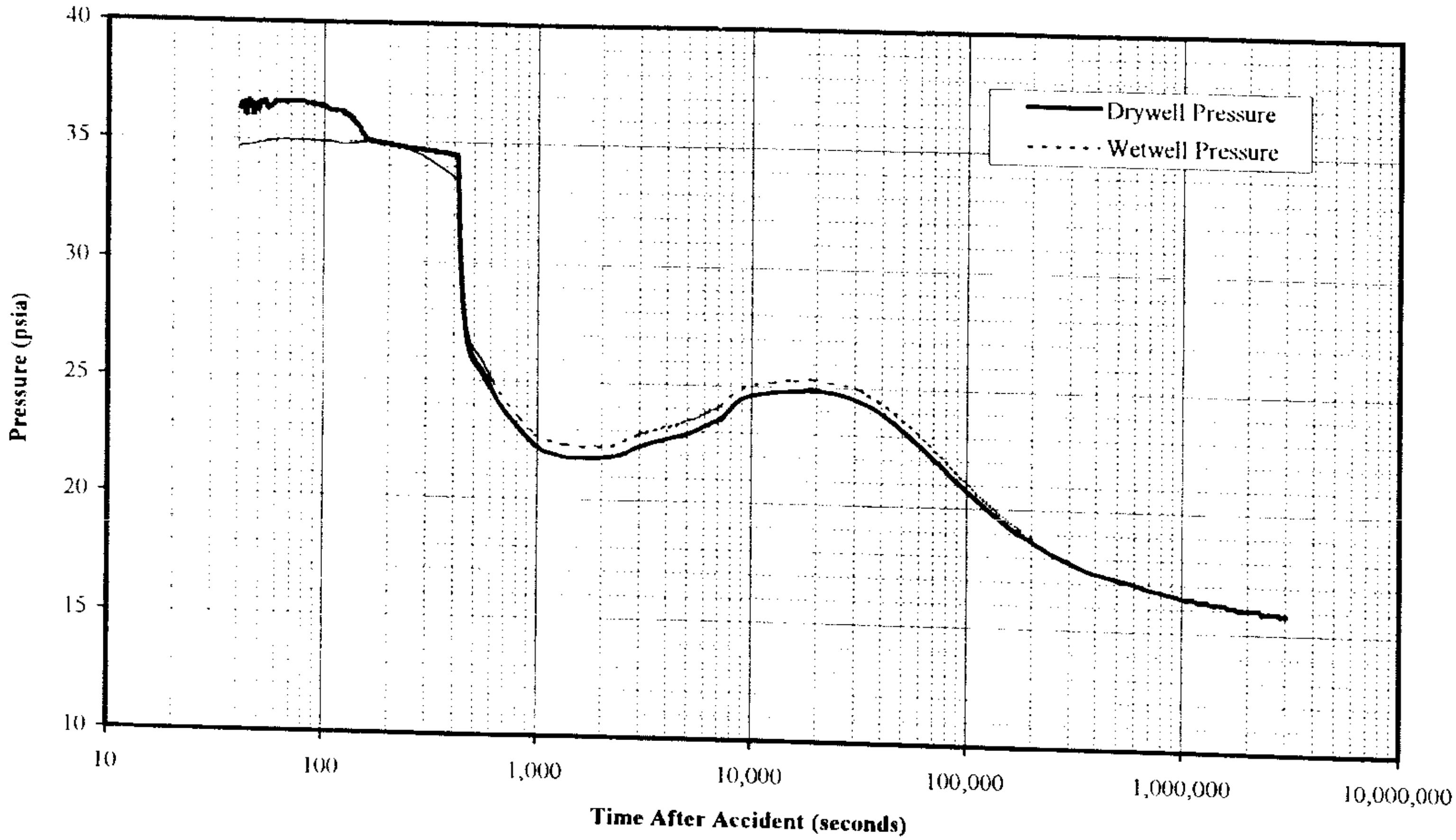

Figure 14.5-16

Design Basis LOCA Long-Term Containment Pressure Response 75°F SSW Case

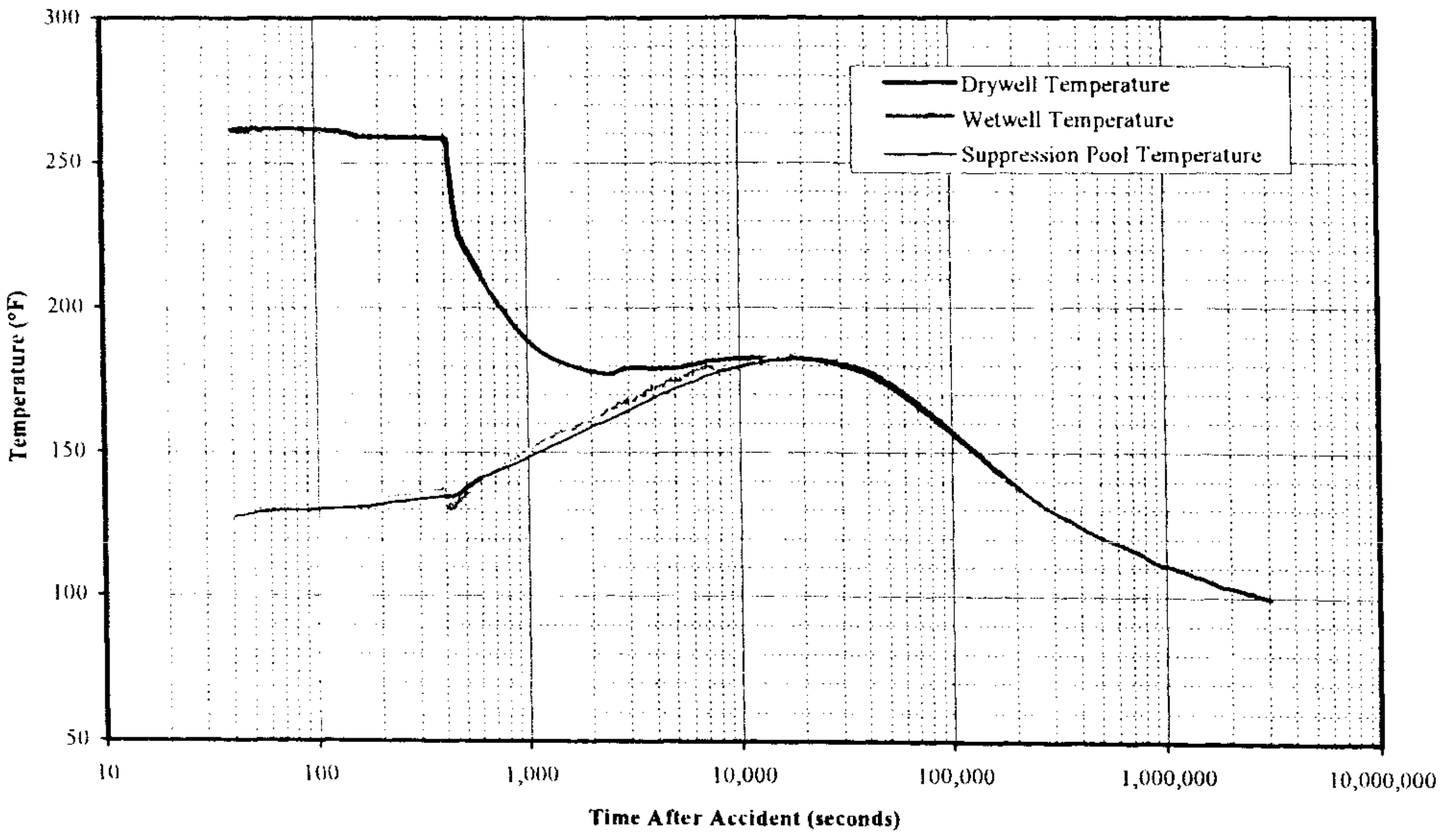

Figure 14.5-17

Design Basis LOCA Long-Term Containment Temperature Response 75°F SSW Case

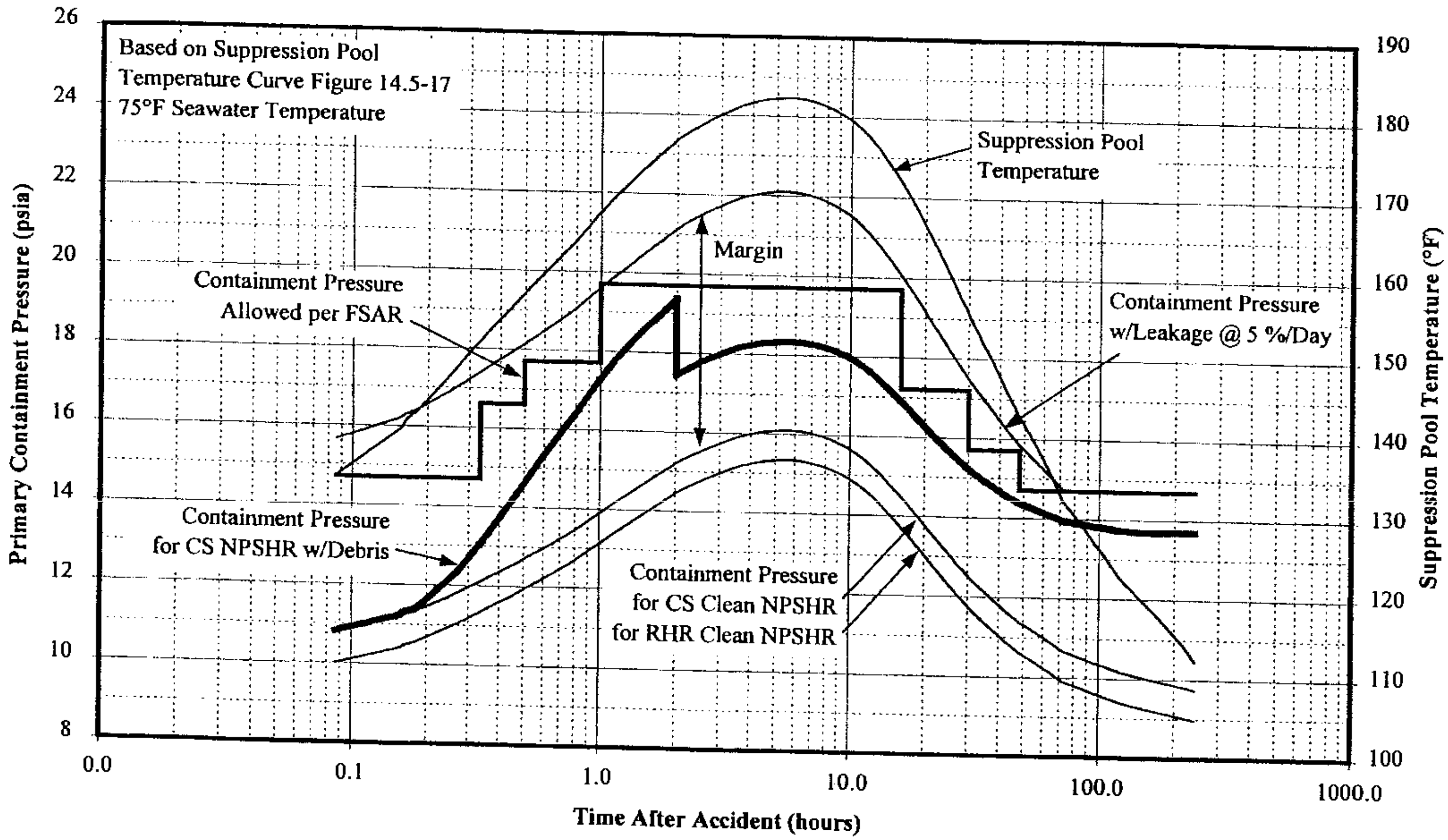

Figure 14.5-18

NPSH Availability and Allowable Limits for RHR and Core Spray Pumps After a DBA-LOCA

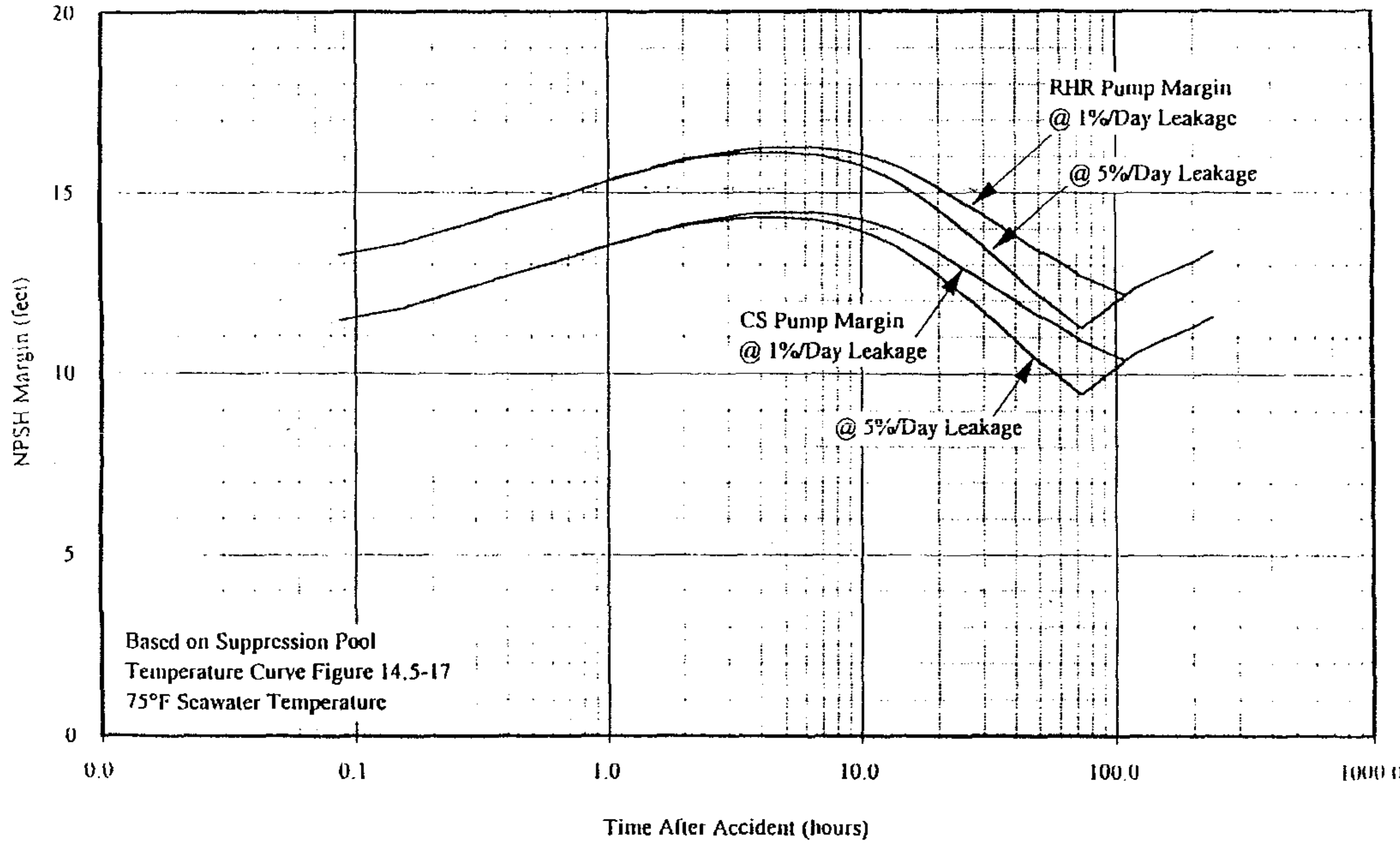

Figure 14.5-19
NPSH Margin for RHR and Core Spray Pumps after a DBA-LOCA

14.6 SPECIAL EVENTS

14.6.1 Station Shutdown from Outside the Control Room

This special event is presented to demonstrate the capability to perform the operations required to maintain the station in a safe condition from outside the control room.

14.6.1.1 Assumptions

1. The station is operating initially at full power.

2. Station personnel evacuate the control room taking time only for those immediate actions within the control room that can be accomplished in seconds.

3. Station personnel take all subsequent action required to bring the reactor to a cold shutdown condition using controls and equipment located outside the control room.

14.6.1.2 Evaluation - Achievement of Hot Shutdown Condition

Emergency operating procedures for the station specify that the only immediate action to be taken (requiring only seconds of delay) prior to control room evacuation is to announce the evacuation of the control room and direct required personnel to assemble at the emergency assembly point. After establishment of emergency communications, the station is shut down by either tripping average power range monitor or venting the control rod drive scram air header.

No further reactivity shutdown action is required following scram. During normal power operation, less than one half of the withdrawn control rods need to be inserted to shut down the reactor. The buildup of core xenon content after scram would provide increased shutdown margin for several hours. The position of the scram valves for the individual drives can be verified in the Reactor Building as a check that control rods were inserted.

No further core cooling action external to the control room is required for hours. The Feedwater System would continue to maintain reactor vessel water level automatically for an extended period of time. The steam bypass valves will continue to dissipate decay heat automatically until reactor pressure falls below the set point of the turbine initial pressure regulator.

The thermal losses from the reactor system combined with the normal steam flow to the turbine seals will exceed the thermal output from reactor decay heat, resulting in gradual cooldown and depressurization of the reactor starting within a few hours after shutdown. This gradual depressurization would require no operator action other than periodic verification of reactor pressure and water level conditions. Reactor vessel water level and pressure are indicated locally in the

Reactor Building so that the operator can monitor these parameters.

It is concluded that the requirements of criterion 1 of Section 14.2.3.2.1 are satisfied by the station design.

14.6.1.3 Evaluation - Achievement of Cold Shutdown Condition

Under most conditions, it is desirable to have the reactor in the hot shutdown condition until the control room can be reentered. This minimizes operator action under adverse conditions. The longer term actions and procedures required to take the station from the hot shutdown condition to the cold shutdown condition can be accomplished from outside the control room by using the high pressure coolant injection (HPCI) in the test mode, the residual heat removal (RHR) in the suppression pool cooling mode, and finally, the RHR in these systems can be operated locally at the switchgear.

Cooling flow to the RHR heat exchangers from the Reactor Building Closed Cooling Water System requires the opening of one motor-operated valve and the startup of one standby closed loop pump. The starting of a standby (salt water) service water pump may also be required. These operations can be done external to the control room. Reactor coolant and reactor vessel water level are indicated in the Reactor Building thus allowing the operator to monitor these principal reactor parameters during cooldown.

When these steps are completed, the reactor has been brought to the cold shutdown condition and could remain in this condition for an unlimited period of time without requiring access to the control room. Alternate sequences of action would be possible considering the operational flexibility of the station design.

It is concluded that the requirements of criterion 2 of Section 14.2.3.2.1 are satisfied by the station design.

14.6.2 Results for Reactor Shutdown Without Control Rods

This special event is presented to demonstrate the capability of the Standby Liquid Control (SLC) System to shut down the reactor. The SLC System is manually initiated and controlled and is not intended to replace control rods for fast scram of the reactor.

Two cases are postulated to evaluate the capability of the SLC System to shut down the reactor.

1. The reactor is scrammed and it is postulated that some of the control rods malfunction and are not fully inserted.

2. The reactor is operating normally and is postulated that all control rods malfunction and remain fixed at their present position.

The maximum number of control rods are withdrawn when the reactor is at full power with equilibrium xenon poisioning, and this condition establishes the maximum total reactivity control requirement for the SLC System based upon the reactivity control required to achieve the cold shutdown condition from the initial conditions assumed in Case 2 above. The SLC System, as designed, has sufficient capacity to control the reactivity difference between the steady state, full power operating condition of the reactor with voids, and the cold shutdown condition, including shutdown margin, to assure complete shutdown from the most reactive condition at any time in the core life.

The maximum rate of core reactivity increase for Case 1 conditions would result if the reactor was scrammed from full power, held at the hot standby condition for about 1 day until the rate of xenon decay was maximum, and then depressed at the maximum allowable cooldown rate of 100#F/hr. Following scram, the available shutdown margin would actually increase as xenon poisoning in the core increases. The maximum xenon decay rate occurs after xenon poisioning has decreased again to values below those present at equilibrium xenon conditions at full power. The combined reactivity effects of maximum xenon decay rate and maximum reactor cooldown rate, with control rods partially withdrawn from the core, result in a maximum rate of change of core reactivity which is aproximately one fifth of the rate of change of core reactivity using the SLC System.

The maximum rate of core reactivity increase for Case 2 conditions would result if the reactor was scrammed, held at the hot standby conditions until the xenon concentration is maximum, and then returned to the full power condition. This sequence maximizes the rate of xenon depletion (burnup) after return to power and results in the maximum rate of increase of core reactivity from inherent nuclear processes. This sequence results in a maximum rate of core reactivity from inherent nuclear processes. This sequence results in a maximum rate of core reactivity change of core reactivity using the SLC System.

The SLC System provides a minimum boron injection rate which substantially exceeds the maximum rate of reactivity insertion based upon the worst case possible conditions associated with Case 1 or Case 2 above. It is concluded that the design of the SLC System is adequate to satisfy the requirements of this special event.

14.7 REFERENCES

1. "General Electric Standard Analysis for Reactor Fuel", NEDE-24011-P-A, Revision Number Listed in Latest Supplemental Reload Submittal in Appendix Q.

2. Moody, F.J., Maximum Flow Rate of a Single Component Two-Phase Mixture. Journal of Heat Transfer, Trans. ASME, Series C, Volume 87, p. 134.

3. Robbins, C.H., Test of a Full Scale 1/48 Segment of Humboldt Bay Pressure Suppression Containment. GEAP-3596, November 1960.

4. Moody, F.J., Two Phase Vessel Blowdown from Pipes. Journal of Heat Transfer, ASME Volume 88, pp. 285, August 1966.

5. Parsly, L.F., Design Consideration of Reactor Containment Spray System -- Part IV; Calculations of Iodine-Water Partition Coefficients. ORNL-TM-2412, Part IV, January 1970.

6. General Electric Company, "Maximum Discharge Rate of Liquid-Vapor Mixtures from Vessels", Moody, F.J., NEDO-21052, September 1975.

7. BECo SUDDS/RF 96-54, 96-55, FFWTR

8. General Electric (D.C. Serell) letter to Boston Edison (B. Hagemeier), dated January 21, 1997, "ATWS Considerations in FFWTR".

9. General Electric Company, "Maximum Extended Load Line Limit Analyses for Pilgrim Nuclear Power Station Reload 9 Cycle 1-," NEDC-32306P, March 1994 (SUDDS/RF94-042).

10. General Electric, "Safety Evaluation for Eliminating the Boiling Water Reactor Main Steam Line Isolation Valve Closure Function and Scram Function of the Main Steam Line Radiation Monitors". NEDO-31400A.

11. Calculation PNPS-1-ERHS-XIII.Y-63 "Design Basis Control Rod Drop Accident for GE14 10x10 Fuel", Rev. 0, February 20, 2001.

12. NRC letter to PNPS dated July 3, 1997 (PNPS Ltr 1.97.078), Issuance of Amendment 173 to Facility Operating License No. DPR-35, PNPS".

13. NRC letter to PNPS dated September 1, 1998 PNPS Ltr 1.98.105), "Supplement and Revisions to Safety Evaluation for Amendment 173 to PNPS".

14. US NRC Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Revision 1, November 1982.

15. Pilgrim License Amendment 89 for Pilgrim Nuclear Power Station, NRC Letter 1.85.287, dated August 30, 1985.

16. NUREG-0133 “Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants.”

17. NUREG-0473 “Radiological Effluent Technical Specifications for Boiling Water Reactors.”

18. PNPS-1-ERHS-XII.D-8 “Maximum Release Rate at Air Ejector Discharge to Produce 2.5 Rem Whole Body at Exclusion Boundary in the Event of an Air Ejector Line Break” Revision 0, dated 4/17/97.

19. NRC Letter to PNPS dated April 12, 2006, (PNPS Ltr 1.06.042), Amendment 219, SER for Single Recirculation Loop Operation.

20. GNF2 Fuel Design Cycle-Independent Analyses for Entergy Pilgrim Nuclear Power Station, GEH-0000-0085-9069-RO, February 2009.

21. NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants, "L. Soffer et al., February 1995.

22. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accident at nuclear power Reactors," July 2000.

23. Regulatory Guide 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at nuclear Power Plants", June 2003.

24. Calculation 32-5052589-01, "Radiological consequences of a Design Basis Fuel Handling Accident based on Alternative Source Term Methodology (2038 MWth)".

25. GE Report NEDE-31917P, "GE11 Compliance With Amendment 22 of NEDE-24011-P-A (GESTAR II)"

26. GE Report NEDC-32868P, "GE14 Compliance With Amendment 22 of NEDE-24011-P-A (GESTAR II)"

27. GE Report NEDC-33270P, "GNF2 Advantage Compliance With Amendment 22 of NEDE-24011-P-A (GESTAR II)"

APPENDIX A

PRESSURE INTEGRITY OF PIPING AND EQUIPMENT PRESSURE PARTS

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PRESSURE INTEGRITY OF PIPING AND EQUIPMENT PRESSURE PARTS

A.1 SCOPE

This Appendix provides information pertinent to the original plant construction concerning the pressure integrity of piping and equipment parts. It is now used as a reference regarding the original requirements since, as described below, repairs, inspections and replacements are now performed using subsequently issued editions of the ASME Boiler and Pressure Vessel Code Section XI for Safety Class 1, 2 or 3 systems. The applicable ASME Code Safety Class 1, 2, or 3 designations are now selected using the In Service Inspection P&ID drawings.

The original classifications used in this Appendix (A, B, etc.) are no longer applicable for purchase and installation of code safety class material. PNPS piping specifications show the material to be in accordance with either ASTM or ASME Standards, but the examination and test requirements for purchase and installation of Safety Class 1, 2, or 3 pressure boundary piping and equipment parts has been updated and reconciled to the 1989 or later Editions of the ASME Boiler and Pressure Vessel Code Section III. These later requirements supersede the original specifications and are consistent with the ASME Code Section III & XI requirements for materials, examinations, fabrication, and testing. The design remains consistent with the original construction code except where ASME Code Section III has been explicitly adopted for replacements or new systems.

PNPS piping systems were designed and fabricated before the ASME Code Section III had issued Subsections NB, NC, and ND that covered piping and components other than the reactor vessel. The original design codes used for vessels, pumps, and piping were as follows:

The original Construction Code for the Reactor Vessel, the Reactor Recirculation Pumps, and other Primary Pressure Boundary components was:

> ASME Boiler & Pressure Vessel Code, Section III, 1965 Edition through Winter 1966 Addenda.

Where necessary, the rules of ASME Code Section VIII were applied to the materials, design, fabrication, inspection, and testing of vessel and pump pressure boundary components with supplementary welding design and inspection requirements in accordance with ASME Code Section III Paragraph N-2110 for any Category A or B welded joints.

The original Construction Code for PNPS piping was:

> USAS B31.1.0, Power Piping, 1967 Edition as supplemented by the additional requirements in Bechtel Specification M300.

This was the original Construction Code for PNPS piping systems along with supplemental requirements contained in Bechtel Specification M300 and its companion piping fabrication Specifications that included M100, M301, and M305.

The original classifications for the PNPS piping systems were based on Bechtel Specification M300. An excerpt from the original M300 shows that PNPS Class I piping is defined as follows:

> "Class I Piping: This is defined on the P.&I.D.'s only by the symbol shown on the P.&I.D. index. This is to designate the systems whose failure could cause significant release of radioactivity or which are vital to a safe shutdown of the plant and to the removal of decay and sensible heat. These systems require tornado protection and Class I seismic considerations as set forth in the general Specification 6498-G-5. A Class I system may also be a nuclear or critical system, but does not have to be either."

PNPS Class I piping was further divided into classifications as Nuclear, Critical, or Non-Critical.

PNPS Class I "Nuclear" included the Reactor Primary Pressure Boundary piping and corresponds to the later ASME Code Section III Subsection NB Class 1 piping, and also included the Primary Containment Pressure Boundary piping into the Core Standby Cooling Systems (CSCS), and the pump discharge piping from these systems, most of which corresponds to the later ASME Code Section III Subsection NC Class 2 piping.

PNPS Class I "Critical" piping included the other piping of the Core Standby Cooling Systems (CSCS), outside of the Reactor Primary Pressure Boundary, most of which corresponds to the later ASME Code Section III Subsection NC Class 2 piping. This also included the Standby Liquid Control (SLC) System, Main Steam to the Turbine & Bypass Valves, Feedwater Pump Discharge, and Off-Gas System piping.

The original Specifications referenced above implemented the requirements that are described in FSAR Appendix A, including the requirements for the design, materials, fabrication, installation, testing, and inspection for all plant piping systems. Each portions of each piping system was assigned into one of the Classes A, B, C, D, E, J, L, or M (FSAR Appendix A.2). The most rigorous requirements were in Class A with progressively less augmentation of the basic USAS B31.1.0 Power Piping Code for each succeeding Class, with Class M being in strict accordance with USAS B31.1.0 but with no supplemental requirements.

The supplementing of the basic USAS B31.1.0 Power Piping Code requirements was done to apply the most rigorous fabrication, inspection, and quality assurance methods to the Nuclear and Critical piping. These additional requirements were generally consistent with the later ASME Code Section III requirements for ASME Code Class 1, 2, and 3 piping systems.

Three "Material Schedules" M1, M2, & M3 invoked the appropriate "Brittle Fracture Control for Ferritic Steels" as supplementary requirements where needed (FSAR Appendix A.4). Fabrication and installation of piping was performed according to four "Fabrication and Erection Schedules" F1, F2, F3, & F4, (FSAR Appendix A.8) and five "Inspection and Test Schedules" T1, T2, T3, T4, & T5 (FSAR Appendix A.10).

Beginning in 1983, the original PNPS Class I systems were reviewed for the purposes of implementing the ASME Code Section XI In-Service Inspection (ISI) Program, which required that all Safety-Related Systems be categorized as ASME Section III Code Class 1, 2, or 3 piping with respect to the ongoing inspection requirements of an ASME Section XI ISI Program. The boundaries for ASME Code Class 1 Reactor Coolant Pressure Boundary (RCPB) components are defined in 10CFR50.2 and the boundaries for Class 2 and 3 components are established using the guidance in Regulatory Guide 1.26. The Code Classifications for the PNPS ISI P&IDs were prepared using Reg Guide 1.26, NUREG-0800, and NUREG-0803 (for CRD) and included only those systems which are important to safety that contain water, steam, or radioactive materials. There are some differences between the Reg Guide and PNPS classifications and these are described in the PNPS ISI Program Plan PNPS-RPT-05-001.

To distinguish between the original PNPS Class I (Roman numeral "I") and ASME Code Class 1, the terms "PNPS Safety Class 1", "ISI Class 1", or "ISI Safety Class 1" are used at PNPS. Specification M300 was subsequently revised to adopt the Safety Class 1, 2, 3 designations in addition to the Nuclear (N), Critical (C), or Non-Critical (NC) classifications that are still listed. The M300 inspection requirements were revised to include materials, fabrication, and examination requirements that are consistent with ASME Code Section III, Subsection NB, NC, & ND for Class 1, 2, & 3 piping in lieu of the original M300 requirements for Nuclear and Critical piping.

Repairs, inspections, and replacements to PNPS piping designated as PNPS Safety Class 1, 2, or 3 are performed under the controls and requirements of ASME Section XI, as stated, but Section XI is not a design or fabrication code and it directs the user to establish and define the design & fabrication code that is to be applied for any given repair or replacement. It is allowed to use the original or later editions of the original construction code, or to adopt the appropriate subsection of ASME Section III.

FSAR Appendix A identifies the PNPS piping systems, or portions thereof, that have been replaced with piping for which the materials, design, fabrication, and examination are in accordance with ASME Code Section III (i.e., piping for which ASME Section III has been adopted as the Construction Code).

The PNPS seismic design stress criteria for piping and components are given in FSAR Appendix A.3 and Bechtel Specification G505 and later PNPS Specification C114ERQE0.

The PNPS material requirements are given in FSAR Appendix A.4 and the Material Schedules in A.9. Fracture Toughness testing and material requirements were not directly included in USAS B31.1.0 such that FSAR Appendix A.4 and Specification M300 imposed applicable requirements from ASME Code Section III at the time for the "Low Temperature Carbon Steel" Pipe Classes DL, EL, GL, HL, & HM. The Pipe Class Data Sheets included Charpy Impact Test criteria, based on ASME Code Section III that was consistent with the impact testing done for low temperature pipe and weld filler metal testing at the time. Impact Test criteria has been removed from the M300 Pipe Class Data Sheets and Fracture Toughness requirements and exemptions are now imposed directly in accordance with the applicable ASME Code Section III Subsections for PNPS Safety Class 1, 2, and 3 piping, which include the applicable Charpy Impact Test criteria for piping, components, weld materials, and weld process qualifications.

The PNPS original fabrication requirements are given in FSAR Appendix A.5 and the Fabrication and Erection Schedules in A.8. These required fabrication details included such items as welding joint design, welding procedures and processes to be used, and specific welding details for groove and socket weld fabrication were included in Specifications M300 along with M100 and M301 for piping fabrication in the shop and in the field, with weld process information in the Welding Procedure Specifications as required by Specification M305. Fabrication requirements are now imposed by these Specifications in accordance with the Entergy Welding Program and its General Welding Standards and related Procedures for Preheat & Postweld Heat Treatment, Inspections, Examinations, and the compilation of Welding Procedure Specifications.

The PNPS inspection and examination requirements that were supplementary to USAS B31.1.0-1967 are given in FSAR Appendix A.6 and the Inspection and Testing Schedules in A.10. These supplementary requirements for radiography and surface examinations were included in Specifications M300 for materials and components and in M100 and M301 for piping fabrication in the shop and in the field. Inspection and examination requirements are now imposed by these Specifications in accordance with the applicable ASME Code Section III Subsections NB, NC, & ND for PNPS Safety Class 1, 2, and 3 piping. For piping that is PNPS Class I but is not Safety Class 1, 2, or 3, the examination requirements of ASME B31.1 are applicable.

For the purposes of this Appendix, the pressure boundary of the process fluid includes, but is not necessarily limited to:

branch outlet nozzles or nipples, instrument wells, reservoirs, pump casing closures, blind flanges and similar pressure closures, studs, nuts, and fasteners in flanged joints between pressure parts and bodies, and pressure parts of inline components such as traps and strainers.

Specifically excluded from the scope of this Appendix are nonpressure parts such as pump motors, shafts, seals, impellers, wear rings, valve stems, gland followers, seat rings, guides, yokes, and operators; any nonmetallic material such as packing and gaskets; fasteners not in pressure part joints such as yoke studs and gland follower studs; washers of any kind.

A.1.1 Codes and Specifications

The piping and equipment pressure parts in this station are designed, fabricated, inspected, and tested in accordance with recognized industrial codes and specifications as far as these codes and specifications can be applied. In some cases these codes and specifications are not stringent enough for nuclear systems and supplementary requirements are applied to increase safety and operational reliability. The application of the industrial codes and specifications is defined in this Appendix, as well as the application of the supplementary requirements. Where conflicts occur between the industrial codes and specifications and the supplementary requirements, the supplementary requirements take precedence.

Repairs, alterations, and inservice inspection are made to the appropriate ASME Boiler and Pressure Vessel Code, Section XI.

The codes and specifications used in the design, fabrication, inspection, and testing of the liquid radwaste system, solid radwaste system, fuel cool cooling and cleanup system, reactor building closed cooling water system, turbine building closed cooling water system, salt service water system, circulating water system, condensate demineralizer system, condensate and demineralized water storage and transfer system, and the diesel fuel oil storage and transfer system are:

a. ASME Boiler and Pressure Vessel Code, Sections I, II, III, VIII, IX, XI, and API-650 Code.

b. USAS B31.1.0 or later editions

c. ASTM Standards

d. Pipe Fabrication Institute Standards

The principal tanks and heat exchangers that are necessary for the proper functioning of the systems listed above are designed, fabricated, inspected, and tested in accordance with the codes and specifications listed on Table A.1-1.

Class I Components are designed such that the stresses in the structural portions shall not exceed the working stress levels allowed by AISC Manual of Steel Construction or other equivalent industrial codes for the Operating Basis Earthquake, and shall not exceed 150 percent of code allowable, provided that primary stresses are less than the yield stress for the Safe Shutdown Earthquake. No supplementary nondestructive testing requirements beyond those required by applicable codes have been specified for Class I tanks and heat exchangers.

The schedules of Appendix A which prescribe the allowable materials apply primarily to the systems which include reactor coolant pressure boundary, extension of containment, and engineered Safeguard Systems. Many of the listed balance of plant systems have unique process conditions requiring a much broader spectrum of allowable materials than those for which Appendix A provides. Essentially all materials needed in these applications are approved for use by USAS B31.1.0 (or later editions). The use of Alloy 20 materials in dilute acid systems such as liquid radwaste and condensate demineralizer systems is not discussed in the code. Replacement pipe materials for Recirculation, Residual Heat Removal (inside the Containment), Core Spray (inside the Containment), Reactor Water Cleanup, and new piping materials for High Pressure Coolant Injection Turbine exhaust vacuum breaker line are approved for use by the ASME Boiler and Pressure Vessel Code Section III Division 1 Appendix I, 1980 Edition through Winter 1980 Addenda. Replaced core spray pipe outside the containment is approved for use by the ASME B&PV Code Section XI, 1980 Edition through Winter 1980 Addenda.

All Torus Mark I Containment analysis and modifications per NUREG-0661 have been completed at PNPS. The criteria used to evaluate the torus structure is the ASME Boiler & Pressure Vessel Code, Section III, Division I, with addenda through Summer 1977 and Code Case N-197. The analysis of Safety Relief Valve piping and supports, Torus Attached Piping (TAP) and branch lines, and TAP and branch line supports was done in accordance with Section III of the ASME B & PV Code, 1977 Edition, including Summer 1977 addenda. Modifications were done under Section XI of the ASME B & PV Code, 1977 Edition through Winter 1980 addenda. Details are provided in Teledyne Report TR-5310-1, Rev. 2: "Mark I Containment Program, Plant Unique Analysis Report of the Suppression Chamber for Pilgrim Station - Unit I", and TR-5310-2, Rev. 1: "Mark I Containment Program, Plant Unique Analysis Report of the Torus Attached Piping for Pilgrim Nuclear Power Station".

*All recirculation pipe was replaced; RHR inside the containment and connected to the recirculation system was replaced; RWCU inside the containment and connected to the RHR supply pipe was replaced and; Core Spray pipe inside containment between the gate valve and containment penetration was replaced. A section of Loop B RHR pipe outside the containment between the drywell penetration and the first valve was also replaced. A section of Loop B core spray pipe outside the containment upstream of valve MO-1400-25B was also replaced.

TABLE A.1-1

TANKS AND HEAT EXCHANGERS - CODES AND SPECIFICATIONS

System	Component	Code	Material Specification
Liquid Radwaste Systems Figures 9.2-3, 9.2-4	T-301 A & B Clean Waste Receiver Tank	API-650	ASTM A-283 Gr. C
	T-304 A & B Treated Water Holdup Tank	API-650	ASTM A-283 Gr. C
	T-312 A & B Chemical Waste Receiver Tank	API-650	ASTM A-283 Gr. C
	T-311 A, B, C Monitor Tank	API-650	ASTM A-283 Gr. C
Solids Recovery System Figure 9.3-1	T-308 Spent Resin Storage Tank	ASME Code Section III Class C	SA-240 Type 304 SS
	T-306 Misc. Waste Tank	API-650	ASTM A-283 Gr. C
	T-307 A & B Sludge Storage Tanks	API-650	SA-240 Type 304 SS
	T-318 Floc Recycle Tank	API-650	SA-240 Type 304 SS
Reactor Building Cooling Water System Figure 10.5-1	HX-E-209 A & B RBCCW Heat Exchanger	ASME Code Section VIII TEMA Class C	Shell: SA-515-70 Tube/Tubesheet: SB-111/SB-171 Channels: SB-402 Nozzles: SA-181-11, SB-402 SA-10
	HX-E-206 A & B Fuel Pool Heat Exchanger	ASME Code Section VIII TEMA Class C	Shell: SA-106B Tube/Tubesheet: SA-249-T304/SA-249-T304L
	T-211 A & B Head Tank	API-650	ASTM A-283 Gr. B
	T-201 A & B Head Tank	API-650	ASTM A-283 Gr. B

TABLE A.1-1

TANKS AND HEAT EXCHANGERS - CODES AND SPECIFICATIONS

System	Component	Code	Material Specification
Turbine Building Cooling Water System Figure 10.6-1	HX-E 122 A & B TBCCW Heat Exchanger	ASME Code Section VII TEMA Class C	Shell: SA-285C Tube/Tubesheet: SB-111/SB-171 Channel: SB-402 Heads: SA-515-70 Nozzles: SA-181-11
	T-104 Head Tank	API-650	A-283 Gr. B
Condensate and Demineralized Water Storage and Transfer System Figure 11.9-1	T-105 A & B Condensate Storage Tank	API-650	A-131C: A-36
	T-108 Demineralized Water Storage	API-650	A-36
Diesel Fuel Oil Storage and Transfer System figure 8.5-1	T-126 A & B Oil Storage Tank	API-650	A-131-C
	T-124 A & B Day Tank	Commercial Standards	A-285-C
SEP Diesel Fuel Oil Storage	T160 A & B Primary and Secondary Oil Storage Tanks	ASTM D4021-81; UL MH 9061 & MH 9061 9; MFPA Sections 30 & 31; Factory Mutual Systems Approval IM7A0AF	FRP

A.2 CLASSIFICATION OF PIPING AND EQUIPMENT PRESSURE PARTS

For the identification and the association of requirements, piping and equipment pressure parts are classified as follows:

Class A	Piping and equipment pressure parts which cannot be isolated from the reactor vessel
Class B	Piping and equipment pressure parts which can be isolated from the reactor vessel by a single isolation valve
Class C	Piping and equipment pressure parts, other than included in Classes A and B for which design considerations require fabrication to Schedule F1 and inspection to Schedule T2
Class D	Piping and equipment pressure parts which serve as an extension of containment and which operate at either pressures greater than 150 psig or temperatures greater than 212°F
Class E	Piping and equipment pressure parts which serve as an extension of containment and which operate at pressures equal to or less then 150 psig and temperatures equal to or less than 212°F
Class J	Piping and equipment pressure parts for which design considerations require fabrication to Schedule F3 and inspection to Schedule T5
Class L	Piping and equipment pressure parts which require materials considerations to maintain deionized water purity
Class M	Power piping and equipment pressure parts not otherwise classified and which are considered within the scope of USAS B31.1.0 (or later editions), Power Piping.

A diagram of piping classification is shown on Figure A.2-1 which includes reactor coolant pressure boundary, extension of containment, and engineered safeguard systems. Table A.2-1 is provided for the classification of additional systems, some of which require a broader spectrum of allowable materials than those which the material schedule provides.

TABLE A.2-1

PIPING & EQUIPMENT PRESSURE PART CLASSIFICATION FOR PIPING IN VARIOUS SYSTEMS
(SUPPLEMENT TO FIGURE A.2-1)

SYSTEM	FSAR FIGURE	CLASSIFICATION (See Note 1)
Diesel Oil Storage and Transfer	8.5-1	M and Manufacturer's standards
Solid Radwaste	9.2-5	J, M and Manufacturer's standards
Radwaste Collection	9.2-2	M
Clean Radwaste	9.2-3	J, M and Manufacturer's standards
Chemical & Non-reclaimable Radwaste	9.2-4	J, M and Manufacturer's standards
Fuel Pool Cooling & Demineralizer	10.4-1	J and M
Reactor Building Closed-Loop Cooling Water	10.5-1	J, M and Manufacturer's standards
Turbine Building Closed-Loop Cooling Water	10.6-1	J and M
Service Water	10.7-1	D, M and Manufacturer's standards
Circulating Water (see note 2)	11.6-1	M and Manufacturer's standards
Condensate Demineralizer	11.7-1	M and Manufacturer's standards
Condensate Demineralizer Resin Regeneration	10.18-1 11.7-2	J, M and Manufacturer's standards
Condensate and Demineralized Water Storage and Transfer	11.9-1	L, J and M

Notes:

(1) "Manufacturer's standards" refers to piping standards specified by an equipment or assembly manufacturer or provider which govern piping responsible for maintaining a portion of the system pressure boundary. For letter designations, refer to Section A.2 for details of classification.

(2) AWWA concrete material standards apply to concrete portions of the Circulating Water System in lieu of material schedules M1 through M3.

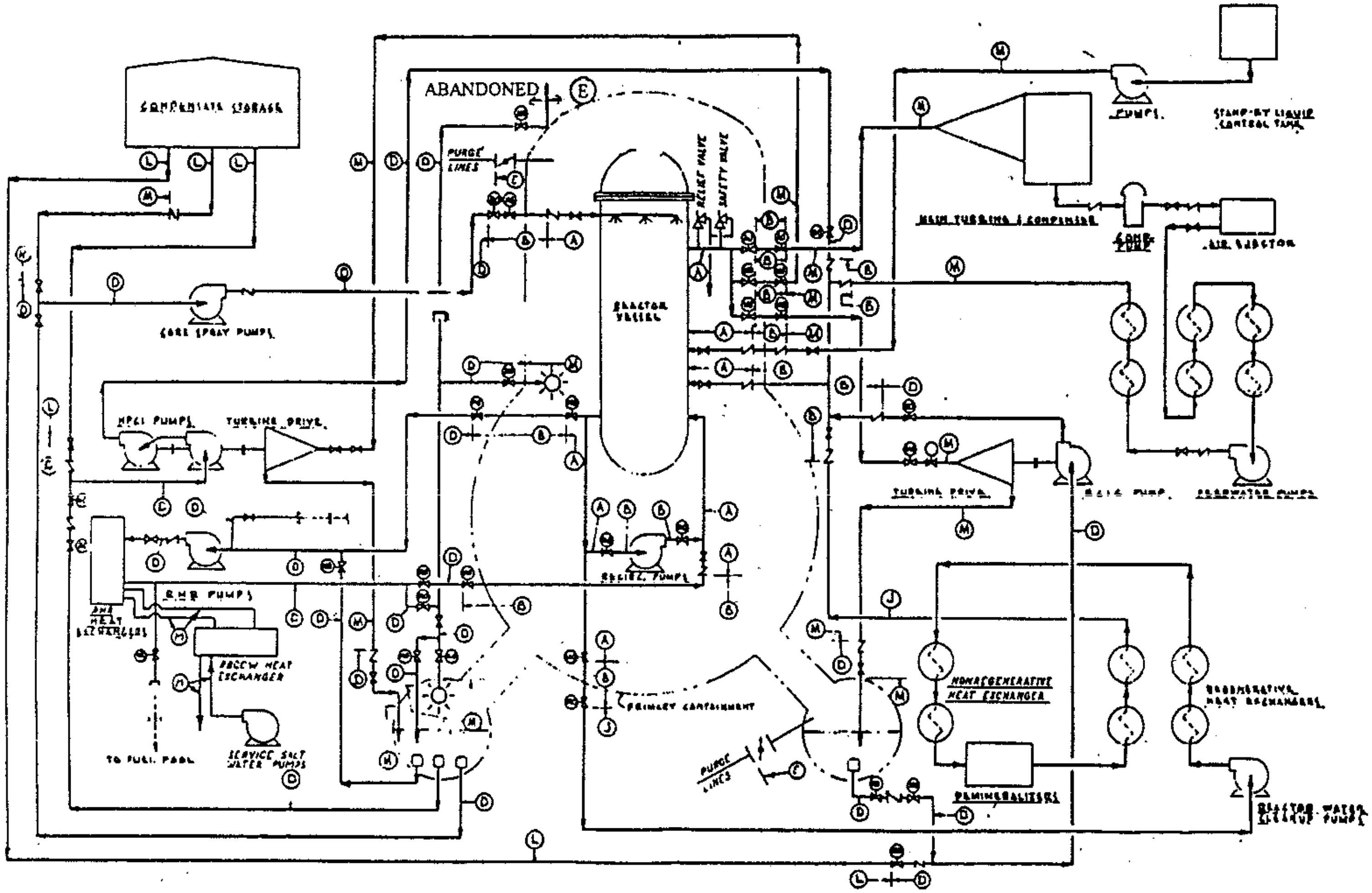

FIGURE A.2-1

TITLE : PIPING CLASSIFICATION DIAGRAM
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

REVISION 21 OCTOBER 1997

A.3 DESIGN REQUIREMENTS

A.3.1 Piping Design

All piping is designed in accordance with USAS B31.1.0, Power Piping, later editions of ANSI/ASME B.31.1, or appropriate parts of ASME Section III. In addition, Class I seismic piping is designed to meet the supplementary requirements included in Section A.3.1.1 or appropriate parts of ASME Section III.

A.3.1.1 Analysis

A.3.1.1.1 Primary Stresses (S_P)

Primary stresses are as follows:

1. Circumferential Primary Stress (S_R)

   Circumferential primary stresses are below the allowable stress (S_h) at the design pressure and temperature

2. Longitudinal Primary Stresses (S_L)

   The following loads are considered as producing longitudinal primary stresses: Internal or external pressures; weight loads including valves, insulation, fluids, and equipment; hanger loads; static external loads and reactions; and the inertia load portion of seismic loads

   When the seismic load is due to the Operating Basis Earthquake (0.08g) the vectorial combination of all longitudinal primary stresses (S_L) does not exceed 1.2 times the allowable stress (S_h)

   When the seismic load is due to the Safe Shutdown Earthquake (0.15g), the vectorial combination of all longitudinal primary stresses does not exceed the yield strength of the material at temperature unless higher allowable limits are calculated and substantiated by the methods outlined in Appendix C

3. For recirculation piping and replaced portions of RWCU and RHR pipe, primary stresses are computed in accordance with ASME Section III, Subsection NB

A.3.1.1.2 Secondary Stresses (S_E)

Secondary stresses are determined by use of the maximum shearing stress theory

$$T\ Max = 1/2\sqrt{S_b^2 + 4S_t^2} = 1/2\ S_E$$

therefore,

$$S_E = \sqrt{S_b^2 + 4S_t^2}$$

(See USAS ANSI B31.1.0)

The following loads are considered in determining longitudinal secondary stresses: (a) thermal expansion of piping, (b) movement of attachments due to thermal expansion, (c) forces applied by other piping systems as a result of their expansion, (d) any variations in pipe hanger loads resulting from expansion of the system, and (e) anchor point movement portion of seismic load.

The vectorial combination of longitudinal secondary stresses (S_E) does not exceed the allowable stress range (S_A), i.e., $S_E \leq S_A$.

Where:

$$S_A = f\ [1.25\ (S_c + S_h) - S_L]$$

(This is Equation 1 from Paragraph 102.3.2 of USAS B31.1.0 modified to include the additional stress allowance permitted when S_L is less than S_h). Secondary stresses are computed where required according to the rules of ASME Section III.

A.3.1.1.3 Allowable Stress Values (S_c and S_L)

The allowable stress values for design are as follows:

1. For austenitic stainless steel, the allowable stress values of USAS (ANSI) B31.1 are used. For materials not covered by USAS (ANSI) B31.1, the higher stress values of the ASME Boiler and Pressure Code, Section I, Appendix A-24 are used

2. For carbon steel, the allowable stress values of USAS (ANSI) B31.1 are used. For types of materials not covered by USAS (ANSI) B31.1, the higher allowable stress values of the ASME Boiler and Pressure Vessel Code, Section I, Appendix A-24 are used

Allowable stresses are used where required according to the rules of ASME Section III.

A.3.1.2 Corrosion and Erosion

Adequate allowances for corrosion and erosion are made according to individual systems requirements for a design life of 40 yr.

A.3.1.3 Wall Thickness

Pipe wall thickness, fitting, and flange ratings are in accordance with USAS (ANSI) B31.1 Power Piping, and the additional requirements in this Appendix and in accordance with ASME Section III.

For recirculation piping and portions of RHR, RWCU and Core Spray pipe; pipe wall thickness, fitting and flange ratings are in accordance with ASME Section III B&PV Code Class 1 requirements, 1980 Edition; Winter 1980 Addenda. The HPCI Turbine exhaust vacuum breaker line piping is in accordance with ASME Section III B&PV Code Class 2 requirements, 1980 Edition; Winter 1980 addenda.

A.3.1.4 Reactor Vessel Nozzle Loads

Piping connections to the reactor pressure vessel nozzles are designed so that forces and moments at those piping connections do not exceed the allowable limits for the nozzle under consideration.

A.3.2 Valve Design

Valves are designed and rated by their manufacturers to meet the highest service pressure and temperature in the piping to which attached. In accordance with USAS B31.1.0, Power Piping (or later editions) valves are designed to USAS B16.5, Steel Pipe Flanges and Flanged Fittings, or Manufacturers Standardization Society, Standard Practice MSS-SP-66, Pressure-Temperature Rating for Steel Butt Welded End Valves, or proven design practice. Valves for the HPCI turbine exhaust condensate pot drain and vacuum breaker line are designed in accordance with ASME Section III Class 2. RHR system containment isolation valves MO-1001-29B and 36A&B are manufactured in accordance with ASME Section III.

A.3.3 Pump Design

The pressure retaining parts of pumps are designed to meet the highest service pressure and temperature in the piping to which they are attached.

1. For pumps used in piping systems classified as A, B, or D, the requirements of Section III of the ASME Boiler and Pressure Vessel Code for Class C vessels are used as a guide in calculating the thickness of pressure retaining parts, and in sizing the cover bolting. When the pump is of such a configuration that code calculations are not applicable or stress levels do not satisfy code criteria and the specific pump is a standard commercial pump, standard design calculations are accepted when supplemented by a documented history of operational reliability.

2. For pumps used in piping systems classified as J, L, or M and operated at pressures above 150 psi or temperatures above 212°F, the requirements of Section VIII of the ASME Boiler and Pressure Vessel Code are used as a guide in calculating the thickness of pressure retaining parts and in sizing the cover bolting. When the pump is of such a configuration that code calculations are not applicable or stress levels do not satisfy code criteria and the specific pump is a standard commercial pump, standard design

calculations are accepted when supplemented by a documented history of operational reliability.

3. Pumps used in Classes J, L, M, or piping systems which operate at pressures below 150 psi and temperatures below 212°F, and are manufacturers' standard pumps for the required service, require no submittal of calculations from the manufacturers.

A.4 MATERIALS

A.4.1 Material Application

The material for piping and equipment pressure parts listed in schedules M1, M2, and M3 are included herein.

These schedules are applied as follows:

Piping and Equipment Pressure Parts Classification	Material Schedules
A, B, C, E, and J	M1 and M2
L and M	M3

A.4.2 Brittle Fracture Control for Ferritic Steels

Classification A and B

The fracture or notch toughness properties and the operating temperature of ferritic materials of the reactor coolant pressure boundary are controlled to ensure adequate toughness when the system is pressurized to more than 20 percent of the design pressure. Such assurance is provided by maintaining a material service temperature at least 60°F above the nil ductility transition (NDT) temperature. Further interpretations and requirements are:

1. Charpy V notch per ASTM A370 Type A or drop weight tests per ASTM E-208 are performed to demonstrate that all materials and weld metal will meet brittle fracture requirements at test temperature. Test specimens shall be prepared and tested with minimum impact energy requirements with Table N-421 and the general provisions of N-313, N-331, N-332, and N-511 of Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. The welding procedures used must be qualified by impact testing of weld metal and heat affected zone to the same requirements as the base metal in accordance with N-541. Consumable insert material does not have to meet the impact test requirements of N-511

2. Piping and equipment pressure parts having a nominal wall thickness of 1/2 in or less are not impact tested provided the material is normalized or has been fabricated to a fine grain melting practice

3. Impact testing will not be required on components or piping within the boundary having a minimum service temperature of 250°F or more

   Example: Steam line is excluded from brittle fracture test requirements since the steam temperature will be over 250°F when the steam line pressure is at the 20 percent design limit

4. Impact testing will not be required on components or piping within the pressure boundary whose rupture could not result in a loss of coolant exceeding the capability of normal makeup systems to maintain adequate core cooling for the duration of a reactor shutdown and orderly cooldown

5. Field welds and shop welds shall have welding procedures qualified by impact testing in accordance with Paragraph 1

6. These criteria apply to nuclear piping and equipment pressure parts of the reactor coolant pressure boundary; and do not apply to related components such as anchors, anchor bolts, hangers, suppressors, and restraints

Classification D and E

The possibility of brittle fracture is considered for ferritic steel piping and equipment pressure parts in Classes D and E if the metal is subjected to temperatures below 40°F during operation or testing. In cases where ferritic metal may be subjected to temperatures below 40°F, brittle fracture control is provided by material evaluation, design, fabrication, or test requirements selected to assure adequate fracture toughness in the piping or component.

A.5 FABRICATION AND INSTALLATION REQUIREMENTS

Fabrication and erection of piping and equipment pressure parts are in accordance with USAS B31.1.0, Power Piping, and the supplementary requirements in schedules F1, F2, F3, and F4 are included herein. These schedules are applied as follows:

Piping and Equipment Pressure Parts Classification	Fabrication and Erection Schedules
A, B, and C	F1
D and E	F2
J and L	F3
M	F4

Fabrication and erection of Recirculation, RHR, RWCU and Core Spray replacement piping and HPCI turbine exhaust vacuum breaker piping, are in accordance with ASME Boiler and Pressure Vessel Code Section XI, 1980 Edition through Winter 1980 Addenda.

A.6 TESTING AND INSPECTION REQUIREMENTS

Testing and inspection of piping and equipment pressure parts are in accordance with USAS B31.1.0, Power Piping (or later editions), and the supplementary requirements Schedules T1, T2, T3, T4, and T5 included herein. These schedules are applied as follows:

Piping and Equipment Pressure Parts Classification	Inspection and Test Schedule
A, B, and C	T1
D	T2
E	T3
L	T4
M and J	T5

Testing and inspection of recirculation, RHR, RWCU and core spray replacement piping is in accordance with ASME B&PV Code Section III, Subsection NB, 1980 Edition through Winter 1980 Addenda.

Testing and inspection of the HPCI turbine exhaust vacuum breaker piping is in accordance with ASME B&PV Code Section II, Subsection NC, 1980 Edition through Winter 1980 Addenda.

Testing and inspection of the salt service water piping replacement project is in accordance with the ASME B31.1 Power Piping Code 1989 through Addenda b.

A.6.1 Methods, Techniques, and Acceptance Standards

A.6.1.1 Radiography

A.6.1.1.1 Welds

The radiography of welds, including acceptability standards, are in accordance with ASME Boiler and Pressure Vessel Code, Section III, Paragraph N-624 (ASME III Subsection NB for replacement recirculation, RHR, RWCU and core spray pipe. ASME III, Subsection NC, for HPCI turbine exhaust vacuum breaker pipe).

A.6.1.1.2 Castings

Methods and Techniques

The radiography of castings employ methods and techniques in accordance with ASTM E94, Tentative Recommended Practices for Radiographic Testing, to the quality level in accordance with ASTM E142, Standard Method for Controlling Quality of Radiographic Testing.

Acceptance Standards

Discontinuities are judged by comparison with ASTM E71, E186, and E280 as appropriate for section thickness. Discontinuity types A through C of severity level 2 are acceptable; discontinuity types beyond C are not acceptable.

A.6.1.2 Ultrasonic Testing

Ultrasonic examination of forgings is done in accordance with ASTM A388, Ultrasonic Testing and Inspection of Heavy Steel Forgings, and meet the following acceptance standards:

A.6.1.2.1 Normal Beam Testing-Acceptance Standards

The materials are considered unacceptable, unless repaired, based on the following test indications:

1. Indications of discontinuities in the material that produce a complete loss of back reflection not associated with the geometric configuration of the piece. A complete loss of back reflection is assumed when the back reflection falls below 5 percent of full screen height

2. Traveling indications of discontinuities with 10 percent or more of the back reflection lost. A traveling indication is defined as an indication which displays sweep movement of the oscilloscope pattern at a relatively constant amplitude as the search unit is moved along the part being examined

A.6.1.2.2 Angle Beam Testing-Acceptance Standards

Materials are unacceptable where oscilloscope indications exceed those produced by the reference standard. The reference standard notch is the smaller of a depth equal to 5 percent of the material thickness or 3/8 in.

A.6.1.3 Liquid Penetrant Testing

Methods, techniques, and acceptance standards for liquid penetrant testing are in accordance with paragraph N-627, Section III, of the ASME Boiler and Pressure Vessel Code (ASME III subsection NB for replacement pipe).

A.6.1.4 Magnetic Particle Testing

Methods, techniques, and acceptance standards for magnetic particle testing are in accordance with paragraph N-626, Section III of the ASME Boiler and Pressure Vessel Code (ASME III subsection NB for replacement pipe).

A.6.1.5 Hydrostatic Testing

Hydrostatic tests of piping and equipment pressure parts were conducted in accordance with B31.1.0, Power Piping, and for current replacement or repairs, hydrostatic testing is performed in accordance with ASME, Section XI.

A.7 FINAL CLEANING AND PROTECTION

To minimize the requirements for cleaning after erection and to prevent damage during shipment, storage, or handling, piping and equipment pressure parts are cleaned, capped, and prepared for shipment.

A.7.1 Austenitic Stainless Steel

Austenitic stainless steel surfaces are in the mechanically cleaned, blast cleaned, or pickled condition and are free of scale and organic contaminants such as grease and oil. Blast cleaned surfaces are free of residual quantities of the cleaning media such as aluminum oxide and silica.

A.7.2 Carbon/Low Alloy Steel

Carbon/low alloy steel surfaces are in the mechanically cleaned, blast cleaned, or pickled condition and are free of scale and organic contaminants such as grease and oil. Blast cleaned surfaces are free of residual quantities of the cleaning media such as aluminum oxide and silica.

After cleaning, all surfaces are suitably protected to prevent excessive corrosion during shipment or storage.

A.8 F1, F2, F3, AND F4 FABRICATION AND ERECTION SCHEDULE

A.8.1 F1 & F2 Fabrication and Erection Schedule

A.8.1.1 Welding

Welding of piping and equipment pressure parts is accomplished according to the following requirements

A.8.1.1.1 Qualification

All welding, including fillet, seal, repair, and attachment welds, is performed in accordance with written welding procedures. Procedure qualification and welder performance qualification are in accordance with USAS B31.1.0, Power Piping (or later editions) (ASME B&PV Code Section XI, 1980 Edition through Winter 1980 Addenda for recirculation, RHR, RWCU and Core Spray replacement piping and HPCI turbine exhaust vacuum breaker piping).

A.8.1.1.2 Qualification Records

Qualification records are in accordance with USAS B31.1.0, Power Piping (or later editions). Application of welder's identification symbols are within the limitations on marking as described in Section A.8.1.7 (ASME B&PV Code Section XI, 1980 Edition through Winter 1980 Addenda for recirculation, RHR, RWCU and Core Spray replacement piping and HPCI turbine exhaust vacuum breaker piping).

A.8.1.1.3 Butt Joints

Joint design and welding procedures for longitudinal and girth butt joints larger than 2 inches in nominal pipe size are complete penetration groove welds.

A.8.1.1.4 Branch Connections (Schedule F1 only)

Branches larger than 4 in nominal pipe size employ welding tees, sweepolets, or extruded outlets of equivalent reinforcement attached with complete penetration groove welds.

Branches 4 in and smaller in nominal pipe size employ weldolets, sockolets, or similar fittings attached with complete penetration groove welds.

Instrument and instrument line branches 2 in and smaller in nominal pipe size employ couplings or half couplings attached with complete penetration welds.

A.8.1.1.5 Branch Connections (Schedule F2 only)

Branch connections are in accordance with the requirements of USAS B31.1.0, Power Piping (or later editions). (ASME B&PV code Section II for HPCI turbine exhaust vacuum breaker piping).

A.8.1.1.6 Socket Welds

Socket welds are employed for nominal pipe size 2 in and smaller, and are in accordance with USAS B31.1.0, Power Piping or ASME B&PV Code Section XI for recirculation, RHR, RWCU and Core Spray replacement pipe and HPCI turbine exhaust vacuum breaker piping.

Socket welds have a minimum of two weld layers for wall thickness of 3/16 in and larger.

A.8.1.1.7 Attachment Welds

Attachment of nonpressure containing parts such as supports and hangers to pressure containing components is by complete penetration groove welds, and is subject to all the requirements and limitations imposed for fabrication of the piping or equipment to which they are attached.

After temporary attachment welds have been removed, the surface is ground smooth and examined by the liquid penetrant or magnetic particle method.

A.8.1.1.8 Weld Reinforcement

Weld reinforcement is in accordance with the requirements of USAS B31.1.0, Power Piping or ASME B&PV Code Section XI for recirculation, RHR, RWCU and Core Spray replacement pipe and HPCI turbine exhaust vacuum breaker piping.

A.8.1.1.9 Welding Procedures and Processes

Welding procedures and processes are employed to produce welds of complete penetration, of complete fusion, and free of unacceptable defects.

Weld layers are built up uniformly around the circumference and across the width of the joint. Weld starts and stops are staggered. Block welding or peening of welds is allowed only after special consideration and approval. Pressure containing and attachment welds may be made by any of the following processes within the limitations described in this Appendix: (The use of other processes or procedures is not allowed without special qualification and approval.)

1. Gas tungsten arc welding (GTAW) (Manual and automatic for recirculation, RHR, RWCU and Core Spray replacement pipe, and HPCI turbine exhaust vacuum breaker piping)

2. Shielded metal arc welding (SMAW)

3. Submerged arc welding (SAW)

4. Gas metal arc welding (GMAW)(1)

(1) GMAW utilizing the short circuiting or globular mode of transfer shall not be used for austenitic stainless welds

A.8.1.1.9.1 Austenitic Stainless Steel Welds

Groove Butt Welds

Austenitic stainless steel groove butt welds are subject to the following limitations:

1. Double welding (welded from both sides) by any of the preceding acceptable processes may be employed provided the first side is back ground to sound metal and visually inspected prior to welding the second side

2. Single welding (welded from one side) may be used, employing the GTAW process with filler metal added or using consumable inserts for the first pass and a protective gas back purge held until a minimum of 3/16 in. of weld thickness is completed. Completion of the weld may be by other acceptable welding processes

Socket Welds

Austenitic stainless steel socket welds are made by the GTAW process with filler metal for at least the root layer. Protective gas back purging is not required.

A.8.1.1.9.2 Carbon Steel Welds

Carbon steel groove welds are subject to the following limitations:

1. Double welding (welded from both sides) by any acceptable processes may be employed Double welded joints are back ground to sound metal and visually inspected prior to welding the second side

2. Single welding (welded from one side) may be used employing the GTAW process with filler metal added or using consumable inserts for the first pass and the GTAW process with filler metal added for the second layer. Completion of the weld may be by other acceptable processes

A.8.1.1.9.3 Dissimilar Metal Welds

Welded connections between austenitic stainless steel and carbon/low alloy steel are considered to be dissimilar metal welds.

Groove Welds

Dissimilar metal groove welds are in accordance with the following requirements:

1. When the carbon/low alloy steel component is over 3/4 in thick, it is "buttered" with Type-309 or 309L filler metal and heat treated in accordance with the requirements of USAS B31.1.0 or ASME B&PV Code Section XI. The minimum thickness of the "buttered" area after end preparation is 3/16 in. Completion of the weld joint is then accomplished with Type-308 or Type-309 filler metal for original fabrication and erection, or Type 308L filler metal for replacement pipe. For replacement pipe, a minimum preheat of 200°F is required for the buttering operation for material with a thickness equal to or greater than 3/4 in. For replacement pipe material less than 3/4 in thick, a minimum preheat of 60°F is required. The completed weld joint is not heat treated

2. For original fabrication when the carbon/low alloy steel component is 3/4 in or less in thickness, it shall be welded utilizing the SMAW, GTAW, or SAW process with Type-309 filler metal. For replacement pipe, carbon/low alloy steel components 3/4 in or less in thickness should be welded in accordance with A.8.1.1.9.3.1.

Socket Welds

Dissimilar metal socket welds employ the GTAW process with filler metal added for at least the root layer. Type-309 filler metal is used for the original fabrication and erection. For replacement pipe, type 309L filler metal is used. Protective gas back purging is not required.

A.8.1.2 Bending and Forming

Bending and forming are in accordance with USAS B31.1.0, Power Piping, and the supplementing requirements as shown. Austenitic stainless steel piping may be hot or cold bent within the following limitations:

1. A pipe may be cold bent within the limitations of USAS B31.1.0 provided the maximum operating temperature does not exceed 200°F for more than 1 percent of the design life and the base material is in the solution annealed condition prior to bending

2. Pipe 2 in and smaller may be cold bent to a minimum radius equal to five nominal pipe diameters regardless of the service temperature. Pipe 2 1/2 in and larger may be cold bent to less than 20 nominal pipe diameters where the normal operating temperature is above 200°F provided the bending operation is followed by solution heat treatment

3. Hot bending is permitted to a minimum radius equal to five nominal pipe diameters regardless of service temperature. Hot bending is followed by solution heat treatment

4. Bending of austenitic stainless steel piping at temperatures below 800°F is considered cold bending

For piping systems and portions of piping systems replaced during 1984, bending, forming, and mechanical sizing is in accordance with the ASME Code, Section III, Class 1 and the additional requirements specified below:

a. Cold bending of piping to any radius within the limitations of the ASME Code, Section III, Subarticle NB-3642 is acceptable, provided the hardness of the bent area does not exceed R_B92 after bending, and the strain induced by bending does not exceed two and one-half percent. If the hardness exceeds R_B92 or the strain exceeds two and one-half percent, then the production pipe is solution heat treated. Bending at temperatures below 800°F is considered cold bending. Other cold worked operations require special approval unless the cold work area is completely removed or solution heat treated.

b. Hot bending of piping to any radius within the limitations of the ASME Code, Section III, Subarticle NB-3642 is acceptable provided the operation is followed by solution heat treating excepting pipe bent by the induction-heated process in accordance with Paragraph (C) below.

c. Pipe formed by the induction-heated process is bent according to a qualified procedure. Pipe is continuously water spray quenched to below 800°F immediately after it exits the induction heated zone such that it is capable of meeting the requirements of tests that show it is not sensitized. Solution heat treatment following bending is not required provided one specimen per heat and heat treat lot is tested for sensitization. A heat treat lot is defined as all pipe bent to the same parameters on the same working shift.

d. Any sizing or swaging operation, except for machining and grinding, which changes the circumference or wall thickness of the piping by more than two percent is followed by solution heat treating.

e. Forged and bored pipe is solution heat treated after the forging and boring operation.

f. Allowable out-of-roundness of bends as determined by the difference between the major and minor diameters is not greater than eight percent of the nominal diameter.

g. All accessible surface areas of piping subjected to bending or to a mechanical sizing operation other than machining or grinding, which changes circumference or diameter dimensions by cver two percent, are examined by the liquid penetrant method. If solution heat treatment is required, the liquid penetrant examination is performed after solution heat treatment and cleaning.

h. All material or subassemblies subjected to hot forming are grit blast cleaned after hot forming and before solution heat treatment. Mechanically polished surfaces are not subjected to sand or grit blasting without special approval. When approval to sandblast is granted, the affected surfaces are subsequently polished to 32 RMS or better.

A.8.1.3 Heat Treatment

A.8.1.3.1 Austenitic Stainless Steel

Austenitic stainless steel piping components and equipment pressure parts are solution heat treated at least once.

Austenitic stainless steel material is degreased and stripped in accordance with ASTM A380, Descaling and Cleaning Stainless Steel Surfaces, paragraph 2, prior to heat treatment.

Material is heat treated by heating to a temperature between 1,900° and 2,050°F and then held for hr/in of thickness but not less than 1/2 hr followed by rapid cooling to below 800°F.

For piping systems and portions of piping systems replaced during 1984, materials are solution heat treated in accordance with the following requirements:

a. Prior to heat treatment, materials are precleaned in accordance with Paragraph 4, ASTM A380, Cleaning and Descaling Stainless Steel Parts, Equipment and Systems.

b. When batch furnaces are used, Type 316 Nuclear Grade fusion welded pipe is heated to temperatures between 1900° and 2000°F for 15 minutes minimum and 30 minutes maximum and then water quenched to below 800°F in 3 minutes. For heat treating utilizing continuous furnaces, the temperature range is between 1900° and 2050°F with no time at temperature restriction.

c. When batch furnaces are used, G.E. designated Type 316K fittings or forged and rolled parts are heated to temperatures between 1900° and 2100°F for 15 minutes per in of thickness but for no less than 15 minutes regardless of thickness. Subsequently, these materials are immediately water quenched to below 400°F. For heat treating utilizing continuous furnaces, the temperature range is between 1900° and 2150°F with no time at temperature restriction.

d. For materials that require a solution heat treatment after bending or forming (reference A.8.1.2), the requirements of (a) and (c), above are imposed except that for quenching, the metal temperature of the slowest cooling surface is not in the range of 1800° to 800°F for more than 3 minutes.

A.8.1.3.2 Carbon and Low Alloy Steel

Carbon and low alloy steel piping components and equipment pressure parts are heat treated in accordance with the requirements of the applicable ASTM or ASME materials specifications and applicable fabrication and erection code.

A.8.1.3.3 Welds

For original fabrication and erection, heat treatment of welds is in accordance with the requirements and recommendations of USAS B31.1.0, Power Piping (or later editions). For added and replacement piping, the heat treatment of welds is in accordance with the requirements of ASME B&PV Code Section XI 1980 Edition through Winter 1980 Addenda. The material is cleaned and degreased prior to heat treatment.

A.8.1.4 Descaling and Cleaning After Heat Treatment

A.8.1.4.1 Austenitic Stainless Steel

Austenitic stainless steel material is descaled and cleaned by one or a combination of the following methods:

1. Machining

2. Brushing (hand or driven) with austenitic stainless steel brushes which have not been used on carbon or low alloy steels

3. Blast cleaning with clean sand or grit not previously used on carbon or low alloy steels. Steel grit is not used

4. Pickling in accordance with ASTM A380 paragraph 4(b)5 with the following restrictions:

   Stainless steel material is pickled only when in the solution heat treated condition. Weldments, weld joints, or weld repairs are pickled only when the document and component have been solution heat treated subsequent to welding.

5. In addition to the methods above, for piping systems and portions of piping systems replaced during 1984 the following additions/alternatives are acceptable:

   a. Grinding using only silicone carbide or aluminum oxide grinding wheels not previously used on carbon or low-alloy steel.

b. Blast cleaning with steel or iron shot is acceptable if it is followed by blasting with new or reconditioned sand.

c. After pickling, materials are rinsed immediately, using water of G.E. specified quality.

d. Alkaline cleaners, vapor degreasers using chlorinated solvents, or surfactants may be used for shop cleaning or parts with no crevices. These parts are subsequently rinsed with high purity water.

e. Solvent cleaning with unused acetone, alcohol, toluene or another approved solvent.

A.8.1.4.2 Carbon and Low Alloy Steels

Carbon and low alloy steels are descaled and cleaned by one or a combination of the following methods:

1. Machining or grinding

2. Blasting in accordance with the Steel Structures Painting Council Standard, SP-5, except anchor patterns need not apply. When used, sandblasting is followed by vacuum or air blast cleaning

3. Pickling in accordance with the Steel Structures Painting Council Standard, SP-8

A.8.1.5 Defect Repair

Repair of base metal or weld metal defects is in accordance with the following requirements:

1. Surface defects such as laps or tears, which do not encroach on minimum wall thickness, are removed by machining or grinding and are blended into the adjacent metal surfaces

2. When defects or defect removal encroaches on minimum wall thickness, repairs are made by welding

A.8.1.5.1 Repair Welding

Repair welding is performed employing welding procedures and welders qualified in accordance with Section IX of the ASME Boiler and Pressure Vessel Code and Section XI for replacement recirculation, RHR, RWCU and Core Spray pipe, and HPCI turbine exhaust vacuum breaker piping.

A.8.1.5.2 Major Repairs

Major repairs, as defined herein, require approval. Records are kept indicating the nature of the defect removed, the location of the defect, subsequent heat treatment, or other pertinent data.

Major repairs in base materials such as plates, forgings, extruded pipe, or castings are defined as (a) a repair which requires excavation of material to a depth greater than 20 percent of the wall thickness, or when the extent of the cavity is greater than 10 in, (b) the repair of any crack in wrought base material, and (c) the repair of defects which are indicative of fundamental materials problems.

Major repairs in welds are defined as (a) the repair of any crack other than crater cracks or (b) the repair of defects which are indicative of either a fundamental materials problem or of a process out of control.

A.8.1.5.3 Inspection of Repair Welds

Major repair welds of a depth greater than 10 percent of the wall thickness must meet the inspection requirements for welds specified for the applicable classification of piping. Other inspection methods are not employed without approval.

A.8.1.5.4 Heat Treatment After Repair by Welding

For original fabrication and erection, base material repair welds are heat treated as required by the applicable material specifications. Weld repairs are heat treated as required by USAS B31.1.0, Power Piping. For recirculation, RHR, RWCU and core spray replacement pipe, and HPCI turbine exhaust vacuum breaker pipe, the base material repairs are heat treated as required by the applicable material specification. Weld repairs are heat treated as required by ASME B&PV Code Section XI.

A.8.1.6 Surface Finish

The surface finish of materials, welds, piping, and equipment pressure parts are suitable for the inspection and testing required by the applicable test schedule. Surface discontinuities which are markedly different from the overall finish are removed or blended into the adjacent surfaces. For replacement recirculation, RHR, RWCU and Core Spray pipe $\geq$ 2", the inside surfaces are finished to a number 4 plate finish or better.

A.8.1.7 Marking

Each section of fabricated piping pipe assembly, fitting, or equipment is clearly marked by identification tabs, bands, painting, or stamping with the appropriate data to indicate its place in the final erected assembly.

A.8.1.8 Submittals

Approval is required for the following fabrication and erection procedures:

1. Welding procedures

2. Repair procedures

3. Heat treatment procedures

4. Cleaning procedures

For replacement pipe, the following additional procedures are submitted for acceptance.

5. Removal procedures for replacement pipe

6. Installation procedure for replacement pipe

A.8.1.9 Inspection and Testing

Inspection and testing of piping and equipment pressure parts, including completed welds, assemblies, and subassemblies, is performed as shown in the applicable schedule for the specific classification of piping and equipment pressure parts. See Section A.6.

A.8.2 F3 Fabrication and Erection Schedule

A.8.2.1 Joint Selection and Limitations

Welded construction is preferred. Expanded or rolled joints, screwed joints, and caulked joints are used only in piping or equipment pressure parts where erosion, crevice corrosion (particularly chemical service), large particulate or fibrous materials, and chemical reactions are shown to be negligible.

A.8.2.2 Welding

All welding, including fillet, seal, repair, and attachment welds, is performed in accordance with written welding procedures. Procedure qualification and welder performance qualification are in accordance with USAS B31.1.0, Power Piping (or later editions).

A.8.2.3 Fabrication, Assembly, and Erection

The requirements for fabrication, assembly, and erection are in accordance with USAS B31.1.0 Power Piping (or later editions).

A.8.2.4 Inspection and Testing

Inspection and testing of piping and equipment pressure parts, including completed welds, assemblies, and subassemblies, are in accordance with the applicable schedule for the specific classification of piping and equipment pressure parts. See Section A.6.

A.8.3 F4 Fabrication and Erection Schedule

The requirements for fabrication, assembly, and erection are in accordance with USAS B31.1.0 Power Piping.

The requirements for fabrication, assembly, and erection are in accordance with ASME B31.1 Power Piping Code 1989 through addenda b for the Salt Service Water Piping Replacement Project.

A.9 M1, M2, M3 MATERIAL SCHEDULES

A.9.1 M1 Material Schedule (Stainless Steel)

Materials with a chemical analysis equivalent to AISI Type 304 or 316 in the following ASTM specifications are used except where furnace sensitization of the material is unavoidable. In this case, materials with a chemical analysis equivalent to Type 304L and Type 316L are used. Austenitic stainless steel is considered to be furnace sensitized if it has been heated by means other than welding within the range of 800°F to 1,800°F, or if it is cooled slowly through this temperature range.

Replacement materials are procured to meet the requirements of the edition of Construction Code to which the original component or part was constructed. Depending on availability, however, it may be necessary to procure materials to a later edition/addenda of the code. In such cases, manufacturer/supplier is required to certify that the design stress intensity values, allowable stresses, material properties, design fatigue curves are equal to or better than that specified in the original construction code.

A.9.1.1 Tubular Products

1. ASTM A312, Grade TP304, TP304L or TP316, TP316L (seamless or welded without filler material)
2. ASTM A376, Grade TP304, TP304L or TP316, TP316L (seamless)
3. ASTM A213, Grade TP304, TP304L or TP316, TP316L (seamless)
4. ASTM A249, Grade TP304, TP304L or TP316, TP316L (welded without filler metal)
5. ASTM A358, Class 1, Grade 304 or 316 (welded with filler metal)
6. ASME SA-358, Class 1 Nuclear Grade 316 (electric fusion welded)
7. ASTM A358, Class 1, Grade 316L (welded with filler metal)

A.9.1.2 Forged or Wrought Fittings, Flanges, or Pressure Retaining Parts

1. ASTM A182
2. ASTM A336
3. ASTM A403
4. ASME SA-182 Nuclear Grade 316 Class 1
5. ASME SA-403 Nuclear Grade 316 Class 1
6. SA 705 GR 630 (Wetwell to Drywell Vacuum Breaker Pallet)
7. SA 564 GR 630 (Wetwell to Drywell Vacuum Breaker Hinge Arms & Hinge Shaft

A.9.1.3 Cast Fittings, Flanges, or Pressure Retaining Parts

1. ASTM A351, Grade CF8, or CF8M

A.9.1.4 Plate

1. ASTM A240
2. ASME SA-240 Nuclear Grade 316, Class 1

A.9.1.5 Bolting Materials

1. ASTM A193, Grade B7, B8
2. ASTM A194, Grade 8, 2H
3. ASTM A453, all grades (if allowable stresses are available)
4. ASTM A540
5. ASTM A564, Grade TP630 (17.4 Ph) precipitation hardened at 1,08515±°F
6. ASTM A461, Grade 688

A.9.1.6 Welding Materials

Filler metals including consumable inserts used for austenitic stainless steel welds are selected and controlled to produce welds that contain a minimum of three percent ferrite based upon one of the following:

1. Chemical analysis of weld metal compared to the Schaeffler Diagram for Stainless Steel Weld Metal (For coated electrodes or for submerged arc wire flux combinations, a chemical analysis of as deposited weld material is required)

2. Aminco-Brenner Magne-Gage measurement

3. Severn-Gage measurement on test weld deposit

Electrodes and Welding Rods

Similar metal welds are made using ASTM A298 or A371 Type-308 or 316 filler metal except where furnace sensitization occurs; in this case Type 308L is used. Dissimilar metal welds are made using ASTM A298 or A371 Type 309.

A.9.1.7 Consumable Inserts

Consumable inserts are of the same nominal chemical composition as the filler metal.

A.9.2 M2 Material Schedule (Carbon Steel)

Note: For HPCI turbine vacuum breaker piping ASME SA materials are used in lieu of ASTM A materials.

A.9.2.1 Tubular Products

1. ASTM A155, Class 1, Grade 70 Firebox (welded with filler metals). Base material shall be ASTM A516, Grade 70
2. STM A106, Grade B (seamless)
3. ASTM A524, Grade 1 or 2 (seamless)
4. ASTM A333, Grade 1 or 6 (seamless)

A.9.2.2 Forged or Wrought Fittings, Flanges, or Pressure Retaining Parts.

1. ASTM A105, Grade II
2. ASTM A350, GR LFI
3. SA 705 GR 630 (Wetwell to Drywell Vacuum Breaker Pallet)

A.9.2.3 Cast Fittings, Flanges, or Pressure Retaining Parts

1. ASTM A216, Grade WCB
2. ASTM A352, GR LCB and LC-2
3. ASTM A395 (Ductile Iron)

A.9.2.4 Plate

1. ASTM A285
2. ASTM A516

A.9.2.5 Bolting Material

1. ASTM A193, Grades B7, B8, or B16
2. ASTM A194, Grades 7, 8, 2H
3. ASTM A540, Grades B23 or B24
4. ASTM A546, Grade TP630 (17.4Ph) precipitation hardened at 1,085 $\pm$15°F
5. ASTM A461, Grade 688

A.9.2.6 Electrodes, Welding Rods, and Fluxes

1. Mild steel covered arc welding electrodes per ASTM A233
2. Low alloy steel covered arc welding electrodes per ASTM A316
3. Electrodes and fluxes for submerged arc welding per ASTM A558
4. Mild steel electrodes (solid only) for gas metal arc or gas tungsten arc welding per ASTM A559

A.9.2.7 Consumable Inserts

Consumable inserts are of the same nominal chemical composition as the filler metal.

A.9.2.8 Shielded Metal Arc Welding

Low hydrogen electrodes are used for shielded metal arc welding of all welds where the pressure boundary material is greater than 1/2 inch in thickness and for all butt joints regardless of thickness.

A.9.3 M3 Material Schedule (Carbon Steel)

A.9.3.1 Tubular Products

1. ASTM A155, Class 2, Grade C55, Firebox
2. ASTM A106, Grade A or B
3. ASTM A53, Grade A or B

A.9.3.2 Forged or Wrought Fittings, Flanges, or Pressure Retaining Parts

1. ASTM A105, Grade I or II
2. ASTM A234, Grade WPA, WPB, or WPW
3. ASTM A181, Grade I or II
4. ASME SA420 WPL6
5. ASME SA105

A.9.3.3 Cast Fittings, Flanges, or Pressure Retaining Parts

1. ASTM A216, Grade WCA or WCB

A.9.3.4 Bolting

1. ASTM A193, Grades B7, B8, or B16
2. ASTM A194, Grades B7, B8, or 2H
3. ASTM A540
4. ASTM A564, Grade TP630 (17-4Ph)
5. ASTM A461, Grade 688

A.9.4 M3 Material Schedule (Titanium)

A.9.4.1 Tubular products

1. ASTM B - 337-87 Grade 2

A.9.4.2 Forged Fittings, Flanges, or Pressure Retaining Parts

1. ASTM B 381-87 Grade F2
2. ASTM B 363-83 Grade WPT 2

A9.4.3 Bolting

1. ASTM F 468-90b Alloy 2 Bolts
2. ASTM F 467-90b Alloy 2 Nuts

A.10 T1, T2, T3, T4, AND T5 INSPECTION AND TESTING SCHEDULES

A.10.1 T1 INSPECTION AND TESTING SCHEDULE

Refer to Section A.6 for application of the Testing and Inspection schedule and for test methods, techniques, and acceptance standards.

A.10.1.1 Certification

The manufacturer of the materials or components certifies that the requirements for which he is responsible, including those of this Appendix as well as those of the specific material specification, are fully satisfied.

A.10.1.2 *Hydrostatic Tests*

Piping and equipment pressure parts are hydrostatically tested. If any repairs are made, the piping or equipment pressure part is retested. If any omissions or modifications of the test requirement are made, the deviation is shown valid before approval.

A.10.1.3 Nondestructive Testing

A.10.1.3.1 Welds

Girth and longitudinal pressure containing complete penetration groove welds are 100 percent examined by radiography. Accessible surfaces of the weld and adjacent base metal are examined by either liquid penetrant or magnetic particle methods.

Fillet welds, socket welds, and nonpressure containing attachment welds such as supports, lugs, anchors, and guides are examined on all accessible surfaces by either liquid penetrant or magnetic particle methods.

Welds attaching branch connections larger than 4 inches in pipe size are 100 percent examined by radiography, and accessible surfaces of the weld and adjacent base metal are examined by either liquid penetrant or magnetic particle methods. Welds attaching branch connections 4 in and smaller are examined by either liquid penetrant or magnetic particle methods on the accessible surfaces of the weld and adjacent base metal.

A.10.1.3.2 Double Welded Joints

The back of the first side welded shall be ground or chipped to sound metal and visually inspected prior to welding the second side.

A.10.1.3.3 Castings

Castings for pressure containing components 4 in and larger in pipe size are 100 percent examined by radiography and all accessible surfaces, including machined surfaces, are examined by either the magnetic particle or the liquid penetrant method.

A.10.1.3.4 Forgings

Forgings for pressure containing components 4 in nominal diameter and larger are examined in the finished condition, on all accessible surfaces including machined surfaces, by either the liquid penetrant or the magnetic particle method. All pressure containing forged components for stainless steel piping systems and portions of piping systems replaced during 1984 are examined by a liquid penetrant test on all accessible internal and external surfaces regardless of size.

A.10.1.4 Submittals

Approval is required for the following inspection and test procedures:

1. Radiography
2. Ultrasonic testing
3. Liquid penetrant testing
4. Magnetic particle testing

A.10.2 T2 INSPECTION AND TESTING SCHEDULE

Refer to Section A.6 for application of the Testing and Inspection schedule and for test methods, techniques, and acceptance standards.

A.10.2.1 Certification

The manufacturer of the materials or components certifies that the requirements for which he is responsible, including those included in this Appendix as well as those of the specific material specification, are fully satisfied.

A.10.2.2 Hydrostatic Tests

Piping and equipment pressure parts are hydrostatically tested. If any repairs are made, the piping or equipment pressure part is retested. If any omissions or modifications of the test requirement are made, the deviation is shown valid before approval.

A.10.2.3 Nondestructive Testing

A.10.2.3.1 Welds

Girth and longitudinal pressure containing complete penetration groove welds are 100 percent examined by radiography.

Fillet welds, socket welds, and nonpressure containing attachment welds such as supports, lugs, anchors, and guides are examined on all accessible surfaces by either the liquid penetrant or the magnetic particle method.

Welds attaching branch connections larger than 4 inches in pipe size are 100 percent examined by radiography, except where configuration does not permit effective radiography; then the root and final pass is examined by liquid penetrant or magnetic particle methods.

Accessible surfaces of the weld and adjacent base metal of branch connections 4 in and less in pipe size are examined by either the liquid penetrant or the magnetic particle method.

A.10.2.3.1.1 Double Welded Joints

The back of the first side welded are ground or chipped to sound metal and visually inspected prior to welding the second side.

A.10.2.3.2 Castings

Castings for pressure containing components 4 in and larger in nominal pipe size are 100 percent examined by radiography unless otherwise indicated herein.

Castings for pressure containing components are examined in the finished condition on all accessible machined surfaces by either the liquid penetrant or the magnetic particle method.

Weld ends only of pump casings are radiographed.

A.10.2.3.3 Forgings

Forgings for pressure containing components larger than 4 inches in nominal pipe size are examined in the finished condition on all accessible surfaces including machined surfaces by either the liquid penetrant or the magnetic particle method.

A.10.2.4 Submittals

Approval is required for the following inspection and test procedures.

1. Radiography
2. Ultrasonic testing
3. Liquid penetrant testing
4. Magnetic particle testing

A.10.3 T3 INSPECTION AND TESTING SCHEDULE

Refer to Section A.6 for application of the Testing and Inspection schedule and for test methods, techniques, and acceptance standards.

A.10.3.1 Certification

The manufacturer of the materials or components certifies that the requirements for which he is responsible, including those included in this Appendix as well as those of the specific material specification, are fully satisfied.

A.10.3.2 Hydrostatic Tests

Piping and equipment pressure parts are hydrostatically tested. If any repairs are made, the piping or equipment pressure part is

retested. If any omissions or modifications of the test requirement are made the deviation is shown to be valid before approval.

A.10.3.3 Nondestructive Testing

A.10.3.3.1 Welds

Girth and longitudinal pressure containing complete penetration groove welds are 100 percent examined by radiography.

Fillet welds, socket welds, and nonpressure containing attachment welds, such as supports, lugs, anchors, and guides are examined on all accessible surfaces by either the liquid penetrant or the magnetic particle method.

Welds attaching branch connections larger than 4 inches in pipe size are 100 percent examined by radiography, except where configuration of the weld does not permit effective radiography; then the root and final pass are examined by liquid penetration or magnetic particle methods. Branch connections 4 in and less have accessible surfaces of the weld and adjacent base metal examined by either liquid penetrant or magnetic particle methods.

A.10.3.3.2 Castings

Castings for pressure containing components 4 in and larger in nominal pipe size are examined in the finished condition on all machined surfaces by either the liquid penetrant or the magnetic particle method.

A.10.3.3.3 Forgings

Forgings for pressure containing components 4 in and larger in nominal pipe size are examined in the finished condition on all machined surfaces by either the liquid penetrant or the magnetic particle method.

A.10.3.3.4 Submittals

Approval is required for the following inspection and test procedures:

1. Radiography
2. Liquid penetrant testing
3. Magnetic particle testing

A.10.4 T4 INSPECTION AND TESTING SCHEDULE

Refer to Section A.6 for application of the Testing and Inspection schedule and for test methods, techniques, and acceptance standards.

A.10.4.1 Certification

The manufacturer of the materials or components certifies that the requirements for which he is responsible, including those described in this Appendix as well as those of the specific material specification, are fully satisfied.

A.10.4.2 Hydrostatic Tests

Piping and equipment pressure parts are hydrostatically tested. If any repairs are made, the piping or equipment pressure part is retested. If any omissions or modifications of the test requirement are made, the deviation is shown valid before approval.

A.10.4.3 Nondestructive Testing

Inspection and testing of piping and equipment pressure parts are in accordance with USAS B31.1.0, Power Piping, and the applicable portions of the ASTM material specifications.

A.10.5 T5 INSPECTION AND TESTING SCHEDULE

Refer to Section A.6 for application of the Testing and Inspection schedule and for test methods, techniques, and acceptance standards. The inspection and testing of piping and equipment pressure parts are in accordance with USAS B31.1.0, Power Piping, and the applicable material specifications.

APPENDIX B

TECHNICAL SPECIFICATIONS

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B.1 TECHNICAL SPECIFICATIONS

The Technical Specifications for Pilgrim Nuclear Power Station were prepared in accordance with 10 CFR 50.36 and submitted to the AEC as part of Application Amendment 13 (2/12/1970). The Technical Specifications were subsequently issued by the AEC as part of License Amendment 1 (9/15/72). The document is entitled The Pilgrim Nuclear Power Station Technical Specifications, which is maintained in a document separate from this FSAR.

B.2 TECHNICAL SPECIFICATIONS RELOCATED TO THE FSAR

The NRC requirements related to the content of Technical Specifications is set forth in 10 CFR 50.36.

Regulation 10 CFR 50.36(c)(2)(ii) sets forth four criteria to be used in determining whether specifications and limiting condition of operations are required to be established for an item of safety significance and those which satisfy any of the criteria of 10 CFR 50.36(c)(2)(ii) must be retained in the Technical Specifications. Those specifications and LCOs that do not satisfy the 10 CFR 50.36(c)(2)(ii) criteria may be relocated to licensee controlled documents, such as FSAR, Core Operating Limits reports, Quality Assurance Manuals, etc.

Pilgrim amended the Technical Specifications by license amendments and relocated to the FSAR certain specifications, LCOs, and surveillance requirements that do not follow the four criteria in 10 CFR 50.36(c)(2)(ii). The Technical Specifications and related Bases relocated to this Appendix began with License Amendment 195.

The relocated Technical Specifications and related Bases contained in the following sections follow the same order as they had been in the Technical Specifications. The definitions and the rules of usage of the Technical Specifications and Bases apply to the relocated Technical Specifications and related Bases. Revision to the relocated Technical Specifications and related Bases may be carried out in accordance with the requirements of 10 CFR 50.59.

B.2.1 License Amendment No. 143: Relocation of Technical Specifications 3/4.12, Fire Protection.

This License Amendment relocated the following Technical Specifications (TS) to the FSAR (Rev. 13). TS 3/4.12.A, “Fire Detection Instrumentation,” to FSAR section 10.8.4.1 (new); TS Table 3.12-1, “Fire Detection Instrumentation,” to FSAR Table 10.8-1 (new); TS 3/4.12.B, “Fire Water Supply System,” to FSAR section 10.8.4.2 (new); TS 3/4.12.C, “Spray and/or Sprinkler Systems,” to FSAR section 10.8.4.3 (new); TS 3/4.12.D, “Halon System,” to FSAR section 10.8.4.4 (new); TS 3.4.12.E, “Fire Hose Stations,” to FSAR section 10.8.4.5 (new); TS Table 3.12-2, “Fire Hose Stations,” to

FSAR Table 10.8-2 (new); and, TS 3/4.12.F, “Fire Barrier System,” to FSAR section 10.8.4.6 (new).

B.2.2 License Amendment No. 195: Relocation of Technical Specification 3/4.6.I, Shock Suppressors (snubbers).

This License Amendment relocates Technical Specifications 3/4.6.I, “Shock Suppressors (snubbers)” and its related Bases to the FSAR. The affected TS contain snubber operability and surveillance requirements for all MODES of operation except cold shutdown and refuel, replacement or repair of inoperable snubbers, and the initiation of an engineering evaluation to determine whether the component by the snubber(s) is and remains capable of meeting its intended function in the specific safety system involved. The snubber inspection and surveillance program shall follow the fourth intervalcurrent ISI program plan.

B.2.3 License Amendment No. 196: Relocation of Technical Specification 3/4.2, Instrumentation that Initiates Rod Block.

This License Amendment relocates certain control rod block functions from Technical Specifications 3/4.2.C, “Control Rod Block Actuation,” Tables 3.2.C.1, 3.2.C-2, and 4.2.C to the FSAR. The instrumentation functions being relocated are those functions that provide information to the operators to help prevent unnecessary automatic reactor protection system actuation. They are used to monitor core reactivity, but they are not relied upon in the accident analysis to ensure specified fuel design limits are met for postulated transients or accidents. The associated Bases pages reflecting these changes are also relocated to the FSAR.

The summary of relocated requirements is as follows:

TS Table 3.2.C.1: The Average Power Range Monitor (APRM), Intermediate Range Monitor (IRM), Source Range Monitor (SRM), Scram Discharge Volume (SDV) and Recirculation Flow Converter Trip Functions that initiate control rod blocks including requirements for Minimum and available Operable Channels per Trip Function, Required Operational Conditions, and Table 3.2.C.1 Notes (1), (3), (4), and (6).

TS Table 3.2.C-2: The APRM, IRM, SRM, SDV, and Recirculation Flow Converter Control Rod Block Trip Functions Trip Setpoints (if applicable) and Note (2).

Table 4.2.C: The APRM, IRM, SRM, SDV, and Recirculation Flow Converter Instrument Channel minimum test and calibration frequency for control rod block actuation, including Table 4.2.C Note (3).

The relocated requirements are placed in the same order as they were in the Technical Specifications and the applicable Bases.

B.2.4 License Amendment No. 202: Relocation of Technical Specification 3/4.6.B, Primary System Boundary Coolant Chemistry.

This License Amendment relocates Technical Specifications 3/4.6.B, Primary System Boundary – Coolant Chemistry to the FSAR. The relocated portions involve limiting condition for operation and surveillance requirements for reactor coolant conductivity and chloride concentration. The associated bases sections are also relocated to the FSAR.

The relocated TS sections and Bases are revised to follow the latest revision to the BWR Water Chemistry Guidelines and BWRVIP Implementation Guide for primary system boundary coolant chemistry.

B.2.5 License Amendment 224: Relocation of Technical Specification 3/4.6.G Structural Integrity.

This license amendment relocated Technical Specification 3/4.6.G and its Bases to the FSAR. The relocated portions include limiting conditions for operation and surveillance requirements for structural integrity of the reactor coolant system boundary. The associated Bases section was also relocated to the FSAR.

B.2.6 License Amendment 225: Relocation of Technical Specification 3/4.6.C drywell equipment drain sump requirements.

This license amendment relocated certain portions of Technical Specification 3/4.6.C and its Bases to the FSAR. The relocated portions include limiting conditions for operation and surveillance requirements for the drywell equipment drain sump. The associated Bases were also relocated to the FSAR.

B.2.7 License Amendment No. 229: Adoption of Technical Specification Task Force (TSTF) Change TSTF-372, "The Addition of Limiting Condition for Operation (LCO) 3.0.8 on the inoperability of snubbers".

This License Amendment adds the following LCO:

When on or more required snubbers are unable to perform their associated support function(s), any affected supported LCO(s) are not required to be declared not met solely for this reason if risk is assessed and managed, and:

a. the snubbers not able to perform their associated support function(s) are associated with only one train or subsystem of a multiple train or subsystem supported system or are associated with a single train or subsystem supported system and are able to perform their associated support function within 72 hours; or

b. the snubbers not able to perform their associated support function(s) are associated with more than on train or subsystem of a multiple train or subsystem supported system and are able to perform their associated support function within 12 hours.

At the end of the specified period the required snubbers must be able to perform their associated support function(s), or the affected supported system LCO(s) shall be declared not met.

The Bases for the LCO is provided in Technical Specification Bases for LCO 3.0.8 on page B3/4.0-1.

B.2.8 License Amendment No.235: Relocation of Technical Specification 3/4.6.D Related to Safety Relief Valve Discharge Pipe Monitoring.

This License Amendment relocates certain instrumentation related to Safety Relief Valve (SRV) discharge pipe temperature monitoring requirements to the FSAR. The relocated portions include requirements related to SRV discharge pipe temperature monitoring, daily temperature logging, and calibration and functioning of discharge pipe temperature monitoring instrumentation.

B.3.0 RELOCATED TECHNICICAL SPECIFICATIONS AND RELATED BASES

B.3.2 INSTRUMENTATION THAT INITIATES ROD BLOCK

B.3.6.B PRIMARY SYSTEM BOUNDARY –COOLANT CHEMISTRY

B.3.6.C COOLANT LEAKAGE

B.3.6.D SAFETY/RELIEF VALVES TEMPERATURE MONITORING

B.3.6.G STRUCTURAL INTEGRITY

B.3.6.I SHOCK SUPPRESSORS (SNUBBERS)

TABLE 3.2.C.1

INSTRUMENTATION THAT INITIATES ROD BLOCKS

Trip Function	Operable Channels per Trip Function		Required Operational Conditions	Notes
	Minimum	Available		
APRM Upscale (Flow Biased)	4	6	Run	(1)
APRM Upscale	4	6	Startup/Refuel	(1)(6)
APRM Inoperative	4	6	Run/Startup/Refuel	(1)(6)
APRM Downscale	4	6	Run	(1)
IRM Downscale	6	8	Startup/Refuel, except trip is bypassed when IRM is on its lowest range	(1)(6)
IRM Detector not in Startup Position	6	8	Startup/Refuel, trip is bypassed when mode switch is placed in run	(1)(6)
IRM Upscale	6	8	Startup/Refuel	(1)(6)
IRM Inoperative	6	8	Startup/Refuel	(1)(6)

TABLE 3.2.C.1 (Cont)

INSTRUMENTATION THAT INITIATES ROD BLOCKS

Trip Function	Operable Channels per Trip Function Minimum	Operable Channels per Trip Function Available	Required Operational Conditions	Notes
SRM Detector not in Startup Position	3	4	Startup/Refuel, except trip is bypassed when SRM count rate is $\geq$ 100 counts/second or IRMs on Range 3 or above	(1)(4)(6)
SRM Downscale	3	4	Startup/Refuel, except trip is bypassed when IRMs on Range 3 or above	(1)(4)(6)
SRM Upscale	3	4	Startup/Refuel, except trip is by-passed when the IRM range switches are on Range 8 or above (4)	(1)(4)(6)
SRM Inoperative	3	4	Startup/Refuel, except trip is by-passed when the IRM range switches are on Range 8 or above (4)	(1)(4)(6)
Scram Discharge Instrument Volume Water Level - High	2	2	Run/Startup/Refuel	(3)(6)
Scram Discharge Instrument Volume-Scram Trip Bypassed	1	1	Refuel/Shutdown	(3)(6)

TABLE 3.2.C.1 (Cont)

INSTRUMENTATION THAT INITIATES ROD BLOCKS

Trip Function	Operable Channels per Trip Function Minimum	Operable Channels per Trip Function Available	Required Operational Conditions	Notes
Recirculation Flow Converter – Upscale	2	2	Run	(1)
Recirculation Flow Converter - Inoperative	2	2	Run	(1)
Recirculation Flow Converter - Comparator Mismatch	2	2	Run	(1)

NOTES FOR TABLE 3.2.C-1

1. With the number of operable channels:

   a. One less than required by the minimum operable channels per trip function requirement, restore an inoperable channel to operable status within 7 days or place an inoperable channel in the tripped condition within the next hour.

   b. Two or more less than required by the minimum operable channels per trip function requirement, place at least one inoperable channel in the tripped condition within one hour.

2. Deleted

3. If the number of operable channels is less than required by the minimum operable channels per trip function requirement, place the inoperable channel in the tripped condition within one hour.

4. SRM operability requirements during core alterations are given in Technical Specification 3.10.

5. Deleted

6. When the reactor mode switch is in the Refuel position, the reactor vessel head is removed, and control rods are inserted in all core cells containing one or more fuel assemblies, these Rod Block functions are not required.

7. Deleted

TABLE 3.2.C-2
CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

Trip Function	Trip Setpoint
APRM Upscale	(1) (2)
APRM Inoperative	Not Applicable
APRM Downscale	$\geq$ 2.5 Indicated on Scale
IRM Downscale	$\geq$ 5/125 of Full Scale
IRM Detector not in Startup Position	Not Applicable
IRM Upscale	$\leq$ 108/125 of Full Scale
IRM Inoperative	Not Applicable
SRM Detector not in Startup Position	Not Applicable
SRM Downscale	$\geq$ 3 counts/second
SRM Upscale	$\leq 10^5$ counts/second
SRM Inoperative	Not Applicable
Scram Discharge Instrument Volume Water Level - High	$<$ 17 gallons
Scram Discharge Instrument Volume - Scram Trip Bypassed	Not Applicable
Recirculation Flow Converter - Upscale	$\leq$ 120/125 of Full Scale
Recirculation Flow Converter - Inoperative	Not Applicable
Recirculation Flow Converter - Comparator Mismatch	$<$ 8% Flow Deviation
Mode Switch in Shutdown	Not Applicable

(1) The trip level setting shall be as specified in the CORE OPERATING LIMITS REPORT.

(2) When the reactor mode switch is in the refuel or startup positions, the APRM rod block trip setpoint shall be less than or equal to 13% of rated thermal power, but always less than the APRM flux scram trip setting.

TABLE 4.2.C

MINIMUM TEST AND CALIBRATION FREQUENCY FOR CONTROL ROD BLOCKS ACTUATION

Instrument Channel	Instrument Functional Test	Calibration	Instrument Check
APRM - Downscale	Once/3 Months	Once/3 Months	Once/Day
APRM - Upscale	Once/3 Months	Once/3 Months	Once/Day
APRM - Inoperative	Once/3 Months	Not Applicable	Once/Day
IRM - Upscale	(2) (3)	Startup or Control Shutdown	(2)
IRM - Downscale	(2) (3)	Startup or Control Shutdown	(2)
IRM - Inoperative	(2) (3)	Not Applicable	(2)
SRM - Upscale	(2) (3)	Startup or Control Shutdown	(2)
SRM - Inoperative	(2) (3)	Not Applicable	(2)
SRM - Detector Not in Startup Position	(2) (3)	Not Applicable	(2)
SRM - Downscale	(2) (3)	Startup or Control Shutdown	(2)
IRM - Detector Not in Startup Position	(2) (3)	Not Applicable	(2)
Scram Discharge Instrument Volume Water Level-High	Once/3 Months Refuel	Not Applicable	
Scram Discharge Instrument Volume-Scram Trip Bypassed	Once/Operating Cycle	Not Applicable	Not Applicable
Recirculation Flow Converter	Not Applicable	Once/Operating Cycle	Once/Day
Recirculation Flow Converter-Upscale	Once/3 Months	Once/3 Months	Once/Day
Recirculation Flow Converter-Inoperative	Once/3 Months	Not Applicable	Once/Day
Recirculation Flow Converter-Comparator Off Limits	Once/3 Months	Once/3 Months	Once/Day
Recirculation Flow Process Instruments	Not Applicable	Once/Operating Cycle	Once/Day

NOTES FOR TABLE 4.2.C

1. Not Applicable

2. Functional tests, calibrations and instrument checks are not required when these instruments are not required to be operable or are tripped. Functional tests shall be performed before each startup with a required frequency not to exceed once per week. Calibrations of IRMs and SRMs shall be performed during each startup or during controlled shutdowns with a required frequency not to exceed once per week. Instrument checks shall be performed at least once per day during those periods when the instruments are required to be operable.

3. This instrumentation is excepted from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel.

4. Not Applicable

5. Not Applicable

6. Not Applicable

7. Not Applicable

BASES:

3.2 PROTECTIVE INSTRUMENTATION

The APRM system provides a control rod block to prevent rod withdrawal beyond a given point, thereby possibly avoiding an APRM Scram. The rod block setpoint is automatically reduced with recirculation flow to form the upper boundary of the Pilgrim power/flow map. The flow biased APRM rod block is not necessary to prohibit fuel damage and is not included in the analysis of anticipated transients.

The RBM rod block function provides local protection of the core, for a single rod withdrawal error from a limiting control rod pattern. The RBM bypass time delay (t_{d2}) is the delay between the time the signal is normalized to the reference signal and the time the signal is passed to the trip logic. Control rod withdrawal is unrestricted during this interval. The RBM bypass time delay is low enough to assure that control rod movement is minimized during the time RBM trips are bypassed.

The IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that trip setting is less than a factor of 10 above the indicated level.

A downscale indication on an APRM, RBM or IRM is an indication the instrument has failed or the instrument is not sensitive enough. In either case the instrument will not respond to changes in control rod motion and thus, control rod motion is prevented. The downscale trips are as shown in Table 3.2.C-2.

The flow comparator and scram discharge volume high-level components have only one logic channel and are not required for safety.

The refueling interlocks also operate one logic channel, and are required for safety only when the mode switch is in the refueling position.

Control Rod Block and PCIS instrumentation common to RPS instrumentation have surveillance intervals and maintenance outage times selected in accordance with NEDC-30851P-A, Supplements 1 and 2 as approved by the NRC and documented in SERs (letters to D. N. Grace from C. E. Rossi dated September 22, 1988 and January 6, 1989).

B.3.6.B PRIMARY SYSTEM BOUNDRY – Coolant Chemistry

1. Reactor coolant chemistry control limits, associated sampling frequencies and associated actions taken when reactor coolant chemistry control limits are exceeded shall be adhered to in accordance with the latest revision of the BWR Water Chemistry Guidelines.

2. Deviations from limits, sampling frequencies and actions taken pertaining to reactor coolant chemistry control limits as specified in the latest revision to the BWR Water Chemistry Guidelines, shall be evaluated and documented in accordance with the latest revision of the BWR Vessel and Internals Project Program Implementation Guide.

LIMITING CONDITIONS FOR OPERATION

3.6 PRIMARY SYSTEM BOUNDARY (Cont)

C. Coolant Leakage

2. Leakage Detection Systems

a. The following reactor coolant system leakage detection systems shall be Operable:

1. The drywell equipment drain sump monitoring system.

SURVEILLANCE REQUIREMENTS

4.6 PRIMARY SYSTEM BOUNDARY (Cont)

C. Coolant Leakage

2. Leakage Detection Systems

The following reactor coolant leakage detection systems shall be demonstrated Operable:

a. For the drywell equipment drain sump monitoring system perform:

1. An instrument functional test at least once per 31 days, and

2. An instrument channel calibration at least once per operating cycle.

BASES:

3/4.6 PRIMARY SYSTEM BOUNDARY (Cont)

C. Coolant Leakage

Two leakage collection sumps are provided inside primary containment. Identified leakage is piped from pump seal leakoffs, reactor vessel head flange seal leakoff, selected valve stem leakoffs including recirculation loop and main steam isolation valves, and other equipment drains to the drywell equipment drain sump. The second sump, the drywell floor drain sump, receives leakage from the drywell coolers, control rod drives, other valve stems and flanges, floor drains, and closed cooling water system drains. Drainage into the drywell floor drain sump is generally considered unidentified leakage. Both sumps are equipped with level and flow monitoring equipment to alert operators if allowable leak rates are approached.

The drywell floor drain sump monitoring system, as required in 3.6.C.2, consists of one floor drain sump pump, plus associated instrumentation. The basic instrument system for the drywell floor drain sump is comprised of a flow integrator that is used to record the flow of liquid from the drywell floor drain sump. The drywell equipment drain sump is equipped similarly. A manual system whereby the time interval between sump pump starts is utilized to provide a back-up to the flow integrator if the instrumentation is found to be inoperable. This time interval determines the leakage flow using the tested capacity for the pump.

The 2 gpm limit for coolant leakage rate increase within any 24 hour period is a limit specified by the NRC in Generic Letter 88-01: "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping". This limit applies only during the RUN mode to accommodate the expected coolant leakage increase during pressurization.

The total leakage rate consists of all leakage, which flows to the drywell equipment drain sump (Identified leakage) and floor drain sump (Unidentified leakage).

FUNCTIONAL REQUIREMENTS FOR SRV/SSV TEMPERATURE MONITORING

3.6 PRIMARY SYSTEM BOUNDARY

D. Safety/ Relief Valves Temperature Monitoring

1. If the discharge pipe temperature of any safety relief valve (SRV) measured at 4.5 to 6 feet exceeds 180°F during normal reactor power operation for a period of greater than 24 hours, an engineering evaluation shall be performed justifying continued operation for the corresponding SRV temperature increases.

2. Any SRV whose discharge pipe temperature measured by the far tailpipe thermocouple exceeds 212°F for 24 hours or more shall be removed at the next cold shutdown of 72 hours or more, tested in the as-found condition, and recalibrated as necessary prior to reinstallation.

3. Whenever SRVs are required to be operable, at least one of the dual thermocouples at each of the following locations shall be functional.

   a. Bellows monitoring temperature (Applicable only to 3-Stage SRVs. Not required for 2-Stage SRVs)

   b. The discharge pipe temperature monitoring (Thermocouple 4.5 to 6 feet down stream from discharge point)

SURVEILLANCE REQUIREMENTS

4.6 PRIMARY SYSTEM BOUNDARY

D. Safety/ Relief Valves Temperature Monitoring

1. Whenever the safety relief valves are required to be operable, the safety relief valve discharge pipe temperature of each safety relief valve shall be logged daily.

2. Whenever the safety relief valves are required to be operable, the bellows thermocouple temperature of each safety relief valve shall be logged daily (Applicable only to 3-Stage SRVs. Not required for 2-Stage SRVs).

3. Instrumentation shall be calibrated and checked once per cycle during refueling outage.

4. First Stage Thermocouple, Second Stage Thermocouple, and safety relief valve discharge thermocouple located 16 to 22 feet down stream from discharge may be used to collect information for an engineering evaluation justifying continued plant operation for the corresponding temperature increases (First Stage Thermocouples and Second Stage Thermocouples are applicable only for 3-Stage SRVs. Not required for 2-Stage SRVs)

3/4.6 D TECHNCIAL BASES FOR LEAKAGE MONITORING

The purpose of monitoring Safety Relief Valve (SRV) discharge pipe temperature is to determine if the SRV is leaking so that appropriate corrective action can be taken to achieve acceptable SRV performance and compatibility with other requirements such as maintaining suppression pool temperature.

For the 3-Stage SRVs eEnhanced temperature monitoring thermocouples have been installed to detect, monitor, and evaluate degraded SRV performance from leakage. In addition to the thermocouples located on the discharge pipe 4.5 to 6 feet down stream of SRV discharge and the bellows monitoring thermocouples, there are thermocouples on the SRV first stage, second stage, and 16 to 22 feet down stream of SRV discharge. The bellows monitoring thermocouple will detect a degraded condition with the bellows and any leakage. The thermocouple 4.5 to 6 feet down stream of SRV discharge will detect any valve leakage due to the first stage, second stage or main stage. The first and second stage thermocouples can determine if valve leakage is due to first stage or second stage pilot leakage or main stage leakage. The thermocouple located 4.5 to 6 feet down stream of valve discharge is a backup to the acoustic monitor. The thermocouple located 16 to 22 feet down stream of the valve discharge is used to monitor larger leakage rates. A dual thermocouple will be installed at each temperature monitoring location (first stage, second stage, 4.5 to 6 feet down stream on SRV discharge pipe, 16 to 20 feet down stream on SRV discharge pipe, and bellows monitoring). Under Section 3.6.D.3, only one of the dual thermocouples at each location is necessary to perform the leakage monitoring function for 4.5 to 6 feet down stream on SRV discharge pipe and bellows monitoring. For the 2-Stage SRVs, only near (4.5 to 6 feet) and far (16 to 22 feet) tailpipe thermocouples are used for leakage monitoring.

SRV Temperature Detectors for Two and Three Stage SRVs

Temperature Detector Location	**Function**	**3-Stage SRV**	**2-Stage SRV**
Pilot Bellows Leakage Installed on bonnet discharge line	Leakage Detection Indication and Alarm	Required	Not Installed & Not Applicable
First Stage Pilot Installed in Valve Body Thermowell	Leakage Detection Indication Only	Required	Not Installed & Not Applicable
Second Stage Pilot Installed in Valve Body Thermowell	Leakage Detection Indication Only	Required	Not Installed & Not Applicable
Near Tailpipe (Discharge Line) 4.5 to 6 feet from SRV discharge	Leakage Detection Indication and Alarm	Required	Required
Far Tailpipe (Discharge Line) 16 to 22 feet from SRV discharge	Leakage Detection Indication Only	Required	Required

General Electric (GE) Service Information Letter (SIL) No. 196, Supplement 11, dated October 31, 1977, provides a recommendation to install thermocouples 4.5 to 6 feet down stream of the discharge flange to achieve good sensitivity to determine SRV leakage with an alarm point setting for all SRVs. Thermocouples installed 4.5 to 6 feet down stream of the discharge point meet this recommendation for Pilgrim. This instrumentation provides indication and an alarm in the Control Room. This instrumentation replaces the temperature indication from the 16 to 22 feet down stream of the SRV discharge for providing a control room alarm that an RV is leaking.

GE SIL 196. Supplement 5, dated October 31, 1977, recommends SRV bellows integrity monitoring with a pressure switch connected to the bonnet that surrounds the bellows assembly. Pilgrim has elected to monitor bellows integrity with a dual thermocouple connected to a sensing line attached to the bonnet surrounding the bellows assembly. Bellows assembly leakage will be considered to be present if there are indications of higher than normal temperature at the bellows thermocouple in combination with an unidentified drywell leakage rate increase. The thermocouples are more sensitive than pressure switches, as recommended by SIL 196, Supplement 11. The SRV bellows monitoring thermocouple will provide temperature indication and an alarm in the Control Room. The alarm setpoint will be determined during power ascension for operating Cycle 19 (restart from Refueling Outage 18) based upon the ambient temperature profile at the thermocouple location. The bellows temperature detector and alarm are applicable only to 3-Stage SRVs and not required for 2-Stage SRVs.

The SRV discharge pipe thermocouple located 4.5 to 6 feet down stream of the SRV discharge provides an alarm in the Control Room, which would indicate SRV leakage, but would not confirm SRV inoperability. Based on this indication an engineering evaluation would be required for continued operation. The SRV first stage, second stage and the discharge pipe thermocouple located 16 to 22 feet down stream of the discharge will provide additional information to determine the condition of SRV performance and leakage, which would be used in an engineering evaluation to determine the corrective actions.

LIMITING CONDITIONS FOR OPERATION

3.6 PRIMARY SYSTEM BOUNDARY (Cont)

G. Structural Integrity

1. The structural integrity of the primary system boundary shall be maintained at the level required by the ASME Boiler and Pressure Vessel Code, Section XI "Rules for Inservice Inspection of Nuclear Power Plant Components," Articles IWA, IWB, IWC, IWD and IWF and mandatory appendices as required by 10CFR50.55a(g), except where specific relief has been granted by the NRC pursuant to 10CFR50.55a(g)(6)(i).

SURVEILLANCE REQUIREMENTS

4.6 PRIMARY SYSTEM BOUNDARY (Cont)

G. Structural Integrity

Inservice inspection of components shall be performed in accordance with the PNPS Inservice Inspection Program. The results obtained from compliance with this program will be evaluated at the completion of each ten year interval. The conclusions of this evaluation will be reviewed with the NRC

BASES:

3/4.6 PRIMARY SYSTEM BOUNDARY (Cont)

G. Structural Integrity

The Pilgrim Nuclear Power Station Inservice Inspection Program conforms to the requirements of 10CFR50.55a(g). Where practical, the inspection of ASME Section XI Class 1, 2, and 3 components conforms to the edition and addenda of Section XI of the ASME Boiler and Pressure Vessel Code required by 10CFR50.55a(g). When implementation of an ASME Code required inspection has been determined to be impractical for PNPS, a request for relief from the inspection requirement is submitted to the NRC in accordance with 10CFR50.55a(g)(5)(iii).

Requests for relief from the ASME Code inspection requirements will be submitted to the NRC prior to the beginning of each 10 year inspection interval for which the inspection requirement is known to be impractical. Requests for relief from inspection requirements which are identified to be impractical during the course of the inspection interval will be reported to the NRC on an annual basis throughout the inspection interval.

LIMITING CONDITIONS FOR OPERATION

3.6 PRIMARY SYSTEM BOUNDARY

I. Shock Suppressors (Snubbers)

1. During all modes of operation except Cold Shutdown and Refuel, all safety-related snubbers listed in PNPS Procedures shall be operable except as noted in 3.6.I.2 through 3.6.I.3 below.

   An Inoperable Snubber is a properly fabricated, installed and sized snubber that cannot pass its functional test.

   Upon determination that a snubber is either improperly fabricated, installed or sized, the corrective action will be as specified for an inoperable snubber in Section 3.6.I.2.

2. Limiting Condition for Operability (LCO) 3.0.8 on the inoperability of snubbers is provided in TS LCO 3.0.8.

   Further corrective action for this snubber, and all generically susceptible snubbers, shall be determined by an engineering evaluation.

   The provisions of this section (3.6.I.1) are not applicable to snubbers removed from a piping system for maintenance purposes. For snubbers that are removed from service for maintenance purposes, Sections 3.6.I.1 and 3.6.I.3 apply.

SURVEILLANCE REQUIREMENTS

4.6 PRIMARY SYSTEM BOUNDARY

I. Shock Suppressors (Snubbers)

The following surveillance requirements apply to all safety-related hydraulic and mechanical snubbers listed in PNPS Procedures.

1. The snubber inspection and surveillance program shall follow the current ISI Program Plan.

2. Snubber Service Life Monitoring

   A. A record of the service life of each snubber, the date at which the designated service life commences and the installation and maintenance records on which the designated service life is based shall be maintained.

   B. This Snubber Service Life Monitoring Program shall become effective July 1, 1982.

LIMITING CONDITIONS FOR OPERATION

3.6 PRIMARY SYSTEM BOUNDARY (cont)

I. Shock Suppressors (Snubbers)

3. From and after the time a snubber is determined to be inoperable, improperly fabricated, improperly installed or improperly sized, if the requirements of Section(s) 3.6.I.1 and 3.6.I.2 cannot be met, then the affected safety system, or affected portions of that system, shall be declared inoperable, as provided by LCO 3.0.8.

4. Snubbers may be added to, or removed from, per 10 CFR 50.59, safety-related systems without prior NRC approval. The addition or deletion of snubbers shall be reported to the NRC in accordance with 10 CFR 50.59.

TABLE 4.6.I-1

(Deleted)

3/4.6.I

BASES 3/4.6 PRIMARY SYSTEM BOUNDARY

I. Shock Suppressors (Snubbers)

Snubbers are designed to prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or severe transient, while allowing normal thermal motion during startup and shutdown. The consequence of an inoperable snubber is an increase in the probability of structural damage to piping as a result of a seismic or other event initiating dynamic loads. It is therefore required that all snubbers required to protect the primary coolant system and all other safety-related systems or components be operable during reactor operation.

The visual inspection frequency is based on maintaining a constant level of snubber protection to systems. The cumulative number of inoperable snubbers detected during any inspection interval is the basis for establishment of the subsequent inspection interval and the existing inspection interval should remain in effect until its completion.

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and verified by inservice functional testing, that snubber may be exempted from being counted as inoperable.

Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to rejection of the snubber by visual inspection, and are exposed to the same environmental conditions such as temperature, radiation, and vibration.

When a snubber is found inoperable, an engineering evaluation is initiated, in addition to the determination of the snubber mode of failure, in order to determine if any safety-related component or system has been adversely affected by the inoperability of the snubber. Initiating this evaluation within 72 hours ensures that prompt corrective action will be afforded.

Hydraulic snubbers and mechanical snubbers may each be treated as a different entity for the above surveillance programs.

The service life of a snubber is evaluated via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc.). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life. The requirements for the maintenance of records and the snubber service life review are not intended to affect plant operation. Due to the number and complexity of the relevant interacting factors it was necessary to develop a comprehensive Service Life Program. This program became effective July 1, 1982.

B.4 REFERENCES (DELETED)

APPENDIX C

STRUCTURAL LOADING CRITERIA

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STRUCTURAL LOADING CRITERIA

C.1 SCOPE

This Appendix provides additional information pertinent to the preceding sections concerning the structural loading criteria applied to Class I structures and components. Station structures and components are classified according to service and location. Those which must be designed to retain their structural or mechanical integrity under improbable loading conditions are classified as Class I in accordance with the definitions presented in Section 12.2.1.1. Class I includes those structures and components whose failure or malfunction might cause or increase the severity of an accident. Class I structures and components are listed in Section 12.2.1.2.

The loads, loading combinations and allowable limits described in this Appendix apply only to Class I structures and components. The criteria in this Appendix are intended to supplement applicable industry design codes where necessary. Compliance with these criteria is intended to provide design safety margins which are appropriate to extremely reliable structural components when account is taken of rare event potentialities such as a Safe Shutdown Earthquake or postulated Loss of Coolant Accident (LOCA) or a combination of both.

Class I components may not always be designed to satisfy the criteria using analytical techniques; alternately, the design of some components may be based upon test results, empirical evidence, or experience.

C.2 CONCRETE AND STEEL STRUCTURES

C.2.1 Description

The Class I concrete and steel structures are designed considering three interrelated primary functions for the design loading combinations described in Section C.2.3. The first consideration is to provide structural strength equal to or greater than that required to sustain the combination of design loads, and provide protection to other vital Class I structures and components. The second consideration is to maintain structural deformations within limits such that Class I components will not experience loss of function. The third consideration is to preclude excessive leakage by preventing excessive deformation and cracking when containment integrity is required.

In general, the load combinations considered and their allowable limits are formed on a "quasi-probabilistic" basis. This means that the higher the probability that a given set of conditions could occur, the lower the allowable limits. This also forms the basis for neglecting some combinations of loads. In general, only one highly improbable event is considered in any load combination, since the probability of two unrelated, highly improbable events occurring simultaneously is vanishingly small.

The consequence of dropping a heavy load from various lifting devices has been evaluated to satisfy NUREG 0612, "Control of Heavy Loads at Nuclear Power Plants." Refer to FSAR Section 12.2.3.7 for a discussion of the heavy load handling program.

C.2.2 Loads

The loads considered in the design of Class I concrete and steel structures include the following:

D = Dead load of the structure and related equipment plus any other permanent loads contributing stress, such as soil or hydrostatic loads. Live loads expected to be present when the station is operating, and the loads due to thermal expansion under normal operating conditions. This load takes into account any deviations from normal operating conditions which are reasonably expected to occur during the design lifetime of the station

R = Loads resulting from jet forces and pressure and temperature transients associated with rupture of a single pipe within the primary containment. This load is considered as indicated in the tables

E = Loads due to the Operating Basis Earthquake (0.08g) horizontal ground acceleration; 2/3 of horizontal ground spectrum acceleration applied simultaneously for vertical seismic acceleration

E' = Loads due to the Safe Shutdown Earthquake (0.15g) horizontal ground acceleration; 2/3 of horizontal

ground spectrum acceleration simultaneously applied for vertical seismic acceleration

Flood = Loads due to flooding the drywell up to el 116 ft

W = Design wind loading conditions (Refer to Section 12.2.3.3)

T = Loads due to the effects of a tornado. (Refer to Section 12.2.3.3 and Appendix H)

C.2.3 Load Combinations and Allowable Limits

Load Combination	Limits
1. D + E	Stresses remain within normal code allowable stresses (AISC for structural steel, ACI for reinforced concrete, ASME Pressure Vessel Code Section III Class B for the primary containment). The customary increase in design stress for earthquake loadings is not permitted.
2. D + W	Maximum allowable stresses may be increased one- third above normal code allowable stresses.
3. D + R + E	Stresses remain within normal code allowable stresses (AISC for structural steel, ACI for reinforced concrete, ASME Pressure Vessel Code Section III Class B for the primary containment). The customary increase in design stress for earthquake loadings is not permitted. In the case of jet impingement loading on the primary containment, where it is backed up by concrete, the general primary membrane stress, plus the primary bending stress, plus the secondary membrane, plus bending stress must be less than either twice the yield stress or 90 percent of the ultimate stress, whichever is lower. For jet impingement loading on the primary containment (including containment penetration assemblies) where the primary containment is not backed up by concrete, the primary stresses must not exceed 90 percent of the yield strength of the material at 300°F. The steel drywell shell is backed up by concrete from the sand pocket above the embedment of the spherical portion at elevation 9 ft-2 in up to elevation 90 ft on the cylinderical portions (See Appendix L, Figures L.1-1 and 12.1-13).
4. D + E	Local membrane stresses in the primary containment

+ Flood	may exceed the yield point, but with a margin against rupture.
5. D + T	Maximum allowable stresses are as follows: Steel - 150 percent of AISC code allowable stress provided that the primary stress is smaller than the yield stress Concrete - 0.75 F'c (compressive strength of concrete) where the "Working Stress Design" and the "Ultimate Strength Design" methods are used When the "Ultimate Strength Design" method is used, a load factor of 1.0 is applied to this load combination with appropriate reduction factors as described in ACI-318-63
6. D + R + E'	Maximum allowable stresses are as follows: Steel - 150 percent of AISC code-allowable stress provided that the primary stress is smaller than the yield stress; or in case of jet impingement loading on the primary containment. Where it is backed up by concrete, the general primary membrane stress, plus the primary bending stress, plus the secondary membrane plus bending stress must be less than either twice the yield stress or 90 percent of the ultimate stress, whichever is the lesser. For jet impingement loading on the primary containment (including containment penetration assemblies) where the primary containment is not backed up by concrete, the primary stresses must not exceed 90 percent of the yield strength of the material at 300°F The steel drywell shell is backed up by concrete from the sand pocket above the embedment of the spherical portion at elevation 9 ft-2 in up to elevation 90 ft on the cylindrical portions (see Appendix L, Figure L.1-1 and 12.1-13) Concrete - 0.75 f'c (compressive strength of concrete) where the "Working Stress Design" and the "Ultimate Strength Design" methods are used. Reinforcement - 0.90 fy (yield strength of reinforcement) When the "Ultimate Strength Design" method is used, a load factor of 1.0 is applied to this load combination with appropriate reduction factors as described in ACI-318-63

As stated above, the ultimate strength method is to be used with a load factor of 1.0, and when this is done, the equivalent factor of safety for both concrete and reinforcing steel in flexure is 1/0.9 = 1.11, recognizing the effect of the capacity reduction factor of the ACI code.

When the working-stress method is used and comparing stresses at yield to maximum allowable values, the factor of safety for concrete becomes 0.85f'c / 0.75f'c = 1.13, and that for reinforcing steel 1.0 fy /0.9 fy = 1.11. Thus, the result is essentially the same by either method, and if the ultimate strength method is acceptable with the stated load factor, then the working stress method is acceptable with the stated permissible unit stresses. The level of safety provided is considered adequate because of the low probability of simultaneous occurrence of these load combinations.

C.2.4 Method of Analysis

A dynamic seismic analysis is performed for Class I structures as described in Section 12.2.3.5.2. The structural analysis is then performed using various calculational methods and techniques. Much of the structural analysis is performed utilizing the "Working Stress Design" method as defined in the ACI Standard Building Code Requirements for Reinforced Concrete (ACI 318-63) and in the AISC Manual of Steel Construction (Sixth Edition). Some portions of the Class I structures are designed by the "Ultimate Strength Design" method described in Chapters 15 through 19 of ACI 318-63. "Finite Element Stress Analysis" is also used for such items as drywell shielding concrete.

Load combinations and allowable limits on stresses are as shown in the preceding Section C.2.3. Moment reversal conditions are provided for in reinforced concrete members, by specifying the reinforcing steel for the positive moment capacity at a support as not less than 40 percent of the negative moment capacity provided at the same support. Structural ductility of concrete members is provided by proportioning the structure so that a calculated value of 0.9 fy for the main tension reinforcement is the limiting stress value at the critical cross section in flexure.

C.2.5 Implementation of Criteria

C.2.5.1 Reactor Building Foundation

The soil bearing stress under the Reactor Building on Table C.2-1 is derived from the application of all dead loads plus a 50 psf uniformly distributed live load on each main floor, to allow for temporary loads during station operation and refueling. Seismic loads for the Operating Basis and Safe Shutdown Earthquakes are also considered.

The primary objective is to establish whether soil bearing stresses will be within the allowable stresses, that there is no possibility of excessive uneven settlement, and that there will be no possibility of soil liquefaction when subject to seismic vibrations.

The analysis shows that maximum stresses are below allowable values, relative settlement will not exceed 1 in, and liquefaction will not occur.

C.2.5.2 Reactor Building Floor System

All floor systems (Table C.2-2) are designed for the loadings tabulated in Section 12.2.3.2. Because of its critical function and particularly heavy loading, the floor system in the spent fuel transfer truck access lock is selected as an example of the implementation of the criteria in the design of floor systems for the station.

Composite design is used in order to reduce deflection, and therefore reduce potential cracking of concrete. The design considers two cases of loading: uniform 1 ksf and AASHO H-20 truck loading upgraded to 40k single axle load to allow for the anticipated maximum possible cask transporter load to be carried. The worst condition of these two loadings is used for design. The conservative assumption is made that the seismic vertical load could occur simultaneously with other vertical live loads. Wind, (W) tornado (T), and accident (R) loads do not apply, and therefore are omitted. The resulting stresses are within the code allowable values, therefore, the floor system is structurally adequate without excessive deflections or cracking of concrete.

C.2.5.3 Reactor Building Concrete Walls

The concrete walls of the Reactor Building (See Tables C.2-3 and C.2-4) are analyzed for the design loads (D) and combined with seismic loads due to the Operating Basis Earthquake (E) or Safe Shutdown Earthquake (E'). Accident loads (R) are not applicable. Tornado wind (T) loads were checked and found not governing. Stresses due to seismic moments and shears were calculated, and combined together with the gravity stresses. When the Operating Basis Earthquake (E) is considered, the ACI Code permissible one third increase in allowable stresses was not used. Allowable stresses for the Safe Shutdown Earthquake (E') are limited to the minimum yield point as a general case. However in a few cases, stresses may exceed yield point. In this case an analysis, using the Limit-Design approach, is made to determine the energy absorption capacity which should be such that it exceeds the energy input. This method has been discussed in the AEC publication TID-7024 Nuclear Reactor and Earthquakes Section 5.7. The resulting distortion is limited to assure no loss of function and adequate factor of safety against collapse.

The dynamic seismic analysis of the Reactor Building included an allowable of 2 percent for structural damping, which, added to the soil damping and radiation damping, resulted in a total damping of 5 percent for the soil and structure composite system. The 2 percent structural damping is associated with about one-half the ultimate stress values. The calculated stresses associated with the Safe Shutdown

Earthquake are generally less than one-half the ultimate values, therefore cracking of concrete will be minimized and the resultant potential leakage will be negligible.

Tables C.2-3 and C.2-4 present three examples covering the various types of wall construction. These types consist of precast concrete panels; precast concrete panels as the outer sheathing with poured-in-place walls on the inside; and interior poured-in-place reinforced concrete walls.

Precast concrete panels are constructed of a very dense concrete having f ' = 6,000 psi in 28 days minimum and heavy reinforcement of ASTM designation: A 615 grade 60 steel. All panels are manufactured under controlled factory conditions which insure a homogeneous, crack resistant concrete product.

The precast concrete panels perform several functions. The panels, exposed to wind, will resist direct design tornado (T) wind pressure and suction, while the panels oriented parallel to the wind direction will act as shear walls. The ASCE Paper No. 3269, Wind Forces on Structures is used for establishing wind forces and their distribution on the structure. The allowable stresses are increased for the short duration of the tornado load up to the same limits as for the Safe Shutdown Earthquake.

It is assumed for the structural analysis of shear walls that two panels and two steel columns act together forming a rectangular frame resembling a flattened "Vierendeel" frame. These frames joining each other vertically and horizontally form a shear wall bounded by the roof at top, refueling floor at bottom, and corner column at sides.

For the structural analysis of the panels subjected to the direct wind pressure, it is assumed that each panel acts as a slab supported by columns and spanning in one direction only. Any fixity of supports due to embedment in concrete pilasters is conservatively not considered.

The shear and tension connection of panels to the columns is accomplished by embedding steel angles anchored with studs in the panels, and welding these angles to the column flanges. The stresses in weldments are shown in the presented example.

It is concluded that since the highest panel stresses are close to normal code allowable stresses, panel deformations will be within the elastic limit without permanent distortions or leak producing cracks. Therefore, the integrity of containment is assured.

C.2.5.4 Reactor Building Structural Steel Columns

The structural steel internal columns (See Table C.2-5) are controlled by vertical loading (D) only. Full design live loads are considered using code allowable reductions of live

loads carried by columns. Within the Reactor Building, below the refueling floor, horizontal loads due to seismic accelerations, wind, or tornado do not contribute to column loads. This kind of loading is carried by the rigid concrete shear walls and floor diaphragms which constitute a structural frame considerably stiffer than the steel frame. Wherever beams framing into columns exert unsymmetrical loads, such as a heavy girder connected to one column flange and a light beam connected to the other flange, the resulting bending moments in the column are considered. In all cases the computed combined stresses do not exceed code allowable stresses. Therefore, the structure has an adequate margin of safety. The column presented on Table C.2-5 represents one of the heaviest loaded columns in the Reactor Building.

C.2.5.5 Reactor Building Steel Roof Truss-Frame Above El. 117 ft

The structural steel truss frame (See Table C.2-6) and columns supporting roof and side concrete panels are analyzed for design loads (D) and for simultaneous Operating Basis Earthquake load (E). In accordance with PDAR Section II, paragraph 7.6 the customary one-third increase in allowable stresses for design load and Operating Earthquake loads was not used. The resultant stresses calculated are below the code allowable stresses; therefore, the integrity of secondary containment will be assured.

In the analysis for design load and Safe Shutdown Earthquake load or for design load and tornado wind load (T) the allowable stresses are limited to the minimum yield point as a general case. However in a few cases, stresses may exceed yield point. In this case an analysis, using the Limit-Design approach, is made to determine the energy absorption capacity which should be such that it exceeds the energy input. This method has been discussed in the AEC publication TID-7024 Nuclear Reactor and Earthquakes Section 5.7. The resulting distortion is limited to assure no loss of function and adequate factor of safety against collapse. It was obvious by inspection that the increase in stresses caused by the Safe Shutdown Earthquake would be small because of the relatively small masses of the concrete panels and framing above el+117. Therefore, containment integrity will be assured. On the other hand tornado winds will produce stresses approaching the allowable values. Two thirds of the roof may be blown off relieving internal building pressure. It is assumed that the remainder of the roof will stay on and be subjected to wind suction. This will produce reversal of the stresses in some of the frame members as shown on Table C.2-6. Precast concrete panels will remain in place and provide missile protection for the structure and portions of the structure which house equipment for safe plant shutdown.

Local crushing of concrete is expected at the missile impact zone with the possibility of spalling on the opposite face. The reinforcing can be expected to yield. Even in the event

of extensive concrete cracking, the curtains of reinforcing steel will restrain the missile from further penetration.

Structural design analysis was made of all exterior portions of structures to assure that they will be able to withstand missile penetration.

The method of analysis selected will either follow the method developed by Amirikian(1) or Gwaltney(2). For further discussion of the effects of horizontal and vertical missiles, refer to Appendix H.

C.2.5.6 Drywell Shielding Concrete

The comments regarding the Reactor Building concrete walls generally apply to the drywell concrete shield (See Table C.2-7) with some exceptions and additions as follows. The loss of coolant accident (LOCA) loading condition (R) is considered averaging operational and accident temperature gradients across the concrete shield wall. The temperature gradients are calculated using ACI Standard 505-54, Specification for the Design and Construction of Reinforced Concrete Chimneys. The case of a lined chimney with unventilated air space between lining and shell is considered similar to the drywell concrete shield wall where the steel drywell plates correspond to the chimney lining, the 2 in. air gap to the chimney unventilated space, and the concrete wall to the chimney shell. Air movement inside the drywell is analogous to the flow of gases in a chimney. The analysis shows that the stresses inside the drywell shielding concrete will not exceed the allowable values.

C.2.5.7 Reactor Concrete Pedestal

The reactor pressure vessel concrete pedestal (See Table C.2-8) is designed conservatively using allowable stresses in accordance with ACI Standard 505.54, Specification for the Design and Construction of Reinforced Concrete Chimneys. This standard is applicable to the pedestal which is similar to a chimney in that it has a hollow, circular cross section with openings and the shell thickness is small in proportion to the diameter. The thermal gradient through the shell is considered at the top of the pedestal where the shell thickens at the haunch below the vessel skirt. It can be expected that during an accident (R) a temperature difference between surfaces of the shell will occur somewhere near the vessel. The forces required to restrain a ruptured recirculation pipe are also considered. The calculated maximum stresses are close to the normal code allowable values indicating that excessive cracking will not occur.

As shown on Figure C.2-1, the horizontal shears on the reactor pressure vessel skirt flange are transferred to the top flange of the ring girder by 52 A490 high strength bolts in the same friction-type connection as is described in the AISC Code. During an earthquake, some of the bolts lose a

part of their clamping force due to applied tension; however, they suffer no overall loss of frictional resistance because the bolt tension produced by the moment is coupled with a compensating compressive force on the other side of the axis of bending (Figure C.2-2).

The applied tension force is 41 kips for the Operating Basis (0.08g) Earthquake load plus operating loads and 152 kips for Safe Shutdown Earthquake (0.15g) load plus operating loads and jet reaction while the installation clamping force is 239 kips. The applied shear force is 10.8 kips for Operating Basis Earthquake load plus operating loads, and 24.0 kips for Safe Shutdown Earthquake load plus operating loads and jet reaction, while the developed frictional resistance is 83.5 kips.

Since the amount of clamping force and developed frictional resistance in the connection are always greater than the applied tension and shear forces, respectively, the flanges cannot separate from, nor slip on each other. The friction-type connection is therefore structurally adequate.

The horizontal shear at the ring girder base is transferred to the top of the concrete pedestal by 104 anchor bolts in a bearing-type connection. The effects of the oversize holes on the design of the connection are eliminated by filling the space between the anchor bolts and ring girder with an epoxy material.**

The allowable stresses used in the design of the reactor pressure vessel support are shown below:

LOAD	ALLOWABLE STRESS
D+E	AISC allowable stress without usual increase for earthquake loads.
D+E'	150 percent AISC allowable stress
D+E'+R	AISC yield stress.***

D = Normal Operating Loads
E = Operating Basis Earthquake
E'= Safe Shutdown Earthquake
R = Jet Reaction Force
AISC = American Institute of Steel Construction, Latest Edition

In the design of the ring girder anchor bolts, a maximum thermal gradient of 80°F is conservatively assumed between the ring girder and concrete pedestal. Of this 80°F, 40°F is considered to be included in the operation loads, and the other 40°F is considered to be transient loads as could occur during the initial start-up phase of the reactor. For the maximum case (the maximum gradient and the Safe Shutdown Earthquake), the stresses will be within yield stresses. Hence, it can be seen that even for this extremely unlikely combination of loadings, the stresses will be satisfactory.

**The epoxy material used has a compression ultimate strength of 16,000 psi.

***There are no specific deformation criteria applied to this design.

D+E'+R load is designed to result in 100 percent of AISC yield stress, which is defined as the stress which results in 0.2 percent strain.

The steel ring 1 in. x 6 in. high is continuously welded inside the drywell using 1/4 in and 3/8 in fillet welds. The capacity of these welds is 7.0 kip/linear in. The seismic shear of 1,640 kip and seismic moment of 66,000 kip-ft will produce a load of 4.9 kip/linear in on the ring, which is less than 7.0 kip/linear in allowable. In addition, the vertical dead load of 8,250 kip, assuming a coefficient of friction of 0.2, will produce a friction force of 1,650 kip. Any bond force has been disregarded.

C.2.5.8 Primary Containment

The design, fabrication, erection, inspection, and testing of the primary containment (See Table C.2-9 thru C.2-13) conforms to ASME Pressure Vessel Code Section III (for Class B Vessel) including addenda up to June 6, 1967, and Code Cases 1177 and 1330. Elements such as jet deflectors, platforms, and accessories conform to AISC-1963 specification. Piping conforms to USASI Pressure Piping Code B31.1. Safety and construction requirements conform to the codes and regulations of the Commonwealth of Massachusetts.

The design analysis was based on the following general references:

- Formulas for Stress and Strain by Roark (4th Edition)
- Theory of Plates and Shells by Timoshenko, Woinowsky, Krieger, (2nd Edition)
- Beams on Elastic Foundations by Hetenyi (7th printing) University of Michigan Press
- Process Equipment Design by Brownell & Young (1959)
- AISC Manual of Steel Construction (6th Edition)

Some problems were solved using other references.

Some examples covering membrane stresses in the shell, jet loads, drywell stabilizer, and torus seismic ties are presented on Tables C.2-9 thru C.2-12. The design adequacy of the structure has been demonstrated by the successful initial pressure test.

Dynamic loads from LOCA were reevaluated in reference 4 to consider effects of operation in the maximum extended load line limit (MELLL) region. No design limits were exceeded due to operation in the MELLL region.

C.2.5.9 Biological Shield Wall

The biological shield wall (See Table C.2-13) is 27 in thick and consists of 27 in vertical WF beam columns, tied together by horizontal WF beams, 1-3/4 and 1/4 in steel plates. These plates are welded to both the inside and outside column flanges, thereby forming a walled shell. The shell is filled with concrete to form the shield. Reactor vessel penetrations which penetrate the biological shield are closed with removable shield plugs which fit around the penetration pipe. The removable shield plugs allow access for inservice inspections. Several of these shield plugs have been permanently removed to facilitate ISI nozzle inspections.

This configuration was conservatively analyzed to determine the capability of the shield wall to withstand pressures generated in the annulus between the reactor pressure vessel and the shield. The criteria utilized to estimate shield wall capability are: (a) that only the two steel plates, acting as a thin cylindrical shell resist the pressure forces with no credit for WF beam or concrete strength; (b) that shear yield stress of 1.67 times code allowable stress, or approximately 20 ksi is limiting, whereas ultimate shear is in the order of 40 ksi; and (c) that the pressure differential across the shield wall is a constant load although the differential pressure would continually decrease as the drywell is pressurized.

For these assumptions the shield wall is capable of withstanding a differential pressure of 54 psi.

The only breaks considered credible within the annulus region are those lines whose circumferential safe-end weld is inside the annulus. The minimum wall thickness for the various piping systems occurs at the safe-end joint to the piping. All other sections from this joint back to the vessel have thicker wall sections and therefore lower stresses. Lines having the safe-end weld within the annulus region in Pilgrim Nuclear Power Station include jet pump instrumentation (4 in.), core differential pressure instrumentation (2 in.) and control rod drive system return line (2-7/8 in.). The largest line break that could credibly pressurize the annulus is the 4 in jet pump instrumentation line which would result in a pressure differential across the shield wall of 8 psid. The effects of such pressurization of the annulus would not result in generation of missiles because the shield plugs would be restrained.

The static pressure differential and appropriate jet impingment pressures associated with other breaks within the annular region and breaks at safe-ends located within the biological shield have been evaluated and the results are as follows:

Line Size	Effective Blowdown Area Into Annulus (ft^2)	Annulus Static Pressure (psid)	Jet Impingement On Shield Plug (psid)	Total Pressure on Shield Plug (psid)
28	2.5	54	0	54
12	1.6	34	0	34
10	1.1	24	0	24
4	.17	8	9.5	17.5
3	.1	5	8	13

The above results are based on the following assumptions:

a) Complete severance at the safe-end weld location with lateral pipe displacement until the pipe wall contacts the shield plug casing

b) The effective blowdown area for the 28 in recirc. line is the total cross section of the displaced pipe exposed to the annulus plus one-half of the area overlapped by the nozzle end and pipe end. This total effective blowdown area and pipe displacement is shown on Figure C.2-3

c) The effective blowdown area from safe-end breaks in lines smaller than the recirculation lines has been conservatively assumed to be from a double ended break disregarding safe- end location and flow direction, although the safe-end weld is within the biological shield penetration for all except the 4 in and smaller lines

d) The total combined pressure of 54 psid would be reached by a postulated 28 in break within the annulus including both the resulting static pressure within the annulus and jet impingement directly on the shield plug

e) For all lines with the safe ends actually located within the biological shield, the blowdown jet from a safe end break would be essentially parallel to the shield plug casing. Therefore, no resulting jet impingment forces would exist tending to force the shield plugs outward into the drywell. For all lines with the safe ends located within the annulus region, the blowdown jet is assumed to impinge directly on the shield plugs

f) Based on the above, the static pressures and appropriate jet impingement pressures have been evaluated and all shield plugs will be restrained for the controlling annulus pressure of 54 psid, which is more than adequate to prevent the shield plugs from becoming missiles due to combined static pressure and appropriate jet impingement pressures

The shield plug details and restraint are shown on Figure C.2-4. This typical restraint mechanism

consists of beams across the outside face of the shield plugs. The restraining beams are held in place by brackets which are welded to the biological shield penetration liner. The beams will be fastened to the brackets but removable for access and inservice inspection of pipe weldments within the biological shield

C.2.5.10 Other Class I Structures

Other Class I structures such as the main stack, the control room, the Diesel Generator Building, Reactor Building auxiliary bay substructure, and the intake structure have been reviewed in a manner similar to the above for applicable combinations of design loads. The stresses in these structures have been found to be within the appropriate allowable limits.

C.2.6 References

1. Amirikian, A., Design of Protective Structures, Bureau of Yards and Docks, Department of the Navy, Washington, D.C., August 1950.

2. Gwaltney, R.C., Missile Generation and Protection in Light Water Reactors, Oak Ridge National Laboratory, March 1, 1967.

3. Deleted

4. G.E., "Maximum Extended Load Line Limit Analyses for Pilgrim Nuclear Power Station Reload 9 Cycle 10," NEOC-32306P, March 1994. (SUDDS/RF94-042)

TABLE C.2-1

REACTOR BUILDING FOUNDATION

Description/Criteria	Method of Analysis	Load Combination[1]	Maximum Allowable Stress	Maximum Stress	Location
Design load (D) includes all dead and equipment loads plus 50 psf live load.	Max. soil stress = $\frac{D+\Delta P}{A} + \frac{M}{S}$ (ksf)	D+E	15.0 ksf per PDAR Amendment 3 Appendix D	11.7 ksf	See Fig. 12.1-5 Under the edge of foundation mat at column lines 5,17 H or P
	where: D=Design load, kips ΔP=vert seismic load, kips M=Horiz. seismic overturning moment kip-ft A=foundation mat area, ft S=section modulus of foundation mat ft³				
Vertical seismic load (ΔP) is assumed to be produced by a vertical acceleration equal to 2/3 of the horizontal ground spectrum acceleration, both components of seismic motion acting simultaneously. Overturning moment (M) is produced by the horizontal seismic acceleration.		D+E[1]	15.0 ksf per PDAR Amendment 3 Appendix D	13.4 ksf	See Fig 12.1-5 Under the edge of foundation mat at H or P

[1]Floor design load (D) includes dead loads, equipment loads, and 50 psf live loads (appropriate for operating conditions) for combination with seismic loads.
Floor design is also checked for maximum live loads (200 to 1,000 psf) not in combination with seismic loads.

TABLE C.2-2

REACTOR BUILDING FLOOR SYSTEM

Description/Criteria	Method of Analysis	Load Combination	Maximum Allowable Stress (ksi)	Maximum Stress (ksi)	Location
Floor system is designed for two alternate loadings: Uniformly distributed load of 1.0 ksf. AASHO H-20 truck except maximum axle load assumed 40 kips. Higher stresses due to either of the above govern the design.	Working Stress Design Method and Composite Construction	D+E	F_c =1.8 F_c =1.8 F_v =0.126 F_t =24.0 F_b =24.0	f_c =1.28 concrete compression f_c =1.09 composite concrete compression f_v =0.117 concrete shear f_t =18.3 reinforcing tension f_b =22.9 structural steel bending, compact section	See Fig. 12.1-6 Spent Fuel Transfer Truck Access Floor at el. 23 ft in NW corner of reactor building concrete slab 1 ft-0 in thick; Steel beam 30 WF172 (compact)
Materials conform as follows: Concrete f' =4,000 psi at 28 days maximum strength per ACI 318-63 Reinforcing ASTM Designation: A615 Grade 60 per ACI 318-63 Structural Steel ASTM Designation: A36-67 per AISC Manual & Specification, 1963					
Vertical seismic load is assumed to be produced by a vertical acceleration equal to 2/3 of horizontal ground spectrum acceleration, both components of seismic motion acting simultaneously.					

TABLE C.2-2 (CONT)

Description/Criteria	Method of Analysis	Load Combination	Maximum Allowable Stress (ksi)	Maximum Stress (ksi)	Location
Maximum allowable stresses for D+E′ load combination are: Concrete F_c =0.75f_c' Reinforcing F_t =0.90fy Structural steel 150% of F_b (Code allowable) ≤Fy	Same as for D+E	D+E′	F_c =3.0	F_c =1.34 concrete compression	Same as for D+E
			F_c =3.0	f_c =1.15 composite concrete compression	
			F_v =0.248	f_v =0.123 concrete shear	
			F_t =54.0	f_t =19.2 reinforcing tension	
			F_b 36.0	f_b =24.1 structural steel bending	
Maximum allowable stresses for D+E combination are not increased above code allowable values, i.e. customary 1/3 increase not used.					

TABLE C.2-3

REACTOR BUILDING CONCRETE WALLS

Description/Criteria	Method of Analysis	Load Combination[1]	Maximum Allowable Stress (ksi)	Maximum Stress (ksi)	Location
Design Load (D) includes all dead and equipment loads, plus 50 psf live load.	Finite Element Stress Analysis	D + E	F_t=24.0 F_c=1.80	f_t=8.8 f_c=1.78	See Fig. 12.1-6 Internal Diagonal concrete Wall at J-7.4 between el. 23 ft and 51 ft
Materials Concrete f'_c =4000 psi at 28 days max. strength per ACI 318-63 Reinforcing ASTM Designation; A 615 Grade 60 per ACI 318-63	Working Stress Design Method used for tension and compression		F_v=0.070	f_v=0.068	
	Ultimate Strength Design Method used:	D + E'	F_t=54.0 F_c=3.0	f_t=9.2 f_c=1.87	See Fig. 12.1-6 Internal Diagonal concrete Wall at J-7.4 between el. 23 ft and 51 ft
			F_v=0.132	f_v=0.072	
Maximum allowable stesses for D+E load Concrete F_c=0.75 f'_c Reinforcing F_t=0.90f_y	For shear stresses only assume load factor for D + E loading per UBC (1967) Sec. 2632	D + E	F_t=24.0 F_c=1.80	f_t=17.8 f_c=0.74	See Fig. 12.1-5 External North wall on Line P above el.(-) 17 ft 6 in acting as Shear Wall
Maximum allowable stresses for D + E load combination are not incresed above code allowable values, i.e. customary 1/3 increase not used.	For D + E loading a coefficient 0.45/0.75 is used for conversion of 0.75 load factor to make it compatible with increase of allowable stresses used in the Working Stress Design Method		F_v=0.260	f_v=0.137	
		D + E'	F_v=0.260	f_v=0.138	See Fig. 12.1-5 External North wall on Line P above el.(-) 17 ft 6 in acting as Shear Wall
		D + E	F_t=24.0	Nominal reinforcing	See Fig. 12.1-6

TABLE C.2-3 (CONT)

Description/Criteria	Method of Analysis	Load Combination[1]	Maximum Allowable Stress (ksi)	Maximum Stress (ksi)	Location
			F_c=1.80	f_c=0.152	External North Wall on Line P above el. + 23 ft acting as Shear Wall
				f_c=0.152	
			F_v=0.260[2]	f_v=0.131	
		D + E′	F_v=0.260[2]	f_v=0.131	See Fig. 12.1-6 External North Wall on Line P above el. + 23 ft acting as Shear Wall

(1) Floor design load (D) includes dead loads, equipment loads, and 50 lb/ft² live loads (appropriate for operating conditions) for combination with seismic loads. Floor design is also checked for maximum live loads (200 to 1000 lb/ft²) not in combination with seismic loads.

According to the Uniform Building Code, Section 2617(j), the allowable shear stress carried by the concrete in a shear wall shall not exceed $v_c = (3.7\text{-}H/D)2\phi\sqrt{f'_c}$

where

- H = height of wall = 183 ft
- D = width of wall parallel to shear force = 142.5 ft
- ϕ = capacity reduction factor = 0.85
- f'_c = compressive strength of concrete = 4,000 psi

Hence $v_c = (3.7 - 1.28) \times 2 \times 0.85\sqrt{4000} = 260$ psi
Maximum calculated $v_c = f_v = 138$ psi

NOTE:

- F & f are tension in reinforcing
- F_c & f_c are compression in concrete
- F_v & f_v are shear in concrete

TABLE C.2-4

REACTOR BUILDING PRECAST CONCRETE WALL PANELS

Description/Criteria	Method of Analysis	Load Combination(1)	Maximum Allowable Stress (ksi)	Maximum Stress (ksi)	Location
		Acting as a Simple Span			
Precast concrete panels ~8" thick x ~24' x7' (approx.) are designed to carry seismic shear load and wind forces including tornado winds, tornado induced internal pressures and missiles. Horizontal joints are caulked and sealed and vertical joints are welded to columns to prevent air leakage, see Amendment No. 6 to PDAR, Comment No. 10.	Where panels are used as shear wall elements, two horizontal concrete panels and two steel columns are considered to form a "Vierendeel" truss. "STRESS" computer program, "Stiffness" method was used for analysis.	D+E D+E D+E D+E' D+E' D+E' D+T D+T D+T D+T	F_c=1.8 F_v=0.70 F_t=24.0 F_c=4.5 F_v=.110 F_t=54.0 F_c=4.5 F_v=.110 F_t=54.0 F_s=15.8	f_c=1.20 concrete compression f_v=.013 concrete shear f_t=16.4 reinforcing tension f_c=1.6 concrete compression f_v=.097 concrete shear f_t=35.6 reinforcing tension f_c=3.0 concrete compression f_v=0.037 concrete shear f_t=46.3 reinforcing tension f_s=1.5 weldment shear	See Fig. 12.1-10 & PDAR Amend. 6, Comment No. 10. Reactor Building panels above el. 117 ft
		Acting as a Shear Wall			
Welding of Steel angles embedded in panels to steel columns ensures structural continuity. Materials: Concrete f'_c =6000 psi at 28 days: Reinforcing ASTM Designation: A 615 Grade 60 per ACI 318-63	Where panels resist direct wind load or seismic load each panel is considered a simple span in one direction. Deflections for D+E' load found by inspection to be within elastic properties of materials, therefore neither permanent deformation nor excessive leakage is expected.	D+E D+E D+E D+E' D+E' D+E' D+T D+T D+T D+T	F_c=1.8 F_v=0.070 F_t=24.0 F_c=4.5 F_v=.110 F_t=54.0 F_c=4.5 F_v=.110 F_t=54.0 F_s=15.8	f_c=.9 concrete compression f_v=0.052 concrete shear f_t=19.0 reinforcing tension f_c=2.1 concrete compression f_v=0.25 concrete shear f_t=32.0 reinforcing tension f_c=1.6 concrete compression f_v=.102 concrete shear	Reactor Building panels above el. 117 ft

TABLE C.2-4 (CONT)

Description/Criteria	Method of Analysis	Load Combination[1]	Maximum Allowable Stress (ksi)	Maximum Stress (ksi)	Location
Tornado wind load (T) on walls averages a pressure at 300 psf (assumed 185 psf pressure, or 115 psf suction)				f_t=37.2 reinforcing tension f_s=12.7 weldment shear	
Allowable Stresses: No increase allowed for load D+E. For load D+E′ or D+T: Concrete F_c=0.75f'_c Reinforcing Ft=0.9f_y Welding F_s=15.8 ksi for Electrode 70 (no increase allowed)					

NOTE:

(1) D+W was considered and found not governing; for D+W the margins between the maximum allowable stresses and the calculated stresses were greater than for D+E or D+Tornado.

TABLE C.2-5

REACTOR BUILDING STRUCTURAL STEEL COLUMN

Description/Criteria	Method of Analysis	Load Combination	Maximum Allowable Stress (ksi)	Maximum Stress (ksi)	Location
Material: ASTM Designation: A 36 per AISC Manual & Specification 1963	Working Stress Design Method considering unsymmetrical beam loading where applicable	D+E	F_a =18.3 per AISC	f_a =16.4 axial compression	See Fig. 12.1-6 Column N-15.5 at el 23 ft 14 WF 320 plus 2/22 in X 1 5/8 in cover plates
Structural Steel Column designed for all dead, equipment, and live loads as follows: 200 psf on floors at el 51 ft, 74 ft, & 91 ft 800 psf on floor at el 117 ft		D+E'	F_a =33.0	f_a =17.3 axial compression	See Fig. 12.1-6 Column N-15.5 at el 23 ft 14 WF 320 plus 2/22 in X 1 5/8 in cover plates
Vertical seismic load is assumed to be produced by a vertical acceleration equal to 2/3 of the horizontal ground spectrum acceleration, both components of seismic motion acting simultaneously					

TABLE C.2-6

REACTOR BUILDING STEEL ROOF TRUSS-FRAME ABOVE EL 117 FT

Description/Criteria	Method of Analysis	Load Combination*	Maximum Allowable Stress ksi	Maximum Stress ksi	Location
Material: Structural steel ASTM Designation: A 36 per AISC Manual & Specification 1963	"Stress" computer program using "stiffness" method	D+E	F_a= -17.4	f_a= -16.8	See Fig. 12.1-10 & 12.1-15
		D+Tornado	F_t= +36.0	f_t= +27.8	Structural Steel Framing
		D+E	F_t= +22.0	f_t= +15.5	
		D+Tornado	F_a= -23.6	f_a= -23.6	
		D+E	F_a= -14.6	f_a= -12.6	Truss top chord
		D+ETornado	F_a= -26.8	f_a= -26.8	Truss top chord
		D+E	F_a= -11.8	f_a= -10.9	Truss bottom chord
	Working Stress Design Method	D+Tornado	F_a= -22.6	f_a= -22.6	Truss bottom chord
		D+E	F_b= -22.3	f_b= -14.5	Truss diagonal chord
Design wind loading: 44 psf on walls due to 110 mph wind; uplift 33 psf on roof due to 110 mph wind dead weight of built-up roofing 13 psf		D+Tornado	F_b= +36.0	f_b= +13.2	Truss diagonal chord
		D+E	F_b= -22.3	f_b= -5.2	Lower chord bracing
		D+Tornado	F_b= +36.0	f_b= +23.0	Lower chord bracing
					Cols on lines P & J
					Cols on lines P & J
					Cols on lines 5 & 17
					Cols on lines 5 & 17
Tornado wind load (T) on walls averages a pressure of 300 psf (assumed 185 psf pressure, or 115 psf suction); only 1/3 of the tornado load (T) acts on the roof because it is assumed that 2/3 of the roof area will be blown off.					Roof truss and columns above refueling floor el. 117 ft 0 in
100 ton capacity crane on column bracket.					
E' load disregarded because tornado is governing at similar allowable stresses of 150%F (code allowable)≤Fy					

*D+W was considered and found not governing; for D+W the margins between the maximum allowable stresses and the calculated stresses were greater than for D+E or D+Tornado.

TABLE C.2-7

DRYWELL SHIELDING CONCRETE

Description/Criteria	Method of Analysis	Load Combination (1)	Maximum Allowable Stress (ksi)	Maximum Stress (ksi)	Location
Drywell shield acts as a structural wall carrying floors. Design load (D) consists of all dead loads, equipment loads, and a 50 psf live load.	Finite Element Stress Analysis. The wall was divided into 89 elements approx. 2 ft thick x 4 ft high x 1 ft wide	D + E	F_t= 24.0	f_t= 23.8 reinforcing tension	Elements @ el 40 ft
		D + E	F_c= 1.80	f_c= 1.23 concrete compression	@ base of cone above equator
		D + E	F_v= .126	f_v= .113 concrete shear	Elements @ el 115 ft below, slab
Seismic loads (E & E') are according to the response spectra for the reactor building.	Thermal stresses are included in the finite element analysis. Seismic forces are superimposed on the results.	D + R + E'	F_t= 54.0	f_t= 51.2 reinforcing tension	Elements @ el 40 ft
		D + R + E'	F_c= 3.00	f_c= 1.78 concrete compression	Elements @ base of cone above equator
		D + R + E'	F_v= .248	f_v= .247 concrete shear	Elements @ el 115 ft below slab
Accident load (R) of 90°F (averaged) thermal gradient across the wall is considered. Operational thermal load of 45°F (averaged) thermal gradient across the wall is considered a part of the design load in D + E combination only.					
Materials conform as follows: Concrete f'_c = 4,000 psi at 28 days maximum strength per ACI 318-63 Reinforcing ASTM Designation: A 615 Grade 60 per ACI 318-63 Structural Steel ASTM Designation: A 36-67 per AISC Specification & Manual 1963.					

TABLE C.2-7 (Cont)

Description/Criteria	Method of Analysis	Load Combination	Maximum Allowable Stress (ksi)	Maximum Stress (ksi)	Location
Maximum allowable stresses for D + E' load combination are: Concrete F_c=0.75 f_c' Reinforcing F_t=0.90f_y					

(1) Floor design load (D) includes dead loads, equipment loads, and 50 psf live loads (appropriate for operation conditions) for combination with seismic loads.

Floor design is also checked for maximum live loads (200 to 1,000 psf) not in combination with seismic loads.

TABLE C.2-8

REACTOR CONCRETE PEDESTAL

Description/Criteria	Method of Analysis	Load Combination	Maximum Allowable Stress (ksi)	Maximum Stress (ksi)	Location
The Reactor Vessel Pedestal consists of a 4 ft - 0 in thick x 26 ft 3 in high cylindrical wall cantilevering from a concrete base of average thickness of 9 ft. This base has spherical shaped bottom matching the sphere of drywell. The shears and moments are transferred to the drywell through a welded steel shear ring and the friction between the drywell and the concrete.	Working Stress Design Method For seismic loads response spectra are used.	D + R + E	F_t=12.5	f_t=10.9 reinforcing tension	See Fig. 12.1-15 at base el 9' 2"
		D + R + E	F_c=1.0	f_c=.52 concrete compression	at base el 9' 2"
		D + R + E	F_v=.070	fv=.037 concrete shear	at base el 9' 2"
	Circumferential stresses due to temperature are in accordance with ACI 505	D + R + E	F_t=28.2	f_t=13.9 reinforcing tension	at base el 9' 2"
		D + R + E	F_c=1.68	f_c=1.29 concrete compression	at base el 9' 2"
		D + R + E	F_v=.132	f_v=.077 concrete shear	at base el 9' 2"
Materials Concrete f_c'=4000 psi at 28 days max. strength per ACI 318-63 Reinforcing ASTM Designation: A 615 Grade 60 per ACI 318-63 Allowable Stresses are in accordance with ACI 505-54		D + R + E'	F_t=20.0	f_t=13.2 circumferential tension in reinforcing	at top el 32' 6" due to 100°F temp. gradient
		D + R + E'	F_c=1.6	f_c=.376 circumferential compression in concrete	at top el 32' 6" due to 100°F temp. gradient
Maximum allowable stresses for D + R + E' load Concrete F_c=0.42 f_c' Reinforcing F_t=0.47 f_y					
Maximum allowable stresses for D + R + E load combination are not increased above code allowable values, i.e. customary 1/3 increase not used.					

TABLE C.2-8

REACTOR CONCRETE PEDESTAL (Cont)

Description/Criteria	Method of Analysis	Load Combination	Maximum Allowable Stress (ksi)	Maximum Stress (ksi)	Location
Accident load (R) includes jet forces due to a ruptured pipe and temperature gradient in concrete.					

*The maximum allowable stresses for concrete and reinforcing are $0.42f'_c$ and $0.47f_y$ respectively. These values were determined by ratioing the allowable values from ACI 505-54 (Specification for the Design and Construction of Reinforced Concrete Chimneys) and ACI 318-63 (Building Code for Reinforced Concrete Structures) in the following manner:

$$F_c = 0.75f'_c\,[f_{cw}/f'_c] = 0.75f'_c[0.25f'_c/0.45f'_c] = 0.42f'_c = 0.42 \times 4{,}000 = 1{,}680 \text{ lb/in}^2$$
$$F_t = 0.90f_y[f_{sw}/f_s] = 0.90f_y[12{,}500/24{,}000] = 0.47\,f_y = 0.47 \times 60{,}000 = 28{,}200 \text{ lb/in}^2$$

The actual calculated concrete and reinforcing working stress f_c and f_t are 1,290 lb/in² ($0.32\,f'_c$) and 13,900 psi ($0.23f_y$), respectively, acting vertically and are significantly below the maximum allowable stresses. These calculated stresses are for the D+R+E' loading combination which includes jet impingement loads. The maximum internally induced differential compartment pressure loading results in a calculated reinforcing circumferential stress of 13,880 lb/in² ($0.23f_y$). The maximum externally induced differential compartment pressure loading does not increase the maximum calculated concrete stress.

TABLE C.2-9

DRYWELL MEMBRANE STRESSES

Description/Criteria	Method of Analysis	Load Combination[1]	Maximum Allowable stress (ksi)	Maximum Stresses (ksi)				Location
The vessel is bulb shaped and houses the primary nuclear reactor vessel, the coolant recirculation lines, pumps, etc. In case of an operating accident the vessel must contain the steam released within the drywell, and conduct this steam to the suppression chamber.	ASME Section III including Code Cases 1330-1 and 1177-5 and addenda as of June 9, 1967 for Vessel Class "B".		Primary General Membrane PM = 17.5 @ 281°F	σ_t Circumferential	σ_l Meridional	σ_r radial	Critical Stress (total) P_m =	See Fig. 12.1-13
		D + R + E		7.74		-.028	7.768	Head
		D + R + E			3.98	-.028	4.008	Cylinder on head
	Stress intensities are defined per Code para. N-413 and their limits are per Code para. N-413.	D + R + E		9.128		-.028	9.156	Cylinder at top flange
		D + R + E		17.42		-.028	17.448	Cylinder
		D + R + E		16.25		-.028	16.278	Knuckle @el 56 ft
Structural steel material is ASME SA-516 fabricated to ASTM Designation: A 300, minimum service temperature 30°F, with Charpy impact requirements at maximum 0°F.	End conditions are found with methods described in the book Theory of Plates and Shells by Timoshenko.	D + R + E		13.41		-.028	13.438	Sphere
Seismic design load includes load due to vertical acceleration equal to 2/3 of the horizontal ground spectrum acceleration, both components of seismic motion acting simultaneously. For design criteria see Table 5.2-1.		D + R + E	Primary Local Membrane PL=1.5 PM = 26.25 @ 281°F	16.52		-.056	P_l=16.58	Cylinder at flange. (Much lower stresses are at other locations)

TABLE C.2-9 (Cont'd)

DRYWELL MEMBRANE STRESSES

Description/Criteria	Method of Analysis	Load Combination[1]	Maximum Allowable stress (ksi)	Maximum Stresses (ksi)			Location
After an accident the drywell may be flooded up to el. 116 ft; stresses shall be below yield point (without seismic load), or may exceed yield but with a margin against rupture if seismic is considered.		D + R + E	Primary + Secondary + Bending Q=3 PM 52.50 @ 281°F	18.03	.056	Q=18.09	Knuckle (stresses are much lower at other locations)
Accident load (R) includes pressure and temperature in the primary containment.		D + E + Flood	Yield 38.0 @ ambient Ultimate 70.0	25.64	7.23	32.87	Embedment @ el. 9 ft
			Critical buckling 23.77 (meridional)	29.94IAm	11.64IAm		

NOTE: ONLY ADDITIVE STRESSES SHOWN ABOVE FOR SIMPLICITY

NOTE:

(1) Maximum calculated stresses for D+E′ +Flood

TABLE C.2-10

JET IMPINGEMENT FORCE STRESSES

Description/Criteria	Method of Analysis	Load Combination	Maximum Allowable Stress (ksi)	Maximum Stress (ksi)	Reference	Location
A jet force is assumed to occur in any direction within the drywell.	Find maximum load on shell prior to breaking	D + R	30.33	28.80	1/(Case 20)	See Fig 12.1-13 Equipment door
		D + R	23.17	20.50	1/(Case 20)	Jet defelector support at vent
The force is calculated as 1,250 psi pressure acting on the area equal to the cross section of ruptured pipe. See PDAR V2.3.6 and Amendment No. 2 to PDAR, Question No. 3	Compare with given jet load	D + R D + R	90 (90% of yield of T-1 Steel at 300°F)	76.94	1/(Case 60)	Jet deflector baffle plate
		D + R	30.33	30.24	1/(Case 20)	Top closure head
The jet impingement force is considered to act coincidentally with the		D + R	30.33	29.46	3	Cylinder above flanges
design internal pressure and 150°F shell temperature	Apply the smaller load of either of the above; calculate deformation,	Experimental test in 1964	CB&I experimental investigation proved that 3/4 in thick plate can deform 3 in without failure			Spherical Shell See Amendment No. 5 to PDAR comment No. 11
Temperature of the shell and welds are assumed to be 300°F if hit directly by jet.	limiting stresses to prevent a progressive deformation or strain as follows:	D + R	30.33	22.72	2	Cylindrical Shell
	(a) $P_m + P_b$ ≤0.9 yield	D + R	27.90	23.12		Upper spray header pipe
	(b) $P_m + P_b + Q$ ≤0.9 Su (Ultimate Stress) (1)		21.23	17.7		Upper spray header weld
Local thermal effects and dynamic jet effects are disregarded						
	P_m=general primary membrane stress P_b=primary bending stress	D + R	30.33	15.51		Upper spray header support

TABLE C.2-10

JET IMPINGEMENT FORCE STRESSES (Cont)

Description/Criteria	Method of Analysis	Load Combination	Maximum Allowable Stress (ksi)	Maximum Stress (ksi)	Reference	Location
There is 2 in air gap between drywell shell and backup concrete.	Q=Secondary membrane + bending stress Su =Ultimate Stress is the minimum tensile strength at room temperature specified in the ASME Sec. III Yield taken from ASME Sec. III Table N-424 at pertinent temperature	D + R	27.90	27.23		Upper spray header inlet
		D + R	27.90	23.21		Lower spray header pipe
Material is ASME SA-516 Grade 70 fabricated to ASTM Designation: A-300 or T-1 [2] by USS where noted.	Assume shear type failure of weld, and its stress equal to 7/10 of parent material	D + R	21.23	15.83		Lower spray header weld
		D + R	30.33	29.52		Lower spray header support
The load combination D+R is a lesser case of D+R+E or D+R+E', the effect of E or E' is insignificant when compared to the effect of the jet impingement force.		D + R + E	63.0	47.65	4	Upper Beam Seats
		D + R + E	63.0	51.96	4	Lower Beam Seats

[1] The ultimate criterion allowed by the ASME Section III Code of twice the yield stress is not used because this stress is greater than 90% of the ultimate stress.

[2] T-1 steel was used only for the jet deflector baffle plate located in the drywell at the inlet to the vent piping. The baffle plate (piece marks 53 and 54), fabrication, and welding details are shown on Figure L.2-7 of Appendix L. Certified mill test reports were reviewed and are filed as part of the Pilgrim Station QA documentation. Fabrication and field assembly were inspected and certified by Hartford Steam Boiler Insurance Company (see L.5, Appendix L). The T-1 steel is not an integral part of the primary containment pressure boundary. The steel baffle was constructed to ASME Section III Code.

Reference:
1. Formulas for Stress & Strain by Roark
2. Analysis of Shells of Revolution by A. Kalnin, Journal of Applied Mechanics, Sept., 1964
3. Stresses from Radial Loads by P.P. Bijlaard, Welding Journal Dec., 1954
4. Theory of Plates by Timoshenko

TABLE C.2-11

TORUS SEISMIC TIES

Description/Criteria	Method of Analysis	Load Combination	Maximum Allowable Stress ksi	Maximum Stress ksi	Location
The saddles, located below the torus at 90° intervals are oriented so that either set of two opposite saddles will withstand a 0.12 g horizontal seismic acceleration (E).	Structural Design in Metals by C.D. Williams & E.C. Harris, p. 315	D+E	F_v=18.0	f_v=4.58 shear	At el.(-)17ft.6in below torus 5in Pin
		D+E	F_b=18.0	f_b=8.58 bending	
		D+E	F_c=1.0	f_c=2.13 concrete bearing	1 1/2in thick shear bar
		D+E	F_t=22.0	f_t=13.1 tension	1 1/2in thick shear bar
		D+E	F_p=27.0	f_p=11.99 bearing	1 1/2in anchor bolt
		D+E	F_p=27.0	f_p=15.00 bearing	1 1/2in thick upper plate 1 1/2in thick lower plate

Materials:
Concrete f'_c =4,000 psi at 28 days, ACI 318-63

Pin AISI-1081 (F_y=45 ksi)

Bolts ASTM Designation: A 36

Plates ASTM Designation: A 283 Grade C

Vertical seismic load is carried by supporting columns, not by saddles.

Accident condition for water volume equals 94,000 ft³

Maximum stresses may not exceed normal code allowable values.

TABLE C.2-12

DRYWELL STABILIZER SHEAR LUGS

Description/Criteria	Method of Analysis	Load Combination	Maximum Allowable Stress ksi	Maximum Stress ksi	Location
The stabilizer mechanism transfers into building the reaction due to seismic loads or seismic plus jet loads acting on the drywell, reactor and shield, or seismic plus flooding of the drywell.	ASME Code Section III including addenda as of June 9, 1967, Vessel Class "B"	D + E	Mate Lug SB=17.5 (plate) SS=12.25 (plate) SC=15.8 (weld)	σB=16.37 (plate) σS=4.22 (plate) σC=14.11 (weld)	See Fig. 12.1-13 Stabilizer Shear Lug for Drywell at el. 81 ft to 64ft
		D + R + E	SB=23.33 SS=16.33 SC=21.07	σB=22.34 σS=5.75 σC=19.26	
The geometry of the stabilizer allows for radial and vertical movements due to pressure and temperature	Formulas for Stress and Strain by Roark, Case 22 for plate.	D + E + Flood	SB=23.33 SS=16.33 SC=21.07	σB=19.16 σS=4.93 σC=16.51	
Materials: components attached to the drywell are ASME SA-516 Grade 70 fabricated to ASTM Designation: A 300, per ASME Code Section III: components outside the drywell are ASTM Designation: A 36 per AISC-1963	σC=combined stress $\sqrt{(\sigma B + \sigma T)^2 + \sigma S}$ σB=bending stress σT=tensile stress σS=shear stress σC=Fb (AISC)	D + E	Female Lug SB=22.0 SS=14.5 SC=15.8	σB=17.25 σS=3.74 σC=13.6	
		D + R + E	SB=29.33 SS=19.33 SC=21.07	σB=23.54 σS=5.1 σC=18.56	
		D + E + Flood	SB=29.33 SS=19.33 SC=21.07	σB=20.18 σS=4.37 σC=15.92	

NOTES:

Stress increase by 1/3 is allowed for jet loading or flooding.

Accident loads (R) considered include jet loads, temperature and pressure.

TABLE C.2-12

The margins at the drywell embedment in the case of flooding up to elevation 116 ft are calculated as follows:

Load Combination	Maximum Allowable Stress kip/lin in		Maximum Calculated Stress kip/lin in
D+E+Flood	Circumferential	38.00	25.64
D+E+Flood	Buckling	23.77	7.33
D+E+Flood	Circumferential	38.00	29.94
D+E+Flood	Buckling	23.77	11.64

Flooding to a lower elevation such as the top of the core will produce lower calculated stresses.

TABLE C.2-13

BIOLOGICAL SHIELD

Description/Criteria	Method of Analysis	Load Combination	Maximum Allowable Stress ksi	Maximum Stress ksi	Location
The biological shield wall is 27 thick and consists of 12-27 WF columns, tied together by horizontal welded plate girders, and 1 3/4 in and 1/2 in steel plates welded on both flanges of 27WF columns thereby forming a walled shell. The shell is filled with concrete to form the shield. The concrete fill was not considered for any Structural purpose.	Working Stress Design Method was used except as indicated otherwise. For seismic design the structure was modeled as a beam fixed at the bottom and hinged at the top. For jet forces WF columns were designed as fixed at both ends. After the integrity of the overall structure was ascertained, local stresses, connections and discontinuities were investigated.	D+R+E	F_b =24.0	f_b =18.6 bending	See Fig. 12.1-8 & Fig. 12.1-13, WF Col. at El.35 ft 0 in
		D+R+E	F_v =14.5	f_v =11.0 shear	See Fig. 12.1-8 & Fig. 12.1-13 WF Col at El. 35ft 0in
		D+R+E	F_a =18.3	f_a =2.53 axial compression	
		D+R+E'	F_b =36.0	f_b =29.90 bending	See Fig. 12.1-8 & Fig. 12.1-13 WF Col at El. 35ft 0in
		D+R+E'	F_v =18.0	f_v =13.1 shear	
		D+R+E'	F_a =32.0	f_a =5.0 axial compression	
		D+R	F_b =22.0	f_b =14.7 bending	Horizontal Restraint at E. 52 ft
		D+R	F_v =14.5	f_v =12.5 shear	Horizontal Restaint at E. 52 ft
Material: Structural steel ASTM Designation: A 36 per AISC Manual & Specification 1963	The shield wall was analyzed to withstand 54 psid pressure using the plastic design concept, and the allowable stress for the fillet weld would be increased by the factor of 1.67 times code allowable stress (AISC-1963, Sec. 2.7). Was assumed that only two steel plates, each acting as a thin cylindrical shell, would resist the pressure forces with no credit credit for WF beam or concrete strength.	D+R+E'	F_v =26.4	f_v =26.4 shear	Fillet welds attaching shell plates to WF framing

TABLE C.2-13

BIOLOGICAL SHIELD (Cont)

Description/Criteria	Method of Analysis	Load Combination	Maximum Allowable Stress ksi	Maximum Stress ksi	Location
	The shield plug restraint mechanism was designed for a static pressure inside the biological shield of 54 psid, using plastic design methods and a load factor of 1.0	D+R+E'	F =36.0	f =36.0 bending	Biological Shield Penetrations

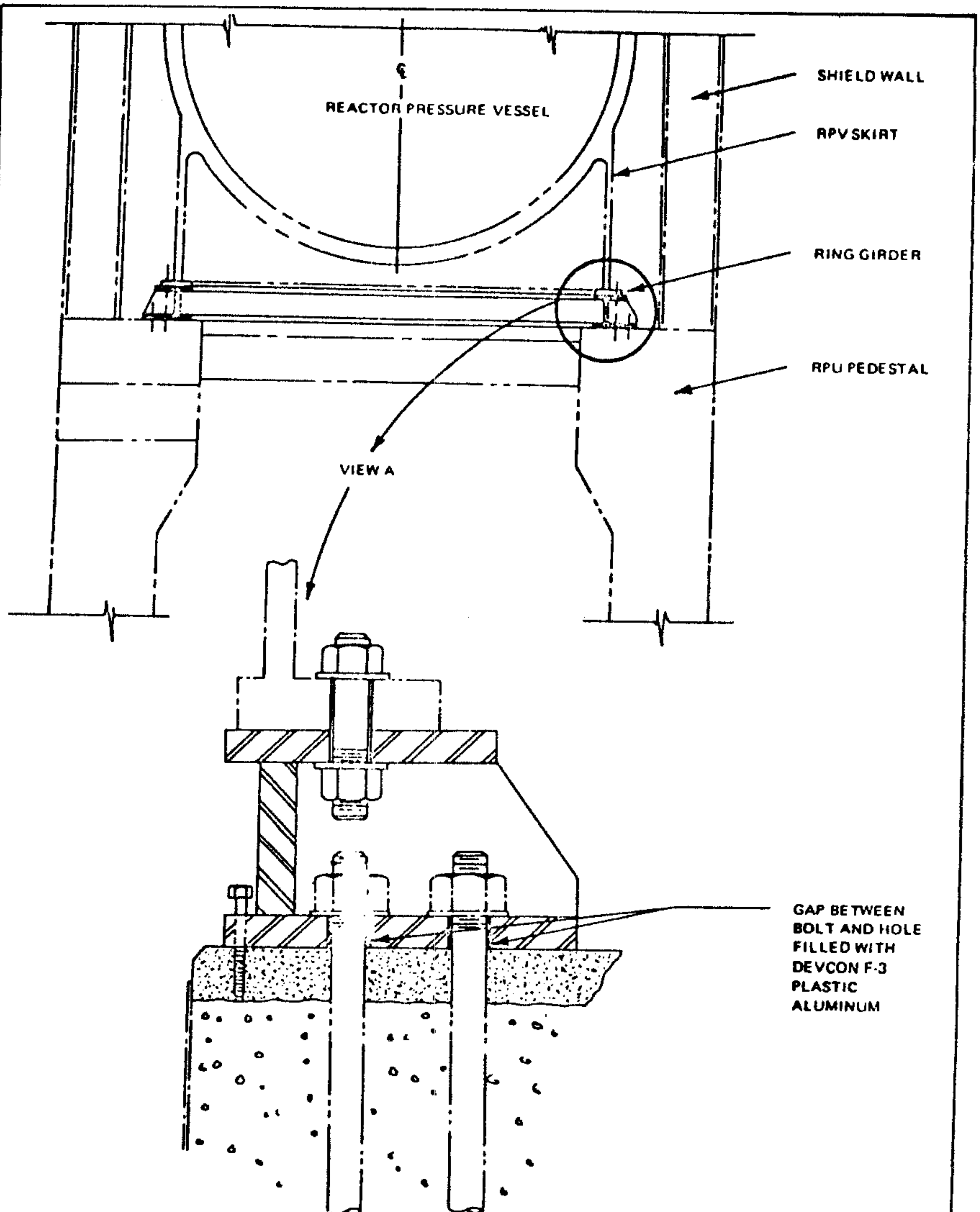
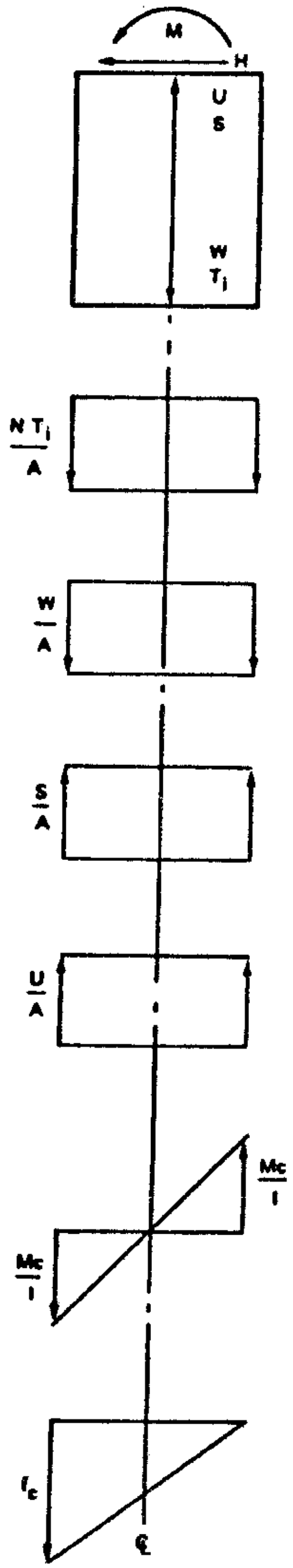


FIGURE C.2-1
REACTOR VESSEL SUPPORT ASSEMBLY
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT



$$\Sigma V = 0$$

$$\frac{N T_i}{A} + \frac{W}{A} = \frac{Mc}{I} + \frac{U}{A} + \frac{S}{A}$$

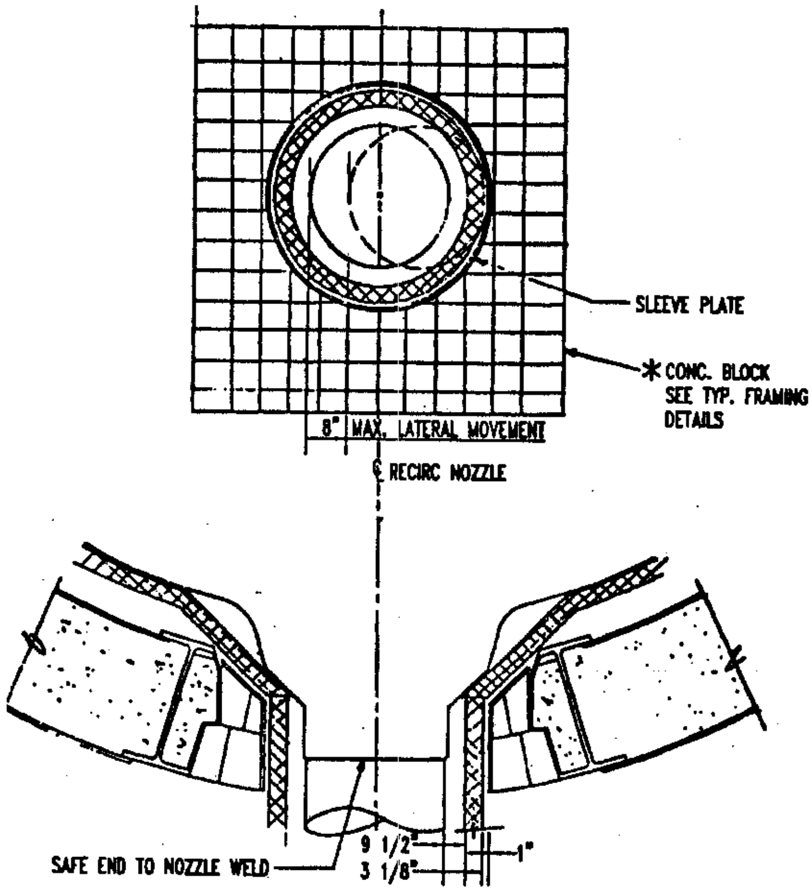
$$T_i = \frac{1}{N}\left(\frac{8 D_o M}{D_o^2 + D^2} + U + S - W\right)$$

$$T_b > T_i$$

WHERE

H	=	SEISMIC HORIZONTAL FORCE – KIPS
T_b	=	INSTALLATION TENSION – KIPS
T_i	=	TENSION W/O SEPARATION – KIPS
N	=	NUMBER OF H.S. BOLTS
M	=	SEISMIC MOMENT – INCH KIPS
U	=	SEISMIC UPLIFT – KIPS
S	=	OPERATING LOADS – KIPS
W	=	WEIGHT OF RPV – KIPS
D_o	=	O.D. OF SKIRT FLANGE – IN.
D	=	I.D. OF SKIRT FLANGE – IN.
V	=	VERTICAL STRESSES
f_c	=	NET COMPRESSIVE STRESS
A	=	AREA (sq in.)
I	=	MOMENT OF INERTIA
C	=	DISTANCE (in.)

FIGURE C.2-2
SKIRT FLANGE TO RING
GIRDER TOP FLANGE
FRICTION-TYPE CONNECTION
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT



* SEVERAL SHIELD BLOCKS WERE PERMANENTLY REMOVED AT SEVERAL NOZZLE PENETRATIONS.

FIGURE C.2-3
CONCEPTUAL RECIRCULATION NOZZLE BIO-SHIELD PENETRATION

PILGRIM NUCLEAR POWER STATION
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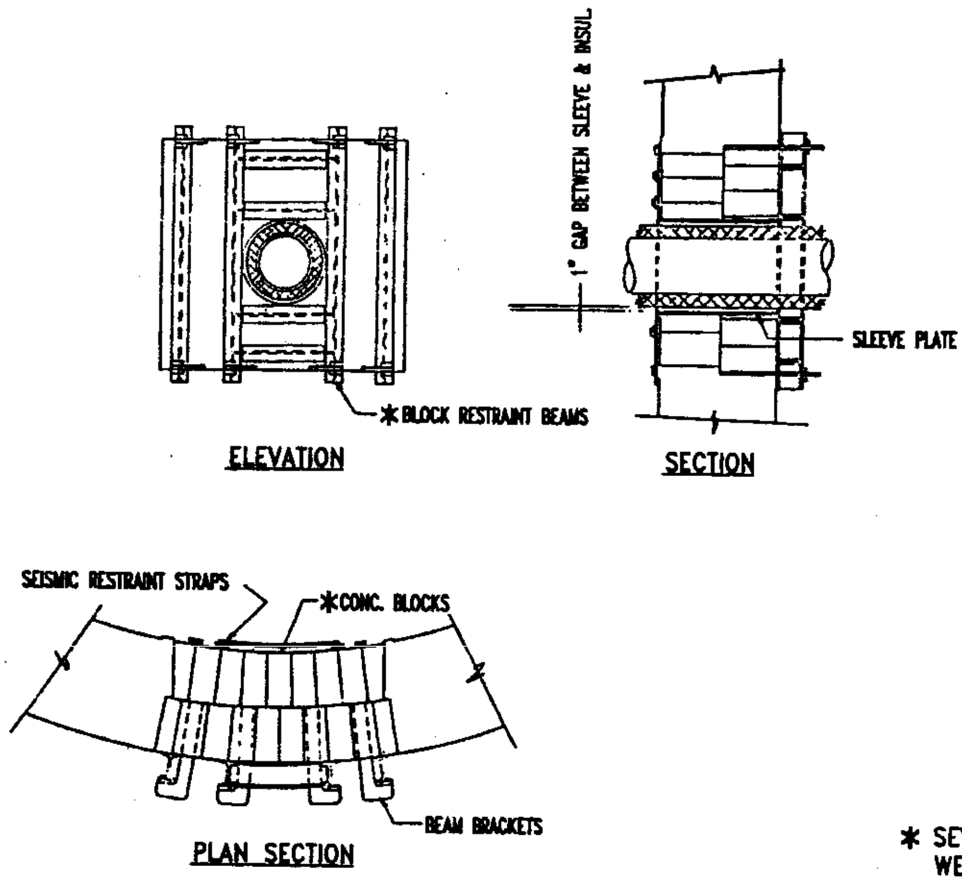

* SEVERAL SHIELD BLOCKS AND RESTRAINT BEAMS WERE PERMANENTLY REMOVED AT SEVERAL NOZZLE PENETRATIONS.

FIGURE C.2-4

CONCEPTUAL RECIRCULATION NOZZLE BIO-SHIELD PENETRATION

PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

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C.3 COMPONENTS

C.3.1 INTENT AND SCOPE

C.3.1.1 Components Designed by Rational Stress Analysis

These general design criteria are intended to apply to those ductile metallic structures or components which are normally designed using rational stress analysis techniques such as pressure vessels, reactor internal components, etc. The criteria may also be applied to those components or structures whose ultimate loading capability is determined by tests. These criteria are intended to supplement applicable industry design codes where necessary. Compliance with these criteria is intended to provide design safety margins which are appropriate to extremely reliable structural components when account is taken of rare event potentialities which might be associated with a Safe Shutdown Earthquake or primary pressure boundary coolant pipe rupture, or a combination of events.

C.3.1.2 Components Designed Primarily by Empirical Methods

There are many important Class I components or equipment which are not normally designed or sized directly by stress analysis techniques. Simple stress analyses are sometimes used to augment the design of these components, but the primary design work does not depend upon detailed stress analysis. These components are usually designed by tests and empirical experience. Complete detailed stress analysis is currently not meaningful nor practical for these components. Examples of such components are valves, pumps, electrical equipment and mechanisms. Field experience and testing are used to support the design. Where the structural or mechanical integrity of components is essential to safety, the components referred to in these criteria must be designed to accommodate the events of the Safe Shutdown or Operating Basis Earthquake, or a design basis pipe rupture, or a combination where appropriate. The reliability requirements of such components cannot be quantitatively described in a general criterion because of the varied nature of each component and its specific function in the system.

Class I seismic criteria were applied in the design of Class I piping inside the Diesel Generator Fuel Oil Storage tanks. However, seismic calculations were not performed for tanks because they are essentially as qualified as the ground they are buried in.

C.3.2 Loading Conditions and Allowable Limits

The loading conditions established herein are expressed in generic terms and are related in a probabilistic manner to the loads which are to be investigated for safety considerations. Related probabilistic definitions are used to determine an appropriate minimum safety factor which is used to establish structural design allowable limits and functional design allowable limits. Certain of the limits described in these criteria, i.e., deformation limit, and fatigue limit, are included for completeness, but do not necessarily require application to all components. Where it is clear to the designer that fatigue or excess deformation are not of concern for a particular structure or component, a formal analysis with respect to that limit is not required.

C.3.2.1 Loading Conditions

The loading conditions may be divided into four categories; normal, upset, emergency, and faulted conditions. These categories are generically described.

Normal Conditions

Any condition in the course of operation of the station under planned and anticipated conditions, in the absence of upset, emergency, or faulted conditions.

Upset Conditions

Any deviations from normal conditions anticipated to occur often enough that design should include a capability to withstand these conditions. The upset conditions include abnormal operational transients caused by a fault in a system component requiring its isolation from the system, transients due to loss of load or power, and any system upset not resulting in a forced outage. The upset conditions may include the effect of the Operating Basis Earthquake.

Emergency Conditions

Any deviations from normal conditions which require shutdown for correction of the conditions or repair of damage in the system. The conditions have a low probability of occurrence but are included to provide assurance that no gross loss of structural integrity will result as a concomitant effect of specific damage developed in the system.

Faulted Conditions

Those combinations of conditions associated with extremely low probability postulated events whose consequences are such that the integrity and operability of the nuclear system may be impaired to the extent where considerations of public health and safety are involved. Such considerations require compliance with safety criteria.

C.3.2.2 Allowable Limits

In addition to the generic definition of loading conditions in the preceding paragraphs, the meaning of those terms is expanded in quantitative probabilistic language. The purpose of this expansion is to clarify the classification of any hypothesized accident or sequence of loading events so that the appropriate limits or safety margins are applied. Knowledge of the event probability is necessary to establish meaningful and adequate safety factors for design. Table C.3-1 illustrates the quantitative event classifications.

These probabilities have been assigned to establish the appropriate structural design limits for the loading conditions in Section C.3.2.1. A summary of these limits is shown in the tables listed below:

Deformation Limit	Table C.3-2
Primary Stress Limit	Table C.3-3
Buckling Stability Limit	Table C.3-4
Fatigue Limit	Table C.3-5

There are many places where, through the exercise of designer judgment, it is unnecessary to actually carry out a formal analysis for each of these limits. A simple example consists of the case where two pieces of pipe of differing wall thickness are joined at a butt weld. If they are both subjected to the same loading, only the thinner piece would require a formal analysis to demonstrate that the primary stress limit has been satisfied.

The term SF min is defined as the minimum safety factor on load or deflection and is related to the event probability by the following equation:

$$SF\ min = \frac{9}{3 - \log10\ P_{60}}$$

where:

$$10^{-1} > P_{60} \geq 10^{-5}$$

For event probabilities smaller than 10-5 or greater than 10-1 the following apply:

$$10^{-5} > P_{60} \geq 10^{-6}\ (SFmin = 1.125)$$
$$1.0 > P_{60} \geq 10^{-1}\ (SFmin = 2.25)$$

These expressions show the probabilistic significance of the classical safety factor concept as applied to reactor safety. The SF min values corresponding to the event probabilities are summarized on Table C.3-6.

The loadings which occur as a result of the conditions listed are factored into the design of the components in accordance with the requirements of the applicable design code, or to the requirements of these criteria. Where permitted by the applicable code and by these criteria, the SF min may be progressively lowered to a minimum acceptable level on the basis that there is a lesser need for design margin for loading conditions which have a diminishing probability of occurrence.

C.3.3 Method of Analysis

C.3.3.1 Piping Systems

Where appropriate, the piping systems were dynamically analyzed using the "response spectrum method" of analysis. For each of the piping systems, a mathematical model consisting of lumped masses at discrete joints connected together by weightless elastic elements was constructed. Valves were also considered as lumped masses in the pipe, and valve operators as lumped masses acting

through the operator center of gravity. Where practical, a support is located on the pipe at or near each valve. Stiffness matrix and mass matrix were generated and natural periods of vibration and corresponding mode shapes were determined. The acceleration response spectra was used as input to the dynamic analysis for the piping anchors. The increased flexibility of the curved segments of the piping systems was also considered. The results for earthquakes acting in the x and y (vertical) directions simultaneously, and z and y directions simultaneously were computed separately. Maximum joint displacements, member forces and support reactions were determined by a square root of the sum of the square combination of each of these parameters for each mode and for each set of earthquake directions. For the replacement recirculation, RHR, and RWCU piping the inputs to the dynamic analyses were the 2.0 percent damped Operating Basis Earthquake acceleration response spectra and the 3.0 percent damped Safe Shutdown Earthquake acceleration response spectra for the piping supports. Colinear responses due to the orthogonal components for the simultaneous application of the three spatial directions of the seismic excitation were combined by the square root sum of the squares (SRSS) method. Modal responses were combined using the double sum or grouping method which includes the effects due to closely spaced modes. The member forces thus obtained were combined with the member forces produced by other loading conditions to compute the stresses. Analysis and allowable stress limits are in accordance with USAS B 31.1.0, Appendix A, or Appendix C, replacements are in accordance with the appropriate provisions of ASME, Section XI.

C.3.3.2 Equipment

The equipment was analyzed to determine equipment adequacy for earthquake loading. The equivalent static coefficients for the equipment were obtained from the amplified floor response spectra corresponding to the support elevations of the equipment. In lieu of determining the natural frequency of the equipment, the peak value of the applicable floor response spectrum was used in calculating the earthquake induced loads. Alternately, the natural frequency of the equipment was determined and corresponding input acceleration was obtained from the appropriate amplified floor response spectra. For the replacement piping valves and pumps the static coefficient method was not used. Instead, the equipment was analyzed as part of the piping system.

The extent of stress analyses performed on equipment due to seismic forces is dependent upon the type of equipment and the type of fabrication. Fabricated shapes are generally made from plate or rolled shapes with uniform thickness and shapes with regular geometric configurations. These can be more readily analyzed by rational means. Included in this category are tanks, certain parts of heat exchangers, etc. Cast shapes are generally made with non-uniform material thickness in complicated shapes that are not regular geometric configurations. Manufacturers have traditionally designed cast shapes conservatively since these do

not lend themselves to rational analysis. This conservatism has been demonstrated by extensive test and experience. Included in this category are pumps, valves, etc.

C.3.4 Implementation of Criteria

C.3.4.1 Reactor Vessel

Criteria

The reactor vessel has been designed, fabricated, inspected, and tested in accordance with the ASME Boiler and Pressure Vessel Code, Section III, its interpretations, and applicable requirements for Class A vessels as defined therein, as of the date that the reactor vessel order was placed.

Stress analysis requirements and load combinations for the reactor vessel have been evaluated for the primary loading and cyclic conditions expected throughout the vessel life, with the conclusion that ASME code limits are satisfied.

Primary Stress

Selected components, considered to possibly have high primary stresses as a result of rare events or a combination of rare events, have been analyzed in accordance with the requirements of the loading criteria in this appendix. Results of the most critical of those analyses are included on Table C.3-7. The conclusion is that the limits in the criteria have been met.

Vessel Fatigue Analysis

An analysis of the reactor vessel shows that all components are adequate for cyclic operation by the rules of Section III of the ASME Code. Operating cycles are specified on reactor thermal cycle Figure C.3-1 and nozzle thermal cycle Figures C.3-2 through C.3-8. Additional transient events based on operating experience and not on thermal cycle figures were also considered. The results given on Table C.3-8 indicate that the primary plus secondary stress intensity range is less than 3S for the more critical pressure boundary components on the vessel. A plastic analysis is, however, applied to two components. Also, the usage factors including Environmental Fatigue for the specified operating cycles is substantially less than the code allowed 1.0 for 60 years of operation.

C.3.4.2 Reactor Vessel Internals

Criteria

Although not mandatory, the design of the reactor vessel internals is in accordance with the intent of Section III of the ASME Boiler and Pressure Vessel Code. The material used for fabrication of most of the materials is solution heat treated, unstabilized type 304 austenitic stainless steel conforming to ASTM specifications. Allowable stresses for the internals materials under normal operating conditions are taken directly from Section III.

For rare events or a combination of rare events, the internals have been analyzed in accordance with the requirements of the loading criteria in this appendix, and results of the most critical of those analyses are included on Table C.3-8. The conclusion is that the limits in the criteria have been met.

In RFO #10 a reactor shroud repair was implemented as described in Section 3.3.4.1.1. The shroud stabilizer design meets the structural criteria as described in Section 3.3 and with primary stress limits as specified in Table C.3-3 as a minimum. Seismic analysis was performed in accordance with the methods described in Sections 3.3.6.9 and 12.2. IN addition, as a design guide, the 1989 ASME Code Section III, Subsections NB "Class 1 Components" and NG "Core Support Structures" were used.

Together, the shroud stabilizer tie rod, upper support bracket, and lower spring form the load path from the shroud top edge down to the shroud support plate gusset. Stress limits for these components are based on the ASME Code Subsection NG limits for threaded structural fasteners. These limits ensure that material yielding (permanent deformation) will not occur for all normal and upset conditions so that fastener preload is maintained and no uplift of the shroud head or separation of existing cracks will occur.

The threaded structural fastener limits were also applied to shroud stresses that are within the load path affecting the tie rod preload. This effectively prevents yielding within the shroud that would cause the tie roads to lose their preload as a result of normal or upset conditions. The limiting upset event is the loss of feedwater pumps with HPCI/RCIC injection. This event creates the greatest differential expansion between the shroud and the tie rod assembly.

Internals Deformation Analysis

Control Rod System

If there were excessive deformation of the control rod system, made up of the control rod drive, control rod drive housing, control rod, control rod guide tube and fuel channels, and the core structural elements which support them (top guide, core support, and shroud and shroud support) it could possibly impede control rod insertion. The maximum loading condition that would tend to deform these long, slender components is the Safe Shutdown Earthquake. Analyses of the internal components which have the highest calculated stresses are included in a following section. The highest calculated stresses occur where the Safe Shutdown Earthquake and loads resulting from the Design Basis Accident (DBA) line break are considered to occur simultaneously. Even in these cases, the general stress levels are relatively low. No significant deformation is associated with these calculated stresses; therefore, rod insertion would not be impeded even after an assumed simultaneous Safe Shutdown Earthquake and line break accident.

Core Support

The core support sustains the pressure drop across the core support plate and the fuel. This pressure drop is the only load which causes significant deflection of the core plate. Excessive core support deflection could lift the control rod guide tubes off their seats on the control rod drive housings and thereby increase core bypass leakage. This upward deflection would have to be 1/2 in to begin to lift guide tubes. The maximum deflection under normal operation conditions for the Pilgrim core support is calculated to be 0.053 in. Under pipe rupture differential pressures the deflection is calculated to be 0.080 in. The guide tubes will not be lifted off, although even if they were, this would not be of concern because bypass leakage at this time is not important.

Fatigue Analysis

A fatigue analysis was performed using as a guide the ASME Boiler and Pressure Vessel Code, Section III. The method of analysis used to determine the cumulative fatigue usage is described in APED-5460, Design and Performance of GE-BWR Jet Pumps, September 1968. The most significant fatigue loading occurs in the jet pump-shroud-shroud support area of the internals. The analysis was performed for a plant where the configuration (gusset type shroud support) was almost identical to the Pilgrim station. Therefore, the calculated fatigue usage is expected to be a reasonable approximation for this station.

Loading Combinations and Transients Considered

1. Normal start up and shutdown
2. Operating Basis and Safe Shutdown Earthquakes
3. Ten minute blowdown from a stuck relief valve
4. HPCI operation
5. LPCI operation (DBA)
6. Improper start of a recirculation loop

Cumulative Fatigue Usage

$U_{allowable} = 1.0$

$U_{calculated} = 0.65$

The location of maximum fatigue usage is at the inside diameter of the jet pump diffuser adapter, at the thin end of the tapered transition section.

C.3.5 Miscellaneous Components

Test results and analyses of miscellaneous components are shown on Table C.3-9.

TABLE C.3-1

LOADING CONDITION PROBABILITIES

	P_{60} = 60 yr event encounter probability			
Upset (likely)	1.0 >	P_{60}	≥	10^{-1}
Emergency (low probability)	10^{-1} >	P_{60}	≥	10^{-1}
Faulted (extremely low probability)	10^{-3} >	P_{60}	≥	10^{-6}

TABLE C.3-2

DEFORMATION LIMIT

Either One of (Not Both)		General Limit
a.	$\frac{\text{Permissible Deformation, DP}}{\text{Analyzed Deformation Causing Loss of Function, DL}}$	$\leq \frac{0.9}{\text{SF min}}$
b.	$\frac{\text{Permissible Deformation, DP}}{\text{Experimental Deformation Causing Loss of Function, DE}}$	$\leq \frac{1.0}{\text{SF min}}$

where

DP = Permissible deformation under stated conditions of normal, upset, emergency or fault

DL = Analyzed deformation which would cause a system loss of function[(1)]

DE = Experimentally determined deformation which would cause a system loss of function[(1)]

(1) Loss of Function can only be defined quite generally until attention is focused on the component of interest. In cases of interest, where deformation limits can affect the function of equipment and components, they will be specifically delineated. From a practical viewpoint, it is convenient to interchange, with the loss of function condition, some deformation condition at which function is assured if the required safety margins from the functioning condition can be achieved. Therefore, it is often unnecessary to determine the actual loss of function condition because this interchange procedure produces conservative and safe designs. Examples where deformation limits apply are: control rod drive alignment and clearances for proper insertion, core support deformation causing fuel disarrangement or excess leakage of any component.

TABLE C.3-3

PRIMARY STRESS LIMIT

Any One of (No More than One Required)		General Limit
a.	$\frac{\text{Elastic Evaluated Primary Stresses, PE}}{\text{Permissible Primary Stresses, PN}}$	$\leq \frac{2.25}{SF_{min}}$
b.	$\frac{\text{Permissible Load, LP}}{\text{Largest Lower Bound Limit Load, CL}}$	$\leq \frac{1.5}{SF_{min}}$
c.	$\frac{\text{Elastic Evaluated Primary Stress, PE}}{\text{Conventional Ultimate Strength at Temperature, US}}$	$\leq \frac{0.75}{SF_{min}}$
d.	$\frac{\text{Elastic Plastic Evaluated Nominal Primary Stress, EP}}{\text{Conventional Ultimate Strength at Temperature, US}}$	$\leq \frac{0.9}{SF_{min}}$
e.	$\frac{\text{Permissible Load, LP}}{\text{Plastic Instability Load, PL}}$	$\leq \frac{0.9}{SF_{min}}$
f.	$\frac{\text{Permissible Load, LP}}{\text{Ultimate Load from Fracture Analysis, UF}}$	$\leq \frac{0.9}{SF_{min}}$
g.	$\frac{\text{Permissible Load, LP}}{\text{Ultimate Load or Loss of Function Load from Test, LE}}$	$\leq \frac{1.0}{SF_{min}}$

TABLE C.3-3 (Cont)

Where:

PE= Primary stresses evaluated on the elastic basis. The effective membrane stresses are to be averaged through the load carrying section of interest. The simplest average bending, shear or torsion stress distribution which will support the external loading will be added to the membrane stresses at the section of interest.

PN= Permissible primary stress levels under normal or upset conditions under applicable industry code.

LP= Permissible load under stated conditions of emergency or fault.

CL= Lower bound limit load with yield point equal to $1.5S_M$' where S_M is the tabulated value of allowable stress at temperature as contained in the ASME III code or its equivalent. The lower bound limit load is here defined as that produced from the analysis of an ideally plastic (non-strain hardening) material where deformations increase with no further increase in applied load. The lower bound load is one in which the material everywhere satisfied equilibrium and nowhere exceeds the defined material yield strength using either a shear theory or a strain energy of distortion theory to relate multiaxial yielding to the uniaxial case.

US= Conventional ultimate strength at temperature or loading which would cause a system malfunction, whichever is more limiting.

EP= Elastic plastic evaluated nominal primary stress. Strain hardening of the material may be used for the actual monotonic stress strain curve at the temperature of loading or any approximation to the actual stress strain curve which everywhere has a lower stress for the same strain as the actual monotonic curve may be used. Either the shear or strain energy of distortion flow rule may be used.

PL= Plastic instability load. The plastic instability load is defined here as the load at which any load bearing section begins to diminish its cross sectional area at a faster rate than the strain hardening can accommodate the loss in area. This type analysis requires a true stress true strain curve or a close approximation based on monotonic loading at the temperature of loading.

TABLE C.3-3 (Cont)

UF= Ultimate load from fracture analyses. For components which involve sharp discontinuities (local theoretical stress concentration > 3) the use of a fracture mechanics analysis where applicable, utilizing measurements of plane strain fracture toughness may be applied to compute fracture loads. Correction for finite plastic zones and thickness effects as well as gross yielding may be necessary. The methods of linear elastic stress analysis may be used in the fracture analysis where its use is clearly conservative or supported by experimental evidence. Examples where fracture mechanics may be applied are for fillet welds or end of fatigue life crack propagation.

LE= Ultimate load or loss of function load as determined from experiment. In using this method, account shall be taken of the dimensional tolerances which may exist between the actual part and the tested part or parts as well as differences which may exist in the ultimate tensile strength of the actual part and the tested parts. The guide to be used in each of these areas is that the experimentally determined load shall use adjusted values to account for material properties and dimension variations, each of which has no greater probability than 0.1 of being exceeded in the actual part.

TABLE C.3-4

BUCKLING STABILITY LIMIT

Any One of (No More than One Required)		General Limit
a.	$\frac{\text{Permissible Load, LP}}{\text{Code Normal Event Permissible Load, PN}}$	$\leq \frac{2.25}{\text{SF min}}$
b.	$\frac{\text{Permissible Load, LP}}{\text{Stability Analysis Load, SL}}$	$\leq \frac{0.674}{\text{SF min}}$
c.	$\frac{\text{Permissible load, LP}}{\text{Ultimate Buckling Collapse Load from Test, SE}}$	$\leq \frac{1.0}{\text{SF min}}$

where:

LP= Permissible load under stated conditions of normal, upset, emergency, or fault.

PN= Applicable code normal event permissible load.

SL= Stability analysis load. The ideal buckling analysis is often sensitive to otherwise minor deviations from ideal geometry and boundary conditions. These affects shall be accounted for in the analysis of the buckling stability loads. Examples of this are ovality in externally pressurized shells or eccentricity of column members.

SE= Ultimate buckling collapse load as determined from experiment. In using this method, account shall be taken of the dimensional tolerances which may exist between the actual part and the tested part. The guide to be used in each of these areas is that the experimentally determined load shall be adjusted to account for material property and dimension variations, each of which has no greater probability than 0.1 of being exceeded in the actual part.

TABLE C.3-5

FATIGUE LIMIT

		General Limit
Summation of mean fatigue (1) damage usage including emergency or fault events with design and operation loads following Miner's hypotheses. either one (not both)	a. Fatigue cycle usage from analysis	≤.056[2]
	b. Fatigue cycle usage from test	≤0.33

NOTES:

1. Fatigue failure is defined here as a 25% area reduction for a load carrying member which is required to function or excess leakage causing loss of function, whichever is more limiting. In the fatigue evaluation, the methods of linear elastic stress analysis may be used when the 3S range limit of ASME III has been met. If 3S is not met, account will be taken of (a) increases in local strain concentration, (b) strain ratcheting, (c) re-distribution of strain due to elastic plastic effects. The January, 1969 draft of the USAS B31.7 Piping Code may be used where applicable or detailed elastic plastic methods may be used. With elastic plastic methods, strain hardening may be used not to exceed in stress for the same strain, the steady state cyclic strain hardening measured in a smooth low cycle fatigue specimen at the average temperature of interest.

2. It is acceptable to use the ASME Section III Design Fatigue curves in conjunction with a cumulative usage factor of 1.0 (using Miner's Hypothesis) in lieu of using the mean fatigue data curves with a limit on fatigue usage of 0.05, since the two methods are approximately equivalent.

TABLE C.3-6

MINIMUM SAFETY FACTOR

Loading Conditions	Loads	P_{60}	SF_{min}
Upset 10^{-1}	N and A_D	10^{-1}	2.25
	or N and U	10^{-1}	2.25
Emergency	N and R	10^{-3}	1.5
	N and A_M	10^{-3}	1.5
	Other combinations in this probability range	$<10^{-3}$ to 10^{-3}	<2.25 to 1.5
Fault	N and A and R	1.5×10^{-6}	1.125
	Other combinations in this probability range	$<10^{-3}$ to 10^{-6}	<1.5 to 1.125

where

N = Normal loads
U = Upset loads (result in maximum system pressure)
A_D = Operating basis Earthquake
A_M = Safe Shutdown Earthquake

R = Loads resulting from jet forces and pressure and temperature transients associated with rupture of a single pipe within the primary containment. This load is considered as indicated in the tables.

The minimum safety factor decreases as the event probability diminishes and if the event is too improbable (incredible: $P_{60}<10^{-6}$) then no safety factor is appropriate or required.

TABLE C.3-7

REACTOR VESSEL INTERNALS AND ASSOCIATED EQUIPMENT

Criteria	Loading	Primary Stress Type	Results: Allowable Stress	Results: Calculated Stress
		STABILIZER BRACKET AND ADJACENT SHELL		
Primary Stress Limit ASME Boiler and Pressure Vessel Code Sect. III defines shear stress limit for SA 302-Gr. B bracket and A-533 Gr. B, C1.1 shell S_M =26.7 Ksi For normal and upset condition shear stress limit=0.6S_M =16.00 Ksi For emergency condition shear stress limit=1.5x16.00=24.00 Ksi For faulted condition shear stress limit=2.0x16.00=32.00 Ksi	Normal and upset condition loads	Pure shear	16.00 Ksi	15.68 Ksi
	1. Operating Basis Earthquake emergency condition load	Pure shear	24.00 Ksi	20.07 Ksi
	1. Safe Shutdown Earthquake faulted condition loads 1. Safe Shutdown Earthquake 2. Jet reaction forces	Pure shear	32.00 Ksi	22.04 Ksi
		VESSEL SUPPORT SKIRT		
Primary Stress Limit ASME Boiler and Pressure Vessel Code, Sect.III, defines stress limit for SA 516 Gr. 70 material For normal and upset condition S_M =19.10 Ksi	Normal and upset condition loads 1. Dead weight 2. Operating Basis Earthquake	General membrane	19.10 Ksi	6.23 Ksi
For emergency condition S_{limit} =1.5S_M =1.5x19.10=28.65 Ksi	Emergency condition loads 1. Dead Weight 2. Safe Shutdown Earthquake	General membrane	28.65 Ksi	9.47 Ksi
For faulted condition S_{limit} =2.0S_M 2.0x19.10=38.2 Ksi	Faulted condition loads 1. Dead Weight 2. Safe shutdown Earthquake 3. Jet reaction forces	General membrane	38.20 Ksi	11.0 Ksi

TABLE C.3-7 (Cont)
REACTOR VESSEL INTERNALS AND ASSOCIATED

Criteria	Loading	Location	Results: Allowable Stress	Results: Calculated Stress
		RPV STABILIZER		
Primary Stress Limit AISC specification for the construction, fabrication, and erection of structural steel for buildings	Upset condition 1. Spring preload 2. Operating Basis Earthquake	Rod	81,600 psi	f_t =49,000 psi*
		Bracket	22,000 psi	f_b =14,300 psi
			14,000 psi	f_v =5,000 psi
For normal and upset conditions AISC allowable stresses, but without the usual increase for earthquake loads	Emergency condition 1. Spring preload 2. Safe Shutdown Earthquake	Bracket	33,000 psi	f_b =18,200 psi
			21,000 psi	f_v =6,300 psi
For emergency conditions 1.5xAISC allowable stresses	Faulted condition 1. Spring preload 2. Safe Shutdown Earthquake 3. Jet reaction load	Bracket	36,000 psi	f_b =20,400 psi
			21,500 psi	f_v =7,000 ps
For faulted conditions material yield strength		*The ratio calculated stress/allowable stresses limit is highest for upset loading conditions.		
		RPV SUPPORT (RING GIRDER)		
Primary Stress Limit AISC specification for the design, fabrication, and erection of structural steel for buildings	Normal and upset condition 1. Dead loads 2. Operating Basis Earthquake 3. Loads due to scram	Top flange	27,000 psi	f_b =10,700 psi
		Bottom flange vessel to girder bolts	27,000 psi	f_b =8,600 psi
			60,000 ps	f_b =21,500 psi
For normal and upset conditions AISC allowable stresses, but without the usual increase for earthquake loads.			22,000 psi	f_b =4,500 psi
For faulted conditions 1.67xAISC allowable stresses for structural steel members yield strength for high strength bolts (vessel to ring girder)	Faulted condition 1. Dead loads 2. Safe Shutdown Earthquake 3. Jet reaction load	Top flange	45,000 psi	f_b =40,500 psi
		Bottom flange	45,000 psi	f_b =32,500 psi
		Vessel to girder bolts	125,000 psi	f_t =79,500 psi
			75,000 psi	f_v =10,000 psi

TABLE C.3-7 (Cont)
REACTOR VESSEL INTERNALS AND ASSOCIATED

Criteria	Loading	Location	Results: Allowable Stress	Results: Calculated Stress
		SHROUDS SUPPORT GUSSETS Primary Stress Type	Allowable Stress Ksi	Calculated Stress Ksi
Primary Stress Limit ASME Boiler and Pressure Vessel Code, Sect. III defines allowable primary membrane stress plus bending for SB-168 material stress	Normal and Upset 1. Operating Basis Earthquake 2. Pressure drop across shroud (normal 3. Subtract dead weight	Local membrane plus bending	34.95 Ksi	22.7 Ksi
For normal and upset condition $S_A = 1.5S_M = 1.5x23.30 = 34.95$ Ksi	Emergency Condition Loads 1. Safe Shutdown Earthquake 2. Pressure drop across 3. Subtract dead weight	Local membrane plus bending	52.43 Ks	44.9 Ksi
For emergency condition $S_{limit} = 1.5S_A = 1.5x34.95 = 52.43$ Ksi	Faulted Condition Loads 1. Safe Shutdown Earthquake 2. Pressure drop across shroud during faulted condition 3. Subtract dead weight	Local membrane plus bending	69.90 Ksi	48.6 Ksi
For faulted condition $S_{limit} = 2.0S_A = 34.95 = 34.95 = 69.90$ Ksi				
		TOP GUIDE-LONGEST BEAM		
Primary Stress Limit The allowable primary membrane stress plus bending stress is based on ASME Boiler and Pressure Vessel Code, Sect. III for type304 stainless steel plate.	Normal and Upset 1. Operating Basis Earthquake 2. Weight of structure 3. Weight of temporary control curtains	General membrane plus bending	24,000 psi	23,790 psi
For normal and upset condition Stress Intensity $S_A = 1.5S_M = 1.5x16,000$ psi - 24,000 psi	Emergency condition loads 1. Safe Shutdown Earthquake 2. Weight of structure 3. Weight of temporary control curtains	General membrane plus bending	36,000 psi	33,100 psi
For emergency condition(N+A) $S_{limit} = 1.5 S_A 1.5x24,000 = 36,000$ psi				
For faulted condition $S_{limit} = 2S_A = 2x24,000 = 48,000$ psi	Faulted Condition Loads (same as emergency condition)	General membrane plus bending	48,000 psi	33,100 psi

=

TABLE C.3-7 (Cont)
REACTOR VESSEL INTERNALS AND ASSOCIATED

Criteria	Loading	Location	Results	
			Allowable Stress	Calculated Stress
		TOP GUIDE-BEAM END CONNECTIONS		
Primary Stress Limit ASME Boiler and Pressure Vessel Code, Sect. III, defines material stress limit for type 304 stainless steel	Normal and Upset Condition Load 1. Operating Basis Earthquake 2. Weight of structure 3. Weight of temporary control curtains	Pure shear	9,600 psi	5,580 psi
	Emergency Condition Loads 1. Safe Shutdown Earthquake 2. Weight of structure 3. Weight of temporary control curtains	Pure shear	14,400 psi	8,800 psi
	Faulted Condition Loads (same as emergency condition)	Pure shear	19,200 psi	8,800 psi
		CORE PLATE STRUCTURE		
Primary Stress Limit The allowable primary membrane stress plus bending stress is based on ASME Boiler and Pressure Vessel Code, Sect. III for type 304 stainless steel plate.	Normal and Upset Condition Loads 1. Normal operation pressure drop 2. Operating Basis Earthquake	General membrane bending	24,000 psi	12,170 psi
For allowable stresses see top guide longest beam, above	Emergency Condition Loads 1. Normal operation pressure drop 2. Safe Shutdown Earthquake	General membrane plus bending	3,600psi	17,740 psi
	Faulted Condition Loads 1. Pressure drop after recirculation line rupture 2. Safe Shutdown Earthquake	General membrane plus bending	48,000 psi	23,140 psi
		CORE PLATE ALIGNERS		
Primary Stress Limit ASME Boiler and Pressure Vessel Code, Sect III, defines material stress limit for type 304stainless steel aligner pin	Normal and upset condition load 1. Operating Basis Earthquake	Pure shear	9,600 psi	0
For allowable shear stresses see top guide beam end connections, above	Emergency condition loadl 1. Safe Shutdown Earthquake	Pure shear	14,400 psi	9,330 psi
	Faulted condition load 1. Safe Shutdown Earthquake	Pure shear	19,200 psi	9,330 psi

TABLE C.3-7 (Cont)
REACTOR VESSEL INTERNALS AND ASSOCIATED

Criteria	Loading	Location	Results	
			Allowable Moment (in.-lb)	Calculated Moment (in.,-lb)
		FUEL CHANNELS		
Primary Stress Limit Allowable stress S_M for Zircaloy determined according to methods recommended by ASME Boiler and Pressure Vessel Code, Sect. III. Allowable moment determined by calculating limit moment using Table C-2, equation (b), then applying SF for emergency loads $(N+A_M)$ S_M =9,270 psi $1.5S_M$ =13,900 psi Emergency Limit load= 1.5xNormal limit load calculated using 1.5 S_M = r yield	Emergency condition load Bending moment resulting from the Safe Shutdown Earthquake	Highest primary bending stress results from maximum moment at mid-span of the channel	42,350	18,900

TABLE C.3-7 (Cont)
REACTOR VESSEL INTERNALS AND ASSOCIATED

Criteria	Loading	Location	Results: Allowable Stress	Results: Calculated Stress
		CONTROL ROD DRIVE HOUSING		
Primary Stress Limit The allowable primary membrane stress is based on the ASME Boiler and Pressure Vessel Code, Sect III, for Class A vessels, for type 304 stainless steel For normal and upset conditions S =15,800 psi @575# °F	Normal and upset condition loads 1. Design Pressure 2. Stuck rod scram loads 3. Operating Basis Earthquake	Maximum membrane stress intensity occurs at the tube to tube weld near the center of the housing for normal, upset, and emergency conditions	15,800 psi	14,480 psi
For emergency conditions $(N+A_M)$ S_{limit}= 1.5S =1.5x15,800 = 23,700 psi	Emergency Condition Loads 1. Design pressure 2. Stuck rod scram loads 3. Safe Shutdown Earthquake		23,700 psi	22,030 psi
		CONTROL ROD DRIVE		
Primary Stress Limit The allowable primary membrane stress plus bending stress is based on ASME Boiler and Pressure Vessel Code, Sect. III for SA-212 TP316 tubing Stress intensity S_A=1.5S_M S_M@575BF=17,375 psi S_A=26,060 psi	Normal and upset condition loads. Maximum hydraulic pressure from the control rod drive supply pump **NOTE**: Accident conditions do not increase this loading. Earthquake loads are negligible.	Maximum stress intensity occurs at a point of the Y-Y axis of the indicator tube.	26,060 psi	20,790 psi
		CONTROL ROD GUIDE TUBE		
Primary Stress Limit The allowable primary membrane stress plus bending stress is based on the ASME Boiler and Pressure Vessel Code, Sect. III for type 304 stainless steel tubing. For normal and upset conditions S_M =15,800 psi @575#F S_A =1.5S_M For faulted condition S_{limit}=2.0S_A= 3.0x15,800=47,400 psi	Faulted Condition Loads 1. Dead weight 2. Pressure drop across guide tube due to failure of recirculation line 3. Safe Shutdown Earthquake	The maximum bending stress under faulted loading conditions occurs at the center of the guide tube	47,400 psi	7,535 psi

TABLE C.3-7 (Cont)
REACTOR VESSEL INTERNALS AND ASSOCIATED

Criteria	Loading	Location	Results	
			Allowable Stress	Calculated Stress
		INCORE HOUSING		
Primary Stress Limit The allowable primary membrane stress is based on the ASME Boiler and Pressure Vessel Code, Sect. III, for Class A vessels for type 304 stainless steel Primary Stress Limit The allowable primary membrane stress plus bending stress is based on ASME boiler and Pressure Vessel Code, Sect. III for SA-212 TP316 tubing For normal and upset conditions S_M=15,800 psi @575°F For emergency condition $(N+A_M)$ S_{limit}=1.5 S_M=1.5 x15,800=23,700	Emergency condition load 1. Design Pressure 2. Safe Shutdown Earthquake	Maximum membrane stress intensity occurs at the outer surface the vessel penetration	23,700 psi	15,290 psi

TABLE C.3-8

RESULTS OF VESSEL FATIGUE AND STRESS ANALYSIS-
SUMMARY OF CRITICAL COMPONENTS

Component	Calculated	$3S_M$ Allowable	Usage Factor (1) (3)
Vessel Shell in Core Region	48.1	80.0	0.038
Closure Studs	77.9	129.9	0.01 (4)
Closure Flanges	37.9	80.0	0.197
Bottom Head-Support Skirt Junction	71.9	80.0	0.391
Shroud Support	120.7 (2)	69.3	0.242 (2)
Feedwater Nozzle	77.0	80.0	0.506
Recirc. Inlet Nozzle Thermal Sleeve	63.0 (2)	47.4	0.477 (2)
CRD Housing to Stub Tube Junction	36.2	47.4	0.795

1. Based on ASME design fatigue curves.
2. These components were justified by a simplified elastic plastic analysis per N-417.6 because the primary plus secondary stress exceeded $3S_M$
3. Environmentally Assisted Fatigue Cumulative Usage Factor (CUF_{EN}) for 60 years, EC12412
4. The Studs do not come in contact with reactor coolant and there is no environmental fatigue component.

TABLE C.3-9

MISCELLANEOUS COMPONENTS

Criteria	Loading	Location	Results: Allowable Stress	Results: Calculated Stress
CRD HOUSING SUPPORT Primary Stress Limit AISC specification for the design, fabrication and erections of structural steel for buildings	Faulted Condition 1. Dead weights 2. Impact force from failure of CRD housing (Earthquake load is negligible)	Beams (top cord) Beams (Bottom cord) Grid Structure	33,000 psi 33,000 psi 33,000 psi 33,000 psi 41,500 psi 27,500psi	F_a = 10,000 F_b = 19,000 F_a = 22,200 F_b = 10,600 F_a = 40,700 F_v = 11,100
For normal and upset condition F_a = 0.60 F_y (Tension) F_b = 0.60 F_y (Bending) F_v = 0.40 S_y (Shear) For faulted conditions F_a limit = 1.5 F_a (Tension) F_b limit = 1.5 F_b (Bending) F_v limit = 1.5 F_v (Shear) F_y = Material yield strength				

TABLE C.3-9

MISCELLANEOUS COMPONENTS

Criteria	Loading	Location	Results: Allowable Stress	Results: Calculated Stress
RECIRCULATING PIPE AND PUMP RESTRAINTS				
Primary Stress Limit Structural Steel: AISC specification for the design, fabrication and erection of structural steel for buildings For normal or upset conditions $F_a = 0.60\ F_y$ (Tension) For faulted conditions F_a limit = 1.5 F_a (Tension) F_y = yield strength Cable (wire rope) For faulted conditions $F_a = 0.80\ F_u$ (Tension) F_u = Ultimate Strength	Faulted Condition 1. Jet reaction force from a complete circumferential failure (break) of recirculation line.	Brackets on 28 in. pipe Cable on pump restraints	33,000psi 99,000 psi	29,000 psi 79,200 psi

TABLE C.3-9

MISCELLANEOUS COMPONENTS

Criteria	Loading	Location	Results: Allowable Stress	Results: Calculated Stress
FUEL STORAGE RACKS				
Primary Stress Limits ASME Code, Section III Normal or Upset (1) Subsection NF, 1980 Edition	Normal or Upset[1]			
For Normal or Upset Condition $F_t = 0.6S_y$ (Tension) $F_v = 0.4S_y$ (Shear) $F_b = 0.6S_y$ (Bending) $F_a = F(x)S_y$ (Compression)[2]	1. Dead Loads 2. Live Loads developed during lifting 3. Differential temperature induced loads. 4. Operating Basis Earthquake Loads. 5. Loads caused by the upward and sideways forces of a postulated stuck fuel assemble	Entire Structure	F_t = 15,000psi F_v = 1000psi F_b = 15,000psi F_a = 15,000psi	(See Ref. 1 and 2)
For the faulted condition the stress limits are1.2 (Sy/Ft) times the limits for the normal or upset condition	Faulted Condition[1] 1. Dead Loads 2. Live Loads developed during lifting 3. Differential temperature induced loads. 4. Safe shutdown Earthquake Loads	Entire Structure	F_t=30,000psi F_v=20,000psi F_b=30,000psi F_a=30,000psi	(See Ref. 1 and 2)
For the accidental drop condition[3]	1. Loads caused by the accidental drop of the heaviest load from the maximum possible height	Baseplate Top of Rack	The functional capability of racks should be demonstrated	(See Ref. 1 and 2)

TABLE C.3-9

MISCELLANEOUS COMPONENTS

Results of Fuel Rack Analysis

The results of the stress analysis in References 1 and 2 indicate that for both the normal/upset condition and the faulted condition the calculated stresses were well below the allowable ASME stress limits. The results of the fuel drop accident indicates that there would be some local yielding and possibly some permanent deformation. However, the racks will still maintain their functionality to prevent criticality of the adjacent racks and maintain their cross sectional geometry at that level of active fuel. The results of the drop accident calculation also indicates that the fuel pool liner stresses caused by the drop accident would be below the stress levels caused by the safe shutdown earthquake condition.

Notes

1. All loads are not considered to act simultaneously but in various combinations. See Reference 1 for loading combinations.
2. f(x) is a function of various parameters. Refer to Reference 1.
3. The accidental drop condition is not required by ASME.

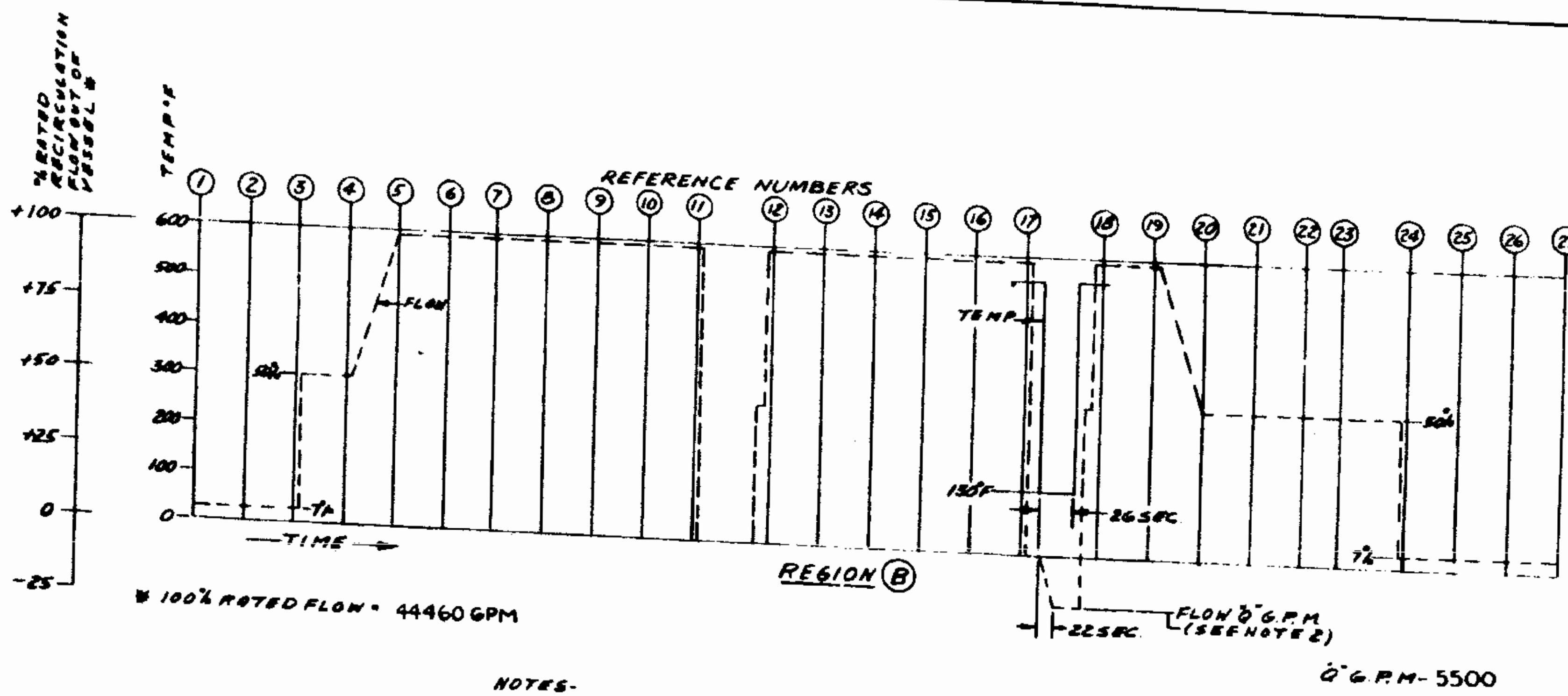
References

1. Joseph Oat Co., Seismic Analysis of High Density Fuel Racks for Pilgrim Nuclear Power Station Unit 1, Report TM-729.

2. Holtec International, "PNPS Spent Fuel Capacity Expansion", Licensing Report #HI92925, January 5, 1993 (SUDDS/RF#93-01).

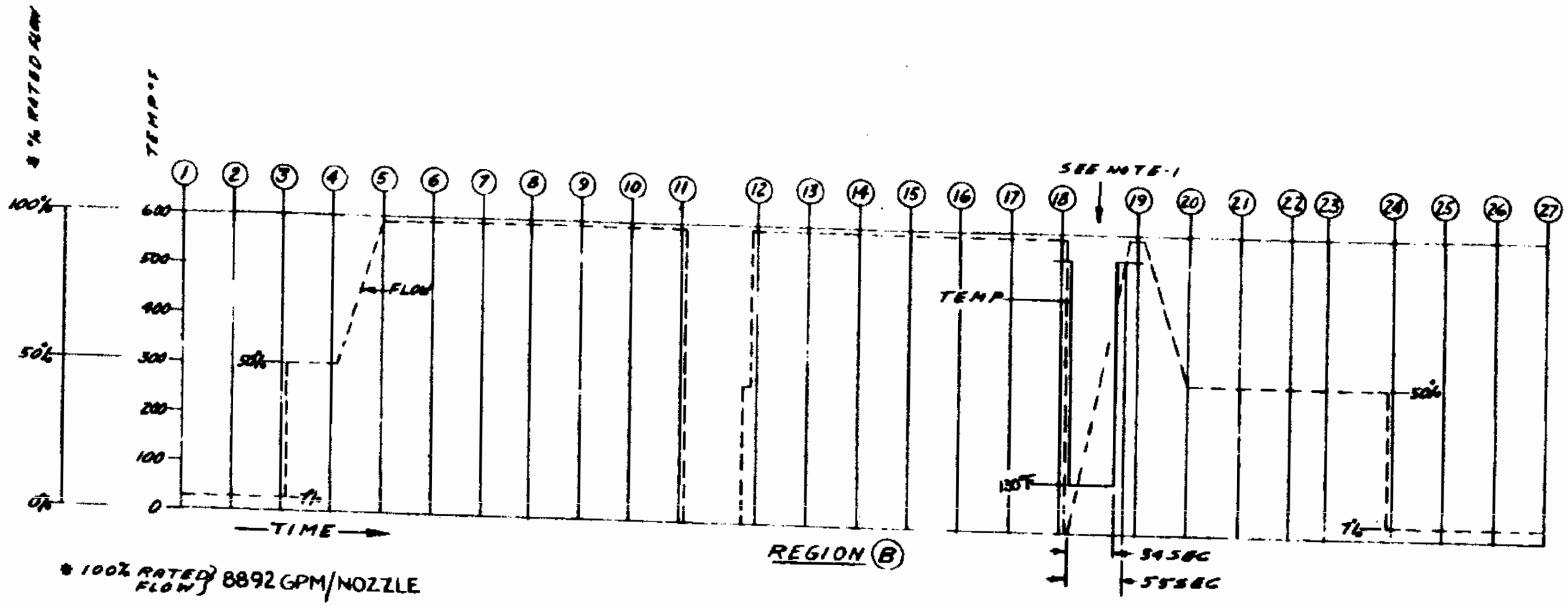
Figure C.3-1 has been removed.

Please refer to BECo Controlled Drawing M1A12-2.



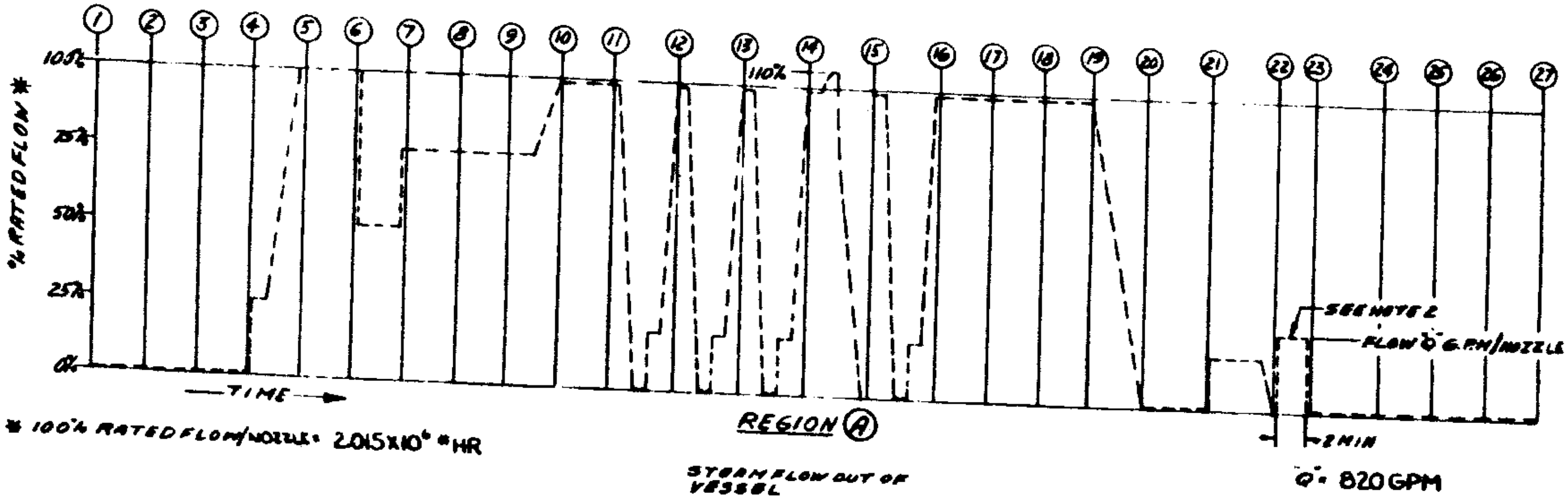
NOTES-
1. TEMPERATURES WILL BE REGION "B" TEMPERATURES EXCEPT AS SHOWN.
2. "Q" GPM FLOWS INTO VESSEL THRU ONE RECIRCULATION OUTLET NOZZLE WHILE THE OTHER RECIRC OUTLET NOZZLE HAS NORMAL OUTFLOW.
3. FOR DEFINITIONS OF REFERENCE NUMBERS & REGIONS FOR ALL SHEETS THIS DWG., SEE REACTOR THERMAL CYCLE DWG.

FIGURE C.3-2
NOZZLE THERMAL CYCLES-
RECIRCULATION OUTLET
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT



NOTES
1- TEMPERATURE OF FLUID ENTERING THE VESSEL THROUGH NOZZLE IS REGION (B) FLUID TEMPERATURE, EXCEPT AS SHOWN.

FIGURE C. 3-3
NOZZLE THERMAL CYCLES-RECIRCULATION INLET
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

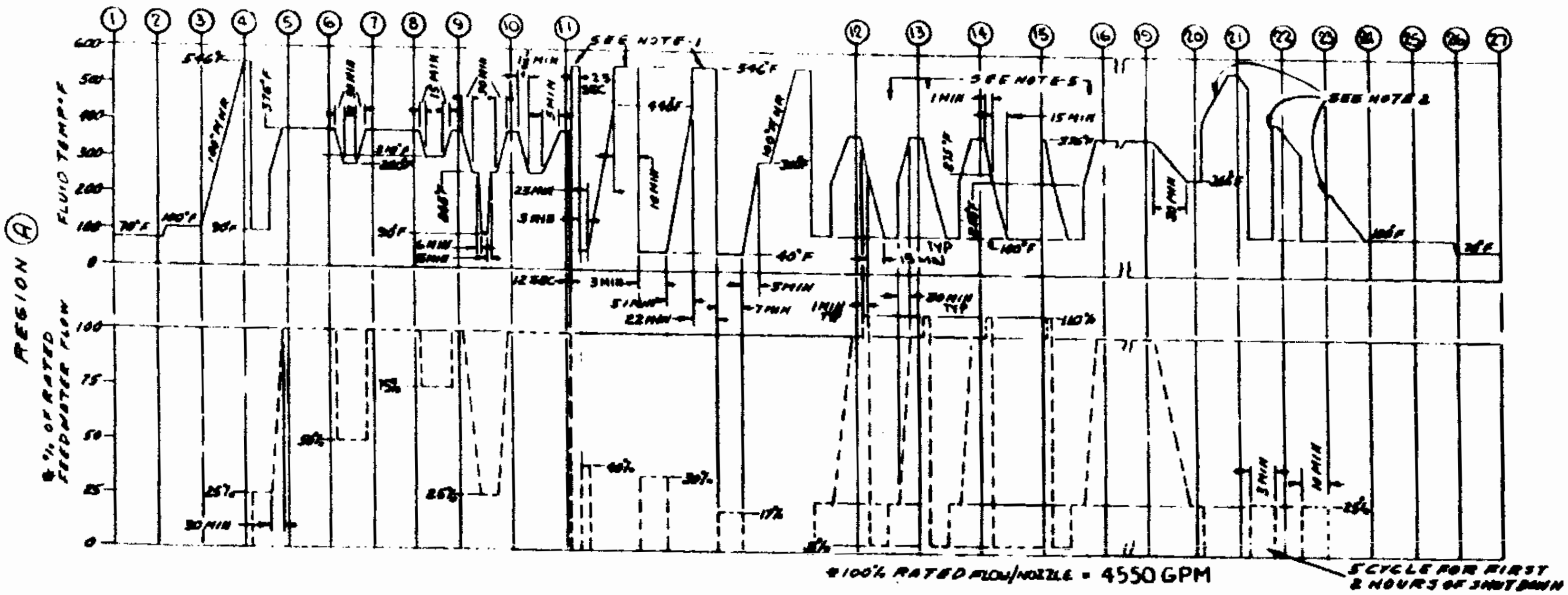


* 100% RATED FLOW/NOZZLE = 2.015X10⁶ #HR

NOTES
1. TEMPERATURES OF FLUID IN STEAM OUTLET NOZZLE ARE IDENTICAL WITH FLUID TEMPERATURES REGION "A". SEE REACTOR THERMAL CYCLE DWG.
2. WATER FLOWS FROM VESSEL INTO STEAM LINE UNTIL STEAM LINE IS FULL DURING VESSEL FLOODING

FIGURE C. 3-4
NOZZLE THERMAL CYCLES-
STEAM OUTLET

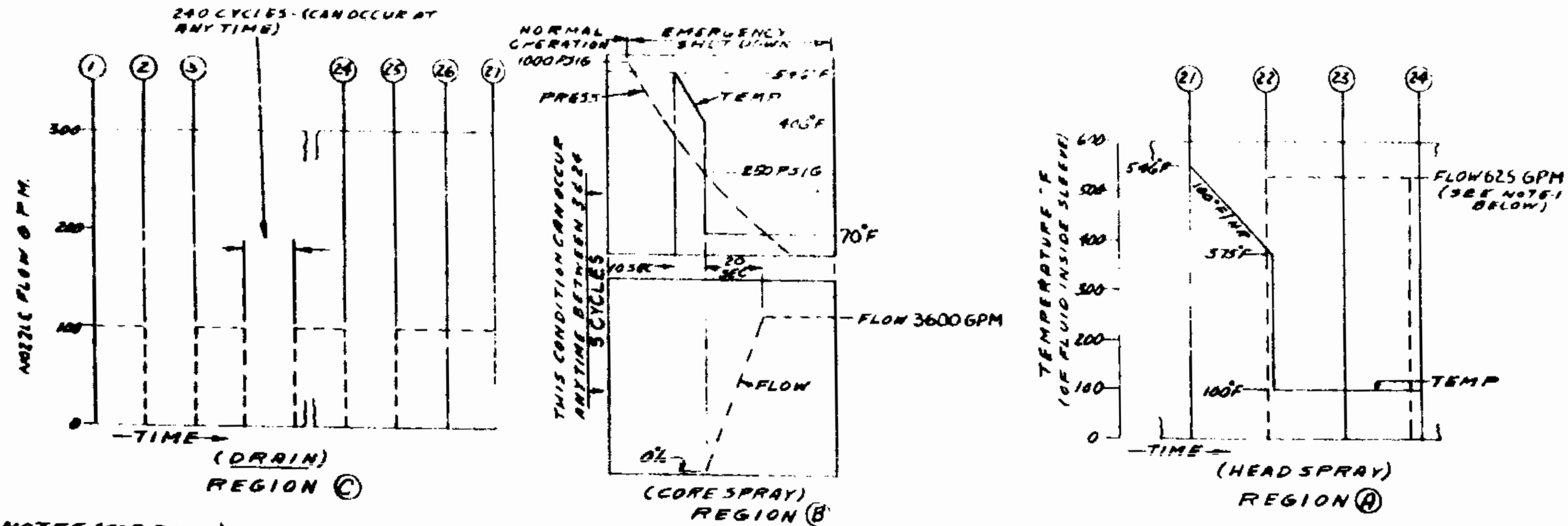
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT



NOTES:

1. DURING THESE TIMES THE NOZZLE IS FILLED WITH STEAM AT REGION "A" TEMPERATURE.
2. DURING THESE TIMES THE WATER IN THE NOZZLE REACHES REGION "A" TEMPERATURES.
3. NUMBER OF CYCLES CORRESPOND TO THAT GIVEN ON REACTOR VESSEL THERMAL CYCLE
4. FLOW CURVES SHOW RECOVERY AFTER CLEARING CONDITION THAT CAUSED SCRAM.
5. TIMES & TEMPERATURES SHOWN IN CYCLE 12 TO 13 ARE TYPICAL FOR CYCLES 13 TO 14 & 15 TO 16

FIGURE C.3-5
NOZZLE THERMAL CYCLES-FEEDWATER
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT



NOTES (FOR DRAIN)

1. DRAIN NOZZLE FLOW IS OUT OF THE VESSEL FOR ALL CONDITIONS SHOWING FLOW
2. TEMPERATURES WILL BE REGION "C" FLUID TEMPERATURES WHEN FLOWING

NOTES (FOR CORE SPRAY)

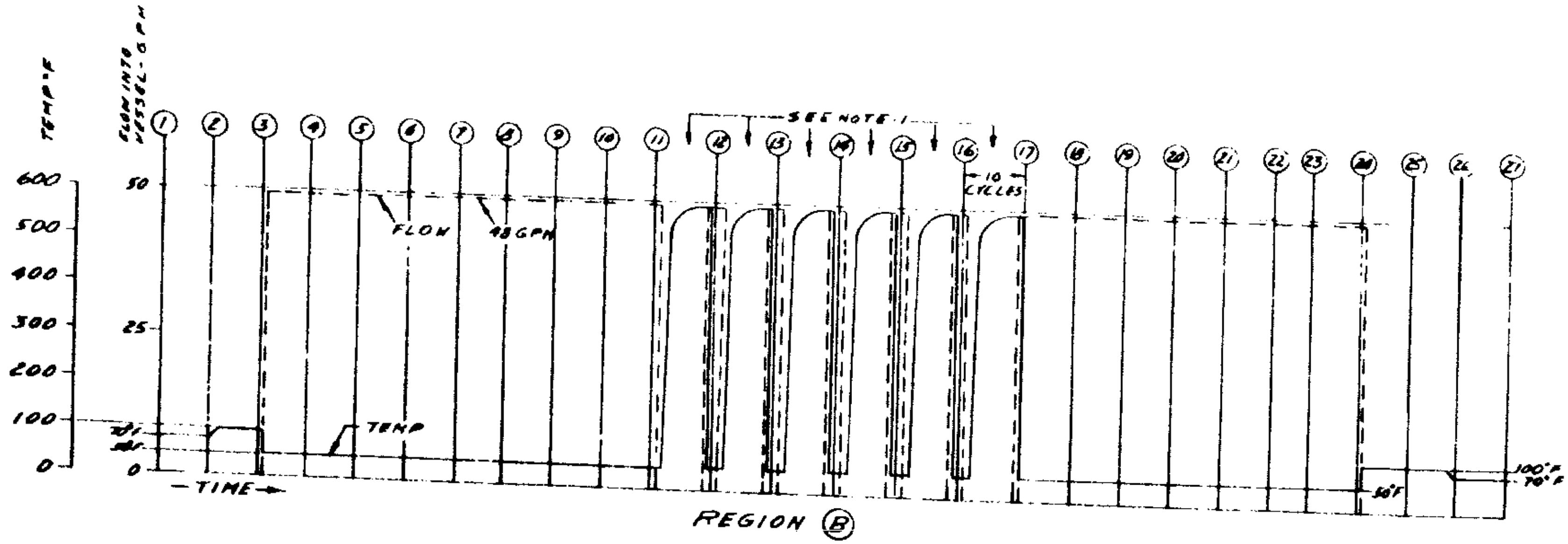
1. AT ALL OTHER TIMES THERE IS NO FLOW IN NOZZLE AND TEMPERATURE IS AS GIVEN AS REGION "B"
2. FLOW IS INITATED AFTER WATER LEVEL DROPS BELOW LOW WATER LEVEL-2 AND PRESSURE IS LESS THAN 250 PSIG DURING EMERGENCY SHUT DOWN REGION "B" AND ANNULUS BETWEEN THE NOZZLE & THERMAL SLEEVE CONTAIN SATURATED STEAM

NOTES (FOR HEAD SPRAY)

1. THIS FLOW IS THRU INNER PIPE SLEEVE ONLY.
2. AT TIMES OTHER THAN SHOWN ABOVE, THERE IS NO FLOW IN NOZZLE AND TEMP. IS AS GIVEN FOR REGION "A"

FIGURE C.3-6
NOZZLE THERMAL CYCLES-DRAIN (CORE SPRAY AND HEAD SPRAY)

PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT



NOTES
1. FLOW IS OFF DURING EACH SCRAM CYCLE AND THE TIME THE FLOW IS OFF IS SUFFICIENT FOR NOZZLE TO COME TO THERMAL EQUILIBRIUM WITH REGION "B"

FIGURE C.3-7
NOZZLE THERMAL CYCLES
CONTROL ROD DRIVE
HYDRAULIC SYSTEM RETURN
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

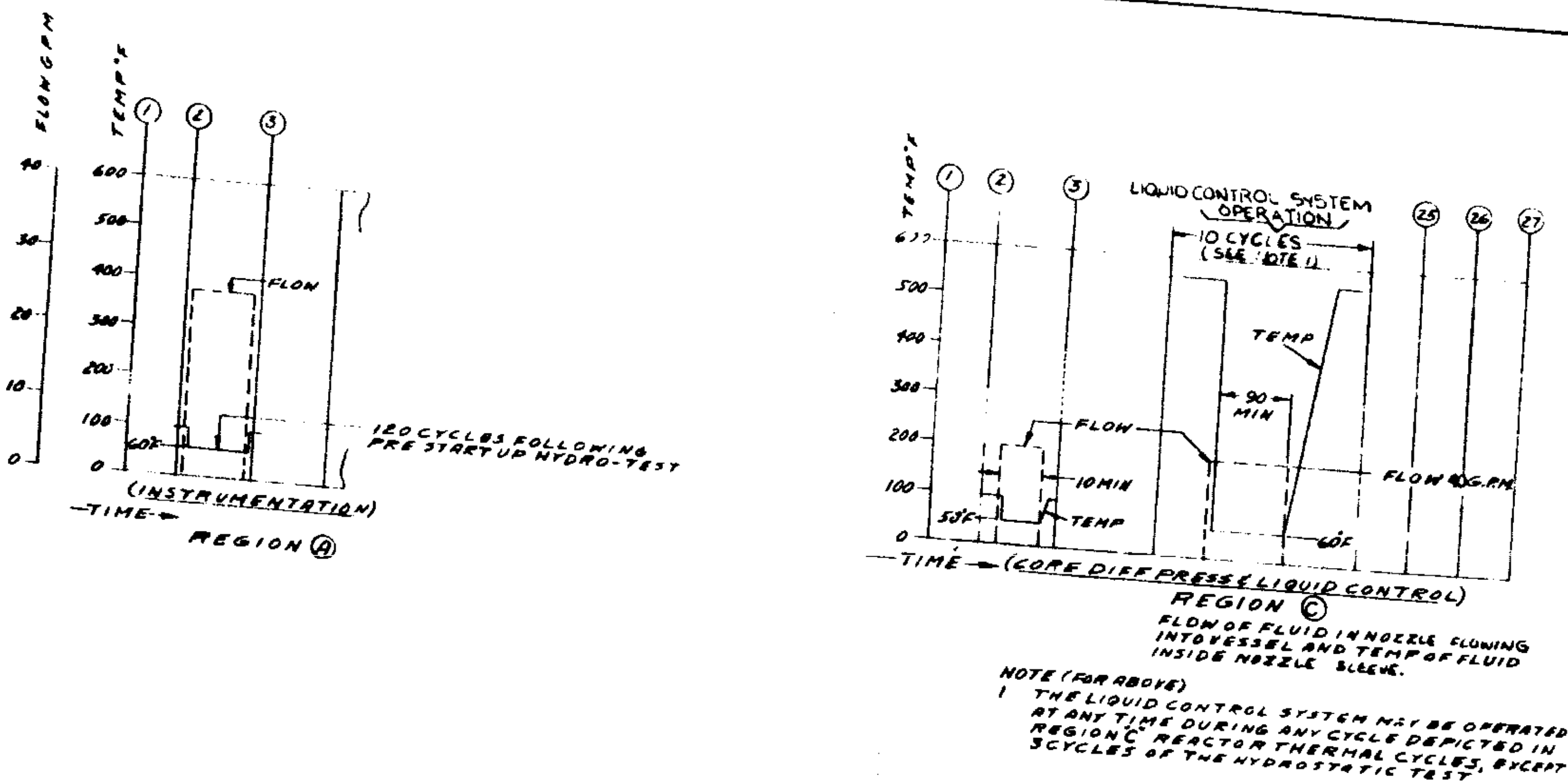


FIGURE C.3-8
NOZZLE THERMAL CYCLES
INSTRUMENTATION AND CORE
DIFFERENTIAL PRESSURE
AND LIQUID CONTROL
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

APPENDIX D

QUALITY ASSURANCE PROGRAM

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QUALITY ASSURANCE PROGRAM
PILGRIM NUCLEAR POWER STATION

D.1 GENERAL

Construction Quality Assurance Program

The Quality Assurance Program set forth in Sections D.1 to D.6 and Attachments D.I to D.III of the Appendix represents the program implemented for the construction stages of the Pilgrim Nuclear Power Station located at Plymouth, Massachusetts. This Program remains intact in the Updated FSAR in order that the NRC may locate previously submitted information in one document.

This requirement to maintain a historical record for the Construction Quality Assurance Program is outlined in question/responses concerning the FSAR Update Rule (10CFR50.71(e) in NRC letter of December 15, 1980.

Operation Quality Assurance Program

The present Quality Assurance Program for Operation of Pilgrim Nuclear Power Station was initially submitted on August 30, 1974. The Boston Edison Operating Assurance Manual for Pilgrim was subsequently revised to (1) ANSI N18.7, draft No. 5 to Revision 1 dated February 12, 1975, and (2) The NRC Standard Review Plan (Operational Phase).

The Manual revision was accomplished as a redescription of the Quality Assurance Program for Operation of Nuclear Power Plants as shown in Enclosure (1) and (2) to Boston Edison letter to the NRC dated August 7, 1975. The Program was further amended for pages 12, 15, 16, 13, 24, 28, and 37 and submitted to the NRC by Boston Edison letter dated October 7, 1975.

The Quality Assurance Program thus developed has the necessary controls to comply with the requirements of Appendix B to 10CFR50 and has been accepted by the NRC for the operations phase of Pilgrim Nuclear Power Station as stated in NRC letter dated November 12, 1975.

Since 1975, Boston Edison has been sensitive to NRC regulations, Regulatory Guide, ANSI Standards, industry information, and its own commitments to excellence and safety. The Boston Edison Quality Assurance Program has evolved accordingly.

The Boston Edison Quality Assurance Program for operation of nuclear power plants is defined in Volume II of the Boston Edison Quality Assurance Manual. It is the governing document for quality related activities of the Boston Edison Company relating thereto. This document has continually reflected the evolution of the Boston Edison Quality Assurance Program, and has been approved pursuant to

10CFR50.54(a) as stated in NRC letter 1.84.034 dated February 7, 1984. All subsequent changes to the NRC approved QA program description shall be submitted to the NRC for approval according to the requirements of 10CFR50.54(a)(3).

D.1.1 Introduction and Summary

Pilgrim Nuclear Power Station is owned and will be operated by Boston Edison Company at Plymouth, Massachusetts. As owner and operator, the Company assumes full responsibility and authority for the project and will take appropriate action to assure that the plant is designed, constructed, and operated in accordance with sound engineering principles and safe operating practices.

The Company's Quality Assurance Organization and the Nuclear Engineering Department are responsible for assuring that systems and structures affecting the safety and integrity of the station are specified, fabricated, inspected, shipped, stored, installed, and tested in accordance with sound engineering principles, appropriate codes, specifications, and procedures.

Independent programs in Quality Control and Quality Assurance have been established by the Architect-Engineer and by the Nuclear Steam System Supplier. Control of quality is accomplished through communications with project contractors, consultants, inspectors, and equipment manufacturers, and by surveillance and audit of quality control records, inspection reports and audits, test reports, certifications of code compliance, and by any other means necessary to accomplish the quality assurance objectives.

Functional responsibility for quality control, quality assurance, engineering design, procurement, expediting, inspection, construction, systems checkout, and acceptance testing are indicated on Table D.1-1, Project Functional Responsibility Summary. This table covers the nuclear steam supply system, turbine generator equipment, nuclear fuel, and balance of plant.

D.1.2 Program Purpose and Objectives

The Pilgrim Nuclear Power Station Project quality assurance and quality control requirements are established to meet the following objectives:

1. Minimize delays in the scheduled completion dates for construction, preoperational rise to power, or warranty tests due to the failure of structures or equipment, or to system deficiencies, or due to failure to meet the applicable requirements of federal, state, and local governmental agencies

2. There shall be reasonable expectation of achieving a full power station availability factor of 90 percent or better during the first 12 months of commercial operation will be achieved

3. There shall be reasonable expectation of achieving a full power station availability factor, excluding refueling time, of 95 percent in succeeding years

There shall be assurance at the end of project that essential items of equipment, systems, and structures that comprise Pilgrim Station conform with the specifications and criteria established for the station and that they are capable of operation at full power without undue risk to health and safety of the general public or to the station operating personnel.

D.1.3 Definitions

Definitions of terms, as used in the Boston Edison Company Quality Assurance Program, are as follows:

Quality Assurance is the functional activity of providing surveillance and documentation to assure that appropriate quality control measures have been specified and properly performed. The surveillance phase checks the adequacy of the quality control function for performance.

Quality Control encompasses specific test and inspection activities conducted by the organization having first line responsibility for the quality of the product. The QC activities determine conformance with approved specifications and drawings. The first line responsibility may shift with time on a particular piece of equipment where the Nuclear Steam System supplier is responsible for quality of process specifications, detailed design by the Architect-Engineer; purchase specifications on performance, materials, inspection, codes and standards, cleaning, testing, etc., by the Architect-Engineer; shop drawings and manufacture by a vendor; and receiving and storing by the field construction forces. The same quality control organization of one company may also be the quality assurance organization over his subvendors' products.

Reactor Primary System Pressure Boundary encompasses the Reactor Coolant System (RCS), including the reactor vessel, its associated control rod housings, and all other pressure retaining components and piping which are subject to RCS conditions during reactor operation (including startup, hot standby, and shutdown). The RCS boundary extends to and includes the second containment isolation valve in the main steam and feedwater lines. It also includes the portions of associated auxiliary systems connected to the RCS, extending to and including the second containment isolation valve normally open or closed during reactor operation or the second block valve normally closed during reactor operation.

Essential items are those systems, components, and structures which are required to maintain the integrity of the primary system pressure boundary and engineered safeguards systems

Classification of Structures, Systems, and Components is the listing of items classified to establish the base for quality level identification, quality specification, and determination of quality documentation requirements.

D.1.4 Classification of Structures, Systems, and Components

Structures, systems, and components are classified for purposes of identifying and assigning quality control and quality assurance requirements. The classification is intended to establish quality levels consistent with the requirements of Boston Edison and to comply with the intent of the quality related AEC General Design Criteria.

TABLE D.1-1

PROJECT FUNCTIONAL RESPONSIBILITY SUMMARY
PILGRIM NUCLEAR POWER STATION

Project Part	Quality Control	Quality Assurance	Engineering Design	Procurement	Mfg Inspection	Mfg Expediting	Construction QA	Construction QC	Systems Checkout and Acceptance Testing
I Nuclear Power Station*	B	Entergy	B	B	B	B	B	B	Entergy(1)
II Nuclear Stream Supply System	GE	Entergy	GE	GE	GE, B	GE, B	B(2)	B(2)	Entergy(2)
III Turbine Generator Equipment	GE	Entergy	GE	GE	GE, B	GE, B	B(2)	B(2)	Entergy(2)
IV Nuclear Fuel	GE	Entergy	GE	GE	GE Entergy	GE	--	--	Entergy(2)
V Recirc., RHR RWCU, Core Spray Piping Replacement (RFO #6)	GE Entergy	GE	GE	GE	GE Entergy	GE	GE	GE Entergy	Entergy(2)

NOTES:

Entergy = Entergy Corporation
B = Bechtel Corporation
GE = General Electric Company
*Except Parts II, III, and IV

(1) Technical direction provided by Bechtel Corp.

(2) Technical direction provided by GE Corp.

D.2 PROGRAM ORGANIZATION AND RESPONSIBILITY

D.2.1 Prime Responsibility and Assignments

Pilgrim Nuclear Power Station is owned and will be operated by Boston Edison Company. The Company assumes full responsibility and authority for the project and has ultimate responsibility for quality assurance.

Bechtel Corporation is Architect-Engineer with additional responsibility for all construction on site.

General Electric Company provides design engineering and procurement of equipment associated with the Nuclear Steam Supply System (NSSS), together with technical direction of installation for this system.

Consultants' services have been procured for assistance in quality assurance, meteorology, marine and ecological studies, and inservice inspection studies. Technical consultants will be employed in areas where expertise is deemed desirable or necessary.

Figure D.2-1, Quality Assurance - Quality Control Responsibility Chart, presents a pyramid of organizational quality assurance responsibility covering materials and equipment subvendors, upward through various contractors, suppliers, Architect-Engineer, Constructor, and ultimately to the prime responsible company, Boston Edison.

Figure D.2-2, Site Construction Quality Assurance Organization Chart, presents the organization and interrelationships of quality oriented personnel at the Pilgrim construction site. Functional lines of reporting and communication are established to coordinate quality control and quality assurance, and to assure that all site construction activities are carried out in accordance with project requirements for inspection, testing, and documentation.

D.2.1.1 Applicant - Boston Edison Company

Boston Edison assumes responsibility for the project and takes appropriate action necessary to assure that the plant is constructed in accordance with applicable codes, specifications, and procedures, and is operated in a safe and reliable manner.

The Nuclear Quality Assurance Organization (NQAO) has the responsibility to assure that quality requirements of essential items affecting the integrity and safety of the station are identified, specified, and performed in accordance with sound engineering principles and appropriate codes and standards. This NQAO consists of a Senior Quality Assurance Engineer who reports to the Assistant Vice President - Nuclear. Reporting to the Senior Quality Assurance Engineer are the Quality Assurance Engineers assigned with mechanical and electrical experience and background. Items pertaining to Quality Assurance are reviewed by the staff and the Senior Quality Assurance Engineer.

It is the prime responsibility of the NQAO to verify that requirements and procedures are executed as specified. This is accomplished by surveillance over the Architect-Engineer, the Nuclear Steam System supplier, contractors, inspectors, and equipment manufacturers. A permanent onsite Quality Assurance Group consisting of experienced graduate engineers and technicians is maintained for construction quality assurance surveillance and auditing.

Emphasis is placed on spot visits to manufacturers' fabrication shops by Boston Edison Quality Assurance personnel to examine product quality, to audit required documentation, and to ascertain that manufacturers' Quality Control Programs are functioning properly.

Company organizations supporting the quality assurance effort for Pilgrim Project include the Engineering and Construction Department, Boston Edison Laboratory, Operations Department, Resident Engineer and Staff, Independent Consultants, and Pilgrim Station Operating Staff.

D.2.1.1.1 Engineering and Construction Department

Responsibility for review of principal specifications, drawings, and procedures is centered in the Nuclear Project Group. Assisting this group are engineering personnel within the Mechanical and Structural Design Division, the Station Electrical Design Division, and the Protection Division.

Technical review assistance is also provided by the Boston Edison Laboratory facility and personnel, specifically from its Mechanical and Chemical Division, Laboratory Electrical Division, Systems Electrical Division, and its Electronics Division, as requested.

Feedback of experience from engineering personnel is factored into the design review, and comments are presented to the Architect-Engineer.

D.2.1.1.2 Operations Department

A parallel design review effort is made by Boston Edison Operations Department engineers and staff who contribute operational and maintenance experience feedback. The Mechanical Technical Division, the Electrical Technical Division, and the Department's Staff Engineers contribute to the review of machine and equipment layout drawings; valve, piping, and equipment accessibility; control board layout and operational safety requirements.

D.2.1.1.3 Site Construction Organization

The Company has assigned a Resident Engineer to the Pilgrim Station Site. It is the responsibility of the Resident Engineer to monitor the onsite construction activities of the prime construction contractor and subcontractors. He is a construction division head and has the authority to stop any work that he deems improper.

An onsite Inspection Staff reports directly to the Boston Edison Resident Engineer. These inspectors provide liaison and surveillance over construction activities for Boston Edison.

D.2.1.1.4 Preliminary Operations and Startup Testing

Each system, subloop, and operating component will be appropriately tested insofar as practical by Edison startup and operating personnel during the test period preceding fuel loading. This phase of projects is coordinated by Boston Edison's Startup Engineer, with technical direction from Bechtel and GE's Startup Engineers on site. The Pilgrim Division personnel prepare written preoperational and acceptance test procedures and participate in testing. Procedures are established for reporting deficiencies and follow-up corrective action. During this same period, Boston Edison personnel perform a checkout of instrumentation and controls and electrical systems.

GE assists Edison in training and familiarizing personnel with nuclear instrumentation and computers.

D.2.1.2 Architect-Engineer, Bechtel Corporation

Bechtel Corporation is the Architect-Engineer for Pilgrim Station, with responsibility for all construction on site. This corporation has been assigned by Boston Edison to perform Quality Assurance functions on designated GE furnished items included in the Nuclear Steam Supply System, the main turbine generator, and station computer.

Bechtel has three organizational functions relating to Quality Assurance and Quality Control; namely, project engineering, procurement, and construction. Coordination and implementation of the Bechtel Quality Assurance Program is the responsibility of the Project Engineer. An organization chart for the Project with regard to Bechtel Quality Assurance related elements is shown in Figure D.2-3. This chart illustrates the Project interrelationships of Bechtel management, departments, and outside quality control testing laboratories.

Attachment D.II, Bechtel Quality Assurance Program, summarizes methods for control and assurance of quality within their scope of Project.

D.2.1.3 Nuclear Steam System Supplier - General Electric Company

GE is the Nuclear Steam System supplier having responsibility within their scope of Project for conceptual and engineering design, specifications, procurement, and components manufacture. Technical direction of installation and startup testing are accomplished by GE for items within the NSSS package.
GE's Quality Assurance Program, which is shown as Attachment D.I, defines in summary form the approach to their overall quality program for supplying nuclear systems and components which will operate in a

safe and reliable manner. It describes organization structure and responsibility assignments, design control, quality assurance, and quality control methods for manufactured and procured products, and the GE activities relative to installation control, preoperational testing, and plant startup.

D.2.1.4 Consultants

The services of specialist consultants have been procured from time to time for guidance in areas of quality assurance, ecology, meteorology, and inservice inspection. When such consulting work is within the scope of Project controlled by Bechtel and GE, then their quality control and assurance programs apply.

Consultants who may be hired directly by Boston Edison will be guided and controlled by the Nuclear Engineering Department or by Edison's Quality Assurance Organization.

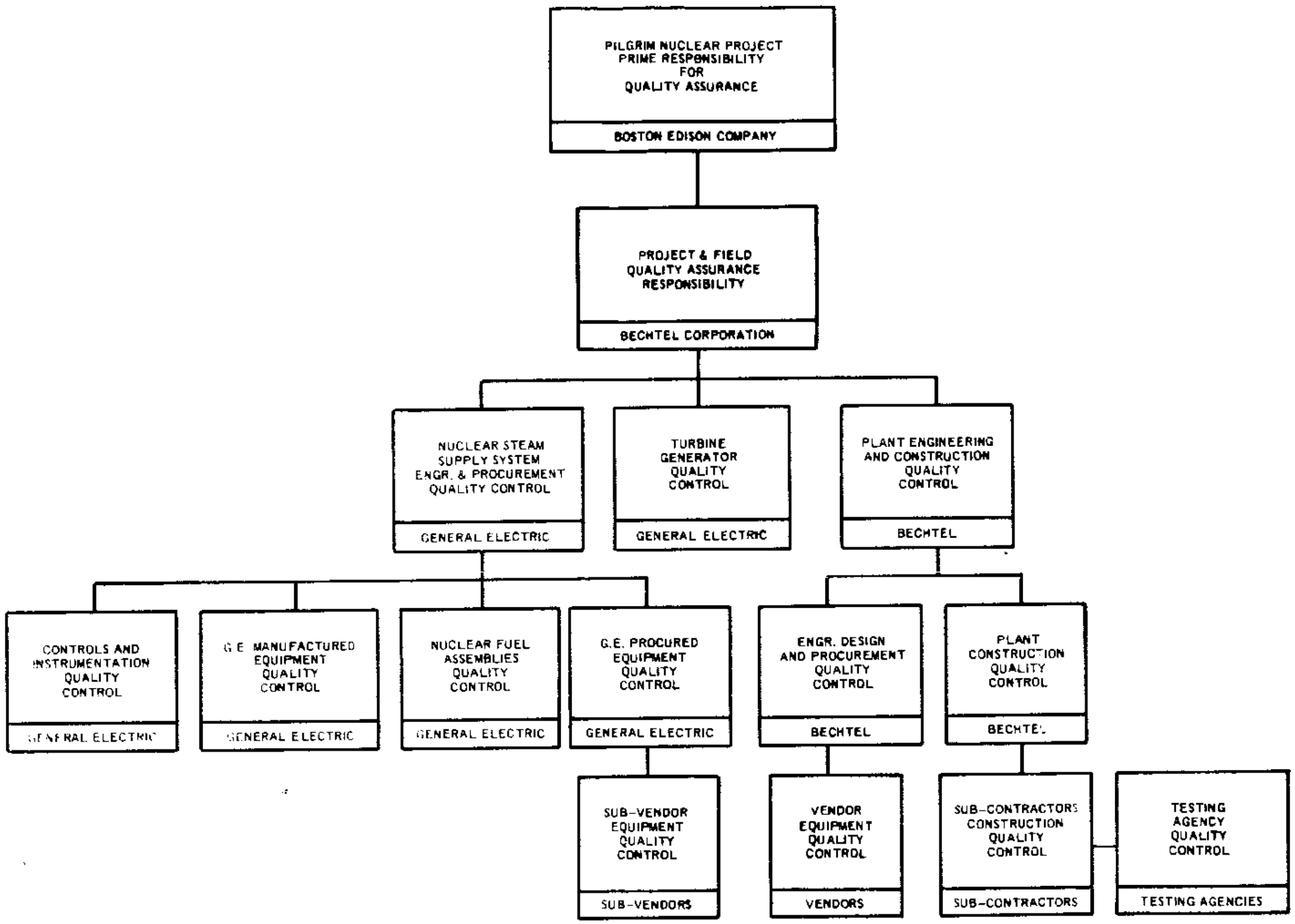

FIGURE D.2-1
QUALITY ASSURANCE-
QUALITY CONTROL
RESPONSIBILITY CHART
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

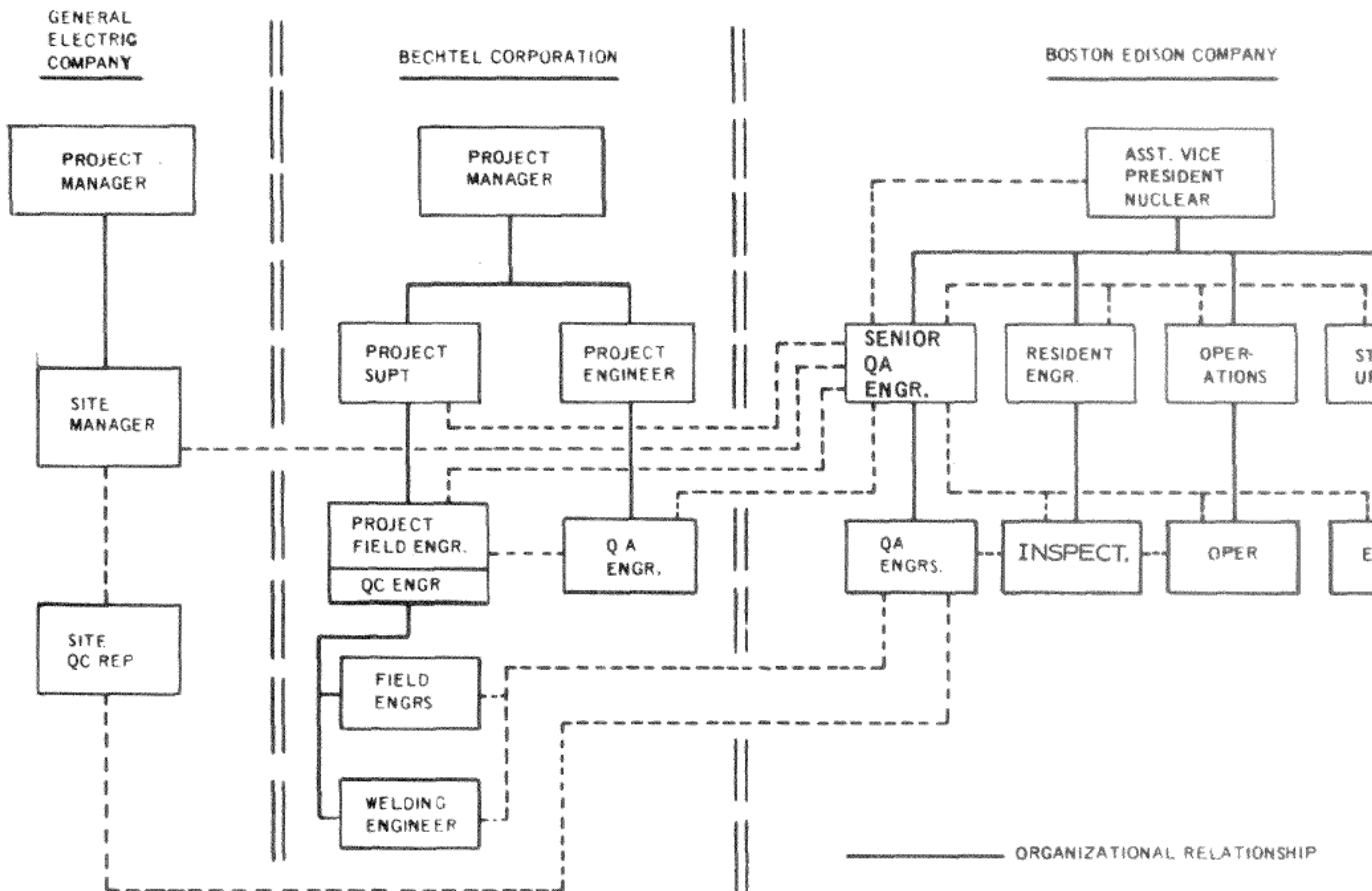

FIGURE D. 2-2
SITE CONSTRUCTION QUALITY ASSURANCE ORGANIZATION CHART
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

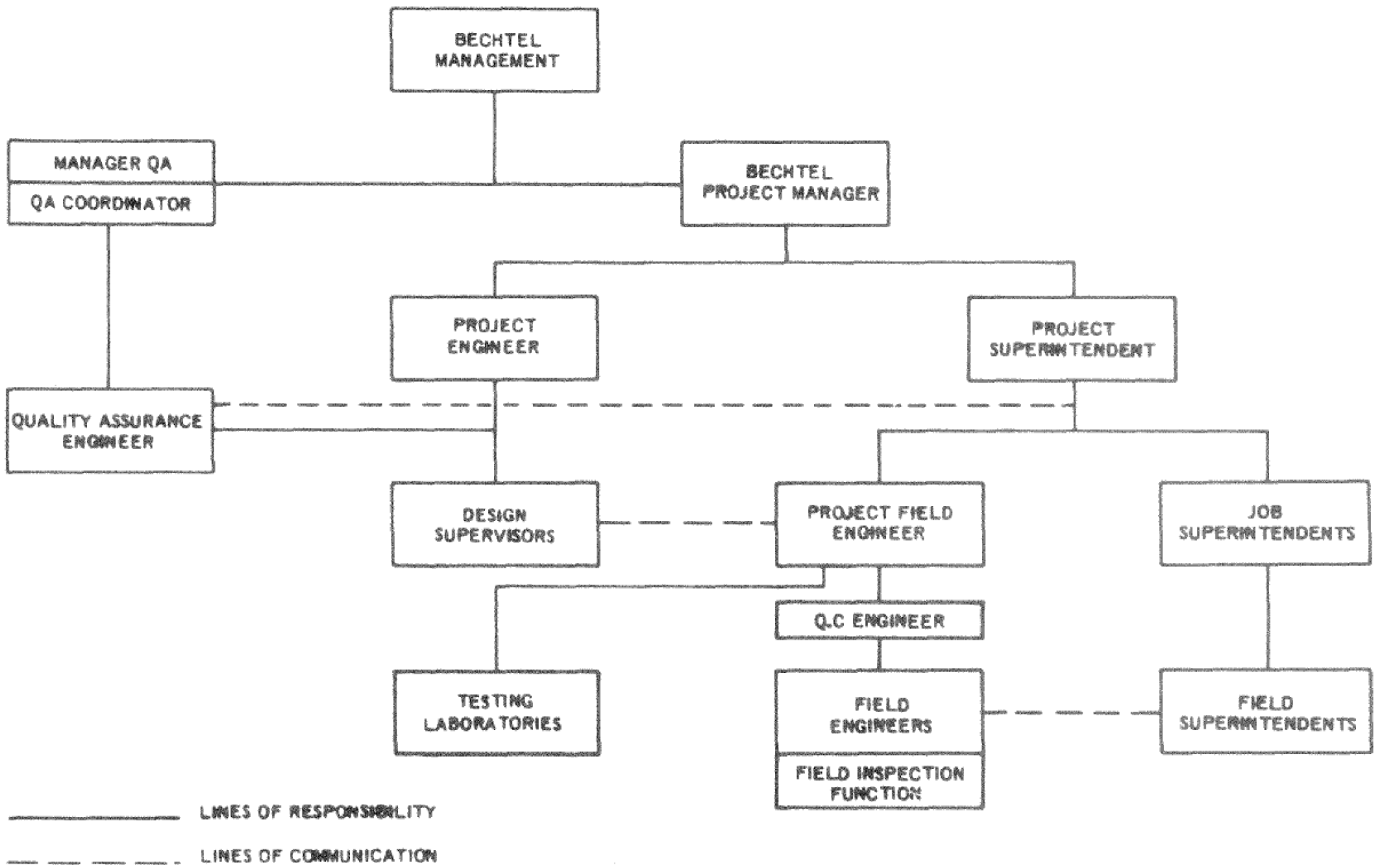

FIGURE D.2-3
QUALITY ASSURANCE RESPONSIBILITY AND LINES OF COMMUNICATION
BECHTEL CORPORATION
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

D.3 QUALITY SURVEILLANCE AND AUDITING

D.3.1 Bechtel Corporation

Bechtel, Architect-Engineer and Constructor for Pilgrim Nuclear Power Station, provides quality measures in areas of conceptual and engineering design, procurement, inspection, installation and erection, testing, and maintenance prior to acceptance of systems and structures by Boston Edison. Quality related reference documents are Bechtel Pilgrim Project Nuclear Quality Assurance Manual, Bechtel Field Inspection Manual, and Bechtel Procurement Department Inspection Manual.

D.3.1.1 Engineering Design Review

Engineering design review is accomplished by Bechtel to assure the compatibility of materials and equipment to meet systems requirements by checking cost, safety features, maintainability, vendors' reputation for performance, proven design, and adequate capacity.

Equipment location plans and arrangement layout drawings are reviewed for compliance with safety and fire protection codes, personnel protection, accessibility, and maintainability.

Design specifications and requirement drawings are reviewed for adequate definition of components and systems, criteria for adequate performance, description of operating conditions, descriptions of maintenance requirements, definition of fabrication checkpoints, description of fabrication procedure requirements, qualification test and sample requirements, and requirements for cleanliness and handling.

D.3.1.2 Procurement Control

In selecting bidding lists, the qualifications of potential vendors are examined for reliability, for past performance on promised delivery dates, and past response to purchasers' overhaul and maintenance problems.

Quality is maintained during the procurement and fabrication phase of the project by Bechtel through its Procurement Department. Vendors and subvendors are required to conform to the specifications concerning the procurement and receipt of raw materials and purchased parts.

After an award of contract to a vendor, a review of documents submitted for approval is made by Bechtel to assure that the original intent and quality requirements have been met. No quality related changes by a vendor are acceptable unless approved by Bechtel. Communication is established and maintained by Bechtel with all vendors in regard to quality control methods and responsibilities.

Bechtel's Procurement Department shop inspectors inspect for product quality, and adherence to specifications and code requirements during

fabrication in the vendors' shops. Specific attention is paid to quality of workmanship, accuracy, finishes; cleaning procedures and facilities; interface setup of connections; and adequacy of cleanliness of shop assembly and test areas. Their shop inspection activities are governed by the Bechtel Procurement Department Inspection Manual.

When specified, final acceptance tests are witnessed and evaluated at the completion of fabrication prior to release for shipment.

D.3.1.3 Field Fabrication and Construction Surveillance

Field fabrication and construction activities are controlled by Bechtel Field Engineers, following instructions in the Field Inspection Manual. The manual covers structures and civil, electrical, and mechanical work. A coordinated inspection reporting system is followed by a qualified Bechtel Field Engineers for all units and systems designated to receive Quality Control inspection.

Construction installation procedures for structural, civil, mechanical, piping, electrical, and instrumentation are followed and results reported back to the cognizant engineering groups. A final quality control construction report on essential items and systems is made after construction is complete and covers tests, cleanliness, and operational checkout prior to startup. Pertinent report forms are sent to the Bechtel home office in San Francisco for review and approval by their Engineering Department to assure that the design requirements are in fact achieved by construction forces.

Bechtel assigns several qualified Quality Assurance Engineers to the jobsite, whose reporting function is back to the Project Engineer in the home office. Close liaison is maintained through the Quality Assurance Engineers, whose specific responsibilities are to review and audit the site quality program and to assure a proper level of quality of field construction. Their duties include surveillance checks, audits, and reviews of QC-QA activities and documentation. They monitor the Bechtel reporting system that records data pertinent to inspection of work under construction, act as a feedback to the Project Engineer on quality of work being accomplished in the field, and monitor inspection and witness testing. Detailed responsibilities of Quality Assurance Engineers are outlined in the Bechtel Pilgrim Project Nuclear Quality Assurance Manual.

Bechtel provides the services of a welding engineer onsite to monitor Project Welding Procedures, to qualify welders, and to document welding and nondestructive testing on all essential items and systems.

D.3.2 Nuclear Steam System Supplier Quality Program

D.3.2.1 General Electric Company Quality System

A quality system is provided by the Nuclear Energy Division (NED) of GE to assure that the required effort, equipment, procedures, and

management are directed toward satisfying the intent of the AEC Quality Assurance Criteria for Nuclear Power Plants and the quality objectives of providing safe and reliable systems and components within the GE scope of supply.

The quality system is designed to assure that the quality related work elements for systems and components supplied by NED are identified, assigned, and controlled from conceptual design through station startup. Specific responsibilities are assigned for quality related activities through the major steps of a power station project encompassing the broad phases of:

Conceptual design

Systems and components specification

Systems and components design

Vendor selection

Material and component procurement

Fabrication of components

Material and component testing

Shipping, site receiving, and storage

Installation of systems and components

Preoperational testing

Startup testing

The quality system recognizes that systems and components supplied by NED come from a number of sources. Components within NED manufacture such items as control rods, control rod drives, control rod drive system components, steam separators, reactor servicing equipment, nuclear fuel assemblies, and instrumentation and control systems and equipment. Equipment within the NED for installation at reactor sites includes such items as the reactor pressure vessel and internals, pumps, motors, piping, valves, and heat exchangers.

The NED staff of technical personnel, experienced in the nuclear industry, supplies designs or design specifications appropriate for nuclear applications for NED manufactured or procured equipment. Quality control organizations direct or audit the quality related work for NED manufactured products and vendor supplied materials and components. NED quality related preoperational testing and startup activities are planned by engineering components within NED.

Line components with assigned quality assurance/quality control responsibilities conduct continuing audits of their activities to assure compliance with the quality system within their assigned

responsibility scope. A BWR quality control staff component is responsible for conducting overall quality system audits to assure integration of, and compliance with, the quality system within the various NED organizational components contributing to the BWR business.

A more detailed description of the GE, NED organizational structure, and functional responsibility assignments is contained in Attachment D.I of this Appendix.

D.3.3 Boston Edison Quality Assurance Program

D.3.3.1 Overall Quality Assurance Program

Boston Edison is responsible for providing a Quality Assurance Program to assure quality related surveillance over engineering design, procurement, installation and construction, and startup testing of equipment and systems for Pilgrim Nuclear Power Station. General description of the overall program is outlined in Edison's Nuclear Quality Assurance Program and supplemental guides and procedures for site construction surveillance outlined in Edison's Nuclear Quality Assurance Manual for Site Construction Activities.

Boston Edison maintains surveillance over the quality control and assurance programs of its contractors, and reviews the applicability and implementation of such programs. Bechtel is responsible for engineering design, procurement, and site construction. GE is the supplier of the Nuclear Steam Supply System (NSSS), turbine-generator equipment, and nuclear fuel. Boston Edison provides a quality assurance surveillance function over nuclear fuel fabrication, and the turbine-generator equipment manufacture.

To maintain a proper level of Project quality, Boston Edison has a Nuclear Quality Assurance Organization consisting of a Senior Quality Assurance Engineer and a staff of fulltime Quality Assurance Engineers and technicians; the Nuclear Quality Assurance Organization is directed by the Senior Quality Assurance Engineer who reports directly to the Assistant Vice President (Nuclear). This Quality Assurance Organization functions independently from the Nuclear Project Group.

A permanent Edison Onsite Quality Assurance Group has been established to maintain surveillance of construction site quality related activities. The plan and objective of this onsite quality effort is to assure that components and systems are constructed in compliance with established Project specifications and criteria.

D.3.4 Consultants Activities

D.3.4.1 Guidance and Control

The services of specialist consultants may be necessary for the proper guidance of the Project. When such consultant activities are within the Bechtel scope of project, then Bechtel will provide

guidance and control. Consultants who may be hired directly by Boston Edison will be guided and controlled by the Nuclear Project Group, except in quality related matters. Quality related matters are controlled and guided by the Nuclear Quality Assurance Organization.

D.3.5 Combined Vendor Shop Audits

Visits to manufacturers' shops by Boston Edison Quality Assurance Personnel are conducted to establish Edison's interest in product quality, and to affirm that Bechtel's and GE's inspection and audit functions and the Quality Control Programs of the manufacturers are functioning.

D.3.5.1 Three Party Vendor Shop Audits

Materials and equipment supplied by GE in the NSSS scope are subjected to quality control shop audit or surveillance by GE whether manufactured by GE or purchased from vendors. Quality assurance surveillance on these designated items is conducted by Bechtel Procurement Department Inspectors who check for compliance with code and specification requirements, and observe general shop practices and adherence to the quality control programs in effect. This is in addition to audit or surveillance conducted by GE.

On a selected list of major equipment and materials associated with the nuclear steam supply system, Boston Edison's Quality Assurance Organization conducts a three party quality assurance audit at various vendor shops. During these audits, documentation is reviewed for accuracy and fulfillment of codes and standards, and specification requirements. Material heat numbers are examined on hardware and components to assure that documentation matches the manufactured equipment. Operational and/or performance tests are usually witnessed during the audit, toward the completion of manufacture.

D.3.5.2 Two Party Vendor Shop Audits

Audits similar to those described above are conducted by Boston Edison on a substantial number of major equipment items in the Bechtel scope of supply.

D.4 QUALITY CONTROLS AND ASSURANCE MEASURES

Quality controls and assurance measures are imposed on the principal phases of engineering design, procurement of materials and components, fabrication and manufacture, shipping, storage and preservation, installation and construction, field inspection, preoperational testing, and systems checkout and acceptance testing. Identification and control of materials, parts and components, and controls on special processes are maintained.

The degree to which the quality controls and assurance measures are imposed and enforced are a function of the integrity and safety requirements of the various structures, systems, and components involved.

D.4.1 Engineering Design

Station engineering design, except the nuclear steam supply system, nuclear fuel, and turbine-generator are the direct responsibility of Bechtel.

Engineering design of the Nuclear Steam Supply System (NSSS), nuclear fuel, and turbine-generator are provided and controlled by GE.

Coordination and quality control measures are reviewed by Bechtel, GE, and Boston Edison to assure proper interface matching of the various project engineering design phases.

D.4.1.1 Bechtel Engineering Design Control

Bechtel performs engineering work including preparation of mechanical, electrical, civil and structural designs, plans, specifications, drawings, bills of material, schedules, and estimates as required to properly describe and detail the project.

Bechtel advises Boston Edison in the determination of basic conditions surrounding the choice of components for Pilgrim Station, other than the exceptions stated above, and is responsible for the environmental studies and reports and the related model studies, as well coordinating the design efforts for the station computer.

Bechtel performs its engineering work on the basis of instructions furnished by Boston Edison. In the absence of such instructions, Bechtel employs its own standards and performs the work, or causes the work to be performed, in accordance with its highest quality engineering skill and judgment. Such work is subject to Boston Edison's review.

Engineering designs are performed by qualified engineering personnel under the supervision of experienced design engineering supervisors. The familiarity that these engineers have with national codes and standards and their previous experience helps assure that the design drawings, specifications, and instructions will reflect and correctly

include the appropriate design considerations, code requirements, and quality control measures.

Specifications and design drawings are approved by both the appropriate design engineering supervisor and the Bechtel Project Engineer before being issued for construction. Principal specifications and drawings are reviewed and commented on by Boston Edison prior to issue for construction.

Significant changes in approved engineering design drawings and specifications are not permitted without the approval of the responsible design engineering supervisor and the Bechtel Project Engineer and must be accompanied by reissued approved drawings and/or specifications.

D.4.1.2 General Electric Design Control

D.4.1.2.1 Nuclear Steam Supply System Design Control

A design control system is provided by the Nuclear Energy Division of GE to assure that design bases, regulatory criteria, and industry codes and standards are appropriately translated and documented in drawings, specifications, and instructions. The system provides for review, approval, release and distribution control of design documentation and changes thereto. A detail description of the GE design control program is included in Attachment D.I of this Appendix.

D.4.1.2.2 Main Turbine Generator

The turbine generator equipment is supplied by GE. Quality control for engineering design and manufacture are the direct responsibility of GE. Bechtel has quality assurance responsibility for this equipment. Special attention is given to safety related items. Overall quality assurance is maintained by Boston Edison.

D.4.2 Procurement of Materials and Components

Bechtel is designated the Agent of Boston Edison Company for procurement of materials, equipment, and supplies for Pilgrim Station (except the Nuclear Steam Supply System equipment, nuclear fuel, and turbine generator).

GE procures materials and equipment within the NSSS, and for nuclear fuel and turbine generator equipment.

It is the direct responsibility of Bechtel and GE to assure that proper instructions are issued to all vendors and subvendors down through four tiers or more if necessary, so that vendors and subvendors of materials and components are cognizant of the total quality requirements for their products and particularly the nuclear safety requirements of parts supplied.

Bechtel is authorized by Boston Edison to perform inspection and expediting services for material, equipment, and supplies for Pilgrim Nuclear Power Station including the NSSS computer, and turbine generator equipment.

D.4.2.1 Bechtel Procurement Controls

Bechtel controls the quality of its purchased materials and components through source selection, submitted data and drawing review, progressive inspection at vendors' shops, witnessing of tests, and job site receiving inspection.

Up to date bidders' lists are maintained by the Procurement Department. From this list, a project bidders' list was prepared and approved by Boston Edison.

When appropriate, purchase specifications include a section delineating the Quality Control Requirements. Project Engineering sets these requirements consistent with the functional importance and complexity of the individual item or system.

Bids are compared on a technical and economic basis to determine compliance with specifications and intended use. Comparisons of bids for major items are reviewed by the Nuclear Project Group, and the Purchasing Department of Boston Edison before release for purchase.

Progressive inspections during manufacture for the control of quality are described in Section D.4.3 and in Attachment D.II of this Appendix and in Bechtel Procurement Department Inspection Manual.

D.4.2.2 General Electric Procurement Controls

The procurement of materials and equipment with associated quality control of components and systems supplied by GE for the NSSS, turbine generator, and nuclear fuel are the first line responsibility of GE.

Details of procurement controls are outlined in Attachment D.I - General Electric Quality Assurance Program, and cover both GE manufactured items and purchased items.

D.4.2.2.1 General Electric Manufactured Products

Purchased material control is exercised for the purpose of assuring that vendor supplied materials, equipment, and services are provided at proper quality levels and conform to the specifications of purchase.

Requests for production material are reviewed by quality control personnel before submittal for purchasing, and necessary quality requirements are added to the material request. These requirements include, where applicable, review of vendor planning, documentation required, objective evidence of quality to be supplied by the vendor, hold points for source inspection by NED quality control personnel,

qualification and certification requirements, and receiving inspection coding.

Vendors selected are qualified by virtue of past performance, vendor survey, or a specific program in which the vendor is qualified by demonstrating the ability to perform. Vendors are required to maintain a quality system commensurate with design and quality assurance requirements. On designated items, quality control maintains surveillance of fabrication and testing at vendors' plants and furnishes reports of source inspections and tests.

Incoming vendor items are inspected and approved by quality control in accordance with the quality requirements as specified in the purchase order, material specifications, standard practices, and receiving inspection plans.

Production materials used must be as specified on applicable approved drawings and specifications unless prior written approval for substitution is obtained from the responsible design engineering component.

Procurement of materials, manufacture, and inspection of nuclear fuel bundle subassemblies are under the quality control standards of GE. Quality control documentation on fuel fabrication will be audited by Boston Edison.

D.4.2.2.2 General Electric Purchased (Engineered) Equipment

GE provides a purchased material control system for those items of equipment purchased from vendors for direct shipment to a station site. The program includes audit and surveillance by GE Field Representatives of vendor activities during manufacture, inspection and test, and is designed to assure vendor compliance with GE design and purchase requirements prior to release for shipment.

D.4.2.2.3 Steam Turbine Generator

The procurement of materials and components associated with the main turbine generator are the first line responsibility of GE, Large Steam Turbine Division. The Quality Control System employed by GE on large steam turbine generator equipment will control the procurement and manufacture of components.

Purchased material control will be exercised to guarantee that vendor supplied materials, equipment, and services are provided at proper quality levels and conform to the specifications of purchase.

GE will initiate quality assurance measures including appropriate audit and surveillance on any and all materials and manufacture originating in foreign lands. Incoming vendor items will be inspected and approved by quality control personnel in accordance with the quality requirements as specified in the purchase order, material specifications, standard practices, and receiving inspection plans.

Boston Edison will impose its quality assurance surveillance on both the turbine and generator during manufacture.

D.4.3 Fabrication and Manufacture Inspection

Documented inspection of fabricated and manufactured equipment and components is performed by Bechtel, GE, and Boston Edison on selected items. The degree and scope of shop inspection was determined at the beginning of the project, and are reviewed and augmented as circumstances require.

D.4.3.1 Bechtel Procurement Inspection

Bechtel's inspection capability is based on established standards of acceptance. The Engineering Department sets the specific standards and incorporates them into specifications. These specifications and other applicable code requirements are reviewed and agreed upon by the vendor before production is permitted.

Inspection is conducted at the vendor's shops to accomplish the following:

1. To assure Bechtel and Boston Edison that material and equipment as shipped by vendors is of the same design, dimensions, material, and quality as specified by the engineers and complies fully within the minimum requirements of the purchase orders, applicable specifications, data sheets, and drawings

2. To minimize delays in the field construction schedule caused by time lost correcting material or equipment incorrectly supplied

3. To minimize delays in startup and potential for future shutdowns caused by incorrectly supplied material, or items containing defective material or workmanship

Inspection requirements, assignments, and preparation and distribution of shop inspection reports are covered in the Bechtel Procurement Department Inspection Manual.

D.4.3.2 General Electric Procurement Audit and Surveillance

Equipment purchased by GE includes such items as reactor pressure vessel and internals, pumps, motors, piping valves, and heat exchangers.

Implementation of quality control in vendors' shops is accomplished by GE Quality Control Field Representatives whose principal functional responsibilities are to review purchased part drawings, and specifications, conduct preproduction reviews with vendors' personnel to assure mutual understanding of code and quality requirements, witness and audit various qualifications, tests, and inspections, and to review quality control records.

Product quality checklists are completed by these inspectors prior to release for shipment.

Attachment D.I to this Appendix outlines details of GE procurement inspection.

D.4.4 Handling, Packaging, and Shipping

Instructions for prepackaging cleaning and preparation, identification and cautionary markings, protection against weathering, corrosion, damage, and the avoidance of undue stressing, and other cautionary instructions and requirements for assuring that materials and equipment arrive at the jobsite as intended, are identified and specified in the Purchase and Installation Specifications. If specifications fail to include such instructions then supplementary instructions may be issued by Bechtel or by GE before materials and equipment are deemed acceptable for shipment.

Bechtel is responsible for the issuance of such specification instructions on applicable balance of plant items. GE is responsible for the issuance of similar specification instructions for applicable items in the NSSS, nuclear fuel, and turbine generator equipment. GE is also responsible for the issuance of specification instructions for the replacement, recirculation, RHR, RWCU and core spray piping inside the drywell.

D.4.5 Storage and Preservation

Site receiving, inspection, handling, and storage control is provided by Bechtel on equipment and materials arriving at the jobsite including turbine-generator, and nuclear steam supply system equipment.

Inspection and storage control are the prime responsibility of the Bechtel Material Supervisor and Project Field Engineer.

Equipment and materials are stored primarily in protected areas outside active construction zones and when required, inside waterproof heated buildings. Equipment deemed suitable for outside storage is protected by heavy-duty construction tarpaulins or other adequate protection.

The handling and storage of components and equipment supplied by GE within the NSSS are in accordance with written aids and guides provided by GE, and summarized in Attachment D.III to this Appendix.

Bechtel provides for necessary protection against deterioration or damage to materials and equipment in storage at the site. Inspection for deterioration or damage is provided. When appropriate, packaging includes means for control of proper environments within packages, e.g., moisture content levels, gas pressures.

The proper maintenance of essential items of equipment while in storage at the site is controlled by Bechtel and audited by Boston Edison Quality Assurance Engineers.

The following describes the storage and preservation controls and responsible organizations for the pipe replacement program.

Site receiving, inspection, handling and storage control was provided by BECo/GE-DA&ESO* on equipment and materials arriving at the job site.

Inspection and storage control were the prime responsibility of BECo.

Piping, equipment and materials were stored primarily in protected areas outside active construction zones and when required, inside waterproof heated buildings. Equipment deemed suitable for outside storage was protected by heavy-duty construction tarpaulins or other adequate protection.

The handling and storage of components and equipment supplied by GE were in accordance with written aids and guides provided by GE.

BECo provided for necessary protection against deterioration or damage to materials and equipment in storage at the site. Inspection for deterioration or damage was provided. When appropriate, packaging included means for control of proper environments within packages, e.g., moisture content levels, gas pressures.

The proper maintenance of essential items of equipment while in storage at the site was controlled by BECo through BECo's Quality Assurance Program.

D.4.6 Cleanliness During Installation and Erection

Aggregate for concrete mix is graded for size, stockpiled, and maintained free from dirt and construction dust. Final examination and washing is accomplished prior to admittance to the concrete batch plant. Mixing of concrete is computer controlled in an automatic batch plant.

Structural backfill materials are carefully graded to fulfill specified compaction requirements and cleanliness.

Components and equipment are inspected, and protected during assembly and erection and laid up after installation in a clean and protected manner awaiting systems startup. Preservatives, additives, heat, inert gas, etc, are used for temporary protection prior to startup.

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*BECo = Boston Edison Company
GE = General Electric Company
DA&ESO = General Electric's Domestic Apparatus and Engineering Services Operation

Essential items in piping systems are inspected during erection and fitup to assure proper levels of internal cleanliness. Piping which does not meet predetermined cleanliness standards is returned to the pipe fabricator or recleaned on site to remove heavy mill scale, oil, and dirt. Special precautions are used in the handling of Type 304 stainless steel, including pressure vessels, piping, tanks, liners, and components.

Structural materials such as steel reinforcing bar, structural steel and supports, concrete building panels, concrete formwork, and imbedded pipe and conduits are checked for cleanliness during installation.

D.4.7 Installation and Construction

Quality control and assurance at Pilgrim Site, with respect to civil, structural, mechanical, and electrical construction, is the assigned responsibility of Bechtel. Such quality control is accomplished by Bechtel Field Engineers, Quality Control Engineers, and qualified outside contractors.

Technical direction is provided by the GE Nuclear Energy Division for the installation of equipment and components supplied within the scope of the NSSS package. In addition, NED provides instructions for site receiving, inspection, handling, and storage control for NED supplied equipment and components. NED also supplies separate installation instructions for reactor recirculation piping, primary steam piping to second isolation valve and reactor vessel internals and appurtenances. Also, necessary installation instructions are provided in equipment manuals for assemblies and components supplied by NED. Furthermore, NED implements preplanned audits of field installation of NED supplied equipment.

Bechtel maintains control of construction and quality through instructions contained in the Field Inspection Manual and the Pilgrim Project Nuclear Quality Assurance Manual. Additional controls are maintained in the following general categories.

D.4.7.1 Removal and Installation - Piping Replacement Project

The removal and installation activities for the recirculation, RHR, RWCU and core spray piping were carried out by the GE's Domestic Apparatus and Engineering Services Operation (DA&ESO) under the direction of a Site Manager reporting to the GE Project Manager. The Site Manager and his organization were dedicated solely to the project.

Bechtel Construction provided craft labor to GE. Craft labor were trained and supervised by GE, with Bechtel administering the craft labor agreements. Bechtel Power Corporation provided engineering support to GE, primarily in areas where Bechtel was the original designer.

General Electric utilized qualified subcontractors in such areas as piping decontamination and disposal, automatic welding equipment, drywell decontamination and non-destructive examination.

The GE Quality Assurance Engineer was responsible for coordinating all project quality assurance/control activities, including those of the various QA/QC components within GE-NEBO.

The repair was performed in accordance with GE-NEBO's BWR Quality Assurance Program Description NEDO-11209-4A and the BWR QA Manual NEDE-20586. GE DA&ESO was responsible for on-site installation, including site receipt inspection and technical support activities, which was performed in accordance with its Nuclear Production Control Manual P1A-AE-11.

An Authorized Nuclear Inspector (ANI) was retained by BECo to certify that applicable work is performed in accordance with American Society of Mechanical Engineer's Boiler and Pressure Vessel Code requirements.

BECo provided radiological control of all project activities and managed the ALARA program. GE designed and supplied temporary shielding, developed the details of the ALARA program and supplied ALARA and health physics coordination.

BECo procedures that must be complied with for the project were defined by the BECo Project Manager. BECo exercised QA control of the overall project activities via a comprehensive audit program.

The removal and installation of the recirculation, RHR, RWCU and core spray system piping were in accordance with the applicable requirements of 10CFR50, including 10CFR50.59, 10CFR50.54(f) and 10CFR50 Appendix B and ASME Section XI, 1980 Edition through Winter 1980 Addenda, IWA 4000, IWA 7000, IWB 4000 and IWB 7000.

D.4.7.2 Earth Compaction

Bechtel field engineering personnel supervise earth compaction operations which in turn are monitored by the Project Field Engineer.

Tests are performed on selective structural backfill by a qualified outside testing contractor under the supervision of Bechtel's Project Field Engineer. Complete laboratory type test facilities have been maintained at the jobsite.

D.4.7.3 Reinforcing Steel

The inspection of steel reinforcing rods and bars prior to concrete placement, for adequacy of installation, splicing details, blockouts, cleanliness, etc, are under the direct supervision of Bechtel's Field Engineers.

Testing of reinforcing steel is accomplished by an outside independent subcontractor to Bechtel. Test results are reviewed and approved by Bechtel.

D.4.7.4 Concrete Mix and Placement

The inspection of form work, reinforcing steel, and embedments prior to concrete placement is performed by Bechtel under a checkout system to assure compliance with drawings, cleanliness, placement procedures including adequacy of placement equipment, curing, and protection of concrete.

The various quality tests including curing and strength, are performed by an outside independent subcontractor to Bechtel Corporation. Surveillance of test procedures, testing, and results are the responsibility of Bechtel.

Complete laboratory type concrete testing facilities have been maintained at the jobsite.

D.4.7.5 Welding

Structural and mechanical systems welding and welding tests at the jobsite fall into two categories; those which are the direct responsibility of Bechtel and those performed by Bechtel's Subcontractors. Welding procedures for subcontractors are reviewed and approved by the Bechtel Metallurgy and Welding Department prior to start of welding at the site. Radiographic testing of materials and welds is accomplished by qualified experts independent from the above mentioned subcontractors. subcontractors for radiographic testing, with Hartford Insurance Inspection on code welds, perform initial film interpretations. Final interpretation of radiographic films and weld approval are the direct responsibility of Bechtel. Boston Edison reviews radiographs on a sampling basis and audits welding records periodically.

D.4.7.6 Welding - Piping Replacement Project

Structural piping, and equipment welding, welding test, and NDE procedures were the responsibility of GE-DA&ESO. These procedures were reviewed and approved by GE-NEBO. Cutting, beveling, grinding, cleaning, descaling, rigging, removing of piping and equipments, mock-ups, fit-up requirements, purging, weld-preps, welding, weld processes, weld-rods, testing quality control and non-destructive examination and general replacement instruction procedures requirements were the responsibility of GE-NEBO.

Quality control of weld end preparations includes inspection of machined ends, alignment, proper erection, cleanliness, and acceptability of welding and preheat equipment. Approved welding techniques are under direct supervision of the welding supervisor of Bechtel and its subcontractors.

Preheat and postheat treatment are supervised by the welding supervisory personnel listed above.

Controls are imposed for protection against weathering and corrosion, joining techniques, housekeeping and cleanliness practices, stressing of components, and qualification of NDT technicians.

D.4.7.7 Welding Electrode Material Controls

Controls for welding electrode material disbursal and use are in effect on the Pilgrim Project, and govern the receipt, storage, disbursal, and documentation of welding filler material.

Procedural details have been adopted to assure that:

1. Welding rod material received is stored in an enclosed area (80°F to 100°F) in the warehouse. Boxes are stacked according to type and size within the warehouse but separated from other storage

2. Each craft draws from the central warehouse on regular requisition forms except in the case of rod to be used on essential items and systems, in which case Bechtel form WR-6 is used

3. Low hydrogen and stainless steel electrodes are stored in separate 300-lb ovens immediately on removal from shipping containers

4. Each welder using low hydrogen or stainless electrodes is assigned an individual rod warmer at his job. These rod warmers are returned at the end of the shift to each craft's central oven area, rods removed, and placed in 300-lb cylinder ovens

Bechtel's welding supervisors monitor these activities for compliance.

For replacement pipe, controls for welding electrode material dispersal and use were controlled on the Pilgrim project and govern the receipt, storage, dispersal and documentation of welding filler material. Procedural controls were used to assure that:

a. Welding materials received were properly identified and in undamaged containers. The material certification was checked against specification requirements.

b. Low hydrogen covered electrodes were removed from the hermetically sealed containers and stored in properly controlled heating ovens prior to issue. Returned, undamaged and uncontaminated electrodes were properly baked prior to reissue.

c. The distribution of the welding materials were controlled to assure that the proper materials were issued for the specific weld.

d. Low hydrogen covered electrodes were kept in portable rod warmers upon removal from the storage oven.

D.4.7.8 Construction Drawing and Document Control

Control of construction drawings and other such documents at the jobsite is necessary to prevent the use of outdated or unrevised copies. Only limited copies of each document are made available and each of these copies is assigned to specific location. When a revision to a drawing or document is received at the jobsite the required number of copies is distributed to the specific locations where responsible personnel destroy or void the old document and replace it with the new revision.

A Design Documentation Control Procedure has been provided by Project Engineering for onsite control of documents and engineering field changes.

D.4.7.9 Calibration of Instruments, Tools, and Equipment

Certain tools, measuring devices, and calibration equipment employed in construction and testing of systems and equipment are periodically calibrated to assure required accuracy. Calibration schedules are established for such equipment.

D.4.8 Field Inspection

Field inspections are performed by Bechtel to assure compliance with approved engineering designs, specifications, and drawings. GE, NED, performs preplanned audits of field installation of NED-supplied equipment. Construction and installation surveillance and documentation audits at the jobsite are accomplished by Boston Edison Quality Assurance Engineers for the purpose of assuring compliance with quality requirements. Edison's Construction Inspection Staff provides additional surveillance capability.

D.4.8.1 Bechtel Field Inspections

Inprocess and final inspection of field work is performed by Field Engineers under the supervision of the Project Field Engineer. These Field Engineers follow instructions given in the project drawings and specifications, and inspection procedures as prescribed in the Field Inspection Manual. Records of field inspections are generated by the Field Engineers and maintained by the Quality Control Engineer.

Subcontractors at the jobsite are required to provide inspection capability and evidence of their quality control performance. The quality control work performed by subcontractors is monitored by Bechtel.

Bechtel has employed independent testing laboratories to perform soils testing, concrete testing, and radiographic inspection. Records of tests and inspections performed by these laboratories are submitted to Bechtel for permanent files. The activities of the laboratories are under the control of the Project Field Engineer and are monitored by the Quality Control Engineer.

D.4.8.2 General Electric Field Surveillance

Technical direction and surveillance of NSSS equipment and systems are provided by GE (NED) through the GE Field Engineer, Quality Control Site Representative, and the GE Quality Control Engineer. Their function is to provide technical guidance, advice, and counsel, based upon current engineering and installation practices, and to audit conformance with NED-approved installation procedures and inprocess controls relative to NED-supplied equipment.

D.4.8.3 Boston Edison Field Inspections

Boston Edison maintains a Resident Engineer at Pilgrim Site who provides surveillance over the onsite construction activities of the prime construction contractor and all subcontractors, and supervises Edison's Construction Inspectors. The Resident Engineer has the authority to stop any work that he deems improper.

An onsite Inspection Staff reports directly to the Boston Edison Resident Engineer. These experienced power plant Construction Inspectors provide liaison and surveillance over construction and installation activities.

Boston Edison's onsite Quality Assurance Group maintains an overall surveillance of construction activities by means of audit oriented field inspections in support of quality documentation auditing.

D.4.8.4 Independent Consultants' Activities

Independent consultants are requested, periodically, to assist in evaluating situations which require specialist qualifications. Their activities are mainly concerned with the environmental sciences and nondestructive testing of materials. Inspections may be a fundamental part of their activities. When appropriate, Boston Edison's quality assurance personnel audit the consultants' work program to assure that the intent of the program is fulfilled.

D.4.9 Preoperational Testing

The Station is made operational by Boston Edison operating personnel under the technical direction of GE and Bechtel using written test procedures and operating instructions.

Preoperational and acceptance test procedures identify the systems and equipment which are to be tested and state the requirements of the tests. Test procedures and test results are reviewed and approved by Boston Edison and either Bechtel or GE, as appropriate.

D.4.10 Startup Testing

After each preoperational test is completed, GE or Bechtel and Boston Edison operating personnel check off that the tests have been completed, and that the results are in accordance with acceptance criteria of the test procedure.

Initial fuel loading and startup testing are performed by Boston Edison operating personnel under the technical direction of GE. Startup test specifications define the test program for safe and efficient startup with defined limits and changes permitted during startup test activities.

Startup test procedures present the recommended test method and describe the steps necessary to perform the tests defined in the startup test specification.

D.4.11 Control of Special Processes

Special processes are done in accordance with written procedures incorporating applicable codes and standards. These special processes include welding, heat treating, magnetic particle and liquid penetrant examination. These procedures have been qualified and personnel involved in performing these processes have passed qualification tests. Records are kept on qualification tests. Such qualifying activities are performed under the supervision of the Bechtel Field Welding Engineer. Radiography is performed by an independent engineering company whose personnel have passed appropriate qualification tests.

Boston Edison audits welder qualification records and reviews radiographs and other nondestructive test records on a sampling basis.

D.4.12 Identification and Control of Materials, Parts, and Components

Equipment, piping, and valves will be tagged or coded to permit readily available identification. Structures and pressure vessels, such as the containment vessel and essential equipment, will be identifiable by nameplates as required by the applicable codes and standards. Materials, parts, and components will be traceable from such identifications or markings back to a specific purchase order or requisition and thence back through manufacturers' records to the original code required documentation.

Piping 1 in and over and associated fittings and valves within the reactor coolant pressure boundary will be identified by permanently attached mark numbers, and other identification for traceability of the item back to appropriate code requirements.

Bulk materials such as bolts, random lengths of pipe, and electrical cable are identified where required as to type and size. This is accomplished by tags, labels, markings, or other means either on the item, its container, or records traceable to the item.

D.4.13 Nonconforming Materials, Parts, and Components

Nonconforming materials, parts, and components are not permitted to remain in any structure or system of Pilgrim Station without concurrence and approval of Bechtel and Boston Edison.

D.4.13.1 Identification and Labeling

Whenever a process is performed in such a manner as to cast doubt on the acceptability of a product, or whenever a material is found by shop or field inspection or test not to meet applicable drawings or specifications, the items are considered rejected and labeled to identify the item as a rejected part. A controlled tagging and identification system is in effect to identify unsuitable components and equipment pending resolution of the discrepancy.

D.4.13.2 Disposition

Discrepancies found during manufacture through shop inspection activities or damaged during shipping or installation are either reworked to meet all the requirements of the specifications, drawings, and applicable codes and standards, or are scrapped.

Nonconforming materials, parts, or components discovered in the field during tests and inspections are reviewed by both Bechtel Field Engineering and home office Project Engineering. The course of corrective action for minor deviations is decided upon by Field Engineering. Major or unusual deviations are referred to and resolved by Project Engineering. Standard repair methods are governed by approved procedures and major defect repairs are documented.

D.5 QC-QA DOCUMENTATION AND RECORDS

D.5.1 Documents and Records Requirements

Quality oriented records and documentation on engineering design, fabrication, and manufacturing, shop and field testing, erection and installation, and construction of structures, which are associated with essential items and systems, will be compiled and maintained throughout the life of the reactor plant.

In general, the requirements for documentation by Boston Edison for essential items are those specified in the many applicable national codes and standards. Additional documentation may be required as a need is identified.

National and Nuclear Codes and Standards which are used for guidance in the design, fabrication, and construction of Pilgrim Station are those approved and accepted by industry from the following responsible organizations:

- American Concrete Institute
- American Petroleum Institute
- American Society of Civil Engineers
- American Society of Mechanical Engineers
- American Society for Testing and Materials
- American Standards Association
- American Welding Society
- Feedwater Heaters Manufacturers Association
- Heat Exchanger Institute
- Institute of Electrical and Electronic Engineers
- Institute of Nuclear Materials Management
- Instrument Society of America
- National Bureau of Standards
- National Council on Radiation Protection and Measurement
- National Electrical Manufacturers Association
- Underwriters Laboratories, Inc.
- U.S. Atomic Energy Commission
- U.S. Coast Guard
- U.S.A. Standards Institute

Table D.5.1 outlines typical Pilgrim Project document and record requirements.

A file of operating and maintenance manuals for essential equipment is being compiled and will be maintained. Included in this file are the manufacturers' equipment drawings, data sheets, operating curves, and any data likely to be required for safe operation, effective maintenance, or engineering study.

D.5.2 Documentation Procurement

Engineering drawings and specifications, procurement and manufacturing records, receiving and storage records, construction and installation records, preoperational testing records, and

equipment operating and maintenance manuals are obtained as soon as final revisions are completed or when construction and testing programs are documented. These documents are furnished by GE for the turbine generator, nuclear steam supply system components, and nuclear fuel. Bechtel is responsible for records procurement on the remainder of plant items. The documentation of finalized information on the Pilgrim Project is a prime responsibility of Bechtel, GE, and vendors supplying equipment and services to the Project.

D.5.3 Documentation Retention

Quality control and assurance documentation associated with essential items, systems, and structures will be assembled in a data retrieval system and retained for the life of the reactor plant by Boston Edison.

TABLE D.5-1

DOCUMENT AND RECORD REQUIREMENTS
PILGRIM NUCLEAR POWER STATION
BOSTON EDISON COMPANY

I. Engineering Design Records

1. Engineering design and layout drawings and revisions
2. Specifications and revisions
3. Engineering data sheets
4. Procedures and instructions
5. Manufacturers' drawings, curves, and equipment data sheets (final revision)

II. Procurement and Manufacturing Records

1. Mill test reports on materials
2. Chemical, mechanical, and physical analyses
3. Nondestructive test reports (U-T, radiograph, liquid penetrant, etc.)
4. Material identification and marking
5. Heat treatment reports
6. Welding procedures and certifications
7. Shop test and acceptance records
8. Safety and relief valve set point data

III. Construction and Installation Records

1. Welder's qualification test records
2. Field welding records (radiographs)
3. Concrete slump and strength test records
4. Reinforcing steel test reports
5. Fill compaction test results
6. Vessel and containment leakage and pressure test records
7. Pump and equipment alignment records
8. Lubrication records
9. Hanger adjustment check data
10. Identification and verification of cable routing
11. Piping and system hydrostatic test records
12. Motor operated valve data sheets
13. Switchgear inspection records
14. Motor control circuit records which include rotation checks
15. Circuit test record

IV. Preoperational Test Records

1. Containment leakage test records
2. Air- and motor-operated valve data sheets
4. High potential test data
5. Electrical relay test and calibration data
6. Instrumentation and control calibration test records

D.6. PROJECT COMMUNICATIONS

Communications among the involved departments of Boston Edison, Bechtel, and GE are accomplished throughout all levels of management and at working levels in the coordination of planning, engineering design, procurement, and construction for Pilgrim Station.

D.6.1 Intracompany Communications

Project direction control is maintained by Boston Edison executives and upper management through executive review discussions conducted by the Assistant Vice President (Nuclear) to inform and obtain guidance from the Chairman of the Board, President, Executive Vice President, Vice President of Operations and Engineering, and other top management personnel.

Project Group coordinating and discussion meetings are conducted by the Vice President (Nuclear) with members of the Pilgrim Project management organization. These meetings define and assign responsibility at the working level of management of project design engineering, quality control and quality assurance, construction at the site, and coordination of operator training and preparation for startup and operation. Onsite Quality Assurance contacts are maintained to coordinate quality related activities of Boston Edison, Bechtel, and GE.

Direct lines of communication exist among Boston Edison's Engineering and Construction, Operations and Purchasing Departments.

Bechtel's project channels of communication are formally established by written Bechtel instructions covering the home office engineering design effort, the site construction effort and the quality assurance effort, and are outlined on Figure D.2-3, Quality Assurance Responsibility and Lines of Communication.

GE's intracompany communications for quality control are established generally in accord with Figure D.6-1. Functional responsibility for quality assurance and quality control at GE's engineering, procurement, and manufacturing facilities, and the necessary communications in effect to assure quality systems and products are enumerated in their Quality Assurance Program-Attachment D.I of this appendix.

D.6.2 Interagency Communications

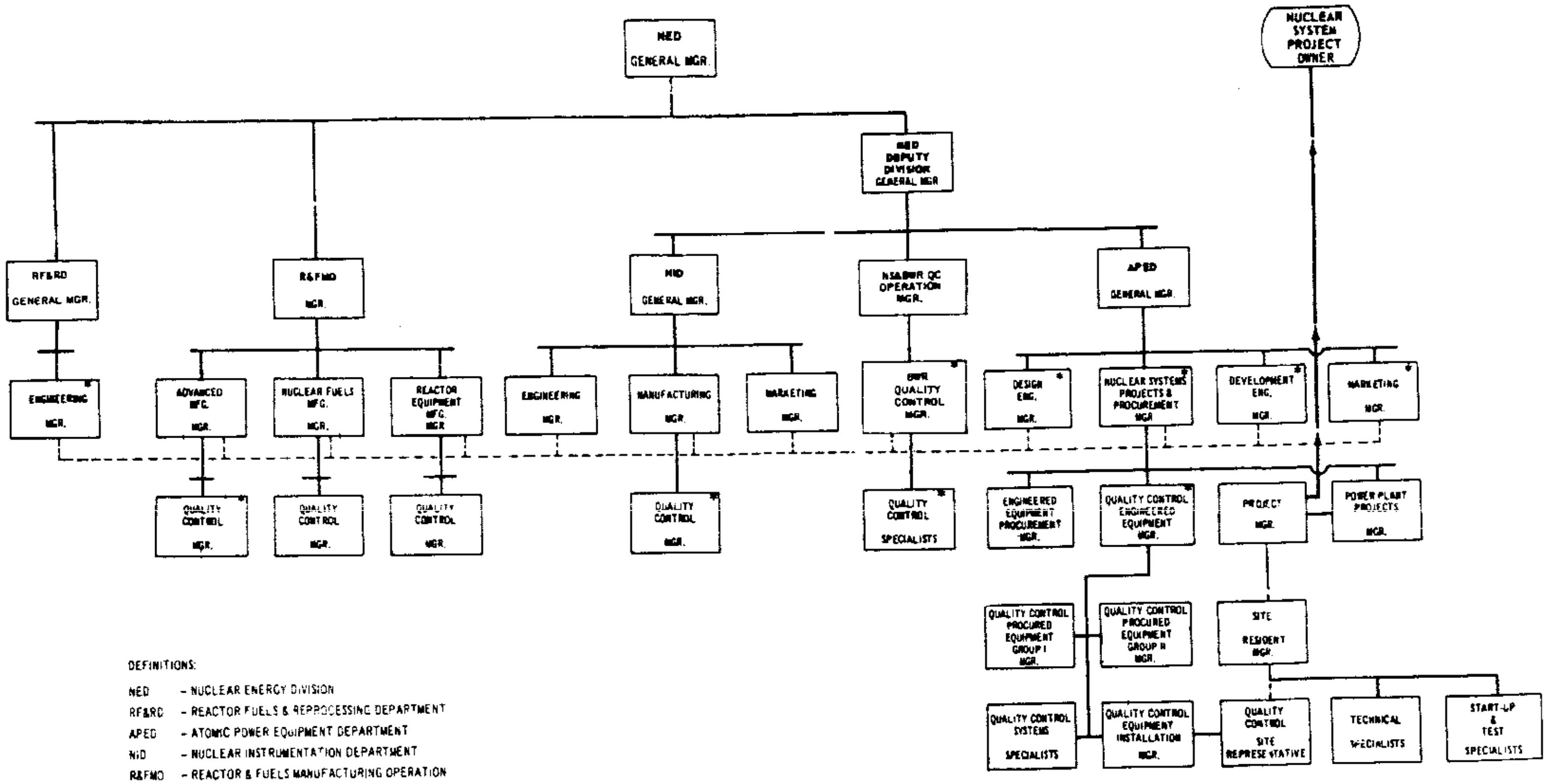
Routine communications are in effect between project and upper management groups at Boston Edison, Bechtel, and GE via mail, telephone, and prearranged conferences in Boston, in San Francisco, in San Jose, and at the Site.

Frequent review conferences are undertaken by the three companies. Auditing visits to the construction site to check on quality of construction and the operability of onsite quality control and

assurance programs are conducted by Bechtel and GE project and quality assurance personnel.

Executive Review Meetings are convened for the purpose of maintaining top level control of policy matters and coordination of quality, scheduling, and cost by the executives of Boston Edison, Bechtel, and GE.

The entire Boston Edison effort is intended to establish, promote, and maintain clear cut lines of communication.



DEFINITIONS:

NED – NUCLEAR ENERGY DIVISION
RF&RD – REACTOR FUELS & REPROCESSING DEPARTMENT
APED – ATOMIC POWER EQUIPMENT DEPARTMENT
NID – NUCLEAR INSTRUMENTATION DEPARTMENT
R&FMO – REACTOR & FUELS MANUFACTURING OPERATION
NS&BWR QC – NUCLEAR SAFETY & BWR QUALITY CONTROL OPERATION

NOTE: THE BWR QUALITY COUNCIL CONSISTING OF REPRESENTATIVES FROM KEY ORGANIZATIONAL COMPONENTS (*) IS CHAIRED BY THE MANAGER, BWR QUALITY CONTROL

FIGURE D.6-1
BWR NUCLEAR SYSTEMS PROJECTS QUALITY SYSTEMS ORGANIZATIONAL STRUCTURE
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

ATTACHMENT D.I

GENERAL ELECTRIC QUALITY ASSURANCE PROGRAM

D.I GENERAL ELECTRIC QUALITY SYSTEM FOR BWR NUCLEAR STEAM SUPPLY PROJECTS

D.I.1 Foreword

GE recognizes that a responsible and comprehensive system to attain the desired level of quality of systems and components is a necessary part of nuclear power station safety and reliability. To that end the Nuclear Energy Division of GE has expended considerable effort in developing a quality system consistent with product requirements. The elements and details of the quality system have been evolutionary and the system is broader in scope now than it was several years ago by virtue of the many improvements in methods and practices which NED has developed and by continued intensive study of the requirements of the nuclear industry and the regulatory agencies.

The information which follows reflects the GE quality system as it is currently structured and implemented. Changes may be made to the system as a continuing effort to further improve system efficiency and product quality.

D.I.2 Introduction

A quality system is provided by the Nuclear Energy Division (NED) of GE to assure that the required effort, equipment, procedures, and management are directed toward satisfying the intent of the proposed NRC Quality Assurance Criteria for Nuclear Power Stations and the quality objectives of providing safe and reliable systems and components within the GE scope of supply.

This document describes, in summary, the quality system established by NED for application to those projects for which GE supplies the nuclear system. The principal objectives of the quality system and the key functions and elements which it contains are not expected to change over the duration of a nuclear system project. However, circumstances may make advisable changes in the organization or emphasis in the quality system; and such changes will be made in accordance with normal management practice. The owner* will be notified of significant changes in the quality system as set forth in this section.

* The term "owner" is herein used to identify the Applicant and/or his designated representative.

The quality system is designed to assure that the quality related work elements for systems and components supplied by NED are identified, assigned, and controlled from conceptual design through station startup. Specific responsibilities are assigned for quality

related activities through the major steps of a power station project encompassing the broad phases of:

Conceptual design
Systems and components specifications
Systems and components design
Vendor selection
Material and component procurement
Fabrication of components
Material and component testing
Shipping, site receiving, and storage
Installation of systems and components
Preoperational testing
Startup testing

The quality system fully recognizes that systems and components supplied by NED come from a number of sources. Components within NED manufacture such items as control rods, control rod drives, control rod drive system components, steam separators, reactor servicing equipment, nuclear fuel assemblies, and instrumentation and control systems and equipment. Equipment within the NED scope of supply purchased from outside NED for installation at reactor sites includes such items as the reactor pressure vessel and internals, pumps, motors, piping, valves, and heat exchangers.

The NED staff of technical personnel, experienced in the nuclear industry, supplies designs or design specifications appropriate for nuclear applications for NED manufactured or procured equipment. Quality control organizations direct or audit the quality related work for NED manufactured products and vendor supplied materials and components. NED quality related preoperational testing and startup activities are planned by engineering components within NED.

Line components with assigned quality assurance/quality control responsibilities conduct continuing audits of their activities to assure compliance with the quality system, within their assigned responsibility scope. A BWR quality control staff component is responsible for conducting overall quality system audits to assure integration of, and compliance with, the quality system within the various NED organizational components contributing to the BWR business.

D.I.3 GE Organization and Responsibilities for Boiling Water Reactor Quality Control

An abbreviated NED organization chart showing specifically the quality-related functions concerned with supplying Boiling Water Reactor (BWR) nuclear systems and components is shown as Figure D.4-1. The managers of the Atomic Power Equipment Department (APED), Reactor Fuels and Reprocessing Department (RD&RD), Reactor and Fuels Manufacturing Operation (R&FMO), Nuclear Instrumentation Department (NID), and Nuclear Safety and BWR Quality Control Operation report either directly to the NED General Manager or to the Deputy Division General Manager.

The manager of the Nuclear Safety and BWR Quality Control Operation has the responsibility for integrating the quality programs for the BWR business. A BWR Quality Control component reporting to the manager, Nuclear Safety and BWR Quality Control Operation, is a staff component assigned responsibility for establishing, documenting, and directing an overall quality system, and for integrating, measuring, and auditing the quality related work across the entire spectrum of the BWR business as conducted by line components.

A BWR Quality Council, with the Manager, BWR Quality Control, as chairman, provides for intradivision communication and integration of quality assurance policies, procedures, and practices.

The BWR Quality Council consists of representatives from each of the contributing organizational components of the BWR business and regularly meets to review status of the overall quality system and to provide management reports of quality related activities.

On a specific project, contact with the owner is through the project manager in the Nuclear Systems Projects and Procurement Section of APED. The project manager provides liaison with the owner on project quality related matters and determines that the required documents and records are channeled into predetermined locations.

The Manager, APED Design Engineering, is responsible for establishing overall quality objectives and quality requirements for systems and components within APED scope. The respective Quality Control Line organizations have responsibility for assuring compliance with these quality requirements.

Table D.I-1 identifies the NED management having responsibilities for attainment of the quality requirements established by APED.

Detailed designs of nuclear systems and components, whether fabricated by NED or by manufacturers outside NED, must meet the specified nuclear system requirements. APED Design Engineering maintains a continuity of engineering control from conceptual design phase through material procurement, manufacturing, field installation, preoperational testing, and station startup. To assure that the nuclear system requirements are met, and to assure compliance with the nuclear system design and design requirements, APED Design Engineering reviews and approves the appropriate detailed design documents, including selected owner documents. APED Design Engineering has design change responsibility and authority for all nuclear systems and components for all phases of a project, except when detailed design is provided for fuel and for nuclear instrumentation and control systems by RF&RD or NID Engineering components. When detailed design is provided by RF&RD or NID Engineering, these engineering components have detailed design change responsibility and authority. However, APED Design Engineering approval is required if deviation in the system design or the design of other components is involved.

APED Development Engineering contributes to the overall quality system by providing basic technical information and advanced inspection techniques resulting from development programs conducted.

The quality control activities related to NED manufactured products are under the direction of managers of quality control Sections who report directly to the managers of manufacturing sections. The quality programs for NED procured equipment, including audits of field installation of this equipment is under the direction of the manager, Quality Control Engineered Equipment. Quality control managers' responsibilities are divorced from those related to production scheduling and to the meeting of production schedules.

Technical direction for field installation of NED supplied systems and components, in accordance with supplied installation instructions, is provided by the APED site resident manager and staff. Preoperational testing and startup engineering specialists assigned to the site resident manager have the responsibility for planning and providing technical direction for the preoperational testing and startup activities.

D.I.4 Design Control

D.I.4.1 General

Design review, approval, release, and change control systems are documented and are in current use in the engineering components contributing to the BWR business. System and components designs are controlled from concept through the startup of a nuclear power station to assure consideration of performance, safety, and reliability.

Each GE BWR is identified as a particular member or variation of the BWR "product line". Each member or variation of the BWR product line is reviewed for safety and conformance to the applicable NRC regulations prior to its being committed to sale. Since several sales may be based on a particular member of a "product line" design, review is not performed for each sale.

D.I.4.2 System Design

Nuclear system data sheets for a specific system contract are issued by the Systems Engineering Section of Design Engineering to the responsible system design components within Design Engineering in the early months following the reciept of an order. The system design controlling documents prepared by the responsible system design engineers typically include the system design specification, piping and instrument diagrams, process flow diagrams, function control diagrams, and the instrument engineering diagrams. These documents are available for issuance to the station designer, in preliminary form, over a period of approximately 9 to 18 months after the receipt of an order.

The nuclear system data sheets and the system design controlling documents incorporate the design and safety requirements for each station. It is the responsibility of the manager, Design Engineering, to conduct safety analyses and to audit the station design to ascertain conformance to established design criteria and design safety requirements. The system design controlling documents are subject to a technical review, and approval by the Requisition Engineering Section of Design Engineering prior to being issued as a basis for component design.

Design Engineering has issued a series of general standard design specifications which are based on standard station equipment requirements. These standard design specifications provide the base for design engineers in APED to use in designing components which satisfy the system requirements. The general design specifications identify the industry codes and any supplemental requirements to be utilized to assure compliance with safety criteria, quality levels, and specific requirements which APED imposes to meet accepted reliability goals. Changes to this group of design specifications must receive the approval of all affected Section managers within APED Design Engineering. The original issue and any changes are subject to review and approval by Requisition Engineering subsection.

D.I.4.3 Overall Design Review

The component design is initiated using the system data sheets, system design controlling documents, and the respective general design specification for all components with the exception of the nuclear instrumentation components. The nuclear instrumentation design is not initiated until after completion of a design review which is discussed below.

An overall design review is made of the engineering work after a period of 20 to 24 months from the award of a contract. The overall design review is made by a task force which is directed by the requisition engineer who is responsible for total project coordination in Design Engineering, but who is not directly responsible for the generation of the systems controlling documents or the design specificiations. Licensing and safety representatives participate to make a comparison with the PSAR and any modifications to it which may have occurred. The responsible design engineers participate in the review to provide necessary design information and to initiate and to follow through on any required changes. Following implementation of any changes required as a result of the review, the design is frozen. Following the overall design review, further changes to the system documentation will only be made for the following reasons: (1) requests by the customer for changes from the original power station sold, (2) feedback from earlier station startups, or (3) other information to indicate that the systems as originally designed will not perform as required by contract, internal criteria, or systems design requirements.

Documents covered by the overall design review are the system design specifications, piping and instrumentation diagrams, process flow

diagrams, functional control diagrams, and instrument engineering diagrams.

Prior to and following the overall design review, a rigid control change procedure established within Design Engineering is followed. The control procedure requires through documentation and approval at appropriate levels of management before a change of any system-controlling document can be implemented. The responsible systems design engineer is charged with the responsibility for defining all documents affected by any such change, for coordinating with other design engineers whose documents are affected, and for obtaining the necessary management approval. Distribution is made to all responsible engineers, their unit and Section managers, the requisition engineer, and to personnel in other Sections who have a need to know. Changes at interfaces between the owner and APED-supplied equipment will be reviewed between the two parties as appropriate.

D.I.4.4 Component Design

D.I.4.4.1 Design of APED Purchased (Engineered) Equipment

The design documentation for the GE APED purchased items normally consists of equipment procurement specifications which specify the general requirements, purchased part drawings which show the outline and interface requirements, and specific data sheets which define the project unique requirements of the equipment. The responsible design engineer approves the specifications, drawings, and data sheets after they have been generated, and a review is made by the requisition engineer to determine that the design documents meet the station data sheets, system design documents, and the general design specifications. Equipment purchase specifications and/or purchase part drawings are reviewed by Quality Control - Engineered Equipment prior to supplier bidding.

The specification outlines the engineering documents such as drawings, procedures, and calculations which must be submitted by the supplier for review and approval by Design Engineering. Any subsequent changes to the approved vendor documents are formally reviewed, and require approval by Design Engineering.

D.I.4.4.2 Design of Mechanical Components Manufactured by Reactor Equipment Manufacturing (R&FMO)

The design documentation for equipment manufactured by R&FMO normally consists of specific detailed design drawings augmented by design and procedure specifications necessary to fabricate, inspect, and test the finished product. Design Engineering identifies the relative importance of the various design requirements defined in the controlling documents as a guide for the quality control organization in Reactor Equipment Manufacturing in establishing the quality plan. The design engineer is responsible for having the design documents reviewed for conformance to the system design control documents, and the general design specifications referenced previously before

releasing them to manufacturing. This review is performed by design engineers in other components with which there is an interface plus Materials Engineering (APED), Manufacturing Engineering (Reactor Equipment Manufacturing), and Quality Control (Reactor Equipment Manufacturing). The rigid change control system described in Section D.I.3 applies to any changes required in component design.

D.I.4.4.3 Design of Fuel

Nuclear system requirements for fuel are specified by APED Design Engineering and are transmitted to RF&RD Engineering. These requirements are reviewed and accepted by the Fuel Engineering component of RF&RD. Detail fuel drawings and specifications are produced by RF&RD Engineering and, prior to release for manufacture, are reviewed by APED Design Engineering to assure conformance with mechanical design requirements. Reviews are also conducted by RF&RD Engineering with Manufacturing Engineering (Nuclear Fuel Manufacturing), and Quality Control (Nuclear Fuel Manufacturing), to assure compatibility with manufacturing and quality control technology and capability.

Product requirements are transmitted from RF&RD Engineering to Nuclear Fuels Manufacturing through issuance of engineering instructions which specify applicable drawings and specification. Changes to drawings are made through use of engineering change notices (ECN) that are reviewed for consistency within RF&RD Engineering and R&FMO components. Prior to release for manufacturing, a review is held with APED Design Engineering for consistency where required.

Design review, release, and change control systems have been developed and implemented and are documented in formal document systems.

D.I.4.4.4 Design of Instrumentation and Controls

The system design controlling documentation for items supplied by NID consists of design specifications, instrument engineering diagrams, function control diagrams, piping and instrument diagrams, specification control drawings, instrument data sheets, and general requirements incorporated in an instrumentation and control purchase specification prepared by APED Design Engineering. The instrument data sheets define the required environment, ranges, accuracies, and set points of instruments required by the system design. The responsible APED engineer is required to obtain review and approval of the drawings and documents he has initiated. Reviewers normally must include all APED engineers responsible for equipment with which there exists an interface. The controlling documents are measured against the requirements of the station data sheets, transient data sheets, and the nuclear engineering data book. After the overall system design review, the final review, and approval of the completed documents is by the responsible APED engineer who reviews to determine that the approved changes have been accurately and properly made. Distribution of the approved documents is to the responsible

APED engineer, to the unit manager of the APED units responsible for them, to the APED requisition engineer, and to NID through APED Purchasing.

Upon receipt of the APED design controlling documentation from APED Purchasing, NID Marketing reviews the documentation package for completeness and technical content and transmits copies to NID Quality Control, and to NID Requisition Engineering. NID Requisition Engineering performs the detail design which encompasses the generation of manufacturing drawings, drawings for customer use, purchase part drawings, and instruction manuals. The detail design makes use of NID standard products and purchased components. The NID standard products are designed to a NID approved functional specification, qualified for performance and design adequacy by a separate testing group and reviewed for critical applications (if appropriate) by a safeguards specialist. The purchased components are bought by NID Manufacturing to purchased part drawings prepared by NID Engineering and controlled by quality control procedures similar to those discussed above for APED purchased (engineered) equipment.

Certain NID documents require review and approval by APED Design Engineering prior to the start of manufacture. Upon receipt of this approval, NID Requisition Engineering transmits the production drawings along with any special assembly and testing instructions to NID Manufacturing. NID Manufacturing orders the necessary materials and plans the shop operations. NID Quality Control plans the inspection, testing, and other quality control requirements. During and following production, NID Quality Control monitors the products to assure conformance to the drawings and specifications.

The design change control system for changes generated within NID (and which do not affect APED-approved documents), is initiated with the generation of an engineering change notice (ECN) by the responsible NID engineer, which is reviewed and approved by NID Engineering Release Control, and by the cognizant NID design engineer. NID Manufacturing, upon receipt of the engineering change notice, acknowledges receipt and implements the change. Those changes which affect APED approved documents are transmitted to APED by NID Marketing. APED approved documents may not be changed without the approval of APED Design Engineering.

D.I.4.5 Field Change Control

Field changes fall into two general classes; first, those generated by design changes originating in the home office; and second, those initiated in the field as a result of unique field conditions. Design changes originating in the home office are generally the result of changes in licensing requirements, changes in customer requirements, or information feedback from the construction of other stations or components being constructed, tested, started up, or in operation. In this case, the responsible design engineer generates a field disposition instruction (FDI) which defines in detail the component or components affected, the changes to be made, the parts

which must be replaced, and the disposition of parts replaced. The responsible design engineer is also responsible for providing instructions for the manufacture or procurement of the replacement parts, and for assuring that instructions are issued for other projects requiring such changes. Review and approval of the FDI is by the responsible design engineer, the requisition engineer, and project management.

Field changes initiated by field organizations are generally the result of deviations from the expected construction conditions. The field organization generates a field design change notice (FDCN) which identifies the deviation(s) and the proposed method for making changes in the established design to compensate for the deviation. Design Engineering is responsible to review the proposed field design change for compliance with the established criteria and the performance and functional design requirements. When a proposed method for correction does not comply with these criteria and requirements, it is the responsibility of Design Engineering to disapprove the FDCN and propose an alternate, acceptable solution to the problem. Final approval of the FDCN for a given project is by the responsible design engineer, the requisition engineer, and project management.

When an FDCN indicates an inherent design problem which will affect more than one project, the responsible design engineer issues appropriate FDI's to effect changes in other projects where the work has already been completed, and initiates and institutes changes to the basic design of stations where construction has not reached the same stage of completion.

D.I.5 NED Manufactured Products

D.I.5.1 Product Scope

NED manufactured products include such items as control rod drives, control rods, steam separators, reactor servicing equipment, nuclear fuel, and instrumentation and control systems.

D.I.5.2 General

A quality system encompassing the effort, equipment, procedures, policies, and management required to manufacture and deliver quality products is formally documented and implemented within the manufacturing components of NED. Prime elements of the manufacturing quality system include:

Preproduction quality evaluation

Quality planning

Purchased material control

Product and process control

Inspection and test

Discrepant material control

Training, qualification, and certification of personnel

Quality information equipment control

Quality information feedback and corrective action

Work instructions, procedures, and drawings

Identification, status, and control of material, parts and components

Special processes control

Shipment release control

Handling, storage, preservation, packing and shipping control

Related activities

D.I.5.3 Preproduction Quality Evaluation

Preproduction quality evaluations are conducted jointly by engineering, manufacturing, and quality control components on all new and revised product designs prior to release for full production. Evaluations are conducted to assure that:

Design quality requirements are clearly defined

The required manufacturing equipment, processes, and methods are available or are developed to a capability consistent with product quality requirements

The quality system by which required product quality characteristics are attained, maintained, and measured, is planned, documented, and implemented

The required personnel, facilities, equipment, and materials are made ready for production

D.I.5.4 Quality Planning

Product and process quality planning is provided by NED quality control components for all major items and manufactured or procured by manufacturing components to assure conformance with applicable drawings, specifications, acceptance criteria, and special instructions. Formally documented product and process quality planning is oriented toward assuring product safety, reliability, and compliance with applicable codes and standards requirements. The planning emphasizes prevention of discrepancies; control of

materials, products, processes and procedures; corrective action; and appraisal of product quality characteristics.

Typical planning documents generated are:

Master product quality control plans

Process quality plans

Product quality control plans

Receiving inspection plans for standard raw material

Receiving inspection plans for specific parts

Purchased material quality control plans

uality control inspection instructions

Quality control test instructions

Quality inspection standards

Audit plans

Quality standing instructions (detailed operating instructions)

Manufacturing administrative and operative procedures

D.I.5.5 Purchased Material Control

Purchased material control is exercised for the purpose of assuring that vendor supplied materials, equipment, and services are provided at proper quality levels and conform to the specifications of purchase.

Requests for production material are reviewed by quality control personnel before submittal for purchasing, and necessary quality requirements are added to the material request. These requirements include, where applicable, review of vendor planning, documentation required, objective evidence of quality to be supplied by the vendor, hold points for source inspection by NED quality control personnel, qualification and certification requirements, and receiving inspection coding.

Vendors selected are qualified by virtue of past performance, vendor survey, or a specific program in which the vendor is qualified by demonstrating the ability to perform. Vendors are required to maintain a quality system commensurate with design and quality assurance requirements. On certain designated items, Quality Control maintains surveillance of fabrication and testing at vendors' stations and furnishes reports of source inspections and tests.

Incoming vendor items are inspected and approved by NED quality control components in accordance with the quality requirements as specified in the purchase order, material specifications, standard practices, and receiving inspection plans.

Acceptable parts and materials are appropriately identified at receiving inspections and released for production control accumulation.

D.I.5.6 Product and Process Control

Product and process control activity is oriented toward assuring that quality planning is properly understood and followed; equipment and processes meet quality requirements; materials, parts, and components are identified and controlled; and only finished goods that meet all quality requirements, including those of applicable codes and standards, are released for shipment. Audits are conducted to assure that quality planning is properly implemented and that products meet applicable quality requirements.

D.I.5.7 Inspection and Testing

Receiving, inprocess, and final inspection and testing are performed in compliance with planned, documented inspection and test instructions, and standards to assure conformance with applicable drawings, specifications, and special instructions. As required by quality planning, inspection and test results are documented to provide objective evidence that quality requirements have been satisfied.

D.I.5.8 Discrepant Material Control

Quality control components document, maintain, and implement systems for the handling and control of discrepant material. These systems provide for the physical identification, segregation, documentation, and timely disposition of nonconforming materials and products. Whenever a process is performed in such a manner as to cast doubt on the acceptability of a product, or whenever a material or product is found by inspection or test not to meet applicable drawings or specifications, the affected items are rejected and processed in accordance with the discrepant material control system.

D.I.5.9 Training, Qualification and Certification of Personnel

Formal programs are provided for the training, qualification, and certification of shop operations and quality control personnel to assure that these individuals are qualified and certified to perform key quality related activities including those identified in applicable codes, standards, and regulatory requirements.

D.I.5.10 Quality Information Equipment Control

A system has been developed, documented, and implemented to assure that gauges, instruments, and measuring devices necessary to control

and measure product quality are calibrated, adjusted, repaired, or replaced in a systematic and timely manner. The system provides for calibration of such equipment against measurement standards that are traceable to national standards.

D.I.5.11 Quality Information Feedback and Corrective Action

Quality information feedback systems make provision for timely reporting to management and others having quality system responsibility, the overall status of product quality and quality related plans and programs. Quality information feedback provides input for the initiation of corrective action deemed necessary to correct or improve quality system performance.

D.I.5.12 Work Instructions, Procedures, and Drawings

The NED quality system provides for the documentation in formally controlled document systems, of those work instructions, procedures, and drawings having an effect on product quality.

D.I.5.13 Identification, Status, and Control of Materials, Parts, and Components

The NED quality system provides for appropriate identification and control of materials, parts, and components from receiving inspection through fabrication steps to assure that only finished goods that meet quality requirements are released for shipment. The system is designed to assure that incorrect or defective items are not incorporated in the product, and that the inspection and test status of raw, inprocess, and finished goods are known at all times.

D.I.5.14 Control of Special Processes

Special processes such as welding, heat treating, cleaning, and nondestructive testing are controlled by formalized procedures and practices. The quality system provides for performance of such special processes by qualified personnel using qualified procedures in accordance with applicable codes, standards, regulatory criteria, and other special instructions.

D.I.5.15 Shipment Release Control

A product quality checklist system is used by quality control components to assure that significant quality related work elements have been completed prior to product shipment release. A responsible engineer reviews and verifies by signature that items on the checklist have been accomplished and that the product is suitable for shipment.

D.I.5.16 Handling, Storage, Preservation, Packing, and Shipping Control

The quality system for NED manufactured products provides for control of the handling, storage, preservation, packing, and shipping of

supplied systems and equipment to prevent their inadvertent damage or deterioration prior to delivery to the owner.

D.I.5.17 Related Activities

Related quality control activities such as special quality studies, postproduction quality service, record accumulation and maintenance, and management of the quality control functions are a part of the quality system and are documented in formal document systems.

D.I.6 APED Purchased (Engineered) Equipment

D.I.6.1 Product Scope

APED purchased equipment includes such items as the reactor pressure vessels and internals, pumps, motors, piping, valves, and heat exchangers.

D.I.6.2 General

The quality system for engineered equipment is initiated with the review of purchase specifications and/or purchased part drawings, and is implemented during vendor selection, and during phases of vendor design, fabrication, test, inspection, cleaning, and packaging.

Quality system policies, procedures, and instructions are documented for the following:

Advance review of purchase specifications and/or purchased part drawings

Quality control plans for engineered equipment

Implementation of quality control plan on bid request and purchase orders

Vendor evaluation and selection

Product quality control checklists

Document change control

Review and approval of special processes

Nonconforming material control

Quality information feedback

D.I.6.3 Quality Planning

Quality control plans are issued by Quality Control - Engineered Equipment for all major components, are directly referenced in bid requests and purchase orders, and are designed to complement and assure conformance with the purchase specification and/or purchased

part drawings. Design Engineering reviews and approves the quality control plans prior to issuance.

Quality control plans require that the vendor establish and maintain a quality system commensurate with the complexity and importance of the component. Major equipment suppliers are required to demonstrate adequate control in areas such as:

Document review and change control

Preproduction quality planning

Purchased material quality control

Material identification

Control of processes such as welding, heat treating, cleaning, and nondestructive examination

Qualification of equipment, procedures, and operators

Product quality records

Deviating material

Calibration of measurement equipment

Handling, storage, preservation, packing, and shipping

D.I.6.4 Quality Implementation

Responsibility for quality planning and implementation for purchased equipment is assigned to Quality Control - Engineered Equipment. Quality control engineers and quality control field representatives within Quality Control - Engineered Equipment are typically assigned responsibilities as listed below:

Quality Control Engineers

Conduct preprocurement reviews with Design Engineering and Procurement to ensure clear understanding of quality requirements

Provide quality related review of design specifications, drawings, manufacturing procedures, and quality control procedures

Establish quality control plans and checklists which define quality audit requirements to vendors and the quality control representative

Conduct or direct preproduction reviews with vendor personnel to assure clear and mutual understanding of quality requirements

Provide technical guidance to the quality control field representatives

Audit the vendor quality control program before and during fabrication

Review quality problems encountered during fabrication and during or after installation of equipment, and initiate corrective action

Quality Control Field Representatives

Review purchased part drawings and specifications.

Conduct preproduction reviews with vendor personnel to assure mutual understanding of quality requirements.

Review vendor's detail drawings and manufacturing and quality control procedures.

Review vendor's detailed fabrication process sheets to assure proper sequencing and adequate inprocess inspection, test, and control.

Witness and audit the various qualifications, tests, and inspections.

Audit cleaning, preserving, packing, and packaging activities.

Audit vendor conformance with established procedures such as:

Product and process quality planning

Document change control

Deviating material control

Material identification and traceability

Complete product quality checklists prior to release of assigned equipment.

D.I.6.5 Discrepant Material Control

APED Design Engineering and Quality Control - Engineered Equipment review and approve all vendor deviations for specified requirements. When it is inappropriate to revise the approved drawings, specifications, or procedures, a deviation disposition request (DDR) form or equivalent vendor form is used to formally document and control deviation approvals.

D.I.6.6 Product Quality Checklists

A product quality checklist system is used by Quality Control Engineered Equipment as a means of assuring systematic verification of vendor conformance during fabrication and prior to release for shipment. The checklists record the degree and nature of NED verification during and after fabrication, test, and inspection of purchased equipment.

Examples of items to be verified by NED Quality Control are:

Use of approved drawings and specifications

Use of approved welding, heat treating, cleaning, and nondestructive examination procedures

Procedure and operator qualification

Material test results

Dimensional measurements

Radiographs and other examination results

Performance testing

Heat treatment

Cleaning

Deviation approvals

Product quality records

On selected equipment, the use of product quality checklists is extended to include NED verification of materials such as castings, forgings, pipe fittings, and plate.

The APED quality control representative dates and initials each checklist. All items on the checklist must be verified prior to APED release for shipment unless specific arrangements are made to complete fabrication and/or verification after shipment. Such arrangements must have the advance approval of the project manager and the manager, Quality Control - Engineered Equipment. A copy of each completed checklist is forwarded to the project manager and other NED personnel, as required.

The final checklist can be used by the owner and the project manager as objective evidence of compliance with the planned quality requirements.

D.I.7 NED Installation Control

D.I.7.1 General

It is the responsibility of NED to deliver systems and components to a predesignated location. It is the responsibility of the owner to receive and install the supplied equipment under the technical direction of NED. The following paragraphs identify those contributions to installation quality control made by NED.

D.I.7.2 Component Receiving, Inspection, Handling, and Storage Control

Instructions for site receiving, inspection, handling, and storage control are provided by NED for the equipment and components supplied by NED.

D.I.7.3 Installation Specifications and Instructions

NED supplies separate installation instructions and specifications for reactor recirculation piping, primary steam piping to second isolation valve and reactor vessel internals and appurtenances. Also, necessary installation instructions are provided in equipment manuals and/or system specifications for assemblies and components supplied by NED. These system specifications are issued by the responsible Design Engineering component and define the requirements which are to be met for equipment installation. Installation instructions, which are reviewed by Quality Control and reviewed and approved by the responsible Design Engineer component, define the methods for installing the equipment to meet the requirements of the specification.

D.I.7.4 Technical Direction

The site resident manager and staff provide technical direction of the installation of NED supplied equipment and components in accordance with approved specifications and instructions. Technical direction is technical guidance, advice, and counsel, based upon current engineering and installation practices, given to the owner's staff. The objective of the technical direction is to provide reasonable assurance to NED that supplied nuclear systems and equipment are properly installed by the owner or his agents.

D.I.7.5 Quality Planning and Implementation

Quality planning is provided by the owner or his agents to meet the quality criteria objectives of the installation specifications and instructions provided by NED.

Responsibility for the NED contribution to installation quality control, which is to provide quality requirements and audit of the owner or his agents' quality activity, is assigned to Quality Control - Engineered Equipment. Within this organization, quality control engineers are assigned responsibility for planning and directing the

installation-related quality activity. Quality control site representatives are provided and are responsible for assuring proper understanding of, and conformance with, the requirements of the NED quality system. Activities assigned to the quality control engineer and the quality control site representative are as follows:

D.I.7.5.1 Quality Control Engineer Activity

Conducts preinstallation reviews with the responsible Design Engineering component, installation technical specialists, and quality control management to ensure clear delineation and understanding of NED quality requirements

Provides quality related review of detailed drawings, specifications, installation instructions, and procedures

Reviews the adequacy of the installation quality control plan with respect to NED supplied equipment

Develops plans for APED audit of installation quality activities

Provides guidance to the quality control site representative

D.I.7.5.2 Quality Control Site Representative Activity

For NED supplied equipment, conducts quality related review of NED approved drawings, specifications, instructions, procedures, and manuals.

Conducts preinstallation reviews with APED site resident manager, technical specialists, to ensure mutual understanding of quality requirements.

Reviews NED installation instructions and owner/constructor's procedures.

Audits conformance with approved installation procedures and inprocess controls.

Audits site quality control records.

Witnesses and/or audits various installation test, qualifications, and inspections.

Ensures orderly processing and formal disposition of quality related discrepancies and deviations.

D.I.8 Preoperational Testing

D.I.8.1 General

The nuclear system is made operational by the owner under the technical direction of NED. NED provides technical direction of the

owner's preparation of preoperational test procedure for NED-supplied systems and equipment.

Preoperational test procedures identify the systems and equipment which must be tested and state the requirements of the tests necessary to assure safe performance during testing.

Preoperational test instructions provide the necessary information and the essentail steps to be taken to fulfill the requirements of the preoperational test specifications.

D.I.8.2 Preoperational Test Implementation

As provided by the contract, NED supplies field engineers with extensive product knowledge and wide startup experience, to provide technical direction of preoperational tests. Technical direction for preoperational testing is technical guidance, advice and counsel, based on current engineering and test practices, given to the owner's staff.

D.I.8.3 Preoperational Testing Check Off

Upon completion of preoperational tests, the NED technical specialist and an owner representative formally verify that the tests have been completed and that the results are in accordance with acceptance criteria of the test procedure.

D.I.9 Station Startup

D.I.9.1 General

Initial fuel loading, nuclear system startup, and operational testing is performed under the technical direction of NED personnel. To facilitate technical direction of initial fuel loading and power testing, startup test specifications, analyses, and instructions are provided by NED.

D.I.9.2 Startup Test Specifications, Analyses, and Instructions

Startup test specifications define the minimum test program for safe and efficient startup, and authorize the required performance of the described tests. The specifications limit and define the freedom for changes during the startup test activities and are reviewed, and approve by the responsible design engineering component. Each required test must be performed to the extent specified.

Startup test analyses contain the results of analyses made to facilitate startup testing activities required by startup test specifications.

Startup test instructions present the recommended test method and describe the steps necessary to perform the test defined in the startup test specification. Other test methods may be employed. However, the resulting data must be equivalent in quality and

quantity to the data which would result from the recommended test method. The startup test instructions also contain criteria for judging the test results, where applicable, and data and calculation sheets for site analysis of the data.

D.I.9.3 Technical Direction

NED supplies field engineers with extensive product knowledge and wide startup experience to provide technical direction for the startup test program. Results of the startup test program are analyzed at the reactor site as the data become available, and periodic reports of the results of the program are issued during the course of testing activities.

D.I.9.4 Startup Testing Check Off

Upon completion of startup testing, the NED field engineers and an owner representative formally check off that the startup tests have been completed, and report that the results have met the intent of the specifications, analyses and instructions.

D.I.10 Quality Records

D.I.10.1 General

For equipment which is under the NED scope of supply, quality related records will be provided to, or maintained for, the owner in accordance with the contractual agreement with the owner.

TABLE D.1-1

PROJECT FUNCTIONAL RESPONSIBILITY SUMMARY
PILGRIM NUCLEAR POWER STATION

Project Part		Quality Control	Quality Assurance	Engineering Design	Procurement	Mfg Inspection	Mfg Expediting	Construction QA	Construction QC	Systems Checkout and Acceptance Testing
I	Nuclear Power Station*	B	Entergy	B	B	B	B	B	B	Entergy$^{(1)}$
II	Nuclear Stream Supply System	GE	Entergy	GE	GE	GE, B	GE, B	B$^{(2)}$	B$^{(2)}$	Entergy$^{(2)}$
III	Turbine Generator Equipment	GE	Entergy	GE	GE	GE, B	GE, B	B$^{(2)}$	B$^{(2)}$	Entergy$^{(2)}$
IV	Nuclear Fuel	GE	Entergy	GE	GE	GE Entergy	GE	-	-	Entergy$^{(2)}$
V	Recirc., RHR RWCU, Core Spray Piping Replacement (RFO #6)	GE	GE Entergy	GE	GE	GE Entergy	GE	GE	GE Entergy	Entergy$^{(2)}$

NOTES:

Entergy = Entergy Corporation
B = Bechtel Corporation
GE = General Electric Company
*Except Parts II, III, and IV

(1) Technical direction provided by Bechtel Corp.

(2) Technical direction provided by GE Corp.

ATTACHMENT D.II

BECHTEL QUALITY ASSURANCE PROGRAM PILGRIM NUCLEAR POWER STATION

D.II.1 Organization

The Bechtel organization employed for implementation of its Quality Assurance Program on the Pilgrim Nuclear Power Station Project during engineering design, procurement, and construction phases is shown on Figures D.2-3 and D.II-1, Bechtel Organization Chart. Personnel having significant Quality Assurance related functions include:

a. Design Phase

   Project Engineering Team

   Chief Engineers and their Technical Staff

   Cognizant Engineering Manager

   Manager Quality Assurance

   Quality Assurance Coordinator

   Metallurgy and Quality Control Department

b. Procurement Phase

   Shop Inspectors

   Chief Inspectors

   Project Engineering Team

   Chief Engineers and their Technical Staff

   Cognizant Engineering Manager

   Manager Quality Assurance

   Quality Assurance Coordinator

   Metallurgy and Quality Control Department

c. Construction Phase

   Field Engineers

   Quality Control Engineer

   Project Field Engineer

Material Supervisor

Quality Assurance Engineer

Project Superintendent

Project Engineering Team

Chief Engineers and their Technical Staff

Cognizant Engineering Manager

Manager Quality Assurance

Quality Assurance Coordinator

Metallurgy and Quality Control Department

The Project Engineering Team prepares drawings, specifications, purchase requisitions, bid evaluation and all other tasks associated with the Bechtel engineering design portion for the Pilgrim Nuclear Power Station Project. The Project Engineering Team also prepares the Quality Assurance Program that will be carried out in the balance of plant and construction phases of the project. The following basic principles are applied:

a. The Project Engineering Team has primary responsibilities for quality in the design phase

b. Vendors and Subcontractors have primary responsibilities for quality of materials, equipment, and services furnished by them

c. The Bechtel project field organization supervised by the Project Superintendent has primary responsibility for quality of construction performed directly by Bechtel

d. One or more levels of inspection are provided as required within the organization having primary quality control responsibilities

The Bechtel Quality Assurance Program also provides for at least one level of monitoring and verification by individuals, not under the direct control of the group having primary responsibility for quality control (e.g., the Quality Assurance Engineer monitors construction, Bechtel Shop Inspectors monitor Vendors, etc.). Beyond this, Quality Assurance Program audits of engineering and field operations are carried out under the direction of the Quality Assurance Coordinator. The Manager Quality Assurance has broad surveillance of the program for this and all other nuclear projects.

D.II.2 Program

The Bechtel Quality Assurance Program for Pilgrim Nuclear Power Station is carried out in accordance with the Nuclear Quality Assurance Manual, modified, to meet specific Applicant requirements. This manual describes the overall Program and identifies management and administrative procedures and individual responsibilities. General instructions, guidelines, and check lists for inspections are contained in the following documents.

a. Bechtel Procurement Department Inspection Manual

b. Bechtel Field Inspection Manual

c. Bechtel Field Procurement Procedures

Structures, systems and components to be covered by the Quality Assurance Program are identified by a Q-List which is prepared by the Project Engineering Team, and reviewed and approved by Chief Engineers. The specific level of inspection and control afforded items on the Q-List is determined on a case by case basis by the Project Engineering Team, through consultation with the Chief Engineers and Bechtel's technical specialists. Factors considered in establishing the degree of control include: Nature of the item, importance of the item to plant safety and reliability, previous experience with this or comparable items, capabilities of potential vendors or subcontractors and requirements of applicable codes or standards. Where necessary, Bechtel engineering documents include specific procedures and requisites for the production and Quality Assurance of the item. Requests may be made for documented procedures from equipment vendors and subcontractors for written Bechtel approval.

In implementing the program in the construction phase, Quality Assurance related responsibilities are assigned to the following personnel:

a. The Project Field Engineer supervises Quality Control inspection at the jobsite. In carrying out this assignment, he assigns qualified Field Engineers to perform Quality Control inspections. He supervises the preparation of inspection check lists, verifies accuracy and completeness of inspection reports and ascertains that defects are removed and that repairs are carried out in accordance with applicable approved specifications, instructions, and procedures

b. The Quality Control Engineer reports to and assists the Project Field Engineer in carrying out Quality Control inspection responsibilities. He normally is assigned responsibility for review of inspection reports, coordination, training, and advising Field Engineers performing Quality Control inspection assignments, coordination of testing laboratories, and overall detailed

execution of field inspection and maintaining the field QC files

c. Field Engineer/Inspectors carry out the inspection assignments and are responsible for filling out the appropriate inspection forms. Field Engineer/Inspectors function on a disciplinary basis, e.g., mechanical equipment, civil/structural, electrical/power, instrumentation/control, welding/metallurgy. The number of Inspectors assigned depends upon the requirements of the variable Quality Control inspection workload and construction schedule. Inspectors have access to all the design drawings, applicable codes and sampling and testing and are thoroughly familiar with the requirements

d. The Quality Assurance Engineer is the field representative of the Project Engineering Team reporting to the Project Engineer. He provides surveillance of engineering and Quality Control activities in the field. The Quality Assurance Engineer does not perform inspection per se; he reviews and approves selected inspection reports, monitors, the permanent field QA/QC documentation files, and monitors important construction inspection. The Quality Assurance Engineer has the authority to stop the work for which Bechtel Corporation is Prime Contractor in the event of nonconformance with drawings, specifications, and procedures established for structures, systems, and units on the Q-List. He serves as field contact with the Applicant's Quality Assurance organization and others concerned with Quality Assurance in the field. He receives supervision and technical support from the Project Engineer

D.II.3 Design Control

Several levels of design review and approval are applied to significant design aspects of the Bechtel work. These standard practices include:

a. Checking and review by design and engineering level personnel within the Project Engineering Team having technical qualifications comparable to those of the engineer or designer who originated the work

b. Review and approval by the originating engineer's Design Group Supervisor

c. Review and approval by the Project Engineer

d. Review and/or approval by the appropriate Chief Engineer or certain key designs and calculations, specifications, and drawings, e.g., electrical single line drawings, containment design specification

In implementing Design Control, Quality Assurance functions performed by the Chief Engineers and Cognizant Engineering Manager are as follows:

Chief Engineers

The Chief Engineers qualify for independent review of engineering, since the Project Engineering Team is under the sole direction of the Project Engineer. The Chief Engineers coordinate and assure necessary technical review by specialists and consultants. Chief Engineers may delegate their review responsibilities to qualified specialists on their staff

Cognizant Engineer Manager

The Cognizant Engineering Manager provides management guidance and surveillance of the Project Engineering activities. His QA related functions involve verification through project reviews that Project Engineering is carrying out its QA responsibilities.

Drawing and specification which have a bearing on the Nuclear Steam System Supplier's equipment or result from criteria supplied by him are routinely submitted to the Supplier for review. In a similar manner, information developed by the Nuclear Steam System Supplier affecting the Bechtel-NSSS interface is submitted by the Supplier to Bechtel for review. Drawings specifications and procurement packages are routinely submitted to the Applicant for appropriate review and approval.

The Project Engineering Team employs several documents to establish design requirements for the project. These documents include or incorporate applicable AEC regulatory requirements, and design bases included in the license application; basic Applicant furnished data defining plant requirements; basic engineering data amplifying the basic Applicant directed project data; NSSS Supplier furnished criteria and data; and project criteria sheets prepared for each discipline.

D.II.4 Procurement Document Control

Technical aspects of Procurement Documents are prepared by the Project Engineering Team in accordance with the procedures described in the preceding section. Appropriate Vendor Quality Assurance Program requirements are incorporated in the Procurement Documents. Provisions are made for periodic and final inspection in Vendor shops as appropriate. All equipment procurement and important material procurement, whether carried out by Home Office Procurement Department or the Field Procurement Organization employs specifications and Quality Assurance Requirements established by the Project Engineering Team.

D.II.5 Instructions, Procedures, and Drawings

General procedures for carrying out engineering, procurement, and construction are contained in the following manuals and documents:

Project Engineer's Manual containing requirements for carrying out engineering activities

Nuclear Quality Assurance Manual defining responsibilities and outlining Quality Assurance activities and procedures

Field Inspection Manual describing general guidelines and procedures for Field Inspection

Procurement Department Inspection Manual containing shop inspection instructions, guidelines, and procedures

Field Procurement Procedures specifying field purchases and material receiving inspection

These are supplemented by specific instructions and procedures which may be prepared by Engineering, obtained from Vendors, or prepared by Vendor, Subcontractor or Bechtel Construction personnel as required. When appropriate, such instructions and procedures are reviewed and approved by the Project Engineering Team and/or Bechtel specialists.

D.II.6 Document Control

The Bechtel Quality Assurance Program includes a comprehensive system of Document Control which assures that documents and changes thereto are reviewed for adequacy in accordance with procedures previously described. Approved drawings are promptly issued to organizations and individuals carrying out the work and to those responsible for inspection. Existing field procedures control the distribution of approved drawings, and assure that obsolete drawings are appropriately voided or destroyed. Changes to approved drawings whether made by the Project Engineering Team or Field Engineering are reviewed and approved by the Project Engineering Team in accordance with procedures for review of the initial issue. Where significant changes are involved, review by cognizant Chief Engineers and/or technical specialists is performed. Drawing control logs indicating current status of drawings, and specifications are prepared monthly and distributed to responsible Project Engineering and Field personnel. Shop Inspectors are advised of current status of approved Vendor shop drawings. The Bechtel Quality Assurance Engineer monitors field engineering activities to verify that field engineering changes are reviewed and approved by the Project Engineering Team.

D.II.7 Control of Purchased Material, Equipment, and Services

The Bechtel Quality Assurance Program includes a comprehensive system to assure that purchased material, equipment, and services conform to the procurement documents. It provides for evaluation, where

appropriate, of Vendor's Quality Assurance Program and preparation of Procurement Specifications incorporating Quality Assurance requirements. The Quality Assurance requirements include an appropriate Vendor Quality Assurance Program, purchaser surveillance as required, Vendor preparation and maintenance of appropriate test and inspection records, certificates and other Quality Assurance documentation, and Vendor submittal of Quality Control records considered necessary for purchaser retention to verify quality of completed work.

Bechtel Shop Inspectors review and verify Vendor Quality Assurance records and prepare reports documenting Vendor Data not submitted to purchaser. Where Bechtel shop inspection is performed, a final inspection of the finished item is carried out in Vendor's shop prior to release for shipment. Bechtel Field Procurement Procedures provide for receipt inspection by Material Supervisor on all materials and equipment delivered to the jobsite. For significant items or critical materials which form parts of Q-List systems, Bechtel Field Engineers carry out an independent receipt inspection and prepare receiving reports which supplement receiving reports required by standard Field Procurement Procedures.

Bechtel Procurement Procedures also provide for periodic audits by the Inspection Department of Vendor Quality Assurance activities as appropriate.

D.II.8 Identification and Control of Materials, Parts, and Components

As applied to Vendors, appropriate requirements for identification and control of materials, parts, and components will be established through review of Vendor's Quality Assurance Program and procedures. Bechtel Field Procedures and practices incorporate measures for material control, including segregation of non-conforming items, and for marking and identification as required. Traceability will be accomplished where required by code, standard, or specification. In other cases, measures such as physical separation or appropriate marking procedures are used to identify and control materials or components of specific type, specification, class, etc. Positive control of nonconforming items are maintained by suitable markings and/or segregation.

D.II.9 Control of Special Processes

Use of qualified procedures and application thereof as required by established codes and standards are enforced on all Bechtel Vendors, Subcontractors, and Bechtel personnel. For other special processes identified by equipment suppliers or Project Engineering, procedures are prepared by equipment suppliers, Project Engineering, or Field Engineering and approved by appropriate personnel in the organization which identified the process. Personnel performing such operations are trained and carefully supervised by personnel familiar with the specific special process.

D.II.10 Inspection

Bechtel performs periodic and final inspection of Vendor work as described previously. This is normally performed by Shop Inspectors; however, in special cases engineering personnel participate in such inspections. Inspection practices normally include witnessing of significant tests, as appropriate, and requirements for Vendors to accept mandatory hold points where, in the opinion of Bechtel Inspectors, work should not proceed without prior examination by the Bechtel Inspector.

Field operations, in process and final inspection of activities affecting quality are carried out by Bechtel Field Engineers with knowledge of the discipline involved, project requirements, and the inspection process. Field inspection is carried out in accordance with guidelines and procedures contained in the Bechtel Field Inspections Manual, supplemented by approved procedures for special processes and specific project requirements. Field inspection operations are supervised by the Project Field Engineer, assisted by the Quality Control Engineer, and are monitored by the Quality Assurance Engineer. Reports or records of inspection operations are prepared by the Field Engineers/Inspectors, approved by the Quality Control Engineer and accepted and distributed by the Quality Assurance Engineer.

D.II.11 Test Control

Supplier and Subcontractor test operations, including procedures as appropriate, are reviewed in accordance with procurement procedures previously described.

In the field, certain test operations are carried out by Construction organization in the course of plant construction. Subsequent to completion of construction, the Bechtel Startup organization is involved in system checkout and startup operations. Construction tests are normally performed in accordance with standard construction practice or per specific test procedures as specified by the Project Engineering Team.

D.II.12 Calibration of Measurement and Test Equipment

Vendor procedures for control of measurement and test equipment are reviewed as appropriate in evaluating the Vendor Quality Assurance Program. In the field, a standard written procedure is followed to provide control and periodic calibration of special tools, measuring and test equipment.

D.II.13 Handling, Storage, Shipping, and Preservation

Special handling, storage, shipping, and preservation requirements are identified in procurement specifications for Vendor's work. In the field, materials and equipment are stored in accordance with standard procedures as well as specific requirements, and any special procedures issued by the Project Engineering Team.

D.II.14 Inspection, Tests, and Operating Status

Records of inspections and tests are provided in the form of written inspection reports or other appropriate records for QA specified items which show inspection performed and results and clearly identify the item. Specific details for marking, tagging or otherwise indicating inspection and acceptance status depend upon the nature of the work performed. it is the responsibility of the Project Field organization to develop specific procedures to comply with this requirement during construction.

D.II.15 Non-conforming Material, Parts, or Components

The Bechtel Quality Assurance Program provides measures which control materials, parts, or components not conforming to prescribed requirements to prevent their inadvertent use or installation. Materials are physically controlled in accordance with procedures described under Identification and Control of Materials Section. For nonconforming items which may be made usable through rework, repair, or modification of requirements, reports are prepared and submitted to the Project Engineering Team for appropriate guidance. Records of resolution for these cases are prepared and incorporated in Quality Assurance files. Where rework, repair, or approval for use is not feasible, nonconforming materials are either removed from the construction site or utilized in systems or structures where their characteristics satisfy requirements for such systems.

D.II.16 Corrective Action

The Bechtel Quality Assurance Program incorporates procedures for identification and reporting of situations which are deemed adverse to quality through preparation of Significant Deviation Reports. These include reports of significant failures, malfunctions, deficiencies, deviations, defective material, etc. Routine occurrences of rework generally anticipated for the activity involved are not normally included in the corrective action program. These reports are prepared by either Project Field Engineering or the Quality Assurance Engineer, and reviewed by the Project Engineering Team and Engineering Specialists. These reports provide for documentation of findings and corrective measures taken. Summaries of Significant Deviation Reports will be incorporated into Project Quality Assurance records.

D.II.17 Quality Assurance Records

Copies of all reports described in the preceding sections, as prepared by Bechtel and obtained from Vendors and Subcontractors are collected in project Quality Assurance files. These files are available for audit by Applicant during the design and construction period and are turned over to the Applicant at the completion of the Bechtel contract. Detailed Bechtel Quality Assurance audit reports are not included in such files; however, summary records and conclusions will be made available.

D.II.18 Audits

The Bechtel Quality Assurance Program includes four specific audit activities:

> Periodic audits of Project Engineering activities and records are carried out by, or under the direction of, the Quality Assurance Coordinator, and under the supervision of the Manager Quality Assurance
>
> Bechtel Shop Inspectors carry out periodic audits of Vendor's Quality Assurance Program and records
>
> Quality Assurance Engineer carries out frequent audits of field inspection activities and Quality Assurance reporting
>
> Periodic audits of field Quality Assurance and inspection activities are carried out under the direction of the Quality Assurance Coordinator

All of the above are carried out, on a sampling basis, periodically during the design and construction period. At the completion of construction for systems and structures subject to the Quality Assurance Program, a final inspection is performed on the work and associated Quality Assurance records to assure necessary inspections and records have been prepared and are on file. A final inspection report is prepared confirming this final examination audit.

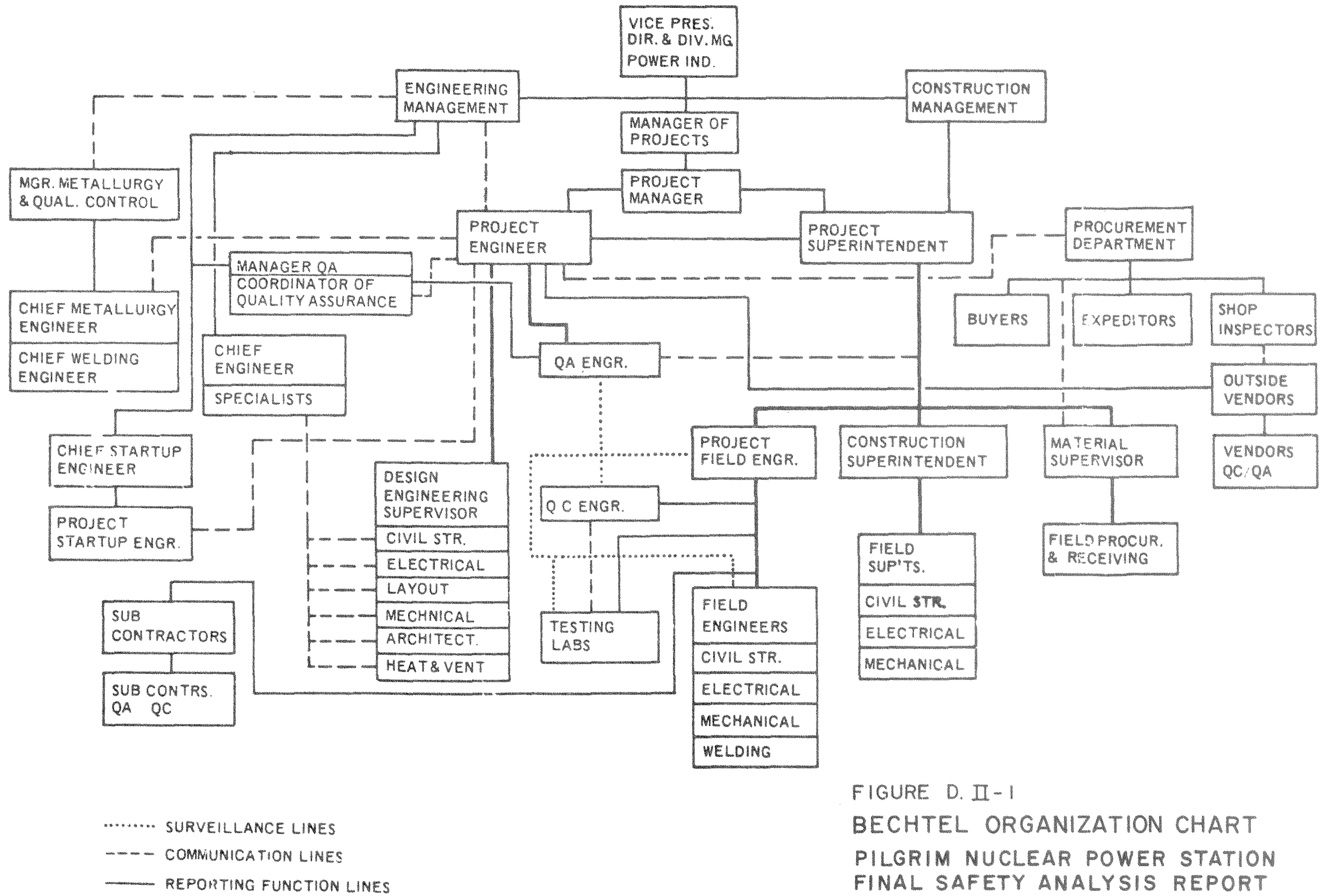


FIGURE D. II-1

BECHTEL ORGANIZATION CHART

PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

ATTACHMENT D.III

SITE HANDLING AND STORAGE OF NUCLEAR STEAM SUPPLY SYSTEM EQUIPMENT PILGRIM NUCLEAR POWER STATION

D.III.1 Introduction

Handling and site storage are important parts of Quality Control and Quality Assurance. Procedures are developed and revised, as necessary, throughout the life of Pilgrim Project concerning onsite storage.

D.III.2 Nuclear Steam Supply System

The responsibility for handling and site storage of Nuclear Steam Supply System (NSSS) equipment rests with Bechtel. Aids and guides are furnished by GE for the handling and storage of APED supplied equipment as follows:

D.III.2.1 General

Three storage categories are assumed; storage inside (heated), storage inside (unheated) and storage outside (unheated).

An itemized list defining where equipment shall be stored is included. It is preferable that those items marked with an asterisk be stored in the category in which they are listed; however, they may be stored in the next lower category provided that storage criteria are met.

Small parts, whether stored inside or outside, which are of the same system or Master Parts List (MPL) category must be grouped together to prevent loss. A typical grouping would be fuel orifices, jet pump sensing lines, jet pump riser braces, thermal sleeves, jet pump supports, core spray clamp, incore tie bars, access hole covers, etc., all the above being vessel internal parts in the nuclear boiler system.

All equipment, whether crated, boxed or stored inside without covering, shall be plainly tagged or marked with purchase order number, equipment piece or master parts list number, such that the equipment is identifiable without opening or disturbing crating, boxing or equipment.

D.III.2.2 Storage Inside - Heated

On equipment where internal cleanliness or operation can be adversely affected, the openings shall be plugged, capped or sealed. Butt weld openings of valves, etc., shall have plywood or fiberboard against the prep and shall be tape sealed. Where metal shipping caps were provided, these shall be reinstalled over tape to protect the prep. Small pieces of equipment shall be bagged and sealed in polyethylene or stored in clean, closed boxes. Masking tape is not recommended.

If dirt producing operations are carried on in the storage warehouse, more elaborate measures may have to be taken to prevent contamination of equipment.

All equipment should be skidded, palletted or stored on racks where practical. Vertical, solid shaft motors shall rest upon a base high enough to prevent the shaft extension from touching the floor. Generally, the timber skid upon which the motor is shipped will be sufficient for this purpose.

Care shall be taken in placement of equipment so that damage from fork lifts, etc., is eliminated.

Sufficient measures shall be taken to prevent rodents and termites from damaging or dirtying equipment, either by direct control or wrapping equipment to prevent entry.

Should the warehouse heating system fail, measures shall be taken to keep the temperature of stored equipment above the dew point of the surrounding air, especially motor windings and electronic circuity.

Motors whose operating voltage is 2,400 v or greater shall be inspected regularly and the insulation resistance of the stator windings meggered monthly and values recorded. The value, when measured at room temperature, should not be less than

$$\frac{\text{Rated Voltage}}{(0.75 \text{ X hp Rating}) + 1{,}000} \text{ Megohms}$$

If the insulation resistance if lower than this value, notify the GE Site Quality Control Engineer or other responsible GE site representative immediately.

To properly protect oil lubricated bearings in large motors, it is recommended that the bearing oil reservoirs on sleeve bearing machines be filled to the proper oil level with a good grade of rust inhibiting oil. Consult the manufacturer's instruction book for the proper oil recommendations and other instructions. On two bearing motors with horizontal shafts. The shaft shall be rotated at three month intervals.

Batteries shall be stored off the floor on wood or other suitable insulating material.

D.III.2.3 Storage Inside - Unheated

On equipment where internal cleanliness or operation can be adversely affected, the openings shall be plugged, capped or sealed. Butt weld openings of valves, etc., shall have plywood or fiberboard against the prep and tape sealed. Where metal shipping caps were provided, these shall be reinstalled over tape to protect the prep. Small pieces of equipment shall be bagged or sealed in polyethylene or other approved vapor dirt barrier such as clean, closed boxes. Tape

used to seal openings, bags, etc., should be weatherproof, such as plastic impregnated cloth tape. Masking tape is not recommended.

If dirt producing operations are carried on in the storage warehouse, more elaborate measures may have to be taken to prevent contamination of equipment.

All equipment should be skidded, palletted or stored on racks where practical. Where the floor is damp, the foregoing shall be done.

On motors and equipment with internal cavities, such as valves, measures shall be taken to keep the dew point of the motor or equipment above the dew point temperature of the surrounding air. For motors some safe and reliable heating means to protect the windings shall be used such as heater tape, space heaters, etc. On tightly capped valves, desiccant may be used. External bagging may be necessary to adequately seal valves, because stem packing, etc., may be shipped loose. Desiccant, if used, shall be equivalent to 1/6th the internal volume of the respective equipment. Desiccant such as silica gel is recommended; calcium chloride is strictly prohibited. When desiccant is used, means should be provided to maintain the effectiveness of desiccant.

Instructions pertaining to motors in Storage Inside Heated, shaft support, shaft rotation, filling oil reservoirs, meggering winding insulation, etc., shall apply to this classification, Storage Inside Unheated.

Sufficient measures shall be taken to prevent rodent and termites from damaging or dirtying equipment.

D.III.2.4 Storage - Outside

On equipment where internal cleanliness or operation can be adversely affected, the openings shall be plugged, capped or otherwise sealed. All plugs and caps shall be taped securely to the equipment to prevent their inadvertent removal, using waterproof tape such as plastic impregnated tape. Masking tape shall not be used.

Verification shall be made that all water or other liquid capable of freezing is drained from equipment prior to outside storage; this includes heat exchangers, tanks, etc.

On tightly capped valves and other similar equipment, desiccant shall be used, although external bagging may be necessary to adequately seal valves as stem packing, etc., may be shipped loose. Desiccant, if used, shall be equivalent to 1/6th the internal volume of the respective equipment. Desiccant such as silica gel is recommended; calcium chloride is strictly prohibited.

In most cases, vendors have applied a rust preservative to their equipment prior to shipping. If possible, the preservative shall be left intact.

Equipment such as valves, control rod hydraulic units, control rod drive housings, etc., shall be suitable crated. Suitable crated is defined as:

Equipment to be crated shall have all openings suitable capped
External machined carbon steel surfaces shall be protected by a rust preservative coating such as a heavy, hard drying oil
External crating shall be tight with butted or overlapping joints

Green lumber shall not be used for crating.

Tanks and heat exchangers with tight flange covers and suitable covers for other openings need not be crated. Tarpaulins shall cover the above equipment to form complete drainage of precipitation without formation of any pools. Clearance between the tarpaulin and equipment shall be sufficient to allow air circulation, thus preventing sweating of equipment.

Piping subassemblies containing valves or instruments not subject to damage from outside storage shall be covered with polyethylene or other approved vapor barrier, tape sealed to the piping, and covered with lashed down tarpaulin. Tarpaulin shall provide complete drainage of precipitation without formation of any pools.

Reactor internals such as core structure parts shall have a house or other similar structure constructed around the part. The structure shall then be covered with heavy duty polyethylene sheeting and tarpaulins.

Polyethylene sheeting or other approved vapor barrier material shall be fixed to crating or housing structures so that wind, hail, snow, etc., cannot remove the barrier and water, dust, dirt, etc., cannot enter. Tarpaulins shall cover all polyethylene sheeting or approved substitute so that the sheeting is not in direct exposure to the environment.

Cribbing, racks, or raised platforms shall support equipment for outside storage at a minimum height of 12 in above ground and in such a manner as not to distort equipment.

Hydraulic control units or control/rod drive housings stored outside shall be stored in groups such that handling, covering and stacking rigidity of the individual units in the group is satisfactory.

Equipment stacked for storage shall be so stacked and crated that the crating bears the full weight of the equipment.

Measures shall be taken to prevent vandalism, damage from rodents, termites, or nesting of fowl on equipment.

TABLE D.III-1

CATEGORY I - STORAGE INSIDE
(HEATED) - ITEMIZED LIST

DESCRIPTION

Refueling Platform (control & trolley system only)
Explosive Valve
Flow Control Valve
RCIC Turbine
RHR Pump Motor
Recirc Valve Motor Oper 28 in
Recirc Valve Motor Oper 28 in
Recirc Valve Motor Oper 4 in
Recirc Valve Motor Oper 22 in
Gate Valve - Motor Oper 2 in By-Pass
CRD Drive Water Pump
Motor - RHR Pump
Solenoid Valves
Stabilizing Valves
HPCI Turbine
Pump-Standby Liquid Control
Pump - Core Spray
Pump Motor - Recirc
Pump - Cleanup
Equipment Handling Platform Motors
MG Set
Control Rod Grapples and Tools
Instrument Handling Tool
Channel Transfer Grapple
Power Wrench
Inst. Transformer Cubicle
Exciter and Field Breaker Cubicle
Relay Panels
Control Curtain Grapple
Fuel Handling Grapples and Tools
Underwater TV
Any Hydraulic Press or other Hydraulic Operated Equipment
Hydraulic System Tools (electronic equipment)
Service Platform - Motor Only
Jib Crane - Motor Only
Jet Pump Tools
Elec-Inst. Panels (with inst installed)

TABLE D.III-2

CATEGORY II - STORAGE INSIDE
(UNHEATED) - ITEMIZED LIST

DESCRIPTION
Surveillance Program Samples
Jet Pump Sensing Line Supports
Pump Assembly - HPCI System
Pump - RHR System
Isolation Valves
Flow Elements
CRD Housings
Shroud Head Bolts
Hydraulic Control Units
Control Rods
Incore Guide Tubes
Spacers
Head Cooling Spray Nozzle
CRD Servicing Tools
Control Rod Drives
Control Rod Guide Tube Seal
Incore Guide Tube Seal
Incore Flange Seal Test Plug
Feedwater Valve
CRD Disc Springs
Jet Pump Sensing Line Support Brackets and Braces
Blade Guide
Underwater Lights and Supports
Temporary Control Curtain
Pressure Control Valve - CRD
Relief Valve Main Steam
Vessel Internals, Supports, Suspensions, Restraints, Etc.

TABLE D.III-3

CATEGORY III - STORAGE OUTSIDE (UNHEATED)

In general, items not listed under Categories I and II fall in Category III.

This includes the RHR heat exchanger, recirc loop piping, RCIC pump, regenerative and nonregenerative heat exchangers, primary steam piping, standby liquid control tank, filter demineralizer, pre-coat tank, and similar items.

Most items of non furnace sensitized stainless steel such as recirc pump bowl castings may be stored outside with proper precautions. Specific questions on storage requirements may be directed to the GE Site Representative.

APPENDIX E

STACK RELEASE LIMIT CALCULATIONS FOR PILGRIM STATION SITE

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APPENDIX E

STACK RELEASE LIMIT CALCULATIONS FOR PILGRIM STATION SITE

E.1 ANALYTICAL MODEL

The model described below is primarily concerned with calculating the annual gamma dose rate at ground level resulting from a continuous release of radioactive materials. As a direct consequence, a method is also obtained for calculating the annual average concentration at ground level.

In essence, the gamma dose model considers the integrated dose rate from a continuously distributed gaseous source (the plume). The source distribution is treated by a standard dispersion model that relates the dispersion of airborne particles to downwind distance and to the meteorological conditions that exist during the release intervals. The annual gamma dose is obtained by weighting the gamma dose rate associated with a given meteorological condition by the frequency of occurrence of that condition.

E.1.1 Meteorological Factors

The air concentration per unit amount released at a point (x, y, z) in the cloud at any instant is given by Equation E.1-1 which is Sutton's equation corrected by Cramer[1] for depletion by ground deposition and radioactive decay:

$$(\chi) = \frac{Q_o}{2\pi\sigma_y\sigma_z\bar{u}_h} \exp\left[\frac{y^2}{2\sigma^2_y} + \frac{z^2}{2\sigma^2_z}\right]\left[\frac{Q}{Q_o}\right] \exp\ [\ \lambda t\ ] \quad (E.1\text{-}1)$$

where

(χ) =average air concentration (Ci/m^3 or Ci/cc)

Q_o =release rate (Ci/sec)

$\bar{u}_h$ =average wind speed at height of release (m/sec)

σ_z, σ_y =standard deviation of cloud width in vertical and horizontal direction,

t =time after release $(sec)^{-1}$

λ =radioactive decay constant

The factor Q/Q_o is the correction for cloud depletion due to deposition and is equal to the fraction of the initial amount released which is present at a down wind distance x. According to Watson and Gamertsfelder[2] Q/Q_0 is given by

$$Q/Q_o = \exp\left[- \frac{V_d(\bar{u}_o/\bar{u}_h)}{\bar{u}_o \pi/2} \int_o^{t} \frac{\bar{u}_h}{\sigma_z} \exp\left(\frac{z^2}{2\sigma_z^2}\right) dt \right] \quad \text{(E.1-2)}$$

Values of the deposition "velocity" (V_d) are obtained from Table E.1-1.

It is considered a reasonable approximation to assume that throughout the year all the plumes which travel anywhere within a given sector direction do not have a skewed frequency distribution within the sector. Then, the average cloud concentration in the sector is found by integrating Equation E.1-1 in the crosswind direction and dividing by the sector width.

$$(\chi)_{ave} = \frac{\int_{-\infty}^{\infty} \chi \, dy}{\theta_x} \quad \text{(E.1-3)}$$

$(\theta_x$ sector width)

Equation E.1-3 cannot be integrated since the interelationship between the variables σ_y, σ_z and $\bar{u}_h$ with respect to their average values is not generally known. However, for any specific combination of wind speed and stability at a given downwind distance, all these variables are known and can be treated as constants, and the integration can then be performed. Thus, the average concentration in the sector for all occurrences of any specific condition is given by:

$$(\chi)_{ave}^{ij} = \frac{Q_o[Q/Q_o]}{\sqrt{2\pi}\theta x \sigma_z \bar{u}_h} \exp\left[\frac{-z^2}{2\sigma_z^2}\right] \exp[-\lambda t] \quad \text{(E.1-4)}$$

where:

θ =sector angle ($\pi/8$ or 22-1/2° is used in this report)

x =downwind distance and is equal to $\bar{u}_h t$

σ_z =a function of stability, wind speed, and downwind distance (x)

Thus, the average cloud is seen to have a uniform concentration distribution vertically which is of the Gaussian form.

The standard deviation in the vertical direction is described by Watson and Gamertsfelder[2] as:

$$\sigma_z^2 = \alpha[1 - \exp(-k^2t^2)] + bt \quad \text{stable condition} \tag{E.1-5}$$

$$\sigma_z^2 = \frac{C_z^2 \, x^{(2-n)}}{2} \quad \text{neutral, unstable conditions} \tag{E.1-6}$$

The expression for σ_z in Equation E.1-6 is easily recognized as the standard Sutton equation. The expression for σ_z in Equation E.1-5 was derived from Hanford Field measurements of the vertical concentration taken at several downwind locations under stable conditions. The constants for Equation E.1-5 and E.1-6 were evaluated from the Hanford measurements for a source height of 200 ft and correlated with vertical temperature gradients at the point of emission.

Since the concentration measurements were averaged over 30 to 60 min intervals the constants used to evaluate σ_z are considered to be more appropriate for long term releases rather than the shorter term or "puff" releases. Figures E.1-1 through E.1-4 show vertical cloud width (σ_z) as a function of distance for each stability category.

The following stability classification is used along with vertical temperature lapse rates for each:

Very stable	$\Delta T \geq 1.5$ °C/100 m
Moderately stable	$-0.5 \leq \Delta T < 1.5$
Neutral	$-1.5 \leq \Delta T < -0.5$
Unstable	$-1.5 < \Delta T$

Table E.1-1 shows the deposition velocity coefficients for each stability category. Table E.1-2 shows the appropriate values of a, b, k^2, C_z and n used with each stability condition and wind speed. Such values are used to calculate the vertical dimensions of the plume (σ_z) and as stated earlier, were constants derived from the Hanford field measurements.

The conventional "reflection" factor of two usually applied for releases is not included. For the passing cloud which is primarily a

gamma dose, the entire plume volume is integrated as an "infinite" number of point sources to plus and minus infinity in the z direction. This ignores the interception by the ground so that the entire cloud volume is included.

Inhalation doses are a function of concentration at the ground and subject to "reflection" effects if they exist. Since the materials of interest in inhalation effects deposit on the ground, it is doubtful that "perfect" reflection will occur, but rather that the cloud will expand distorting the Gaussian mass distribution of the cloud resulting in, at most, a small increase in concentration. In addition, no account was taken of the better diffusion at the ground (effective on the portion of the cloud near the ground) compared to the stack exit elevation used. Meteorology and Atomic Energy (AECU 3066) shows that compared to an elevation of 200 m, ground level diffusion coefficients are larger by about a factor of 2 plus proportionally increasing dispersion. In any event, an increase by a factor of slightly more than 1.0, but less than 2, would account for this "reflection" effect.

E.1.2 Radiological Factors

The ground level gamma dose rate from an elevated plume of radioactive materials having a spatial distribution as given in Equation E.1-3 may be considered as the sum of the dose rates from all the points in the plume. The source strength of each point is (X)dV and the total source is:

$$S = \int_{-\infty}^{\infty} (\chi)\, dV \qquad \text{(E.1-7)}$$

where:

$dV=dxdydz$

The flux from a point source, considering buildup in the air, is given by Glasstone[3] as

$$\Phi = \frac{SBe^{-\mu R}}{4\pi R^2} \ \frac{\text{photons per}}{\text{m}^2 \text{ per sec}} \qquad \text{(E.1-8)}$$

where:

S =source strength

B =buildup factor=$1+K\mu R$ (See Figure E.1.5)

$\kappa = \frac{\mu - \mu_a}{\mu_a}$ π

μ =Total (linear) attenuation coefficient $(\text{meters})^{-1}$

μ_a =Energy absorption coefficient $(\text{meters})^{-1}$

R =distance from source equal to $(x^2 + y^2 + z^2)^{1/2}$

x_1, y_1, z_1 =coordinates of dose point at ground level relative to the incremental volume (dV)

Gamma Radiation Absorption Coefficients and Buildup Constants for Air are shown on Figure E.1-5.

The gamma dose-rate from a flux of a given energy (E) from Glasstone is

$$(DR)_\gamma = 5 \times 10^{-3}\ \phi\ E\mu_a (\text{units of mR/hr}) \qquad \text{(E.1-9)}$$

so that the total dose-rate from the plume at any point is found by combining Equations E.1-7, E.1-8, and E.1-9. Hence, the gamma-dose rate

$$(DR)_\gamma = \frac{5 \times 10^{-3}}{4\pi} E\mu_a \int_{-\infty}^{\infty} \frac{(\chi)_{ave}\ Be^{-\mu R}}{R^2}\, dV \quad (\text{mR/hr}) \qquad \text{E.1-10)}$$

As Equation E.1-10 is written, it assumes a monoenergetic source. For a mixture of isotopes, it is proper to perform the calculation for each gamma energy present considering its abundance. Since μ and μ_a are energy dependent and appear in an exponential term, care must be exercised if an average energy is to be used. A listing of each of the noble .gas isotopes and significant particulate daughter products is shown on Tables E.1-3 and E.1-4. Also shown are the gamma energies, total attenuation and linear absorption coefficients. This analysis used an "average" isotope representing the mixture at 30 min decay. Values used for E , μ, and μ_a are shown on Table E.1-4.

In general, Equation E.1-10, cannot be solved analytically and must be solved numerically. While integration to infinity is indicated, in practice finite bounds are placed on the cloud. Integrating Equation E.1-10 to $\pm 3\sigma_z$ includes more than 99.97 percent of the entire matter per unit length; hence, the dose contributions from points in

the cloud when vertical displacement is more than three standard deviations from the plume center line can be ignored. Likewise, due to the geometric and material attenuation shown in Equation E.1-8, one can usually ignore the dose contribution from source points that are more than 400 to 500 m downwind or upwind of the receptor point without significant error. The integration proceeds by reducing the distributed source (the plume) into a large array of point sources. This is done by dividing the cloud into cubical volume elements. The assumption is made that the concentration at the center of the cube is average for the volume element.

The total source strength is preserved by multiplying the concentration at the center (μCi/cc) by the volume of the element (cc). The dose rate from each point source is calculated by Equation E.1-10 and summed over all points. Equation E.1-10 then becomes a finite series.

Mathematically the numerical integration can be expressed as:

$$DR^{ij}(P) = \sum_{P'} \sum_{I} G^{ij}(I, P' : P) \qquad \text{(E.1-11)}$$

where $G^{ij}(I, P', P)$ is the dose rate contribution from isotope (I) to point (P) from a source at (P′) as described by Equation E.1-10.

Equation E.1-10 or E.1-11 gives the average dose rate for the (ij)th meteorological condition for a point (P) which may be immersed in the cloud or at some point outside the cloud. This is a significant item since the gamma dose at ground level from a stack plume is not merely existent when the receptor is immersed in the plume. Dose is also received when the plume is traveling in some other sector than the one in which the receptor point is located. The effect is particularly important at points close to the stack where the receptor remains at a nearly constant distance from the plume regardless of angular separation.

E.1.3 Engineering Factors

From Equations E.1-4 and E.1-10 it is evident that the dose-rate is significantly affected by the height of the plume above ground level. This height is made up of the physical stack height plus plume rise due to exit velocity and bucyancy. Many formulae are available to calculate plume rise. The method used here is the Holland formula as modified by Moses, et al.[4]

$$\Delta H = \frac{K(1.5V_s d + 4 \times 10^{-5} Q_h)}{\bar{u}_h} \qquad \text{(E.1-12)}$$

where:

V_s =exit velocity (m/sec)

d =stack diameter (m)

Q_h =heat emission of effluent (cal/sec)

$\bar{u}_h$ =wind speed at stack exit (m/sec)

K =correction factor for stack diameter[4] (Stumke regression coefficient)

In proposing the correction factor "K" in the plume rise formula, Moses used data from an experimental stack at Argonne with a diameter of about 0.46 m and from a stack at Duisburg, Germany, which has a diameter of 3.5 m. His conclusions are that a value of 3 for the correction factor is proper for large stacks with appreciable buoyancy, whereas a factor of 2 is recommended for small stacks with modest buoyancy. In applying the Moses correction to individual situations, a linear interpolation is made from the actual stack diameter compared to those from which data were obtained. See Figure E.1-6.

The USAEC document Meteorology and Atomic Energy - 1968[(5)] points out similar results in Section 5.2 discussed by Gary A. Briggs. He states that "both the Stumke formula and Holland formula times a factor of 3 seem to give good agreement (calculated versus observed plume rise) for the moderate sized sources (heat emissions of about 10^6 cal/sec) but grossly underestimate rise in the case of the large Colber plant (heat emissions of about $7X10^6$ cal/sec)."

E.1.4 Averaging Techniques

One is usually interested in the cumulative dose over some appropriate time interval, such as a year. To compute the annual gamma dose, the gamma dose rate for a given meteorological condition must be weighted by the frequency distribution F^{ijk}. F^{ijk} describes the frequency of the i^{th} stability condition with j^{th} wind speed occurring in direction sector k. The average annual gamma dose rate in sector k is given by:

$$DR_\gamma^{ij}(P) = C \sum_{k'} \left[\sum_{k} DR_\gamma^{ijk;k'}(P) F^{ijk} \right] \qquad (E.1\text{-}13)$$

where:

$DR_\gamma^{ijk;k'}(P)$ is the gamma dose rate to a point (P) in sector k from a plume traveling in Sector k' and C = 8760 hr/yr.

Equation E.1-13 indicates a finite summation over the variables of stability, wind speed, and direction. For stability and direction it has already been indicated how these variables can be grouped into four stability classes and 16 directions. The spectrum of wind speeds can also be grouped into representative ranges. One such grouping that has proven useful, especially when using U.S. Weather Bureau summaries, is as follows:

Wind Speed Range (mph)	Average Wind Speed (m/sec)
0-3	1
4-7	2
8-12	5
13-18	7
19-24	10
>25	>13

Also included above is the average wind speed that is representative of each speed range.

E.1.5 Average Air Concentration

For doses other than the whole body gamma dose, the annual average concentration at ground level is of interest. This is easily obtained from the preceding material presented by substituting plume height for z. The air concentration during any meteorological condition has been described by Equation E.1-4. However, for materials other than noble gases, the depletion factor (Q/Q_0) is not unity and must be accounted for. For the calculations made in the report, the deposition rates shown on Table E.1-1 were used.

Using the joint frequency distribution F^{ijk} defined previously, computations of the annual average concentration at the ground can be made from:

$$(\chi)^{k}_{gr} = \sum_{ij} (\chi)^{ij}_{gr} F^{ijk} \qquad \text{(E.1-14)}$$

E.1.6 Shielding and Occupancy Factors

Radiation doses calculated are usually performed for certain distances from the point of release and often are calculated for locations where no actual dose would be received by a human receptor. In fact it is not too uncommon to see radiation doses from the passing cloud calculated as if the dose receptors were outside day and night. This is certainly possible, but it does not lead to particularly accurate dose estimates for the great majority of people. For this reason occupancy by individuals should be considered in arriving at reasonable dose estimates. Credit for this is allowed by the USNRC's 10 CFR 20.

Additionally, it seems rather incongruous to assume that a person would stay in one place all of the time without being inside some type of shelter. For this reason, the shielding effect for various types of structures was evaluated. The shielding value of such typical structures is shown on Table E.1-5.

It is easily seen that the error introduced by omitting this effect can be a factor of two or more. Where larger urban complexes are concerned, such an error may be far greater.

TABLE E.1-1

DEPOSITION VELOCITY COEFFICIENTS

Stability Condition	$V_d/\bar{u}_o$ [1] Particulates	Halogens
Very Stable	1.5×10^{-4}	2.4×10^{-3}
Moderately Stable	2.2×10^{-4}	3.4×10^{-3}
Neutral	3.0×10^{-4}	4.6×10^{-3}
Unstable	6.0×10^{-4}	8.0×10^{-3}

NOTE:

(1) To obtain the deposition velocity, multiply this ratio of deposition velocity to ground wind speed by the ground speed ($\bar{u}_o$).

TABLE E.1-2

DIFFUSION COEFFICIENTS

Constants	Very Stable	Moderately Stable	Neutral	Unstable
$a(m^2)$	34	97		-
$b(m^2/sec)$	0.025	0.33		-
$K^2(sec^{-2})$	8.8×10^{-4}	2.5×10^{-4}		-
$C_z(\bar{u}=1m/sec)$	-	-	0.15	0.30
$(\bar{u}=5m/sec)$	-	-	0.12	0.26
$(\bar{u}=10m/sec)$	-	-	0.11	0.24
n	-	-	0.25	0.20

TABLE E.1-3

NOBLE GAS ISOTOPES
CONSTITUTING MIXTURE

Isotope Name	Half Life [1]	E (MeV)		
Noble Gases				
Kr-83m	1.86h	0.032	0.045	0.015
		0.009	0.8	0.7
Kr-85m	4.4h	0.15	0.016	0.0032
		0.305	0.013	0.0038
Kr-85	10.76y	0.522	0.011	0.004
Kr-87	76m	2.05	0.006	0.0028
		2.57	0.005	0.0026
		0.847	0.009	0.0038
		0.347	0.013	0.0039
Kr-88	2.8h	2.4	0.0055	0.0027
		2.21	0.006	0.0028
		0.19	0.015	0.0034
		1.55	0.007	0.0032
		0.85	0.009	0.0038
		0.17	0.015	0.0032
		0.02	0.1	0.063
Xe-131m	12d	0.164	0.015	0.0032
Xe-133m	2.3m	0.233	0.014	0.0037
Xe-133	5.27d	0.081	0.02	0.0032
Xe-135m	16m	0.53	0.011	0.004
Xe-135	9.2h	0.604	0.01	0.004
		0.36	0.013	0.0039
		0.244	0.014	0.0037
Xe-138	14m	0.42	0.012	0.004

[1] Radiological Half-Life

TABLE E.1-4

PARTICULATE DAUGHTER PRODUCTS AND "AVERAGE" ISOTOPE

Isotope Name	Half Life[1]	$E\gamma$ (MeV)	μ	μ_a
Particulate Daughters				
Rb-88	18m	0.91	0.0085	0.0037
		1.28	0.0072	0.0034
		1.85	0.0060	0.0032
		2.18	0.0050	0.0030
		4.2	0.0038	0.0024
Cs-138	32.2m	0.14	0.018	0.0033
		0.19	0.016	0.0035
		0.23	0.015	0.0037
		0.41	0.0122	0.0037
		0.46	0.0116	0.0038
		0.55	0.0108	0.0038
		0.87	0.0088	0.0037
		1.01	0.0082	0.0036
		1.43	0.0068	0.0034
		2.21	0.0055	0.003
		2.63	0.0050	0.0039
		3.34	0.0043	0.0026
"Average" Isotope				
0-12 Hr Decay		0.62	0.0099	0.0039
12-48 Hr Decay		0.30	0.0135	0.0038
>48 Hr Decay		0.020	0.092	0.059

[1] Radiological Half-Life

TABLE E.1-5

SHIELDING - DOSE REDUCTION FACTORS[1]

Structure	Location	Dose Reduction Factor
One story frame house	First floor, middle	0.5
	Basement, middle	0.15
Two story brick veneer	First floor, middle	0.3
	Basement, middle	0.06
Three story brick veneer	First floor, middle	0.13
	Basement, middle	0.04
Multi-floor reinforced concrete	Upper floors	0.02
	First floor	0.1
	Basement	0.001

NOTE:

1. Glasstone, S. Effects of Nuclear Weapons, Rev. ed. Defense Atomic Support Agency, Dept. Defense, April 1962.

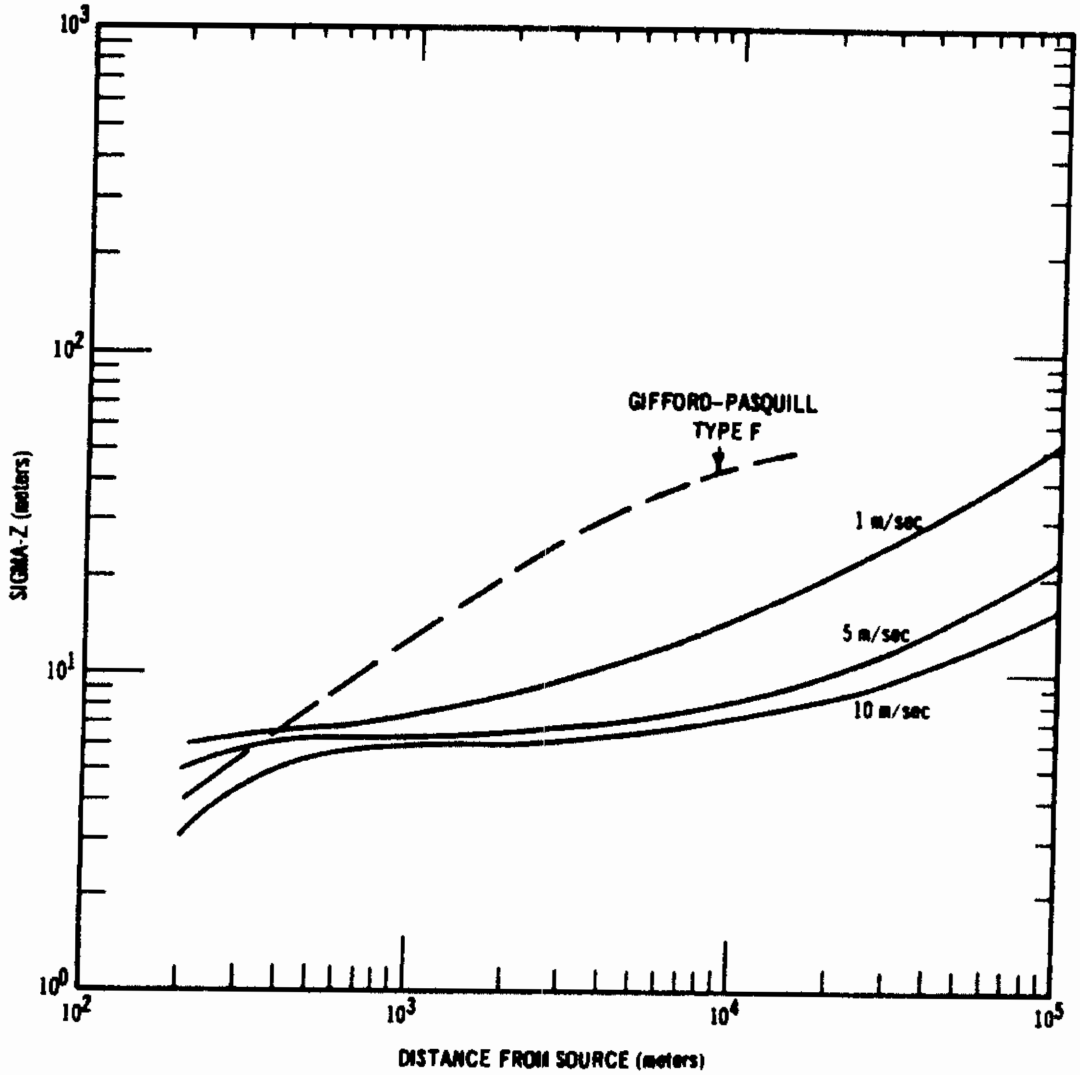

FIGURE E.1-1
VERTICAL CLOUD WIDTH VERSUS
DISTANCE: VERY STABLE
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

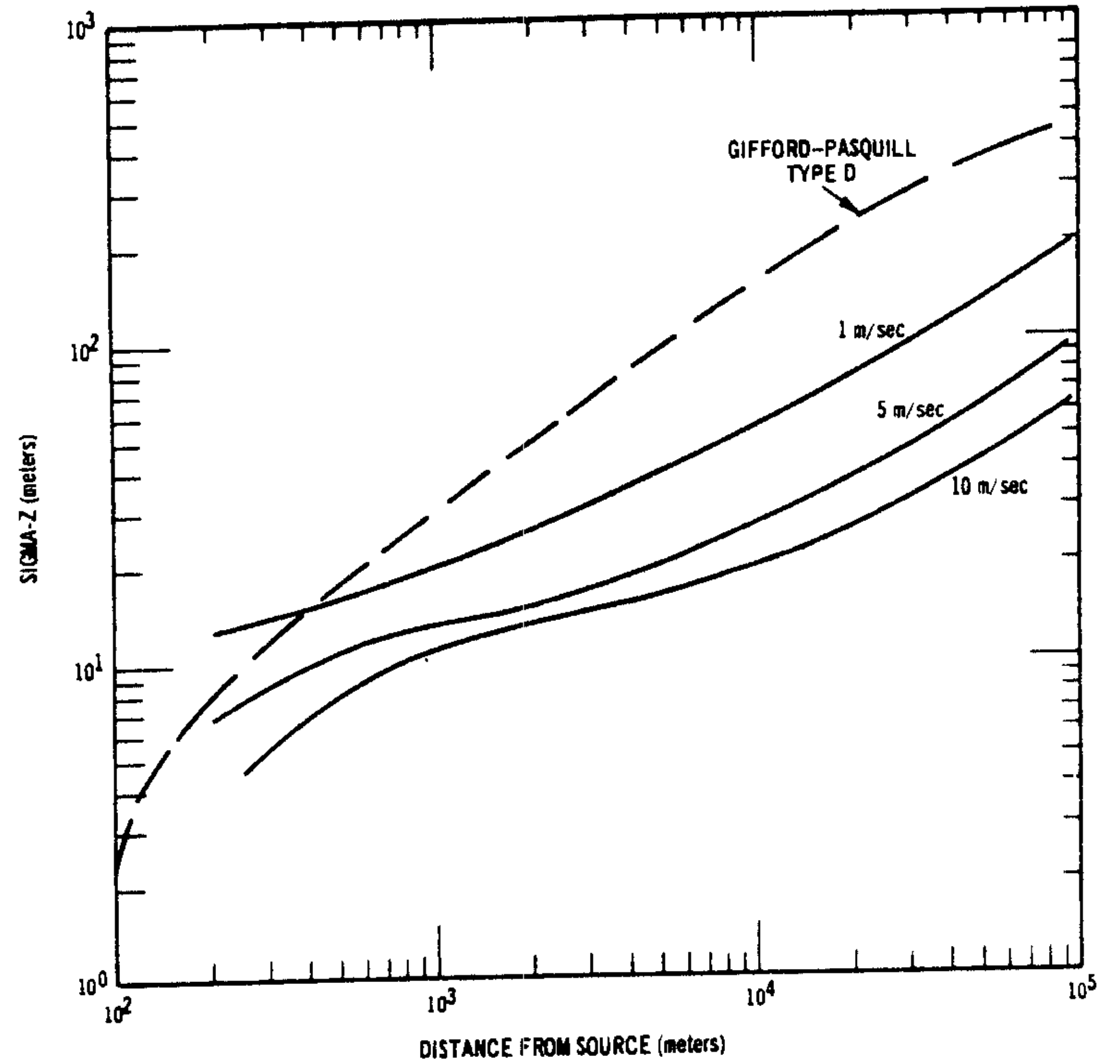

FIGURE E.1-2
VERTICAL CLOUD WIDTH VERSUS DISTANCE: MODERATELY STABLE

PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

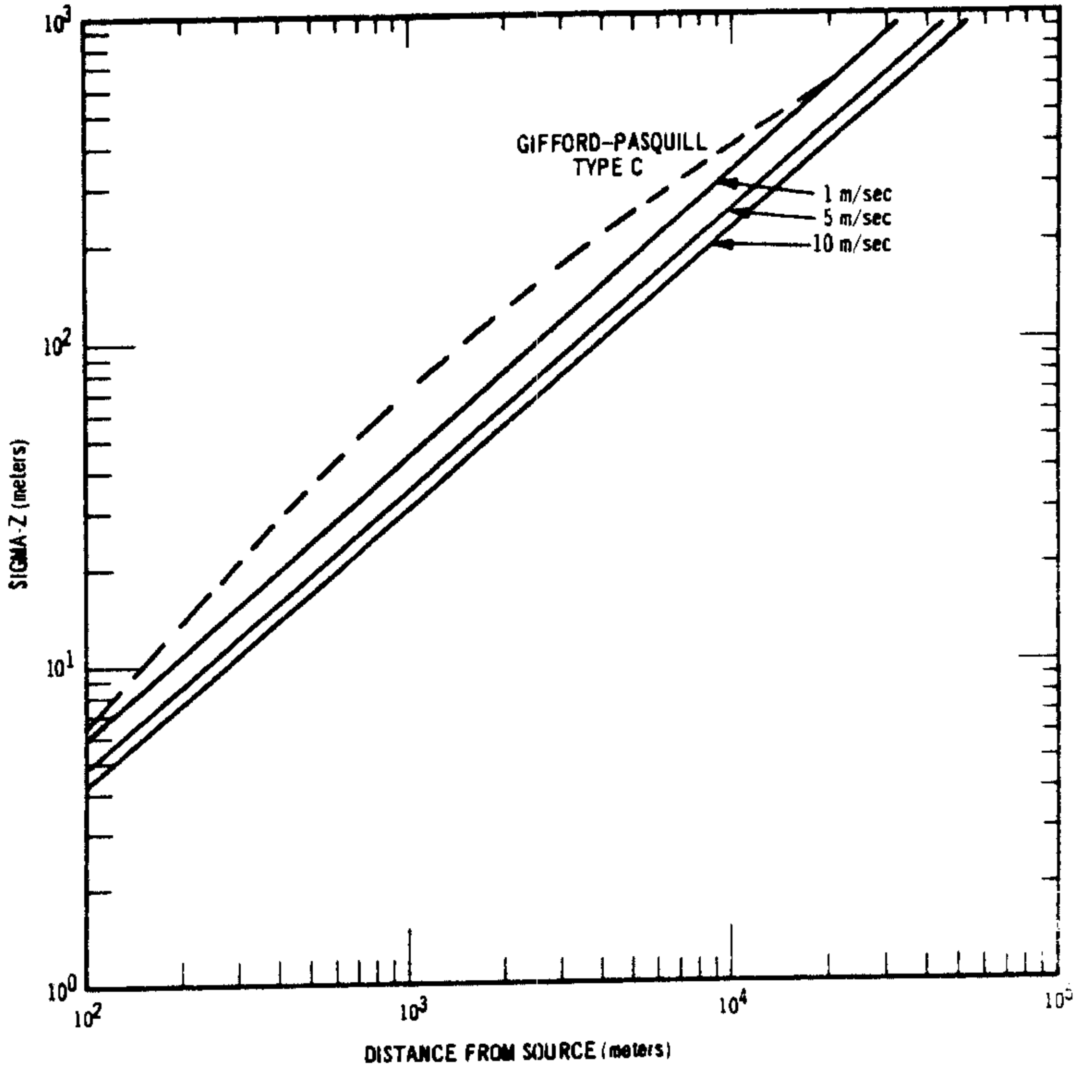

FIGURE E.1-3
VERTICAL CLOUD WIDTH
VERSUS DISTANCE: NEUTRAL
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

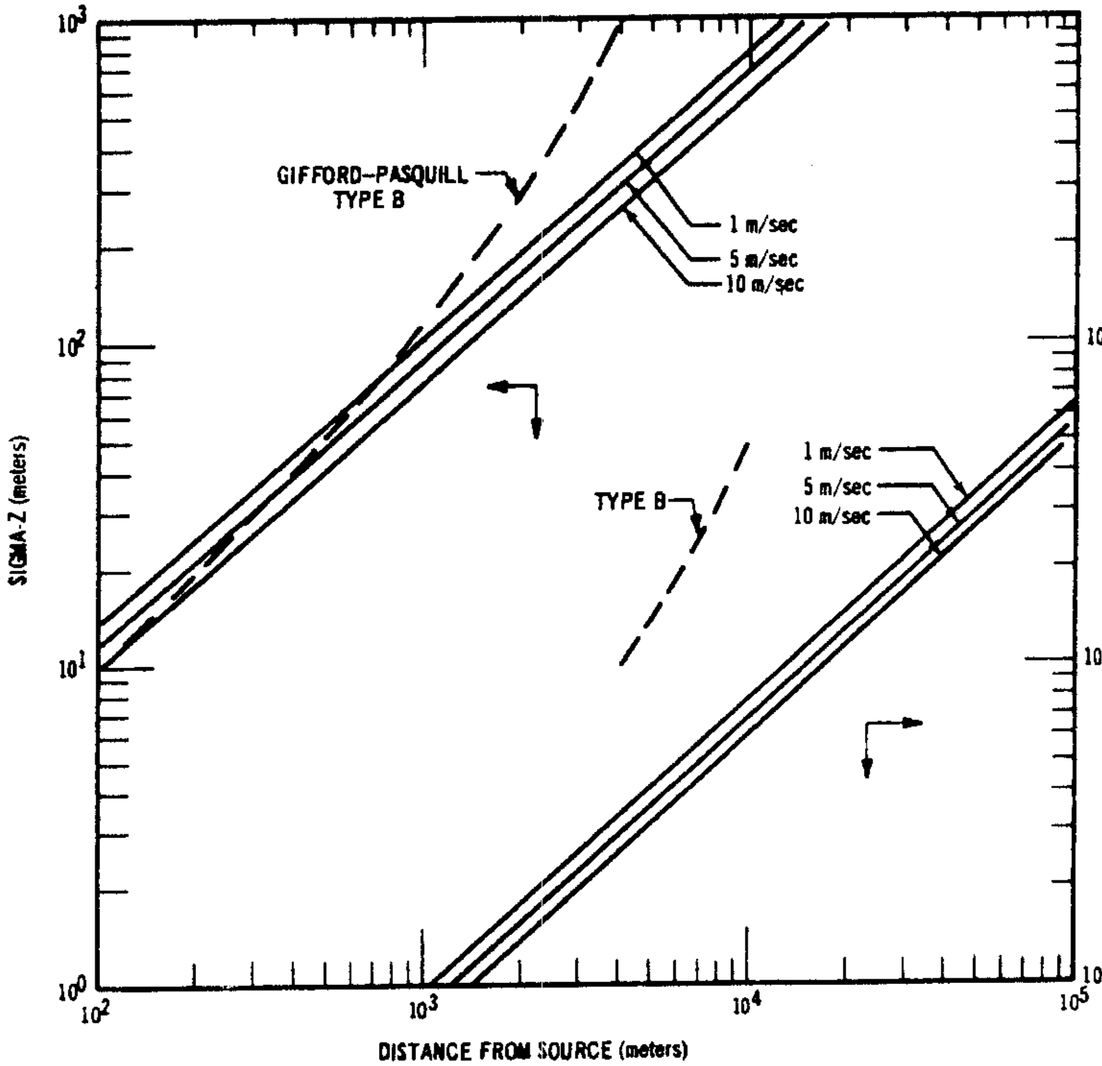

FIGURE E. I-4
VERTICAL CLOUD WIDTH
VERSUS DISTANCE: UNSTABLE

PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

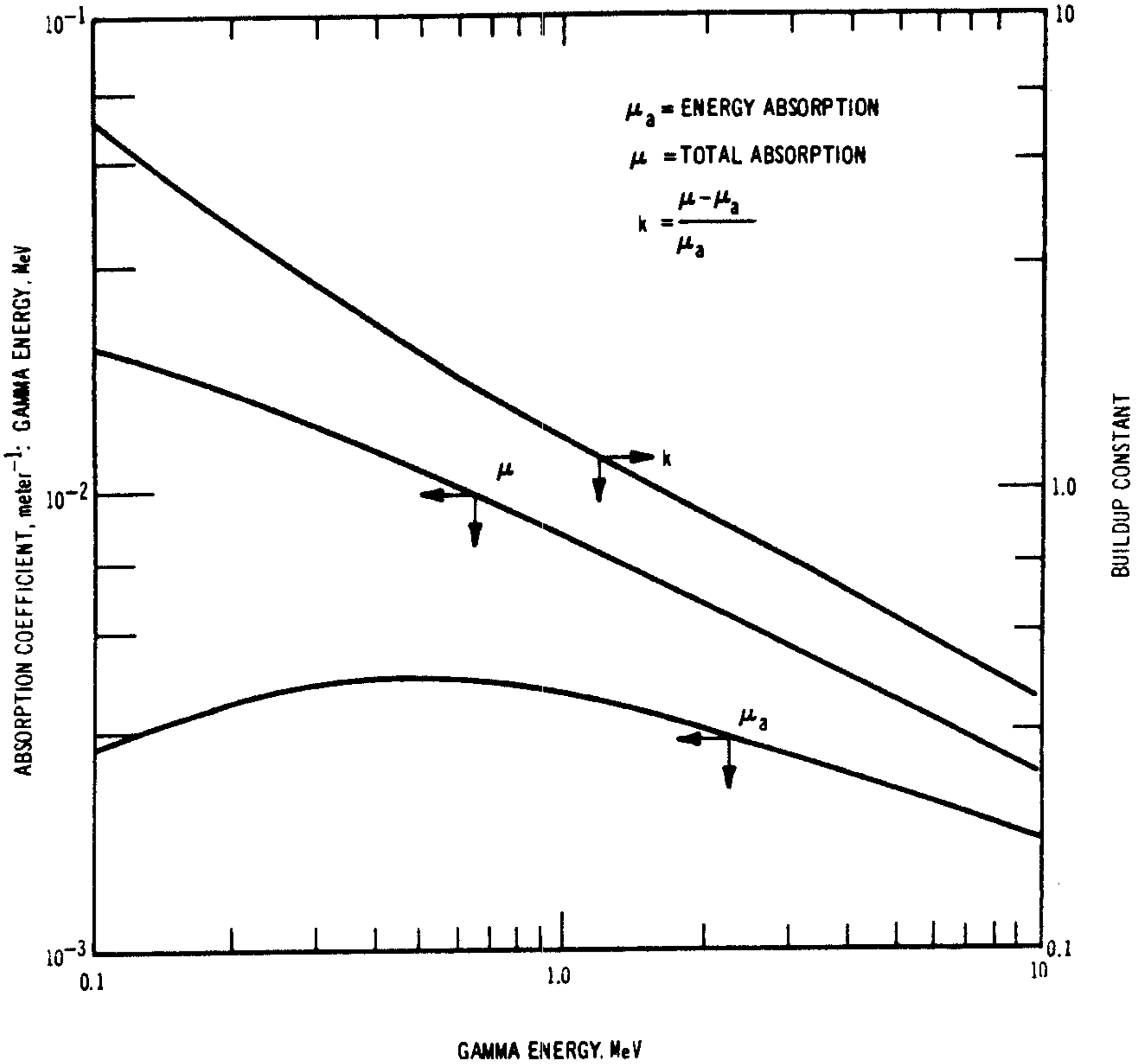

FIGURE E.1-5
GAMMA RADIATION ABSORPTION COEFFICIENTS AND BUILDUP CONSTANTS FOR AIR, STP

PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

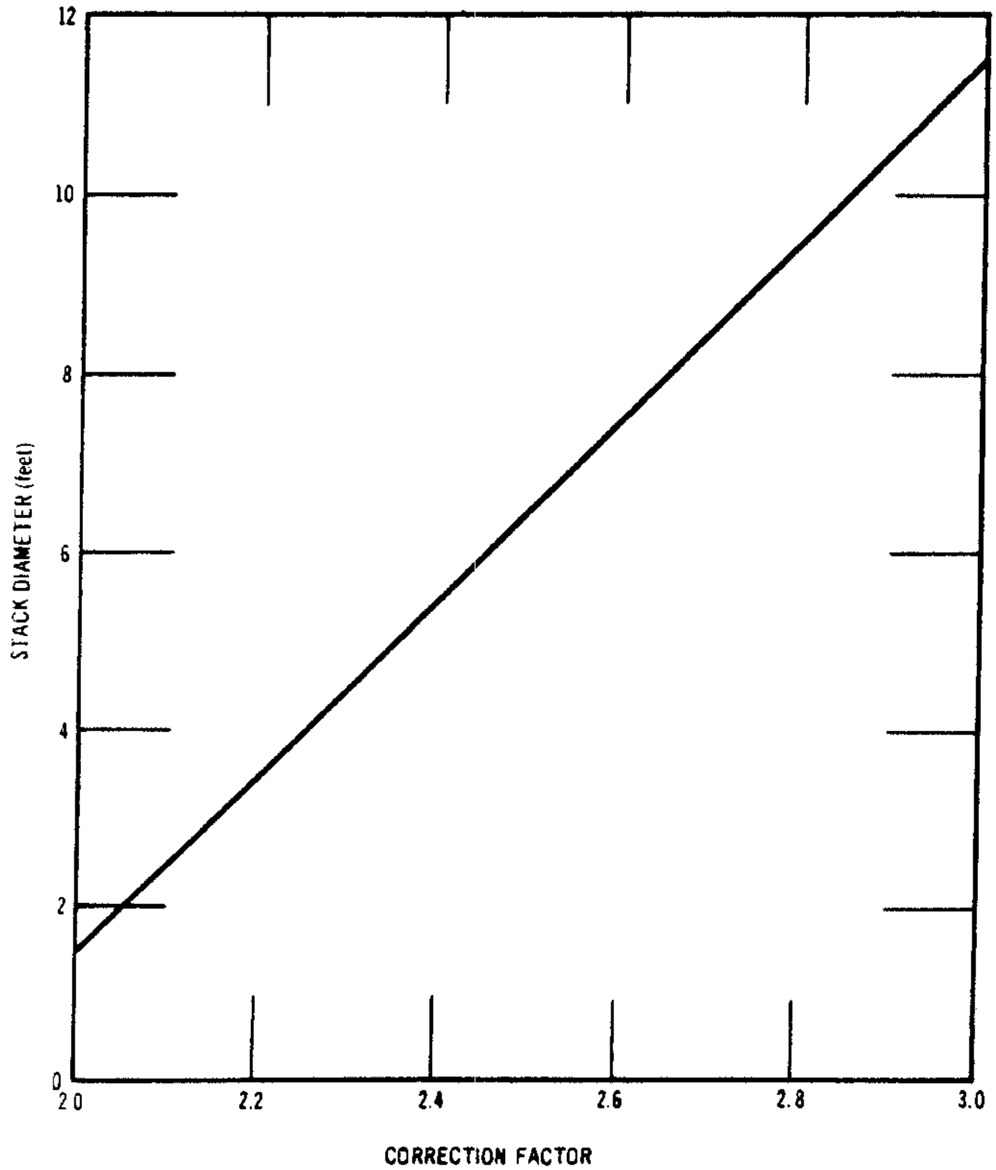

FIGURE E.1-6
HOLLAND PLUME RISE
CORRECTION FACTOR

PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

E.2 VERIFICATION OF ANALYTICAL MODEL

E.2.1 Meteorology Data

Micrometeorological data for 1963 were obtained from Brookhaven National Laboratory. The data were in the form of computer input cards containing hourly observations of average wind speed and direction at the 37, 150, and 335 ft levels; and the air temperature at the 37, 75, 150, 300, and 410 ft levels. The measurements at the 355 ft level were summarized in terms of frequency of occurrence according to wind speeds, direction, and atmospheric stability. The stability was determined according to the method described in Section E.1 by using the temperature gradient measured between the 410 ft and 37 ft levels.

The summaries are presented on Tables E.2-1 to E.2-6. The frequency of occurrence was based on 6,464 hr of good observations. Of the missing 2,296 hr of 1963, August and September account for 1,464 missing hours, the rest being scattered throughout the year. A total of 12 hr were observed to have a wind speed less than 0.5 mph. These "calm" conditions were included in the wind speed category (0-3) mph.

E.2.2 Radiological Data

As is discussed by Hull[7], the radiation dose was measured at several stations around the Brookhaven Graphite Research Reactor (BGRR) in 1963 using 6 liter atmospheric pressure ion chambers. The dose rate from the release of Ar^{41} (Argon) was determined from the total dose measurement by subtracting from it the contribution from natural background and operation of the forest ecology station. The resultant dose rate is shown on Table E.2-7.

It was necessary to adjust the measured values of annual gamma dose to account for the absence of meteorological data during August and September. The average dose rate (mR/wk) was averaged over the 10 months for which meteorological data were available and multiplied by 52 to get annual dose (mR/yr). The exception to this is station E-2 which was moved in December. For this station, nine months were used to determine the annual dose. These normalized values are shown in comparison with calculated values on Table E.2-8.

E.2.3 Gamma Dose Calculations

The methods described in Section E.1 were used to analyze the effects of the BGRR stack effluent in the Brookhaven environs. The following is a discussion of the calculations leading to the gamma dose rate matrix, $Dr_{\gamma}ijk,k'$.

E.2.3.1 Plume Rise

The BGRR has a 350 ft stack (107 m) with an exit velocity of 6 m/sec and an effluent temperature difference of 50°C above ambient. For use in Equation E.1-12 these values correspond to:

$Q_h = 1.62 \times 10^6$ cal/sec - Heat rate
d = 5.18 - Stack exit diameter
K = 3.47 - Correction factor in equation

Using these values in Equation E.1-12 the plume rise formula becomes:

$$\Delta H = \frac{377}{\bar{u}_h} \text{ meters} \tag{E.2-1}$$

Using the six standard wind speed groups described earlier, the effective stack heights were computed and are shown below.

BGRR Plume Rises for Various Wind Speeds

Wind Speed Range (mph)	Average Speed (m/s)	Plume Rise (meters)	Effective Height (meters)
0-3	1	377	484
4-7	2	189	295
8-12	5	75	182
13-18	7	54	161
19-24	10	38	145
>25	>13	29	136

E.2.3.2 Isotopic Data

The BGRR during full power operation releases about 12,960 curies of Argon 41 per day (0.15 Ci/sec). However, the actual average release rate during 1963 was 0.127 Ci/sec as determined from personal communications with the BGRR staff, which represents an 85 percent operation factor.

Pertinent radiological properties of Ar^{41} are:

E = 1.29 Mev Gamma energy

$\mu = 6.93 \times 10^{-3} M^{-1}$ Total attenuation coefficient

$\mu_a = 3.3 \times 10^{-3} M^{-1}$ Energy absorption coefficient

$\gamma = 1.1 \times 10^{-4} sec^{-1}$ Decay constant

E.2.3.3 Dose Rate Calculations

From the above information, the gamma dose rate as given by Equations E.1-10 and E.1-11 was evaluated using a digital computer program to evaluate Equation E.1-10. The dose rate was evaluated for downwind distances of 10, 100, 400, 1,400, 2,400, 3,200, and 6,400 meters, using the six wind speeds shown earlier and all four stability conditions. The results are shown on Figures E.2-1 to E.2-3. The dose rates for the very stable and moderately stable conditions are essentially identical, because for the distances used here the vertical spread of the plume is small in each case. Hence,

the difference in cloud dimensions between the two stable conditions are not great compared to the attenuation distances involved.

Another important feature to notice is that there is very little variation in dose rate between any of the stability classes for the plume height considered here. Figure E.2-2 illustrates this point more clearly by showing the dose rate for a 5 m/s wind speed for each of the stability conditions. The variation of dose rate between stability conditions is very small for downwind distances less than 400 m, and is less than a factor of 2 even to a distance of 6 mi. From the shape of the dose rate curves, it can be seen that the maximum usually occurs within 1,000 m and decreases rapidly thereafter.

The dose rates shown on Figures E.2-1 to E.2-3 are for points on the ground directly below the centerline of the sector averaged plume. As previously mentioned, significant dose contributions can also occur in sectors other than the one in which the plume is traveling. Due to symmetry, there are only 9 unique sectors for which dose rate calculations can be made.

If the sector in which the plume is traveling is designated as Sector 1, shown on Figure E.2-4, then the dose to Sector 16 from the plume is equal to the dose to Sector 2; the dose to Sector 15 is the same as the dose to Sector 3 and so on. In terms of the dose rate matrix the following equalities can be listed:

$$
\begin{aligned}
DR_\gamma^{ij1,l} &= DR_\gamma^{ij1,l} \\
DR_\gamma^{ij16,l} &= DR_\gamma^{ij2,l} \\
DR_\gamma^{ij15,l} &= DR_\gamma^{ij3,l} \\
&\vdots \\
DR_\gamma^{ij9,l} &= DR_\gamma^{ij9,l}
\end{aligned}
$$

However, for distances greater than 100 m, the dose rate to adjacent sectors is very small because of the large separation distances. This is illustrated by Figure E.2-3 which shows the sector variation

of dose rate with distance for one particular meteorological condition. In practice the dose rate to a point in Sector k is not calculated if the dose rate is less than 0.1 percent of the dose rate to a point at the same downwind distance in Sector 1.

Figures E.2-1 to E.2-3 indicate how the dose rate matrix DR_{γ} ijk, k' is constructed. It now remains to find the joint frequency distribution F^{ijk} to calculate the annual dose rate.

E.2.3.4 Conclusions About Gamma Dose Calculations

From the data presented on Figure E.2-4 it is concluded that the analytical model provides a fairly precise correlation between stack release rate and ground level gamma radiation dose. It is seen that the maximum dose is at the closest point to the stack. This should not be surprising since at the base of the stack, for example, the dose rate is continuous and independent of plume direction travel. This would be expected from the dose rate curves presented on Figures E.2-1 through E.2-3.

Further examination of Figures E.2-1 through E.2-3 showing dose rate during each meteorological condition leads to additional interesting conclusions. The dose rate does not seem to be very sensitive to the atmospheric stability condition. This is markedly in contrast to the air concentration differences at ground level during the various stability regimes. It is widely known that during very stable conditions near zero air concentration exists at ground level from an elevated plume since it remains very narrow and highly concentrated aloft. On the other hand, unstable conditions promote rapid effluent growth and dispersion, and highest ground level air concentrations.

It appears that while the gamma dose rate is quite insensitive to atmospheric stability, it is quite dependent on plume height and wind speed. This is to be expected intuitively from Equation E.1-4 where the average concentration which is used to obtain dose rate is inversely proportional to wind speed and that the attenuation distances increase with plume height. In practice, buoyant effluents are typical (although not universal) so that effluent buoyance centers the calculations. That is to say, plume height is made up of stack height plus plume rise due to buoyancy. The latter is greatest for smallest wind speeds. Thus, the smallest wind speed conditions do not a priori yield the largest dose rates. In fact, experience with calculations using this analytical model verifies this fact.

Calculations have also shown that most of the dose over a long period of time comes from the conditions where the wind speed is about at the average speed of 4 to 7 m/sec (9 to 16 mph), which most locations are observed to have. The calculation for Brookhaven is no exception. This can partially be explained by the fact that for elevations considered here (300 to 400 ft) low winds speeds, for example, are rather infrequent, accounting for about 3 percent of the time.

A final conclusion drawn from the comparison of calculated and measured doses refers to the dose pattern depicted on Figure E.2-5. It is observed that for distances out to about 1/2 mi (typical large reactor site) the iso-dose contours exhibit a smooth, rather than a peaked, pattern. This is quite different from the wind direction distribution (wind rose) or where total direction frequency is indicated. See Table E.2-5. However, the smooth gamma dose pattern, as indicated on Figure E.2-5, is attributed to the fact that the total dose at each point is made up of the dose from plumes traveling in all directions. At distances of 2 mi and beyond the gamma dose contours exhibit a peaked pattern similar to the wind rose. At these distances only plumes traveling in the direction of a dose point contribute significantly to the gamma dose at the point.

E.2.3.5 Ground Level Air Concentration Calculations

For some kinds of radiation dose only the groundlevel air concentration is of interest. Examples of these are dose from inhalation, external beta dose and deposition. In each of these, concentration at the dose point determines the dose regardless of the concentration at other points in the plume. This method of calculating the correlation between stack emission rate and groundlevel air concentration is also of interest in assessing environmental effects of a stack effluent.

Some limited air concentration measurements are also made at Brookhaven (BNL-915). These are measurements of small quantities of iodine released from the BGRR. Three monitoring stations were operated in 1963, although since then the scope of this program has been augmented. The release of iodine-131 from the BGRR was about 0.1 μ Ci/sec continuously.

As indicated previously, the analytical model used calculates average air concentration at any point in the cloud, including groundlevel. Thus, the calculation is similar to that done for the gamma dose, but only at the dose point (groundlevel) can the calculation be performed.

The calculation of long term average groundlevel air concentration is as described in Section E.1.1. This involves weighting each calculated average concentration during each meteorological condition by its frequency of occurrence and summing over all conditions.

The highest concentration calculated for the Brookhaven case is 0.6 x 10^{-15} μ Ci/cc. As indicated previously, only three iodine monitoring points existed during the year 1963. All of these locations showed annual concentrations below detectable limits of about 2 x 10^{-15} μCi/cc. Thus, only a qualitative comparison of the analytical model and the data can be made at this time for this type of calculation. The calculated values, however, appear to be about the correct order of magnitude, but the comparison is inconclusive.

TABLE E.2-1

PERCENT OCCURRENCE OF GOOD OBSERVATIONS FROM THE BNL SITE
FOR VARIOUS DIRECTIONS AND WIND SPEEDS

(6,464 hr during 1963)
Atmospheric stability: Very Stable
Stability based temp. diff. taken at 410 ft and 37 ft

Speeds (mph) at 355 ft

Direction	0-3	4-7	8-12	13-18	19-24	>24	All Speeds
N	0.0619	0.108	0.433	0.449	0	0	1.05
NNE	0.0619	0.139	0.155	0.124	0	0	0.48
NE	0.0309	0.0464	0.155	0.0309	0	0	0.26
ENE	0	0.0928	0.124	0.0309	0	0	0.25
E	0.0464	0.0774	0.155	0.0774	0.0619	0	0.42
ESE	0.108	0.201	0.263	0.0155	0	0	0.59
SE	0.0464	0.0309	0.0464	0.0155	0.0155	0	0.15
SSE	0.0619	0.139	0.201	0.124	0	0	0.53
S	0.0464	0.201	0.248	0.356	0.232	0.0309	1.11
SSW	0.0619	0.294	0.665	1.13	1.01	0.0928	3.25
SW	0.108	0.170	0.433	1.22	0.897	0	2.83
SSW	0.0928	0.124	0.340	1.11	0.557	0.0155	2.24
W	0.0464	0.186	0.804	0.712	0.541	0.309	2.32
WNW	0.0774	0.186	0.572	0.433	0.139	0	1.41
NW	0.0309	0.186	0.433	0.603	0.186	0	1.44
NNW	0.0155	0.124	0.433	0.789	0.0774	0	1.44
All Directions	0.90	2.31	5.46	7.22	3.71	0.17	19.77

TABLE E.2-2

PERCENT OCCURRENCE OF GOOD OBSERVATIONS FROM THE BNL SITE
FOR VARIOUS DIRECTIONS AND WIND SPEEDS

(6,464 hr during 1963)
Atmospheric stability: Moderately Stable
Stability based temp. diff. taken at 410 ft and 37 ft

Speeds (mph) at 355 ft

Direction	0-3	4-8	8-12	13-18	19-24	>24	All Speeds
N	0	0.186	0.433	0.433	0.232	0.0464	1.33
NNE	0.0464	0.278	0.557	0.464	0.186	0.0309	1.56
NE	0.0774	0.232	0.557	0.124	0.0928	0	1.08
ENE	0.0309	0.139	0.139	0.201	0.155	0	0.66
E	0.0309	0.186	0.263	0.278	0.232	0.0619	1.05
ESE	0.0928	0.139	0.371	0.278	0.186	0.0619	1.13
SE	0.0309	0.0774	0.155	0.387	0.0619	0	0.71
SSE	0.0309	0.0619	0.495	0.696	0.418	0.449	2.15
S	0.0928	0.108	0.402	1.08	0.804	0.201	2.69
SSW	0.139	0.232	0.913	1.90	1.36	0.108	4.66
SW	0.0464	0.232	0.495	1.53	0.572	0.0619	2.94
WSW	0.0464	0.108	0.449	1.44	0.480	0.0774	2.60
W	0	0.139	0.387	1.42	1.01	0.170	3.12
WNW	0.0464	0.139	0.371	0.727	1.07	0.0464	2.40
NW	0.0464	0.139	0.371	0.743	0.727	0.0309	2.06
NNW	0.0464	0.155	0.655	1.25	0.309	0.0619	2.49
All Directions	0.80	2.55	7.02	12.96	7.89	1.41	32.63

TABLE E.2-3

PERCENT OCCURRENCE OF GOOD OBSERVATIONS FROM THE BNL SITE
FOR VARIOUS DIRECTIONS AND WIND SPEEDS

(6,464 hr during 1963)
Atmospheric stability: Neutral
Stability based temp. diff. taken at 410 ft and 37 ft

Speeds (mph) at 355 ft

Direction	0-3	4-7	8-12	13-18	19-24	>24	All Speeds
N	0.0619	0.217	0.526	0.325	0.170	0.0309	1.331
NNE	0.0928	0.340	0.495	0.774	0.0464	0.0155	1.764
NE	0.0774	0.480	0.526	0.217	0.0155	0	1.316
ENE	0.0928	0.186	0.371	0.325	0.155	0.0155	1.145
E	0.0464	0.0928	0.263	0.155	0	0	0.557
ESE	0.139	0.449	0.743	0.278	0.0774	0.0155	1.702
SE	0.0309	0.217	0.464	0.124	0.0309	0.0155	0.882
SSE	0.0309	0.248	1.45	0.665	0.232	0.201	2.827
S	0.155	0.402	1.58	1.42	0.603	0.0774	4.237
SSW	0.186	0.172	1.67	0.93	0.774	0.0619	5.334
SW	0.139	0.294	0.712	0.804	0.433	0.139	2.521
WSW	0.124	0.263	0.882	1.18	0.990	0.325	3.764
W	0.155	0.248	0.789	1.39	1.73	0.975	5.287
WNW	0.124	0.248	0.851	1.01	1.30	0.619	4.152
NW	0.0464	0.294	0.511	0.851	0.866	0.263	2.831
NNW	0.0619	0.464	0.619	0.990	0.402	0.928	2.630
All Directions	1.56	5.15	12.45	12.44	7.83	2.85	42.28

TABLE E.2-4

PERCENT OCCURRENCE OF GOOD OBSERVATIONS FROM THE BNL SITE
FOR VARIOUS DIRECTIONS AND WIND SPEEDS

(6,464 hr during 1963)
Atmospheric stability: Unstable
Stability based temp. diff. taken at 410 ft and 37 ft

Speeds (mph) at 355 ft

Direction	0-3	4-7	8-12	13-18	19-24	>24	All Speeds
N	0	0	0.0928	0.0464	0	0	0.1392
NNE	0	0	0.0464	0.0155	0	0	0.0619
NE	0	0	0.0619	0	0	0	0.0619
ENE	0	0	0	0	0.0619	0.0155	0.0774
E	0	0	0	0.0155	0.0155	0	0.031
ESE	0	0	0.0155	0	0	0	0.0155
SE	0	0	0	0	0	0	0
SSE	0	0	0.0155	0.0309	0	0	0.0464
S	0	0	0.0928	0.248	0.232	0.0464	0.6192
SSW	0	0	0.0155	0.217	0.0774	0	0.3099
SW	0	0	0.0619	0.0619	0	0.0155	0.1393
WSW	0	0	0.0309	0.155	0.170	0.0928	0.4487
W	0	0	0.0619	0.402	0.541	0.186	0.1909
WNW	0	0.0309	0.0619	0.495	0.402	0.201	1.1908
NW	0	0	0.186	0.309	0.139	0.0155	0.6495
NNW	0	0.0155	0.0928	0.155	0.0619	0	0.3252
All Directions	0.046	0.835	2.15	1.70	0.5724	0.5737	5.87

TABLE E.2-5

BNL SITE METEOROLOGICAL DATA (1963)

PERCENT OCCURRENCE OF ALL WIND SPEEDS FOR 16 DIRECTIONS
AND 4 ATMOSPHERIC STABILITY CONDITIONS AT 355 FT

(Wind Rose)

Stability (ΔT = T410-T37)

Direction	VS	MS	N	U	All Stabilities
N	1.05	1.33	1.33	0.14	3.86
NNE	0.48	1.56	1.76	0.06	3.86
NE	0.26	1.08	1.31	0.06	2.73
ENE	0.25	0.66	1.14	0.08	2.13
E	0.42	1.05	0.56	0.03	2.06
ESE	0.59	1.13	1.70	0.01	3.43
SE	0.15	0.71	0.88	0	1.75
SSE	0.52	2.15	2.83	0.05	5.55
S	1.11	2.69	4.24	0.62	8.66
SSW	3.25	4.66	5.34	0.31	13.55
SW	2.83	2.94	2.52	0.14	8.43
WSW	2.24	2.60	3.76	0.45	9.05
W	2.32	3.12	5.29	1.19	11.92
WNW	1.41	2.40	4.15	1.19	9.14
NW	1.44	2.06	2.83	0.65	6.98
NNW	1.44	2.49	2.63	0.32	6.88
All Directions	19.77	32.64	42.28	5.31	100.00

6,464 Total Hours
12 hr of calm (less than 0.5 mph)

VS = very stable
MS = moderately stable
N = neutral
U = unstable

TABLE E.2-6

PERCENT OCCURRENCE OF GOOD OBSERVATIONS FROM THE BNL SITE
FOR VARIOUS DIRECTIONS AND WIND SPEEDS

(6,464 hr during 1963)
All Stabilities

Speeds (mph) at 355 ft

Direction	0-3	4-8	8-12	13-18	19-24	>24	All Speeds
N	0.124	0.510	1.48	1.25	0.402	0.077	3.85
NNE	0.201	0.758	1.25	1.38	0.232	0.046	3.87
NE	0.186	0.758	1.30	0.371	0.108	0	2.72
ENE	0.124	0.418	0.634	0.557	0.371	0.031	2.13
E	0.124	0.356	0.681	0.526	0.303	0.062	2.06
ESE	0.340	0.789	1.39	0.572	0.263	0.077	3.43
SE	0.108	0.325	0.665	0.526	0.108	0.015	1.75
SSE	0.124	0.449	2.16	1.52	0.650	0.650	5.55
S	0.294	0.712	2.32	3.11	1.87	0.356	8.66
SSW	0.387	1.24	3.26	5.18	3.22	0.263	13.55
SW	0.294	0.696	1.70	3.62	1.90	0.216	8.43
WSW	0.263	0.495	1.70	3.98	2.20	0.510	9.05
W	0.201	0.572	2.04	3.93	3.82	1.36	11.93
WNW	0.247	0.603	1.86	2.66	2.91	0.866	9.14
NW	0.124	0.619	1.50	2.51	1.92	0.309	6.98
NNW	0.124	0.758	1.81	3.19	0.851	0.155	6.88
All Directions	3.42	10.05	25.77	34.78	21.13	5.00	100.00

TABLE E.2-7

1963 BNL ENVIRONMENTAL MONITORING

Monthly Average Ar^{41} Radiation Levels, mR/wk(1)
Station Locations

Month	Onsite E-10	Onsite E-11	Onsite E-12	Perimenter E-2	Perimenter E-4	Perimenter E-7	Perimenter E-9	Offsite 0-6
January	1.46	2.08	2.59	0.45	0.26	0.28	0.76	0
February	0.06	2.22	2.92	0.06	0.11	0.76	0.76	0.03
March	0.68	2.58	2.25	0.58	0.05	0.57	0.57	0.03
April	0.78	1.94	2.59	0.14	0.19	1.08	0.74	0.01
May	0.44	6.55	5.19	0.43	0.24	0.41	1.86	0.01
June	0.85	2.31	2.43	0.82	0.32	0.57	0.74	0.02
July	0.35	2.56	4.30	0.47	0.25	0.42	1.49	0.03
August	0.64	3.18	5.02	0.17	0.01	0.48	1.02	0
September	1.63	3.07	3.83	0.21	0.70	0.27	0.55	0.03
October	1.51	2.68	3.46	0.41	0.57	0.53	0.80	0.02
November	0.90	2.16	3.40	0.31	0.39	0.45	0.58	0.04
December	0.58	1.60	1.17	0.19	0.25	0.39	0.35	0.04
Average	0.82	2.74	3.26	0.35	0.28	0.52	0.85	0.02
Peak Weekly Average	3.23	12.91	7.57	1.97	1.94	1.63	2.29	1.08

Estimated error at 90% confidence level, ±0.25 mR/wk.

(1) From Brookhaven National Laboratory Publication BNL 915.

TABLE E.2-8

AVERAGE ANNUAL GAMMA DOSE (mRad/yr) FOR
BGRR 1963 - PREDICTED AND OBSERVED

Station	Sector	Distance (Meters)	Dose (mRad/yr) Measured(1)	Calculated(3)
E-2	NW	1100	21(2)	20
E-4	WSW	2200	14	13
E-7	SE	2500	28	30
E-9	NE	2750	45	34
E-10	W	520	40	42
E-11	S	420	140	122
E-12	NNE	460	158	156

(1) Based on a 10 month average

(2) Based on a 9 month average

(3) Based on an 85% operation factor giving a release rate of 0.127 ci/sec

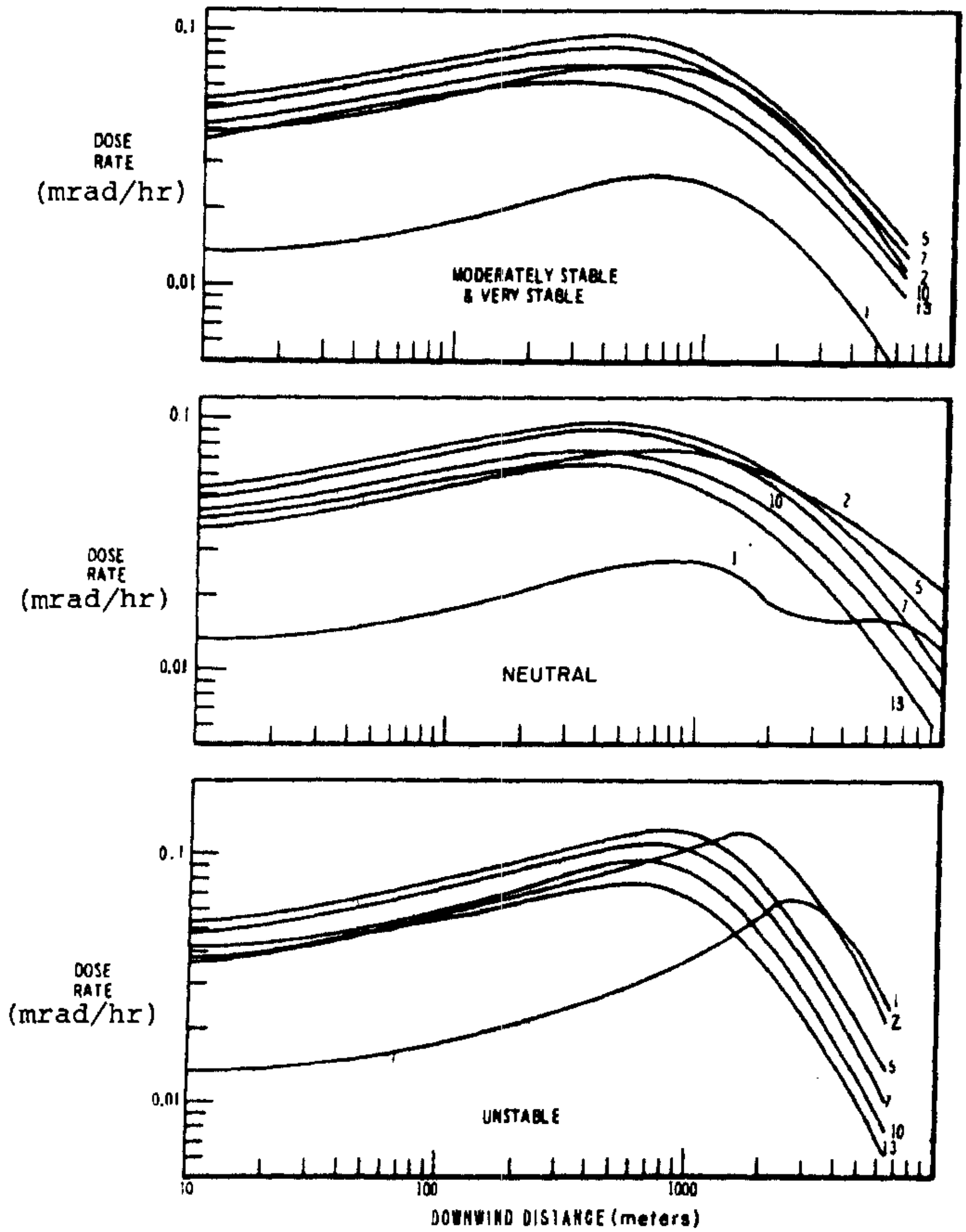

FIGURE E. 2-1
GAMMA DOSE RATE FOR
VARIOUS WIND SPEEDS AND
STABILITIES FOR BGRR STACK
(RELEASE RATE 0.127 Ci/sec)
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

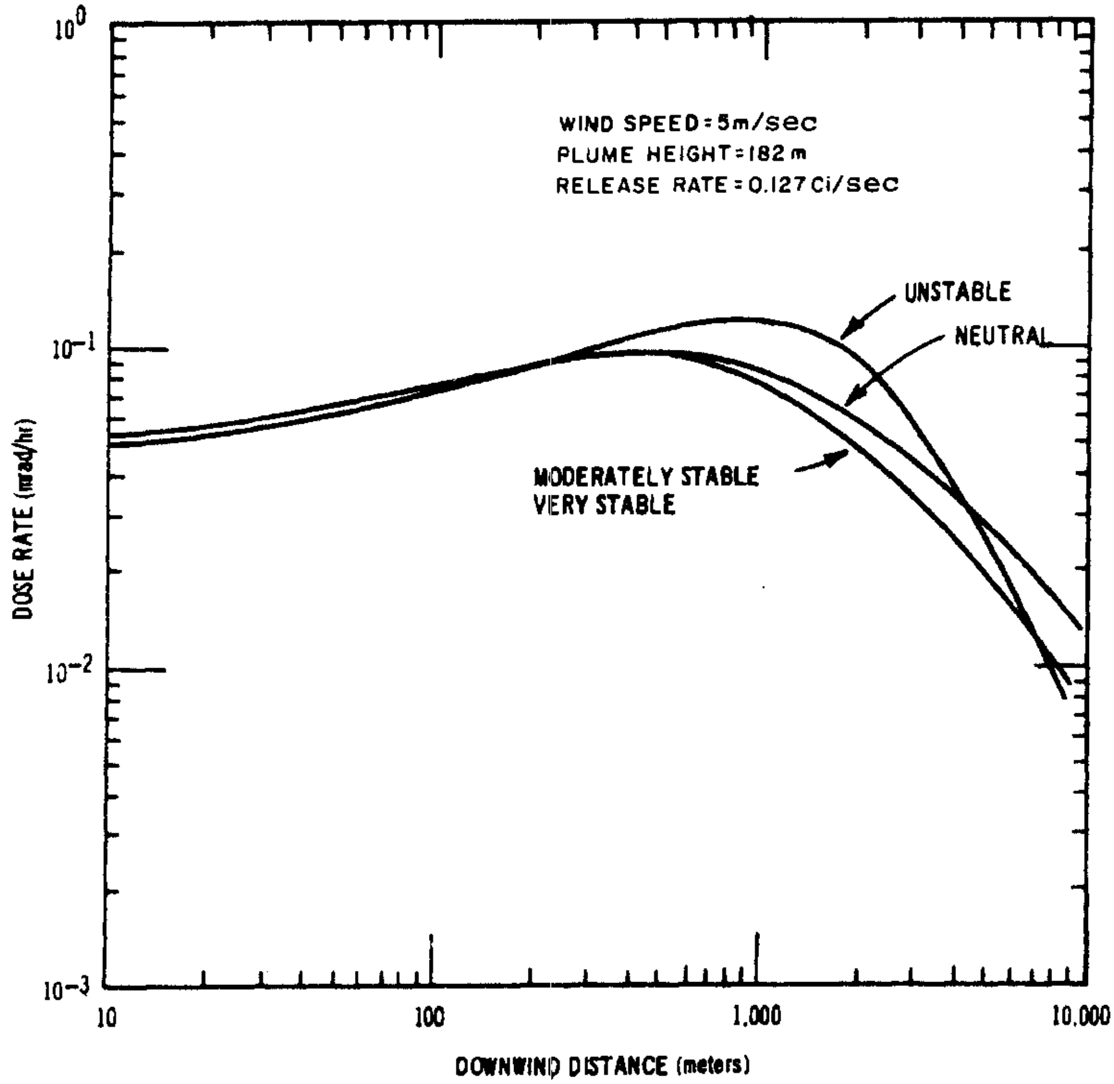

FIGURE E. 2-2
GAMMA DOSE RATE IN AIR FOR VARIOUS STABILITY CONDITIONS
PILGRIM NUCLEAR POWER STATION
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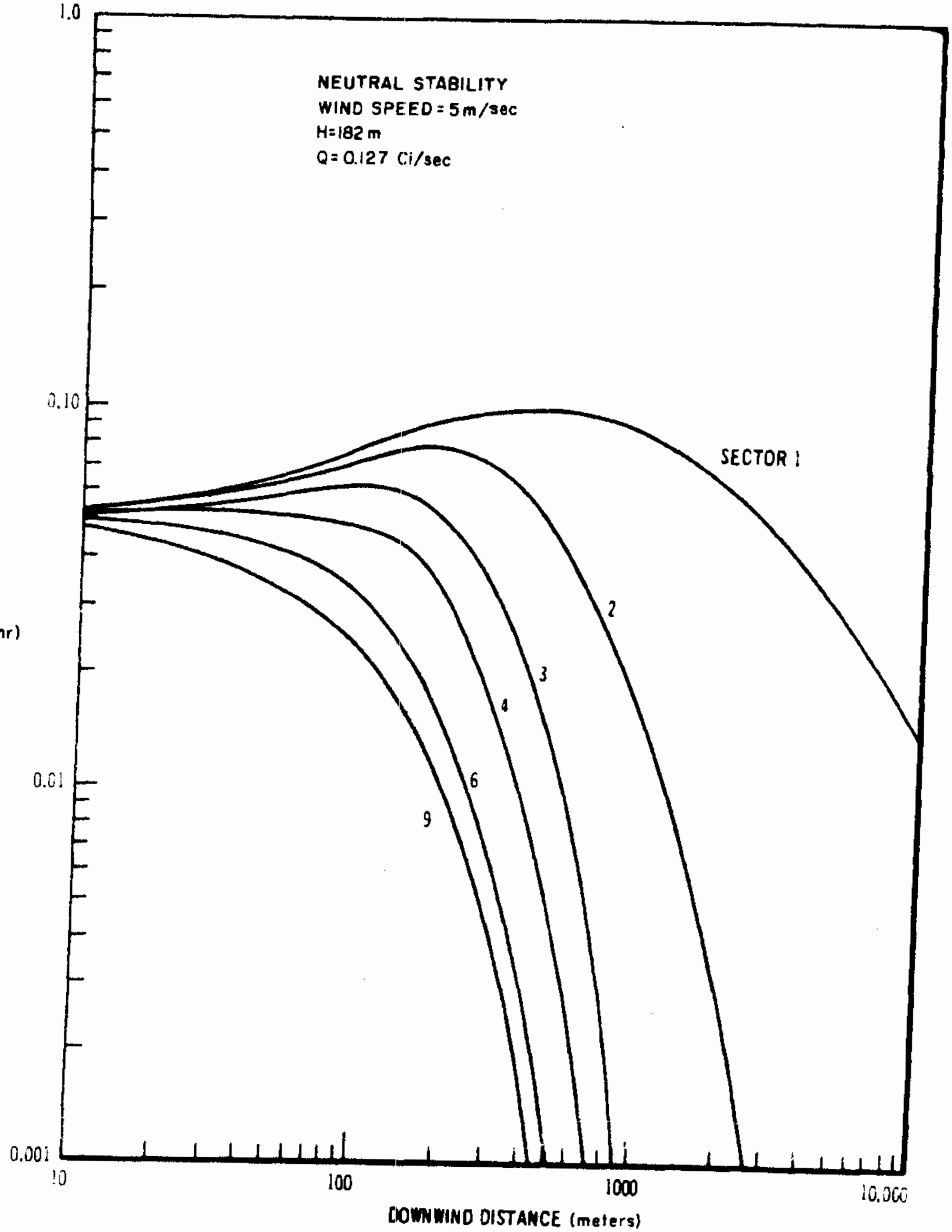

FIGURE E.2-3
DOSE RATE IN EACH SECTOR
PILGRIM NUCLEAR POWER STATION
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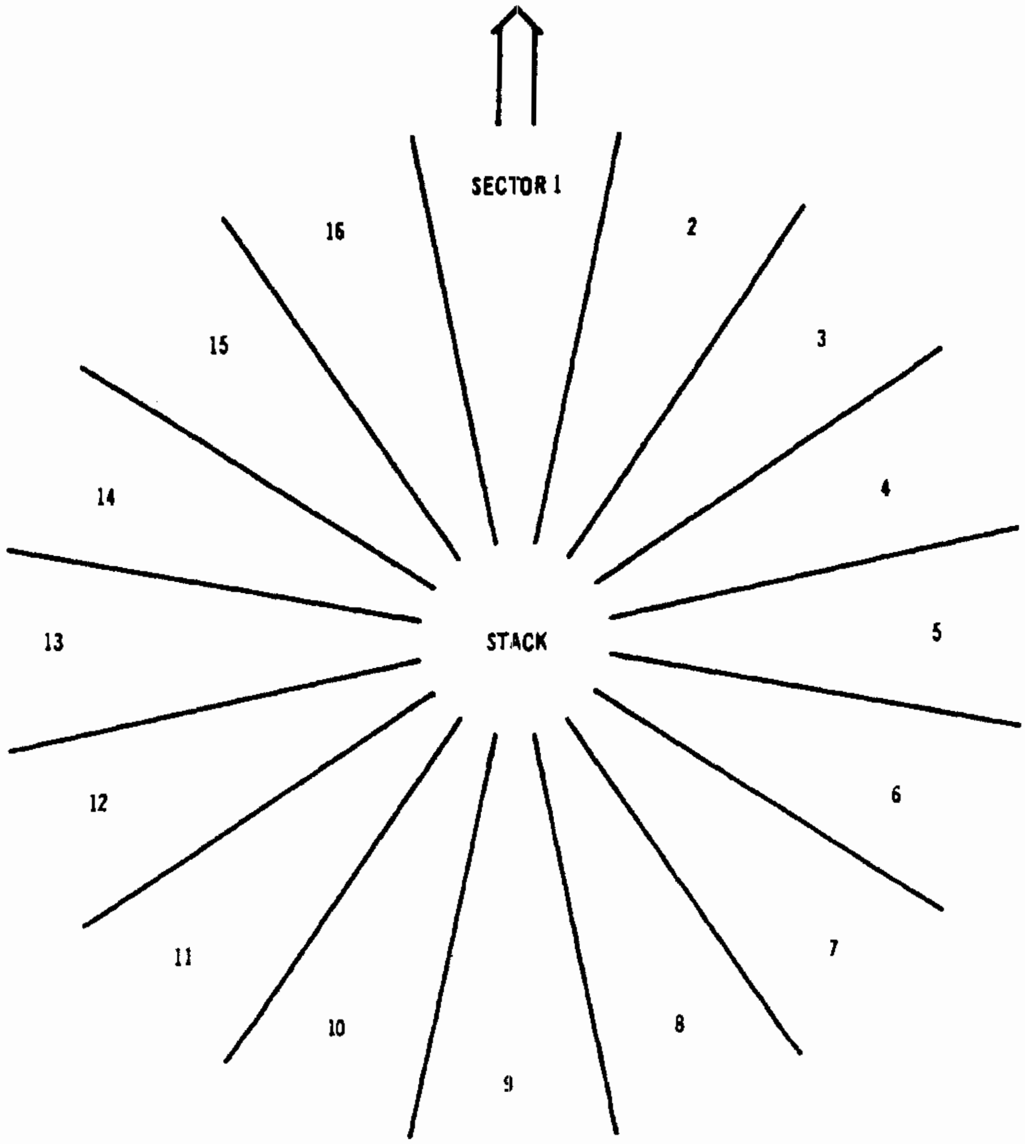
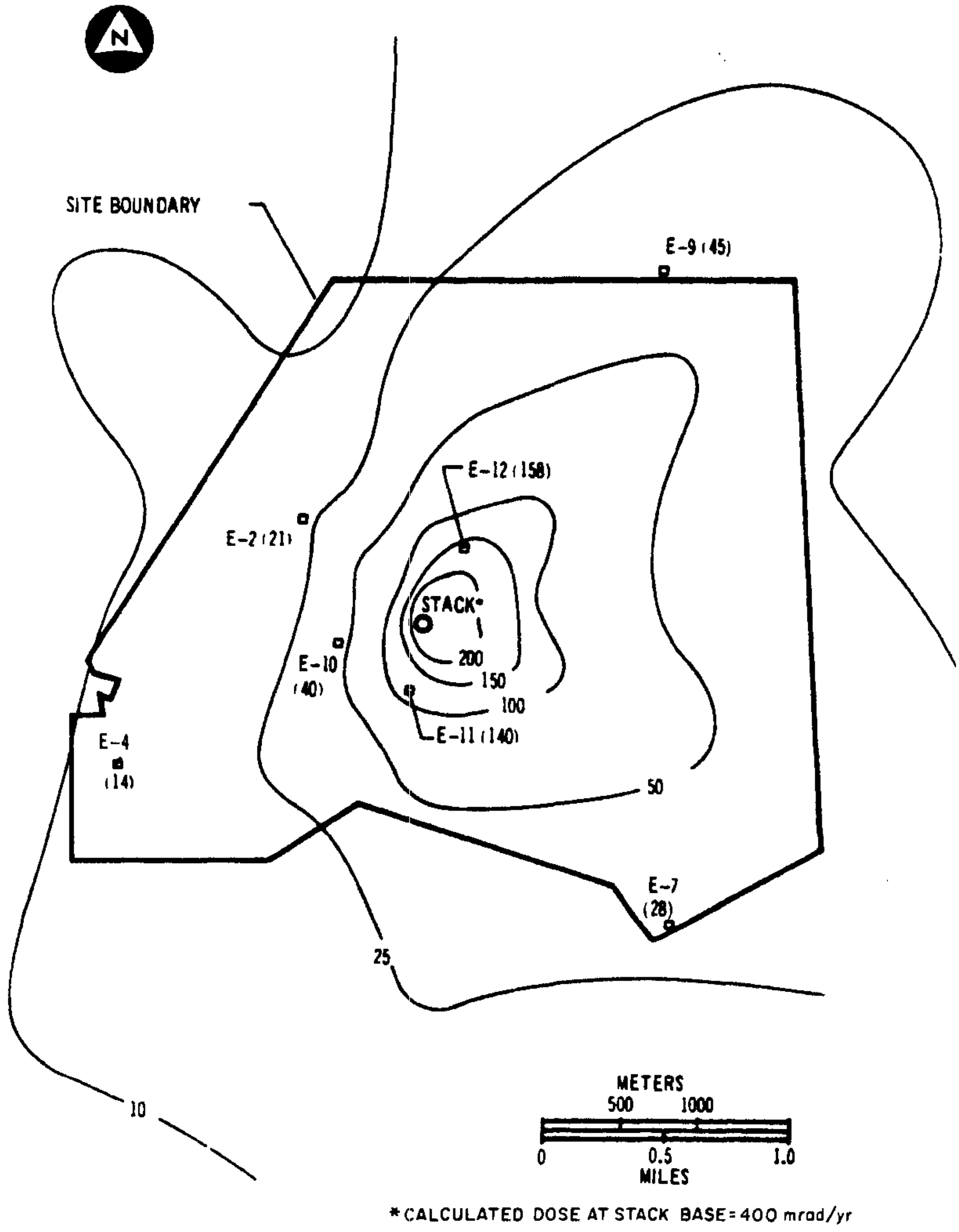


FIGURE E.2-4
NOMENCLATURE OF SECTORS USED FOR AVERAGING
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT



*CALCULATED DOSE AT STACK BASE=400 mrad/yr

FIGURE E. 2-5
WHOLE BODY GAMMA DOSE (mrad/yr)
PATTERN AROUND BGRR STACK
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

E.3 STACK RELEASE LIMIT CALCULATIONS FOR PILGRIM STATION SITE

The same methods as described above were used in the following calculations for the Pilgrim Station site.

E.3.1 Plume Rise

Characteristics of the stack offgas exhaust system design are significant to the calculation of groundlevel doses. Plume height above the stack is a function of vertical momentum and buoyancy effects. Greater plume heights result in lower groundlevel dose effects. Credit was taken for momentum effects only since relatively little heat would be discharged through the stack.

The following values were used as input to the Holland plume rise model (modified) Equation E.1-12:

Stack height (H)	330 ft (100.5m): above 70 ft MSL stack grade
Inside Diameter	2.44 ft (0.744m)
Exit Velocity	62 ft/sec (18.9m/sec)
Correction factor (K)	2.1

Using the above data, the effective plume rise above the stack exit was calculated for various wind speeds at stack height as follows:

Wind Speed ($\bar{u}$) mph	m/sec	Plume Rise(ΔH) (m)	Effective Plume Height ($H+\Delta H$) (m)
0-3	1	44	144
4-7	2	22	122
8-12	5	8	108
13-18	7	6	106
19-24	10	4	104
>25	>13	3	103

E.3.2 Terrain Effects

In calculating downwind ground level concentrations and doses, the effective plume height above the ground must be known. As the overhead plume travels downwind toward rising terrain, the plume will tend to follow the general contours of the ground. Usually the plume would flow over and/or around any significant obstacle. However, for the sake of conservatism, the height of the terrain (Z) is subtracted from the stack height (H) plus plume rise height (ΔH). This is done for the 16 wind directions and for various distances. Figures E.3-1 and E.3-2 illustrate terrain cross sections around the site. The wind directions not plotted correspond to trajectories that are over the ocean. It is from these figures that the plume height above the ground was determined.

E.3.3 Whole Body Dose Calculations

E.3.3.1 Gamma Dose

The procedure for calculating annual gamma dose consists of calculating the dose rate at various points during each different meteorological condition, weighting the dose rate by the frequency of occurrence and summing over the year to determine total dose. Calculations have shown that gamma dose rate results are not strongly dependent on atmospheric stability. This is in contrast to ground level air concentration calculations where stability is important. See Figure E.3-3. As an example, Figure E.3-4 shows the difference in gamma dose rate at distances beyond 100 m. This difference does not exceed a factor of about 2. Figure E.3-4 is for an average wind speed of 5 m/sec and a plume height of 100 m, and a continuous stack release rate of 1 Ci/sec of noble gases.

Gamma dose rates as a function of wind speed for stable, neutral, and unstable conditions are shown on Figures E.3-5 through E.3-7. All curves assume a continuous stack release rate of 1 Ci/sec of noble gases. For distances less than about 800 to 1000 m, the dose rate contribution from adjacent sectors should be considered. Figure E.3-8 shows such dose rates for the direction in which the plume is traveling (Sector 1) and for the adjacent directions to the right (Sectors 2, 3 etc.). As previously discussed, dose rates in Sectors 2 and 3 are equivalent to dose rates in Sectors 16 and 15, respectively. See Figure E.2-4.

Gamma dose rate calculations were done for many downwind dose points. These dose rates were weighted by the frequency of occurrence of wind speed, direction, and atmospheric stability in accordance with the observed meteorological data (May 1968 to April 1969) and summed to give a total "air dose" for the year. Gamma dose rate calculations and integrated doses over a one year period resulting from the radioactive source in the metal stack were calculated as a function of distance from the base of the stack. A tabulation of the sum of these doses is included on Table E.3-1. In all cases dose is calculated for a fixed point (air dose) with no consideration of human occupancy or shielding. However, credit will be taken for such effects in defining the actual "stack release rate limit" in the results below (Section E.3.3.4).

The closest site boundary in the direction of the highest dose is usually taken as the basis for determining the continuous stack release rate limit. The direction of the "maximum" calculated dose may or may not be in the direction of existing population centers. For the Pilgrim site the maximum offsite dose occurs 530 m away from the stack towards the southwest. See Table E.3-1. Terrain effects were considered by reducing the effective plume height for each distance and direction.

The whole body gamma doses (annual average) at the site boundary resulting from a 1 Ci/sec stack release rate are shown on Figure E.3-9 for each of the 16 directions. Wind direction frequency

toward each of these site boundary dose points is also shown on the figure. Table E.3-1 lists the annual dose and site boundary distance for these 16 dose points.

E.3.3.2 Beta Dose

The range of beta particles in air is only a few meters distance. Hence, for beta calculations, a cloud of material released via a stack and which expands to large dimensions at downwind distances where the cloud has reached groundlevel, is frequently considered an "infinite" cloud. In such a cloud, the air dose rate is calculated assuming that the rate of energy release per unit volume in the cloud is equal to the rate of absorption in that volume (no buildup). The body is considered a small volume within the flux of the cloud and causes no perturbation in the flux.

Beta flux incident on the human body comes from one direction only, so that the air dose rate at the surface of the body is only 1/2 of that in the air. In addition, the cloud is not infinite since the ground represents a boundary to the cloud, such that at the ground the cloud is a hemisphere of "infinite" radius. It approaches the "infinite" cloud at some height above the ground equal to the range of the betas in air. There will be a variation in the dose rate from the head to the foot of an individual with the highest dose rate at the head. This factor varies from one half at the ground to one at heights greater than the range of betas in air. Taylor[8] has computed this effect to show that the average dose to the body of a person 1.8 m tall is about 0.64 times the semi-inifinite cloud dose. This factor applies for mixed fission products with maximum energies of about 1-2 Mev.

The following beta dose equation[9] is used and modified:

$$D_\beta = 0.457\bar{E}_\beta \chi \qquad \text{(E.3-1)}$$

This equation is multiplied by 0.5 for the beta flux factor discussed above and by 0.64 to account for the average dose to the body. Converting Equation E.3-1 into a dose rate yields the equation used in the analysis.

$$(DR)_\beta = 0.53 \times 10^6 \ (\chi) \ \bar{E}(\text{mRad/hr}) \qquad \text{(E.3-2)}$$

Substituting $(\chi)^i_{avg}$ for (χ) gives the average beta dose rate for the ith meteorological condition. Since the range of betas in air is quite short, the annual total beta dose in a given direction is the sum of the dose rates (in mRad/hr) during each ith condition accompanied by wind blowing in that direction weighted by the annual frequency (in hours) of occurrence. Conversion of this dose into a

dose delivered to an individual requires adjustments to take the shielding effect of clothing into account.

Table E.3-2 shows the beta dose calculated for the direction having the highest gamma dose (southwest sector) which is the direction in which the stack release limit is based. There is no dose contribution from adjacent sectors due to the short range of betas in air.

E.3.3.3 Sea Breeze Effects

The Pilgrim Station site due to its proximity to water experiences sea breeze flow circulations during certain synoptic weather patterns. A study[10] was performed at the Pilgrim Station site to determine the behavior of an elevated plume discharged into the transition zone from over water to over land air flow and, in particular, to investigate the occurrence of "pseudo-fumigation" referred to by Hewson and Van derHoven [11]. The conclusions discussed in the study[10] referred to above indicate that the "Pseudo-fumigation" conditions were observed during the studies, however, the point at which fumigation occurs varies considerably over a short span of time. From the results and conclusions of the study (see Appendix I), it was recommended that short term dosage calculations for accident analyses, consider the effects of fumigation occurring at or beyond the site boundary.

Since this phenomenon appears to be of short duration, no seabreeze effects are included directly in the whole body dose calculations reported in this appendix. Such effects were considered in the accident analyses reported in Section 14, Station Safety Analyses.

E.3.3.4 Results

Utilizing Pilgrim site meteorological data (220 ft tower), the maximum offsite dose to a fixed point can be seen on Table E.3-1 to be southwestward. The total ground level whole body radiation dose in air from a continuous stack release of 1.0 Ci/sec all year is shown on Table E.3-2. The total represents the sum of gamma and beta doses for a year to a fixed point in the southwestward direction. Slightly higher annual doses are observed in the north-northeast direction, but this direction is out to sea.

The stack release limit is based on the total dose to an individual from all dose contributions at a distance of 530 m towards the southwest. The maximum dose (in air) at this point is calculated to be 500 mRem/yr from a 0.44 Ci/sec continous stack release rate. The 500 mRem/yr value is established by 10 CFR 20.105.

If an individual were present at the maximum dose point for every hour of a year, he would receive a whole body dose of 500 mRem from the above stack release rate. It is incongruous to assume that an individual would be at such a location all of the time. Therefore, credit has been taken for such effects in evaluating dose effects to persons beyond the site boundary. To receive the maximum dose a

person would have to stay at the appropriate site boundary point continuously for the entire year without benefit of shelter of any type. It is estimated that the use of normal occupancy and shielding factors would permit increasing the release rate by a factor of two or three without exceeding the 10CFR20 limits. Therefore, the stack release limit calculated above (0.44 Ci/sec) is increased to 0.90 Ci/sec. This is the annual average limit which is proposed as an operating limit.

Since the 500 mRem offsite dose is an average over a year, the stack release rate is also an average. In order to ensure that the range of release rates averaged over one year are not excessive, an upper bound to the release rate is applied for short time periods. A factor of 10 times the annual average stack release rate for a period not to exceed one day is used.

The following summarizes the stack release rate limits for the emission of noble radiogases.

Release Period	Stack Release Rate
Annual average continuous	0.90 Ci/sec
Short term (not to exceed one day)	9.0 Ci/sec

E.3.4 Internal Dose Calculations

E.3.4.1 Internal Dose from Inhalation

Internal dose from inhalation may be related directly to an annual average groundlevel air concentration. The average air concentration at groundlevel is as given in Equation E.1-4 for any specific meteorological condition. Figure E.3-3 illustrates this for a 5 m/sec wind speed for various stability conditions. The annual average concentration is the sum of each of the averages for various wind speeds and stabilities weighted by their frequency of occurrence. This weighted concentration may then be compared to the MPC_a (Maximum Permissible Concentration in air) given in 10CFR20, Appendix B, Table II for the isotope or mixture of isotopes of interest. These MPC_a values are equivalent to an annual dose to an individual of 500 mRem.

The annual average ground level air concentrations were calculated using the 220 ft tower meteorological data for the site. Table E.3-3 shows the results of calculations for nine distances (ranging from the nearest site boundary to 16,000 m) and for 16 directions.

E.3.4.2 Internal Dose from Ingestion

Radioactive materials which deposit on vegetation and on the ground can cause radiation dose from consumption of certain agricultural products. For certain food chains, concentration effects exist. One radioisotope which exhibits such effects is I-131. The appropriate food chain is air-pasture-cow-milk-infant thyroid.

On the other hand, the MPC_a for I-131 in air is based on exposure via the air-lung-thyroid route. The milk exposure mode is far more limiting. That is, the thyroid dose from breathing air of any given I-131 content is much less than the thyroid dose to an infant drinking milk solely from cows feeding from pastures exposed to the same air. This is a result of deposition of iodine on pasture grass, concentrating the iodine due to the large area of grass eaten by the cow, and relatively efficient transfer to the milk. This effect must be considered when relating an emission rate for iodine to an environmental dose where there are cows involved. Current U.S. practice, in context of USNRC licenses associated with stack emission, assigns a reconcentration factor of 700 to I-131. Thus for example, the MPC_a for I-131 in 10CFR20 is 1×10^{-10} μCi/cc for inhalation but is

$$\frac{1 \times 10^{-10}}{700}$$

for ingestion consideration for a baby with an assumed 2 gram thyroid drinking 1 liter of milk per day.

The maximum annual average offsite air concentration is calculated to occur in a direction SW of the stack.

TABLE E.3-1

PILGRIM STATION SITE
ANNUAL AVERAGE GAMMA DOSE AT GROUNDLEVEL FROM CONTINUOUS RELEASE RATE of 1.0 Ci/sec (mrem/yr)
Distance From Stack (km)

Direction From Stack	Site Boundary Distance		1.0	2.0	3.0	5.0	8.0
N[1]	(0.4)[2]	361	226	110	60	26	10
NNE[1]	(0.25)	1088	672	337	190	85	36
NE[1]	(0.23)	588	366	183	103	46	20
ENE[1]	(0.24)	689	373	188	106	48	21
E[1]	(0.34)	736	470	238	135	61	26
ESE[1]	(0.75)	515	434	219	123	55	24
SE	(0.87)	362	363	168	85	35	15
SSE	(2.15)	125			74	28	13
S	(1.07)	527		261	264	54	20
SSW	(0.54)	810	743	352	217	35	21
SW	(0.53)	1081	722	350	100	42	20
WSW	(0.63)	932	756	223	97	53	26
W	(0.38)	710	562	132	72	38	16
WNW	(0.33)	763	586	157	86	37	15
NW	(0.34)	531	224	112	63	29	12
NNW	(0.41)	319	152	77	44	20	9

NOTES:

(1) The calculated gamma doses are of academic interest only since they are for directions over the ocean. Site boundary doses are at the shoreline.

(2) Numbers in parentheses are site boundary distances in km.

TABLE E.3-2

TOTAL ANNUAL GROUNDLEVEL RADIATION DOSE IN AIR
FROM STACK RELEASE RATE OF 1.0 Ci/sec

Distance		Dose (mrad/yr)		
Miles	Meters	Gamma(γ)	Beta(β)	Total
0.33	530	1081	47	1128
0.62	1000	737	27	764
1.24	2000	353	11	364
1.86	3000	102	5	104
3.11	5000	42	1	43
4.97	8000	20	1	21

NOTE:

1. The doses shown are for the direction southwest of the stack which give the maximum values for the closest offsite land location.

TABLE E.3-3

PILGRIM STATION SITE
ANNUAL AVERAGE INTEGRATED GROUNDLEVEL AIR CONCENTRATION
($10^{-[illegible]}$ µCi/cc per Ci/sec released)
Distance From Stack (km)

Direction From Stack	Site Boundary Distance		0.8	1.2	1.6	2.4	3.2	4.8	8.0	16.0
N [3]	(0.4) [2]	9.3 [1]	9.7	7.2	5.3	2.1	2.0	0.99	0.40	0.12
NNE [3]	(0.25)	0.10	4.0	3.9	3.7	3.0	2.2	1.2	0.55	0.20
NE [3]	(0.23)	0.09	2.2	2.1	2.0	1.5	1.1	0.63	0.28	0.10
ENE [3]	(0.24)	0.09	1.9	1.9	1.9	1.6	1.2	0.68	0.30	0.11
E [3]	(0.34)	0.27	2.1	2.3	2.4	2.1	1.6	0.92	0.41	0.15
ESE [3]	(0.75)	2.2	2.3	2.3	2.4	2.0	1.5	0.87	0.39	0.14
SE	(0.87)	2.8	-	2.7	2.3	1.5	1.0	0.49	0.22	0.09
SSE	(2.15)	1.6	-	-	-	1.4	0.95	0.45	0.20	0.08
S	(1.07)	8.4	-	7.4	5.2	5.5	3.3	1.6	0.72	0.15
SSW	(0.54)	7.4	14.2	8.7	6.1	5.9	3.6	0.78	0.44	0.21
SW	(0.53)	8.7	17.9	10.7	7.4	4.3	1.3	0.95	0.52	0.24
WSW	(0.63)	8.1	7.6	11.7	8.1	1.8	1.2	0.98	0.59	0.17
W	(0.38)	8.6	7.0	5.0	1.8	1.3	0.92	0.59	0.30	0.14
WNW	(0.33)	9.9	8.6	6.2	2.2	1.6	1.1	0.63	0.28	0.11
NW	(0.34)	1.1	2.1	1.2	1.1	0.88	0.65	0.37	0.18	0.08
NNW	(0.41)	0.56	0.91	0.54	0.60	0.53	0.40	0.24	0.12	0.07

NOTES:

1. Each number is to be multiplied by $10^{-[illegible]}$. Example: $9.3 \times 10^{-[illegible]}$ µCi/cc.
2. Numbers in parentheses are site boundary distances.
3. The calculated concentrations are of academic interest only since they are for over water directions. Site boundary concentrations are at the shoreline.

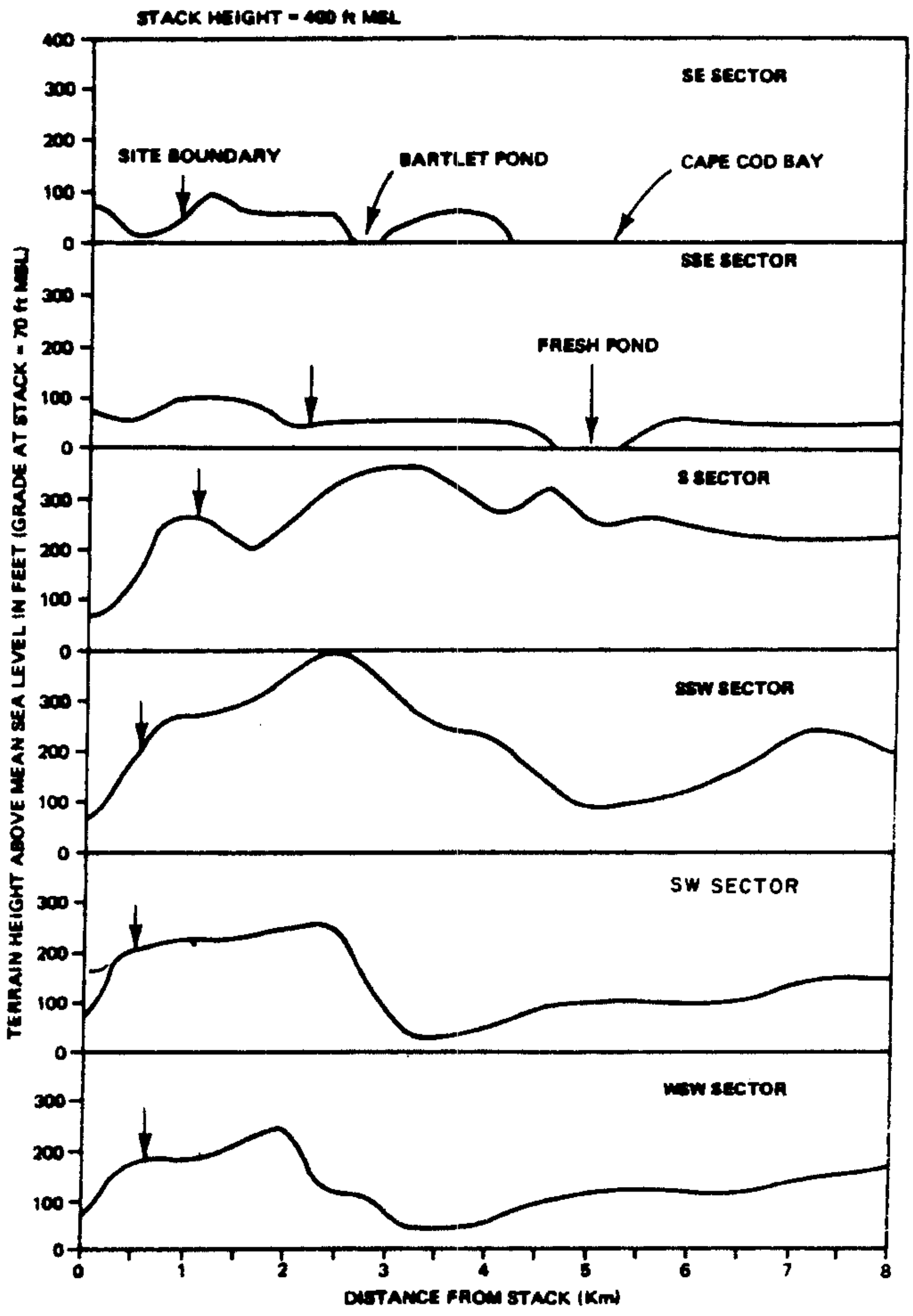


FIGURE E.3-1
TERRAIN CROSS SECTIONS AROUND THE PILGRIM STATION SITE

PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

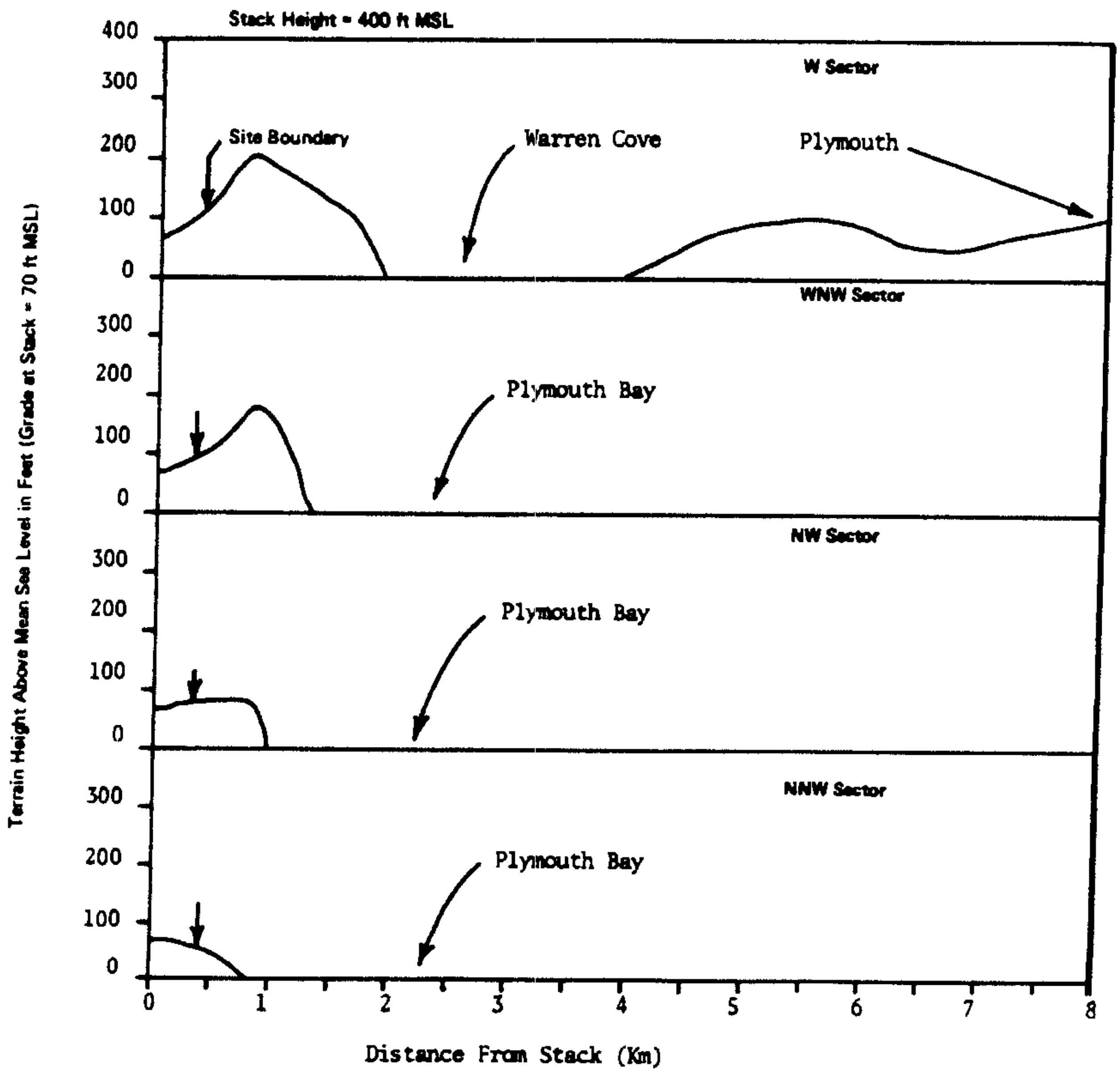


FIGURE E.3-2
TERRAIN CROSS SECTIONS AROUND THE PILGRIM STATION SITE
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

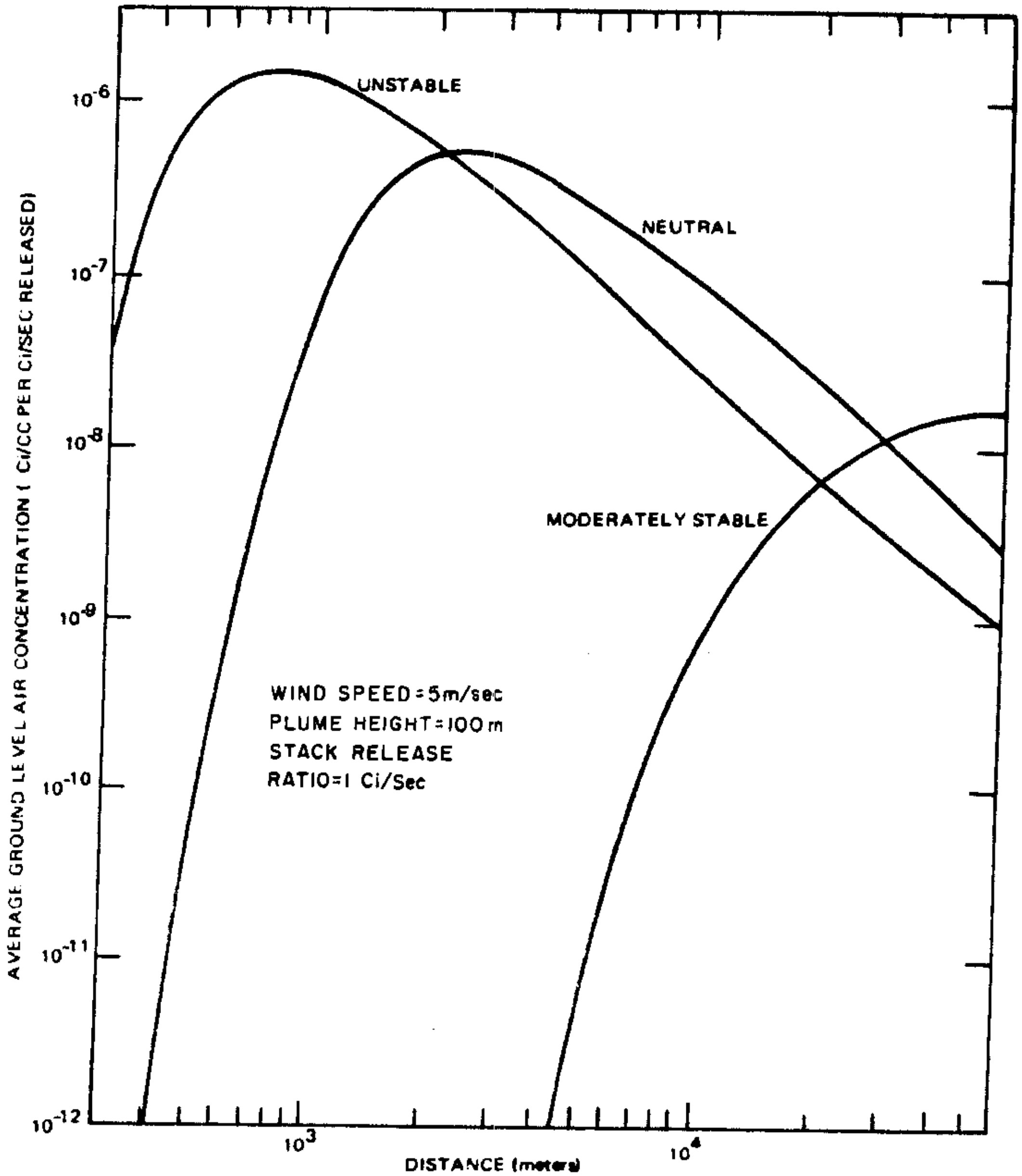


FIGURE E.3-3
AVERAGE GROUND-LEVEL AIR CONCENTRATION VERSUS DISTANCE FOR THREE STABILITY CONDITIONS
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

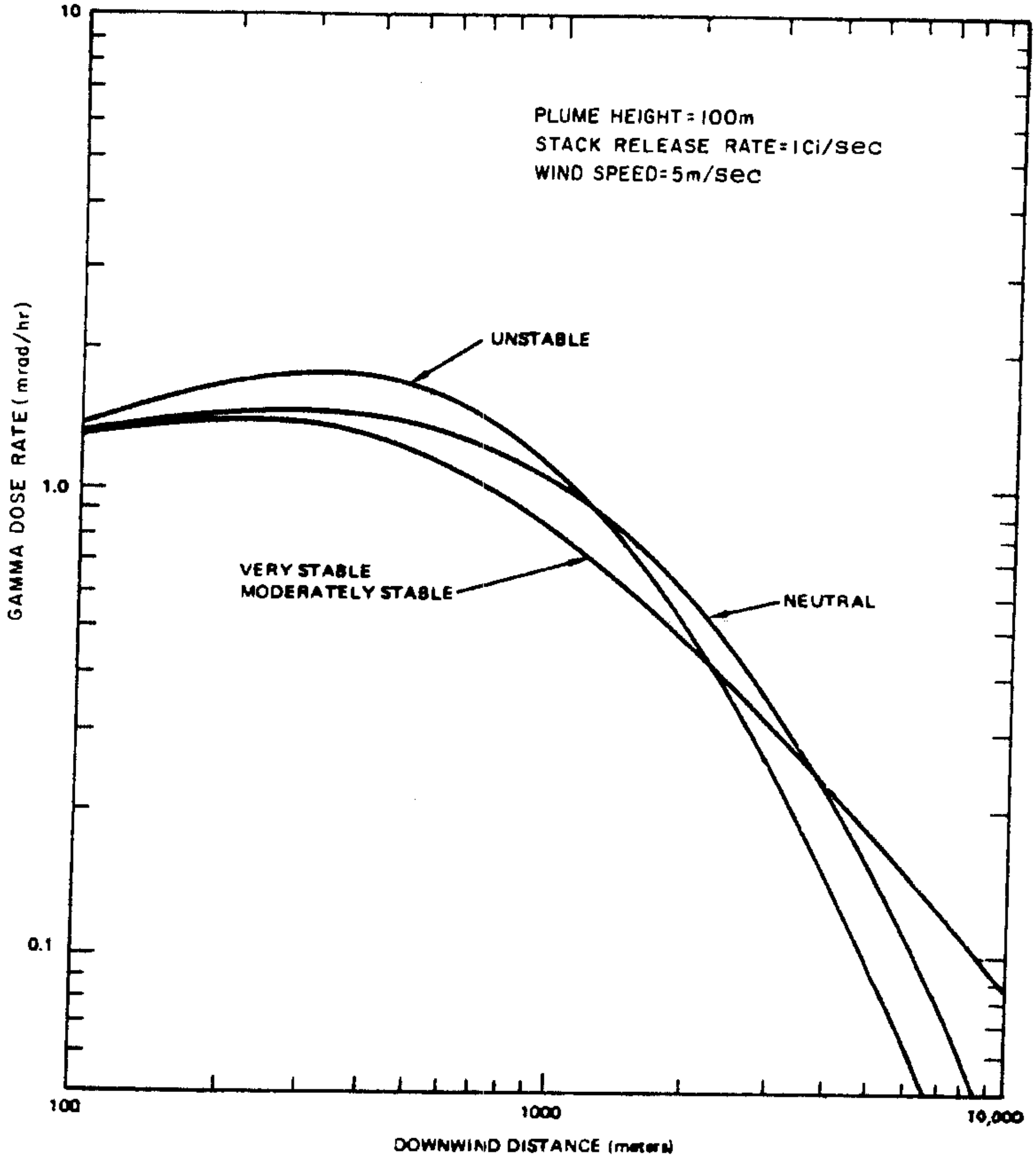


FIGURE E.3-4
GAMMA DOSE RATE FOR VARIOUS STABILITIES
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

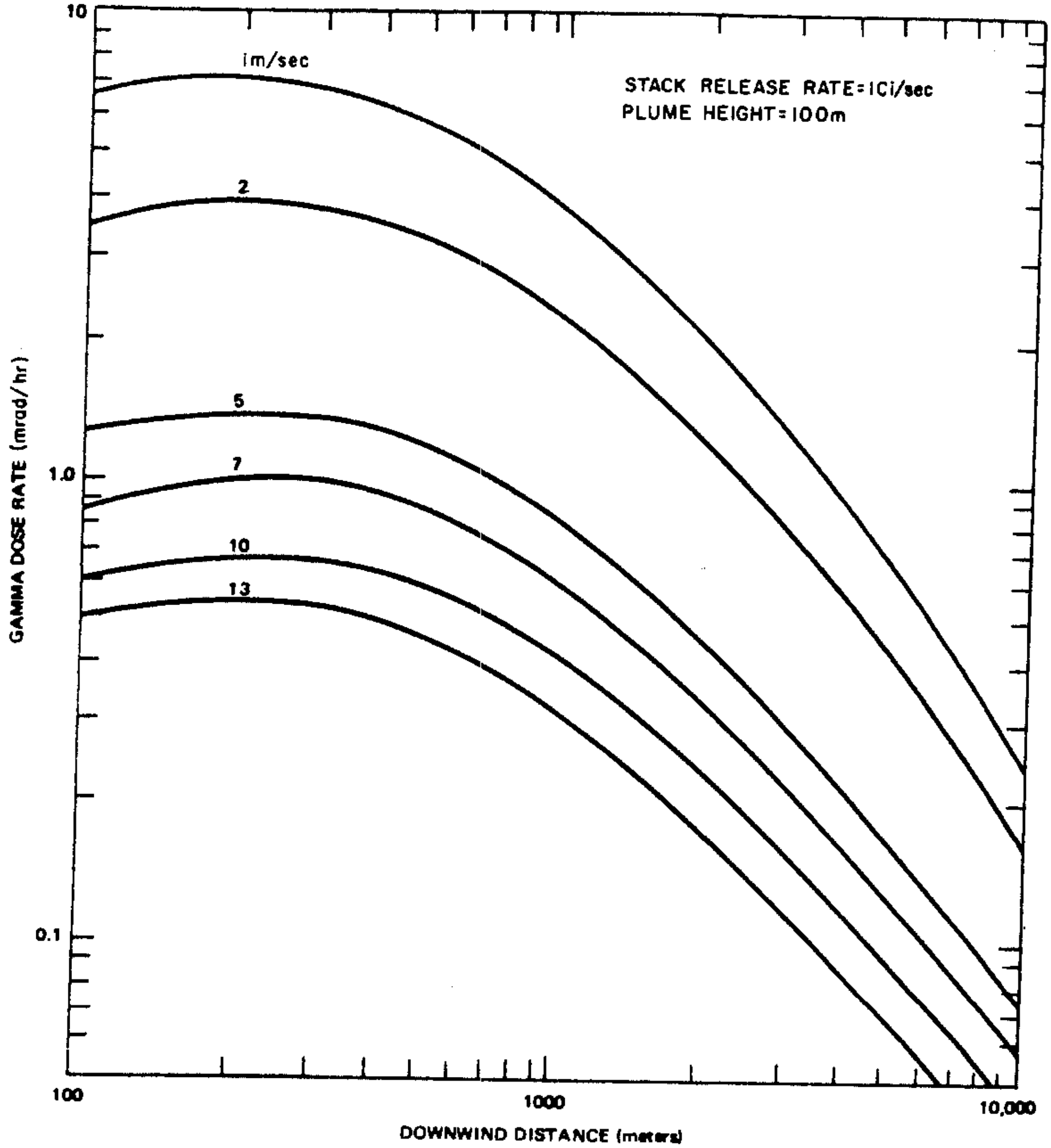


FIGURE E.3-5
GAMMA DOSE RATES FOR VARIOUS WIND SPEEDS (MODERATELY STABLE)

PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

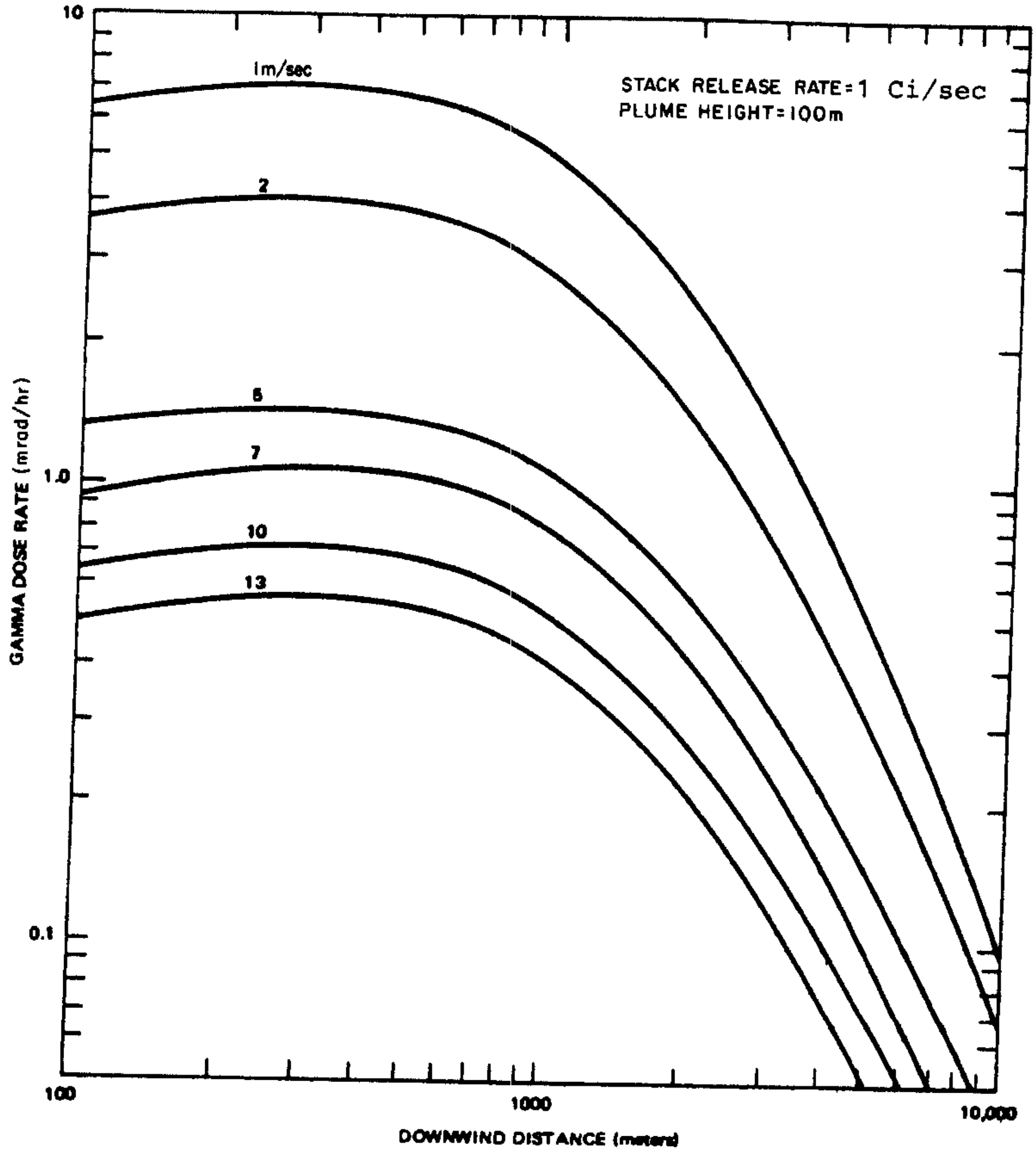


FIGURE E.3-6
GAMMA DOSE RATE FOR VARIOUS WIND SPEEDS (NEUTRAL)
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

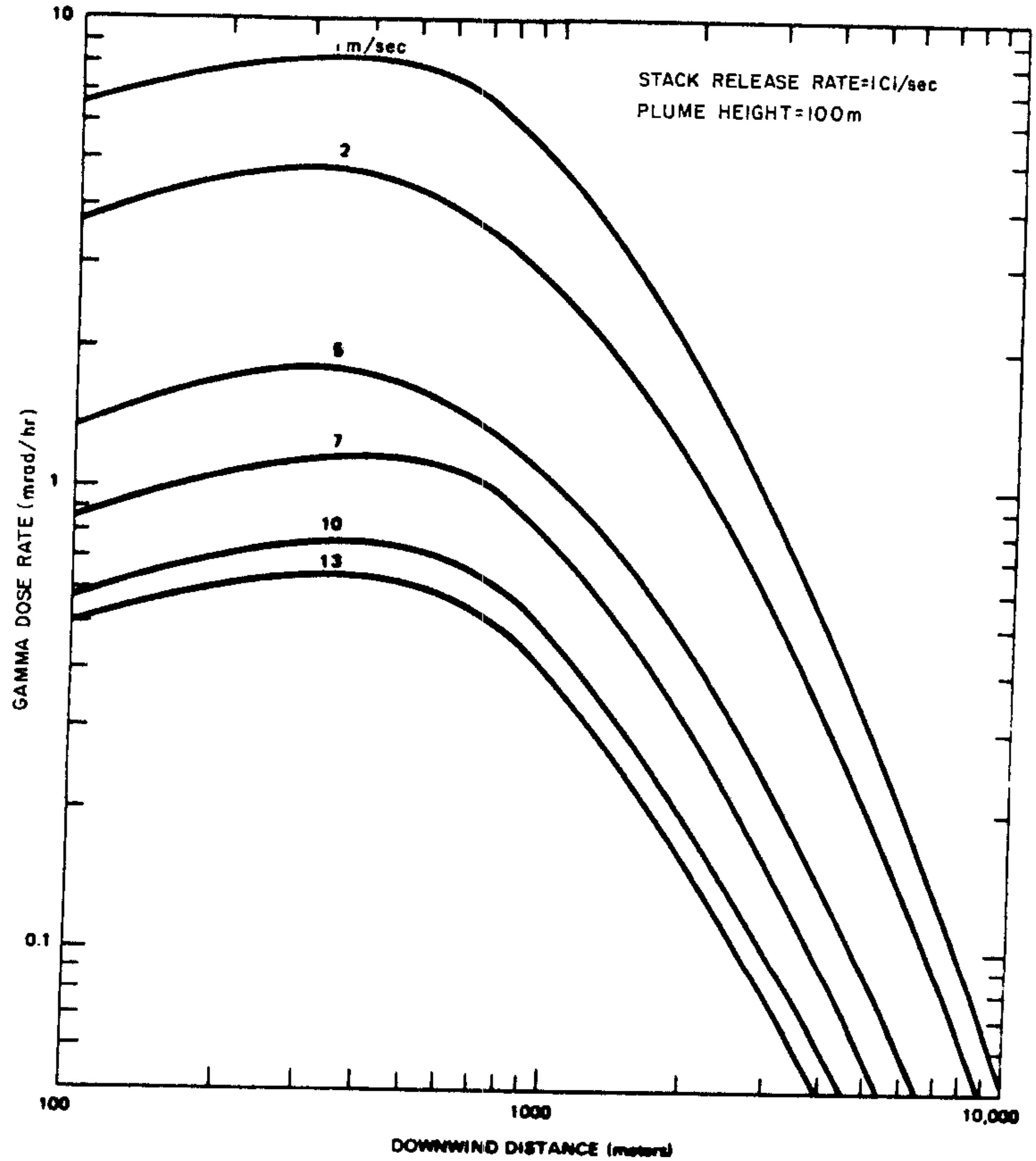


FIGURE E.3-7
GAMMA DOSE RATE FOR VARIOUS WIND SPEEDS (UNSTABLE)
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

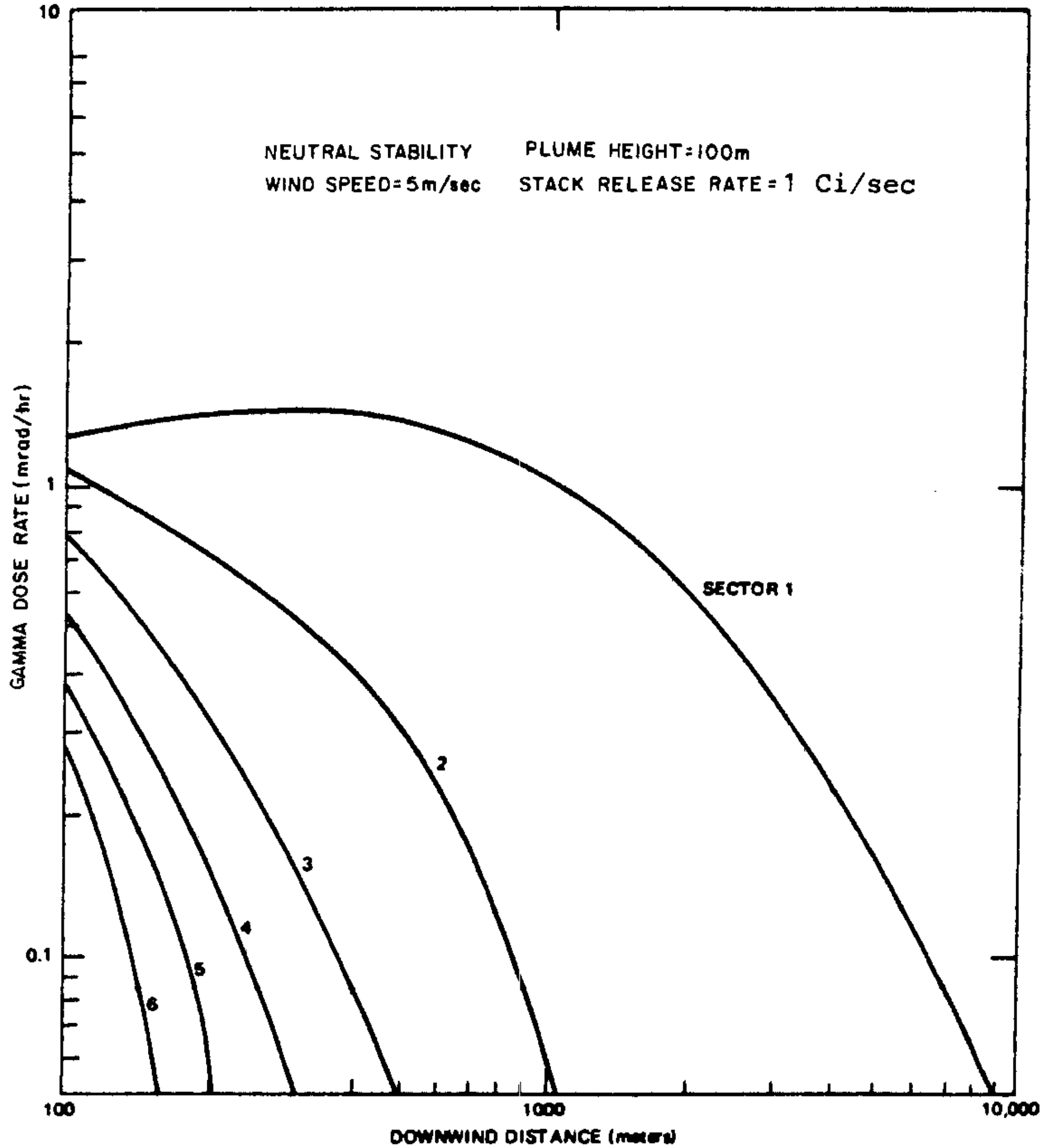


FIGURE E. 3-8
GAMMA DOSE RATES
IN NEIGHBORING SECTORS
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

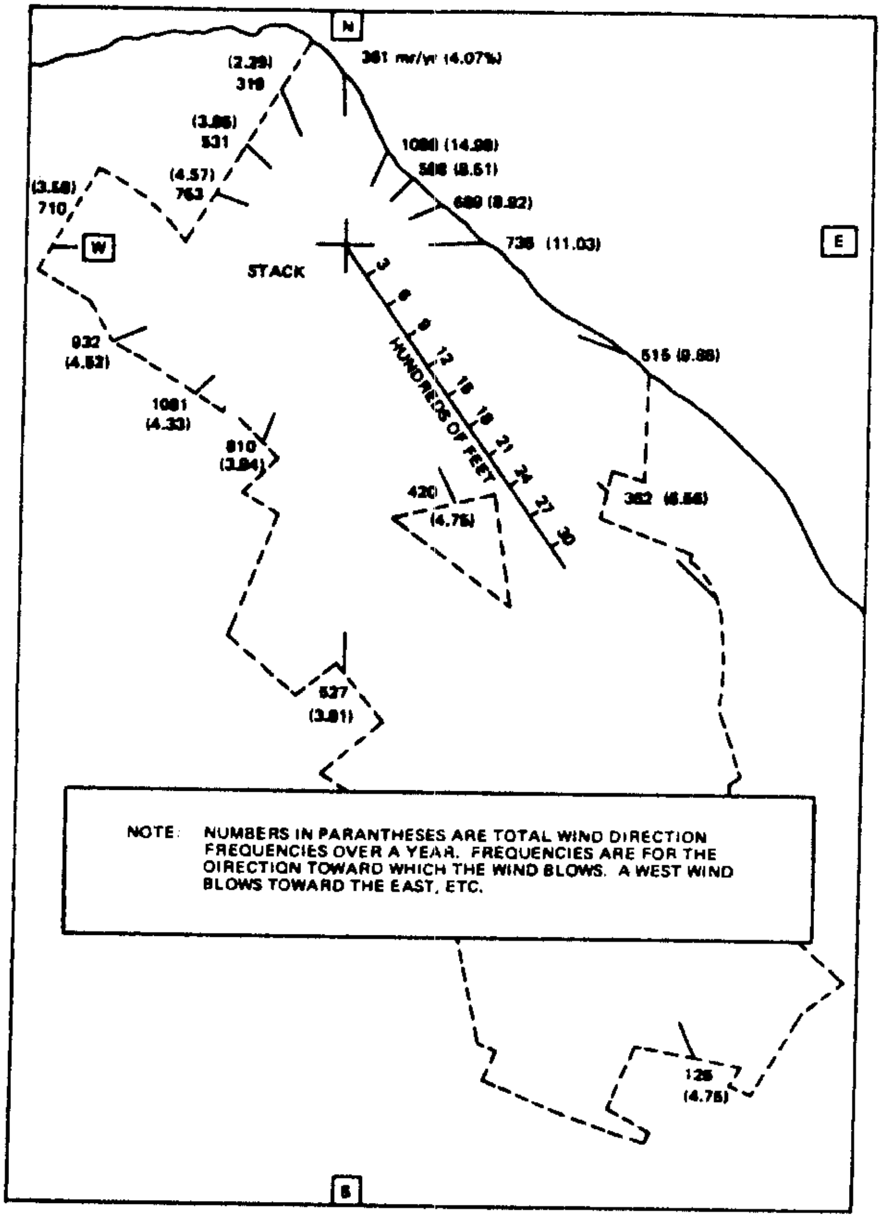


FIGURE E.3-9
ANNUAL AVERAGE GAMMA DOSE AT GROUND LEVEL AROUND THE SITE PERIMETER FROM A CONTINUOUS RELEASE OF 1.0 Ci/sec

PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

E.4 BUILDING EXHAUST VENT RELEASE

The ventilation air from the reactor building is routed to a common release location just higher than the top of the reactor building. The air within the secondary containment reactor building is designed to be relatively free from radioactive material. In this case the ventilation air leaving such a clean environment would contribute nothing to the annual average doses beyond the site perimeter. However, recent licensing trends by the NRC/DRL have required that an analysis be performed for such a "potential" release. Such an analysis has been done assuming that the roof top release of "clean" air contained a certain amount of activity resulting in an offsite dose of 500 mRem/yr.

In performing this analysis, it was assumed that one-half of the time the "plume" traveled horizontally downwind from the duct top; the rest of the time the plume experienced downwash in the turbulent wake of the reactor building complex. This certainly is conservative since the plume would rise above the duct top prior to bending over and traveling downwind. The resulting release rate over the year could amount to 0.14 Ci/sec without exceeding the 10CFR20 annual dose. This release rate, in combination with the stack release rate of 0.9 Ci/sec, would not exceed 500 mRem/yr.

E.5 SUMMARY

The method of calculating a stack release limit is given along with partial verification of the method using data from Brookhaven National Laboratory. The whole body gamma dose calculations are quite close to that observed at Brookhaven. The groundlevel integrated air concentration calculations give an order of magnitude type of verification due to the lack of sensitive field measurements.

The calculated annual average stack release rate limits assume conservative human occupancy and shielding factors. The short term release rate limits did not assume such factors, however.

The annual average release rate of I-131 for consideration of postulated exposure via the milk production and consumption mode is calculated to be 0.80μ Ci/sec.

It is recognized that a precise determination of dose from a certain emission from the stack is only possible by direct measurement. Such information will be provided by the environs surveillance program conducted at and around the site. If the stack emission ever reaches a level such that it is measureable in the environment, such measurements will provide a basis for adjusting the proposed stack limit long before the effect in the environment is of any safety concern. In this regard, it is important to realize that averaging the emission rate over a period of one year as permitted by 10CFR20 represents a very large safety margin between conditions existing at any one instant (any minute, hour, or day) and the long term dose of interest.

The stack release limit calculation was performed for the Pilgrim Station site using the 220-ft tower meteorological data. Calculations include whole body dose from the noble gases and internal dose from iodine 131. It is concluded that the noble gases will dominate and that the control of emission should be on these constituents of the stack effluent. From the calculations, a maximum dose of 500 mRem/yr to an individual about 530 m southwest of the stack is equivalent to a continuous stack release rate of 0.90 Ci/sec.

A short term release rate for a period not to exceed one day, of ten times the annual average release rate, will bound the range of release rates used in obtaining an average release value.

E.6 REFERENCES

1. Cramer, H.E. A Brief Survey of the Meteorological Aspects of Atmospheric Pollution. Bulletin of the American Meteorological Society, 40 (4), p 165-171.

2. Watson, H.C. and Gamertsfelder, C.C. Environmental Radioactive Contamination as a Factor in Nuclear Plant Siting Criteria. HW-SA2809, Feb 1963.

3. Glasstone, S. and Sesonke, A. Nuclear Reactor Engineering. D. Van Nostrand Co., New York, NY, 1968.

4. Moses, H.; Strom, G.H.; and Carson, J.E. Effects of Meteorological and Engineering Factors on Stack Plume Rise. Nuclear Safety, Vol 6(12), Fall 1964.

5. Slade, D.H. (ed.). Meteorology and Atomic Energy 1968. TID 24190, July 1968, pp 189-198.

6. Glasstone, S. Effects of Nuclear Weapons, rev. ed. Defense Atomic Support Agency, Dept. of Defense, April 1962.

7. Hull, A.P. 1963 Environmental Radiation Levels at Brookhaven National Laboratory. BNL 915, November 1964.

8. Meteorology and Atomic Energy. AECU 3066, 1955, p 100.

9. Slade, D.H. (ed.). Meteoeology and Atomic Energy 1968. TID 24190, July 1968.

10. Collins, G.F.; Scoggins, M.F.; Magyar, J.C. An Investigation of the Sea Breeze at Pilgrim Nuclear Site, Plymouth, Mass. The Travelers Research Corporation, Hartford, Connecticut, October 1969.

11. Van derHoven, I. Nuclear Safety. Vol. 8, No. 5, Sept-Oct 1967, pp 490-499.

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F.1 SUMMARY DESCRIPTION

This Appendix contains an evaluation of the design bases of the Pilgrim Nuclear Power Station (PNPS) nuclear facility by means of the July, 1967 draft of the proposed 70 General Design Criteria for Nuclear Power Plant Construction Permits issued by the Atomic Energy Commission. These proposed criteria were issued for comment in July 1967 on a draft basis and were addressed by PNPS prior to being adopted by the AEC as Appendix A to 10CFR50.

This Appendix used the draft as a basis for conducting a reference audit by subject matter and contains an evaluation of the design basis of PNPS relative to each of the nine groups of proposed criteria. (The then current draft of criteria were separated into nine groups by subject matter.) A statement of interpretation of the intent of the criteria and a discussion of the station design conformance is made for each group. A list of references where the subject material of the individual criterion is found in the FSAR is presented in Tables F.2-1 through F.2-9.

It was concluded by the AEC in an SER dated May 20, 1968 that PNPS conforms to these criteria.

F.2 CRITERIA CONFORMANCE

F.2.1 Group I-Overall Plant Requirements

If adopted, the proposed criteria in Group I would have established quality standards applicable to the design, fabrication, and erection of reactor systems and components essential to accident prevention or the mitigation of their consequences taking into account the forces which might be imposed by natural phenomena. These proposed criteria include design requirements for fire and explosion control and protection requirements applicable to shared systems or components and record keeping requirements. See Criteria 1 through 5 on Table F.2-1.

The BECo Quality Assurance Program covers the design, procurement, fabrication, manufacture, erection, and testing of components and systems for the plant. This program assures the use of all applicable design and construction codes and standards as a minimum (Criterion 1). The station seismic design criteria for Class I structures and equipment is based on appropriate static or dynamic analyses for the two postulated earthquakes, the "Operating Basis Earthquake" and the "Safe Shutdown Earthquake." In addition, all Class I structures and equipment except the stack and noncritical portions of the reactor building are designed to maintain their integrity when subjected to the forces of postulated tornado loading. All Class I structures are designed for flood protection in the event of a maximum probable flood (Criterion 2). Separation of redundant critical equipment is utilized in the design to minimize the effects of occurrences such as fires and explosions. Noncombustible and fire resistant materials are used throughout the facility (Criterion 3).

Criterion 4-"Sharing of Systems"-does not apply since PNPS is a single nuclear facility.

Records of design, fabrication, and construction for PNPS are either stored or maintained under BECo's control or are available for BECo's inspection if necessary (Criterion 5).

F.2.2 Group II-Protection by Multiple Fission Product Barriers

If adopted, the Proposed criteria in Group II would have required that nuclear power facilities be provided with multiple barriers to protect the public against the inadvertent release of radioactive material to the environs. See Criteria 6 through 10 on Table F.2-2.

The radioactive material barriers are the basic features which minimize release of radioactive materials and associated radiation levels. The Pilgrim Nuclear Power Station, like other boiling water reactors of General Electric design, provides the following means of containing or mitigating the release of fission products: (1) the high density ceramic UO_2 fuel; (2) the high integrity Zircaloy cladding; (3) the reactor vessel and its connected piping, pumps and valves which make up the nuclear system process barrier; (4) the drywell-suppression chamber primary containment; (5) secondary

containment-the Reactor Building, the Reactor Building Standby Gas Treatment System, and the main stack. The primary containment is designed, fabricated and erected to accommodate without failure, the maximum pressure and temperature resulting from or subsequent to any failure of any coolant pipe within the primary containment.

The secondary containment, encompassing the primary containment, provides secondary containment when the primary containment is closed and in service, and provides for primary containment when the primary containment is open (e.g., during refueling periods). The two containments and such other associated engineered safeguards as may be necessary are designed and maintained so that offsite doses which result from postulated design basis accidents are below the guideline values set forth in 10CFR100 (Criterion 10).

The reactor core is designed such that no inherent tendency exists for: sudden divergent oscillation of operating characteristics; divergent power transients in any mode of plant operation; uncontrollable oscillations (Criteria 6, 7). The basis of the reactor core design, in combination with the plant equipment characteristics, nuclear instrumentation system, and the Reactor Protection System, is to provide margins to ensure that fuel damage will not occur during normal operation or operational transients caused by single operator error or equipment malfunction (Criteria 6,7). The reactor is designed so that the overall power coefficient in the power operating range is not positive (Criterion 8).

The Reactor Coolant System (RCS) is designed to carry its dead weight and specified live loads separately or concurrently; e.g., pressure, temperature, and vibration stresses, with the concurrent seismic loads prescribed for the station. Provisions are made to control or shutdown the RCS in the event of malfunction of operating equipment or leakage of coolant from the system. The reactor vessel and support structures are designed to withstand the forces that would be created by any postulated Loss of Coolant Accident (LOCA) inside the drywell concurrent with the plant design earthquake loads (Criterion 9).

F.2.3 Group III-Nuclear and Radiation Controls

If adopted, the proposed criteria in Group III would have identified and defined the station instrumentation and control systems necessary to maintain the station in a safe operational status and also to provide adequate radiation shielding, radiation monitoring, fission process controls, and the effective sensing of abnormal conditions for initiation of engineered safety features. See Criteria 11 through 18 on Table F.2-3.

The station is provided with a centralized control room having adequate shielding, fire protection, air conditioning, and facilities to permit access and continuous occupancy during all design basis accident situations. The station design, therefore, does not contemplate the necessity for evacuation of the control room. If it were necessary to evacuate the control room, steps can be taken to

bring the plant to a safe cold shutdown from outside the control room (Criterion 11).

The necessary station controls, instrumentation, and alarms for safe and orderly operation are located in the control room. These include such controls and instrumentation as the control rod position indication, Residual Heat Removal System, and the Reactor Core Isolation Coolant System (RCICS) (Criteria 11, 12, 13, 16).

The performance of the reactor core and the indication of reactor power level are continuously monitored by the Nuclear Instrumentation System (Criterion 13). The Reactor Protection System, independent from the plant process control systems, overrides all other controls to initiate any required safety action. The RPS automatically initiates appropriate action whenever the plant conditions approach preestablished operational limits. The system acts specifically to initiate the Core Standby Cooling Systems (CSCS) (Criteria 12, 13, 14, 15). The station radiation and process monitoring systems are provided for monitoring significant parameters from specific plant process systems and specific areas including the plant effluents to the site environs and to provide alarms and signals for appropriate corrective actions. Monitoring and alarm instrumentation are provided for fuel storage and handling areas. Area monitors are provided in the Radwaste Process and Radwaste Truck Block areas. (Criteria 17, 18).

F.2.4 Group IV-Reliability and Testablility of Protection System

If adopted, the proposed criteria in Group IV would have identified and established requirements with regard to the functional reliability, inservice testability, redundancy, physical and electrical independence and separation, and fail-safe design of the reactor protection and control instrumentation systems. See Criteria 19 through 26 on Table F.2-4.

The RPS acts to shutdown the reactor, close primary containment isolation valves and initiate the operation of the core standby cooling systems. The RPS automatically overrides the plant normal operational control systems (i.e., functions independently) to initiate appropriate protective action whenever the plant conditions monitored by the system (e.g., neutron flux, containment pressure, reactor vessel pressure) exceed preestablished limits (Criterion 22). By means of a dual channel protection system with complete redundancy in each channel, no loss of the protection systems can occur by either component failure or removal from service. The RPS is designed so that a plant transient or accident is sensed by different parametric measurements (e.g., LOCA is detected by high drywell pressure and reactor low water level monitors). At least two instrument channels are provided to initiate each protection function (Criterion 20). Components of the redundant subsystems can be removed from service for testing and maintenance without negating the ability of the protection system to perform its protection functions (even when subjected to a multiple failure as a result of a single event) upon receipt of the appropriate signals (Criteria 19, 20, 21).

The design of the RPS provides means for testing and facilitates maintenance and trouble shooting while the reactor is at power operation without impeding the plant's operation or impairing its safety function (Criterion 25). The system electric power requirements are supplied from independent, redundant sources. Alternate sources of power are provided so as to permit the required functioning of the RPS in the event of loss of all offsite power (Criterion 24). The system circuits are separated to preclude a circuit fault from inducing a fault in another circuit and to reduce the likelihood that adverse conditions will encompass more than one circuit. The system sensors are electrically and physically separated, with special attention given to assure that the sensors in any one trip channel are not placed in the same local area or connected to the same power source or process measurement line. The system internal wiring or external cable routing arrangements are arranged to reduce any external influence on the systems performance (Criteria 23, 24). Systems essential to the protection function are designed to fail-safe in their likely failure modes. A failure of any one RPS input or subsystem component will produce a trip in 1 of the 2 channels; this condition is insufficient to produce a reactor scram, but the system is ready to perform its protective function upon another trip (either by failure or by exceeding the preset trip) in the other channel (Criterion 26).

F.2.5 Group V-Reactivity Control

If adopted, the proposed criteria in Group V would have established reactor core reactivity insertion and withdrawal rate limitations and the means to control the plant operations within these limits. See Criteria 27 through 32 on Table F.2-5.

The plant design contains two, independent, different principle reactivity control systems. Control of reactivity is operationally provided by a combination of movable control rods, and reactor coolant recirculation system flow. These systems accommodate fuel burnup, load changes, and long term reactivity changes. Reactor shutdown by the Control Rod Drive System is sufficiently rapid to prevent violation of fuel damage limits for normal operation and all abnormal operating transients. A Standby Liquid Control System is provided as an independent backup shutdown system to cover situations which limit the use of the operational reactivity control system. This system is designed to shut down the reactor and maintain the shutdown condition during reactor cooldown (Criteria 27, 28, 29, 31, 32).

The reactor core is designed to have (a) a reactivity response which regulates or damps changes in power level and spatial distributions of power production to a level consistent with safe and efficient operation, (b) a negative reactivity feedback consistent with the requirements of overall plant nuclear-hydrodynamic stability, and (c) have a strong negative reactivity feedback under severe power transient conditions (Criteria 27, 31). The reactivity control system is designed such that under conditions of normal operation, sufficient reactivity compensation is always available to make the

reactor adequately subcritical from its most reactive condition, and means are provided for continuous regulation of the reactor core excess reactivity and reactivity distribution. Shutdown margins greater than the maximum worth of the most effective control rod when fully withdrawn are provided (Criteria 29, 30). This system is also designed to be capable of compensating for positive and negative reactivity changes resulting from changing nuclear coefficients, fuel depletion, and fission product transients and buildup (Criterion 29). The system design is such that control rod worths, and the rate at which reactivity can be added, are limited to assure that the design basis reactivity accident is not capable of damaging the Reactor Coolant System or disrupting the reactor core, its support structures, or other vessel internals sufficiently to impair the Core Standby Cooling System effectiveness if needed. Acceptable fuel damage limits are not exceeded for any reactivity transient resulting from a single equipment malfunction or operator error.

F.2.6 Group VI-Reactor Coolant Pressure Boundary

If adopted, the Proposed criteria in Group VI would have established the reactor coolant pressure boundary design requirements and identified the means used to satisfy these design requirements. The "reactor coolant pressure boundary" is referred to in the Final Safety Analysis Report, Section 1.2, as the "nuclear system primary barrier." See Criteria 33 through 36 on Table F.2-6.

The inherent safety features of the reactor core design in combination with certain engineered safety features (control rod velocity limiter and control rod housing) and the station reactivity control system are such that the consequences of the most severe potential nuclear excursion accident, caused by a single component failure within the reactivity control system (rod drop accident), cannot result in damage (either by motion or rupture) to the reactor coolant pressure boundary (Criterion 33). The ASME and USASI Codes are used as the established and acceptable criteria for design, fabrication, and operation of components of the reactor coolant pressure boundary. The reactor coolant pressure boundary is designed and fabricated to meet the following as a minimum (Criterion 34):

1) Reactor Vessel-ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels, Subsection A.

2) Pumps-ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels, Subsection C

3) Piping and Valves-ANSI B.31.1.0, Code for Pressure Power Piping

ASME B&PV Code Section III Subsection NB 1980 Edition through Summer 1981 Addenda for fabrication and Winter 1981 addenda for design for piping replaced during 1984 replacement program. ASME B&PV Code Section XI 1980 Edition through Winter 1980 Addenda is also applicable to the 1984 replacement piping program.

The brittle fracture failure mode of the reactor coolant pressure boundary system components is prevented by control of the notch toughness properties of the ferritic steel components. This control is exercised in the selection of materials and fabrication of equipment and components. In the design, appropriate consideration is given to the different notch toughness requirements of each of the various ferritic steel product forms, including weld and heat-affected zones. In this way, assurance is provided that brittle fracture is prevented under all potential service loading temperatures. A temperature-based rule was used, with modifications drawn from fracture mechanics technology, to establish the requirements for brittle fracture prevention. The approach, which is generally accepted by materials specialists, establishes the requirements for brittle fracture prevention. These requirements are less stringent, when measured in terms of NDT requirement, for thin section materials than thick sections (Criterion 35).

The reactor coolant pressure boundary will be given a hydrostatic test at 1,560 psig in accordance with code requirements prior to initial reactor startup. Subsequent to the recirculation pipe replacement program in 1984, the Reactor Recirculation System was given a pressure test at 110 percent of reactor vessel design pressure, in accordance with ASME Section XI 1980 Edition, including Winter 1980 Addenda. A hydrostatic test, in accordance with ASME Section XI, is made every 10 yr and a leak test is made on the reactor coolant pressure boundary following each removal and replacement of the reactor vessel head. The system will be checked for leaks, and abnormal conditions will be corrected before reactor startup. The minimum vessel temperature during hydrostatic test shall at least be 60°F above the calculated NDT temperature prior to pressurizing the vessel. Extensive quality control assurance programs will also be followed during the entire fabrication of the reactor coolant pressure boundary (Criterion 36). Vessel material surveillance samples will be located within the reactor primary vessel to enable periodic monitoring of material properties with exposure. The program will include specimens of the base metal, heat affected zone metal, and standards specimens. Leakage from the reactor coolant pressure boundary is monitored during reactor operation (Criterion 36).

F.2.7 Group VII-Engineered Safety Features

If adopted, the Proposed criteria in Group VII would have established requirements with respect to: (1) incorporation of engineered safety features; (2) independence, redundance, capability, testability, inspectability, and reliability of engineered safety features; (3) suitability of each engineered safety feature for its intended duty; and (4) justification that each engineered safety feature's capability envelops all the anticipated and credible phenomena associated with the plant operational transients or design basis accidents considered. The engineered safety features are referred to in the Final Safety Analysis Report, Section 1.2, as "engineered safeguards" and "nuclear safety systems." See Definitions in

Introduction and Summary, Section 1.2. See Criteria 37 through 65 on Table F.2-7.

The normal plant control systems maintain plant variables within narrow operating limits. These systems are thoroughly engineered and backed up by a significant amount of experience in system design and operation. Even if an improbable maloperation or equipment failure were to occur, including a circumferential rupture of any pipe in the reactor coolant boundary with unobstructed discharge from both ends, an extensive system of engineered safety features limits the transient and the effects to below the guideline values set forth in

10CFR100 (Criterion 37). These engineered safety features include those which offer protection against a reactivity excursion, (control rod velocity limiters, and control rod housing supports), those which act to reduce the consequences of postulated Design Basis Accidents (DBA's) (main steam line flow restrictors, and containment), and those which provide core cooling in the event of a loss of normal cooling (Core Standby Cooling Systems-Criterion 37). Sufficient offsite and standby (redundant, independent, and testable) auxiliary sources of electrical power are provided to attain prompt shutdown and continued maintenance of the plant in a safe condition. The capacity of the offsite and onsite power sources are each independently adequate to accomplish all required engineered safety features functions assuming a failure of a single active component in each power system (Criterion 39).

The engineered safety features are designed to provide high reliability and ready testability. Specific provisions are made in each engineered safety feature to demonstrate operability and performance capabilities (Criterion 38). Components of the engineered safety features which are required to function after DBA's are designed to withstand the most severe forces and credible environmental effects, including missiles from plant equipment failures anticipated from the events without impairment of their performance capability (Criteria 40, 42, 43). The CSCS are designed to provide at least two different subsystems of different principles to prevent clad melt over the entire spectrum of postulated coolant breaks. Such capability is available notwithstanding the loss of all offsite power. The CSCS subsystems are designed to various levels of component redundancy such that no single active component failure in addition to the accident can prevent core cooling (Criteria 41, 44). To assure that the CSCS will function properly, specific provisions have been made to provide capability for testing the sequential operability and functional performance of each individual system (Criteria 46, 47, 48). Design provisions have also been made to facilitate physical and visual inspection of the CSCS components (Criterion 45).

The primary containment structure, including access openings and penetrations, is designed to withstand the peak pressure and temperatures which could occur due to the postulated design basis LOCA. The containment design includes allowance for energy addition from metal water or other chemical reactions beyond conditions which would occur with normal operation of CSCS's (Criterion 49). Plates, structural member, forgings, and pipe associated with the drywell have an initial NDT temperature of approximately 0°F when tested in accordance with the appropriate code for the materials. It is intended that the drywell will not be pressurized or subjected to substantial stress at temperatures below 30°F. The integrity of the complete plant containment is designed and maintained so that offsite doses resulting from postulated DBA's will be below the guideline values stated in 10CFR100 (Criteria 50, 51).

Provisions are made for the removal of heat from within the plant containment and to isolate the various process system lines as may be

necessary to maintain the integrity of the plant containment systems as long as necessary following the various postulated design basis accidents. All pipes or ducts which penetrate the primary containment and which connect to the Reactor Coolant System, or to the primary containment free space, are provided with at least two isolation valves in series (Criterion 53). The plant design includes preoperational pressure and leak rate testing of the primary containment, and a capability for leak testing at design pressure after the plant has commenced operation (Criteria 54, 55). Provisions are also made for demonstrating the functional performance of the plant containment isolation valves and leak testing of penetrations which have resilient seals or expansion bellows (Criteria 56, 57). The Suppression Pool Cooling System and the Containment Spray Cooling System provide two different means for containment heat removal under accident conditions so that the peak containment pressure would be substantially less than the primary containment maximum allowable pressure (Criterion 52). Demonstration of operability and ability to test the functional performance and inspect the active components in the Containment Spray Cooling System is provided (Criteria 58, 59, 60, 61). The Secondary Containment System is designed to permit periodic testing of the system performance using tracer injection and sampling (Criterion 64). This system can be physically inspected and its operability demonstrated (Criteria 62, 63, 65).

F.2.8 Group VIII-Fuel and Waste Storage Systems

If adopted, the proposed criteria in Group VIII would have established requirements applicable to fuel and waste storage systems. See Criteria 66 through 69 on Table F.2-8.

Appropriate plant fuel handling and storage facilities are provided to preclude accidental criticality and to provide sufficient cooling for spent fuel (Criteria 66, 67). The new fuel storage vault racks (located in the Reactor Building) are top entry, and are designed to prevent an accidental critical array even if the vault were flooded. Vault drainage is provided to prevent possible water collection (Criterion 66). The handling and storage of spent fuel takes place entirely within the Reactor Building (which provides containment) (Criterion 69). The spent fuel storage pool has provisions to maintain water clarity, control temperature, and instumentation to monitor water level. Water depth in the pool provides sufficient shielding for normal reactor building occupancy (10CFR20) by operating personnel. The racks in which spent fuel assemblies are placed are designed and arranged to ensure subcritically in the storage pool (Criteria 66, 67, 68, 69). The Spent Fuel Pool Cooling and Demineralizer System is designed to maintain the pool water temperature (decay heat removal) to control water clarity (safe fuel movement), and to reduce water radioactivity (shielding and effluent release control-Criteria 66, 67, 68). Accessible portions of the Reactor and Radwaste Buildings have sufficient shielding to maintain dose rates within 10CFR20 (Criterion 68). The Radwaste Building is designed to preclude accidental release of radioactive materials to environs (Criterion 69).

F.2.9 Group IX-Plant Effluents

If adopted, the proposed criterion in Group IX would have established requirements to limit releases of radioactive materials. See Criteria 70 on Table F.2-9.

The station Radioactive Waste Control Systems (which include the liquid, gaseous, and solid radwaste subsystems) are designed to limit the offsite radiation exposure to levels below those of 10CFR20 and Appendix I to 10 CFR50. The plant engineered safety systems (including the containment barriers) are designed to limit the offsite dose under various postulated DBA's to levels below 10CFR100 guideline values. The Air Ejector Offgas System is designed with sufficient holdup retention capacity so that during normal plant operation the controlled release of radioactive materials does not exceed the established release limits at the plant stack (Criterion 70).

TABLE F.2-1

AEC GENERAL DESIGN CRITERIA - GROUP I
(OVERALL PLANT REQUIREMENTS)

Criterion	Conformance (References to Sections of FSAR)
1. Quality Standards	1.5, 1.10, 3.2, 3.3, 3.4, 3.5 3.7, 3.8, 4.1, 4.2, 4.4, 4.5 4.6, 4.7, 4.8, 5.0, 6.0, 7.2, 7.3, 8.0, Appendix D
2. Performance Standards	1.5, Appendix C
3. Fire Protection	5.0, 10.8, 12.0
4. Sharing of Systems	Not applicable to Single Unit Reactor Facility
5. Records Requirements	13.7, Appendix D

TABLE F.2-2

AEC GENERAL DESIGN CRITERIA - GROUP II
(PROTECTION BY MULTIPLE FISSION PRODUCT BARRIERS)

Criterion	Conformance (References to Sections of FSAR)
6. Reactor Core Design	1.5, 1.7, 3.2, 3.6, 3.7, 4.3, 4.7, 4.8, 7.2, 14.2, 14.5, 14.6, 14.7
7. Suppression of Power Oscillations	1.5, 3.4, 3.6, 3.7, 4.4, 7.2, 7.5 7.7, 7.17, 14.6
8. Overall Power Coefficient	1.5, 1.7, 3.6, 3.7, 7.17
9. Reactor Coolant Pressure Boundary	1.5, 4.2, 4.3, 4.4, 4.10, 4.11, 7.8, 14.6, 14.7, Appendix A, Appendix C
10. Containment	5.2, 5.3, 14.5, 14.7

TABLE F.2-3

AEC GENERAL DESIGN CRITERIA - GROUP III
(NUCLEAR AND RADIATION CONTROLS)

Criterion	Conformance (References to Sections of FSAR)
11. Control Room	1.5, 7.2-7.5, 7.7-7.10, 7.12, 12.0
12. Instrumentation and Control System	1.5, 3.4, 3.8, 4.10, 7.2, 7.3, 7.4, 7.5, 7.7, 7.8, 7.9, 7.10, 7.12, 7.13, 7.14, 7.16, 9.2, 9.3, 9.4
13. Fission Process Monitors and Controls	1.5, 3.4, 3.8, 7.2, 7.5, 7.7, 7.8, 7.9, 7.16
14. Core Protection Systems	1.5, 3.4, 3.5, 4.4, 4.5, 4.6, 4.7, 4.8, 6.1-6.7, 7.2-7.5, 7.7, 7.12, 14.1-14.7
15. Engineered Safety Features Protection Systems	1.5, 7.2-7.5, 7.12
16. Monitoring Reactor Coolant Pressure Boundary	1.5, 4.10, 5.2, 7.8, 10.1
17. Monitoring Radioactive Releases	1.5, 7.12, 7.13, 7.14, 9.2, 9.4
18. Monitoring Fuel and Waste Storage	7.6, 7.12, 7.13, 9.2, 9.4, 10.5

TABLE F.2-4

AEC GENERAL DESIGN CRITERIA - GROUP IV
(RELIABILITY AND TESTABILITY OF PROTECTION SYSTEMS)

Criterion	Conformance (References to Sections of FSAR)
19. Protection Systems Reliability	1.5, 3.4, 7.2-7.5, 7.12, 14
20. Protection Systems Redundancy and Independence	1.5, 3.4, 7.2-7.5, 7.12, 14
21. Single Failure Definition	1.2, 14.5
22. Separation of Protection and Control Instrumentation Systems	1.5, 3.4, 7.2-7.5, 7.12
23. Protection Against Multiple Disability for Protection Systems	1.5, 3.4, 7.2-7.5, 7.12, 14
24. Emergency Power for Protection Systems	1.5, 3.4, 6.4, 7.2-7.5, 7.12, 8.4, 8.5, 14
25. Demonstration of Functional Operability of Protection Systems	1.5, 3.4, 4.6, 4.8, 5.2, 5.3, 6.7, 7.2-7.5, 7.12
26. Protection Systems Fail-Safe Design	1.5, 6.1-6.5, 7.2-7.5, 8.4, 8.5

TABLE F.2-5

AEC GENERAL DESIGN CRITERIA - GROUP V (REACTIVITY CONTROL)

Criterion	Conformance (References to Sections of FSAR)
27. Redundancy of Reactivity Control	1.5, 3.4, 3.8, 7.7
28. Reactivity Hot Shutdown Capability	1.5, 3.4, 3.8, 7.7, 14
29. Reactivity Shutdown Capability	1.5, 3.4, 3.6, 7.2, 14
30. Reactivity Holddown Capability	1.5, 3.4, 3.6, 3.8
31. Reactivity Control Systems Malfunction	1.5, 3.4, 3.6, 3.7, 7.2, 7.7, 14
32. Maximum Reactivity Worth of Control Rods	1.5, 3.4, 3.6, 7.7, 14

TABLE F.2-6

AEC GENERAL DESIGN CRITERIA - GROUP VI
(REACTOR COOLANT PRESSURE BOUNDARY)

Criterion	Conformance (References to Sections of FSAR)
33. Reactor Coolant Pressure Boundary Capability	1.5, 3.3-3.6, 4.2, 4.4, 4.5, 4.6, 14.5-14.7, Appendix A, Appendix C
34. Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention	3.3, 4.2, 4.3, 7.8, Appendix A, Appendix C, Appendix D
35. Reactor Coolant Pressure Boundary Brittle Fracture Prevention	4.2, Appendix A
36. Reactor Coolant Pressure Boundary Surveillance	4.2, 4.3, 4.10, Appendix A

TABLE F.2-7

AEC GENERAL DESIGN CRITERIA - GROUP VII
(ENGINEERED SAFETY FEATURES)

Criterion	Conformance (References to Sections of FSAR)
37. Engineered Safety Features Basis for Design	1.5,3.3,3.4,4.2,4.4,4.6,5.2,5.3, 6.1-6.7,7.2,7.3,7.4,8.4,8.5,8.6, 14.1-14.7
38. Reliability and Testability of Engineered Safety Features	1.5,3.4,3.5,4.6,5.2,5.3,6.6,7.2, 7.3,7.4,7.5,7.12,8.4,8.5,8.6
39. Emergency Power For Engineered Safety Features	7.2,7.3,7.4,8.5,8.6
40. Missile Protection	5.2,12.2,Appendix C
41. Engineered Safety Features Performance Capability	6.1-6.5,7.4,14.1-14.7
42. Engineered Safety Features Components Capability	3.4,5.2,5.3,6.1-6.5,7.2,7.3 7.4,8.4,8.5,8.6,14.7
43. Accident Aggravation Protection	3.4,5.2,5.3,6.1-6.5,7.3,7.4, 8.4,8.5,8.6
44. Emergency Core Cooling Systems Capability	6.1-6.5,7.4,14.7
45. Inspection of Emergency Core Cooling Systems	3.3,4.2,6.6
46. Testing of Emergency Core Cooling Systems Components	1.5,6.6,7.4
47. Testing of Emergency Core Cooling Systems	7.4,6.6
48. Testing of Operational Sequence of Emergency Core Cooling Systems	1.5,6.4,6.6,7.4,8.5,8.6, 10.8
49. Containment Design Basis	1.5,4.6,5.2,5.3,6.1,6.2,6.5 7.3,7.4,14.2-14.7,Appendix A, Appendix C
50. NDT Requirement for Containment Material	5.2

TABLE F.2-7 (Cont)

Criterion	Conformance (References to Sections of FSAR)
51. Reactor Coolant Pressure Boundary Outside Containment	1.5,2.2,2.3,5.2,4.6,7.3 14.7
52. Containment Heat Removal Systems	1.5,4.8,5.2,6.1-6.5,7.4, 10.8,14.7
53. Containment Isolation Valves	1.5,4.6,5.2,7.3
54. Containment Leakage Rate Testing	5.2
55. Containment Periodic Leakage Rate Testing	5.2
56. Provisions for Testing of Penetrations	5.2,5.3
57. Provisions for Testing of Isolation Valves	4.6,5.2,7.3,7.12
58. Inspection of Containment Pressure-Reducing Systems	4.8,5.2,5.3,6.4,6.6,10.7 10.8,12.0
59. Testing of Containment Pressure Reducing Systems Components	4.8,5.2,5.3,6.4,6.6,7.3, 7.4,10.7,10.8
60. Testing of Containment Spray Systems	6.4,6.6,7.4
61. Testing of Operational Sequence of Containment Pressure-Reducing Systems	5.2,5.3,6.4,6.6,7.4,8.4, 8.5,8.7
62. Inspection of Air Cleanup Systems	5.2,5.3,10.10
63. Testing of Air Cleanup Systems Components	5.2,5.3,10.10
64. Testing of Air Cleanup Systems	5.2,5.3,10.10
65. Testing of Operational Sequence of Air Cleanup Systems	5.3,7.12,13.4

TABLE F.2-8

AEC GENERAL DESIGN CRITERIA-GROUP VIII
(FUEL AND WASTE STORAGE SYSTEMS)

	Criterion	Conformance (References to Sections of FSAR)
66	Prevention of Fuel Storage Criticality	7.6, 10.2, 10.3
67	Fuel and Waste Storage Delay Heat	10.5
68	Fuel and Waste Storage Radiation Shielding	9.1, 9.2, 9.3, 9.4, 10.3, 10.5, 12.0
69	Protection Against Radioactivity Release From Spent Fuel and Waste Storage	5.1-5.3, 9.2, 9.3, 9.4, 10.2, 10.3, 10.5, 12.0

TABLE F.2-9

AEC GENERAL DESIGN CRITERIA-GROUP IX
(PLANT EFFLUENTS)

Criterion	Conformance (References to Sections of FSAR)
1. 70 Control of Releases of Radioactivity to the Environment	1.5, 5.2, 5.3, 7.12, 7.13, 9.2, 9.4, 14.2-14.7, Appendix E

APPENDIX G

STATION NUCLEAR SAFETY OPERATION ANALYSIS SUPPORTING NUCLEAR SAFETY REQUIREMENTS FOR PLANT OPERATION

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APPENDIX G

STATION NUCLEAR SAFETY OPERATIONAL ANALYSIS SUPPORTING NUCLEAR SAFETY REQUIREMENTS FOR PLANT OPERATION

G.1 ANALYTICAL OBJECTIVE

The objective of the station nuclear safety operational analysis is to systematically identify the requirements for and limitations on station operations necessary to satisfy nuclear safety operational criteria. This was used in the development of the Station Technical Specifications.

Key terms used in this Appendix are defined in Section 1, Introduction and Summary.

G.2 BASES FOR SELECTING OPERATIONAL REQUIREMENTS FOR PLANT OPERATION

An operational requirement is a requirement on either the value of a process variable or the operability of a station system. Such requirements must be observed during all modes of station operation (not just at power) to ensure that the station is operated safely. There are two kinds of operational requirements for station hardware:

1. Limiting condition for operation: the required condition for a system while the reactor is operating in a specified state

2. Surveillance requirements: the nature and frequency of tests required to ensure that the system is capable of performing its essential functions

Operational requirements are systematically selected for one of two basic reasons:

1. A requirement is considered essential if it is necessary to ensure that some specified condition (unacceptable result) is avoided during or following some specified station event

2. A requirement is considered essential if it is necessary to avoid some specified condition (unacceptable result) in spite of a single failure during or following some specified station event

G.2.1 Unacceptable Safety Results

Table G.2-1 lists the unacceptable safety results used as the major reasons for selecting system operational requirements. These unacceptable safety results are associated with different event categories. Those unacceptable safety results that are superior in importance to the others are marked with an asterisk.

G.2.2 Nuclear Safety Operational Criteria

The following nuclear safety operational criteria are used to select operational requirements:

Applicability	Nuclear Safety Operational Criteria
General	The station shall be operated so as to avoid unacceptable safety results.

Applicability	Nuclear Safety Operational Criteria
Abnormal operational transients and accidents	The station shall be operated in such a way that no single active component failure can prevent the safety actions essential to avoiding the unacceptable safety results associated with abnormal operational transients and accidents. This requirement is not applicable during system repair if the availability of the safety action is maintained either by restricting the allowable repair time or by more frequently testing a redundant system.

The unacceptable safety results have been associated with the different categories of station operation and events to facilitate the systematic selection of operational requirements. All the criteria must be satisfied at all times.

G.2.3 Origin of Unacceptable Safety Results and Criteria

Most of the unacceptable safety results and nuclear safety operational criteria represent an extension of the general intent of station hardware design criteria to station operations. Thus, where design criteria require that hardware design offer a specified degree of protection for a radioactive material barrier under certain circumstances, so operational criteria require that actual station operation offer the same degree of protection under the same circumstances.

Unacceptable safety result 1-4 differs in origin from the other criteria. See Table G.2-1. This criterion requires, in effect, that the station be operated only under conditions for which safety analysis has been performed. In a case where a system has not been shown to be nonessential to some safety action, the system would be considered essential under this criterion until proven otherwise. Thus, definitive safety analysis is a prerequisite to a finding that a system or system action is nonessential.

Piece parts of safety system components may be essential or nonessential depending on their contribution to safety actions. In order for a part to be designated as nonessential, definitive analyses must be performed to show that the part does not perform a safety function, the part is not required for the performance of a safety action, and the part's failure does not prevent the accomplishment of a safety action. The analyses can assume that all parts (including the part under consideration) are present, are installed correctly, and are designed correctly. Parts analyses shall use a logical, reasonable, qualitative, and conclusive assessment. Explicit calculations may be used (but are not necessarily required) to determine a part's contribution to a safety action. Credible failure

modes and effects analyses (FMEA) may be used to support determinations of whether a part is safety-related or not but FMEA shall not constitute the sole basis for concluding that a part is nonsafety-related.

TABLE G.2-1

SELECTING SYSTEM OPERATIONAL REQUIREMENTS

Station Event Category		Unacceptable Safety Results
1. Planned Operation	*1-1.	Release of radioactive material to the environs that exceeds the limit of 10CFR20
	1-2.	Fuel failure to such an extent that were the freed fission products released to the environs via the normal discharge paths for radioactive material, the limits of 10CFR20 would be exceeded
	1-3.	Nuclear system stress in excess of that allowed for planned operation by applicable industry codes
	1-4.	Existence of a station condition not considered by station safety analyses
2. Abnormal Operational Transients	*2-1.	Release of radioactive material to the environs that exceeds the limits of 10CFR20
	2-2.	Any fuel safety limit as a direct result of the transient analyses
	2-3.	Nuclear system stress exceeding that allowed for transients by applicable industry codes
3. Accidents	*3-1.	Radioactive material release exceeding the guideline values of 10CFR100
	3-2.	Excessive fuel cladding temperature
	3-3.	Nuclear system stresses exceeding that allowed for accidents by applicable industry codes
	3-4.	Containment stresses exceeding that allowed for accidents by applicable industry codes when containment is required

TABLE G.2-1 (cont)

	3-5. Overexposure to radiation of station personnel in the control room
4. Special Event - shutdown from outside control room	4-1. Inability to shut down reactor by manipulating local controls and equipment outside the control room
	4-2. Inability to bring the reactor to the cold shutdown condition from outside the control room
5. Special Event - shutdown without control rods	5-1. Inability to shut down the reactor independent of control rods
	5.2. Inability to automatically trip the recirculation pumps to reduce reactor power while shutting down the reactor independent of control rods.

NOTE:

*Unacceptable results superior in importance

G.3 BASES FOR SELECTING SURVEILLANCE TEST FREQUENCIES FOR NUCLEAR SAFETY SYSTEMS AND ENGINEERED SAFEGUARDS FOR PLANT OPERATION

G.3.1 Normal Surveillance Test Frequencies

After the essential Nuclear Safety Systems and engineered safeguards have been identified by applying the nuclear safety operational criteria, surveillance requirements are selected for these systems. In the course of selecting surveillance test frequencies, the various systems are considered in terms of relative availability, test capability, station conditions necessary for testing, and engineering experience with the system type. The surveillance test frequency selected represents the application of engineering judgment integrating all of these considerations. However, the selected frequencies are conservative with respect to the surveillance requirements needed to maintain the reliability of the system as provided by the basic system design. Surveillance requirements are contained in the Technical Specifications.

G.3.2 Allowable Repair Times

Originally, allowable repair times were selected by computing these times based upon the surveillance test frequencies using as a guideline availability analysis methods (Reference 1) for redundant standby systems. The resulting maximum average allowable repair times assure that a system's long term availability, including allowance for repair, is not reduced below the availability that would be achieved if repairs could be made in zero time.

The original times were selected using the analytical methods developed by I.M. Jacobs in 1969. Jacobs analytical methods used to assess risk in 1969 were crude compared to the sophisticated tools currently used, and required the use of many conservative assumptions. For example, generic failure rates and repair times had to be assumed because no plant specific failure data was available. Our current risk assessment tools take advantage of Pilgrim's twenty year operating history to calculate plant specific failure rates and repair times that accurately reflect our overall performance, including the impact of planned as well as corrective maintenance activities. The Jacobs generic "failure" rates did not include margin for voluntary unavailability events; hence, the restriction only to failure events.

However, the major obsolete feature of the Jacobs method is that it is not capable of evaluating the integrated and cumulative impact on plant risk from the collective performance of individual systems. Our initial allowed outage times were computed before the development of the WASH-1400 (1975) study and its introduction of a sophisticated methodology to assess overall plant risk. The Jacobs method used reliability and availability goals established at the system level, but the method could not translate these individual system goals into a meaningful measure of overall plant safety. The method employed today uses Individual Plant Evaluation (IPE) to evaluate the integrated impact of system reliability and availability on overall plant risk. We use the critical figure of merits of Core Damage Frequency (CDF) and Large Early Release Frequency (LERF) to ensure that the work performed on our safety systems maintain the high reliability of the systems used in support of plant safety.

Approved allowable repair times are contained in the Technical Specifications.

G.3.3 Repair Time Rule

Repair of a safety system may be carried out while the reactor is in operation for a time equal to the maximum allowable average repair time. If repair is not complete when the allowable repair time expires, the reactor station must be placed in its safest mode (with respect to the protection lost).

To maintain the validity of the assumptions used to establish the superseded repair time rule as developed in accordance with I.M. Jacobs, the following restrictions must be observed:

1. The allowable repair time should only be used as required to restore failed equipment to operation, not for routine maintenance. Using this time should be an event as rare as failure of the equipment itself. Routine maintenance should be scheduled when the equipment is not needed.

2. When a failure is discovered by test, all the redundant channels should be tested to establish that they are good at the beginning of the repair time for the failed channel and do not suffer from the same failure mode discovered in the failed channel. If there are multiple failures of the same mode, the repair time allowance does not apply.

3. At the conclusion of the repair, the repaired channel must be retested and placed in service. The redundant channels must also be retested, not only to validate the assumptions but to assure that the repair did not inadvertently invalidate a good channel.

4. Once the need for repair of a failed device is discovered, repair should proceed as quickly as possible consistent with good craftsmanship.

Alternately, if a system is expected to be out for repair for an extended length of time, the availability of the remaining systems may be maintained at the prefailure level by testing them more often. This technique is fully developed in APED-5736, Guidelines for Determining Safe Test Intervals and Repair Times for Engineered Safeguards.

Current

The Repair Time Rule was originally established to maintain the validity of the original allowed outage time calculations performed more than twenty six years ago. The purpose for the restriction on voluntary entry into LCOs is no longer needed nor valid in light of improved techniques and greater operational experience. The current program to control voluntary entry into LCOs provides a superior means to assure safety system reliability is maintained and the plant is operated safely and supersedes the original repair time rule.

The current basis for allowable repair times is defined in the Technical Specifications. Such repair times are based on the current analytical methods described above and not on I.M. Jacob's methodology.

G.3.4 References

Jacobs, I.M., Guidelines for Determining Safe Test Intervals and Repair Times for Engineered Safeguards, General Electric Company, Atomic Power Equipment Department, April 1969 (APED-5736). Allowable repair times for CSCS, diesel generators, and containment cooling systems are based on NUREG-0123, Standard Technical Specifications.

G.4 METHOD OF ANALYSIS

The nuclear safety operation analysis is performed after detailed design has been established for the station. The end products of the analysis are the operational nuclear safety requirements, restrictions, and limits on station hardware and its operation that must be observed to satisfy the nuclear safety operation criteria. The key steps in this analysis are as follows:

1. Identify and define the physical states (operating states) in which the BWR may exist

2. For each operating state, identify the types of operations (planned) and events (transients, accidents, and special events) that the station must accommodate within the nuclear safety operational criteria

3. For each operating state, identify the safety actions essential to accommodating each applicable type of operation and event within the nuclear safety operational criteria

4. For each operating state, identify the system or variables (to be limited) that are essential to achieving each required safety action. Systems that are needed for the achievement of a safety action with a specified degree of redundancy are considered essential to the safety action. Limitations on process variables should be associated with the applicable unacceptable result and with the related station system originating the need for the limit

5. For each system identified in step 4, identify the specific system functional requirements and restrictions that must be observed within each operating state. For each key process variable identified in step 4, establish the limits that must be observed in each operating state

6. Identify the minimum amount of system hardware that must be operable (or restricted from operation) to accomplish the functional requirements (and restrictions) identified in step 5

7. For each system, identify the conditions (operability, numbers of components, out of service times, inspection, and test frequencies) that must be met to accomplish, with an acceptable level of redundancy and availability, the system functional requirements (and restrictions) identified in step 5

The results of steps 1, 2, 3, and 4 are presented in this appendix. The results of steps 5 and 6 for each system and variable identified in step 4 are presented in the subsections of the safety analysis report that describe the system. The results of step 7 are contained in the Technical Specifications referenced in Appendix B.

Together the station design and the observation of the operational nuclear safety requirements derived in this analysis assure that the nuclear safety operational criteria are satisfied. When an operational nuclear safety requirement for a system is combined with the action to be taken if the requirement cannot be met, a technical specification is formed. Figure G.4-1 shows in block form the process by which technical specifications are derived.

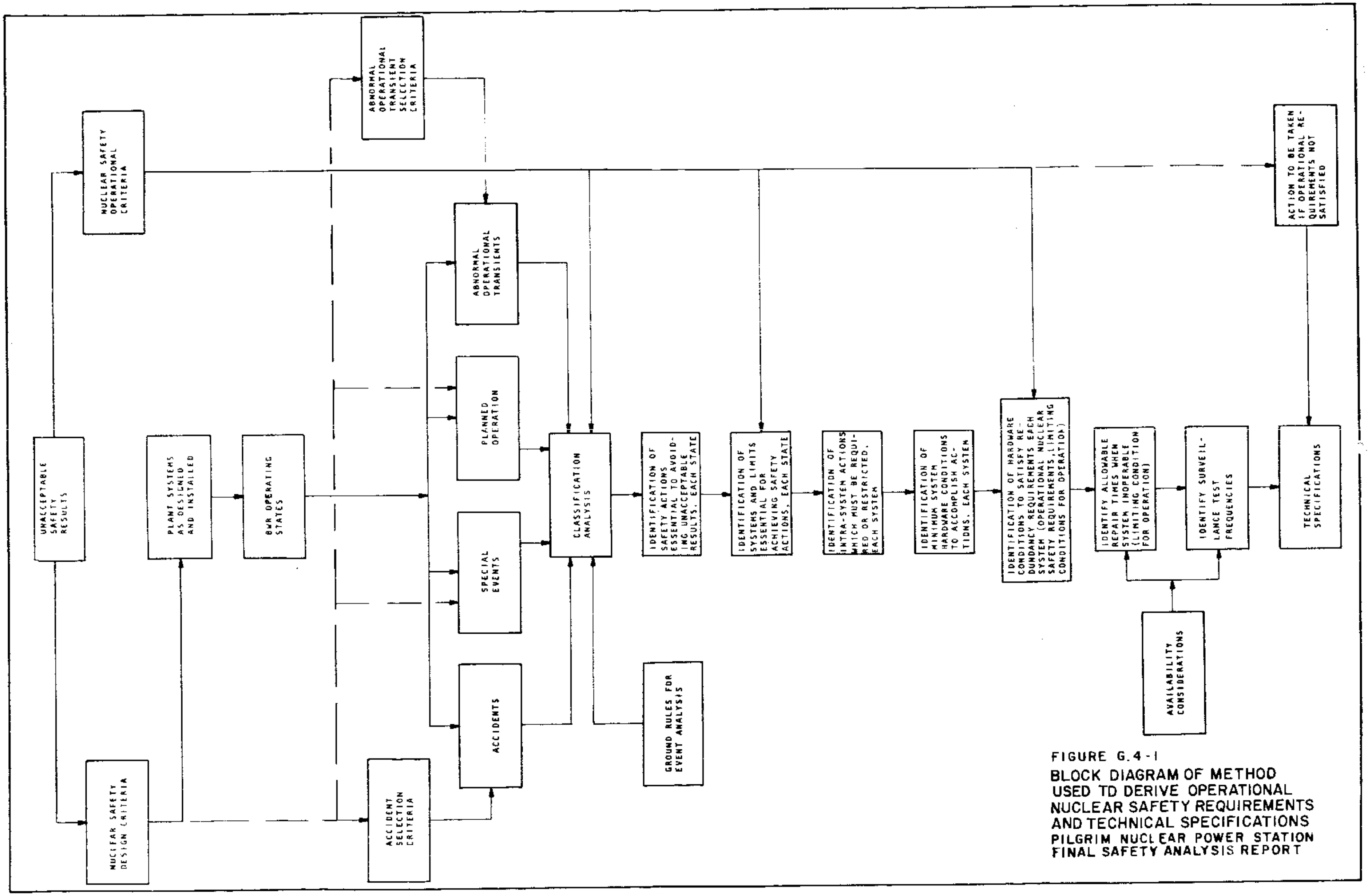


FIGURE G.4-1
BLOCK DIAGRAM OF METHOD USED TO DERIVE OPERATIONAL NUCLEAR SAFETY REQUIREMENTS AND TECHNICAL SPECIFICATIONS
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

G.5 ANALYSIS AND RESULTS

G.5.1 IDENTIFICATION OF BWR OPERATING STATES

Six BWR operating states are identified and defined on Matrix 1 of Table G.5-1. The main objective in selecting operating states is to divide the BWR operating spectrum into a few major conditions to facilitate consideration of various events in each state. The matrix includes all the conditions in which the reactor can exist.

Each operating state includes a wide spectrum of values for important station parameters. Within each state these parameters are considered over their entire range, to determine the limits on their values necessary to satisfy the operational nuclear safety criteria. Such limitations are presented in the sections of the Safety Analysis Report that describe the systems originating the requirement for the parameter limit. The station parameters to be considered in this manner include the following:

Reactor coolant temperature
Reactor vessel water level
Reactor vessel pressure
Nuclear system leakage
Reactor coolant forced circulation flow rate
Reactor power level (thermal and neutron flux)
Core neutron flux distribution
Fuel pool water level
Drywell pressure and temperature
Suppression pool level and temperature

G.5.2 TYPES OF OPERATION AND EVENTS APPLICABLE IN EACH OPERATING STATE

G.5.2.1 Identification Method

Matrix 2 on Table G.5-2 identifies the planned operations, abnormal operational transients, accidents, and special events to be considered in determining station operational nuclear safety requirements. Planned operations are to be considered without regard to the need for anticipating abnormal operational transients, accidents, or special events, because these events are considered separately. The abnormal operational transients and accidents listed on the matrix were selected and categorized by the same methods as those described in Section 14, Station Safety Analysis. In each case the listed events cause the most severe demand for protective action of any event of a similar nature.

G.5.2.1.1 Planned Operation Definition

Planned operation refers to normal station operation under planned conditions in the absence of significant abnormalities. Operations subsequent to an incident (transient, accident, or special event) are not considered planned operations until the actions taken or equipment used in the station are identical to those that would be

used had the incident not occurred. The established planned operations can be considered as a chronological sequence: refueling outage achieving criticality heatup power operation achieving shutdown cooldown refueling outage.

The planned operations are defined below.

1. Refueling Outage

   Includes all of the planned operations associated with a normal refueling outage:

   a. Planned, physical movement of core components (fuel, curtain, control rods, etc.)

   b. Refueling test operations

   c. Planned maintenance

2. Achieving Criticality

   Includes all the station actions normally accomplished in bringing the station from a condition in which all control rods are fully inserted to a condition in which nuclear criticality is achieved and maintained

3. Heatup

   Begins where achieving criticality ends and includes all station actions normally accomplished in approaching nuclear system rated temperature and pressure by using nuclear power (reactor critical). Heatup extends through warmup and synchronization of the turbine-generator

4. Power Operation

   Begins where heatup ends and includes continued station operation at power levels in excess of heatup power

5. Achieving Shutdown

   Begins where power operation ends and includes all station actions normally accomplished in achieving nuclear shutdown (more than one rod subcritical) following power operation

6. Cooldown

   Begins where achieving shutdown ends and includes all station actions normally accomplished in the continued removal of decay heat and the reduction of nuclear system temperature and pressure

The entries in Matrix 2 on Table G.5-2 indicating the applicability of the planned operations are based on the definitions of the planned operations.

G.5.2.1.2 Selection of Abnormal Operational Transients

To select abnormal operational transients, eight nuclear system parameter variations are considered as potential initiating causes of threats to the fuel and the nuclear system process barrier. The parameter variations are as follows:

1. Nuclear system pressure increase
2. Reactor vessel water (moderator) temperature decrease
3. Positive reactivity insertion
4. Reactor vessel coolant inventory decrease
5. Reactor core coolant flow decrease
6. Reactor core coolant flow increase
7. Core coolant temperature increase
8. Excess of coolant inventory

These parameter variations, if uncontrolled, could result in damage to the reactor fuel or nuclear system process barrier, or both. A nuclear system pressure increase threatens to rupture the nuclear system process barrier for internal pressure. A pressure increase also collapses the voids in the moderator, causing an insertion of positive reactivity that threatens fuel damage from overheating. A reactor vessel water (moderator) temperature decrease results in an insertion of positive reactivity as density increases. This could lead to fuel overheating. Positive reactivity insertions are possible from causes other than nuclear system pressure or moderator temperature changes. Such reactivity insertions threaten fuel damage caused by overheating. Both a reactor vessel coolant inventory decrease and a reduction in the flow of coolant through the core threaten to overheat the fuel as the coolant becomes unable to adequately remove the heat generated in the core. An increase in coolant flow through the core reduces the void content of the moderator, resulting in an increased fission rate.

The eight parameter variations listed above include all of the effects within the nuclear system caused by abnormal operational transients that threaten the integrities of the reactor fuel or nuclear system process barrier. The variation of any one parameter may cause a change in another listed parameter; however, for analysis purposes, threats to barrier integrity are evaluated by groups according to the parameter variation originating the threat. For example, positive reactivity insertions resulting from sudden pressure increases are evaluated in the group of transients stemming from nuclear system pressure increases.

Abnormal operational transients result from single equipment failures or single operator errors that can be reasonably expected during any modes of station operation. The following types of operational single failures and operational errors are identified:

1. Opening or closing any single valve (a check valve is not assumed to close against normal flow)

2. Starting or stopping any single component

3. Malfunction or maloperation of any single control device

4. Any single electrical failure

5. Any single operator error

Operator error is defined as an active deviation from written operating procedures or nuclear station standard operating practices. A single operator error is the set of actions that is a direct consequence of a single, reasonably expected erroneous decision. The set of actions is limited as follows:

1. Those actions that could be performed by only one person

2. Those actions that would have constituted a correct procedure had the initial decision been correct

3. Those actions that are subsequent to the initial operator error and have an effect on the designed operation of the station but are not necessarily directly related to the operator error

Examples of single operator errors are as follows:

1. An increase in power above the established flow control power limits by control rod withdrawal in the specified sequences

2. The selection and complete withdrawal of a single control rod out of sequence

3. An incorrect calibration of an average power range monitor

4. Manual isolation of the main steam lines caused by operator misinterpretation of an alarm or indication

The five types of single errors or single malfunctions are applied to the various station systems, with a consideration for a variety of station conditions to discover events that directly result in any of the listed undesired parameter variations. Once discovered, each event is evaluated for the threat it poses to the integrity of the radioactive material barriers.

G.5.2.1.3 Selection of Accidents

For analysis purposes, accidents are categorized as follows:

1. Accidents that result in radioactive material release from the fuel with the nuclear system process barrier, primary containment, and secondary containment initially intact

2. Accidents that result in radioactive material release directly to the primary containment

3. Accidents that result in radioactive material release directly to the secondary containment with the primary containment initially intact

4. Accidents that result in radioactive material release directly to the secondary containment with the primary containment not intact

5. Accidents that result in radioactive material release outside the secondary containment

Accidents are defined as hypothesized events that affect one or more of the radioactive material barriers and that are not expected during station operations. The accident types considered are as follows:

1. Mechanical failure of a single component leading to the release of radioactive material from one or more barriers. The components referred to here are not those that act as radioactive material barriers. Examples of mechanical failure are breakage of the coupling between a control rod drive and the control rod, and failure of a spring used to close an isolation valve

2. Arbitrary rupture of any single pipe up to and including complete severance of the largest pipe in the nuclear system process barrier. This kind of accident is considered only under conditions in which the nuclear system is significantly pressurized

The effects of the various accident types are investigated, with a consideration for the full spectrum of station conditions, to examine events that result in the release of radioactive material. The accidents resulting in potential radiation exposures greater than any other accident considered under the same general accident assumptions are designated design basis accidents.

G.5.2.1.4 Selection of Special Events

Two special events are evaluated to demonstrate station capability required by arbitrary nuclear safety criteria. The adequacy of the redundant reactivity control capabilities is demonstrated by evaluating the event "shutdown without control rods." The capability to perform a safe shutdown from outside the control room is demonstrated by evaluating the event "shutdown from outside control room."

G.5.2.2 Detailed Explanations of Matrix 2 Entries on Table G.5-2

The Matrix 2 entries on Table G.5-2 indicating the applicability of an event (transient, accident, or special event) to each state are based on whether the event can occur, starting from any of the initial conditions represented by the set of planned operations that are applicable in the corresponding BWR operating state.

Note that even though a given operation or event is not applicable while the reactor is in a certain operating state, operational restrictions on certain station systems may be necessary to ensure that the given operation or event remains inapplicable. The requirements for such restrictions are identified in later matrices.

The explanations for the entries in Matrix 2 on Table G.5-2 are given item-by-item in the following paragraphs.

G.5.2.2.1 Planned Operations

The Matrix 2 entries on Table G.5-2 for the planned operations all follow directly from the definitions of the planned operation and the definitions of the BWR operating states.

G.5.2.2.2 Abnormal Operational Transients

The abnormal operational transients, listed in Matrix 2 on Table G.5-2 as events 12 through 36, are the same ones selected by the methods described in Section 14, Station Safety Analysis. The following paragraphs explain why certain events are applicable in certain operation states but not in others.

Events 12 and 13 - Generator and Turbine Trips

A turbine or generator trip can occur in operating states D (during heatup) or F (during power operation).

Events 14 and 15 - Main Steam Line Isolation

Isolation of the main steam lines can result in a transient for which some degree of protection is required only in operating states C, D, E, and F. In operating states A and B, the main steam lines are continuously isolated.

Event 16 - Loss of Vacuum (turbine trip without bypass)

Because the main condenser is normally used for removing decay heat under any condition in which steam is being generated, this event applies in operating states C, D, E, and F. The more significant cases are in operating state F, when the condenser is used during power operations.

Event 17 - (not used)

Event 18 - Loss of Feedwater Heating

A loss of feedwater heating must be considered with regard to the nuclear safety operational criteria only in operating state F, because significant feedwater heating does not occur in any other operating state.

Event 19 - Shutdown Cooling (RHRS) Malfunction

A shutdown cooling malfunction causing a moderator temperature decrease must be considered in operating states A, B, C, and D. This event is not considered in operating states E and F because nuclear system pressure is too high to permit shutdown cooling (RHRS) operation.

Event 20 - Inadvertent Pump Start (temperature decrease)

The addition of cold water via an inadvertent start of a pumping system can occur under any operating condition.

Event 21 - Control Rod Withdrawal Error

The results of adding positive reactivity via a control rod withdrawal error must be considered in all operating states. This error can occur under any operating condition.

Events 22, 23, and 24 - Fuel Assembly Insertion and Removal of Control Rod or Control Curtain

An inadvertent positive reactivity insertion can result from erroneous control rod removal, control curtain removal, or fuel assembly insertion. Because these actions can occur only when the reactor vessel head is removed and manipulation of the refueling equipment over the reactor core is possible, these events must be considered only in operating state A.

Event 25 - Pressure Regulator Failure

A pressure regulator failure causing a coolant inventory decrease applies only in operating states C, D, E, and F, because the reactor is not pressurized in other states.

Event 26 - Inadvertent Opening of a Relief Valve

The inadvertent opening of a relief valve is possible in any operating state.

Event 27 - Loss of Feedwater Flow

Because continuous feedwater flow is neither required nor provided in operating states A and B, a loss of feedwater flow need only be considered in operating states C, D, E, and F.

Event 28 - Total Loss of Offsite AC Power

The effects of a loss of offsite ac power are considered in each operating state. This event also includes lesser electrical failures involving the loss of ac power busses.

Events 29, 30, 31 and 32 - Core Coolant Flow Decrease

Because forced coolant circulation would be present as a planned operation only in operating states C, D, E, and F, events causing loss of forced circulation flow need be considered only in these states.

Event 33 - Recirculation Flow Control Failure - Increasing Flow

Because a recirculation flow control failure causing an increased coolant flow through the core can occur only when a recirculation pump is initially operating during planned operation, this event is applicable only in operating states C, D, E, and F.

Event 34 - Startup of Idle Recirculation Pump

A startup of an idle recirculation pump can potentially occur in any operating state.

Event 35 - Loss of Shutdown Cooling

Malfunctions causing loss of RHR shutdown cooling are considered in operating states A, B, C, and D, because only in these states would the RHR Shutdown Cooling System be in use as part of one of the planned operations.

Event 36 - Feedwater Controller Failure - Maximum Demand

A feedwater controller failure causing an excess coolant inventory in the reactor vessel must be considered in operating states C, D, E, and F, because only in these states can the feedwater controller be operating as part of planned operations.

Event 37 - (Not Used)

G.5.2.2.3 Accidents

The accidents listed in Matrix 2 on Table G.5-2 as items 38 through 41 are the same ones selected by the methods described in Section 14, Station Safety Analysis. The following paragraphs explain why certain accidents are applicable in certain operating states but not in others.

Event 38 - Control Rod Drop Accident

A control rod drop accident is considered possible under any condition in which a control rod drive is withdrawn and no specific verification of the control rod to drive coupling can be made. The

accident is significant only if it occurs with the reactor not shut down (states D and F). The accident is not considered in states A or B, because control rod to drive coupling verification is required as a part of planned operation with the vessel head off. The rod to drive coupling integrity must be verified prior to withdrawing more than one control rod in states A or B.

Event 39 - Pipe Breaks Inside Primary Containment

A pipe break inside the primary containment is considered in all operating states except A and B, because the nuclear system is not significantly pressurized in the two latter states.

Event 40 - Fuel Handling Accident

Because a fuel handling accident can potentially occur any time when fuel assemblies are being manipulated, either over the reactor core or in the spent fuel pool, this accident is considered in all operating states.

Event 41 - Pipe Breaks Outside Primary Containment

A pipe break outside the primary containment is applicable in all operating states except A and B; the nuclear system is not significantly pressurized in the two latter states.

Event 42 and 43 - (Not Used)

G.5.2.2.4 Special Event - Shutdown from Outside the Control Room (Event 44)

Shutdown from outside the control room is a special event investigated to evaluate the capability of the station to be controlled from outside the control room. Special criteria, given in Section 14, Station Safety Analysis, apply to this event. The event is applicable to any operating state.

G.5.2.2.5 Special Event - Shutdown Without Control Rods (Event 45)

Reactor shutdown without control rods is a special event postulated to evaluate the capabilities of the Standby Liquid Control System. Because this event can occur only when the reactor is initially not shutdown, it is applicable only to operating states B, D, and F. In operating states D and F, the Standby Liquid Control System is supplemented by the Recirculation Pump Trip System which provides the capability to promptly reduce reactor power during certain transients independent of the Reactor Protection System and control rods.

G.5.3 IDENTIFICATION OF SAFETY ACTIONS AND SYSTEMS ESSENTIAL TO SATISFACTION OF NUCLEAR SAFETY OPERATIONAL CRITERIA

G.5.3.1 Introduction

To fully identify the requirements, restrictions, and limitations that must be observed during station operation, station systems must be related to the needs for their actions in satisfying the nuclear safety operational criteria. This Appendix displays these relationships in a series of block diagrams and matrices. Refer to Figures G.5-1 through G.5-44.

For each event a protection sequence diagram, shown on Figures G.5-22 through G.5-44, is presented showing the primary (front line) systems

essential to achieving each safety action. Those station systems which are essential to the operation of the primary systems are shown on the auxiliary block diagrams, Figures G.5-1 through G.5-21. The block diagrams show only that equipment necessary to provide the safety actions in such a way that the nuclear safety operational criteria are satisfied. The total station capability to provide a safety action is not shown, only the minimum capability essential to satisfying the operational criteria. Once all of the protection sequences are identified in block diagram form, the system requirements are superimposed on the operational matrices. Thus, the matrices display the most restrictive requirements from all of the essential protection sequences for any one event. Each matrix of the series considers the following:

1. The BWR operating state

2. Types of operations or events that are possible within the operating state

3. Relationships of certain safety actions to the unacceptable results and to specific types of operation and events

4. Relationships of the actions of certain systems to safety actions and to specific types of operation and events

5. Supporting or auxiliary systems essential to the operation of the front line safety systems

6. Considerations necessary to achieve a minimum level of functional redundancy (the single failure criterion applied functionally at the safety action level)

It is necessary to systematically consider each of the above six aspects of the BWR on a stationwide basis in order to rationally determine operational requirements. Matrices 3 on Table G.5-3 and the block diagrams for the events combine to provide a vehicle for such a systematic analysis. Through the use of Matrices 3 on Table G.5-3 and the block diagrams, any operational requirement can be traced to the unacceptable result, criterion, or safety action originating its need.

All of the entries in Matrices 3 on Table G.5-3 represent a finding of essentiality for the safety action, system, or limit under consideration. Essentiality in this context means that the safety action, system, or limit is essential to satisfying the nuclear safety operational criteria. Essentiality is found through an analysis in which the safety action, system, or limit being considered is completely disregarded in the analyses of the applicable operations or events. If the nuclear safety operational criteria are satisfied without the safety action, system, or limit, then the safety action, system, or limit is not essential, and no operational nuclear safety requirement would be indicated. When disregard of a safety action, system, or limit results in violation of one or more nuclear safety operational criteria, the safety

action, system, or limit is considered essential, and the resulting operational nuclear safety requirements can be related to specific criteria and unacceptable results.

There is a difference between classification analyses, which provide bases for findings of essentiality, and the analyses of Section 14, Station Safety Analysis. Although the events analyzed are the same, the analyses of Section 14 represent a real response of the station under certain limiting assumptions; whereas a classification analysis strips away all nonessential actions and systems in the effect to determine essentiality. A classification analysis represents essential station response. The analyses of Section 14 emphasize "worst cases" with regard to the fuel thermal hydraulic conditions, nuclear system pressure or radioactivity release. The classification analyses emphasize protection sequences.

G.5.3.2 Presentation of Information on Matrices 3, Table G.5-3

Figure G.5-45 shows the concept used for presenting information in Matrices 3 on Table G.5-3. The left side relates safety actions to the unacceptable results for each specific event. The right side of each matrix relates hardware (systems) requirements to safety actions and specific events. Each matrix applies only to one BWR operating state. A safety action that is essential to avoiding one or more unacceptable results for a given event is identified with the number of the appropriate unacceptable result in the matrix block corresponding to the safety action and the event. The example shows that for a turbine trip the scram safety action is essential to avoiding unacceptable results 2-2 and 2-3, and the pressure relief safety action is essential to avoiding unacceptable result 2-3. Section G.2.1, Unacceptable Results, lists the reasons why scram and pressure relief are needed.

A system that is essential to carrying out a safety action for a given event is identified by placing the column number of the safety action in the matrix block corresponding to the system and event. This number also can indicate that the system is an auxiliary (support system) to the system with that column number. Other symbols used in the system matrix blocks to indicate various requirements of the system are explained in the notes to Figure G.5-45.

G.5.3.3 Rules for Block Diagrams and Matrices

The block diagrams and the entries in Matrices 3 on Table G.5-3 represent the consistent application of the rules listed below.

1. Entries are made only when an action, limit, or system is essential to satisfying the nuclear safety operational criteria and avoiding the unacceptable safety results.

2. Entries are made only for all actions, limits, and systems essential for the event through the full range of initial conditions within an operating state. Consideration is not

limited to worst cases; lesser cases sometimes require actions or systems different from the worst case

3. For planned operations, entries are made only for actions, limits, and systems essential to avoiding the unacceptable results during operation in that state (as opposed to transients, accidents, and special events, which are followed through to completion). Planned operations are treated differently from other events because the transfer from one state to another during planned operations is deliberate. For events other than planned operations the transfer from one state to another may be unavoidable

4. Limits are indicated on the matrices only for those essential parameters that are continuously monitored by the operator. Parameter limits associated with the required performance of an essential system are considered to be included in the requirement for the operability of the system. Limits on continuously monitored parameters are called "envelope limits," and limits on periodically monitored parameters are called "operability limits." Only the envelope limits and the associated indicators for the envelope limits will be indicated on the matrix. Systems associated with the control of the envelope parameters are considered nonessential as long as it is possible to place the station in a safe condition without using the system in question

5. Entries for transients, accidents, and special events are made for the entire duration of the event and aftermath until planned operation is resumed. Planned operation is considered resumed when the procedures being followed are identical to those used during any one of the planned operations.

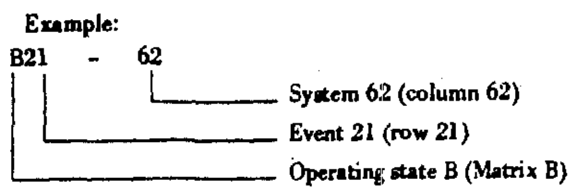
6. The initial conditions for transients, accidents, and special events are limited to the conditions that would exist during the planned operations applicable within the operating state.

7. Because transients, accidents, and special events are considered through the entire duration of the event until planned operation is resumed, manual operation of certain systems is sometimes required following the more rapid portions of the event. Credit for operator action is taken on a case basis, depending on the conditions that would exist at the time operator action would be required. Credit for operator action is taken only when the operator can reasonably be expected to accomplish the required action under the existing conditions. When credit for operator action is taken, a "P" is entered in the appropriate matrix block.

8. Matrix entries for transients, accidents, and special events are made only for those actions, limits, and systems for which there arises a unique requirement as a result of the event. For instance, if a system that was operating prior to the event (during planned operation) is to be employed in the same manner following the event, and if the event did not affect the operation of the system, then no matrix entries for the system would be made.

9. Where an operational nuclear safety requirement for a system is based on a certain event, the corresponding matrix block is framed with dark lines. Operational nuclear safety requirements for a system are given in the FSAR subsection describing the system and each requirement is referenced to the most significant block in Matrices 3 on Table G.5-3. The references are given as coordinates in Matrix 3 on Table G.5-3 as follows:

Example:



G.5.3.4 Meaning of Matrix 3 on Table G.5-3

The entries corresponding to a given event (horizontally across the entire width of Matrix 3 on Table G.5-3) form a comprehensive statement of the safety actions and station systems that must be the subject of operational nuclear safety requirements to satisfy the nuclear safety oprational criteria. System requirements and safety actions are related to the criteria for which they are essential. The entries corresponding to a given system (vertically down Matrix 3) form a comprehensive statement of the needs for, or restrictions against, the system's actions in the designated operating state. Note that requirements for indications refer to either direct or indirect indications of the listed process variable.

With the information in Matrices 3 on Table G.5-3 it is possible to determine for each system the detailed functional requirements and the detailed conditions to be observed regarding system hardware in each operating state. The detailed conditions to be observed regarding system hardware include such operational nuclear safety requirements as the number of components that must be operable and test frequencies.

G.5.3.5 Detailed Explanation of Matrix 3 Entries on Table G.5-3

The following paragraphs and the associated block diagrams describe the various events from a functional and system level viewpoint. A

more detailed analysis of the transients, accidents, and special events is presented in Section 14, Station Safety Analysis, to give the event results in terms of key station parameters.

The block diagrams of the protection sequences show only the front line systems that must perform in a protection sequence. Systems that act as auxiliaries to the front line safety systems are identified in the block diagrams of safety system auxiliaries given on Figures G.5-1 through G.5-21. Safety system auxiliaries are shown as required in Matrices 3 on Table G.5-3 for any event for which the Front Line Safety System is required. The notation used on Matrices 3 on Table G.5-3 for safety system auxiliaries reflects the need, when applicable, to ensure that a safety system auxiliary is single failure proof relative to some combination of front line safety systems. Thus, the notation 60-65SF in column 89 of a matrix would indicate that the dc power system (column 89) must be single failure proof relative to the system pair consisting of the RCICS (column 60) and the HPCIS (column 65). Thus, Matrices 3 on Table G.5-3 reflect an indepth analysis of the auxiliaries that support more than one front line safety system.

If a front line safety system fails safe following failure of an auxiliary system, the auxiliary system is considered nonessential and is not indicated on the block diagrams or the matrices. Auxiliaries are not shown for indications or systems needed only for planned operations.

The treatment on the matrices of the Offsite AC Power System versus the Standby AC Power System (onsite diesel generators) is worth noting. Most of the transients and accidents do not necessarily involve loss of the offsite ac power supply; however, the Standby AC Power System is by itself capable of accommodating the events within the nuclear safety operational criteria. But the protection sequences resulting from considering only the use of the Standby AC Power System are all similar to the sequence for event 28, loss of all offsite ac power. To reveal the characteristic differences in the protection sequences, offsite ac power is assumed available for all transients except event 28. For those transients in which the use of off-site power is used in the protection sequence, appropriate symbols are entered in column 96 (offsite ac power), but the single failure criterion is not applied because with offsite power a lesser case of event 28 results. For accidents the protection sequences shown are those that assume the use only of the Standby AC Power System.

The conventions used on the protection sequence diagrams associated with each event are illustrated on Figure G.5-22. A separate protection sequence diagram is shown for each essential safety action requiring the operation of two or more systems.

G.5.3.5.1 Planned Operations

The requirements for the planned operations normally involve using limits on certain key process variables. The matrices generally

display the process variable limits, associate the limits with the system for which the limit is essential, and show the indications that are necessary for the station operator to comply with the limits.

The following list relates the safety actions for planned operation with unacceptable results for planned operation.

Event 1 - Refueling Outage

Refueling outage operations include all planned operations pertaining to the nuclear core that are normally accomplished when the reactor vessel head is removed. These operations apply to operating states A and B only.

The limits associated with the essential safety actions are usually obvious. The power level control needed for these states refers to a minimum neutron source level. This minimum must exist prior to withdrawing control rods for a reactor startup. Possible refueling restriction sequences are indicated on Figure G.5-23. This figure shows that either the procedural restrictions or the refueling interlocks can maintain core alterations conditions to within the envelope of conditions considered by station safety analysis.

State B considerations include those shown on the matrices for state A, but because the reactor is critical or subcritical by less than the reactivity worth of any one control rod, an additional requirement - rod worth control - must be observed.

As shown on Figure G.5-24, adequate rod worth control can be achieved either by operator control of rod position via the control rod position indications or by the action of the rod worth minimizer program of the process computer. In state B, power level control requires both a minimum and a maximum boundary on core power level.

Event 2 - Achieving Criticality

Through definition, achieving criticality is applicable to all operating states. States A, C, and E each consist of that part of "achieving criticality" in which the reactor is shut down (more than one rod subcritical), while B, D, and F each consist of that part of "achieving criticality" in which the reactor is not shut down. For states B, D, and F, the actual condition of criticality ($k_{eff}=1$) may or may not exist at any instant. For example, in operating state F it is possible to be not shut down, yet still be in the latter stages of achieving criticality ($k_{eff}<1$). Note that the condition of shutdown for these analyses is a nuclear definition only.

The nuclear system may be subject to its greatest loads under operating state F. Because operating states A through E may be considered an approach of state F for this operation, there are more safety action requirements in state F than in other states.

To achieve the safety actions for this operation, it is essential that certain station systems be operating or available to operate in state F. An example of a system requirement is the entry of "5" in block F2-46. This entry means that the core power level indications must be operating to satisfy safety action 5, core power level control. This system requirement is self explanatory in that the operator must have some indication of power level to control it. Similarly, to satisfy the essential safety action of rod worth control, the process computer and the control rod position indications share this function. That is, to have rod worth control, the operator either must have the process computer operating, or must have some indication of control rod positions. See Figure G.5-24.

It is essential that certain parameters be limited because of various system limitations (symbol "L" on Matrix 3 on Table G.5-3). For example, limits are placed on pressure, water quality, and temperature because of reactor vessel design limitations (entries 8L, 10L, and 9L on matrix). Also, a limit exists on the power level because of fuel design limitations (entry 5L on matrix). Similar reasoning for safety action limits is followed for the remaining systems. These limits are discussed in the section describing each individual system. A restriction (symbol "R" on Matrix 3) is placed on the operation of the recirculation system to avoid the thermal stresses on the reactor vessel that might otherwise arise from the cold loop startup of a recirculation pump.

Event 3 - Heatup

Heatup, which begins where achieving criticality ends and includes all station actions that are normally accomplished in approaching nuclear system rated conditions, begins in state D and continues into state F. Most of the systems required to be operable to accomplish the essential safety actions are obvious. Limits (L) are placed on several process variables because of certain system limitations. For example, there is a limit on the core power level because of the fuel (5L in block F3-53).

For some systems requirements, two or more systems share the responsibility for safety action. In particular, the safety action for core neutron flux distribution control is accomplished through operator observation of either the core flux distribution indications and the Neutron Monitoring System (which drives the indicators) or the control rod position indications. Also, the rod worth control safety action is accomplished through either automatic operation of the process computer or operator observation of the control rod position indications. See Figure G.5-24.

Event 4 - Power Operation

Operating state F is the only state in which the reactor can be under normal station operation in excess of heatup power; therefore, the other states are not applicable to this event.

To achieve certain of the safety actions for this operation, several station systems are indicated in Matrix 3 on Table G.5-3 as essential to safe operation. Of these actions, the core neutron flux distribution control (safety action 6) is accomplished by the reactor operator observing either the control rod position indications (System 52) or the core neutron flux distribution indications (System 47); these flux indications are driven by System 74, the Neutron Monitoring System. Similarly, the control rod position indications (System 52) and the process computer (System 81) are shown to share rod worth control (Action 13). These systems are required to be continuously operating. Other system requirements are more obvious.

Certain parameters are limited because of an individual system limitation. For example, there is a limit on pressure (pressure control, Safety Action 8) because of the reactor vessel. Similarly, there are limits on temperature and leakage because of the reactor vessel. The imposed limits are discussed in the section on each individual system.

A restriction (symbol "R" in Matrix 3) is placed on the operation of the Recirculation System for this event, to avoid the thermal stresses that may arise on the reactor vessel from the cold loop startup of a recirculation pump.

Event 5 - Achieving Shutdown

The planned operation of achieving shutdown applies in states B, D, and F. In states A, C, and E the reactor is in the shutdown condition by definition. The most demanding state is F, which requires the most safety actions. In operating state D the vessel pressure is less than 785 psig, resulting in minor modifications in safety actions and system requirements. In operating state B, when the vessel head is removed, there is no requirement for reactor vessel pressure control or core neutron flux distribution control.

In states F and D shared (S) functions are noted in several instances. The core neutron flux distribution indications share with the control rod position indications the function of controlling, through operator observation, the core neutron flux distributions to avoid unacceptable result 1-4. This requires use of essential indications to assure that operation remains within the envelope of conditions considered by station safety analysis. See F-5-47, F-5-52, and F-5-74 entries. Similarly, control rod position indications and the process computer share (S) the function of rod worth control to assure operation within the envelope of station safety analysis. Because the nuclear system is being depressurized and cooled during this operation, there is a limit on the rate of temperature change.

Event 6 - Cooldown

Since cooldown begins where achieving shutdown ends, by definition, cooldown is applicable in states A, C, and E, in which the reactor

has achieved shutdown. The requirements for limits and parameter indications for the safety actions essential to this planned operation are essentially identical to the requirements for these same safety actions for other planned operations, except that there is a limit on rate of temperature change during cooldown. This is indicated as a limit for temperature control on the reactor vessel and a requirement for temperature indications.

Events 7 - 11 (numbers not used)

G.5.3.5.2 Abnormal Operational Transients

The safety requirements and protection sequences for abnormal operational transients are described in the following paragraphs. The protection sequence block diagrams show only the sequence of front line safety systems. On transferring the information in the sequence diagrams to Matrix 3 on Table G.5-3, the auxiliaries for the front line safety systems are accounted for on the matrices.

The following list relates the safety actions for transients with the unacceptable safety results. Appropriate entries are made in Matrix 3 on Table G.5-3 to indicate when a safety action is essential to avoiding an unacceptable result for a given transient.

Events 12 and 13 - Generator Trip and Turbine Trip

Generator trip and turbine trip (with bypass) are similar abnormal operational transients. Although the turbine trip is a more severe transient than the generator trip, the required safety actions and the systems required to fulfill the safety actions are the same. The state D turbine trip is an insignificant event because the initial power level is low.

Figures G.5-25 and G.5-26 illustrate the different protection sequences pertinent to producing the scram and pressure relief safety actions. Scram is accomplished through operation of the RPS and the control rod drive system. Pressure is relieved by operating the Nuclear System Pressure Relief System. As the cited figures indicate, all systems involved with scram and pressure relief must individually meet the single failure criterion.

Event 14 - Isolation of All Main Steam Lines

Isolation of all main steam lines is the most severe and rapid in operating state F during power operation. In other states, isolation becomes a lesser case of the state F sequence.

Figure G.5-27 shows how scram is accomplished through the actions of the RPS and the Control Rod Drive System. The Nuclear System Pressure Relief System provides pressure relief. Feedwater flow continues and decay heat may cause an increase in nuclear system pressure, eventually lifting relief valves. This core cooling sequence can be used in any situation in which the main heat sink is lost and normal feedwater flow continues.

Event 15 - Isolation of One Main Steam Line

Isolation of one main steam line causes a significant transient only in state F during high power operation. Scram is the only unique action required to avoid excessive fuel damage and nuclear system overpressure. Because the Feedwater System and main condenser remain in operation following the event, no unique requirement arises for core cooling.

As shown on Figure G.5-28, the scram safety action is accomplished through the combined actions of the Neutron Monitoring, Reactor Protection, and Control Rod Drive Systems.

Event 16 - Loss of Condenser Vacuum

A loss of vacuum in the turbine generator condenser can occur at any time steam pressure is available and is therefore applicable to operating states C, D, E, and F. This nuclear system pressure increase transient is the most severe of the pressure increase transients and is similar in analysis to the event 14 (isolation of all main steam lines). However, because this transient becomes a lesser case in the operating states in which the reactor is more than one control rod subcritical, scram protection in states C and E is not needed.

In operating state D at more than 600 psig and in state F, scram is initiated to prevent fuel damage and is accomplished with the actions of the RPS and Control Rod Drive System. Figure G.5-29 shows the sequence. The Nuclear System Pressure Relief System provides pressure relief. Feedwater flow continues and decay heat may cause an increase in nuclear system pressure, eventually lifting relief valves.

Event 17 -(number not used)

Event 18 - Loss of Feedwater Heating

Significant feedwater heating occurs only in operating state F. A loss of feedwater heating causes such a mild transient that no protective actions are required when the reactor is on automatic recirculation flow control. If the reactor is on manual flow control, however, the neutron flux increase associated with this event will reach the scram setting. As shown on Figure G.5-30, the scram safety action is accomplished through the combined actions of the Neutron Monitoring, Reactor Protection, and Control Rod Drive Systems.

Event 19 - Shutdown Cooling (RHRS) Malfunction Temperature Decrease

No unique safety actions are required to avoid the unacceptable safety results for transients as a result of a reactor coolant temperature decrease induced by misoperation of the shutdown cooling heat exchangers. In states B and D, where the reactor is critical or

near critical, the slow power increase resulting from the moderator temperature decrease would be controlled by the operator in the same manner as is normally used to control power in the source of intermediate power ranges.

Event 20 - Inadvertent Pump Start (Temperature Decrease)

An inadvertent pump start (temperature decrease) is defined as an unintentional start of any nuclear system pump that adds sufficient cold water to the reactor coolant inventory to cause a measurable decrease in moderator temperature. While all the safety criteria apply, no unique safety actions are required to control the adverse effects of such a pump start, that is, pressure increase and temperature decrease in states A, C, and E. In these operating states the safety criteria are met through the basic design of the station systems, and no safety action is specified. In states B, D, and F, where the reactor is not shut down, the station operator can control any power changes by the normal manner for controlling power.

Event 21 - Control Rod Withdrawal Error

No unique safety actions are required in operating states A, C, and E because the core is more than one rod subcritical and could not achieve criticality with the full withdrawal of any one control rod.

During high power operation (state F) an uninhibited, erroneous rod withdrawal cannot result in a condition requiring any unique safety action; no fuel damage results. However, during station operation in the intermediate range achieving criticality, heatup, achieving shutdown (states B, D, and F) a high flux scram is required to terminate the increase in power level. As shown on Figure G.5-31, the required scram is accomplished by the Neutron Monitoring, Reactor Protection, and Control Rod Drive Systems.

Events 22, 23, and 24 - Fuel Assembly Insertion, Control Rod Removal, and Control Curtain Removal

An inadvertent, positive reactivity insertion is a result from the erroneous physical operations pertaining to fuel assembly insertion, control rod removal, or control curtain removal is possible only when the reactor vessel head is removed.

During core alterations the mode switch is in the REFUEL position, which allows the refueling equipment to be positioned over the core and also inhibits control rod withdrawal; therefore, this transient applies only to operating state A. No unique safety actions are required because the total worth (positive reactivity) of one fuel assembly, one control rod, or one control curtain is inadequate to cause a criticality. Moreover, the mechanical designs of the control rod assembly and the control curtain assembly physcially prevent their removal without the simultaneous or prior removal of the adjacent fuel assemblies.

Event 25 - Pressure Regulator Failure

A pressure regulator failure is most severe and rapid in operating state F during power operation. In state E, pressure regulator failure becomes a milder case of the state F sequence. In states C and D, this transient is even less severe because main steamline pressure is initially less than 785 psig.

The various protection sequences giving the safety actions are shown on Figure G.5-32. Depending on the station conditions existing prior to the event, scram will be either on turbine stop valve closure (due to high reactor water level), low reactor water level, or on main steam line isolation. The sequence resulting in reactor vessel isolation also depends on initial conditions. In state F, with the mode switch in RUN, isolation is initiated when main steam line pressure decreases to 782 psig. Under other conditions, isolation is initiated by either high or low reactor vessel water level. Feedwater flow continues and decay heat may cause an increase in nuclear system pressure, which may lift the relief valves.

Event 26 - Inadvertent Opening of a Relief Valve

A relief valve can inadvertently open in any state. Due to the normal operation of water makeup systems (feedwater, control rod drive cooling water) the water level cannot be lowered far enough to threaten fuel damage, therefore, no safety actions are required.

Event 27 - Loss of Feedwater Flow

A loss of feedwater flow results in a net decrease in the coolant inventory available for core cooling. A partial or complete loss of feedwater flow can occur in states C, D, E, and F. Appropriate responses to this transient include a reactor scram on low water level and maintenance of reactor vessel water level.

As shown on Figure G.5-33, the Reactor Protection and Control Rod Drive Systems effect a scram on low water level. The Reactor Vessel Isolation Control System and the main steam line isolation valves act to isolate the reactor vessel. After the main steam line isolation valves close, decay heat slowly raises system pressure to the lowest relief valve setting. Pressure is relieved by the Nuclear System Pressure Relief System. Core cooling is necessary to restore and maintain water level. Either the HPCI or the RCIC can maintain adequate water level; as a pair, these systems satisfy the single failure criterion for core cooling.

The requirements for operating state D are the same as for state F. The requirements for operating states C and E are the same as for states D and F, except that the scram action is not required.

Event 28 - Total Loss of Offsite Power

There is a variety of possible electrical failures; the most severe of which is total loss of offsite ac power. Each case is considered under all planned operating conditions. Figure G.5-34 shows the

various electrical sequences considered by this analysis. The sequences are selected by applying the abnormal operational transient selection criteria.

As shown on Figure G.5-34, the Reactor Protection and Control Rod Drive Systems effect a scram on turbine trip or loss of RPS MG sets. The Reactor Vessel Isolation Control System and the main steam line isolation valves act to isolate the reactor vessel. After the main steam line isolation valves close, decay heat slowly raises system pressure to the lowest relief valve setting. Pressure is relieved by the Nuclear System Pressure Relief System.

After the reactor is isolated and feedwater flow has been lost, decay heat may cause an increase in nuclear system pressure, eventually lifting relief valves, and allowing reactor vessel water to decrease. The core cooling sequence shown on Figure G.5-34 shows the short and long term sequence for achieving adequate cooling in spite of any single failure. This same sequence could be used in any situation in which the main heat sink and normal feedwater flow are lost. For initial pressures over 104 psig, either RCIC or HPCI maintain water level in the reactor vessel as steam is relieved via the relief valves to the torus.

The RHR torus cooling mode can be used to remove the heat received by the torus water. When the torus water temperature limit is reached, a controlled depressurization must be started by operating the relief valves through remote manual control, considered part of the Automatic Depressurization System. Starting the depressurization at this temperature limit ensures that the torus water retains its capability to suppress a full blowdown of the nuclear system within the bounds of the experimental data observed in actual pressure suppression tests. Depending on nuclear system pressure, the RCIC, HPCI, LPCI, or Core Spray Systems can be used to maintain reactor vessel water level until the Shutdown Cooling System can be placed in operation (planned operation). For initial pressures between 75 and 104 psig, high pressure cooling is not required. By manually actuating the relief valves, the nuclear system pressure is maintained low enough that the LPCI and Core Spray Systems can cool the core. For initial pressures less than 75 psig, the system is returned to planned operation shutdown cooling.

Event 29 - Recirculation Flow Control Failure (Decreasing Flow)

This recirculation flow control malfunction causes a decrease in core coolant flow. Because such a decrease can be accommodated within the operational nuclear safety criteria without the action of any protection systems, no unique operational nuclear safety requirements arise.

Event 30 - Trip of One Recirculation Pump

The trip of one recirculation pump produces a milder transient than does the simultaneous trip of two recirculation pumps (see event 31). No unique safety actions are required in response to this transient.

Event 31 - Trip of Two Recirculation Pumps

The transient resulting from this two loop trip is not severe enough to require any unique safety action. The transient is compensated for by the inherent stability of the reactor. This event is not applicable to states A and B because the reactor vessel head is off and the recirculation pumps normally would not be in use. The trip could occur in states C through F; however, the absence of matrix entries signifies the reactor's ability to accommodate the transient with no unique safety action requirement.

Event 32 - Recirculation Pump Seizure

A recirculation pump seizure considers the instantaneous stoppage of the pump motor shaft of one recirculation pump. The case involving operation at design power in state F is described in Section 14. While all the safety criteria apply, no safety actions are required to control the adverse effects of pump seizure. MCHFR is maintained at above 1.0, and no damage occurs to the fuel barrier. No scram is required, and no unique safety action is necessary to control temperature and pressure. The safety criteria are met through the basic design of the station systems.

Event 33 - Recirculation Flow Control Failure Increasing Flow

A recirculation flow control failure causing increased flow is applicable in states C, D, E, and F. In state F, the accompanying increase in power level is accommodated through a reactor scram. As shown on Figure G.5-35, the scram safety action is accomplished through the combined actions of the Neutron Monitoring, Reactor Protection, and Control Rod Drive Systems.

Event 34 - Startup of Idle Recirculation Pump

The cold loop startup of an idle recirculation pump is most severe and rapid for those operating states in which the reactor may be critical (states B, D, and F). When the transient occurs in the range of 10 to 60 percent power operation, no safety action responses are required. Reactor power in this case would be limited to approximately 60 percent design power because of core flow limitations while using one working recirculation loop. Above 60 percent power, a high neutron flux scram is initiated. Should the event occur when the reactor is not at power operation, but critical (<10 percent), the resulting transient may produce a high level neutron flux scram of the intermediate range monitors (IRM).

As shown on Figure G.5-36, the scram action is accomplished through the combined actions of the Neutron Monitoring, Reactor Protection, and Control Rod Drive Systems. At power operation (10-60 percent) the high level IRM scram is not initiated, because the core flux monitoring has been shifted to the average power range monitors (APRM).

Event 35 - Loss of Shutdown Cooling

The loss of RHR shutdown cooling can occur only during the low pressure portion of a normal reactor shutdown and cooldown. At this time the RHR System is operating in the shutdown cooling mode which occurs only in states A, B, C, and D.

As shown on Figure G.5-37, for most single failures that could result in loss of shutdown cooling, no unique safety actions are required; in these cases shutdown cooling is simply reestablished using other, normal shutdown cooling equipment. In the cases where the RHRS shutdown cooling suction line becomes inoperative, a unique requirement for cooling arises. In states A and B, in which the reactor vessel head is off, either half of the RHRS LPCI mode can be used to maintain water level. In states C and D, in which the reactor vessel head is on and the system can be pressurized, the Low Pressure Cooling Systems, relief valves (manually operated), and RHRS torus cooling mode can be used to maintain water level and remove decay heat.

Event 36 - Feedwater Controller Failure - Maximum Demand

A feedwater controller failure maximum demand leads to an excess of coolant inventory in states C, D, E, and F. In operating states D and F, any adverse responses of the reactor caused by cooling of the moderator can be compensated by a scram. As shown on Figure G.5-38, the scram safety action is accomplished through the combined actions of the Neutron Monitoring Reactor Protection and Control Rod Drive Systems. Pressure relief is required in states C, D, E, and F and is achieved through the operation of the Nuclear System Pressure Relief System.

G.5.3.5.3 Accidents

The safety requirements and protection sequences for accidents are described in the following paragraphs. The protection sequence block diagrams show only the sequence of front line safety systems. On transferring the information in the sequence diagrams to Matrices 3, the auxiliaries for the front line safety systems are accounted for on the matrices.

Table G.5-3 relates the safety actions for transients with the unacceptable safety results. Appropriate entries are made on Matrices 3 on Table G.5-3 when a safety action is essential to avoiding an unacceptable result for a given accident.

Event 37 - (number not used)

Event 38 - Control Rod Drop Accident

The control rod drop accident results from an assumed failure of the rod to drive coupling after the rod becomes stuck in its fully inserted position. It is assumed that the control rod drive is fully withdrawn before the stuck rod falls out of the core at a maximum

velocity of 5 ft/sec. The control rod velocity limiter, an engineered safeguard, limits the rod drop velocity to less than this value. The resultant radioactive material release is maintained below the requirements of 10CFR100. This accident is analyzed in Section 14.

The control rod drop accident is applicable in operating states C, D, E, and F. The rod drop accident cannot occur in states A and B because rod coupling integrity is checked on each rod if more than one rod is withdrawn. No safety actions are required in states C and E where the station is shut down by more than one rod prior to the accident.

Figures G.5-39, Sheets 1, 2, and 3, show the different protection sequences for the control rod drop accident.

Event 39 - Pipe Break Inside Primary Containment

Pipe breaks inside the primary containment are considered only when the nuclear system is significantly pressurized and result in the release of steam or water into the primary containment. The most severe case is the circumferential break of the largest recirculation system pipe. This is called the design basis accident (DBA) for the loss of coolant from a pipe break inside the primary containment.

As shown on Figures G.5-40, Sheets 1 and 2, in operating states C and E (reactor shut down, but pressurized) a pipe break accident up to the DBA can be accommodated within the operational nuclear safety criteria through the various operations of the main steam line isolation valves, Core Standby Cooling Systems (HPCI, Automatic Depressurization System, LPCI and Core Spray System), Primary Containment and Reactor Vessel Isolation Control System, Primary Containment, Secondary Containment, Standby Gas Treatment System, Main Control Room Environmental Control System, and the incident detection circuitry. In operating states D and F (reactor not shut down, but pressurized) the same equipment is required as in states C and E but, in addition, the RPS and the Control Rod Drive System must operate to scram the reactor. The limiting items, on which the operation of the above equipment is based, are the allowable fuel temperature and the primary containment pressure capability.

The control rod drive housing supports are considered necessary whenever the system is pressurized to prevent excessive control rod movement through the bottom of the pressure vessel following the postulated rupture of one control rod drive housing, (a lesser case of loss of coolant accident).

After completion of the automatic actions of the above equipment, manual operation of RHR (torus cooling mode) is required to maintain primary containment pressure and fuel temperature within limits during long term cooldown following the accident.

Event 40 - Fuel Handling Accident

This unlikely accident, described in Section 14 as the drop of one fuel assembly from the refueling equipment during fuel handling operation, is possible in any state during fuel handling operations.

Because in state A the mode switch is in the REFUEL position, allowing the refueling equipment to be positioned over the core and inhibiting control rod withdrawal, the design basis accident is applicable to operating state A only. Accident considerations include mechanical fuel damage caused by impact and a subsequent release of fission products.

The protection sequences pertinent to this accident are shown on Figure G.5-41.

Event 41 - Pipe Break Outside Primary Containment

Pipe break accidents outside the primary containment are assumed to occur any time the nuclear system is pressurized (states C, D, E, and F). This accident is most severe during operations at high power (state F). In the other states (C, D, and E) this accident becomes a lesser case of the state F sequence.

The protection sequences for the various possible pipe breaks outside the primary containment are shown on Figures G.5-42, Sheets 1, 2, and 3. As shown on Figures G.5-42, Sheets 1 and 2, special consideration must be given to the HPCI steam line break, because this system is otherwise used in response to the other pipe break accidents. The sequences show that for small breaks (breaks not requiring immediate action) the operator can use a large number of process indications to identify the break and isolate it. See Figure G.5-42.

Scram is accomplished through operation of the RPS and the Control Rod Drive System. Reactor vessel isolation is accomplished through operation of the main steam line isolation valves and the Primary Containment and Reactor Vessel Isolation Control System.

Core cooling is accomplished by either the HPCI or the manually actuated Automatic Depressurization System for a break in a main steam line. The break of a HPCI steam line (smaller steam line break accident) requires manual initiation of the Automatic Depressurization System after some time has elapsed. After the vessel has depressurized, the core is cooled by either the Core Spray System or the LPCI mode of RHR in combination with RHR torus cooling. Operation of the incident detection circuitry is required for operation of the HPCI, LPCI, and Core Spray Systems. The loss of reactor coolant for the main steam line break is restricted by the

flow restrictors. Pressure is relieved through the action of the Nuclear System Pressure Relief System.

Events 42 and 43 - (numbers not used)

G.5.3.5.4 Special Event 44 - Shutdown From Outside Control Room

This event is displayed to demonstrate the ability to safely shut down the reactor and subsequently cool the reactor to the cold shutdown state, accomplished entirely from outside the control room.

Figure G.5-43 shows the protection sequences for this event in each operating state. In state A no sequence is shown because the reactor is already in the condition finally required for the event.

A scram from outside the control room can be achieved by opening the ac supply breakers to the Control Rod Drive System. Core cooling is accomplished by continued operation of the Feedwater and the Turbine Bypass Systems. When the reactor pressure falls to 75 psig, the RHRS shutdown cooling mode is started.

G.5.3.5.5 Special Event 45 - Shutdown Without Control Rods

Shutdown without control rods is a special event devised to evaluate the plant capabilities to shut down independent of control rods. By definition, this event can only occur when the reactor is not already shut down. Therefore, the event is considered only in operating states B, D, and F. The Standby Liquid Control System must operate to avoid unacceptable result 5-1 while the Recirculation Pump Trip System must operate to avoid unacceptable result 5-2. . The design basis for the Standby Liquid Control System results from these operating criteria when applied under the most severe conditions - operating state F at rated power. As indicated on Figure G.5-44 and the matrices for states B, D, and F, the Standby Liquid Control System is manually initiated and controlled. The Recirculation Pump Trip System is an automatic system that supplements the Standby Liquid Control System during power operations and is, therefore, applicable only to states D and F.

G.5.4 Remainder of Nuclear Safety Operational Analysis

With the information presented in the protection sequence block diagrams and in Matrix 3 on Table G.5-3, it is possible to determine the functional and hardware requirements for each system. That part of the nuclear safety operational analysis is presented in those portions of the Safety Analysis Report where the system itself is described and evaluated. In each section on operational nuclear safety requirements, the essential actions of the system are identified (using Matrix 3 on Table G.5-3 as a guide). Then availability requirements and operability restrictions are established for system hardware to ensure that the essential actions of the system can be achieved.

Finally, the action that must be taken (should an operational nuclear safety requirement not be met) can be determined by considering the associated unacceptable safety results. It is expected that the measures that must be taken in the station when an operational nuclear safety requirement is not met will reflect the different levels of importance of the unacceptable safety results. The combination of an operational nuclear safety requirement and the action that must be taken, should the requirement not be met form a technical specification.

TABLE G.5-1

BWR OPERATING STATES

MATRIX 1

Conditions	States A	B	C	D	E	F
Reactor vessel head off	X	X				
Reactor vessel head on			X	X	X	X
Shutdown	X		X		X	
Not shutdown		X		X		X
Pressure <785 psig[2]	X[1]	X[1]	X	X		
Pressure >785 psig					X	X

DEFINITION:

Shutdown: K_{eff} sufficiently less than 1.0 that the full withdrawal of any one control rod could not produce criticality under the most restrictive potential conditions of temperature, pressure, core age, and fission product concentrations.

NOTES:

1. Because the reactor vessel head is off in states A and B, pressure is atmospheric pressure.

2. At less than 785 psig, the main steam line isolation valves are interlocked closed when the mode switch is in RUN.

TABLE G.5-2

TYPES OF OPERATIONS AND EVENTS APPLICABLE
IN EACH BWR OPERATING STATE
MATRIX 2

Types of Operation and Events	BWR Operating States					
	A	B	C	D	E	F
Planned Operation						
1. Refueling Outage	X	X				
2. Achieving Criticality	X	X	X	X	X	X
3. Heatup				X		X
4. Power Operation						X
5. Achieving Shutdown		X		X		X
6. Cooldown	X		X		X	
7. (Open)						
8. (Open)						
9. (Open)						
10. (Open)						
11. (Open)						
Abnormal Operational Transients						
Nuclear System Pressure Increase						
12. Generator trip				X		X
13. Turbine trip (with bypass)				X		X
14. Isolation of all main steam lines			X	X	X	X
15. Isolation of one main steam line			X	X	X	X
16. Loss of vacuum (turbine trip without bypass)			X	X	X	X

TABLE G. 5-2 (Cont)

Types of Operation and Events	BWR Operating States					
	A	B	C	D	E	F
Moderator Temperature Decrease						
17. (Open)						
18. Loss of feedwater heating						X
19. Shutdown cooling (RHRS) malfunction (temperature decrease)	X	X	X	X		
20. Inadvertent Pump Start (temperature decrease)	X	X	X	X	X	X
Reactivity Insertion						
21. Control rod withdrawal error	X	X	X	X	X	X
22. Fuel assembly insertion	X					
23. Control rod removal	X					
24. Control curtain removal	X					
Loss of Coolant Inventory						
25. Pressure regulator failure			X	X	X	X
26. Inadvertent opening of a relief valve	X	X	X	X	X	X
27. Loss of feedwater flow			X	X	X	X
28. Total loss of offsite power	X	X	X	X	X	X
Core Coolant Flow Decrease						
29. Recirculation flow control failure decreasing flow			X	X	X	X
30. Trip of one recirculation pump			X	X	X	X
31. Trip of two recirculation pumps			X	X	X	X
32. Recirculation pump seizure			X	X	X	X
Core Coolant Flow Increase						
33. Recirculation flow control failure increasing flow			X	X	X	X

FIGURE G.5-45 (cont)

Event	System or Safety Action (Column Number)	Symbol (Column Entry)	Meaning
40. Loss of coolant accident	Core spray system (45)		The dark frame around the matrix block indicates that this block represents the most significant or demanding condition from which at least one operational nuclear safety requirement for the system is derived. This block would be referenced (using the block coordinates X 40-45) in the operational nuclear safety requirements portion of the FSAR subsection describing the system.
40. Loss of coolant accident	Reactor protection system (48)	17SF	It is essential that the reactor protection system be capable of operating to achieve safety action 17 (scram) and be in a condition to meet the single failure criterion.
40. Loss of coolant accident	Control rod drive system (53)	17SF	It is essential that the control rod drive system be capable of operating to achieve safety action 17 (scram) and be in a condition to meet the single failure criterion.
40. Loss of coolant accident	LPCI (76)	22SF (S45)	It is essential that LPCI be capable of operating to achieve safety action 22 (core cooling) and that LPCI be in a condition to meet the single failure criterion. The (S45) indicates that LPCI shares with system 45 (core spray system) the obligation to satisfy the single failure criterion. The dark frame around the matrix block indicates that this block represents the most significant or demanding condition from which at

TABLE G. 5-2 (Cont)

Types of Operation and Events	BWR Operating States					
	A	B	C	D	E	F
34. Startup of idle recirculation pump	X	X	X	X	X	X
Core Coolant Temperature Increase						
35. Loss of shutdown cooling	X	X	X	X		
Excess of Coolant Inventory						
36. Feedwater controller failure-maximum demand			X	X	X	X
Accidents						
37. (Open)						
38. Control rod drop accident			X	X	X	X
39. Pipe breaks inside primary containment			X	X	X	X
40. Fuel handling accident	X	X	X	X	X	X
41. Pipe breaks outside primary containment			X	X	X	X
42. (Open)						
43. (Open)						
Special Events						
44. Shutdown from outside control room	X	X	X	X	X	X
45. Shutdown without control rods		X		X		X
46. (Open)						
47. (Open)						

PILGRIM NUCLEAR POWER STATION

BWR OPERATING STATE A

FSAR TABLE G.5-3
1 OF 6
REVISION 16–JUNE 1994

REACTOR VESSEL HEAD OFF
REACTOR SHUTDOWN
PRESSURE <785 psig

SAFETY ACTIONS — SAFETY ACTIONS FOR PLANNED OPERATION (columns 1–16)

SAFETY ACTION CODE NUMBER AND COLUMN NUMBERS	1. APPLICABILITY TO OPERATING STATE	2.	3. RADIOACTIVE MATERIAL RELEASE CONTROL	4. CORE COOLANT FLOW RATE CONTROL	5. CORE POWER LEVEL CONTROL	6. CORE NEUTRON FLUX DISTRIBUTION CONTROL	7. REACTOR VESSEL WATER LEVEL CONTROL	8. REACTOR VESSEL PRESSURE CONTROL	9. NUCLEAR SYSTEM TEMPERATURE CONTROL	10. NUCLEAR SYSTEM WATER QUALITY CONTROL	11. NUCLEAR SYSTEM LEAKAGE CONTROL	12. CORE REACTIVITY CONTROL	13. ROD WORTH CONTROL	14. REFUELING RESTRICTIONS	15. PRIMARY CONTAINMENT PRESSURE AND TEMPERATURE CONTROL	16. STORED FUEL SHIELDING, COOLING AND REACTIVITY CONTROL
TYPES OF OPERATION AND EVENTS																
PLANNED OPERATION																
1 REFUELING OUTAGE	YES		1-1		1-4		1-2 1-4			1-4		1-4		1-4		1-2 1-4
2 ACHIEVING CRITICALITY	YES		1-1		1-4		1-2 1-4			1-4		1-4		1-4		1-2 1-4
3 HEATUP	NO															
4 POWER OPERATION	NO															
5 ACHIEVING SHUTDOWN	NO															
6 COOLDOWN	YES		1-1		1-4		1-2 1-4		1-3	1-4		1-4				1-2 1-4
7																
8																
9																
10																
11																
ABNORMAL OPERATIONAL TRANSIENTS																
NUCLEAR SYSTEM PRESSURE INCREASE																
12 GENERATOR TRIP	NO															
13 TURBINE TRIP (WITH BYPASS)	NO															
14 ISOLATION OF ALL MAIN STEAM LINES	NO															
15 ISOLATION OF ONE MAIN STEAM LINE	NO															
16 LOSS OF VACUUM (TURBINE TRIP WITHOUT BYPASS)	NO															
MODERATOR TEMPERATURE DECREASE																
17																
18 LOSS OF FEEDWATER HEATING	NO															
19 SHUTDOWN COOLING (RHRS) MALFUNCTION (TEMPERATURE DECREASE)	YES															
20 INADVERTENT PUMP START (TEMPERATURE DECREASE)	YES															
REACTIVITY INSERTION																
21 CONTROL ROD WITHDRAWAL ERROR	YES															
22 FUEL ASSEMBLY INSERTION	YES															
23 CONTROL ROD REMOVAL	YES															
24 CONTROL CURTAIN REMOVAL	YES															
LINE REFERENCE	1.	2.	3.	4.	5.	6.	7.	8.	9.	10.	11.	12.	13.	14.	15.	16.
LOSS OF COOLANT INVENTORY																
25 PRESSURE REGULATOR FAILURE	NO															
26 INADVERTENT OPENING OF RELIEF VALVE	NO															
27 LOSS OF FEEDWATER FLOW	NO															
28 TOTAL LOSS OF OFF-SITE AC POWER	YES															
CORE COOLANT FLOW DECREASE																
29 RECIRCULATION FLOW CONTROL FAILURE—DECREASING FLOW	NO															
30 TRIP OF ONE RECIRCULATION PUMP	NO															
31 TRIP OF TWO RECIRCULATION PUMPS	NO															
32 RECIRCULATION PUMP SEIZURE	NO															
CORE COOLANT FLOW INCREASE																
33 RECIRCULATION FLOW CONTROL FAILURE—INCREASING FLOW	NO															
34 STARTUP OF IDLE RECIRCULATION PUMP	YES															
CORE COOLANT TEMPERATURE INCREASE																
35 LOSS OF SHUT DOWN COOLING	YES															
EXCESS OF COOLANT INVENTORY																
36 FEEDWATER CONTROLLER FAILURE: MAXIMUM DEMAND	NO															
ACCIDENTS																
37																
38 CONTROL ROD DROP ACCIDENT	NO															
39 PIPE BREAK INSIDE PRIMARY CONTAINMENT	NO															
40 FUEL HANDLING ACCIDENT	YES															
41 PIPE BREAK OUTSIDE PRIMARY CONTAINMENT	NO															
42																
43																
44 SPECIAL EVENT—SHUT DOWN FROM OUTSIDE CONTROL ROOM	YES															
45 SPECIAL EVENT—SHUT DOWN WITHOUT CONTROL RODS	NO															
46																
47																
LINE REFERENCE	1.	2.	3.	4.	5.	6.	7.	8.	9.	10.	11.	12.	13.	14.	15.	16.

SAFETY ACTIONS FOR ABNORMAL OPERATIONAL TRANSIENTS (17–23) / SAFETY ACTIONS FOR ACCIDENTS (24–37) / SAFETY ACTIONS FOR SPECIAL EVENT (38–44)

Line	17. SCRAM	18. PRESSURE RELIEF	19. CORE COOLING	20. REACTOR VESSEL ISOLATION	21. RESTORE AC POWER	22.	23.	24. SCRAM	25. REACTOR VESSEL ISOLATION	26. ESTABLISH PRIMARY CONTAINMENT	27. ESTABLISH SECONDARY CONTAINMENT	28. CORE COOLING	29.	30. CONTAINMENT COOLING	31. STOP CONTROL ROD EJECTION (PASSIVE)	32. LIMIT REACTIVITY INSERTION RATE (PASSIVE)	33. RESTRICT LOSS OF REACTOR COOLANT (PASSIVE)	34.	35. PRESSURE RELIEF	36. CONTROL ROOM ENVIRONMENTAL CONTROL	37.	38. SHUT DOWN REACTOR FROM OUTSIDE CONTROL ROOM	39. COLD SHUTDOWN FROM OUTSIDE CONTROL ROOM	40.	41. SHUTDOWN WITHOUT CONTROL RODS	42.	43.	44. AUTOMATIC TRIP OF THE RECIRCULATION PUMPS
1																												
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47																												

PLANT SYSTEMS (columns 45–81)

LINE REFERENCE	45. CORE COOLANT FLOW RATE INDICATIONS	46. CORE POWER LEVEL INDICATIONS	47. CORE NEUTRON FLUX DISTRIBUTION INDICATIONS	48. REACTOR VESSEL WATER LEVEL INDICATIONS	49. REACTOR VESSEL PRESSURE INDICATIONS	50. REACTOR VESSEL WATER TEMPERATURE INDICATIONS	51. NUCLEAR SYSTEM LEAKAGE INDICATIONS	52. CONTROL ROD POSITION INDICATIONS	53. FUEL	54. REACTOR VESSEL	55. RECIRCULATION SYSTEM	56. FUEL POOL WATER LEVEL INDICATIONS	57. MAIN STEAM LINE ISOLATION VALVES	58. CONTROL ROD DRIVE SYSTEM	59. REACTOR MANUAL CONTROL SYSTEM	60. RCICS	61. RHRS (SHUTDOWN COOLING MODE)	62. RHRS (LPCI MODE)	63. RHRS (CONTAINMENT COOLING MODE)	64. NUCLEAR SYSTEM PRESSURE RELIEF SYSTEM	65. HPCIS	66. AUTOMATIC DEPRESSURIZATION SYSTEM	67. CORE SPRAY SYSTEM	68. STANDBY LIQUID CONTROL SYSTEM	69. CONTROL ROD DRIVE HOUSING SUPPORTS	70. STANDBY COOLANT SUPPLY	71. REACTOR PROTECTION SYSTEM	72. REACTOR MODE SWITCH	73. PRIMARY CONTAINMENT AND REACTOR VESSEL ISOLATION SYSTEM	74. NEUTRON MONITORING SYSTEM	75. REFUELING INTERLOCKS	76. AIR EJECTOR OFFGAS RADIATION MONITORING SYSTEM	77. MAIN STACK RADIATION MONITORING SYSTEM	78. PROCESS LIQUID RADIATION MONITORS	79. REFUELING/VENTILATION EXHAUST MONITORING SYSTEM	80. MAIN STEAM LINE RADIATION MONITORS	81. PROCESS COMPUTER
1.		5		7					7L 12L	10L	9R	16		14 5R 12L														14R		5	14 (S87)		3		3		
2.		5		7					7L 12L	10L	9R	16		14 5R 12L																5	14 (S87)		3		3		
3.																																					
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6.		5		7		9			7L 12L	9L 10L	9R	16		5R 12L																5			3		3		
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PLANT SYSTEMS (columns 82–110)

LINE REFERENCE	82. PRIMARY CONTAINMENT (PASSIVE) (EXCLUDING ISOLATION VALVES)	83. Aux. TORUS WATER STORAGE	84. Aux. CONDENSATE STORAGE SYSTEM	85. LIQUID RADWASTE SYSTEM	86. GASEOUS RADWASTE SYSTEM	87. REFUELING BRIDGE AND HOISTS	88. Aux. STANDBY AC POWER	89. Aux. DC POWER SYSTEM (125/250 VOLT)	90. REACTOR BUILDING (PASSIVE)	91. STANDBY GAS TREATMENT SYSTEM	92. INCIDENT DETECTION CIRCUITRY	93. CONTROL ROD VELOCITY LIMITER	94. RHRS DRYWELL SPRAY MODE	95. MAIN STEAM LINE FLOW RESTRICTOR	96. Aux. OFFSITE AC POWER	97. Aux. REACTOR BUILDING CLOSED COOLING WATER	98. Aux. TURBINE BUILDING CLOSED COOLING WATER	99. Aux. SALT SERVICE WATER	100. MAIN CONTROL ROOM ENVIRONMENTAL CONTROL	101. Aux. INSTRUMENT AND SERVICE AIR	102. REACTOR BUILDING ISOLATION AND CONTROL	103. DRYWELL TEMPERATURE AND PRESSURE INDICATIONS	104. TORUS WATER TEMPERATURE AND LEVEL INDICATIONS	105. MAIN STACK (PASSIVE)	106. NEW AND SPENT FUEL STORAGE	107. Aux. EQUIPMENT AREA COOLING SYSTEM	108. Aux. DC POWER SYSTEM (24 VOLT)	109. Aux. PRIMARY CONTAINMENT VACUUM RELIEF SYSTEM	110. RECIRCULATION PUMP TRIP SYSTEM
1.				3L	3L	14 (S75)																			16L				
2.				3L	3L	14 (S75)																			16L				
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