January 29, 1992

Docket No. 50-293

Mr. Roy A. Anderson
Senior Vice President - Nuclear
Boston Edison Company
Pilgrim Nuclear Power Station
RFD #1 Rocky Hill Road
Plymouth, Massachusetts 02360

Dear Mr. Anderson:

SUBJECT: ISSUANCE OF AMENDMENT NO. 140 TO FACILITY OPERATING LICENSE NO. DPR-35 - PILGRIM NUCLEAR POWER STATION (TAC NO. M80622)

The Commission has issued the enclosed Amendment No. 140 to Facility Operating License No. DPR-35 for the Pilgrim Nuclear Power Station. This amendment is in response to your application dated June 11, 1991.

This amendment revises the thermal and pressurization limit curves of Figure 3.6.1 and 3.6.2 of the technical specifications, and adds a new curve to Figure 3.6.3 to cover operations between 10 and 32 effective power years. Also included are changes to associated limiting conditions of operation, surveillance and bases sections 3.9 by letter of August 5, 1991 and sections 3.1, 3.2, 3.3 and 4.3, sections 3.5.C, D & E, and 3.10 by letter of October 4, 1991.

A copy of our Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly FEDERAL REGISTER Notice.

Sincerely,

Original signed by Ronald B. Eaton

Ronald B. Eaton, Senior Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 140 to License No. DPR-35
2. Safety Evaluation

cc w/enclosures:
See next page

OFC	:LA:PDI-3	:PM:PDI-3	:OGC	:D:PDI-3	:
NAME	:MRushbrook	:REaton:sk	: MZOBLER	:WButler	:
DATE	: 1/ /92	: 1/ /92	:1/16/92	:1/29/92	:

OFFICIAL RECORD COPY
Document Name: PI AMEND M80622

See comments on overclipped (base)

9202060226 920129
PDR ADOCK 05000293
P PDR

CP1
DF01

AMENDMENT NO. 140 TO DPR-35 PILGRIM NUCLEAR POWER STATION DATED January 29, 1992

DISTRIBUTION:
Docket File 50-293
NRC PDR
Local PDR
PDI-3 Reading
S. Varga
J. Calvo
M. Rushbrook
R. Eaton
W. Butler
OGC - 15 B18
Dennis Hagan - MNBB 3206
E. Jordan - MNBB 3701
B. Grimes - 9 A2
G. Hill (4) - P1-37
Wanda Jones - MNBB 7103
C. Grimes - 11 F23
ACRS (10) - P-315
GPA/PA - 2 G5
OC/LFMB - MNBB 11104
J. Linville, Region I
R. Lobel - 17G5
J. Tsao - 7D16

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

January 29, 1992

Docket No. 50-293

Mr. Roy A. Anderson
Senior Vice President - Nuclear
Boston Edison Company
Pilgrim Nuclear Power Station
RFD #1 Rocky Hill Road
Plymouth, Massachusetts 02360

Dear Mr. Anderson:

SUBJECT: ISSUANCE OF AMENDMENT NO. 140 TO FACILITY OPERATING LICENSE NO. DPR-35 - PILGRIM NUCLEAR POWER STATION (TAC NO. M80622)

The Commission has issued the enclosed Amendment No. 140 to Facility Operating License No. DPR-35 for the Pilgrim Nuclear Power Station. This amendment is in response to your application dated June 11, 1991.

This amendment revises the thermal and pressurization limit curves of Figure 3.6.1 and 3.6.2 of the technical specifications, and adds a new curve to Figure 3.6.3 to cover operations between 10 and 32 effective power years. Also included are changes to associated limiting conditions of operation, surveillance and bases sections 3.9 by letter of August 5, 1991 and sections 3.1, 3.2, 3.3 and 4.3, sections 3.5.C, D & E, and 3.10 by letter of October 4, 1991.

A copy of our Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly FEDERAL REGISTER Notice.

Sincerely,

Ronald B. Eaton, Senior Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 140 to
   License No. DPR-35
2. Safety Evaluation

cc w/enclosures:
See next page

cc:
Mr. William C. Rothert, Acting
Vice President of Operations
and Station
Pilgrim Nuclear Power Station
RFD #1 Rocky Hill Road
Plymouth, Massachusetts 02360

Resident Inspector
U. S. Nuclear Regulatory Commission
Pilgrim Nuclear Power Station
Post Office Box 867
Plymouth, Massachusetts 02360

Chairman, Board of Selectmen
11 Lincoln Street
Plymouth, Massachusetts 02360

Office of the Commissioner
Massachusetts Department of
Environmental Protection
One Winter Street
Boston, Massachusetts 02108

Office of the Attorney General
One Ashburton Place
20th Floor
Boston, Massachusetts 02108

Mr. Robert M. Hallisey, Director
Radiation Control Program
Massachusetts Department of
Public Health
305 South Street
Boston, Massachusetts 02130

Regional Administrator, Region I
U. S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, Pennsylvania 19406

Mr. John Dietrich
Licensing Division Manager
Boston Edison Company
25 Braintree Hill Park
Braintree, Massachusetts 02184

Mr. H. Vernon Oheim
Manager, Reg. Affairs Dept.
Pilgrim Nuclear Power Station
RFD #1 Rocky Hill Road
Plymouth, Massachusetts 02360

Ms. Elaine D. Robinson
Nuclear Information Manager
Pilgrim Nuclear Power Station
RFD #1, Rocky Hill Road
Plymouth, Massachusetts 02360

Mr. Dale C. Jenkins, Jr.
Secretary of Public Safety
Executive Office of Public Safety
One Ashburton Place
Boston, Massachusetts 02108

Mr. David Rodham, Director
Massachusetts Civil Defense Agency
400 Worcester Road
P.O. Box 1496
Framingham, Massachusetts 01701-0317
Attn: James Muckerheide

Chairmen, Citizens
Urging Responsible Energy
P. O. Box 2621
Duxbury, Massachusetts 02331

Citizens at Risk
P. O. Box 3803
Plymouth, Massachusetts 02361

W. S. Stowe, Esquire
Boston Edison Company
800 Boylston St., 36th Floor
Boston, Massachusetts 02199

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

BOSTON EDISON COMPANY

DOCKET NO. 50-293

PILGRIM NUCLEAR POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 140
License No. DPR-35

1. The Nuclear Regulatory Commission (the Commission or the NRC) has found that:

   A. The application for amendment filed by the Boston Edison Company (the licensee) dated June 11, 1991 and letters of August 5 and October 4, 1991, requesting bases changes comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-35 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 140, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

9202060233 920129
PDR ADOCK 05000293
P PDR

21

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler, Director
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 29, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 140

FACILITY OPERATING LICENSE NO. DPR-35

DOCKET NO. 50-293

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
40b	40b
70	70
89A	89A
89C	89C
116	116
117	117
118	118
123	123
124	124
124A	124A
125	125
128	128
128A	128A
---	128B *
139	139
139A	139A
198	198
204A	204A

* Denotes new page

Main Steam Line High Radiation

High radiation levels in the main steam line tunnel above that due to the normal nitrogen and oxygen radioactivity is an indication of leaking fuel. A scram is initiated whenever such radiation level exceeds seven times normal background. The purpose of this scram is to reduce the source of such radiation to the extent necessary to prevent excessive turbine contamination. Discharge of excessive amounts of radioactivity to the site environs is prevented by the air ejector off-gas monitors which cause an isolation of the main condenser off-gas line.

Reactor Mode Switch

A reactor mode switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status. Ref. Section 7.2.3.9 FSAR.

Manual Scram

The manual scram function is active in all modes, thus providing for a manual means of rapidly inserting control rods during all modes of reactor operation.

Scram Discharge Instrument Volume

The control rod drive scram system is designed so that all of the water which is discharged from the reactor by a scram can be accommodated in the discharge piping. The two scram discharge volumes accommodate in excess of 39 gallons of water each and are at the low points of the scram discharge piping. No credit was taken for these volumes in the design of the discharge piping as concerns the amount of water which must be accommodated during a scram.

During normal operation the scram discharge volume system is empty; however, should it fill with water, the water discharged to the piping could not be accommodated, which would result in slow scram times or partial control rod insertion. To preclude this occurrence, redundant and diverse level detection devices in the scram discharge instrument volumes have been provided which will alarm when water level reaches 4.5 gallons, initiate a control rod block at 18 gallons, and scram the reactor when the water level reaches 39 gallons. As indicated above, there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or amount of insertion of the control rods. This function shuts the reactor down while sufficient volume remains to accommodate the discharged water and precludes the situation in which a scram would be required but not be able to perform its function adequately.

A source range monitor (SRM) system is also provided to supply additional neutron level information during start-up but has no scram functions. Ref. Section 7.5.4 FSAR.

The APRM's cover the "Refuel" and "Startup/ Hot Standby" modes with the APRM 15% scram, and the power range with the flow dent. With the established setting of 7 times normal background, and main steam line isolation valve closure, fission product release is limited so that 10 CFR 100 guidelines are not exceeded for this accident. Reference FSAR Section 14.5.1 and Appendix R.3.2.5.

Pressure instrumentation is provided to close the main steam isolation valves in Run Mode when the main steam line pressure drops below 880 psig. In the Refuel and Startup Mode this function is replaced by high reactor water level. This function is provided primarily to provide protection against a pressure regulator malfunction which results in the control and/or bypass valves opening. With the trip settings specified, inventory loss is limited so that fuel is not uncovered.

The HPCI high flow and temperature instrumentation are provided to detect a break in the HPCI steam piping. Tripping of this instrumentation results in actuation of HPCI isolation valves. Tripping logic for the high flow is a 1 out of 2 logic, and all sensors are required to be operable.

Temperature is monitored at three (3) locations with four (4) temperature sensors at each location. Two (2) sensors at each location are powered by "A" direct current control bus and two (2) by "B" direct current control bus. Each pair of sensors, e.g., "A" or "B", at each location are physically separated and the tripping of either "A" or "B" bus sensor will actuate HPCI isolation valves.

The trip settings of ≤ 300% of design flow for high flow and 200°F or 170°F, depending on sensor location, for high temperature are such that core uncovery is prevented and fission product release is within limits.

The RCIC high flow and temperature instrumentation are arranged the same as that for the HPCI. The trip setting of ≤ 300% for high flow and 200°F, 170°F and 150°F, depending on sensor location, for temperature are based on the same criteria as the HPCI.

The Reactor Water Cleanup System high flow and temperature instrumentation are arranged similar as that for the HPCI. The trip settings are such that core uncovery is prevented and fission product release is within limits.

The instrumentation which initiates CSCS action is arranged in a dual bus system. As for other vital instrumentation arranged in this fashion, the Specification preserves the effectiveness of the system even during periods when maintenance or testing is being performed. An exception to this is when logic functional testing is being performed.

BASES:

basis is given is subsection 3.5.2 of the FSAR, and the safety evaluation is given in subsection 3.5.4. This support is not required if the reactor coolant system is at atmospheric pressure since there would then be no driving force to rapidly eject a drive housing. Additionally, the support is not required if all control rods are fully inserted and if an adequate shutdown margin with one control rod withdrawn has been demonstrated, since the reactor would remain subcritical even in the event of complete ejection of the strongest control rod.

3. In the course of performing normal startup and shutdown procedures, the reactor operator follows a pre-specified sequence for the withdrawal or insertion of control rods. The specified sequences are characterized by homogeneous, scattered patterns of control rods selected for withdrawal or insertion. The maximum control rod worths encountered in these patterns for the initial core load are presented in FSAR Figure R.3-1. These sequences are developed prior to initial operation of the unit to limit the reactivity worths of individual control rods in the core. These control rod sequences, together with the integral rod velocity limiters which will limit the velocity during free fall to less than five feet per second, limit the potential reactivity insertion such that the consequences of a control rod drop accident will not exceed a peak calculated enthalpy of 280 calories/gram generated in the fuel. The design limit of 280 calories/gram is selected for limiting peak enthalpies in UO_2 and is assumed to be the lower threshold at which rapid fuel dispersal and damaging pressure pulses to the primary system might occur.

As discussed in FSAR Section 14.5.1.3, the calculated radiological consequences of a control rod drop accident are well within the guideline values of 10 CFR Part 100.

BASES:

Above 10% of design power assuming a single operator error, it will not be possible for the maximum rod worth to exceed 0.020 delta K in accordance with Specification 3.3.B.3.b(2).

Specification 4.3.B.3 requires a sequence of checks and tests on the RWM to verify its operability before startup and before reducing power to less than 10% of design power. These checks and tests assure that the actions of the control operator are always monitored and blocked when in error should they lead to a condition which might cause fuel damage during the control rod drop accident.

Under these specification limits, the maximum energy deposition in the fuel and the number of fuel rods damaged resulting from a control rod drop accident, assuming Technical Specification limits on scram times (Specification 3.3.C) and rod drop velocity (5 feet/second), is established to be below the consequences calculated by the licensee for the hot standby critical case. Reference: FSAR Section 14.5.1. Therefore, the assumptions used by the licensee and the NRC in estimating the number of failed fuel rods and fuel damage resulting from the excursion energy generated by the rod drop accident appear conservative within the LCO.

BASES:

3.5.C HPCI

The limiting conditions for operating the HPCI System are derived from the Station Nuclear Safety Operational Analysis (FSAR Appendix G) and a detailed functional analysis of the HPCI System (FSAR Section 6).

The HPCIS is provided to assure that the reactor core is adequately cooled to limit fuel clad temperature in the event of a small break in the nuclear system and loss-of-coolant which does not result in rapid depressurization of the reactor vessel. The HPCIS permits the reactor to be shut down while maintaining sufficient reactor vessel water level inventory until the vessel is depressurized. The HPCIS continues to operate until reactor vessel pressure is below the pressure at which LPCI operation or Core Spray System operation maintains core cooling.

The capacity of the system is selected to provide this required core cooling. The HPCI pump is designed to pump 4250 gpm at reactor pressures between 1100 and 150 psig. Two sources of water are available. Initially, demineralized water from the condensate storage tank is used instead of injecting water from the suppression pool into the reactor.

When the HPCI System begins operation, the reactor depressurizes more rapidly than would occur if HPCI was not initiated due to the condensation of steam by the cold fluid pumped into the reactor vessel by the HPCI System. As the reactor vessel pressure continues to decrease, the HPCI flow momentarily reached equilibrium with the flow through the break. Continued depressurization causes the break flow to decrease below the HPCI flow and the liquid inventory begins to rise. This type of response is typical of the small breaks. The core never uncovers and is continuously cooled throughout the transient so that no core damage of any kind occurs for breaks that lie within the capacity range of the HPCI.

The analysis in FSAR Appendix G shows that the ADS provides a single failure proof path for depressurization for postulated transients and accidents. The RCIC is required as an alternate source of makeup to the HPCI only in the case of loss of all offsite A-C power. Considering the HPCI and the ADS plus RCIC as redundant paths, and considering judgments of the reliability of the ADS and RCIC systems, a 7-day allowable repair time is specified.

The requirement that HPCI be operable when reactor coolant temperature is greater that 365°F is included in Specification 3.5.C.1 to clarify that HPCI need not be operable during certain testing (e.g., reactor vessel hydro testing at high reactor pressure and low reactor coolant temperature). 365°F is approximately equal to the saturation steam temperature at 150 psig.

BASES:

3.5.D RCIC System

The RCIC is designed to provide makeup to the nuclear system as part of the planned operation for periods when the normal heat sink is unavailable. The Station Nuclear Safety Operational Analysis, FSAR Appendix G, shows that RCIC also serves as redundant makeup system on total loss of all offsite power in the event that HPCI is unavailable. In all other postulated accidents and transients, the ADS provides redundancy for the HPCI. Based on this and judgments on the reliability of the HPCI system, an allowable repair time of seven days is specified.

The requirement that RCIC be operable when reactor coolant temperature is greater than 365°F is included in Specification 3.5.D.1 to clarify that RCIC need not be operable during certain testing (e.g., reactor vessel hydro testing at high reactor pressure and low reactor coolant temperature). 365°F is approximately equal to the saturation steam temperature at 150 psig.

BASES:

3.5.E Automatic Depressurization System (ADS)

The limiting conditions for operating the ADS are derived from the Station Nuclear Safety Operational Analysis (FSAR Appendix G) and a detailed functional analysis of the ADS (FSAR Section 6).

This specification ensures the operability of the ADS under all conditions for which the automatic or manual depressurization of the nuclear system is an essential response to station abnormalities.

The nuclear system pressure relief system provides automatic nuclear system depressurization for small breaks in the nuclear system so that the low pressure coolant injection (LPCI) and the core spray systems can operate to protect the fuel barrier.

Because the Automatic Depressurization System does not provide makeup to the reactor primary vessel, no credit is taken for the steam cooling of the core caused by the system actuation to provide further conservatism to the CSCS. Performance analysis of the Automatic Depressurization System is considered only with respect to its depressurizing effect in conjunction with LPCI or Core Spray. There are four valves provided and each has a capacity of 800,000 lb/hr at a reactor pressure of 1125 psig.

The allowable out of service time for one ADS valve is determined as seven days because of the redundancy and because of HPCIS operability; therefore, redundant protection for the core with a small break in the nuclear system is still available.

The ADS test circuit permits continued surveillance on the operable relief valves to assure that they will be available if required.

3.6 PRIMARY SYSTEM BOUNDARY

Applicability:

Applies to the operating status of the reactor coolant system.

Objective:

To assure the integrity and safe operation of the reactor coolant system

Specification:

A. Thermal and Pressurization Limitations

1. The average rate of reactor coolant temperature change during normal heatup or cooldown shall not exceed 100°F/hr when averaged over a one-hour period except when the vessel temperatures are above 450°F. The reactor vessel flange to adjacent reactor vessel shell temperature differential shall not exceed 145°F.

2. The reactor vessel shall not be pressurized for hydrostatic and/or leakage tests, and critical core operation shall not be conducted unless the reactor vessel temperatures are above those defined by the appropriate curves on Figures 3.6.1, 3.6.2, and 3.6.3. (Linear interpolation between curves is permitted). At stated pressure, the reactor vessel bottom head may be maintained at temperatures below those temperatures corresponding to the adjacent reactor vessel shell as shown in Figures 3.6.1 and 3.6.2.

4.6. PRIMARY SYSTEM BOUNDARY

Applicability:

Applies to the periodic examination and testing requirements for the reactor cooling system.

Objective:

To determine the condition of the reactor coolant system and the operation of the safety devices related to it.

Specification:

A. Thermal and Pressurization Limitations

1. During heatups and cooldowns, with the reactor vessel temperature less than or equal to 450°F, the temperatures at the following locations shall be permanently logged at least every 15 minutes until the difference between any two readings at individual locations taken over a 45 minute period is less than 5°F:

   a. Reactor vessel shell adjacent to reactor vessel flange

   b. Reactor vessel shell flange

   c. Recirculation loops A and B

2. Reactor vessel shell temperatures, including reactor vessel bottom head, and reactor coolant pressure shall be permanently logged at least every 15 minutes whenever the shell temperature is below 220°F and the reactor vessel is not vented.

   Test specimens of the reactor vessel base, weld and heat affected zone metal subjected to the highest fluence of greater than 1 Mev neutrons shall be installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The specimens and sample program shall conform to the

3.6.A Thermal and Pressurization Limitations (Cont'd)

In the event this requirement is not met, achieve stable reactor conditions with reactor vessel temperature above that defined by the appropriate curve and obtain an engineering evaluation to determine the appropriate course of action to take.

3. The reactor vessel head bolting studs shall not be under tension unless the temperature of the vessel head flange and the head is greater than 55°F.

4. The pump in an idle recirculation loop shall not be started unless the temperatures of the coolant within the idle and operating recirculation loops are within 50°F of each other.

5. The reactor recirculation pumps shall not be started unless the coolant temperatures between the dome and the bottom head drain are within 145°F.

6. Thermal-Hydraulic Stability

Core thermal power shall not exceed 25% of rated thermal power without forced recirculation.

B. Coolant Chemistry

1. The reactor coolant system radioactivity concentration in water shall not exceed 20 microcuries of total iodine per ml of water.

4.6.A Thermal and Pressurization Limitations (Cont'd)

requirements of ASTM E 185-66. Selected neutron flux specimens shall be removed at the frequency required by Table 4.6.3 and tested to experimentally verify adjustments to Figures 3.6.1, 3.6.2, and 3.6.3 for predicted NDT temperature irradiation shifts.

3. When the reactor vessel head bolting studs are tensioned and the reactor is in a Cold Condition, the reactor vessel shell temperature immediately below the head flange shall be permanently recorded.

4. Prior to and during startup of an idle recirculation loop, the temperature of the reactor coolant in the operating and idle loops shall be permanently logged.

5. Prior to starting a recirculation pump, the reactor coolant temperatures in the dome and in the bottom head drain shall be compared and permanently logged.

B. Coolant Chemistry

1. a. A reactor coolant sample shall be taken at least every 96 hours and analyzed for radioactivity content.

b. Isotopic analysis of a reactor coolant sample shall be made at least once per month.

TABLE 4.6.3
REACTOR VESSEL MATERIAL
SURVEILLANCE PROGRAM WITHDRAWAL SCHEDULE

Capsule Number	Effective Full Power Years (EFPY)
1	4.17
2	15 (approx.)
3	32 (End of Life)

3.6.B Coolant Chemistry (Cont'd)

2. The reactor coolant water shall not exceed the following limits with steaming rates less than 100,000 pounds per hour, except as specified in 3.6.B.3:

   Conductivity ... 2 μmho/cm
   Chloride ion ... 0.1 ppm

3. For reactor startups and for the first 24 hours after placing the reactor in the power operating condition, the following limits shall not be exceeded.

   Conductivity. . . 10 μmho/cm
   Chloride ion. . . 0.1 ppm

4. Except as specified in 3.6.B.3 above, the reactor coolant water shall not exceed the following limits when operating with steaming rates greater than or equal to 100,000 pounds per hour.

   Conductivity. . . 10 μmho/cm
   Chloride ion. . . 1.0 ppm

5. If Specification 3.6.B cannot be met, an orderly shutdown shall be initiated and the reactor shall be in Hot Shutdown within 24 hrs. and Cold Shutdown within the next 8 hours.

3.6.C Coolant Leakage

Any time irradiated fuel is in the reactor vessel and coolant temperature is above 212°F, the following limits shall be observed:

1. Operational Leakage

   a. Reactor coolant system leakage shall be limited to:

      1. No Pressure Boundary Leakage
      2. $\leq$5 gpm Unidentified Leakage
      3. $\leq$25 gpm Total Leakage averaged over any 24 hour period.

4.6.B Coolant Chemistry (Cont'd)

2. During startups and at steaming rates less than 100,000 pounds per hour, a sample of reactor coolant shall be taken every four hours and analyzed for chloride content.

3. a. With steaming rates of 100,000 pounds per hour or greater, a reactor coolant sample shall be taken at least every 96 hours and analyzed for chloride ion content.

   b. When all continuous conductivity monitors are inoperable, a reactor coolant sample shall be taken at least daily and analyzed for conductivity and chloride ion content.

4.6.C Coolant Leakage

Any time irradiated fuel is in the reactor vessel and coolant temperature is above 212°F, the following surveillances shall be performed:

1. Operational Leakage

Demonstrate drywell leakage is within the limits specified in 3.6.C.1 by monitoring the coolant leakage detection systems required to be operable by 3.6.C.2 at least once every 8 hours.

Amendment No. 82, 94, 140

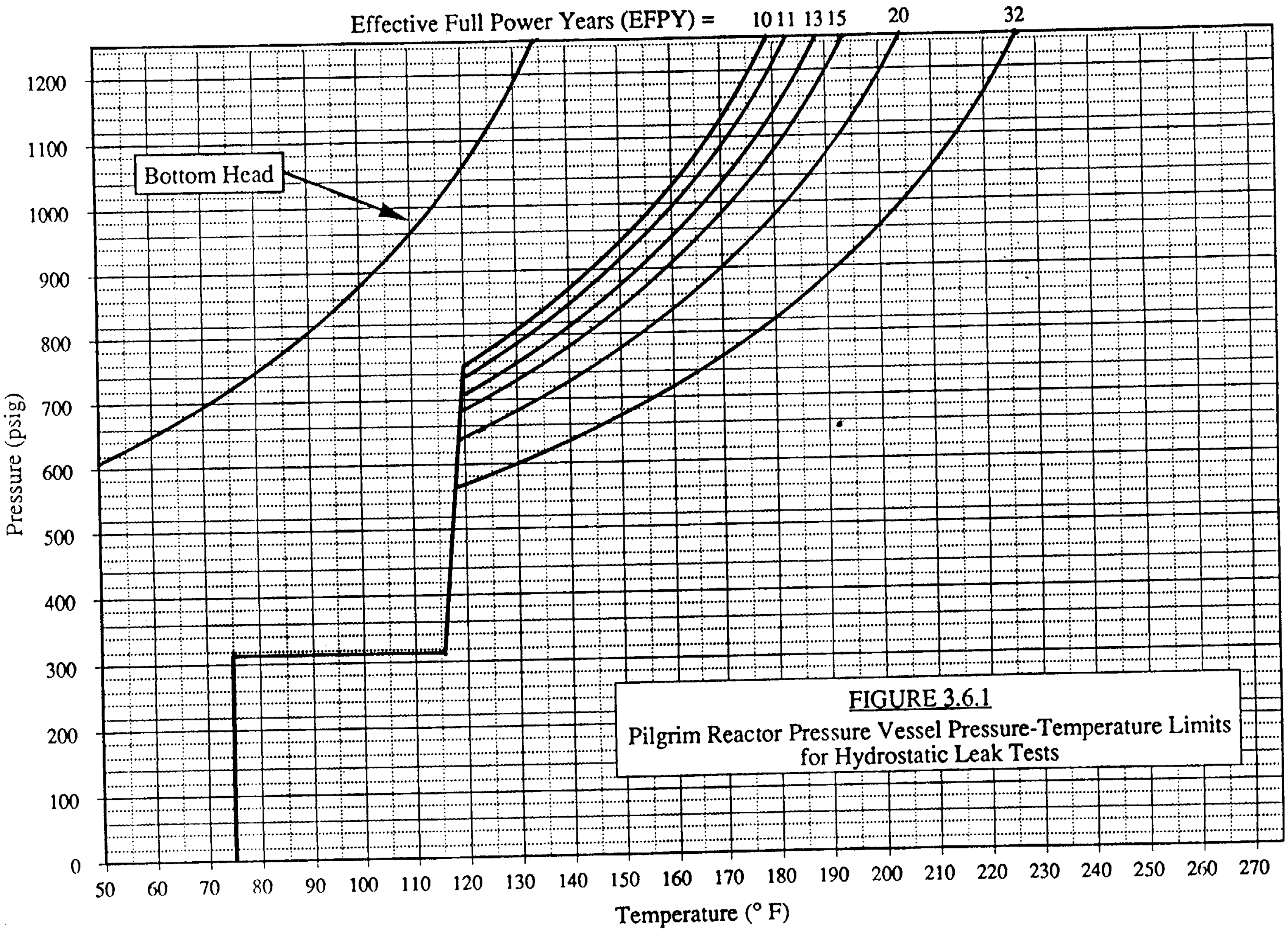


FIGURE 3.6.1

Pilgrim Reactor Pressure Vessel Pressure-Temperature Limits for Hydrostatic Leak Tests

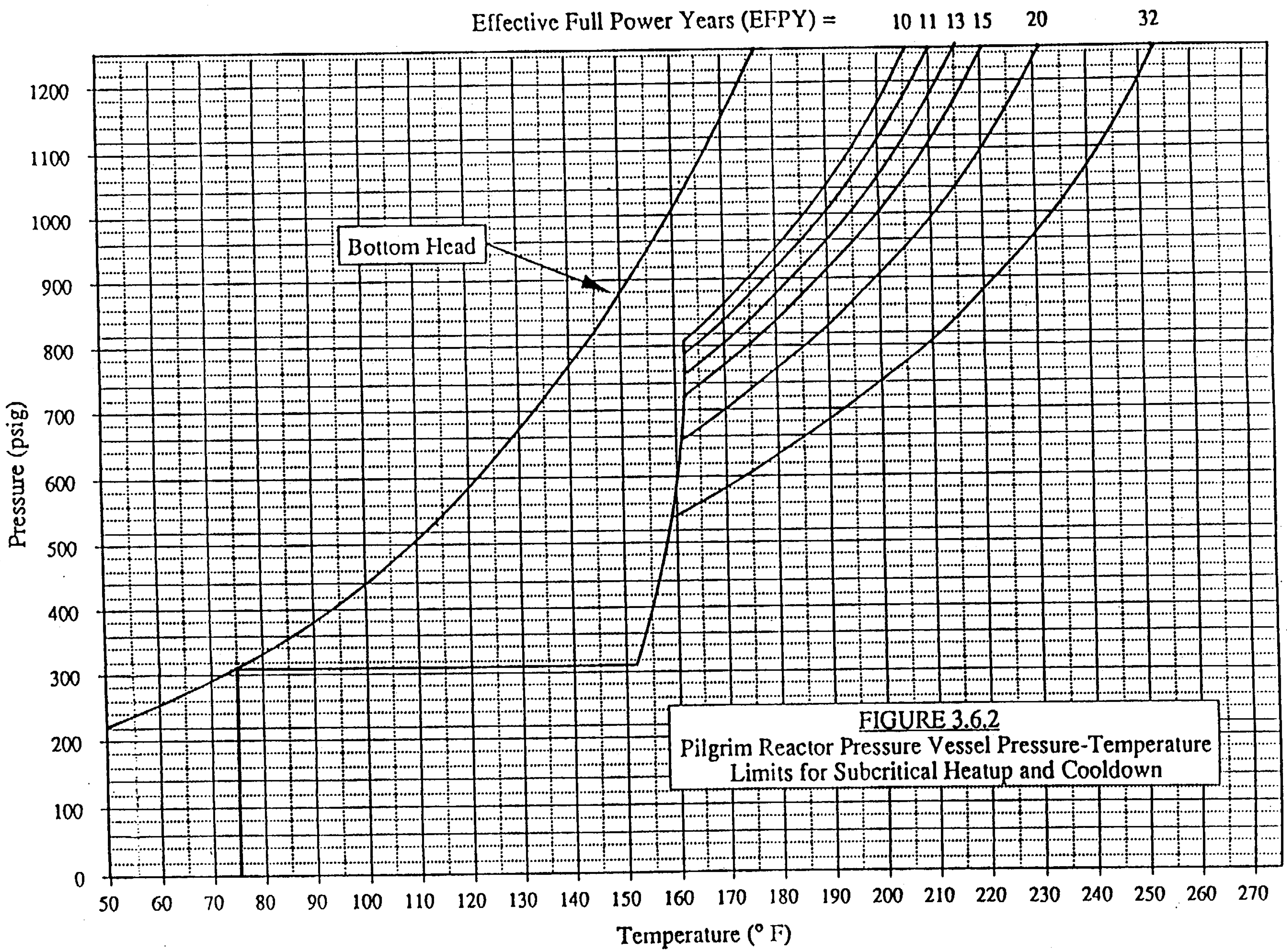
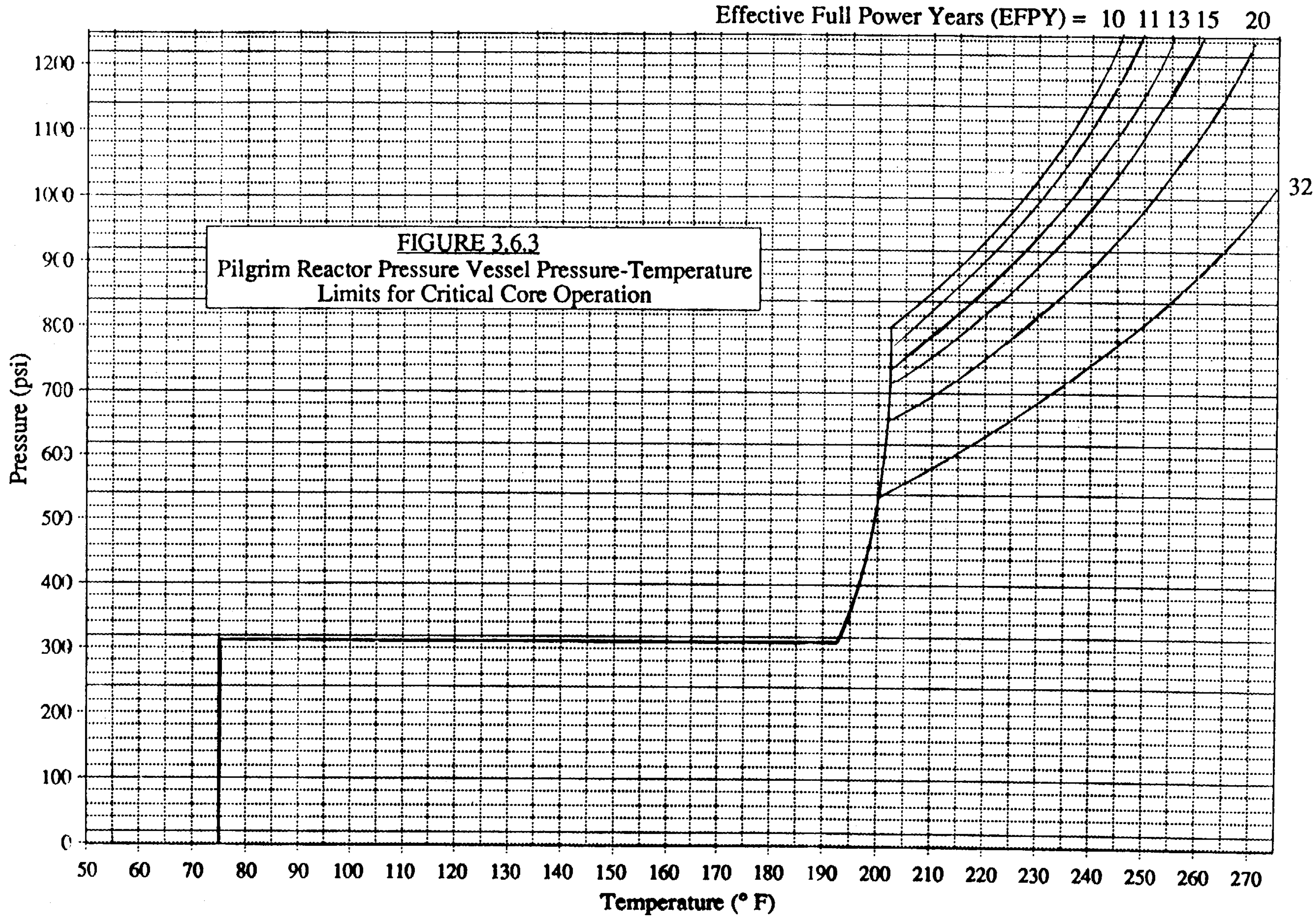


FIGURE 3.6.2
Pilgrim Reactor Pressure Vessel Pressure-Temperature Limits for Subcritical Heatup and Cooldown


Effective Full Power Years (EFPY) = 10 11 13 15 20
32
FIGURE 3.6.3
Pilgrim Reactor Pressure Vessel Pressure-Temperature
Limits for Critical Core Operation
Pressure (psi)
0
100
200
300
400
500
600
700
800
900
1000
1100
1200
50 60 70 80 90 100 110 120 130 140 150 160 170 180 190 200 210 220 230 240 250 260 270
Temperature (° F)

Bases:

3.6.A and 4.6.A

Thermal and Pressurization Limitations (Cont'd)

The reactor coolant system is a primary barrier against the release of fission products to the environs. In order to provide assurance that this barrier is maintained at a high degree of integrity, restrictions have been placed on the operating conditions to which it can be subjected.

Appendix G to 10CFR50 defines the temperature-pressurization restrictions for hydrostatic and leak tests, pressurization, and critical operation. These limits have been calculated for Pilgrim and are contained in Figures 3.6.1, 3.6.2, and 3.6.3.

The bottom head, defined as the spherical portion of the reactor vessel located below the lower circumferential weld, was also evaluated. Reference transition temperatures (RT_{NDT}) were developed for the bottom head and the resulting pressure vs. temperature curves plotted on Figures 3.6.1 and 3.6.2. It has been determined that the bottom head temperatures are allowed to lag the vessel shell temperatures, Reference: Teledyne Engineering Services (TES) report TR-6051C-1, dated June 27, 1986. The referenced analysis utilizes the stress results established in the Combustion Engineering Inc., Pilgrim Reactor Vessel Design Report, No. CENC 1139, dated 1971, and combines the stress analysis results, specific to the bottom head, with the pressurization temperatures necessary to maintain fracture toughness requirements in accordance with the ASME Boiler and Pressure Vessel Code, Section III, the criteria of 10CFR Part 50, Appendix G, and the supplementary guidelines of Reg. Guide 1.99, Rev. 2.

For Pilgrim pressure-temperature restrictions, two locations in the reactor vessel are limiting. The closure region controls at lower pressures and the beltline controls at higher pressures.

The nil-ductility transition (NDT) temperature is defined as the temperature below which ferritic steel breaks in a brittle rather than ductile manner. Radiation exposure from fast neutrons (>1 mev) above about 10^{17} nvt may shift the NDT temperature of the vessel metal above the initial value. Impact tests from the first material surveillance capsule removed at 4.17 EFPY indicated a maximum RT_{NDT} shift of 55°F for the weld specimens.

Neutron flux wires and samples of vessel material are installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The wires and samples will be periodically removed and tested to experimentally verify the values used for Figures 3.6.1, 3.6.2, and 3.6.3. The withdrawal schedule of Table 4.6.3 has been established as required by 10CFR50, Appendix H.

The RT_{NDT} of the closure region is - 5°F. The initial RT_{NDT} for the beltline weld and basemetal are -50°F and 0°F respectively. These RT_{NDT} temperatures are based upon unirradiated test data, adjusted for specimen orientation in accordance with USNRC Branch Technical Position MTEB 5-2.

3.6.A and 4.6.A Thermal and Pressurization Limitations (Cont'd)

The closure and bottom head regions are not exposed to neutron fluence (> 1 Mev) over the vessel life sufficient to cause a shift in RT_{NDT}. The pressure-temperature limitations (Figures 3.6.1, 3.6.2, and 3.6.3) of the closure and bottom head regions will therefore remain constant throughout vessel life. Only the beltline region of the reactor vessel will experience a shift in RT_{NDT} with a resultant increase in Pressure-Temperature limits.

The curves apply to 100% bolt preload condition, but are conservative for lesser bolt preload conditions.

For critical core operation when the water level is within the normal range for power operation and the pressure is less than 20% of the preservice system hydrostatic test pressure (313 psi), the minimum permissible temperature of the highly stressed regions of the closure flange is RT_{NDT} + 60 = 55°F. A conservative cutoff limit of 75°F was chosen as shown on Figure 3.6.3 and as permitted by 10CFR50 Appendix G, paragraph IV. A.3. This same cutoff is included in the limits for hydrostatic and leak tests and for non-critical operation, as shown on Figures 3.6.1 and 3.6.2 respectively, in order to be consistant with the limits for critical operation.

The closure region is more limiting than the feedwater nozzle with regards to both stress intensity and RT_{NDT}. Therefore the pressure-temperature limits of the closure are controlling.

The adjusted reference temperature shift is based on Regulatory Guide 1.99, Revision 2, dated May 1988; the analytical results of General Electric Report MDE 277-1285, Revision 1, dated January 21, 1985, regarding projected neutron fluence; and Teledyne Engineering Services Reports, TR-6052B-1, Revision 1, dated June 26, 1986, as supplemented by TR-7487, dated April 16, 1991, for RT_{NDT} vs. fluence as a function of temperature and pressure, and TR-6052C-1, dated June 27, 1986, for the RPV bottom head pressurization temperatures.

BASES:

3.9

The general objective of this Specification is to assure an adequate source of electrical power to operate the auxiliaries during plant operation, to operate facilities to cool and lubricate the plant during shutdown, and to operate the engineered safeguards following an accident. There are three sources of a-c electrical energy available; namely, the startup transformer, the diesel generators and the shutdown transformer. The d-c supply is required for switchgear and engineered safety feature systems. Specification 3.9.A states the required availability of a-c and d-c power; i.e., an active off-site a-c source, a back-up source of off-site a-c power and the maximum amount of on-site a-c and d-c sources.

The diesel fuel supply consists of two (2) 25,000 gallon tanks. The level alarms will be set to ensure a minimum supply of 19,800 gallons in each tank.

Auxiliary power for PNPS is supplied from two sources; either the unit auxiliary transformer or the startup transformer. Both of these transformers are sized to carry 100% of the auxiliary load. If the startup transformer is lost, the unit can continue to operate since the unit auxiliary transformer is in service, the shutdown transformer is available, and both diesel generators are operational.

If the startup and shutdown transformers are both lost, the reactor power level must be reduced to a value whereby the unit could safely reject the load and continue to supply auxiliary electric power to the station.

In the normal mode of operation, the startup transformer is energized, two diesel generators and the shutdown transformer are operable. One diesel generator may be allowed out of service based on the availability of power from the startup transformer, the shutdown transformer and the fact that one diesel generator carries sufficient engineered safeguards equipment to cover all breaks. With the shutdown transformer and one diesel generator out of service, both 345kV supply lines must be available for the startup transformer.

Upon the loss of one on-site and one off-site power source, power would be available from the other immediate off-site power source and the one operable on-site diesel to carry sufficient engineered safeguards equipment to cover all breaks. In addition to these two power sources, removal of the Isolated Phase Bus flexible connectors would allow backfeed of power through the main transformer to the unit auxiliary transformer and provide power to carry the full station auxiliary load. The time required to perform this operation is comparable to the time the reactor could remain on RCIC operation before controlled depressurization need be initiated.

A battery charger is supplied with each of the 125 and 250 volt batteries and, in addition, (1) a 125 volt shared back-up battery charger is supplied which

3.10 BASES:

B. Core Monitoring

The SRM's are provided to monitor the core during periods of station shutdown and to guide the operator during refueling operations and station startup. Requiring two operable SRM's in or adjacent to any core quadrant where fuel or control rods are being moved ensures adequate monitoring of that quadrant during such alterations. The requirement of 3 counts per second provides assurance that neutron flux is being monitored and ensures startup is conducted only if the source range flux level is above the minimum assumed in the control rod drop accident.

The limiting conditions for operation of the SRM subsystem of the Neutron Monitoring System are derived from the Station Nuclear Safety Operational Analysis (FSAR Appendix G) and a functional analysis of the neutron monitoring system. The specification is based on the Nuclear Safety Requirements for Plant Operation in Subsection 7.5.10 of the FSAR.

A spiral unloading program is one by which the fuel is in the outermost cells (four fuel bundles surrounding a control blade) is removed first. Unloading continues by removing the remaining outermost fuel cell by cell. The center cell will be the last removed.(1) A spiral loading program is one by which fuel is loaded on the periphery of the previously loaded fueled region beginning around a single SRM. Spiral unloading and reloading will preclude the creation of flux traps (moderator filled cavities surrounded on all sides by fuel).

During spiral unloading, the SRM's shall have an initial count rate of $\geq$ 3 cps with all rods fully inserted. The count rate will diminish during fuel removal. Under the special condition of complete spiral core unloading, it is expected that the count rate of the SRM's will drop below 3 cps before all of the fuel is unloaded.

Since there will be no reactivity additions, a lower number of counts will not present a hazard. When all of the fuel has been removed to the spent fuel storage pool, the SRM's will no longer be required. Requiring the SRM's to be operational prior to fuel removal assures that the SRM's are operable and can be relied on even when the count rate may go below 3 cps.

During spiral reload, SRM operability will be verified by using a portable external source every 12 hours until the required amount of fuel is loaded to maintain 3 cps. As an alternative to the above, up to two fuel assemblies will be loaded in different cells containing control blades around each SRM to obtain the required 3 cps. Until these assemblies have been loaded, the 3 cps requirement is not necessary.

(1)During selected refueling outages, prior to initiating spiral unloading, the central controlled cell will be removed to facilitate inspection of the Core Spray Spargers.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 140 TO FACILITY OPERATING LICENSE NO. DPR-35

BOSTON EDISON COMPANY

PILGRIM NUCLEAR POWER STATION

DOCKET NO. 50-293

1.0 INTRODUCTION

By letter dated June 11, 1991, the Boston Edison Company (the licensee) submitted a request for changes to the Pilgrim Nuclear Power Station, Technical Specifications (TS). The requested changes would revise the pressure/temperature (P/T) limits in the Pilgrim Technical Specifications, Section 3.6. The proposed P/T limits were requested for 10, 11, 13, 15, 20, and 32 effective full power years (EFPY). In October 1991, the EFPY is about 9.1. The proposed P/T limits were developed using Regulatory Guide (RG) 1.99, Revision 2. Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Effect on Plant Operations," recommends RG 1.99, Rev. 2, be used in calculating P/T limits, unless the use of different methods can be justified. A P/T limits for the bottom head of the reactor vessel were also requested.

To evaluate the P/T limits, the staff uses the following NRC regulations and guidance: Appendices G and H of 10 CFR Part 50; the ASTM Standards and the ASME Code, which are referenced in Appendices G and H; 10 CFR 50.36(c)(2); RG 1.99, Rev. 2; Standard Review Plan (SRP) Section 5.3.2; and Generic Letter 88-11.

Each licensee authorized to operate a nuclear power reactor is required by 10 CFR 50.36 to provide Technical Specifications for the operation of the plant. In particular, 10 CFR 50.36(c)(2) requires that limiting conditions of operation be included in the Technical Specifications. The P/T limits are among the limiting conditions of operation in the Technical Specifications for all commercial nuclear plants in the U.S. Appendices G and H of 10 CFR Part 50 describe specific requirements for fracture toughness and reactor vessel material surveillance that must be considered in setting P/T limits. An acceptable method for constructing the P/T limits is described in SRP Section 5.3.2.

Appendix G of 10 CFR Part 50 specifies fracture toughness and testing requirements for reactor vessel materials in accordance with the ASME Code and, in particular, that the beltline materials in the surveillance capsules be tested in accordance with Appendix H of 10 CFR Part 50. Appendix H, in turn, refers to ASTM Standards. These tests define the extent of vessel embrittlement at the time of capsule withdrawal in terms of the increase in reference temperature. Appendix G also requires the licensee to predict the effects of neutron irradiation on vessel

9202060235 920129
PDR ADOCK 05000293
P PDR

embrittlement by calculating the adjusted reference temperature (ART) and Charpy upper shelf energy (USE). Generic Letter 88-11 requested that licensees and permittees use the methods in RG 1.99, Rev. 2, to predict the effect of neutron irradiation on reactor vessel materials. This guide defines the ART as the sum of unirradiated reference temperature, the increase in reference temperature resulting from neutron irradiation, and a margin to account for uncertainties in the prediction method.

Appendix H of 10 CFR Part 50 requires the licensee to establish a surveillance program to periodically withdraw surveillance capsules from the reactor vessel. Appendix H refers to the ASTM Standards which, in turn, require that the capsules be installed in the vessel before startup and that they contain test specimens made from plate, weld, and heat-affected-zone (HAZ) materials of the reactor beltline. By letters dated August 5 and October 4, 1991, the licensee requested Bases changes to Sections 3.1, 3.2, 3.3 and 4.3, 3.5.C, D & E, 3.9 and 3.10.

2.0 EVALUATION

The staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the Pilgrim reactor vessel. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Rev. 2. The staff has determined that the material with the highest ART at 32 EFPY was the lower intermediate shell axial weld (1-338A, B, and C) with 0.13% copper (Cu), 1.06% nickel (Ni), and an initial RT_{ndt} of -35°F.

The licensee has removed one surveillance capsule from Pilgrim. The results from capsule 1 were published in Southwest Research Institute Report SwRI 02-5951. All surveillance capsules contained Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

For the limiting beltline material, weld 1-338A, B, and C, the staff calculated the ART to be 90.9°F at 1/4T (T = reactor vessel beltline thickness) and 70.2°F for 3/4T at 32 EFPY. The staff used a neutron fluence of 9.8E17 n/cm^2 at 1/4T and 4.9E17 n/cmY2H at 3/4T. The ART was determined by Section 1 of RG 1.99, Rev. 2, because only one capsule was removed from the Pilgrim reactor pressure vessel.

The licensee used the method in RG 1.99, Rev. 2, to calculate an ART of 91°F at 32 EFPY at 1/4T for the same limiting weld metal. The staff judges that the licensee's ART of 91°F is more conservative than the staff's ART of 90.9°F, and it is acceptable. Substituting the ART of 91°F into equations in SRP 5.3.2, the staff verified that the proposed P/T limits for 32 EFPY for heatup, cooldown, and hydrotest meet the beltline material requirements in Appendix G of 10 CFR Part 50. The staff also verified that P/T limits for 10, 11, 13, 15, and 20 EFPYs meet the Appendix G requirements.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes P/T limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.A.2 of Appendix G states that when the pressure exceeds 20% of the pre-service system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. Paragraph IV.A.3 of Appendix G states "an exception may be made for boiling water reactor vessels when water level is within the normal range for power operation and the pressure is less than 20 percent of the pre-service system hydrostatic test pressure. In this case the minimum permissible temperature is 60°F (33°C) above the reference temperature of the closure flange regions that are highly stressed by the bolt preload." Based on the flange reference temperature of -5°F, the staff has determined that the proposed P/T limits for the 11, 13, 15, 20, and 32 EFPYs satisfy Section IV.A.2 of Appendix G.

In regard to the proposed bottom head P/T limits, the staff believes that because the bottom head of the reactor vessel does not receive significant amount of neutron fluence, embrittlement due to irradiation is not of major concern. The reference temperature calculations for the reactor beltline materials as prescribed in RG 1.99, Rev. 2 are not applicable to the bottom head P/T limits. The licensee calculated the stresses of the bottom head materials due to internal pressure, startup and cooldown transients, deadweight, and seismic loadings. The maximum stress locations are located at the junction between the lower torus and the support skirt and at control rod penetrations. From the maximum stresses, stress intensity factors and P/T limits were calculated based on ASME Code, Section III, Appendix G and 10 CFR 50, Appendix G. The staff finds that the licensee's calculation satisfy 10 CFR 50, Appendix G. However, to safeguard the structural integrity of the reactor beltline materials, the licensee must ensure that the pressure and temperature readings from the P/T sensors at the reactor vessel beltline region must be within the acceptable region of the beltline P/T limit curves when the bottom head P/T limits are being used during heatup and cooldown. The bottom head P/T limits must follow the same heatup and cooldown rate, 100 degrees F per hour, as that of the beltline P/T limits.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Commonwealth of Massachusetts State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (56 FR 31429). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 REFERENCES

1. Regulatory Guide 1.99, Radiation Embrittlement of Reactor Vessel Materials, Revision 2, May 1988

2. NUREG-0800, Standard Review Plan Section 5.3.2: Pressure-Temperature Limits

3. Letter from G. W. Davis (BECo) to USNRC Document Control Desk, Subject: Proposed Changes to the Reactor Pressure Vessel Thermal and Pressurization Technical Specification Limits, June 11, 1991

4. E. B. Norris, "Pilgrim Nuclear Power Station Unit 1 Reactor Vessel Irradiation Surveillance Program, SWRI 02-5951," July 1981

Principal Contributor: John C. Tsao

Date: January 29, 1992