Distribution
Docket
ORB #3
NRR Reading
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DEisenhut
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JHannon
SSheppard
Atty, OELD
OI&E (5)
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BScharf (10)
STSG
ACRS (16)
CMiles, OPA
RDiggs
HDenton
TERA
JRBuchanan

JANUARY 8 1980

Docket No. 50-293

Mr. G. Carl Andognini
M/C NUCLEAR
Boston Edison Company
800 Boylston Street
Boston, Massachusetts 02199

Dear Mr. Andognini:

The Commission has issued the enclosed Amendment No. 39 to Facility Operating License No. DPR-35 for the Pilgrim Nuclear Power Station Unit No. 1. This amendment consists of changes to the Technical Specifications in response to your requests dated March 10, 1977, December 28, 1977, November 13, 1979 and November 21, 1979.

This amendment (1) authorizes an increase in power level from 10% to 20% below which the Rod Worth Minimizer (RWM) must be operable, (2) revises the Technical Specifications to permit ascension to power within the envelope defined by a power/flow limit line, (3) permits the torus to be drained with up to a single control rod drive removed with irradiated fuel in the reactor vessel, and (4) revises the requirements for Source Range Monitor (SRM) minimum count rate during fuel movements.

During our review of the proposed Technical Specifications, we determined that certain changes to your requests were necessary to conform with NRC requirements. The changes were discussed with and agreed to by members of your staff.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Original signed by

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures and cc:
See page 2

8002070428

OFFICE	ORB #3	ORB #4	AD:ORP	OELD	ORB #3	
SURNAME	SSheppard	JHannon:mjf	WGammill	KARMAN	TIppolito	
DATE	1/8/80	1/8/80	1/8/80	1/8/80	1/8/80	

Enclosures:
1. Amendment No. 39 to DPR-35
2. Safety Evaluation
3. Notice

cc w/enclosures:
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Pilgrim Station Acting Manager
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Plymouth, Massachusetts 02360

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Massachusetts Department of Public Health
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

BOSTON EDISON COMPANY

DOCKET NO. 50-293

PILGRIM NUCLEAR POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 39
License No. DPR-35

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

   A. The applications for amendment by the Boston Edison Company (the licensee) dated March 10, 1977, December 28, 1977, November 13, 1979 and November 21, 1979, 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 applications, 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;

   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 License No. DPR-35 is hereby amended to read as follows:

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 39, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

8002070 435

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 8, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 39

FACILITY OPERATING LICENSE NO. DPR-35

DOCKET NO. 50-293

Revise Appendix A as follows:

Remove the following pages and insert identically numbered pages:

82
89B
104
111
112
119
152
152B
166
203
205
205B
205C-6

Insert the following new page:

205H

3.3.B Control Rods

2. The control rod drive housing support system shall be in place during reactor power operation and when the reactor coolant system is pressurized above atmospheric pressure with fuel in the reactor vessel, unless all control rods are fully inserted and Specification 3.3.A.1 is met.

3. a. No control rods shall be moved when the reactor is below 20% rated power, except to shutdown the reactor, unless the Rod Worth Minimizer (RWM) is operable. A maximum of two rods may be moved below 20% design power when the RWM is inoperable if all other rods except those which cannot be moved with control rod drive pressure are fully inserted.

   b. Control rod patterns and the sequence of withdrawal or insertion shall be established such that:

      1) when the reactor is critical and below 20% design power the maximum worth of any insequence control rod which is not electrically disarmed is less than 0.010 delta k.

      2) and when the reactor is above 20% design power the maximum worth of any control rod, including allowance for a single operator error, is less than 0.020 delta k.

4.3.B Control Rods

b. When the rod is fully withdrawan the first time subsequent to each refueling outage or after maintenance, observe that the drive does not go to the overtravel position.

2. The control rod drive housupport system shall be inspected after reassbmely and the results of the inspection recorded.

3. Prior to control rod withdrawal for startup or insertion to reduce power below 20% the operability of the Rod Worth Minimizer (RWM) shall be verified by:

   a. verifying the correctness of the control rod withdrawal sequence input to the RWM computer.

   b. performing the RWM computer diagnostic test

   c. verifying the annunciation of the selection errors of at least one out-of-sequence control rod in each distinct RWM group

   d. verifying the rod block function of an out-of-sequence control rod which is withdrawn no more than three notches.

BASES:

When THERMAL POWER is greater than 20% of RATED THERMAL POWER, there is no possible rod worth which, if dropped at the design rate of the velocity limiter, could result in a peak enthalpy of 280 cal/gm. Thus requiring the RWM to be OPERABLE when THERMAL POWER is less than or equal to 20% of RATED THERMAL POWER provides adequate control.

We are therefore requiring as a limiting condition of operation (LCO) that the Rod Worth Minimizer (RWM) be operable when the reactor is critical and below 20% of design power in accordance with Specification 3.3.B.3a so that the maximum in-sequence control rod worth will be limited to 0.010 delta k as given in Specification 3.3.B.3b(1) even assuming a single failure of the RWM or an operator error. The RWM assists and supplements the operator with an effective backup control rod monitoring routine that enforces adherence to pre-established startup, shutdown, and low power level control rod procedures. The RWM computer prevents the operator from establishing control rod patterns that are not consistent with prestored RWM sequences by initiating appropriate rod select block, rod withdrawal block, and rod insert block - interlock signals to the reactor manual control systems rod block circuitry. Reference: FSAR Section 7.16.4.3. The RWM sequences stored in the computer memory are based on control rod withdrawal procedures designed to limit the individual control rod worths to levels given in Specification 3.3.B.3.b.

Two exceptions to the requirement for RWM operability are permitted. Control rods may be moved to shutdown the reactor, and up to two control rods can be moved provided all other rods, except those which cannot be moved with control rod drive pressure, are inserted. The first exception permits the operator to shutdown the reactor in the event the RWM should become inoperable while the reactor is critical. In this case, the operator is moving the rods to reduce the reactivity in the core. Outward movement of any control rod is limited to a short adjustment and the general sequence of control rod movement is always toward a safer pattern during shutdown operations. The second exception permits the control rod drives to be moved when the RWM is inoperative provided that all but two rods are fully inserted except for those control rods which cannot be moved with control rod drive pressure.

3.5.A Core Spray and LPCI Subsystems (cont'd)

2. From and after the date that one of the core spray subsystems is made or found to be inoperable for any reason, continued reactor operation is permissible during the succeeding seven days, provided that during such seven days all active components of the other core spray subsystem and active components of the LPCI subsystem and the diesel generators are operable.

3. The LPCI Subsystems shall be operable whenever irradiated fuel is in the reactor vessel, and prior to reactor startup from a Cold Condition, except as specified in 3.5.A.4, 3.5.A.5 and 3.5.F.5.

4.5.A Core Spray and LPCI Subsystems (cont'd)

Check	Once/day
Calibrate	Once/3 months
Test	Once/3 months

2. When it is determined that one core spray subsystem is inoperable, the operable core spray subsystem, the LPCI subsystem and the diesel generators shall be demonstrated to be operable immediately. The operable core spray subsystem shall be demonstrated to be operable daily thereafter.

3. LPCI Subsystem Testing shall be as follows:

a.	Simulated Automatic Actuation Test	Once/Operating Cycle
b.	Pump Operability	Once/month
c.	Motor Operated valve operability	Once/month
d.	Pump Flow Rate	Once/3 months

Three LPCI pumps shall deliver 14,400 gpm against a system head corresponding to a vessel pressure of 20 psig.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.5.F <u>Minimum Low Pressure Cooling and Diesel Generator Availability (Cont'd)</u>

3. When irradiated fuel is in the reactor vessel and the reactor is in the Cold Shutdown Condition, both core spray systems, the LPCI and containment cooling subsystems may be inoperable, provided no work is being done which has the potential for draining the reactor vessel.

4. During a refueling outage, for a period of thirty days, refueling operation may continue provided that one core spray system or the LPCI system is operable or specification 3.5.F.5 is met.

5. When irradiated fuel is in the reactor vessel and the reactor is in the Refueling Condition with the torus drained, a single control rod drive mechanism may be removed, if both of the following conditions are satisfied:

   a) No work on the reactor vessel, in addition to CRD removal, will be performed which has the potential for exceeding the maximum leak rate from a single control blade seal if it became unseated.

   b) i) the core spray systems are operable and aligned with a suction path from the condensate storage tanks, ii) the condensate storage tanks shall contain at least 200,000 gallons of usable water and the refueling cavity and dryer/separator pool shall be flooded to at least elevation 114'-0".

3.5.G

(Intentionally left blank)

LIMITING CONDITION FOR OPERATION

3.5.H Maintenance of Filled Discharge Pipe

Whenever core spray subsystems, LPCI subsystem, HPCI, or RCIC are required to be operable, the discharge piping from the pump discharge of these systems to the last block valve shall be filled.

SURVEILLANCE REQUIREMENT

4.5.H Maintenance of Filled Discharge Pipe

The following surveillance requirements shall be adhered to to assure that the discharge piping of the core spray subsystems, LPCI subsystem, HPCI and RCIC are filled:

1. Every month prior to the testing of the LPCI subsystem and core spray subsystem, the discharge piping of these systems shall be vented from the high point and water flow observed.

2. Following any period where the LPCI subsystem or core spray subsystems have not been required to be operable, the discharge piping of the inoperable system shall be vented from the high point prior to the return of the system to service.

3. Whenever the HPCI or RCIC system is lined up to take suction from the torus, the discharge piping of the HPCI and RCIC shall be vented from the high point of the system and water flow observed on a monthly basis.

4. The pressure switches which monitor the discharge lines to ensure that they are full shall be functionally tested every month and calibrated every three months.

BASES:

3.5.F Minimum Low Pressure Cooling and Diesel Generator Availability

The purpose of Specification F is to assure that adequate core cooling equipment is available at all times. If, for example, one core spray were out of service and the diesel which powered the opposite core spray were out of service, only 2 LPCI pumps would be available. It is during refueling outages that major maintenance is performed and during such time that all low pressure core cooling systems may be out of service. This specification provides that should this occur, no work will be performed on the primary system which could lead to draining the vessel. This work would include work on certain control rod drive components and recirculation system. Specification F allows removal of one CRD mechanism while the torus is in a drained condition without compromising core cooling capability. The available core cooling capability for a potential draining of the reactor vessel while this work is performed is based on an estimated drain rate of 300 gpm if the control rod blade seal is unseated. Flooding the refuel cavity and dryer/separator pool to elevation 114' 0" corresponds to approximately 350,000 gallons of water and will provide core cooling capability in the event leakage from the control rod drive does occur. A potential draining of the reactor vessel (via control rod blade leakage) would allow this water to enter into the torus and after approximately 140,000 gallons have accumulated (needed to meet minimum NPSH requirements for the LPCI and/or core spray pumps), the torus would be able to serve as a common suction header. This would allow a closed loop operation of the LPCI system and the core spray system (once re-aligned) to the torus. In addition, the other core spray system is lined up to the condensate storage tanks which can supplement the refuel cavity and dryer/separator pool water to provide core flooding, if required.

Specification 3.9 must also be consulted to determine other requirements for the diesel generators.

LIMITING CONDITIONS FOR OPERATION

3.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the operating status of the primary and secondary containment systems.

Objective:

To assure the integrity of the primary and secondary containment systems.

Specification:

A. Primary Containment

1. At any time that the nuclear system is pressurized above atmospheric pressure or work is being done which has the potential to drain the vessel, the pressure suppression pool water volume and temperature shall be maintained within the following limits except as specified in 3.7.A.2 and 3.7.A.3.

   a. Minimum water volume - 84,000 ft^3

   b. Maximum water volume - 94,000 ft^3

   c. Maximum suppression pool temperature during normal continuous power operation shall be $\leq$ 80°F, except as specified in 3.7.A.1.e.

   d. Maximum suppression pool temperature during RCIC, HPCI or ADS operation shall be $\leq$ 90°F, except as specified in 3.7.A.1.e.

   e. In order to continue reactor power operation, the suppression chamber pool temperature must be reduced to $\leq$80°F within 24 hours.

   f. If the suppression pool temperature exceeds the limits of Specification 3.7.A.1.d, RCIC, HPCI or ADS testing shall be terminated and suppression pool cooling shall be initiated.

   g. If the suppression pool temperature during reactor power operation exceeds 110°F, the reactor shall be scrammed.

SURVEILLANCE REQUIREMENTS

4.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the primary and secondary containment integrity.

Objective:

To verify the integrity of the primary and secondary containment.

Specification:

A. Primary Containment

1. a. The suppression chamber water level and temperature shall be checked once per day.

   b. Whenever there is indication of relief valve operation or testing which adds heat to the suppression pool, the pool temperature shall be continually monitored and also observed and logged every 5 minutes until the heat addition is terminated.

   c. Whenever there is indication of relief valve operation with the temperature of the suppression pool reaching 160°F or more and the primary coolant system pressure greater than 200 psig, an external visual examination of the suppression chamber shall be conducted before resuming power operation.

   d. A visual inspection of the suppression chamber interior, including water line regions, shall be made at each major refueling outage.

LIMITING CONDITIONS FOR OPERATION

2. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing "open vessel" physics tests at power levels not to exceed 5 Mw(t).

3. The suppression chamber can be drained if the conditions as specified in Sections 3.5.F.3 and 3.5.F.5 of this Technical Specification are adhered to.

SURVEILLANCE REQUIREMENTS

2. Integrated Leak Rate Testing

   a. The primary containment integrity shall be demonstrated by performing an Integrated Primary Containment Leak Test (IPCLT) in accordance with either Method A or Method B, as follows:

      Method A

      Perform leak rate test prior to initial unit operation at the test pressure 45 psig, P_t (45), to obtain measured leak rate L_m (45), or

      Method B

      Perform leak rate test prior to initial unit operation at the test pressure of 45 psig, P_t (45), and 23 psig, P_t (23), to obtain the measured leak rates, L_m (45) and L_m (23), respectively.

BASES:

3.7.A & 4.7.A Primary Containment

The integrity of the primary containment and operation of the core standby cooling system in combination limit the off-site doses to values less than those suggested in 10 CFR 100 in the event of a break in the primary system piping. Thus, containment integrity is specified whenever the potential for violation of the primary reactor system integrity exists. Concern about such a violation exists whenever the reactor is critical and above atmospheric pressure. An exception is made to this requirement during initial core loading and while the low power test program is being conducted and ready access to the reactor vessel is required. There will be no pressure on the system at this time, thus greatly reducing the chances of a pipe break. The reactor may be taken critical during this period; however, restrictive operating procedures will be in effect again to minimize the probability of an accident occurring. Procedures and the Rod Worth Minimizer would limit control worth such that a rod drop would not result in any fuel damage. In addition, in the unlikely event that an excursion did occur, the reactor building and standby gas treatment system, which shall be operational during this time, offer a sufficient barrier to keep off-site doses well below 10 CFR 100 limits.

The pressure suppression pool water provides the heat sink for the reactor primary system energy release following a postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat released during primary system blowdown from 1035 psig. Since all of the gases in the drywell are purged into the pressure supression chamber air space during a loss-of-coolant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 62 psig, the suppression chamber maximum pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

Using the minimum or maximum water volumes given in the specification, containment pressure during the design basis accident is approximately 45 psig which is below the maximum of 62 psig. Maximum water volume of 94,000 ft^3 results in a downcomer submergency of 4'9" and the minimum volume of 84,000 ft^3 results in a submergence approximately 12-inches less. The majority of the Bodega tests were run with a submerged length of 4 feet and with complete condensation. Thus, with respect to downcomer submergency, this specification is adequate.

Should it be necessary to drain the suppression chamber, provision will be made to maintain those requirements as described in Section 3.5.F BASES of this Tehcnical Specification.

Experimental data indicates that excessive steam condensing loads can be avoided if the peak temperature of the pressure suppression pool is maintained below 160°F during any period of relief-valve operation with sonic conditions at the discharge exit. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high pressure suppression chamber loadings.

Exhibit A

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
2. The SRM shall have a minimum of 3 cps except as specified in 3 and 4 below. 3. Prior to spiral unloading, the SRM's shall have an initial count rate of $\geq$ 3 cps. During spiral unloading, the count rate on the SRM's may drop below 3 cps. 4. During spiral reload, each control cell shall have at least one assembly with a minimum exposure of 1000 MWD/t.	c. Spiral Reload 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 loaded, the cps requirement is not necessary.

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 assures adequate monitoring of that quadrant during such alterations. The requirement of 3 counts per second provides assurance that neutron flux is being monitored and insures that 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 (Appendix G) and a functional analysis of the neutron monitoring system. The specification is based on the Operational Nuclear Safety Requirements in subsection 7.5.10 of the Safety Analysis Report.

A spiral unloading pattern is one by which the fuel 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. Spiral loading is the reverse of unloading. 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.

C. Spent Fuel Pool Water Level

To assure that there is adequate water to shield and cool the irradiated fuel assemblies stored in the pool, a minimum pool water level is established. The minimum water level of 33 feet is established because it would be a significant change from the normal level (-1 foot) and is well above the level to assure adequate cooling.

4.10 BASES

A. Refueling Interlocks

Complete functional testing of all refueling interlocks before any refueling outage will provide positive indication that the interlocks operate in the situations for which they were designed. By loading each hoist with a weight equal to the fuel assembly, positioning the refueling platform, and withdrawing control rods, the interlocks can be subjected to valid operational tests. Where redundancy is provided in the logic circuitry, tests can be performed to assure that each redundant logic element can independently perform its functions.

B. Core Monitoring

Requiring the SRM's to be functionally tested prior to any core alteration assures that the SRM's will be operable at the start of that alteration. The daily response check of the SRM's ensures their continued operability.

C. Minimum Critical Power Ratio (MCPR)

During power operation MCPR shall be $\geq$ 1.31 for 8x8 fuel. If any time during operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

For core flows other than rated the MCPR shall be $\geq$ 1.31 for 8x8 fuel times K_f, where K_f is as shown in Figure 3.11-8.

As an alternative method providing equivalent thermal-hydraulic protection at core flows other than rated, the calculated MCPR may be divided by K_f, where K_f is as shown in Figure 3.11-8.

C. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined daily during reactor power operation at $>$ 25% rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3B.5.

D. Power/Flow Relationship During Power Operation

The power/flow relationship shall not exceed the limiting values shown in Figure 3.11-9. If at any time during power operation it is determined by normal surveillance that the limiting value for the power-flow relationship is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the power/flow relationship is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

D. Power/Flow Relationship During Power Operation

Compliance with the power/flow relationship in Section 3.11.D shall be determined daily during reactor operation.

The K_f factors shown in Figure 3.11-8 [5] are conservative for the Pilgrim Unit 1 operation because the operating limit MCPR given in Specification 3.11C is greater than the original 1.20 operating limit MCPR used for the generic derivation of K_f.

4.11C MINIMUM CRITICAL POWER RATIO (MCPR) - SURVEILLANCE REQUIREMENT

At core thermal power levels less than or equal to 25%, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. During initial start-up testing of the plant, a MCPR evaluation will be made at 25% thermal power level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR above 25% rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at a thermal limit.

Power/Flow Relationship Bases

The power/flow curve is the locus of core thermal power as a function of flow from which the occurrence of abnormal operating transients will yield results within defined plant safety limits. Each transient and postulated accident applicable to operation of the plant was analyzed along the power/flow line. The analysis justifies the operating envelope bounded by the power/flow curve as long as other operating limits are satisfied. Operation under the power/flow line is designed to enable the direct ascension to full power within the design basis for the plant.

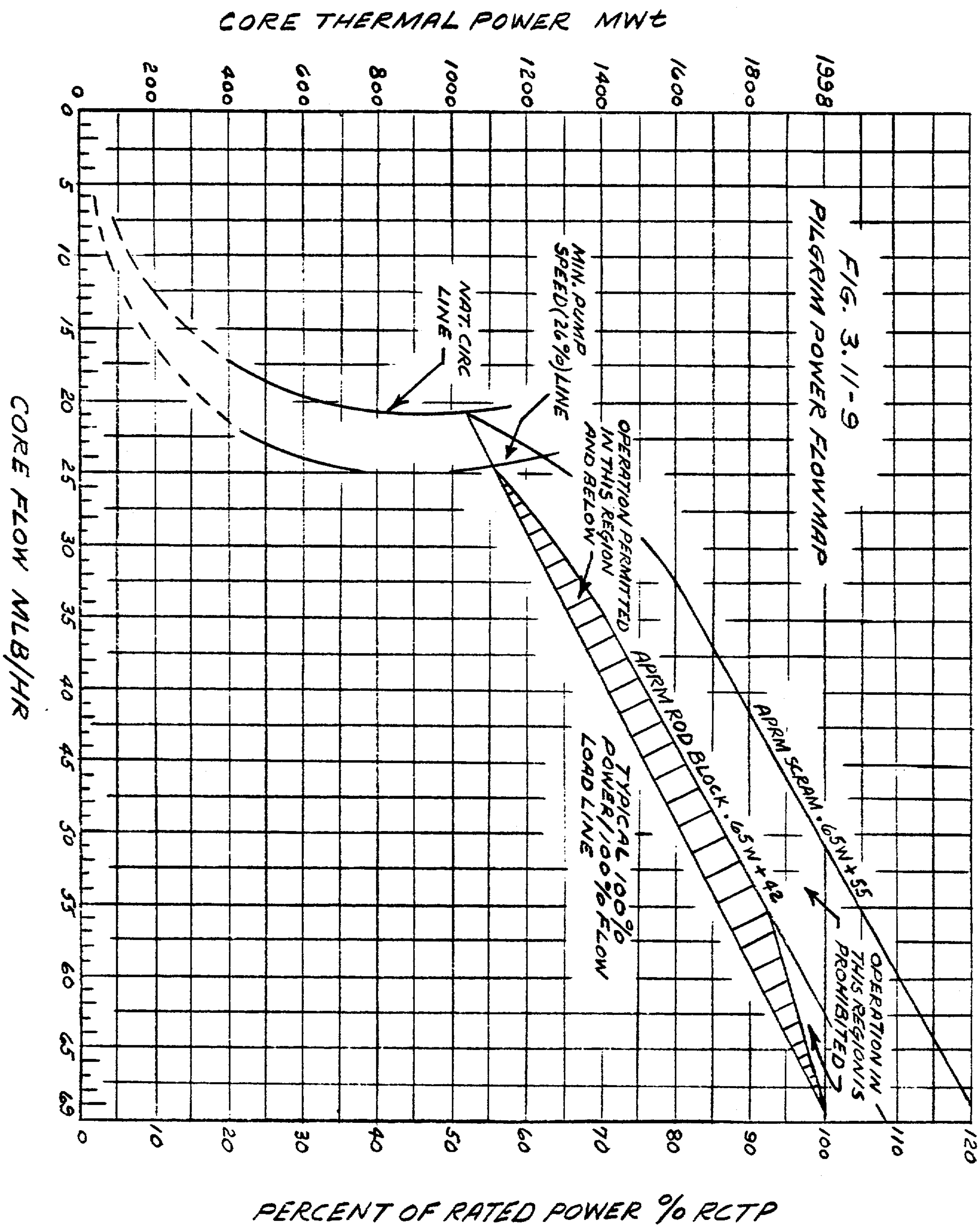

FIG. 3.11-9
PILGRIM POWER FLOW MAP

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 39 TO FACILITY OPERATING LICENSE NO. DPR-35

BOSTON EDISON COMPANY

PILGRIM NUCLEAR POWER STATION, UNIT NO. 1

DOCKET NO. 50-293

I. Power Level for Operability of Rod Worth Minimizer (RWM)

1.0 INTRODUCTION

By application dated March 10, 1977 Boston Edison Company (BECo or licensee) proposed a change to the Pilgrim Unit 1 Technical Specification (TS) 3.3.B.3 to increase the power level, from 10% to 20%, below which the RWM must be operable.

2.0 EVALUATION

The RWM restricts control rod selection to a preprogrammed pattern or order. Use of the RWM prevents the selection of high reactivity worth control rods which, if upon withdrawal, become upcoupled and subsequently dropped out of the reactor core, could cause fuel damage. The basis for requiring RWM operability up to 10% of rated thermal power was an analysis[1] which indicated that above 10% power, even single operator errors cannot result in a dropped rod accident which could cause fuel damage, because of the lower rod worth when significant moderator voiding is present.

In the Fall of 1976 it was found that more recent analyses[2] referenced by BECo in support of their Reload No. 3 application indicated that the RWM is assumed to be operable below 20% of rated thermal power to prevent fuel damage as a result of the postulated dropped rod accident. BECo subsequently committed to the use of the RWM during reactor operation below 20% power and has administratively imposed this requirement, pending TS revision.

3.0 CONCLUSION

The proposed change is acceptable since it is consistent with the assumptions adopted in BECo's Reload No. 3 application which was approved on October 17, 1977 as part of License Amendment No. 27. The language of the TS Bases supporting this revision was modified to conform to the GE-STS[3].

II. Power/Flow Operating Map

1.0 INTRODUCTION

By letter dated December 28, 1977 BECo provided the staff with technical justification for operation limited by a rod block intercept line at power and flow conditions greater than the nominal 100% power/flow control line and less than the current rod block line. Operation in this manner provides additional flexibility for power ascension while still complying with procedures to reduce pellet-clad interaction (PCIOMR's). The licensee believed that operation within the extended envelope is compatible with the current Technical Specifications. BECo, therefore, intended to make use of the added flexibility during power ascension methods and the effect of this change on abnormal operational transients has been considered for the Pilgrim Nuclear Power Station (PNPS).

2.0 EVALUATION

All reactor safety analyses are based on power and flow constraints. These power/flow conditions are such that even with the occurrence of an abnormal operating transient, the core will be operated within safety limits. BECo has provided the results of analyses and sensitivity studies to demonstrate that these criteria were met. These are discussed below.

2.1 Transients

As shown in Reference 4, the three most limiting abnormal operational transients for PNPS are Turbine Trip Without Bypass (TTWOB), Loss of Feedwater Heater, and Rod Withdrawal Error (RWE). The transient analyses and sensitivity studies for the proposed change were performed with the same input parameters as those for the Reload 3 analyses. Because the end-of-cycle 4 (EOC4) scram reactivity insertion function is the most limiting condition, this curve was used for all analyses. Each transient was analyzed at power/flow conditions of 100%/100%, 91%/75%, and 85%/61% (RWE was analyzed at only the first 2 points) to provide verification of transient behavior along the rod block intercept line to the point of rated power and flow. At the rated power/flow point the resultant transient behavior is the same as the previous analysis because the trip and rod block functions were not changed. The ΔCPR derived at the two lower values of power/flow are less than the ΔCPR for rated conditions for all transients except RWE.

2.1.1 Rod Withdrawal Error

Since it is not apparent the RWE will not be the limiting transient at lower power levels along the rod block intercept line, the RWE was anlayzed along the rod block intercept line at the 91% power/75% flow, and the 100%/100% point. At the 91%/75% point the RWE results in a ΔCPR that is the same as the value at the 100%/100% point. At points along the rod block intercept line between the 91%/75% point and the rod block intercept point

(85% power/61% flow) the ΔCPR of the RWE may increase. However, similar analyses [7,8] have shown that any increase in the ΔCPR along this portion of the rod block intercept line can be accommodated by the K_f factor.

The K_f factor is normally used to provide margin for flow increase transients and will be at least 1.065 along the rod intercept line. The product of 1.065 and the MCPR operating limit is high enough to more than compensate for the potential increase in ΔCPR at the 85%/61% point. For example, a conservative increase of 0.02 ΔCPR can be estimated for the rod block intercept point (85%/61%), and the corresponding compensation due to K_f would be about 0.08 in increased initial CPR. This compensation has been previously found acceptable (Reference 5) and is applicable for PNPS.

The APRM rod block setpoint is selected to allow for failed instruments for the worst allowable power profile. It is demonstrated that even if the operator ignores all alarms during the course of this transient, the rod blocks will stop rod withdrawal when the CPR is 1.06 (the CPR safety limit). At powers and flows lower than the 85%/61% condition within the proposed operating envelope, a RWE results in smaller ΔCPR values. The use of the present K_f factors limits the control rod positions such that the resulting MCPR's are conservative and bound the ΔCPR due to a RWE. The consequences of the RWE transient decrease at lower flows and the effective MCPR required by the use of the K_f values become increasingly conservative. Thus, the MCPR of an RWE from the rod block intercept line or the rod block line will be greater than that for the RWE from rated conditions. The analyses presented by BECo, and previous similar analyses reviewed and approved by the staff, show that this mode of operation is acceptable for all future cycles when transients that affect the entire core are considered (e.g., turbine trip without bypass, loss of feedwater heater). However, transients such as the RWE which affect only local portions of the core appear to be sensitive to the particular core configuration. Since the RWE may be the limiting transient at power levels below 100% rated along the rod block intercept line, similar analysis and justification should be provided for future cycles, if this new method of power ascension is to be used.

2.1.2 Peak Pressure Margin (25 psi Below Lowest Set Safety Value)

An analysis of the transient which involves main steam line isolation valve (MSIV) closure with high flux scram is used to evaluate compliance with the ASME pressure vessel code. The GE design criteria for adequacy of the safety valve capacity is a 25 psi margin between the peak vessel pressure and the ASME Boiler and Pressure Vessel code limit of 1375 psig based on a postulated MSIV closure transient with an indirect scram. BECo presented the results of analyses of the postulated transient, which showed the peak vessel bottom pressure at the rod block intercept point (85%/61%) is 1281 psig, 32 psi below that for the 100%/100% point and 94 psi below the code limit. At the intermediate point (91%/75%) on the rod block intercept line, the peak vessel bottom pressure is 1290 psig, also less than the 100%/100% point.

2.1.3 Operating MCPR Limits for Less Than Rated Power and Flow

A statistical analysis was performed to determine the part-load safety limit MCPR requirements along the APRM rod block line (Reference 6). The results of the analysis show a small increase (<.01 at the rod block intercept point) in the safety limit MCPR requirement for part-load conditions due to increase in uncertainty of flow measurement. However, this small increase in the part-load safety limit is more than compensated for by K_f factor-based operating MCPR limit for part-load conditions. (For PNPS, the operating MCPR increases by about 0.08 due to the K_f factor, compared to a 0.01 decrease due to flow measurement uncertainty.) The K_f factor is determined such that any inadvertent increase in core flow results in a MCPR greater than or equal to the safety limit MCPR at 100% power. Therefore, the safety limit will be satisfied when the plant is operating in accordance with the new power-flow operating envelope.

2.1.4 Xenon Transients

The typical FSAR transient analyses assume equilibrium xenon conditions. To investigate the change in MCPR from non-equilibrium conditions, a localized xenon transient is simulated. On the basis of actual rod swap observations, a reactivity swing from equilibrium to peak of -0.05 $\Delta k/k$ was conservatively established to model the effect of xenon. This reactivity swing is input as a local reactivity change to the BWR simulator and associated change in the MCPR is calculated. This analysis is performed for both the rods-in and rods-out core configuration. This conservative, xenon induced, reactivity change through the extremes of rod configurations resulted in a ΔCPR of -0/09.[7] Thus, the safety limit will not be violated for any potential xenon transient.

2.2 Thermal-Hydraulic Stability Analysis

Thermal-hydraulic stability analyses are presented in Reference 9. Previously in order to eliminate staff concerns on this topic PNPS precluded natural circulation operation. With this elimination of natural circulation as a normal mode of operation the results of the stability analysis are acceptable (decay ratio $\leq$0.5).

2.3 Accidents

The staff has reviewed the FSAR accident analysis for PNPS and agrees with the licensee that the change in power ascension methods and conditions will not significantly affect the consequences or probabilities of any accident sequence.

3.0 TECHNICAL SPECIFICATIONS

In order to verify that the reactor is maintained within the analyzed bounds, the staff recommends Technical Specifications similar to those provided for Millstone Nuclear Power Station Unit No. 1 by Amendment No. 52 to DPR-21 dated July 11, 1978. The licensee has agreed to such Technical Specifications.

4.0 CONCLUSIONS

PNPS has shown that the change in power ascension methods do not significantly affect the consequences for any transient or accident previously analyzed and accepted by the NRC. On this basis we find this proposal acceptable with the provisions that (1) a Technical Specification be added to limit the power/flow conditions to within the analyzed limits as suggested in the text, and (2) a RWE analysis must be performed or appropriate justification provided for future cycles which ensures that the change in critical power ratio (ΔCPR) for a RWE from conditions on the rod block intercept line does not exceed the ΔCPR for an RWE from 100% power/100% flow conditions.

III. Core Spary and Containment Cooling System

1.0 INTRODUCTION

By letter dated November 13, 1979, BECo proposed changes to the Technical Specifications for PNPS. The proposed changes would allow the suppression chamber (torus) to be drained and removal of up to one control rod drive mechanism (CRDM) without compromising core cooling capability. The change is required to permit modifications to be installed in the torus that will bring it into conformance with the Mark 1 Containment Long Term Program acceptance criteria and is similar to a previous change approved for Millstone Unit No. 1[1]. The capability to remove a CRD mechanism with the torus drained will permit CRD maintenance and Mark 1 modifications to proceed in parallel, thus reducing the overall outage duration.

The core spray (CS) system is one of the low pressure emergency core cooling systems (ECCS) along with the low pressure coolant injection (LPCI) system, which protects the core in case of a loss-of-coolant accident (LOCA) when the reactor pressure is low. Draining the torus will remove the normal source of water for both CS pumps and the residual heat removal (RHR) pumps when in the LPCI mode. The Containment Cooling System (Suppression Pool Cooling) will also be deactivated with no water in the torus. The proposal includes permitting a single control rod to be withdrawn and its CRDM to be removed while the torus is drained. In this situation, if the seal on the velocity limiter were to fail, a leak from the bottom of the core at a rate as high as 300 gpm could develop. It is therefore necessary that an alternate source of water be available if CS/LPCI is required. The licensee proposed the condensate storage tanks and the refueling cavity and dryer/separator pool as alternate sources of water. To supply this water to the core, both core spray systems must be available when work is being done that has the potential for draining the reactor vessel.

2.0 EVALUATION

In reviewing the proposed new TS 3.7.A.3 and 3.5.F.5, we have considered the possible ways in which water could be lost from the reactor vessel. These include (1) leakage past the seal on the velocity limiter of the control rod, (2) inadvertent operation of valves or pumps in such a way that water flows from the core, or (3) the break of a line connected to the vessel.

The first case, draining of the reactor vessel at a rate up to 300 gpm if the control rod blade seal is unseated, is well within the capability of the core spray systems, assuming a single failure.

The second mechanism for a loss of reactor cooling water is a possible error in which a pump is started or a valve is opened such that there is a decrease in the amount of available water to protect the core. The licensee has addressed this mechanism and has concluded that sufficient controls are in effect to preclude the possibility of this mechanism existing when no work is being done which has the potential for draining the reactor vessel. We concur. Thus, no requirements are placed on the operability of CS and Containment Cooling Systems with the torus drained, unless work is being done which has the potential for draining the reactor vessel. The licensee has analyzed this mechanism for the case when work is being done and concluded that the rate of coolant loss by inadvertent operation of any single pump or valve would be less than the leakage past the control rod flow limiting seal. Therefore, the core would be protected against a loss of water of this magnitude when work is being performed which has the potential for draining the reactor vessel.

The third mechanism for loss of reactor water is a pipe break. Because the system is not pressurized in the refueling mode, the probability of a significant break is negligibly low and was not considered further.

A total capacity of 224,000 gallons of water is available above the reactor vessel in the refueling cavity and another 165,000 gallons are available in the dryer/separator pool. The requirement of a minimum level at 114 feet elevation will assure at least 350,000 gallons of water are available. With no action, a leak from the reactor vessel to the drywell would result in flooding approximately 120,000 gallons into the drywell, at which point the torus would begin to fill by flow into the vent pipes from the drywell. At a leak rate of 300 gpm, sufficient water will accumulate in the torus to provide minimum NPSH requirements for the LPCI and/or core spray pumps after about 14 hours, assuming the torus is intact. We have reviewed the list of modifications to be performed in the torus,[10] and conclude that the capability to store at least 140,000 gallons (minimum NPSH) will not be degraded. Thus, torus integrity will be adequate for this function.

The postulated leak could continue unabated for at least 20 hours before water level would drop below the upper edge of the reactor vessel.

Therefore, there is sufficient time to take other emergency measures, if necessary. In addition, there is a low water alarm for the refueling cavity and automatic activation of the available low pressure ECCS upon a low-low water level in the reactor vessel.

Although no credit is given for the condensate storage tanks (CTS) in the above analysis, TS 3.5.F.5.b will require at least 200,000 gallons of useable water in the CST aligned with a suction path to the CS system. The CST has a low level alarm to alert the operator in time to take corrective action. The CST will be available for ECCS with the torus drained.

3.0 CONCLUSION

The proposed Technical Specifications will provide adequate assurance that the core will remain covered in the refueling mode with the torus drained. Protection is provided for (1) a leak of water past a velocity limiter on a control rod during CRDM maintenance, (2) a break in the reactor system piping, and (3) an inadvertent error in the opening or closing of a valve or starting a pump in such a way that water is lost from the reactor core. The protection is provided by assuring that adequate sources of water and methods of supplying this water are available. With the torus drained, the primary source is the refueling cavity and dryer/separator pool water and the backup source is the CST. This water would be supplied by the Core Spray System or LPCI.

IV. Minimum SRM Count Rate Requirements

1.0 INTRODUCTION

By letter dated November 21, 1979, Boston Edison Company proposed an amendment to the Technical Specifications for the Pilgrim Nuclear Power Station. The effect of the amendment would be to allow the count rate in the Source Range Monitor (SRM) channels to drop below 3 counts per second (cps) when the entire reactor core is being removed or replaced. The present Technical Specifications require that a count rate of at least 3 cps be maintained whenever one or more fuel assemblies are present in the core.

2.0 DISCUSSION

During any core alteration, and especially during core loading, it is necessary to monitor flux levels. In this manner, even in the highly unlikely event of multiple operator errors, there is reasonable assurance that any approach to criticality would be detected in time to halt operations.

The minimum count rate requirement in the Technical Specifications accomplishes three safety functions: (1) it assures the presence of some neutrons in the core, (2) it provides assurance that the analog portion of the SRM channels is operable, and (3) it provides assurance that the SRM detectors are close enough to the array of fuel assemblies to monitor core flux levels.

Unloading and reloading of the entire core leads to some difficulty with this minimum count rate requirement. When only a small number of assemblies are present within the core, the SRM count rate will drop below the minimum due to the small number of neutrons being produced, and due to attenuation of these neutrons in the water and control blades separating the fuel from the SRM detectors. Past practice has been to connect temporary "dunking" chambers to the SRM channels in place of the normal detectors, and to locate these detectors near the fuel.

Besides being operationally inconvenient, dunking chambers suffer from signal variations due to their lack of fixed geometry. Moreover, the use of dunking chambers increases the risk of loose objects being dropped into the vessel.

3.0 EVALUATION

3.1 Minimum Flux in the Core

A multiplying medium with no neutrons present forms the basis for an accident scenario in which reactivity is gradually but inadvertently added until the medium is highly supercritical. No neutron flux will be evident since there are no neutrons present to be multiplied. The introduction of some neutrons at this point would cause the core to undergo a sudden power burst, rather than a gradual startup, with no warning from the nuclear instrumentation.

This scenario is of great concern when loading fresh fuel, but is of lesser concern for exposed fuel. Exposed fuel continuously produces neutrons by spontaneous fission of certain plutonium isotopes, photofission and photodisintegration of deutrium in the moderator. This neutron production in exposed fuel is normally great enough to meet the 3 cps minimum for a full core after a refueling outage with the lumped neutron sources removed.

Thus, there is assurance that a minimum flux level will be present as long as some exposed fuel is present. We therefore find the proposed amendment to be acceptable from the point of view of minimum flux provided the words "spiral reload" in proposed specification 3.10.B.4, pg. 203, are interpreted to mean "reload of fuel which has previously accumulated exposure in the reactor." We do not find the amendment to be applicable to the loading of a new core containing only fresh fuel. Such a loading must use lumped neutron sources and dunking chambers to meet the normal 3 cps minimum count rate.

With the agreement of the licensee, we have therefore modified TS 3.10.B.4 to read: "During spiral reload, each control cell shall have one assembly with a minimum exposure of 1000 MWD/t."[11]

3.2 SRM Operability

Specification 4.10.B requires a functional check of the SRM channels, including a check of neutron response, prior to making any alteration to the core and daily thereafter. This would be sufficient for core unloading and reloading, except that the more extensive fuel handling operations involved imply a greater possiblity of SRM failure. During spiral unloading and reloading, Proposed Specification 3.10.B.4 would increase this frequency to every 12 hours or, as an alternative, allow some exposed fuel to be loaded adjacent to the SRM detectors to provide a minimum 3 cps count rate continuously. We agree that this increased testing is sufficient.

3.3 Flux Attenuation

The four SRM detectors are located, one per quadrant, roughly half a core radius from the center. Although these are incore detectors and thus very sensitive when the reactor is fully loaded, they lose some of their effectiveness when the reactor is partially defueled and the detectors are located some distance from the array of remaining fuel.

GE's spent fuel pool studies have shown[12] that 16 or more fuel assemblies (i.e., four or more control cells) must be loaded together before criticality is possible. In spiral loading sequences in the Pilgrim core, an array containing four or more control cells will be at most two control cells (i.e., about two feet) away from an SRM detector. We have previously examined the sensitivity loss in such a case on another docket,[13] and found it to be at most one decade of sensitivity (i.e., about one fifth of the SRM's logarithmic scale). As in Reference 13, we found this to be acceptable.

4.0 SUMMARY AND CONCLUSION

At this point, we have examined all these safety issues and found the proposed amendment to be acceptable provided it is understood that spiral reload will include a significant quantity of exposed fuel. With the change described in 3.1 above, we find this amendment to be acceptable.

ENVIRONMENTAL CONSIDERATION

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR Section 51.5(d)(4) that an environmental impact statement, negative declaration, or environmental impact appraisal need not be prepared in connection with the issuance of the amendment.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: January 8, 1980

References

1. NEDO-10527 "Rod Drop Accident Analysis for Large Boiling Water Reactors" March 1972.

2. Supplement 4 to NEDO-20360 Revision 1 General Electric BWR Generic Reload Application for 8x8 Fuel, April 1, 1976.

3. General Electric Standard Technical Specifications NUREG-0123 Revision 2 dated August 1979.

4. General Electric Boiling Water Reactor Reload No. 3 Licensing Submittal for Pilgrim Nuclear Power Station Unit 1, May 1977 (NEDO-21462-01).

5. Memo to K. Goller from V. Stello, "Review of Millstone Unit 1 Reload 3 TAR 1775," dated October 30, 1975.

6. Final Safety Analysis Report, Millstone Nuclear Power Station Unit 1, Docket No. 50-245, License No. DPR-21.

7. Millstone Point Nuclear Power Station, Unit 1, Load Line Limit Analysis License Amendment Submittal, June 1976 (NEDO-21285).

8. "Nine Mile Point Nuclear Power Station Unit 1, Load Limit Line Analysis," NEDO-24012.

9. NEDO-24058 "Pilgrim Nuclear Power Station Load Line Limit Analysis" September 1977.

10. BECo letter #79-92 dated May 9, 1979 regarding Schedule for the Implementation and Resolution of the Mark I Containment Long Term Program.

11. Amendment No. 46 to DPR-21 for Millstone Nuclear Power Station Unit No. 1 dated March 10, 1978.

12. General Electric Standard Safety Analysis Report, 251-GESSAR, Section 4.3.2.7, pg. 4.3-27.

13. "Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 27 to Facility Operating License No. DPR-63," Docket No. 50-220, enclosed with letter, T. A. Ippolito (NRC) to D. P. Dise (Niagara Mohawk Power Corporation), dated March 2, 1979.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-293

BOSTON EDISON COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 39 to Facility Operating License No. DPR-35, issued to Boston Edison Company (the licensee), which revised the Technical Specifications for operation of the Pilgrim Nuclear Power Station Unit No. 1 (the facility) located near Plymouth, Massachusetts. The amendment is effective as of its date of issuance.

This amendment (1) authorizes an increase in power level from 10% to 20% below which the Rod Worth Minimizer (RWM) must be operable, (2) revises the Technical Specifications to permit ascension to power within the envelope defined by a power/flow limit line, (3) permits the torus to be drained with up to a single control rod drive removed with irradiated fuel in the reactor vessel, and (4) revises the requirements for Source Range Monitor (SRM) minimum count rate during fuel movements.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR Section 51.5(d)(4), an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of the amendment.

For further details with respect to this action, see (1) the applications for amendment dated March 10, 1977, December 28, 1977, November 13 and 21, 1979, (2) Amendment No. 39 to License No. DPR-35, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C., and at the Plymouth Public Library on North Street in Plymouth, Massachusetts 02360. A single copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland this 8th day of January 1980

FOR THE NUCLEAR REGULATORY COMMISSION

TA Ippolito

Thomas A. Ippolito, Chief
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Division of Operating Reactors