Entergy

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**Mike Bellamy**
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August 16 , 2002

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555-0001

SUBJECT: Entergy Nuclear Operations, Inc.
Pilgrim Nuclear Power Station
Docket No. 50-293
License No. DPR-35

Request for Amendment to the Technical Specifications
Relocation of Certain Control Rod Block Requirements to the Updated Final Safety Analysis Report

REFERENCE: NUREG 1433, Standard Technical Specifications for General Electric Plants, BWR/4.

LETTER NUMBER: 2.02.065

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Entergy Nuclear Operations Inc. (Entergy) hereby proposes to amend the Pilgrim Station Facility Operating License, DPR-35. This proposed license amendment would relocate certain Control Rod Block functions from Technical Specification 3/4.2.C, "Instrumentation that Initiates Rod Blocks," to the Updated Final Safety Analysis Report. This change is consistent with Standard Technical Specifications (NUREG 1433, Revision 2) and changes previously approved by the NRC for other reactor licensees. Pilgrim has reviewed the proposed amendment in accordance with 10 CFR 50.92 and concludes it does not involve a significant hazards consideration.

Entergy requests approval of the proposed amendment by March 31, 2003 to support Pilgrim's upcoming refueling outage (scheduled to commence on April 19, 2003). Once approved, the amendment will be implemented within 60 days.

A001

If you have any questions or require additional information, please contact Bryan Ford at (508) 830-8403.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 16th day of August 2002.

Sincerely,

Richard M. Bellamy

Enclosure: Evaluation of the Proposed Changes – 6 pages

Attachments: 1. Proposed Technical Specification and Bases Changes (mark-up) – 10 pages

2. List of Regulatory Commitments – 1 page

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ENCLOSURE

Evaluation Of The Proposed Changes

Subject: Relocation of Certain Control Rod Block Functions to the Final Safety Analysis Report

1. Description

This letter is a request to amend Operating License DPR-35 for Pilgrim Nuclear Power Station. This proposed license amendment would relocate certain Control Rod Block functions from Technical Specifications (TS) 3/4.2.C, "Instrumentation that Initiates Rod Blocks," to the Updated Final Safety Analysis Report (UFSAR).

This change is consistent with Standard Technical Specifications (NUREG 1433, Revision 2) and changes previously approved by the NRC for other reactor licensees. Entergy has reviewed the proposed amendment in accordance with 10 CFR 50.92 and concludes it does not involve a significant hazards consideration. Entergy requests approval of the proposed amendment by March 31, 2003 to support Pilgrim's upcoming refueling outage (scheduled to commence on April 19, 2003).

2. Proposed Change

The proposed change is to relocate the following Control Rod Block functions from TS 3/4.2.C to the UFSAR.

A. Average Power Range Monitors (APRMs)

1. APRM Upscale (Flow Biased)
2. APRM Upscale
3. APRM Inoperative
4. APRM Downscale

B. Intermediate Range Monitors (IRMs)

1. IRM Upscale
2. IRM Detector not in Startup Position
3. IRM Inoperative
4. IRM Downscale

C. Source Range Monitors (SRMs)

1. SRM Upscale
2. SRM Detector not in Startup Position
3. SRM Inoperative
4. SRM Downscale

D. Scram Discharge Volume Water Level

1. Scram Discharge Instrument Volume Water Level – High
2. Scram Discharge Instrument Volume – Scram Trip Bypassed

E. Recirculation Flow Converter

1. Recirculation Flow Converter – Upscale
2. Recirculation Flow Converter – Inoperative
3. Recirculation Flow Converter – Comparator Mismatch

All associated current TS requirements and TS Bases information will be relocated to the UFSAR as part of this change. This change is consistent with Standard TS, General Electric Plants, BWR/4, (NUREG-1433, Revision 2) and changes previously approved by the NRC for other reactor licensees. Also included in Attachment 1 are the associated Bases changes that will be made as part of implementation of the proposed changes.

3. Background

Control rods provide the primary means for control of reactivity changes. The control rod block instrumentation is designed to ensure that specified fuel design limits are not exceeded for postulated transients and accidents.

During high power operation, the rod block monitor provides protection for control rod withdrawal error events. During low power operations, control rod blocks from the rod worth minimizer enforce specific control rod sequences designed to mitigate the consequences of the control rod drop accident. During shutdown conditions, control rod blocks from the Reactor Mode Switch in Shutdown Position Function ensure that all control rods remain inserted to prevent inadvertent criticalities.

The purpose of the rod block monitor is to limit control rod withdrawal if localized neutron flux exceeds a predetermined setpoint during control rod manipulations. It is assumed to function to block further control rod withdrawal to preclude a MCPR Safety Limit violation. The rod block monitor supplies a trip signal to the reactor manual control system (RMCS) to appropriately inhibit control rod withdrawal during power operation above the low power range setpoint. The rod block monitor has two channels, either of which can initiate a control rod block when the channel output exceeds the control rod block setpoint. One rod block monitor channel inputs into one rod block circuit and the other channel inputs into the second rod block circuit. The rod block monitor channel signal is generated by averaging a set of local power range monitor (LPRM) signals at various core heights surrounding the control rod being withdrawn. A signal from one APRM channel assigned to each Reactor Protection System (RPS) trip system supplies a reference signal for the rod block monitor channel in the same trip system. This reference signal is used to determine which rod block monitor range setpoint (low, intermediate, or high) is enabled. The requirements associated with the rod block monitor function will continue to be controlled by TS 3/4.2.C and are not modified by the proposed TS changes.

The purpose of the rod worth minimizer is to control rod patterns during startup, such that only specified control rod sequences and relative positions are allowed over the operating range from all control rods inserted to 20% rated thermal power. The sequences effectively limit the potential amount and rate of reactivity increase during a control rod drop accident. Prescribed control rod sequences are stored in the rod worth minimizer, which functions to initiate control rod withdrawal and insert blocks when the actual sequence deviates beyond allowances from the stored sequence. The rod worth minimizer determines the actual sequence based on position indication for each control rod. The requirements associated with the rod worth minimizer function will continue to be controlled by TS 3/4.3.F and are not modified by the proposed TS changes.

With the reactor mode switch in the shutdown position, a control rod withdrawal block is applied to all control rods to ensure that the shutdown condition is maintained. This prevents inadvertent criticality as the result of a control rod withdrawal when the reactor is shutown. The reactor mode switch has two channels, each inputting into a separate RMCS rod block circuit. A rod block in either RMCS circuit will provide a control rod block to all control rods. The requirements associated with the rod block monitor function will continue to be controlled by TS 3/4.2.C and are not modified by the proposed TS changes.

In addition to these functions assumed in the accident analysis, control rod block instrumentation also provides information to the operators to help prevent an unnecessary RPS automatic actuation (scram trip signal) and also provides indication of reactor power and core flow conditions and other information. The control rod block instrumentation provides this function by blocking control rod movement when specific conditions occur. These functions (e.g., APRM, IRM, SRM, Scram Discharge Volume Water Level, and Recirculation Flow Converter control rod blocks) are the functions that are to be relocated from the TS to the UFSAR.

The APRMs provide information about average core power. This average core power information is used by the control rod block instrumentation to generate a control rod block prior to the RPS actuation setpoint and in the case of the Flow Biased rod block, provide indication to the reactor operator concerning the boundary of the restricted region for reactor core stability purposes. The APRM rod blocks are not capable of providing the local power information required to mitigate rod withdrawal errors or rod drop accidents and are not credited to mitigate any design basis accident or transient.

The IRMs and SRMs are provided to monitor the neutron flux levels during shutdown, refueling, and startup conditions. This information is used by the control rod block instrumentation to generate a control rod block prior to the RPS actuation setpoint. The IRM and SRM rod blocks are not credited to mitigate any design basis accident or transient.

The purpose of measuring the Scram Discharge Volume Water Level is to ensure there is sufficient volume remaining to contain the water discharged by the CRDs during a scram, thus ensuring the control rods are able to insert fully. This rod block signal provides an indication to the operator that water is accumulating in the Scram Discharge Volume and prevents further rod withdrawals. With continued water accumulation, an RPS automatic scram will occur. Thus, the Scram Discharge Volume water level control rod block signal provides an opportunity for the operator to take action to avoid a subsequent scram. The Scram Discharge Volume Water Level rod blocks are not credited to mitigate any design basis accident or transient.

The Recirculation Flow Converter control rod blocks function to monitor for failure of the Recirculation Flow Converter. Failure(s) of a Recirculation Flow Converter can result in an increase or a mismatch in reactor recirculation flow. An increase in reactor recirculation flow causes an increase in neutron flux that results in an increase in reactor power. This increase in neutron flux is monitored by the APRM RPS instrumentation while flow mismatches are controlled by TS 3.6.F. The Recirculation Flow Converter rod blocks are not credited to mitigate design basis accidents or transients.

Relocating these requirements to the UFSAR will allow revisions in accordance with 10 CFR 50.59 without requiring a license amendment. Any change of the relocated specifications in the UFSAR will be controlled in accordance with the provisions of 10 CFR 50.59.

4. Technical Analysis

Section 182a of the Atomic Energy Act of 1954, as amended (the Act) requires applicants for nuclear power plant operating licenses to include the TSs as part of the license. The Commission's regulatory requirements related to the content for the TSs are set forth in 10 CFR 50.36. That regulation requires the TSs include items in eight specific categories. The categories are (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; (5) administrative controls; (6) decommissioning; (7) initial notification; and (8) written reports. However, the regulation does not specify the particular requirements to be included in a plant's TSs.

The Commission amended 10 CFR 50.36 (60 FR 36593, July 19, 1995), and codified four criteria to be used in determining whether a particular matter is required to be included in a limiting condition for operation (LCO) as follows: (1) Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary; (2) a process variable, design feature, or operating restriction that is an initial condition of a design-basis accident or transient analysis that either assumes that failure of, or presents a challenge to, the integrity of a fission product barrier; (3) a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design-basis accident or transient that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier; or (4) a structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety. LCOs and related requirements that fall within or satisfy any of the criteria in the regulation must be retained in the TSs, while those requirements that do not fall within or satisfy these criteria may be relocated to licensee-controlled documents. Pilgrim's UFSAR is one such licensee-controlled document.

The proposed changes are consistent with the Standard TS for General Electric plants (NUREG-1433) and 10 CFR 50.36. NUREG-1433 does not include the control rod block functions that are requested to be removed from the TS. In addition, the criteria in 10 CFR 50.36 for features required to be retained in TSs do not apply. The NRC's Final Policy Statement recommends that TSs that do not meet the screening criteria for retention may be relocated to a licensee-controlled document. The four criteria of 10 CFR 50.36 are addressed below:

(1) These control rod block functions do not have the ability to detect abnormal degradation of the reactor coolant pressure boundary. Therefore, these control rod block functions do not satisfy Criterion 1.

(2) These control rod block functions provide information to the operators to help prevent unnecessary reactor protection system automatic actuations (scrams) and indication of reactor power and core flow conditions. However, these control rod block functions are not explicitly considered in the accident analysis and are not considered a required initial condition for a design basis accident or transient. Therefore, these control rod block functions do not satisfy Criterion 2.

(3) These control rod block functions provide information to the operators to help prevent unnecessary reactor protection system automatic actuations (scrams) and indication of reactor power and core flow conditions. However, These control rod block functions are not explicitly considered in the accident analysis. These control rod block functions are not a primary success path for accident mitigation; therefore they do not satisfy Criterion 3.

(4) Operating experiences or probabilistic safety assessments have not shown these control rod block functions to be significant to public health and safety. Therefore, These control rod block functions do not satisfy Criterion 4.

These control rod block functions will be relocated to the UFSAR. Any changes to these requirements will be strictly controlled under the provisions of 10 CFR 50.59. Therefore, the relocation of these control rod block functions specifications from the TSs to the UFSAR will continue to provide adequate assurance that functionality and testing of the control rod blocks will be assured as necessary.

In conclusion, the relocated requirements are not required to be in the TS under 10 CFR 50.36 or section 182a of the Atomic Energy Act, and are not required to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety. In addition, sufficient regulatory controls exist under 10 CFR 50.59 to assure continued protection of public health and safety.

5. Regulatory Safety Analysis

5.1 No Significant Hazards Consideration

Pilgrim is proposing to relocate certain Control Rod Block functions from TS 3/4.2.C, "Instrumentation that Initiates Rod Blocks," to the Updated Final Safety Analysis Report (UFSAR).

Pilgrim has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change is administrative in nature and does not involve the modification of any plant equipment or affect basic plant operation. These control rod blocks are not assumed to be an initiator of any analyzed event, nor are they assumed in the mitigation of consequences of accidents. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed change does not involve any physical alteration of plant equipment and does not change the method by which any safety-related system performs its function. As such, no new or different types of equipment will be installed, and the basic operation of installed equipment is unchanged. The methods governing plant operation and testing remain consistent with current safety analysis assumptions. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   The proposed change is administrative in nature, does not negate any existing requirement, and does not adversely affect existing plant safety margins or the reliability of the equipment assumed to operate in the safety analysis. As such, there are no changes being made to safety analysis assumptions, safety limits or safety system settings that would adversely affect plant safety as a result of the proposed change. Margins of safety are unaffected by requirements that are retained, but relocated from the Technical Specifications to the FSAR. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, Entergy concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

6. Environmental Consideration

As defined in 10 CFR 20, a review of this TS change determined that the proposed amendment would change a requirement in respect to installation or use of a facility component located within the restricted area, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the proposed amendment.

7. Coordination With Other Pending TS Changes

The mark-ups of TS submitted by Letter Number 2.02.072 for "Changes to Post-Accident Monitoring Instrumentation Requirements", deletes Note (2) on TS page 3/4.2-41. If the reference TS change is approved prior to the approval of this TS change, the final TS page 3/4.2-41 should be consolidated to delete Note (2), in addition to the deletion of Note (3).

8. References


1. 10 CFR 50.36, "Technical Specifications"

2. NUREG-1433,"Standard Technical Specifications, General Electric Plants, BWR/4"

# ATTACHMENT 1

# PROPOSED TECHNICAL SPECIFICATIONS AND BASES CHANGES (MARK-UP)

PNPS
TABLE 3.2.C.1

## INSTRUMENTATION THAT INITIATES ROD BLOCKS

| Trip Function | Operable Channels per Trip Function: Minimum | Operable Channels per Trip Function: Available | Required Operational Conditions | Notes |
|---|---|---|---|---|
| APRM Upscale (Flow Biased) | 4 | 6 | Run | (1) |
| APRM Upscale | 4 | 6 | Startup/Refuel | (1)(6) |
| APRM Inoperative | 4 | 6 | Run/Startup/Refuel | (1)(6) |
| APRM Downscale | 4 | 6 | Run | (1) |
| Rod Block Monitor(Power Dependent) | 2 | 2 | Run, with limiting control rod pattern, and reactor power > LPSP | (2)(5) |
| Rod Block Monitor Inoperative | 2 | 2 | Run, with limiting control rod pattern, and reactor power > LPSP | (2)(5) |
| Rod Block Monitor Downscale | 2 | 2 | Run, with limiting control rod pattern, and reactor power > LPSP | (2)(5) |
| IRM Downscale | 6 | 8 | Startup/Refuel, except trip is bypassed when IRM is on its lowest range | (1)(6) |
| IRM Detector not in Startup Position | 6 | 8 | Startup/Refuel, trip is bypassed when mode switch is placed in run | (1)(6) |
| IRM Upscale | 6 | 8 | Startup/Refuel | (1)(6) |
| IRM Inoperative | 6 | 8 | Startup/Refuel | (1)(6) |

TABLE 3.2.C.1 (Cont)

INSTRUMENTATION THAT INITIATES ROD BLOCKS

| Trip Function | Operable Channels per Trip Function: Minimum | Operable Channels per Trip Function: Available | Required Operational Conditions | Notes |
|---|---|---|---|---|
| SRM Detector not in Startup Position | 3 | 4 | Startup/Refuel, except trip is bypassed when SRM count rate is ≥ 100 counts/second or IRMs on Range 3 or above | (1)(4)(6) |
| SRM Downscale | 3 | 4 | Startup/Refuel, except trip is bypassed when IRMs on Range 3 or above | (1)(4)(6) |
| SRM Upscale | 3 | 4 | Startup/Refuel, except trip is bypassed when the IRM range switches are on Range 8 or above (4) | (1)(4)(6) |
| SRM Inoperative | 3 | 4 | Startup/Refuel, except trip is bypassed when the IRM range switches are on Range 8 or above (4) | (1)(4)(6) |
| Scram Discharge Instrument Volume Water Level - High | 2 | 2 | Run/Startup/Refuel | (3)(6) |
| Scram Discharge Instrument Volume-Scram Trip Bypassed | 1 | 1 | Refuel/Shutdown | (3)(6) |

PNPS
TABLE 3.2.C.1 (Cont)

INSTRUMENTATION THAT INITIATES ROD BLOCKS

| Trip Function | Operable Channels per Trip Function: Minimum | Operable Channels per Trip Function: Available | Required Operational Conditions | Notes |
|---|---|---|---|---|
| Recirculation Flow Converter - Upscale | 2 | 2 | Run | (1) |
| Recirculation Flow Converter - Inoperative | 2 | 2 | Run | (1) |
| Recirculation Flow Converter - Comparator Mismatch | 2 | 2 | Run | (1) |
| Reactor Mode Switch in Shutdown | 2 | 2 | Shutdown | (7) |

NOTES FOR TABLE 3.2.C-1

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1. With the number of operable channels:

   a. One less than required by the minimum operable channels per trip function requirement, restore an inoperable channel to operable status within 7 days or place an inoperable channel in the tripped condition within the next hour.

   b. Two or more less than required by the minimum operable channels per trip function requirement, place at least one inoperable channel in the tripped condition within one hour.

2. a. With one RBM Channel inoperable:

   (1) restore the inoperable RBM channel to operable status within 24 hours; otherwise place one rod block monitor channel in the tripped condition within the next hour, and;

   (2) prior to control rod withdrawal, perform an instrument function test of the operable RBM channel.

   b. With both RBM channels inoperable, place at least one inoperable rod block monitor channel in the tripped condition within one hour.

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3. If the number of operable channels is less than required by the minimum operable channels per trip function requirement, place the inoperable channel in the tripped condition within one hour.

4. SRM operability requirements during core alterations are given in Technical Specification 3.10.

5. RBM operability is required in the run mode in the presence of a limiting rod pattern with reactor power greater than the RBM low power setpoint (LPSP). A limiting rod pattern exists when:

   MCPR < 1.41 for reactor power ≥ 90%

   MCPR < 1.72 for reactor power < 90%

   The allowable value for the LPSP is ≤ 29% of rated core thermal power.

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6. When the reactor mode switch is in the Refuel position, the reactor vessel head is removed, and control rods are inserted in all core cells containing one or more fuel assemblies, these Rod Block functions are not required.

7. With one or more Reactor Mode Switch - Shutdown Position channels inoperable, suspend control rod withdrawal and initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies immediately.

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TABLE 3.2.C-2
CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

| Trip Function | Trip Setpoint |
|---|---|
| APRM Upscale | (1) (2) |
| APRM Inoperative | Not Applicable |
| APRM Downscale | ≥ 2.5 Indicated on Scale |
| Rod Block Monitor (Power Dependent) | (1) (3) |
| Rod Block Monitor Inoperative | Not Applicable |
| Rod Block Monitor Downscale | (1) (3) |
| IRM Downscale | ≥ 5/125 of Full Scale |
| IRM Detector not in Startup Position | Not Applicable |
| IRM Upscale | ≤ 108/125 of Full Scale |
| IRM Inoperative | Not Applicable |
| SRM Detector not in Startup Position | Not Applicable |
| SRM Downscale | ≥ 3 counts/second |
| SRM Upscale | ≤ $10^5$ counts/second |
| SRM Inoperative | Not Applicable |
| Scram Discharge Instrument Volume Water Level - High | ≤ 17 gallons |
| Scram Discharge Instrument Volume - Scram Trip Bypassed | Not Applicable |
| Recirculation Flow Converter - Upscale | ≤ 120/125 of Full Scale |
| Recirculation Flow Converter - Inoperative | Not Applicable |
| Recirculation Flow Converter - Comparator Mismatch | ≤ 8% Flow Deviation |
| Mode Switch in Shutdown | Not Applicable |

(1) The trip level setting shall be as specified in the CORE OPERATING LIMITS REPORT.

(2) ~~When the reactor mode switch is in the refuel or startup positions, the APRM rod block trip setpoint shall be less than or equal to 13% of rated thermal power, but always less than the APRM flux scram trip setting.~~ Deleted

(3) The RBM bypass time delay ($t_{d2}$) shall be < 2.0 seconds.

PNPS
TABLE 4.2.C

## MINIMUM TEST AND CALIBRATION FREQUENCY FOR CONTROL ROD BLOCKS ACTUATION

| Instrument Channel | Instrument Functional Test | Calibration | Instrument Check |
|---|---|---|---|
| APRM - Downscale | Once/3 Months | Once/3 Months | Once/Day |
| APRM - Upscale | Once/3 Months | Once/3 Months | Once/Day |
| APRM - Inoperative | Once/3 Months | Not Applicable | Once/Day |
| IRM - Upscale | (2) (3) | Startup or Control Shutdown | (2) |
| IRM - Downscale | (2) (3) | Startup or Control Shutdown | (2) |
| IRM - Inoperative | (2) (3) | Not Applicable | (2) |
| RBM - Upscale | Once/3 Months | Once/6 Months | Once/Day |
| RBM - Downscale | Once/3 Months | Once/6 Months | Once/Day |
| RBM - Inoperative | Once/3 Months | Not Applicable | Once/Day |
| SRM - Upscale | (2) (3) | Startup or Control Shutdown | (2) |
| SRM - Inoperative | (2) (3) | Not Applicable | (2) |
| SRM - Detector Not In Startup Position | (2) (3) | Not Applicable | (2) |
| SRM - Downscale | (2) (3) | Startup or Control Shutdown | (2) |
| IRM - Detector Not In Startup Position | (2) (3) | Not Applicable | (2) |
| Scram Discharge Instrument Volume Water Level-High | Once/3 Months | Refuel | Not Applicable |
| Scram Discharge Instrument Volume-Scram Trip Bypassed | Once/Operating Cycle | Not Applicable | Not Applicable |
| Recirculation Flow Converter | Not Applicable | Once/Operating Cycle | Once/Day |
| Recirculation Flow Converter-Upscale | Once/3 Months | Once/3 Months | Once/Day |
| Recirculation Flow Converter-Inoperative | Once/3 Months | Not Applicable | Once/Day |
| Recirculation Flow Converter-Comparator Off Limits | Once/3 Months | Once/3 Months | Once/Day |
| Recirculation Flow Process Instruments | Not Applicable | Once/Operating Cycle | Once/Day |
| Mode Switch in Shutdown | Once/Operating Cycle | Not Applicable | Not Applicable |
| **Logic System Functional Test (4) (6)** | | | |
| System Logic Check | Once/Operating Cycle | | |

NOTES FOR TABLES 4.2.A THROUGH 4.2.G

1. Initially once per month until exposure hours (M as defined on Figure 4.1.1) is $2.0 \times 10^5$; thereafter, according to Figure 4.1.1 with an interval not les than one month nor more than three months.

2. Functional tests, calibrations and instrument checks are not required when these instruments are not required to be operable or are tripped. Functiona tests shall be performed before each startup with a required frequency not t exceed once per week. Calibrations of IRMs and SRMs shall be performed during each startup or during controlled shutdowns with a required frequency not to exceed once per week. Instrument checks shall be performed at least once per day during those periods when the instruments are required to be operable.

3. ~~This instrumentation is excepted from~~ the functional test definition. The functional test will consist of injecting a simulated ~~electrical~~ signal intc the measurement channel.

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4. Simulated automatic actuation shall be performed once each operating cycle. Where possible, all logic system functional tests will be performed using th test jacks.

5. Reactor low water level and high drywell pressure are not included on Table 4.2.A since they are tested on Tables 4.1.1 and 4.1.2.

6. The logic system functional tests shall include a calibration of time delay relays and timers necessary for proper functioning of the trip systems.

7. Calibration of analog trip units will be performed concurrent with functiona testing. The functional test will consist of injecting a simulated electrical signal into the measurement channel. Calibration of associated analog transmitters will be performed each refueling outage.

BASES:

3.2 PROTECTIVE INSTRUMENTATION (Cont)

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.

The control rod block functions are provided to prevent excessive control rod withdrawal. The trip logic for this function is 1 out of n: e.g., any trip on one of six APRM's, two ~~RMB's~~ RBM's, eight IRM's, or four SRM's will result in a rod block.

The minimum instrument channel requirements assure sufficient instrumentation to assure the single failure criteria is met. The minimum instrument channel requirements for the RBM may be reduced by one for not longer than 24 hours without significantly increasing the risk of an inadvertent control rod withdrawal.

Reactor power may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal beyond a given point, thereby possibly avoiding an APRM Scram. The rod block setpoint is automatically reduced with recirculation flow to form the upper boundary of the PNPS power/flow map. The flow biased APRM rod block is not necessary to prohibit fuel damage and is not included in the analysis of anticipated transients.

BASES:

3.2 PROTECTIVE INSTRUMENTATION (Cont)

The RBM rod block function provides local protection of the core, for a sing rod withdrawal error from a limiting control rod pattern.

The RBM bypass time delay ($t_{d2}$) is the delay between the time the signal is normalized to the reference signal and the time the signal is passed to the trip logic. Control rod withdrawal is unrestricted during this interval. Tl RBM bypass time delay is low enough to assure that control rod movement is minimized during the time RBM trips are bypassed.

The IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that trip setting is less than a factor of 1( above the indicated level.

A downscale indication on an APRM, RBM or IRM is an indication the instrument has failed or the instrument is not sensitive enough. In either case the instrument will not respond to changes in control rod motion and thus, contro rod motion is prevented. The downscale trips are as shown in Table 3.2.C-2.

The flow comparator and scram discharge volume high level components have onl one logic channel and are not required for safety.

The refueling interlocks also operate one logic channel, and are required for safety only when the mode switch is in the refueling position.

For effective emergency core cooling for small pipe breaks, the HPCI system must function since reactor pressure does not decrease rapidly enough to allo either core spray or LPCI to operate in time. The automatic pressure relief function is provided as a backup to the HPCI in the event the HPCI does not operate. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are adequate to assure the above criteria are met. The specification preserves the effectiveness of the system during periods of maintenance, testing or calibration, and also minimizes the risk of inadvertent operation; i.e., only one instrument channel out of service.

Four radiation monitors are provided which initiate the Reactor Building Isolation and Control System and operation of the standby gas treatment system. The instrument channels monitor the radiation from the refueling area ventilation exhaust ducts.

Four instrument channels are arranged in a 1 out of 2 twice trip logic.

BASES:

4.2 PROTECTIVE INSTRUMENTATION (Cont)

Control Rod Block and PCIS instrumentation common to RPS instrumentation have surveillance intervals and maintenance outage times selected in accordance with NEDC-30851P-A, Supplements 1 and 2 as approved by the NRC and documented in SERs (letters to D. N. Grace from C. E. Rossi dated September 22, 1988 and January 6, 1989).

A logic system functional test interval of 24 months was selected to minimize the frequency of safety system inoperability due to testing and to minimize the potential for inadvertent safety system trips and their attendant transients.

# ATTACHMENT 2

# LIST OF REGULATORY COMMITMENTS

## List of Regulatory Commitments

The following table identifies those actions committed to by Pilgrim in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

| REGULATORY COMMITMENT | DUE DATE |
|---|---|
| Relocate the control rod block requirements removed from the Technical Specifications and Technical Specifications Bases to FSAR. | 60 days following approval of the amendment. |