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10 CFR 50.90

5928-08-20132
September 29, 2008

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

Three Mile Island Nuclear Station, Unit 1
Facility Operating License No. DPR-50
NRC Docket No. 50-289

Subject: Technical Specification Change Request No. 342
Control Rod Drive Control System Upgrade and Elimination of the Axial Power Shaping Rods

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," AmerGen Energy Company, LLC (AmerGen) requests the following amendment to the Technical Specifications, Appendix A of Facility Operating License Nos. DPR-50 for Three Mile Island Nuclear Station, Unit 1 (TMI-1).

The proposed amendment would revise Technical Specification (TS) Sections 1.4, Protection Instrumentation Logic; 3.5, Instrumentation Systems; Table 4.1-1, Instrumentation Surveillance Requirements; and 4.7, Reactor Control Rod System Tests to reflect design changes resulting from the Control Rod Drive Control System (CRDCS) digital upgrade project. Concurrent with this digital CRDCS project, the proposed amendment would revise TS Sections 3.5, Instrumentation Systems; 4.7, Reactor Control Rod System Tests; and 5.3, Reactor to remove all references to the Axial Power Shaping Rods (APSRs) from the TMI-1 TSs to reflect changes resulting from the proposal to eliminate APSRs from the TMI-1 reactor.

AmerGen specifically seeks NRC approval for the reactor trip breaker (RTB) design change configuration and associated TSs and the deletion of TSs associated with the elimination of the APSRs. The replacement of the RTBs, although included in the overall modification, is not a digital upgrade. The digital upgrade involves the CRDCS only. There are no safety related setpoint changes associated with this proposal.

Attachment 1 provides the Evaluation of Proposed Changes. Attachment 2 provides the proposed Technical Specification marked-up pages. Attachment 3 provides the proposed Technical Specification Bases marked-up pages for information only. Attachment 4 provides a regulatory commitment for control rod patch verification.

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The proposed changes have been reviewed by the TMI-1 Plant Operations Review Committee and approved by the Nuclear Safety Review Board.

Using the standards in 10 CFR 50.92, AmerGen has concluded that these proposed changes do not constitute a significant hazards consideration, as described in the enclosed analysis performed in accordance with 10 CFR 50.91(a)(1). Pursuant to 10 CFR 50.91(b)(1), a copy of this Technical Specification Change Request is being provided to the designated official of the Commonwealth of Pennsylvania, as well as the chief executives of the township and county in which the facility is located.

This amendment is required to support the 2009 fall refueling outage. We request approval of the proposed amendment by September 29, 2009. The amendment will be implemented during the 2009 fall refueling outage.

A new regulatory commitment is established by this submittal and is contained in Attachment 4.

If any additional information is needed, please contact Frank Mascitelli at (610) 765-5512.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 29th day of September 2008.

Respectfully,

Pamela B. Cowan
Director - Licensing & Regulatory Affairs
AmerGen Energy Company, LLC

Attachments:
1) Three Mile Island Nuclear Station Unit 1 Technical Specification Change Request No. 342 - Evaluation of Proposed Changes
2) Three Mile Island Nuclear Station Unit 1 Proposed Technical Specification Marked-Up Pages
3) Three Mile Island Nuclear Station Unit 1 Proposed Technical Specification Bases Marked-Up Pages (For information only)
4) Summary of Regulatory Commitments

cc: S. J. Collins, Administrator, USNRC Region I
D. M. Kern, USNRC Senior Resident Inspector, TMI Unit 1
P. J. Bamford, USNRC Project Manager, TMI Unit 1
D. Allard, Director, Bureau of Radiation Protection-PA Department of Environmental Resources
Chairman, Board of County Commissioners of Dauphin County
Chairman, Board of Supervisors of Londonderry Township

ATTACHMENT 1

Three Mile Island Nuclear Station Unit 1
Technical Specification Change Request No. 342

Evaluation of Proposed Changes

Attachment 1
TMI-1 Evaluation of Proposed Changes for the Control Rod Drive Control System Upgrade and Elimination of Axial Power Shaping Rods

1.0 DESCRIPTION

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," AmerGen Energy Company, LLC (AmerGen) is requesting changes to Facility Operating License No. DPR-50 for Three Mile Island Nuclear Station, Unit 1 (TMI-1). The proposed amendment would revise Technical Specification (TS) Sections 1.4, Protection Instrumentation Logic; 3.5, Instrumentation Systems; Table 4.1-1, Instrument Surveillance Requirements; and 4.7, Reactor Control Rod System Tests to reflect design changes resulting from the Control Rod Drive Control System (CRDCS) digital upgrade project.

Concurrent with this digital CRDCS project, the proposed amendment would revise TS Sections 3.5, Instrumentation Systems; 4.7, Reactor Control Rod System Tests; and 5.3, Reactor to reflect changes resulting from the proposal to remove the Axial Power Shaping Rods (APSRs) from the TMI-1 reactor. The original design of the TMI-1 reactor included eight APSRs to allow operators to control the axial power profile in the core within limits specified in the TSs. However, APSRs have not been used at TMI-1 for axial imbalance control since 1994. Axial imbalance has been successfully maintained within core operating limits using regulating control rod groups alone.

Upon approval, AmerGen requests that the following changed replacement pages be inserted into the existing Facility Operating License:

Revised TMI-1 TS Pages: 1-3, 3-27a, 3-34, 3-35, 4-3, 4-5, 4-48, 5-4
Revised TMI-1 TS Bases Pages (for information only): 3-28, 3-35a, 3-36

2.0 PROPOSED CHANGES

TMI-1 Facility Operating License No. DPR-50, Appendix A, Technical Specifications:

Page 1-3 - Modify TS 1.4.4 to provide reactor trip signals for de-energizing four breakers instead of six control rod drive reactor trip breakers (RTBs).

Page 3-27a - Delete TS 3.5.1.8 and TS 3.5.1.8.1 requirements for Silicon-Controlled Rectifier (SCR) Electronic Trips.

Page 3-28 - Modify TS Bases to describe new logic employing four alternating current (AC) RTBs.

Page 3-34 - Delete TS 3.5.2.2.f reference to axial power shaping and TS 3.5.2.2.g reference to Group 8.

Page 3-35 - Modify TS 3.5.2.7.d to delete reference to APSRs and delete TS 3.5.2.6 requirement to lock patch panels.

Page 3-35a - Delete TS 3.5.2 Bases reference to APSRs.

Page 3-36 - Delete TS 3.5.2 Bases reference to APSRs.

Page 4-3 - Modify Table 4.1-1, Channel Description 2 to delete Regulating Rod Power SCRs.

Page 4-5 - Modify Table 4.1-1, Control Rod Absolute and Relative Positions to add a new surveillance for the Zone Reference Switches, delete the refuel calibration surveillance for Control Rod Relative Position and delete the word "Indicator" in the Remarks column for Control Rod Absolute and Relative Positions.

Page 4-48 - Modify TS 4.7.1.1 to delete reference to APSRs and modify TS 4.7.1.3 to allow use of Zone Reference Switches for locating a Control Rod.

Page 5-4 - Modify TS 5.3.1.4 to delete references to APSRs.

3.0 BACKGROUND

The existing TMI-1 CRDCS is being replaced with a digital CRDCS (DCRDCS). The existing relay-based CRDCS (refer to Figure 1 for a simplified diagram) is being upgraded to a redesigned system (refer to Figure 2 for a simplified diagram) to resolve obsolescence and age-related degradation issues. The RTBs are reconfigured to accommodate the new control system; however, the reactor trip function is independent of the digital system. The replacement of the RTBs is not considered a digital upgrade. The digital system provides Control Rod Position Indication but does not perform any nuclear safety related function.

The DCRDCS will require a TS change to allow implementation, specifically for: 1) replacement of the existing two alternating current (AC) RTBs and four existing direct current (DC) RTBs with four new AC RTBs; 2) deletion of the Electronic Trip Assemblies (ETAs) associated with the DC RTBs; and, 3) replacement of the patch panels with software control as the means to assign control rods to their groups. The TS changes described above are the only aspects of the DCRDCS project that require NRC approval prior to implementation. The DCRDCS design change is being evaluated under Engineering Change Request (ECR) TM 07-01037 (Reference 1).

<u>Description of the System Change</u>

The CRDCS will be replaced with a design that was developed by AREVA and installed at Oconee Nuclear Station. The significant differences between this submittal and the Oconee Nuclear Station submittal are: elimination of the APSRs; use of Zone Reference Switches for CRD position indication; removal of patch panel TS requirements. The AREVA DCRDCS design has greater than five years of reactor operation. The RTB configuration for the new system changes from two AC circuit breakers and four DC circuit breakers to four AC circuit breakers. Other changes that will be incorporated with CRDCS replacement are deletion of Group 8 APSR controls, and indications. The Flat Panel Position Indication Display will be installed on Control Room Panel PC replacing analog meters. The Group Average Meters will be deleted from the console and Group Average Position will be displayed on the PI Panel. Inverter backed In-Limit Light Emitting Diodes (LEDs) will be connected on the Position Indication Panel to provide In-Limit Indication independent of the digital system. The Diverse

Scram System (DSS) will be modified to trip the DCRDCS electronic trip instead of tripping 1G-2A and 1L-2A breakers that provide power to the Control Rod Drive (CRD) system.

Existing CRD System (Figure 1)

The Reactor Protection System (RPS) logic and the interface between the RPS and the CRDCS are described in TMI-1 Updated Final Safety Analysis Report (UFSAR) Section 7.1.2.2. Outputs from each of the four RPS reactor trip modules will trip an associated RTB. In the existing configuration of the CRD power supplies, power is supplied to all of the Control Rod Drive Mechanisms (CRDMs) from two redundant 480-volt AC sources. Each of the sources is connected through an AC RTB, and then either through diode banks and two parallel DC RTBs for safety rod Groups 1 through 4, or through SCR banks for the regulating Group rods. In the existing configuration, RPS Channel A trips CRD AC breaker A, RPS Channel B trips CRD AC breaker B, RPS Channel C trips CRD DC breaker C and SCR ETA E, and RPS Channel D trips CRD DC breaker D and SCR ETA F. Breakers A and B control all of the 3-phase primary power to the rod drives; breakers associated with Channels C and D control the DC power to rod Groups 1 through 4; and ETAs E and F control the gating power to the regulating rod groups.

Interruption of both redundant sources of power is required to release a CRDM and allow its associated control rod to fall into the reactor core. The combination of the AC RTBs in series with the DC RTBs and Electronic Trips provide a trip logic scheme described as one-out-of-two-times-two. Each element of the one-out-of-two-times-two logic is controlled by a separate set of two-out-of-four logic contacts from the four reactor protection channels. When any two out of four protection channels trip, all reactor trip module logics trip, commanding all RTBs to trip. In this scheme, no single breaker, ETA or SCR failure will prevent power being removed from all of the control rods and tripping the reactor.

The undervoltage (UV) coils of the RTBs receive their power from the protection channel associated with each breaker. The manual reactor trip switch is interposed in series between each Reactor Trip module logic and the assigned breaker's UV coil.

As a backup to the breaker's UV coil trip, the breaker's shunt trip coil is energized by action of a voltage-sensing relay, which operates when a trip is initiated via the RPS logic. The shunt trip coils are also actuated by the existing source interrupt device if a system power supply is outside allowed limits. Shunt trips are initiated for DC undervoltage and AC overvoltage conditions in the existing system.

The design of the existing CRDCS employs connector patch panels to align the individual CRDMs and their corresponding Control Rod position indications with the appropriate group power supplies. The cabinets containing these patch panels are locked and the keys administratively controlled to prevent any unauthorized realignment of CRDMs or Control Rod indication.

New CRD System (Figure 2)

The CRD logic and power supplies are being replaced to address age, obsolescence and reliability issues with the existing system. The replacement system is the same base system as that installed at Oconee Units 1 and 3 with changes to accommodate TMI-1 needs. Lessons learned from the Oconee experience have been incorporated into the design.

The DCRDCS utilizes Single Rod Power Supplies (SRPSs) instead of Group Power Supplies and a triple-modular-redundant digital system for control, logic and position indication.

Existing Diverse Scram System (DSS)

The DSS monitors Reactor Coolant System (RCS) pressure with two channels of instrumentation. When the DSS two-out-of-two logic is met, signals are sent to 1G-2A and 1L-2A breakers to remove power from the CRDCS resulting in a reactor trip.

Modified Diverse Scram System

Upon detection of high RCS pressure on both DSS channels, signals are sent to trip Electronic Trip E and F resulting in a reactor trip. The change to DSS does not affect the Technical Specifications and this information is provided for background.

Deletion of APSRs

Concurrent with the DCRDCS upgrade, AmerGen also requests deletion of the TS requirements associated with Group 8 APSRs. The evaluation to eliminate Group 8 APSRs has been completed and documented under ECR TM 07-01037 (Reference 2).

The original design of the TMI-1 reactor included eight APSRs to allow operators to control the axial power profile in the core within limits specified in TS. [Note: These axial power imbalance limits have since been relocated to the Core Operating Limits Report (COLR)]. However, APSRs have not been used at TMI-1 for axial imbalance control since 1994 (Cycle 10). APSRs have been located at an axial position that has neutral impact on axial power imbalance. This operating philosophy was developed based on internal operating experience that showed that APSRs, if not used properly, could exacerbate axial imbalance swings during a power transient. Axial imbalance has been successfully maintained within core operating limits using regulating control rod groups alone.

Furthermore, in the early 2000 timeframe, multiple fuel rod defects occurred in Babcock and Wilcox (B&W) units (including TMI-1 in 2003) due to pellet-clad-interaction (PCI) during End-of-Cycle (EOC) APSR withdrawal maneuvers (i.e., a typical EOC reactivity maneuver employed at B&W units to extend the full power capability of the core for an additional 7-8 days by withdrawing the APSR poison from the core). Therefore, starting with Cycle 16 in 2005, APSRs have been withdrawn from the core at TMI-1 at Beginning-of-Cycle (BOC) and have been parked in this position for the entire cycle with COLR limits preventing insertion.

4.0 TECHNICAL ANALYSIS

Scope

The proposed change is requested in connection with the replacement of the CRDCS. The CRDCS itself is not safety related, and prior NRC approval is not required for modifications to it. However, the CRDCS does interface with the RPS through the RTBs. NRC approval is required for changes to the RTB configuration. The new RTB configuration is required in order to properly interface with the new SRPSs.

Reactor Trip Breaker Replacement

The revised RTB configuration being installed as part of the DCRDCS modification will replace the existing two AC RTBs and four DC RTBs with four new AC RTBs. Breakers A and C are placed in series in one parallel power path, and breakers B and D are in series in the other parallel path. All of the 3-phase primary power to the rod drives is via these parallel paths. In the revised configuration, RPS Channel A trips CRD breaker A, Channel B trips breaker B, Channel C trips breaker C, and Channel D trips breaker D. The one-out-of-two-times-two trip logic is maintained. The new RTB configuration of four qualified breakers maintains the safety design basis of the CRD system in that upon receipt of the RPS signals the RTBs in the power feed circuits are tripped, which removes power upstream of the SRPSs. The new RTBs are fully qualified for this safety related application. Each RTB has diverse trip devices including an undervoltage device that is de-energizd to trip and a shunt trip device that is energized to trip. The RPS and RTBs are diverse from the DSS / Anticipated Transient Without Scram (ATWS) System. As with the existing system configuration, no single breaker failure will prevent a reactor trip when required or cause a reactor trip when not required. The one-out-of-two-times-two logic scheme described above will be maintained utilizing significantly less components.

The existing channel independence, separation, and performance requirements of the RTBs and the RPS system response times are retained.

Channel Independence

All CRDMs in the existing CRDCS configuration are powered from two separate (redundant) 480 VAC power sources (A and B trains) through two AC breakers. Downstream of the AC breakers the safety control rods are powered through four DC Breakers, two per train, and the regulating rods are powered through parallel SCR banks, one bank in each power train. The SCRs are controlled by gating signals supplied through ETA relay contacts.

The RPS design consists of four independent and redundant channels, with a Reactor Trip Module in each channel providing for interruption of voltage to the circuit breaker under-voltage coil and an under-voltage relay that activates the RTB shunt trip coil. In the current Reactor Trip configuration, RPS Channel A is connected to an AC breaker in train A, RPS Channel B is connected to an AC breaker in train B, RPS Channel C is connected to a parallel combination of two DC breakers and Electronic Trip Assemblies in train A, and RPS Channel D is connected to a parallel combination of two DC breakers and Electronic Trip Assemblies in train B.

In the revised reactor trip configuration, RPS Channel A trips CRD breaker A, RPS Channel B trips breaker B, RPS Channel C trips breaker C, and RPS Channel D trips breaker D. Thus, the four-channel power interruption configuration of the existing system is being maintained with each RPS channel associated with a power interruption device dedicated to that channel. The separation and channel independence of the RPS itself is maintained since the RPS is not being changed by this modification.

Separation

The cables associated with RPS Channel C and RPS Channel D are rerouted to the new RTB location in dedicated conduits to maintain electrical separation.

The four replacement RTBs will be housed in two cabinet assemblies, two RTBs in each cabinet assembly. Physical separation is maintained by mounting the breaker assemblies in individual cubicles in each of the two train related replacement RTB cabinets. Within the RTB cabinet, separation of wiring associated with the RPS channels is maintained in accordance with station engineering guidelines.

Common Mode Failure Review

The RTB design was reviewed for credible common mode failures and no credible common mode failures were identified that would prevent the breakers from performing the reactor trip function. Common cause failures that were considered were a seismic event that would result in a loss of geometry of the breaker cubicles. Failure to trip due to a seismic event is not considered credible due to the seismic qualification of the switchgear. Failure due to harsh environment is not considered to be credible since the RTBs are in a mild area not containing high-energy lines and not subject to high radiation. A third possible common cause failure would be as a result of Electromagnetic Interference (EMI) or Radio Frequency Interference (RFI) and this is not considered credible since the undervoltage trip device and the shunt trip device are low impedance devices and are not susceptible to failures from EMI/RFI.

No single failure of an RTB will prevent completion of the Reactor Trip function. Online testing on a rotational basis is intended to detect failures and the TS requires removal of power through any breaker found to be failed.

Performance Requirements

The new RTBs are seismically qualified, seismically installed, and located in an area not subject to harsh environments.

The new design is considered better than the existing design in that the new Square D Masterpact NT breakers are manufactured using a well developed and proven breaker design. These are next generation of Masterpact Power Circuit Breakers from Square D that employs the latest technology. The new breakers offer proven reliability with improved functionality. These breakers are used as RTBs in the United States. These new qualified breakers will be used to replace the original, obsolete, and increasingly difficult to maintain General Electric AK-15 and AK-25 breakers. Overall reliability of the system will be increased with the new equipment.

The new RTBs meet or exceed the performance requirements as evidenced in the table below.

Design Parameter	Requirement	Square D Masterpact NT
Rated Voltage	508 VAC Minimum	600 VAC
Rated Current	600 A Minimum	800 A
Interrupting Rating	22,000 A at 508 VAC	42,000 A at 508 VAC
Opening (Breaking) Time	80 msec maximum	50 msec
Temperature	40 deg F to 104 deg F	-22 deg F and 140 deg F
Humidity, Maximum	95%	95%
Seismically qualified	IEEE 323 and 344	IEEE 323 and 344
Life, Cycles between maintenance intervals	1200 cycles Minimum	2800 Cycles
Shunt Trip Voltage Range	100 VDC to 137 VDC	88 VDC to 137 VDC
Undervoltage Device Dropout	36 VAC Minimum	35% of 120 VAC (42 VAC)

<u>DCRDCS System</u>

The new RTB configuration will continue to provide the same design function as the existing configuration by removing power to the CRDMs to initiate a reactor trip. The new configuration will utilize qualified breakers to accomplish this function, and it will continue to meet the design function given a single failure. The revised configuration also uses fewer components to meet the original design requirements, and is therefore simpler than the existing design. Figure 2 shows the modified simplified diagram for the DCRDCS and RTB configuration.

The CRD logic and power supplies are being replaced to address age, obsolescence and reliability issues with the existing system. The replacement system is the same base system as that installed at Oconee Units 1 and 3 with changes to accommodate TMI-1 needs. Lessons learned from the Oconee experience have been incorporated into the design.

The DCRDCS utilizes SRPSs instead of Group Power Supplies and a triple modular redundant digital system for control, logic and position indication. Operator interface and operating procedures are simplified by eliminating transfers. Each rod can be controlled individually by selecting the rod. The SRPSs have redundant Power Modules that can be replaced on-line. The operator control panel is at the same location and is similar in appearance and function to the existing panel. Changes to the control panel relate to elimination of transfers, elimination of Group 8 and other minor changes.

The DCRDCS does not include power supplies, logic or position indication for Group 8 APSRs since the APSRs are no longer utilized in the core designs for TMI-1.

The DCRDCS does not perform a nuclear safety-related function and failure of the control system will not prevent the RPS from tripping the reactor. The Electronic Trip does not perform any TS required function or nuclear safety-related function. The function of the control system will not prevent the RPS from tripping the reactor. DCRDCS is designed so that no single failure will cause a reactor trip, cause out motion, or prevent in motion of the rods. A Failure Modes and Effects Analysis (FMEA) (Reference 3) demonstrates improved reliability of the DCRDCS control system compared to the existing system. Functions performed by the existing system were compared to the new system and differences were evaluated for acceptability.

The DCRDCS does not perform any safety function but does provide Position Indication that is required by TS. On a loss of offsite power, the reactor will trip and analog Position Indication will be lost because power is lost to the digital system. Inverter backed In-Limit LEDs are provided for each rod to provide indication to the operator that each rod has inserted. In addition, the existing inverter-backed Source and Intermediate Range Nuclear instruments remain available to the operators to verify that the reactor is shutdown.

The DCRDCS provides additional monitoring and alarms compared to the existing system. Absolute and Relative Position indications are compared on an ongoing basis and alarmed upon deviation between the indications for any rod. Power Supply voltages are monitored and alarmed.

The DCRDCS will monitor for system power supply faults and actuate the RTB shunt trip coils if a system power supply is outside allowed limits in place of the existing source interrupt device. DCRDCS will actuate the RTB shunt trip coils on DC undervoltage and AC overvoltage, and an additional trip initiator for AC undervoltage will be added. The addition of this trip initiator helps to ensure the CRDMs are de-energized to prevent damage to the leadscrews on a low voltage condition. With the additional trip feature, there is reasonable assurance that the overall risk of false tripping the reactor is lower with the DCRDCS based on elimination of a Single Point Vulnerability (SPV) and improved hardware. The existing source interrupt device was identified as an SPV that could result in a false reactor trip given a single hardware failure. The DCRDCS hardware that is utilized to detect a power supply fault condition is triple-modular-redundant, and therefore, no single failure will result in a false reactor trip.

The Flat Panel Position Indication Display replaces analog meters for Position Indication. The display provides analog indication (bar graph) for absolute position of each rod. The display provides numerical indication of Absolute Position and Relative Position for each rod simultaneously. With the existing system the operator can switch between absolute and relative position but cannot display the indications simultaneously. Group Average position for all seven groups is provided on the display and the analog meters on the control panel are eliminated.

The Position Indication display was evaluated for human factors and was presented to licensed operators. Asymmetric rod indication is indicated by a change in bar color. Zone Reference Switch indication is provided on the panel, and therefore, is available to the operator for verifying rod position. The Zone Reference Switch status was available in the control cabinet in the existing system and was not readily available to the operator. The operators concluded that the display improves readability and provides additional indication to the operator. The Position Indication display provided improved indication during dynamic plant conditions as well as steady state conditions.

The DSS was modified to trip the Electronic Trips in the DCRDCS. This change makes TMI-1 consistent with the Oconee Nuclear Station design. The change was made to improve reliability since some hardware issues were experienced with the non-safety related breakers utilized in the existing DSS system. Utilizing the Electronic Trips instead of the upstream breakers also avoided the need to find alternate power sources for DCRDCS control power. Control power for the existing and new CRD Control System is fed from the incoming sources to the RTBs.

Up to 12 Control Rods can be assigned to any of seven groups. Assignment in the existing system is controlled by Patch Panels. The TS requires that the panels be locked with access authorized by the Plant Manager. In the DCRDCS, changes to CRDM patching require modification of the system software and rewiring the 0% limit switches for station black out. Changes to CRDM patching will only be available when the system is offline and requires recompiling the application software. Changes to application software are administratively controlled, password protected, and will be verified by a test.

Electronic Trip

Electronic Trips E and F are utilized for equipment protection and for Diverse Scram in the new system. The new system utilizes a pair of Power Modules (A and B) making up a SRPS for each Control Rod rather than using group power supplies. The power supplies consist of 61 SRPSs, each with two identical A and B power supplies. Electronic Trip E turns off the Power Modules on the A side and Electronic Trip F turns off the Power Modules on the B side of the SRPS. When the DSS two-of-two logic is met, signals are sent to DCRDCS to trip Electronic Trips E and F. The Electronic Trips do not perform a nuclear safety related function in the new system. In addition a redundant trip feature was added to DCRDCS to actuate the Electronic Trips if a power supply voltage is outside allowed limits.

Control Rod Patching (Rod Group Assignment)

The DCRDCS will utilize software to assign a CRDM to a specific group. As a result the connector patch panels are no longer needed and are being deleted along with the associated TS 3.5.2.6 requirement that the patch panels be kept locked. The existing TS 3.5.2.6 requirement was part of the original licensing design bases for the plant, and in place because swapping cables in the existing patch panels would change which rods were assigned to which group. Therefore, the implementation of the administrative control of the rod group assignments is being changed from hardware based (lock and key) to software based (password protected in a specified plant configuration). To modify DCRDCS rod patching, the system must be offline with the reactor shutdown (all rods in). The software must be modified, recompiled, and downloaded into the DCRDCS memory with password control. The software is contained on a lap top computer in a locked cabinet in a vital area room. There are two levels of password protection once the laptop is accessed. Additionally, control rod patch verification will be performed by moving each control rod to verify correct group/rod assignment following any software modification and download. These controls provide assurance that the rod groups remain in conformance with the COLR.

Impact on Startup and Rod Withdrawal Accidents

The FMEA postulated a total of 481 independent failures to determine the global effects on the DCRDCS design. The failures that were postulated included failures of entire modules, failure of individual “slices” representing redundant channels of a module and discrete components (e.g., relays, switches, diodes, etc.). The results of the failures were placed into categories corresponding to the negative effects achieved by each fault. The results of the analysis concluded there were zero number of failures in the category of uncontrolled rod withdrawal.

Impact on ATWS Requirements

TMI-1 commitments related to the ATWS rule were reviewed and it was confirmed that implementation of the DCRDCS will not adversely impact ATWS rule compliance at TMI-1. The ATWS rule for B&W designs requires a DSS, "independent from the existing reactor trip system (from sensor output to interruption of power to the control rods)." The current design of the DSS installed at TMI-1 opens the 480 VAC 1G and 1L switchgear breakers on DSS actuation, which supply power to the Train A and Train B AC RTBs and subsequently to all control rods.

Installation of the DCRDCS will not change the sensor and logic portion of the DSS. The actuation portion of the DSS subsystem will be modified by removing DSS output from tripping the 480 VAC 1G and 1L switchgear breakers, and providing the DSS outputs as discrete inputs to the DCRDCS Triplex triple-modular-redundant (TMR) Controller. When both the DSS A and B inputs are provided to the TMR Controller, it will utilize Electronic Trip circuits to disable the A and B train SCR gating in the SRPSs, removing power from the CRDMs and shutting down the reactor. The primary design requirement of the DSS system is that it be diverse from the RPS. The diversity of DSS from RPS is maintained by this modification because the RPS trips the 3-phase incoming power breakers, whereas the DSS signals disable the gating power to the SRPSs. The independence of the RPS and the DSS trips is maintained, because no common components are utilized by both trip systems. The response time of DSS may be increased up to a maximum of 1.5 seconds; however, an evaluation (Reference 4) concluded that original ATWS acceptance criteria are maintained. The peak reactor coolant pressure will remain less than the ATWS Acceptance Criterion 1 limit of 3250 psia with the DSS response time increased to 2.0 seconds.

Position Indication

Position indication is provided by the Flat Panel Position Indication Display. The display provides Absolute and Relative Position Indication. Accuracy and readability of Position Indication is improved.

The new system determines Relative Position Indication by counting pulses sent from the control system to the gate drives for the CRDMs. This indication is displayed to the operators on the same display as the Absolute Position Indication. The shift check surveillance verifies that the system is functioning by comparing Relative Position Indication to Average Position Indication; therefore, this requirement will be retained. However, Relative Position Indication has no adjustable hardware, so calibration is not required and the refueling calibration surveillance requirement for Relative Position Indication will be deleted. The calibration surveillance of the Absolute Position Indicator is retained to verify the analog-to-digital converter.

Zone Reference Switches

The DCRDCS provides indication of the Zone Reference Switches to the operator. The 50% and 75% switch indication is on the Position Indication panel and the Plant Process Computer (PPC). The 25% switch position is on the PPC. The Zone Reference Switches are located in the position indication tube, are actuated by the magnet on the lead screw assembly, and provide direct indication of the rod position. Use of this indication is only effective when the rod is at a zone reference position. Use of this indication to locate a rod is now appropriate since this indication is readily available to the operator.

Conclusions

- The response time of the RPS system to trip the reactor is not adversely impacted.
- The analyzed time for control rod insertion is unchanged and the replacement RTB revised trip times are bounded by the UFSAR safety analysis considerations.
- The existing channel independence, separation, and performance requirements of the RTBs are retained.
- The DSS retains its ability to interrupt power to the CRD mechanisms independently of the RPS/RTB interface.
- The new RTBs are seismically qualified, and are located in areas not subject to harsh environments.
- The new RTBs are an updated version of a type of hardware that is widely and successfully used. They replace original RTBs, which have become obsolete and increasingly difficult to maintain.
- The new design is simpler and more reliable than the existing design, and utilizes a smaller number of components.
- The new design is similar to the design currently in use at Oconee Nuclear Station. This design has been shown to be more reliable than the TMI-1 design.
- The software control of the rod assignment within each group provides improved traceability and verification of rod assignment, and therefore, is considered better than the controls provided by the existing system.

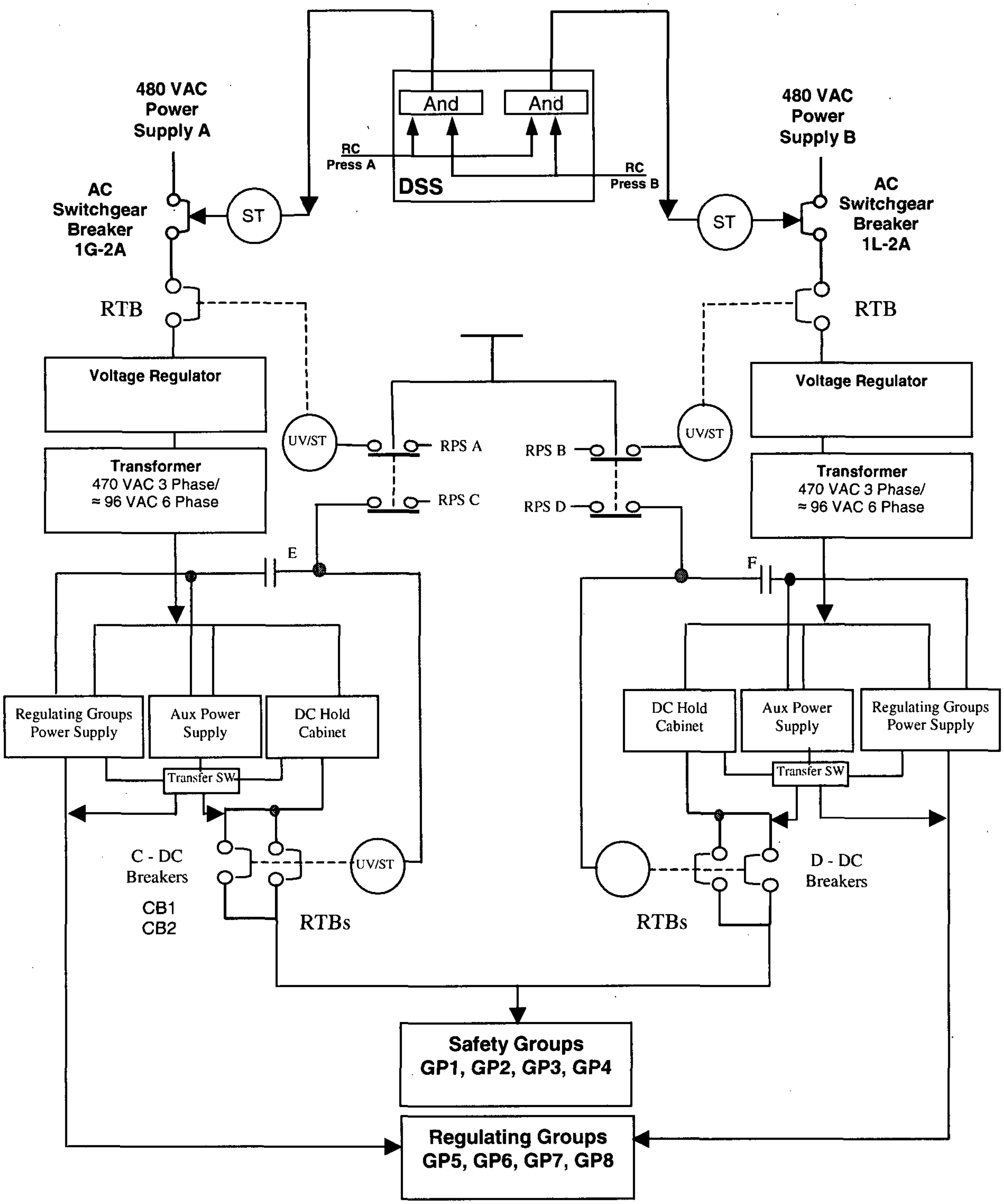

Figure 1

Existing Control Rod Drive Control System and Reactor Trip Breaker Configuration

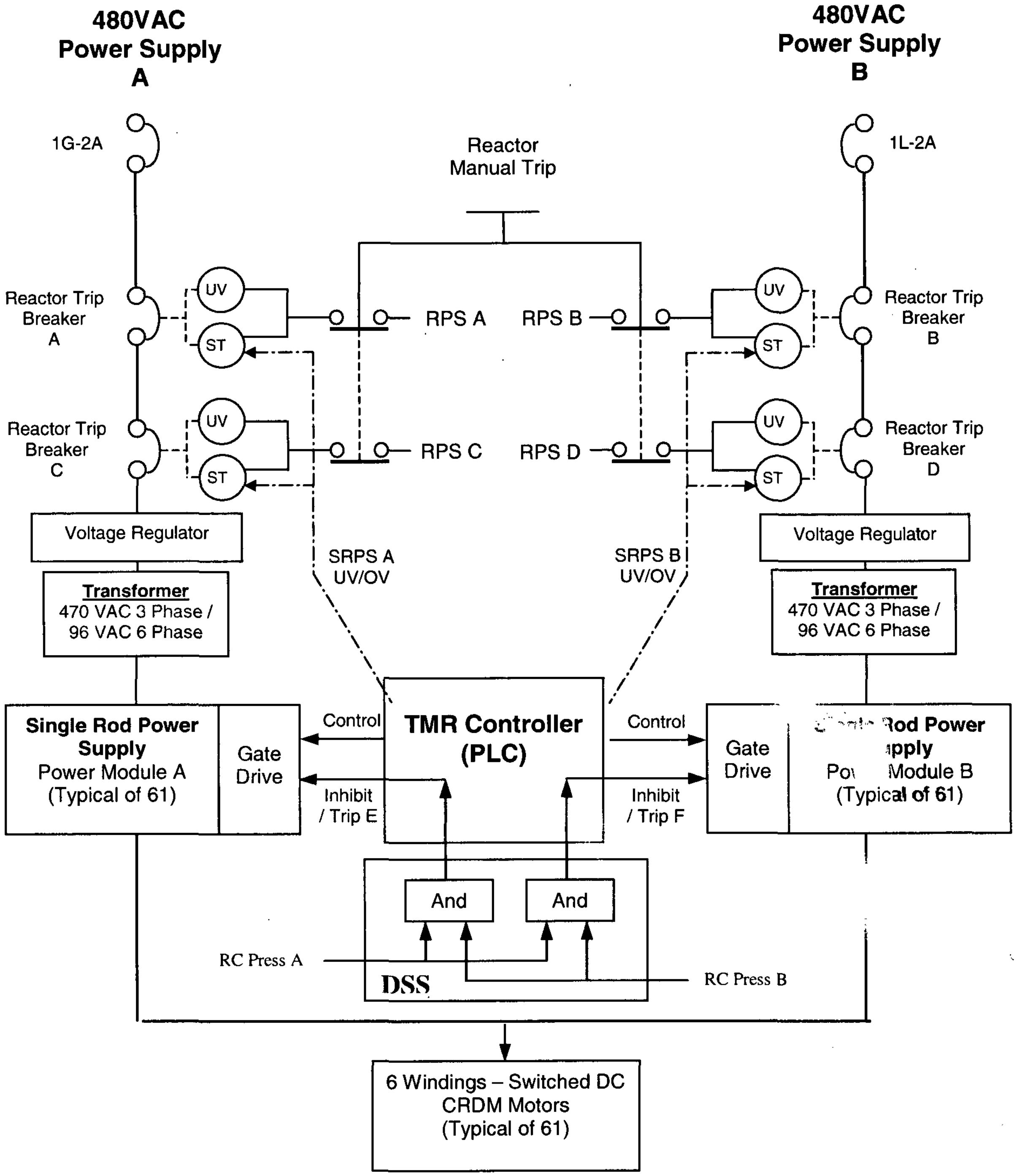

Figure 2

New Digital Control Rod Drive Control System and Reactor Trip Breaker Configuration

APSR Removal

The TMI-1 reactor contains eight APSRs, which were originally designed to control the axial power profile in the core within limits specified in TSs. APSRs are similar to standard control rod assemblies with three major differences: 1) the APSRs use a weaker neutron absorber material (Inconel), 2) the neutron absorber section in APSRs are part-length instead of full-length, and 3) the control rod drive mechanisms for APSRs do not insert on a reactor trip signal.

The original design of B&W reactors included APSRs based on historical analyses, which indicated that axial xenon instabilities could occur for certain core designs and scenarios, and that these instabilities could be damped using APSRs. These conclusions were based on analyses documented in B&W Topical Report BAW-10010, Parts 1-3, (Reference 5) circa 1971. The analyses were performed based on core designs, operating philosophies (i.e., rodded core design) and using neutronic codes from the 1970 timeframe. Analyses summarized in the B&W report indicated that there was no axial instability for a non-rodded core.

TMI-1 transitioned from rodded core designs to non-rodded, feed-and-bleed core designs in 1978 (Cycle 4). Due to internal operating experience that showed the APSRs, if not used properly, could exacerbate axial imbalance swings during a power transient, TMI-1 instituted administrative guidelines circa 1994 to keep APSRs in a stationary position throughout the cycle, including during power transients. Since that time, axial imbalance swings have been successfully maintained within core operating limits using regulating control rods and the naturally damped characteristics of non-rodded core designs.

The design functions of APSRs as described in the TMI-1 UFSAR are to: 1) maintain an acceptable power distribution in the core and control any tendency towards axial oscillations, and 2) create core flux imbalance during the Power Imbalance Detector Correlation (PIDC) test such that measurements can be taken to obtain information regarding the correlation between incore and excore detectors. The ability to suppress power oscillations is a General Design Criterion (#12) and the PIDC test fulfills a TS instrument surveillance requirement for power range channels.

Suppression of Axial Oscillations

The ability to maintain acceptable power distributions and control any tendency towards axial oscillations without the need for APSRs has been demonstrated at TMI-1 over the past 13-plus years of operation. In addition, an evaluation of axial xenon stability and transient imbalance control using regulating control rods was performed using AREVA's NRC-approved Nodal Expansion Method Optimization (NEMO) nuclear design code (Reference 6). The evaluation demonstrated that, for power reductions and for return to full power transients, axial power oscillations are naturally damped and there is no axial xenon instability (i.e., diverging axial power oscillation). Furthermore, the evaluation demonstrated how coordinated movements of regulating Control Rod Group 7 (CRG-7), using planned water adds (either borations or dilutions), can provide further axial imbalance control during power transients. This evaluation was the basis for a revision to the AREVA Power Operations Guidelines to include guidance for axial imbalance control using CRG-7 when APSRs are not available. It is noted that Westinghouse Pressurized Water Reactors (PWRs) are not equipped with axial power shaping rods and use regulating control rod banks similar to TMI-1 Group 7 to control axial xenon oscillations.

As noted in AREVA's NRC-approved reload licensing topical report BAW-10179 (Reference 7), the xenon stability for each reload cycle design is evaluated as part of the standard reload licensing process. Typically, the evaluation has shown that axial xenon oscillations are naturally damped. For example, the xenon stability index values calculated for the current TMI-1 Cycle 17 core range from –0.098 at beginning of cycle to –0.031 at end of cycle; negative values indicate the core is naturally damped. However, if the simulation shows that the criterion is not met, the result would be noted in the safety evaluation and regulating rods would be used to damp any induced xenon oscillations.

Power Imbalance Detector Correlation (PIDC)Test

The PIDC test is performed following each refueling outage at partial reactor power conditions. The purpose of the test is to obtain varying axial offset conditions in the core so that the offset measured by excore power range upper and lower detector chambers can be calibrated against the incore detector system offset measurements. Historically, the axial offset swings have been obtained using a combination of APSRs and regulating Control Rod Group 7 (CRG-7) movements, as permitted by governing test procedures.

A feasibility study was performed using AREVA's NRC-approved NEMO code to determine if CRG-7 could provide the required imbalance swings without using APSRs. The study demonstrated that the PIDC test could be successfully completed using CRG-7 for a variety of scenarios allowed by the test procedure (i.e., 25% RP with deep rod insertion, 40% RP with deep rod insertion, 60% RP with shallow rod insertion, and 80% RP with shallow rod insertion). Upon successful completion of the feasibility study, the PIDC test specification was revised to include an optional procedure for performing the test using CRG-7 only. In 2007, the PIDC test was completed successfully during TMI-1 Cycle 17 initial power escalation testing with APSRs fully withdrawn from the core and using CRG-7 to obtain the axial offset swings. This confirmed the calculation results that the PIDC test can be performed using CRG-7 only and without APSRs.

Safety Analyses

A review of the TMI-1 safety analyses in UFSAR Chapter 14 identified no credit for APSRs in any of the events analyzed for TMI-1. UFSAR Section 3.2.4.3.2.2 indicates that, "For actuating the partial length control rods, which maintain their set position during a reactor trip of the shim safety drive, the CRDM is modified so that the roller nut assembly will not disengage from the lead screw on a loss of power to the stator." Therefore, there is no affect of APSRs on transient analyses, as APSR positions do not change in the event of a reactor trip.

In addition, AREVA reload methods require an evaluation of cycle-specific core parameters (e.g., Doppler and Moderator coefficients, rod worths, boron concentrations, shutdown margin, etc.) against the assumptions used in UFSAR safety analyses to demonstrate the safety analyses remain bounding for the reload. The cycle-specific values are determined based on the nominal position of the APSRs. In cores with APSRs removed, these parameters would be calculated in accordance with AREVA reload methods with no APSR poison in the core.

Elimination of the eight APSRs from the reactor would increase the bypass flow relative to current TMI-1 cycle designs. Bypass flow represents a percentage of reactor coolant flow that takes parallel paths from the core inlet to the core outlet (e.g., through open fuel assembly guide tubes, core shroud, etc.) and is not directly available to remove heat from fuel rods. In the case of APSR elimination, an increase in bypass flow occurs due to reactor coolant passing through the empty fuel assembly guide tubes that had previously been blocked by the eight APSRs. The impact of increased bypass flow was evaluated based on the fuel designs used in current TMI-1 core design (i.e., Mark-B12 and Mark-B-HTP). Bypass flow is an input to three areas of analysis: 1) core Departure from Nucleate Boiling (DNB) analyses, 2) core hydraulic lift analyses, and 3) LOCA analyses. The current licensing bases in these three areas were reviewed to determine the impact of eliminating APSRs.

AREVA reload methods require at least a 95% probability at a 95% confidence level that the hot fuel rod in the core does not experience DNB during normal operation or events of moderate frequency. An increase in bypass flow reduces the volume of coolant that would be available to transfer heat from fuel rods and could adversely affect DNB calculations. A review of the current analyses of record demonstrated that sufficient retained thermal margin is available to offset the small increase in bypass flow that is calculated with APSRs eliminated and assuming there are no Burnable Poison Rod Assemblies (BPRA) in the core. The DNB analysis of record for TMI-1 for the Mark-B-HTP fuel design assumed a bounding bypass flow of 6.7% (nominal value consistent with AREVA Statistical Core Design (SCD) methodology). This assumption bounds the bypass flow of 6.62% calculated for the projected Cycle 18 core design and would remain valid for a full core of Mark-B-HTP fuel (bypass flow of 6.72%) using an insignificant DNB penalty (i.e., for an additional 0.02% bypass flow, the DNB penalty would be an insignificant 0.03 DNB points). The DNB analysis of record for TMI-1 for the Mark-B12 fuel design assumed a bounding bypass flow of 5.72% (nominal). This assumption does not bound the bypass flow of 6.62% calculated for the projected Cycle 18 core design. Consistent with SCD methodology, a DNB penalty of 1.5 points would be applied to account for the higher bypass flow. Sufficient retained thermal margin is available in the Mark-B12 DNB analysis to assess this penalty in Cycle 18 without requiring a revision to the analysis. After Cycle 18, the Mark-B12 fuel design is being phased out and is not expected to be used in future TMI-1 cycle designs. AREVA reload methods include a cycle-specific verification of bypass flow based on the actual number of control rod assemblies and BPRAs in the core.

AREVA reload methods require the fuel assembly holddown spring system to be capable of maintaining fuel assembly contact with the lower support plate during Condition I events. Hydraulic lift analyses conservatively assume a minimum value for bypass flow to maximize the volume of coolant available to lift fuel assemblies out of the lower support plate. Elimination of APSRs would increase bypass flow and reduce hydraulic lift of the fuel; therefore, the current analysis of record would remain bounding with APSRs eliminated.

AREVA LOCA methods require that Peak Cladding Temperatures (PCT) do not exceed 2200 deg F based on an approved evaluation model analysis that incorporates 10 CFR 50 Appendix K models. An increase in bypass flow reduces the volume of coolant that would be available to transfer heat from fuel rods and could adversely affect LOCA PCT calculations. A review of the current analyses of record demonstrated a conservatively bounding bypass of 7.5% (maximum value consistent with AREVA LOCA methodology) was assumed in both the Mark-B12 and Mark-B-HTP LOCA analyses. This value remains bounding for a core with no APSRs, whether the core is all Mark-B12 (maximum bypass flow of 6.78%), all Mark-B-HTP (maximum bypass flow of 7.32%), or mixed Mark-B12 / Mark-B-HTP (maximum bypass flow between 6.78 and 7.32%). Therefore there is no reduction in the current margin to the 2200 deg PCT limit with APSRs removed.

Conclusions

Elimination of APSRs from the TMI-1 reactor is acceptable. APSRs have not been used for transient imbalance control at TMI-1 since Cycle 10 (Cycle 17 is currently in operation) and the axial imbalance swings required for successful performance of the PIDC test were analyzed (Reference 8) and proven to be obtainable using regulating CRG-7.

Analyses have shown that the core designs employed at TMI-1 are stable with respect to axial oscillations and that xenon oscillations initiated during power transients are naturally damped. Actual operating experience at TMI-1 bears out the analysis conclusions that adequate axial imbalance control can be maintained using coordinated movements of CRG-7 using timed water additions.

The current licensing bases will remain valid for a core configuration with no APSRs. Safety analyses do not credit or account for APSRs and reload licensing analyses remain valid for the changes in core bypass flow resulting from removal of APSRs.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

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

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

   Response: No.

   The proposed license amendment modifies the Technical Specifications (TSs) to incorporate new TS requirements associated with the new Digital Control Rod Drive Control System (DCRDCS) and an evaluation to permanently remove the Axial Power Shaping Rods (APSRs) from the reactor core.

   The proposed license amendment will continue to ensure reliability and operability of the control rod drive Reactor Trip Breakers (RTBs) to perform their safety function of tripping the reactor. The existing channel independence, separation and performance requirements of the RTBs and the Reactor Protection System (RPS) response time are retained for the new configuration. The RTB design was reviewed for credible common mode failures and no credible common mode failures were identified that would prevent the breakers from performing the reactor trip function. Reliable RTBs and their associated support circuitry provide assurance that a reactor trip will occur when initiated. The planned DCRDCS modification upgrades the relay-based Control Rod Drive Control System (CRDCS) to a solid state programmable DCRDCS using single rod power supplies assigned to each of the 61 Control Rod Drives (CRDs). The new components will meet the same design requirements (i.e., seismic, environmental, quality, separation, single failure criteria) as the existing components in the CRDCS / RPS interface. The DCRDCS modification will improve the reliability of the system by resolving age-related degradation issues and replacing obsolete equipment.

Malfunction of the CRD control system (or operator error) is an initiator of the startup and rod withdrawal accidents. The new DCRDCS meets the design requirements of the original system including redundancy of critical functions, isolation from safety related systems, reactivity rate limit, and single failure requirements. Electrical ratings, heat loading, structural and environmental aspects have been verified to be acceptable. Therefore, there is no increase in the frequency of occurrence or probability of a malfunction of equipment important to safety. The DCRDCS is not required for accident mitigation, post accident response or offsite release mitigation. The action of the RPS to trip the RTBs, to remove power from the control rods, and drop the rods into the core, remains independent of the DCRDCS. Therefore, there is no increase in the consequences or probability of occurrence of an accident previously evaluated.

The modified Diverse Scram System (DSS) design utilizes the same power sources as the existing DSS, which are independent of reactor trip (i.e., RPS) related power sources. There is no change to the DSS logic circuitry. The DSS sensors and trip setpoint remains unchanged. Updated Final Safety Analysis (UFSAR) Section 7.1.5.4 indicates that: "The DSS provides an independent method of automatically tripping the reactor in the event the RPS related reactor trip system fails. It is designed in accordance with the Anticipated Transient Without Scram (ATWS) rule and, as such, its critical features are independence and diversity from the reactor trip system and emphasis on not failing in a tripped state." However, DSS is not safety related and is not credited in any safety analysis in UFSAR Chapter 14, "Safety Analysis." The assumed DSS response time increase from 1.0 second to 2.0 seconds has been evaluated and the results of the analysis concluded that the original acceptance criteria are maintained. Therefore, the proposed change to the DSS does not increase the consequence of an ATWS event.

The proposed license amendment will continue to ensure the reliability and operation of the reactor core. Analyses have shown that the core designs employed at TMI-1 are stable with respect to axial oscillations and that xenon oscillations initiated during power transients are naturally damped or can be manually suppressed using regulating control rods (i.e., Control Rod Group 7 (CRG-7)). Actual operating experience at TMI-1 bears out the analysis conclusions that adequate axial imbalance control can be maintained using coordinated movements of CRG-7 using timed water additions. A review of the TMI-1 safety analyses found no mention or credit for APSRs in any of the events analyzed for TMI-1, and safety analysis assumptions are verified to bound key core parameters for each reload with explicit accounting for the presence of (or lack of) APSRs in the core. Therefore, there is no affect of APSRs on transient analyses, as APSR positions do not change in the event of a reactor trip.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The systems affected by implementing the proposed changes to the TS are not assumed to initiate design basis accidents. Rather, the CRDCS / RPS interface (i.e., RTBs) is used to mitigate the consequences of an accident that has already occurred. The proposed TS changes do not affect the mitigating function of this system. The failure of any one RTB will not inhibit the reactor trip function. The modification interfaces with the DSS, which mitigates the ATWS event, but the interface function remains the same.

A Failure Modes and Effects Analysis (FMEA) was performed on the DCRDCS design to determine if adverse effects (i.e., loss of reactor control, uncontrolled rod withdrawal, reactor trip, or prevention of reactor trip) could result from the credible failure of a single component. The FMEA concluded that no credible single component failure would cause a total loss of reactor control, an uncontrolled rod withdrawal, a reactor trip, or prevent a reactor trip. All operation critical to the safe and effective performance of the DCRDCS maintained sufficient redundancy such that no credible single failure could compromise the design functionality.

The APSRs' original function was to control any reactor core tendency towards axial oscillations resulting from xenon instabilities that could occur for certain early reactor core designs (i.e., rodded core designs). More recent non-rodded feed-and-bleed core designs have been shown to be self-dampened with respect to axial xenon oscillations such that APSRs have not been moved at TMI-1 for axial power control since 1994, and have been withdrawn from the reactor core since Fall 2005 with Core Operating Limits Report (COLR) limits preventing insertion, consistent with AREVA reload methods. Use of regulating Group 7 has been shown to adequately suppress axial xenon oscillations.

The proposed changes to the CRDCS and APSRs and associated TS changes do not introduce any new accident initiators, nor do they reduce or adversely affect the capabilities of any plant structure, system, or component to perform their safety function.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   Response: No.

   The proposed TS changes do not adversely impact any plant safety limits, setpoints, response times, or design parameters. The changes do not negatively affect the fuel, fuel cladding, reactor coolant system, or containment integrity.

   Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, AmerGen 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.

5.2 APPLICABLE REGULATORY REQUIREMENTS/CRITERIA

10 CFR 50.55a Codes and Standards

Section 50.55a (h)(2) requires that, for plants with construction permits issued prior to January 1, 1971, the design of protection systems must meet the original licensing bases, or may meet IEEE 603-1991, "Criteria for Safety Systems for Nuclear Power Generating Stations," with correction sheet dated January 30, 1995. The construction permit for TMI-1 was issued in 1968. The RTB design specification meets original licensing bases and includes the following:

1. IEEE Standard 141, Recommended Practices for Electric Power Distribution for Industrial Plants
2. IEEE 323-1983, Qualifying Class 1E Equipment for Nuclear Power Generating Stations
3. IEEE 344-1987, Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations
4. IEEE 383-1974, Type Test of Class 1E Electric Cables, Field Splices and Connections for Nuclear Power Generating Stations.
5. ANSI C37.13, Low Voltage AC Power Circuit Breakers and AC Power Circuit Protectors
6. ANSI C37.16, Preferred Ratings, Related Requirements and Application Recommendations for Low-Voltage Power Circuit Breakers and AC Power Circuit Protectors
7. ANSI C37.20.1-1993, Standard for Metal-Enclosed Low-Voltage Power Circuit Breaker Switchgear
8. ANSI C37.50-1989, Switchgear-Low-Voltage AC Power Circuit Breakers Used in Enclosures - Test Procedures
9. ANSI C37.90.1-1989, Standard Surge Withstand Capability (SWC) Tests for Protective Relays and Relay Systems
10. N45.2, Requirements for Quality Assurance Programs for Nuclear Power Plants

In addition, as part of the RPS/CRDCS interface, the RTB design meets the following:

1. IEEE-279, Criteria for Protection Systems for Nuclear Power Plants
2. IEEE-379, Standard Application of Single Failure Criterion to Nuclear Power Generating Class 1E Systems
3. Regulatory Guide 1.53, Application of the Single Failure Criterion to Nuclear Power Plant Protection Systems
4. Regulatory Guide 1.75, Physical Independence of Electrical Systems

10 CFR 50 Appendix A General Design Criterion (GDC)

GDC 12-Suppression of reactor power oscillations

The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

The ability to maintain acceptable power distributions and control any tendency towards axial oscillations without the need for APSRs has been demonstrated at TMI-1 over the past 13-plus years of operation. In addition, an evaluation of axial xenon stability and transient imbalance control using regulating control rods was performed using AREVA's NRC-approved NEMO nuclear design code. The evaluation demonstrated that, for power reductions and for return to full power transients, axial power oscillations are naturally damped and there is no axial xenon instability (i.e., diverging axial power oscillation).

Conclusion

AmerGen has determined that the proposed changes do not require any exemptions or relief from regulatory requirements and do not affect conformance with any General Design Criteria.

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

6.0 ENVIRONMENTAL CONSIDERATION

The proposed amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The proposed amendment revises technical specifications to support the digital upgrade of the CRDCS system and the elimination of APSRs from the reactor core. 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 the 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 need be prepared in connection with the proposed amendment.

7.0 PRECEDENT

Oconee Nuclear Station, Units 1, 2 and 3 RE; Issuance of Amendments (TAC NOS. MC1785, MC1786, and MC1787), dated November 2, 2004 (ML042820458)

8.0 REFERENCES

1. AmerGen ECR TM 07-01037, Attachment 1, "CRDCS Controls Upgrade-Common Engineering Documentation"

2. AmerGen ECR TM 07-01037, Attachment 15, "APSR Removal Evaluation"

3. AREVA Document Identifier 51-9082709-000, "Failure Modes and Effects Analysis for the Three Mile Island Digital Control Rod Drive Control System," June 30, 2008

4. AREVA Document Identifier 51-9091240, "TMI-1 DSS Response Time Evaluation," September 15, 2008

5. BAW-10010, Parts 1-3, "Stability Margin for Xenon Oscillations," August 1969, February 1970, June 1971

6. Nodal Expansion Method Optimization (NEMO), BAW-1018A Rev 1, B&W Fuel Company, Lynchburg, VA, March 1993

7. BAW-10179P-A, Rev. 7, "Safety Criteria and Methodology for Acceptable Cycle Reload Analysis," January 2008

8. AREVA Document Identifier 32-9031517-001 "Feasibility Study for PIDC Test Without APSRs," December 19, 2006

<u>Attachment 2</u>

Three Mile Island Nuclear Station, Unit 1
Technical Specification Change Request No. 342

Proposed Technical Specification Marked-Up Pages

The pages included in this attachment are:

1-3
3-27a
3-34
3-35
4-3
4-5
4-48
5-4

1.4.2 REACTOR PROTECTION SYSTEM

The reactor protection system is **described in Section 7.1** of the Updated FSAR. It is that combination of protection channels and associated circuitry which forms the automatic system that protects the reactor by control rod trip. It includes the four protection channels, their associated instrument channel inputs, manual trip switch, all rod drive control protection trip breakers, and activating relays or coils.

1.4.3 PROTECTION CHANNEL

A PROTECTION CHANNEL **as described in Section 7.1** of the updated FSAR (one of three or one of four independent channels, complete with sensors, sensor power supply units, amplifiers, and bistable modules provided for every reactor protection safety parameter) is a combination of instrument channels forming a single digital output to the protection system's coincidence logic. It includes a shutdown bypass circuit, a protection channel bypass circuit and a reactor trip module.

1.4.4 REACTOR PROTECTION SYSTEM LOGIC

This system utilizes reactor trip module relays (coils and contacts) in all four of the protection four channels **as described in Section 7.1** of the updated FSAR, to provide reactor trip signals for de-energizing the ~~six~~ control rod drive trip breakers. The control rod drive trip breakers are arranged to provide a one-out-of-two-times-two logic. Each element of the one-out-of-two-times-two logic is controlled by a separate set of two-out-of-four logic contacts from the four reactor protection channels.

1.4.5 ENGINEERED SAFETY FEATURES SYSTEM

This system utilizes relay contact output from individual channels arranged in three analog sub-systems and two two-out-of-three logic sub-systems as shown in Figure 7.1-4 of the updated FSAR. The logic sub-system is wired to provide appropriate signals for the actuation of redundant engineered safety features equipment on a two-of-three basis for any given parameter.

1.4.6 DEGREE OF REDUNDANCY

The difference between the number of operable channels and the number of channels which, when tripped, will cause an automatic system trip.

1.5 <u>INSTRUMENTATION SURVEILLANCE</u>

1.5.1 TRIP TEST

A TRIP TEST is a test of logic elements in a protection channel to verify their associated trip action.

3.5.1.7.1 Power may be restored through the breaker with the failed trip feature for up to two hours for surveillance testing per T.S. 4.1.1.

Deleted

3.5.1.8 During STARTUP, HOT STANDBY or POWER OPERATION, in the event that one of the two regulating control rod power SCR electronic trips is inoperable, within one hour:

a. Place the inoperable SCR electronic trip in the tripped condition or

b. Remove the power supplied to the associated SCRs. Specification 3.0.1 applies.

3.5.1.8.1 Power may be restored through the SCRs with the failed electronic trip for up to two hours for surveillance testing per T.S. 4.1.1.

3.5.1.9 The reactor shall not be in the Startup mode or in a critical state unless both HSPS actuation logic trains associated with the Functional units listed in Table 3.5-1 are operable except as provided in Table 3.5-1,D.

3.5.1.9.1 With one HSPS actuation logic train inoperable, restore the train to OPERABLE or place the inoperable device in an actuated state within 72 hours or be in HOT SHUTDOWN within the next 12 hours. With both HSPS actuation logic trains inoperable, restore one train to OPERABLE within 1 hour or be in HOT SHUTDOWN within the next 6 hours.

Bases

Every reasonable effort will be made to maintain all safety instrumentation in operation. The reactor trip, on loss of feedwater may be bypassed below 7% reactor power. The bypass is automatically removed when reactor power is raised above 7%. The reactor trip, on turbine trip, may be bypassed below 45% reactor power (Reference 1). The safety feature actuation system must have two analog channels functioning correctly prior to startup.

The anticipatory reactor trips on loss of feedwater pumps and turbine trip have been added to reduce the number of challenges to the safety valves and power operated relief valve but have not been credited in the safety analyses.

Operation at rated power is permitted as long as the systems have at least the redundancy requirements of Column "B" (Table 3.5-1). This is in agreement with redundancy and single failure criteria of IEEE 279 as described in FSAR Section 7.

There are four reactor protection channels. Normal trip logic is two out of four. Minimum required trip logic is one out of two.

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f. If a control rod in the regulating ~~or axial power shaping~~ group~~s~~ is declared inoperable per Specification 4.7.1.2, operation may continue provided that within 1 hour the rods in the group are positioned such that the rod that was declared inoperable is maintained within allowable group average position limits of Specification 4.7.1.2.

OR 7,

g. If the inoperable rod in Paragraph "e" above is in groups 5, 6, 7, or 8, the other rods in the group may be trimmed to the same position. Normal operation of 100 percent of the thermal power allowable for the reactor coolant pump combination may then continue provided that within 1 hour the rod that was declared inoperable is maintained within allowable group average position limits in 3.5.2.5.

3.5.2.3 The worth of single inserted control rods during criticality is limited by the restriction of Specification 3.1.3.5 and the Control Rod Position Limits defined in Specification 3.5.2.5.

3.5.2.4 Quadrant Tilt:

a. Except for physics tests, the quadrant tilt, as determined using the full incore system (FIS), shall not exceed the values in the CORE OPERATING LIMITS REPORT.

The FIS is OPERABLE for monitoring quadrant tilt provided the number of valid symmetric string individual SPND signals in any one quadrant is not less than the limit in the CORE OPERATING LIMITS REPORT.

b. When the full incore system is not OPERABLE and except for physics tests quadrant tilt as determined using the power range channels for each quadrant (out of core detector system)(OCD), shall not exceed the values in CORE OPERATING LIMITS REPORT.

c. When neither detector system above is OPERABLE and, except for physics tests, quadrant tilt as determined using the minimum incore system (MIS), shall not exceed the values in the CORE OPERATING LIMITS REPORT.

d. Except for physics tests if quadrant tilt exceeds the tilt limit, allowable power shall be reduced 2 percent for each 1 percent tilt in excess of the tilt limit. For less than four pump operation, thermal power shall be reduced 2 percent below the thermal power allowable for the reactor coolant pump combination for each 1 percent tilt in excess of the tilt limit.

e. If quadrant power tilt exceeds the tilt limit then within a period of 10 hours, the quadrant power tilt shall be reduced to less than the tilt limit except for physics tests, or the following verifications and/or adjustments in setpoints and limits shall be made:

1. Verify $F_Q(Z)$ and $F_{\Delta H}^{N}$ are within limits of the COLR once per 2 hours and restore QPT to $\leq$ steady state limit within 24 hours, or perform steps 2, 3, & 4 below.

3.5.2.5 Control Rod Positions

a. Operating rod group overlap shall not exceed 25 percent ±5 percent, between two sequential groups except for physics tests.

b. Position limits are specified for regulating control rods. Except for physics tests or exercising control rods, the regulating control rod insertion/withdrawal limits are specified in the CORE OPERATING LIMITS REPORT.

1. If regulating rods are inserted in the restricted operating region, corrective measures shall be taken immediately to achieve an acceptable control rod position. Acceptable control rod positions shall be attained within 24 hours, and FQ(Z) and $F^N_{\Delta H}$ shall be verified within limits once every 2 hours, or power shall be reduced to ≤ power allowed by insertion limits.

2. If regulating rods are inserted in the unacceptable operating region, initiate boration within 15 minutes to restore SDM to ≥1% ΔK/K, and restore regulating rods to within restricted region within 2 hours or reduce power to ≤ power allowed by rod insertion limits.

c. Safety rod limits are given in 3.1.3.5.

3.5.2.6 ~~The control rod drive patch panels shall be locked at all times with limited access to be authorized by the Plant Manager.~~ Deleted

3.5.2.7 Axial Power Imbalance:

a. Except for physics tests the axial power imbalance, as determined using the full incore system (FIS), shall not exceed the envelope defined in the CORE OPERATING LIMITS REPORT.

The FIS is operable for monitoring axial power imbalance provided the number of valid self powered neutron detector (SPND) signals in any one quadrant is not less than the limit in the CORE OPERATING LIMITS REPORT.

b. When the full incore detector system is not OPERABLE and except for physics tests axial power imbalance, as determined using the power range channels (out of core detector system)(OCD), shall not exceed the envelope defined in the CORE OPERATING LIMITS REPORT.

c. When neither detector system above is OPERABLE and, except for physics tests axial power imbalance, as determined using the minimum incore system (MIS), shall not exceed the envelope defined in the CORE OPERATING LIMITS REPORT.

d. Except for physics tests if axial power imbalance exceeds the envelope, corrective measures (reduction of imbalance by APSR CONTROL ROD movements and/or reduction in reactor power) shall be taken to maintain operation within the envelope. Verify FQ(Z) and $F^N_{\Delta H}$ are within limits of the COLR once per 2 hours when not within imbalance limits.

Amendment No. ~~10, 17, 29, 38, 39, 50, 120, 126, 142, 150, 179, 211~~, 219

TABLE 4.1-1

INSTRUMENT SURVEILLANCE REQUIREMENTS

CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE	REMARKS
1. Protection Channel Coincidence Logic	NA	Q	NA	
2. Control Rod Drive Trip Breaker ~~and Regulating Rod Power SCRs~~	NA	Q	NA	(1) Includes independent testing of shunt trip and undervoltage trip features.
3. Power Range Amplifier	D(1)	NA	(2)	(1) When reactor power is greater than 15%. (2) When above 15% reactor power run a heat balance check once per shift. Heat balance calibration shall be performed whenever heat balance exceeds indicated neutron power by more than two percent.
4. Power Range Channel	S	S/A	M(1)(2)	(1) When reactor power is greater than 60% verify imbalance using incore instrumentation. (2) When above 15% reactor power calculate axial offset upper and lower chambers after each startup if not done within the previous seven days.
5. Intermediate Range Channel	S(1)	P S/U	NA	(1) When in service.
6. Source Range Channel	S(1)	P S/A	NA	(1) When in service.
7. Reactor Coolant Temperature Channel	S	S/A	F	

TABLE 4.1-1 (Continued)

CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE	REMARKS
19. Reactor Building Emergency Cooling and Isolation System Analog Channels				
a. Reactor Building 4 psig Channels	S(1)	M(1)	F	(1) When CONTAINMENT INTEGRITY is required.
b. RCS Pressure 1600 psig	S(1)	M(1)	NA	(1) When RCS Pressure > 1800 psig.
c. Deleted				
d. Reactor Bldg. 30 psi pressure switches	S(1)	M(1	F	(1) When CONTAINMENT INTEGRITY is required.
e. Reactor Bldg. Purge Line High Radiation (AH-V-1A/D)	W(1)	M(1)(2)	F	(1) When CONTAINMENT INTEGRITY is required.
f. Line Break Isolation Signal (ICCW & NSCCW)	W(1)	M(1)	R	(1) When CONTAINMENT INTEGRITY is required.
20. Reactor Building Spray System Logic Channel	NA	Q	NA	
21. Reactor Building Spray 30 psig pressure switches	NA	M	F	
22. Pressurizer Temperature Channels	S	NA	R	
23. Control Rod Absolute Position	S(1)	NA	~~R~~ NA	(1) Check with Relative Position ~~Indicator~~ Deleted
a. Zone Reference Switch	N/A	R(1)	NA	(1) Verify switch functions
24. Control Rod Relative Position	S(1)	NA	~~R~~ NA	(1) Check with Absolute Position ~~Indicator.~~
25. Core Flooding Tanks				
a. Pressure Channels Coolant	NA	NA	F	
b. Level Channels	NA	NA	F	
26. Pressurizer Level Channels	S	NA	R	

Amendment No. ~~24, 78, 156, 157, 175, 189, 200~~, 225

4.7 REACTOR CONTROL ROD SYSTEM TESTS

4.7.1 CONTROL ROD DRIVE SYSTEM FUNCTIONAL TESTS

Applicability

Applies to the surveillance of the control rod system.

Objective

To assure operability of the control rod system.

Specification

4.7.1.1 The control rod trip insertion time shall be measured for each control rod at either full flow or no flow conditions following each refueling outage prior to return to power. The maximum control rod trip insertion time for an operable control rod drive mechanism ~~except for the axial power shaping rods (APSRs),~~ from the fully withdrawn position to ¾ insertion (104 inches travel) shall not exceed 1.66 seconds at hot reactor coolant full flow conditions or 1.40 seconds for the hot no flow conditions (Reference 1). ~~For the APSRs it shall be demonstrated that loss of power will not cause rod movement.~~ If the trip insertion time above is not met, the rod shall be declared inoperable. DELETED DELETED

4.7.1.2 If a control rod is misaligned with its group average by more than an indicated nine inches, the rod shall be declared inoperable and the limits of Specification 3.5.2.2 shall apply. The rod with the greatest misalignment shall be evaluated first. The position of a rod declared inoperable due to misalignment shall not be included in computing the average position of the group for determining the operability of rods with lesser misalignments.

4.7.1.3 If a control rod cannot be exercised, or if it cannot be located with absolute or relative position indications ~~or~~, in or out limit ~~lights~~ INDICATION, OR ZONE REFERENCE SWITCH INDICATION, the rod shall be declared to be inoperable.

Bases

The control rod trip insertion time is the total elapsed time from power interruption at the control rod drive breakers until the control rod has actuated the 25% withdrawn reference switch during insertion from the fully withdrawn position. The specified trip time is based upon the safety analysis in UFSAR, Chapter 14 and the Accident Parameters as specified therein.

Each control rod drive mechanism shall be exercised by a movement of a minimum of 3% of travel every 92 days. This requirement shall apply to either a partial or fully withdrawn control rod at reactor operating conditions. Exercising the drive mechanisms in this manner provides assurance of reliability of the mechanisms.

5.3 REACTOR

Applicability

Applies to the design features of the reactor core and reactor coolant system.

Objective

To define the significant design features of the reactor core and reactor coolant system.

Specification

5.3.1 REACTOR CORE

5.3.1.1 A fuel assembly normally contains 208 fuel rods arranged in a 15 by 15 lattice. The reactor shall contain 177 fuel assemblies. Fuel rods shall be clad with zircaloy, ZIRLO, or zirconium-based M5 alloy materials and contain an initial composition of natural or slightly enriched uranium dioxide as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions. The details of the fuel assembly design are described in TMI-1 UFSAR Chapter 3.

5.3.1.2 The reactor core shall approximate a right circular cylinder with an equivalent diameter of 128.9 inches. The active fuel height is defined in TMI-1 UFSAR Chapter 3.

5.3.1.3 The core average and individual batch enrichments for the present cycle are described in TMI-1 UFSAR Chapter 3.

5.3.1.4 The control rod assemblies (CRA) ~~and axial power shaping rod assemblies (APSRA)~~ are distributed in the reactor core as shown in TMI-1 FSAR Chapter 3. The CRA ~~and APSRA~~ design data are also described in the UFSAR.

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5.3.1.5 The TMI-1 core may contain burnable poison rod assemblies (BPRA) and gadolinia-urania integral burnable poison fuel pellets as described in TMI-1 UFSAR Chapter 3.

5.3.1.6 Reload fuel assemblies and rods shall conform to design and evaluation data described in the UFSAR. Enrichment shall not exceed a nominal 5.0 weight percent of U_{235}.

5.3.2 REACTOR COOLANT SYSTEM

5.3.2.1 The reactor coolant system shall be designed and constructed in accordance with code requirements. (Refer to UFSAR Chapter 4 for details of design and operation.)

Attachment 3

Three Mile Island Nuclear Station, Unit 1
Technical Specification Change Request No. 342

Proposed Technical Specification Bases Marked-Up Pages
(For information only)

The pages included in this attachment are:

3-28
3-35a
3-36

The four reactor protection channels were provided with key operated bypass switches to allow on-line testing or maintenance on only one channel at a time during power operation. Each channel is provided alarm and lights to indicate when that channel is bypassed. There will be one reactor protection system bypass switch key permitted in the control room.

Each reactor protection channel key operated shutdown bypass switch is provided with alarm and lights to indicate when the shutdown bypass switch is being used.

INSERT B3.5.1

Power is normally supplied to the control rod drive mechanisms from two separate parallel 460 volt sources. ~~Redundant trip devices are employed in each of these sources. The AC Trip Breaker is one means to trip a source. The redundant means is a parallel configuration consisting of two DC Trip Breakers and five SCR power supplies. The SCRs are turned off by the "electronic trip relays."~~

Diverse trip features are provided on each breaker. These are the undervoltage relay and shunt trip attachment. Each trip feature is tested separately. Failure of one breaker trip feature does not result in loss of redundancy and a reasonable time limit is provided for corrective action.

Failure in the untripped state of a breaker ~~or SCR electronic trip~~ results in loss of redundancy and prompt action is required. Failure of both trip features on one breaker is considered failure of the breaker.

Power may be restored through the failed breaker ~~(SCRs)~~ for a limited time to perform required testing.

The 4.16kv ES Bus Undervoltage Relays detect a degraded voltage or Loss of Voltage on the associated ES Bus. Detection of low voltage will separate the ES bus from the offsite power, initiate load shedding and start the associated diesel generator. The relays do not function during design basis events where acceptable offsite voltage is available. If the voltage relays on either train are not operable, the time permitted for repair is consistent with other safety related equipment. If both trains are affected then shutdown is initiated in accordance with Specification 3.0.1 since automatic response of the diesel generator is required to assure completion of the safety function if offsite power is degraded or lost.

Automatic initiation of EFW is provided on loss of all reactor coolant pumps, loss of both main feedwater pumps, low OTSG level, and high reactor building pressure. High reactor building pressure would be indicative of a loss of coolant accident, main steam line or feedwater line break inside the reactor building. Operability of these instruments is required in order to assure that the EFW system will actuate and control at the appropriate OTSG level without operator action for those events where timely initiation of EFW is required.

Automatic isolation of main feedwater is provided on low OTSG pressure in order to maintain appropriate RCS cooling (minimize overcooling) following a loss of OTSG integrity and minimize the energy released to the Reactor Building atmosphere.

Insert B3.5.1, to Bases 3.5.1 on page 3-28:

through four AC trip breakers, designated A, B, C and D. The breaker undervoltage trip coils are powered by RPS channels A, B, C and D, respectively. From these circuit breakers, the CRD power travels through voltage regulators and stepdown transformers to complete redundant power buses that feed the CRD Single Rod Power Supplies (SRPSs) A and B.

Two AC breakers (A and C) are arranged in series to feed SRPS power bus A, and the other two AC breakers (B and D) are in series to feed SRPS power bus B. Opening at least one circuit breaker in each of the two parallel paths to the SRPS will cause a reactor trip, in a one-out-of-two taken twice logic. Either path can provide sufficient power to operate all CRDs.

e. If an acceptable axial power imbalance is not achieved within 24 hours, reactor power shall be reduced to ≤40% FP within 2 hours.

f. Axial power imbalance shall be monitored on a minimum frequency of once every 12 hours when axial power imbalance alarm is OPERABLE, and every 1 hour when imbalance alarm is inoperable during power operation above 40 percent of rated power.

3.5.2.8 A power map shall be taken at intervals not to exceed 31 effective full power days using the incore instrumentation detection system to verify the power distribution is within the limits shown in the CORE OPERATING LIMITS REPORT.

Bases

The axial power imbalance, quadrant power tilt, and control rod position limits are based on LOCA analyses which have defined the maximum linear heat rate. These limits are developed in a manner that ensures the initial condition LOCA maximum linear heat rate will not cause the maximum clad temperature to exceed 10 CFR 50 Appendix K. Operation outside of any one limit alone does not necessarily constitute a situation that would cause the Appendix K Criteria to be exceeded should a LOCA occur. Each limit represents the boundary of operation that will preserve the Acceptance Criteria even if all three limits are at their maximum allowable values simultaneously. The effects of the APSRs are included in the limit development. Additional conservatism included in the limit development is introduced by application of:

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a. Nuclear uncertainty factors

b. Thermal calibration uncertainty

c. Fuel densification effects

d. Hot rod manufacturing tolerance factors

e. Postulated fuel rod bow effects

f. Peaking limits based on initial condition for Loss of Coolant Flow transients.

The incore instrumentation system uncertainties used to develop the axial power imbalance and quadrant tilt limits accounted for various combinations of invalid Self Powered Neutron Detector (SPND) signals. If the number of valid SPND signals falls below that used in the uncertainty analysis, then another system shall be used for monitoring axial power imbalance and/or quadrant tilt.

For axial power imbalance and quadrant power tilt measurements using the incore detector system, the minimum incore detector system consists of OPERABLE detectors configured as follows:

Axial Power Imbalance

a. Three detectors in each of three strings shall lie in the same axial plane with one plane in each axial core half.

b. The axial planes in each core half shall be symmetrical about the core mid-planes.

c. The detectors shall not have radial symmetry.

Quadrant Power Tilt

a. Two sets of four detectors shall lie in each core half. Each set of four shall lie in the same axial plane. The two sets in the same core half may lie in the same axial plane.

b. Detectors in the same plane shall have quarter core radial symmetry.

The 25 ± 5% overlap between successive control rod groups is allowed since the worth of a rod is lower at the upper and lower part of the stroke. Control rods are arranged in groups or banks defined as follows:

Group	Function
1	Safety
2	Safety
3	Safety
4	Safety
5	Regulating
6	Regulating
7	Regulating
8	APSR (axial power shaping rod bank)

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Control rod groups are withdrawn in sequence beginning with group 1. Groups 5,6 and 7 are overlapped 25 percent. The normal position at power is for group 7 to be partially inserted. Group 8 position is maintained consistent with the core design which may include withdrawal of APSRs and long-term operation with APSRs fully withdrawn. When APSR withdrawal is specified in the core design, appropriate limits for time of the withdrawal and restrictions for subsequent APSR insertion are included in the COLR.

The rod position limits are based on the most limiting of the following three criteria: ECCS power peaking, shutdown margin, and potential ejected rod worth. As discussed above, compliance with the ECCS power peaking criterion is ensured by the rod position limits. The minimum available rod worth, consistent with the rod position limits, provides for achieving hot shutdown by reactor trip at any time, assuming the highest worth control rod that is withdrawn remains in the full out position (Reference 1). The rod position limits also ensure that inserted rod groups will not contain single rod worths greater than: 0.65% delta k/k at rated power. These values have been shown to be safe by the safety analysis of the hypothetical rod ejection accident (Reference 2). A maximum single inserted control rod worth of 1.0% delta k/k is allowed by the rod position limits at hot zero power. A single inserted control rod worth 1.0% delta k/k at beginning of life, hot, zero power would result in a lower transient peak thermal power and, therefore, less severe environmental consequences than 0.65% delta k/k ejected rod worth at rated power.

The plant computer will scan for tilt and imbalance and will satisfy the technical specification requirements. If the computer is out of service, then manual calculation for tilt above 15 percent power and imbalance above 40 percent power must be performed as specified until the computer is returned to service.

Attachment 4

SUMMARY OF REGULATORY COMMITMENTS

The following table identifies commitments made in this document. (Any other actions discussed in the submittal represent intended or planned actions. They are described to the NRC for the NRC's information and are not regulatory commitments.)

COMMITMENT	COMMITTED DATE OR "OUTAGE"	COMMITMENT TYPE	
		ONE-TIME ACTION (Yes/No)	Programmatic (Yes/No)
Control rod patch verification will be performed by moving each control rod to verify correct group/rod assignment following any software recompiling and download.	T1R18 Refuel Outage	No	Yes