

ATTACHMENT A

NIAGARA MOHAWK POWER CORPORATION
LICENSE NO. NPF-69
DOCKET NO. 50-410

PROPOSED CHANGES TO THE TECHNICAL SPECIFICATIONS

Replace existing Technical Specification pages 2-3, 3/4 2-2, 3/4 3-60, 3/4 3-62, 3/4 3-63, 3/4 3-64, 3/4 3-65 and 6-22 and Bases page B3/4 2-1 with the attached revised pages. The pages have been retyped in their entirety and have marginal markings to indicate the changes to the text.

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

<u>FUNCTIONAL UNIT</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUE</u>
1. Intermediate Range Monitor, - Neutron Flux - High	$\leq 120/125$ divisions of full scale	$\leq 122/125$ divisions of full scale
2. Average Power Range Monitor:		
a. Neutron Flux - Upscale, Setdown	$\leq 15\%$ of RATED THERMAL POWER	$\leq 20\%$ of RATED THERMAL POWER
b. Flow-Biased Simulated Thermal Power - Upscale		
1) Flow-Biased	$\leq 0.58 (W-\Delta W)^{(a)} + 59\%$, with a maximum of $\leq 113.5\%$ of RATED THERMAL POWER	$\leq 0.58 (W-\Delta W)^{(a)} + 62\%$, with a maximum of $\leq 115.5\%$ of RATED THERMAL POWER
2) High-Flow-Clamped		
c. Fixed Neutron Flux - Upscale	$\leq 118\%$ of RATED THERMAL POWER	$\leq 120\%$ of RATED THERMAL POWER
d. Inoperative	NA	NA
3. Reactor Vessel Steam Dome Pressure - High	≤ 1037 psig	≤ 1057 psig
4. Reactor Vessel Water Level - Low, Level 3	≥ 159.3 in. above instrument zero*	≥ 157.8 in. above instrument zero
5. Main Steam Line Isolation Valve - Closure	$\leq 8\%$ closed	$\leq 12\%$ closed
6. Main Steam Line Radiation ^(b) - High	≤ 3.0 x full-power background	≤ 3.6 x full-power background
7. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig

* See Bases Figure B 3/4 3-1.

(a) The Average Power Range Monitor Scram Function varies as a function of recirculation loop drive flow (W). ΔW is defined as the difference in indicated drive flow (in percent of drive flow which produces rated core flow) between two loop and single loop operation at the same core flow. $\Delta W = 0$ for two loop operation. $\Delta W = 5\%$ for single loop operation.

(b) See footnote (***) to Table 3.3.2-2 for trip setpoint during hydrogen addition test.

POWER DISTRIBUTION LIMITS

3/4.2.2 AVERAGE POWER RANGE MONITOR SETPOINTS

LIMITING CONDITIONS FOR OPERATION

3.2.2 The Average Power Range Monitor (APRM) flow-biased simulated thermal power-upscale scram trip setpoint (S) shall be established according to the relationship specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With the APRM flow-biased simulated thermal power-upscale scram trip setpoint less conservative than the value shown in the Allowable Value column for S, as above determined, initiate corrective action within 15 minutes and adjust S to be consistent with the Trip Setpoint value* within 6 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.2 The FRACTION OF RATED THERMAL POWER (FRTP) and the CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY (CMFLPD) shall be determined, the value of T** calculated, and the most recent actual APRM flow-biased simulated thermal power-upscale scram trip setpoint verified to be within the above limit or adjusted, as required:

- a. At least once per 24 hours,
- b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and
- c. Initially and at least once per 12 hours when the reactor is operating with CMFLPD greater than or equal to FRTP.
- d. The provisions of Specification 4.0.4 are not applicable.

* With CMFLPD greater than the FRTP rather than adjusting the APRM setpoints, the APRM gain may be adjusted so that APRM readings are greater than or equal to 100% times CMFLPD provided that the adjusted APRM reading does not exceed 100% of RATED THERMAL POWER and a notice of adjustment is posted on the reactor control panel.

** Definition of T is specified in the CORE OPERATING LIMITS REPORT.

TABLE 3.3.6-1

CONTROL ROD BLOCK INSTRUMENTATION

<u>TRIP FUNCTION</u>	<u>MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION</u>	<u>APPLICABLE OPERATIONAL CONDITIONS</u>	<u>ACTION</u>
1. <u>Rod Block Monitor(a)</u>			
a. Upscale	2	1*	60
b. Inoperative	2	1*	60
c. Downscale	2	1*	60
2. <u>Source Range Monitor</u>			
a. Detector Not Full In (b)	3	2	61
	2	5	61
b. Upscale(c)	3	2	61
	2	5	61
c. Inoperative(c)	3	2	61
	2	5	61
d. Downscale(d)	3	2	61
	2(f)	5	61
3. <u>Intermediate Range Monitor</u>			
a. Detector Not Full In	6	2, 5	61
b. Upscale	6	2, 5	61
c. Inoperative	6	2, 5	61
d. Downscale(e)	6	2, 5	61
4. <u>Scram Discharge Volume</u> Water Level - High, Float Switch	2	1, 2, 5**	62
5. <u>Reactor Coolant System</u> <u>Recirculation Flow</u>			
a. Upscale	2	1	62
b. Inoperative	2	1	62
c. Comparator	2	1	62
6. <u>Reactor Mode Switch</u>			
a. Shutdown Mode	2	3, 4	62
b. Refuel Mode	2	5	62

TABLE 3.3.6-2

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

<u>TRIP FUNCTION</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUE</u>
1. <u>Rod Block Monitor</u>		
a. Upscale	*	*
b. Inoperative	NA	NA
c. Downscale	≥ 5% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
2. <u>Source Range Monitor</u>		
a. Detector Not Full In	NA	NA
b. Upscale	≤ 1 x 10 ⁵ cps	≤ 1.6 x 10 ⁵ cps
c. Inoperative	NA	NA
d. Downscale	≥ 3 cps**	≥ 1.8 cps**
3. <u>Intermediate Range Monitors</u>		
a. Detector Not Full In	NA	NA
b. Upscale	≤ 108/125 divisions of full scale	≤ 110/125 divisions of full scale
c. Inoperative	NA	NA
d. Downscale	≥ 5/125 divisions of full scale	≥ 3/125 divisions of full scale
4. <u>Scram Discharge Volume</u>		
Water Level - High, Float Switch	≤ 16.5 in.	≤ 39.75 in.

Table 3.3.6-2 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

<u>TRIP FUNCTION</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUE</u>
5. <u>Reactor Coolant System Recirculation Flow</u>		
a. Upscale	≤ 108% rated flow	≤ 111% rated flow
b. Inoperative	NA	NA
c. Comparator	≤ 10% flow deviation	≤ 11% flow deviation
6. <u>Reactor Mode Switch</u>		
a. Shutdown Mode	NA	NA
b. Refuel Mode	NA	NA

* Specified in the CORE OPERATING LIMITS REPORT

** For fuel loading and startup from refueling the count rate may be less than 3 cps if the following conditions are met: the signal to noise ratio is greater than or equal to 5, and the signal is greater than 1.3 cps.

TABLE 4.3.6-1

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>TRIP FUNCTION</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL FUNCTIONAL TEST</u>	<u>CHANNEL CALIBRATION(a)</u>	<u>OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED</u>
1. <u>Rod Block Monitor</u>				
a. Upscale	NA	S/U(b)(c), Q(c)	Q	1*
b. Inoperative	NA	S/U(b)(c), Q(c)	NA	1*
c. Downscale	NA	S/U(b)(c), Q(c)	Q	1*
2. <u>Source Range Monitors</u>				
a. Detector Not Full In	NA	S/U(b), W	NA	2, 5
b. Upscale	NA	S/U(b), W	Q	2, 5
c. Inoperative	NA	S/U(b), W	NA	2, 5
d. Downscale	NA	S/U(b), W	Q	2, 5
3. <u>Intermediate Range Monitors</u>				
a. Detector Not Full In	NA	S/U(b), W	NA	2, 5
b. Upscale	NA	S/U(b), W	Q	2, 5
c. Inoperative	NA	S/U(b), W	NA	2, 5
d. Downscale	NA	S/U(b), W	Q	2, 5
4. <u>Scram Discharge Volume</u>				
Water Level - High, Float Switch	NA	Q	R	1, 2, 5**

TABLE 4.3.6-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>TRIP FUNCTION</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL FUNCTIONAL TEST</u>	<u>CHANNEL CALIBRATION(e)</u>	<u>OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED</u>
5. <u>Reactor Coolant System Recirculation Flow</u>				
a. Upscale	NA	S/U(b), Q	Q	1
b. Inoperative	NA	S/U(b), Q	NA	1
c. Comparator	NA	S/U(b), Q	Q	1
6. <u>Reactor Mode Switch</u>				
a. Shutdown Mode	NA	R	NA	3, 4
b. Refuel Mode	NA	R	NA	5

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section assure that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the 2200°F limit specified in 10 CFR 50.46.

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is dependent only secondarily on the rod-to-rod power distribution within an assembly. The peak clad temperature is calculated assuming an LHGR for the highest powered rod which is equal to or less than the design LHGR corrected for densification. This LHGR times 1.02 is used in the heatup code along with the exposure-dependent steady-state gap conductance and rod-to-rod local peaking factor. The limiting value for APLHGR is specified in the CORE OPERATING LIMITS REPORT for two-recirculation-loop operation.

The calculational procedure used to establish the APLHGR specified in the CORE OPERATING LIMITS REPORT is based on a loss-of-coolant accident analysis. The analysis was performed using General Electric (GE) calculational models which are consistent with the requirements of Appendix K to 10 CFR 50. A complete discussion of each code employed in the analysis is presented in Reference 1.

A list of the significant plant input parameters to the loss-of-coolant accident analysis is presented in Bases Table B3.2.1-1.

For plant operations with single recirculation loop the MAPLHGR limits are specified in the CORE OPERATING LIMITS REPORT. The constant factor is derived from LOCA analyses initiated from single loop operation to account for earlier boiling transition at the limiting fuel node compared to the standard LOCA evaluations.

3/4.2.2 APRM SETPOINTS

The fuel cladding integrity Safety Limits of Specification 2.1 were based on a power distribution which would yield the design LHGR at RATED THERMAL POWER. The flow-biased simulated thermal power-upscale scram setting of the APRM instruments must be adjusted to ensure that the MCPR does not become less than the fuel cladding integrity safety limit or that greater than or equal to 1% plastic strain does not occur in the degraded situation. The scram setpoint is adjusted in accordance with the formula in this specification when the combination of THERMAL POWER and CMFLPD indicates a higher peaked power distribution to ensure that an LHGR transient would not be increased in the degraded condition.

ADMINISTRATIVE CONTROLS

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

6.9.1.8 (Continued)

The Semiannual Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM (PCP) and to the OFFSITE DOSE CALCULATION MANUAL (ODCM), pursuant to Specifications 6.13 and 6.14, respectively, as well as any major change to liquid, gaseous, or solid radwaste treatment systems pursuant to Specification 6.15. It shall also include a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 3.12.2.

The Semiannual Radioactive Effluent Release Reports shall also include the following: an explanation of why the inoperability of liquid or gaseous effluent monitoring instrumentation was not corrected within the time specified in Specification 3.3.7.9 or 3.3.7.10, respectively, and a description of the events leading to liquid holdup tanks exceeding the limits of Specification 3.11.1.4.

CORE OPERATING LIMITS REPORT

6.9.1.9

- a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle for the following:
- 1) The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) for Specification 3.2.1.
 - 2) The Average Power Range Monitor (APRM) flow-biased simulated thermal power-upscale scram trip setpoint for Specification 3.2.2.
 - 3) The K_r core flow adjustment factor for Specification 3.2.3.
 - 4) The MINIMUM CRITICAL POWER RATIO (MCPR) for Specification 3.2.3.
 - 5) The LINEAR HEAT GENERATION RATE (LHGR) for Specification 3.2.4.
 - 6) Control Rod Block Instrumentation Setpoint for the rod block monitor upscale trip setpoint and allowable value for Specification 3.3.6.

and shall be documented in the CORE OPERATING LIMITS REPORT.

- b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following document.
- 1) General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDE-20566 latest approved revision.

ATTACHMENT B

NIAGARA MOHAWK POWER CORPORATION
LICENSE NO. NPF-69
DOCKET NO. 50-410SUPPORTING INFORMATION AND NO SIGNIFICANT HAZARDS
CONSIDERATION ANALYSIS**1.0 BACKGROUND**

The proposed amendment consists of several changes that will improve utilization of the Extended Load Line Limit Analysis (ELLLA) region. Operation above the rated rod line in the ELLLA region is discussed and analyzed in Appendix 15G of the Nine Mile Point Unit 2 (NMP2) Updated Safety Analysis Report (USAR). Full power operation at core flows between 87% and 100% of rated takes advantage of the boiling water reactor operating characteristics to allow improved fuel management.

Currently, operation in the ELLLA region is restricted because at low flows the average power range monitor (APRM) flow-biased scram encroaches on the ELLLA region. To allow full utilization of the increased operating domain, modifications to the reactor protection system (RPS) APRM flow-biased simulated thermal power (STP) - upscale scram setpoints are required. The proposed change would revise the slope of the APRM flow-biased STP scram line on the power/flow map to $[0.58(W - \Delta W) + 59 \text{ percent}]$ while maintaining the current maximum value of 113.5%. This proposed APRM flow-biased scram setpoint provides adequate margin for the Safety Limits and yet allows operating margin to reduce the possibility of unnecessary shutdowns.

In addition, the proposed change would relocate reference to the APRM rod block instrumentation and setpoints from the Technical Specifications to USAR Section 7.6, "All Other Instrumentation Systems Required for Safety." The setpoint for the APRM flow-biased rod block, currently specified in the Core Operating Limits Report, would be relocated to USAR Section 7.6. The setpoints for the other APRM rod block functions specified in the Technical Specifications would also be relocated to USAR Section 7.6. This change is consistent with NUREG 1433, "Improved Standard Technical Specifications for BWR/4." During formulation of the improved technical specifications (ITS), the APRM rod block instrumentation was determined to have minimal safety impact on plant operations and it was deleted from the ITS. Niagara Mohawk has reviewed the NMP2 licensing basis and confirmed the applicability of this conclusion to NMP2.

2.0 DESCRIPTION OF PROPOSED CHANGES

2.1 APRM Flow-Biased Simulated Thermal Power Upscale Scram Setpoint

Table 2.2.1-1, "Reactor Protection System Instrumentation Setpoints"

	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUE</u>
Revise From:		
2. Average Power Range Monitor		
b. Flow-Biased Simulated Thermal Power - Upscale		
1) Flow-Biased	$\leq 0.66(W-\Delta W)^{(a)} + 51\%$, with a maximum of 113.5% RATED THERMAL POWER	$\leq 0.66(W-\Delta W)^{(a)} + 54\%$, with maximum of ≤ 115.5 RATED THERMAL POWER
2) High-Flow-Clamped		

Revise To:

2. Average Power Range Monitor		
b. Flow-Biased Simulated Thermal Power - Upscale		
1) Flow-Biased	$\leq 0.58(W-\Delta W)^{(a)} + 59\%$, with a maximum of $\leq 113.5\%$ RATED THERMAL POWER	$\leq 0.58(W-\Delta W)^{(a)} + 62\%$, with a maximum of $\leq 115.5\%$ RATED THERMAL POWER
2) High-Flow-Clamped		

2.2 APRM Flow-Biased Neutron Flux-Upscale Rod Block Instrumentation System

Specification 3/4.2.2, "Average Power Range Monitor Setpoints"

Reference to the APRM flow-biased neutron flux-upscale control rod block trip setpoint (S_{RB}) has been deleted from Limiting Condition for Operation 3.2.2, its associated Action Statement, and from Surveillance Requirement 4.2.2.

Table 3.3.6-1, "Control Rod Block Instrumentation"

Delete Parameter 2.a, "APRM Flow-Biased Neutron Flux - Upscale"
 Delete Parameter 2.b, "APRM Inoperative"
 Delete Parameter 2.c, "APRM Downscale"
 Delete Parameter 2.d, "APRM Neutron Flux - Upscale, Startup"

Table 3.3.6-2, "Control Rod Block Instrumentation Setpoints"

Delete Parameter 2.a, "APRM Flow-Biased Neutron Flux - Upscale"
 Delete Parameter 2.b, "APRM Inoperative"
 Delete Parameter 2.c, "APRM Downscale"
 Delete Parameter 2.d, "APRM Neutron Flux - Upscale, Startup"

Table 4.3.6-1, "Control Rod Block Instrumentation Surveillance Requirements"

- Delete Parameter 2.a, "APRM Flow-Biased Neutron Flux - Upscale"
- Delete Parameter 2.b, "APRM Inoperative"
- Delete Parameter 2.c, "APRM Downscale"
- Delete Parameter 2.d, "APRM Neutron Flux - Upscale, Startup"

Section 6.9.1.9, "Core Operating Limits Report"

Revise From:

- a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle for the following:
- 1) The AVERAGE ...
 - 2) The Average Power Range Monitor (APRM) flow biased simulated thermal power-upscale scram trip setpoint and flow-biased neutron flux-upscale control rod block trip setpoint for Specification 3.2.2.
 - 3) The K_f ...
 - 4) The MINIMUM ...
 - 5) The LINEAR ...
 - 6) Control Rod Block Instrumentation Setpoints for the rod block monitor upscale and APRM flow biased neutron flux upscale trip and allowable values for Specification 3.3.6.

and shall be documented in the CORE OPERATING LIMITS REPORT.

Revise To:

- a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle for the following:
- 1) The AVERAGE ...
 - 2) The Average Power Range Monitor (APRM) flow biased simulated thermal power-upscale scram trip setpoint for Specification 3.2.2.
 - 3) The K_f ...
 - 4) The MINIMUM ...
 - 5) The LINEAR ...
 - 6) Control Rod Block Instrumentation Setpoint for the rod block monitor upscale trip setpoint and allowable value for Specification 3.3.6.

and shall be documented in the CORE OPERATING LIMITS REPORT.

Bases Section B3.4.2.2, "APRM Setpoints"

Changes to Section B3/4.2.2 are proposed to reflect the change to the APRM rod block instrumentation.

2.3 Editorial Changes

Table 3.3.6-2, "Control Rod Block Instrumentation Setpoints," Parameter 3.a

Present Wording:	Detector not full in
Proposed Wording:	Detector Not Full In

3.0 EVALUATION

The analysis supporting operation in the ELLLA region is provided in Appendix 15G of the NMP2 USAR. Appendix A, "Reload Analysis," contained in Vol. 28 of the USAR, documents the inclusion of ELLLA operating conditions in the cycle specific transient analyses. The Maximum Critical Power Ratio (MCPR) operating limits provided in the current Core Operating Limits Report bound operation in the ELLLA region and assure that the Safety Limit MCPR will not be violated. The ELLLA region is presented graphically in USAR Figure 15.0-2.

3.1 APRM Flow-Biased Simulated Thermal Power Upscale Scram Setpoint

The RPS APRM system provides two upscale scram trips. In the first one, the APRM fixed neutron flux trip, the APRM signal (which is proportional to thermal neutron flux) is compared with a fixed reference. The neutron flux leads the fuel cladding heat flux during fast transients because of the fuel time constant. Therefore, the fixed neutron flux trip provides adequate thermal margin for fuel cladding during fast transients. However, during relatively slow transients such as the loss of feedwater heater, the heat flux and neutron flux are almost in equilibrium. To provide adequate margin against thermal stress during slow transients, the neutron flux trip setpoint would have to be lowered significantly. This would adversely impact operating margin during fast transients.

To avoid lowering the neutron flux trip, a trip based on thermal power is provided. In the APRM simulated thermal power (STP) trip, the APRM signal is passed through a low pass RC filter to provide a trip dependent on heat flux rather than neutron flux. The low pass filter provides a six second time constant, which simulates the time constant of the fuel. The APRM STP trip is conservative since the RC time constant is shorter than the actual fuel thermal time constant. The STP signal is compared with a flow-biased reference which decreases approximately parallel to the flow control lines. Since the threshold of concern for neutron flux is significantly higher than that for thermal heat flux, the fixed neutron flux trip can be set higher than the STP trip, thereby allowing increased operating margin for fast transients.

The current RPS APRM STP trip setpoint is $[0.66(W - \Delta W) + 51\%]$, with a maximum value of 113.5%. The APRM flow-biased STP trip is clamped so that the maximum value of the trip setpoint remains less than the value of the fixed neutron flux trip. The proposed change raises the APRM flow-biased STP trip setpoint to $[0.58(W - \Delta W) + 59\%]$, while maintaining the maximum value of 113.5%. Although the proposed setpoint reaches the maximum value at a lower flow, the APRM flow-biased STP scram setpoint will remain below the fixed neutron flux scram.

The transient analyses for NMP2 uses a fixed analytical value of 117% for the APRM flow-biased STP trip. That is, the transient analysis does not adjust the APRM trip to account for lower setpoints at lower flows. This yields conservative results in the transient analysis since the APRM STP trip will occur at a lower value (i.e., sooner) than that assumed in the analysis. The analytical value of 117% corresponds to the maximum nominal trip setpoint of 113.5%. As stated above, the maximum trip setpoint is not changing. Therefore, the proposed change does not impact the results of any transient analyses. The results of the Loss of Feedwater Heating (Manual Recirculation Flow Control mode) transient are unaffected, with a reactor scram still occurring on high thermal power at approximately 57 seconds into the event (USAR Table 15.1-2). Therefore, operating limit MCPR values calculated based on the current setpoint bound those for the proposed setpoint.

3.2 APRM Flow-Biased Neutron Flux-Upscale Rod Block Instrumentation System

The proposed changes will delete reference to the APRM rod block instrumentation from the Technical Specifications. NMP2 takes no credit for the APRM rod blocks and the effect of the proposed change on operator action to correct unexpected situations is insignificant. Two events which could involve operation of the APRM rod block are the Loss of Feedwater Heating transient and a general operator error in which control rods are withdrawn beyond the upper rod line. These events result in relatively slow total power increases such that preventive action to avoid a scram is possible if the APRM flow-biased rod block were to actuate. However, the USAR takes no credit for the APRM rod block in terminating these transients. Loss of feedwater heating, while in the automatic recirculation flow control mode, results in a runback of core flow such that a new steady state operating condition is achieved without any safety system initiation. In the manual flow control mode, loss of feedwater heating without operator action will result in a reactor scram on high thermal power (USAR Table 15.1-2). The APRM flow-biased rod block can provide the operator with sufficient time to terminate this transient. However, NMP2 USAR analyses have demonstrated that the Δ CPR resulting from a scram would not result in exceeding any thermal safety limits.

During a fast flux transient such as the Main Steam Isolation Valve closure event, a reactor scram without a preceding APRM alarm would not impact plant safety since even with an alarm the operator would not have enough time to initiate a corrective action. There is no safety impact since the absence of a rod block has no significance after the scram initiation. For rod withdrawal errors, the primary protection for local power effects is provided by the Rod Block Monitor (RBM). In this respect, the APRM rod blocks and associated alarms are essentially backup functions and are not used in any licensing basis event. In general, scram prevention for slow transients is accomplished by 1) initiating a control rod withdrawal block via the RBM to inhibit further increases in power level due to rod withdrawals and 2) actuating an alarm on the Reactor Operator Console to enable the operator to take any necessary action to reduce the power level to within the normal operating envelope. All safety aspects associated with control rod withdrawal errors are addressed by the RBM.

Removal of the APRM rod block reference is consistent with NUREG 1433, "Improved Standard Technical Specifications for BWR/4." During the development of the Improved Technical Specifications (ITS), a methodology was developed for use in determining which requirements from the existing Standard Technical Specifications would be retained in the ITS. This methodology involved the application of three screening criteria to each TS requirement. Those requirements which met any of the three criteria were retained in the ITS. The APRM rod block functions did not meet any of the criteria and were deleted from the ITS. The NMP2 licensing basis has been reviewed to confirm the validity of this conclusion for NMP2:

1. Criterion 1

Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary

Evaluation:

The APRM rod block instrumentation does not detect or indicate degradation of the reactor coolant pressure boundary. The APRM flow-biased rod block and its associated alarm assist the operator in recognizing and terminating unexpected power excursions by initiating a rod block and alarm when neutron flux exceeds normal neutron flux levels. However, NMP2's transient analyses have demonstrated that the resulting CPR will remain above the Safety Limit CPR without actuation of the APRM flow-biased rod block. Similarly, the downscale and neutron flux upscale, startup (i.e., setdown) rod blocks merely alert the operator to

abnormal conditions in order to provide time for corrective action. In addition, the APRM inoperative rod block actuates concurrent with the APRM inoperative scram. Any transient resulting in an APRM inoperative rod block would also result in a scram. For rod withdrawal errors the primary protection for local power effects is provided by the Rod Block Monitor (RBM). The APRM rod block functions and associated alarms are essentially backup functions.

2. *Criterion 2*

A process variable that is an initial condition of a design basis accident (DBA) or transient analyses that either assume the failure of or presents a challenge to the integrity of a fission product barrier

Evaluation:

The APRM rod blocks are not process variables. It is an instrumentation system that responds to unexpected neutron flux excursions by alerting operators to the condition. It does not affect the initial conditions of any DBA or transient analysis nor does its function directly affect the integrity of any fission product barrier

3. *Criterion 3*

A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Evaluation:

The APRM flow-biased rod block is a system for a function to minimize the severity of pressurization transients analyzed in the USAR. However, it is not part of the primary success path for those transients. The downscale and neutron flux upscale, startup (i.e., setdown) rod blocks function merely to alert the operator to abnormal conditions in order to provide time for corrective action and the APRM inoperative rod block actuates concurrent with the APRM inoperative scram function. The primary protection for local power effects during rod withdrawal errors is provided by the RBM. The APRM rod block instrumentation is essentially a backup system and is not credited in any licensing basis event. NMP2 transient analyses have demonstrated that resulting CPR will remain above the Safety Limit CPR without actuation of any of the APRM rod blocks.

Consistent with the above evaluation, the reference to APRM rod block instrumentation has been deleted from the ITS. Niagara Mohawk proposes a similar change for NMP2. APRM rod block instrumentation setpoints currently in the Technical Specifications will be relocated to USAR Section 7.6. The APRM flow-biased neutron-flux upscale rod block setpoints, currently controlled in the Core Operating Limits Report, will also be relocated to USAR Section 7.6. The removal of the APRM rod blocks from the Technical Specifications does not alter plant design or system operation. The proposed change will permit administrative control of any future changes to the system design without processing of a license amendment. Any change to systems described in the USAR is subject to the requirements specified in the Administrative Controls Section of the Technical Specifications on modifications to the plant. Therefore, appropriate control of system configuration and operation is assured by the Administrative Controls Section of the Technical Specifications, which requires that all changes be evaluated in accordance with 10 CFR § 50.59.

3.3 Editorial Change

Item 3.a in Table 3.3.6-2 has been revised to maintain consistency with the other tables in Specification 3/4.3.6. The change involves capitalization only and does not change the intent of the specification.

4.0 CONCLUSION

The proposed amendment consists of several changes that will improve utilization of the Extended Load Line Limit Analysis (ELLLA) region. The proposed change would revise the APRM flow-biased flux scram line on the power/flow map and delete reference to the APRM rod block instrumentation.

[APRM Setpoint]

The proposed APRM flow-biased scram setpoint would maintain the same operating margin as the current setpoint. Both the current and proposed formulation of the flow-biased APRM scram equation are clamped at the same value such that the maximum value of the trip setpoint is less than the trip setpoint of the fixed neutron flux scram. The proposed formulation does, however, reach the maximum at a lower flow condition. Since the transient analyses utilize the maximum value, and the maximum value is not changing, the MCPR operating and safety limits are not affected by the change.

[APRM RB]

The APRM flow-biased rod block is designed primarily to detect any significant increase in reactor power and to help the control room operator prevent the reactor power level from reaching the APRM scram setpoint level. As such, the flow-biased APRM rod block provides a buffer in power and flow conditions from the APRM flow-biased scram function. While the APRM flow-biased rod block can help the operator terminate slow thermal transients, the NMP2 USAR analyses take no credit for APRM rod block actuation. During a fast flux transient, a reactor scram without a preceding APRM rod block or alarm has no safety impact since even with an alarm the operator would not have enough time to initiate a corrective action. Other events, such as a general operator error of withdrawing control rods beyond the upper rod line, produce only very mild power increases and the primary protection for local power effects is provided by the RBM. The APRM downscale and neutron flux upscale, startup (i.e., setdown) rod blocks are essentially backup functions and are not credited in any licensing basis event. They alert the operator to abnormal conditions and allow for corrective action. The APRM inoperative function serves no purpose since it actuates concurrent with the RPS APRM inoperative trip. Essentially, all safety aspects associated with control rod withdrawal errors are addressed by the RBM. Therefore, relocating the APRM rod blocks out of the Technical Specifications has minimal impact on plant safety.

[Editorial Changes]

Editorial changes by their nature do not change the intent or interpretation of the Technical Specifications. The proposed change to the Technical Specifications addresses only capitalization.

The aggregate affect of the proposed changes has been evaluated and found to have no resulting impact on system reliability or performance. The proposed changes assure that the system response to postulated accidents remains within accepted limits and will not cause existing Technical Specification Safety Limits, operational limits, or system performance criteria to be exceeded. Therefore, there is reasonable assurance that operation of Nine Mile Point Unit 2 in the proposed manner will not endanger the public health and safety and that issuance of the proposed amendment will not be inimical to the common defense and security.

10 CFR § 50.91 requires that at the time a licensee requests an amendment, it must provide to the Commission its analysis using the standards in 10 CFR § 50.92 concerning the issue of no significant hazards consideration. Therefore, in accordance with 10 CFR § 50.91, the following analysis has been performed:

The operation of Nine Mile Point Unit 2, in accordance with the proposed amendment, will not involve a significant increase in the probability or consequences of an accident previously evaluated.

APRM Flow-Biased Simulated Thermal Power Upscale Scram Setpoint

The proposed change will enhance utilization of the expanded operating domain by relaxing the restrictions imposed by the APRM flow-biased scram trip setpoint. The proposed formulation of the APRM flow-biased scram trip equation provides the same operating margin for the ELLLA region as the current equation provided at the rated flow condition. Further, the change to the flow-biased APRM scram trip does not affect any accident precursors or initiators. The trip serves to terminate certain transients. Therefore, the proposed change does not affect the probability of any accident. The transient analyses for NMP2 use a fixed analytical value of 117% for the APRM flow-biased simulated thermal power trip, corresponding to the maximum nominal trip setpoint of 113.5%. Since the analytical value of 117% is not changing, the proposed change does not impact the results of any transient analyses. MCPR operating and safety limits are not affected, and therefore there is no increase in the consequences of any accident. MCPR values calculated based on the current setpoint bound those for the proposed setpoint. Thus, the proposed changes do not adversely affect the response to previously analyzed accidents.

APRM Flow-Biased Neutron Flux-Upscale Rod Block Instrumentation System

The probability of accidents is not a function of the APRM rod block instrumentation since the failure of this system does not initiate or help to initiate any accident. Therefore, removing reference to the APRM rod block instrumentation from the Technical Specifications will not increase the probability of any accident previously evaluated. The APRM flow-biased rod block is not used to mitigate any accident in the USAR. While actuation of the APRM rod block can result in early termination of some slow pressurization transient events, USAR transient analyses take no credit for APRM rod block and the resulting Δ CPR will not cause CPR to exceed its safety limit. Further, during a fast flux transient a reactor scram without a preceding APRM rod block or alarm has no safety impact since the operator would not have enough time to initiate a corrective action even with an alarm. Other events such as a general operator error of withdrawing control rods beyond the upper rod line produce only very mild power increases. The downscale and neutron flux upscale, startup (i.e., setdown) rod blocks can alert the operator to these conditions, however the primary protection for local power effects is provided by the RBM. Essentially, all safety aspects associated with control rod withdrawal errors are addressed by the RBM. Finally, the APRM inoperative rod block actuates concurrent with the APRM inoperative scram and any transient resulting in an APRM inoperative rod block would also initiate a scram. Therefore, removing reference to the APRM rod block instrumentation will not increase the consequences of any transient previously evaluated.

Editorial Change

The proposed change to the Technical Specifications addresses only capitalization. Editorial changes by their nature do not change the intent or interpretation of the Technical Specifications and have no effect on accident probabilities or consequences.

The changes to the RPS will not affect plant response to previously analyzed transients or accidents. The MCPR limits previously evaluated remain valid. Therefore operation of Nine Mile Point Unit 2, in accordance with this proposed change, will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The operation of Nine Mile Point Unit 2, in accordance with the proposed amendment, will not create the possibility of a new or different kind of accident from any accident previously evaluated.

APRM Flow-Biased Simulated Thermal Power Upscale Scram Setpoint

Operation with the proposed APRM flow-biased scram line setpoint does not affect the assumptions (initial conditions) used in existing analyses and does not provide any new accident modes. Changing the formulation of the flow-biased APRM scram trip setpoint does not change its respective functions. The APRM scram trip setpoint will continue to initiate the scram if the power flow condition exceeds that specified by the APRM rod block setpoint. Modifying the APRM flow-biased upscale trip does not create any new (1) operating modes, (2) accident scenarios, (3) equipment failure modes, or (4) fission product release paths.

APRM Flow-Biased Neutron Flux-Upscale Rod Block Instrumentation System

The proposed changes remove reference to the APRM rod block instrumentation system from the Technical Specifications. Deletion of the APRM rod block reference does not create any new (1) operating modes, (2) accident scenarios, (3) equipment failure modes, or (4) fission product release paths. The APRM rod block functions are essentially backup functions and, while not used in any licensing basis event, will still be functional to assist operators during certain transients. In addition, the RBM will still actuate to terminate control rod withdrawal errors. Therefore, the effect of the proposed change on operator action to correct unexpected situations is insignificant.

Editorial Change

Editorial changes by their nature do not change the intent or interpretation of the Technical Specifications. The proposed change addresses only capitalization. The proposed change has no effect on any accident, analyzed or unanalyzed.

The aggregate affect of these proposed changes has been evaluated and found to have no resulting impact on system reliability or performance. Thus, the proposed changes do not adversely affect the response of any component or system to previously analyzed accidents. The response to previously evaluated accidents remains within previously assessed limits of temperature and pressure. Further, all safety-related systems and components remain within their applicable design limits. Thus, system and component performance is not adversely affected by these changes, thereby assuring that the design capabilities of those systems and components are not challenged in a manner not previously assessed so as to create the possibility of a new or different kind of

accident. Therefore, operation of Nine Mile Point Unit 2, in accordance with the proposed amendment, will not create the possibility of a new or different kind of accident from any previously assessed.

The operation of Nine Mile Point Unit 2, in accordance with the proposed amendment, will not involve a significant reduction in a margin of safety.

APRM Flow-Biased Simulated Thermal Power Upscale Scram Setpoint

The proposed change will facilitate utilization of the expanded operating domain by relaxing the restrictions imposed by the formulation of the APRM flow-biased simulated thermal power scram trip setpoint. The transient analyses for NMP2 use a fixed analytical value of 117% for the APRM flow-biased trip, corresponding to the maximum nominal trip setpoint of 113.5%. Since the analytical value of 117% is not changing and remains below the fixed neutron flux trip value, the proposed change does not impact the results of any transient analyses. In addition, the APRM flow-biased trip setpoint remains below the APRM Fixed Neutron Flux - Upscale trip setpoint. MCPR values calculated based on the current setpoint bound those for the proposed setpoint. Since MCPR operating and safety limits are not affected, there is no significant decrease in any margin of safety.

APRM Flow-Biased Neutron Flux-Upscale Rod Block Instrumentation System

Relocation of the APRM rod block references will not alter plant response to any transient. Fast flux transients, such as MSIV closure, are terminated by the APRM scram without any operator action and therefore are not affected by this change. The APRM flow-biased rod block will still be available to respond to slow pressurization transients such as loss of feedwater heating and provide time for proper operator action. However, should operators fail to respond or the rod block fail to actuate, the APRM scram will still terminate the transient before any safety limits are impacted. Other events such as a general operator error of withdrawing control rods beyond the upper rod line produce only very mild power increases. The downscale and neutron flux upscale, startup (i.e., setdown) rod blocks can alert the operator to these conditions, however the primary protection for local power effects is provided by the RBM. In addition, the APRM inoperative rod block actuates concurrent with the APRM scram. Therefore, any transient which would result in an APRM inoperative rod block would also initiate a scram signal. Since operability of the APRM scram functions and the RBM is still assured under these proposed changes, MCPR operating and safety limits are not affected. Therefore, the proposed change will not involve a reduction in a margin of safety.

Editorial Change

The wording of the Technical Specifications has not changed. The proposed change addresses only capitalization. Editorial changes by their nature do not change the intent or interpretation of the Technical Specifications and do not affect any margin of safety.

The aggregate affect of these proposed changes has been evaluated and found to have no impact on plant response to transients and accidents. The proposed changes do not affect the basis for any Technical Specification and previously established safety limits remain valid. Therefore, the operation of Nine Mile Point Unit 2, in accordance with the proposed amendment, will not involve a significant reduction in a margin of safety.