ATTACHMENT A
NIAGARA MOHAWK POWER CORPORATION
LICENSE NO. DPR-63
DOCKET NO. 50-220

PROPOSED CHANGES TO TECHNICAL SPECIFICATIONS

Replace existing pages with the attached revised and additional pages. These pages have been retyped in their entirety with marginal markings to indicate changes to the text.

Pages changed

191-195
196a
197-201
203
204
Add page 204a
205-210
212
212a
214-217
221
223
225
225a
228
230-232a
237
237a
Add page 237b
241ff
241hh
241ii

The attached revised pages 191 and 196a do not reflect changes contained in Niagara Mohawk's Technical Specification submittal for reactor coolant pressure testing requirements dated September 25, 1992.

Table 3.6.2a

INSTRUMENTATION THAT INITIATES SCRAM

Limiting Condition for Operation

	Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels per Operable Trip System	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable			
					Shutdown	Refuel	Startup	Run
(1)	Manual Scram	2	1			x	x	x
(2)	High Reactor Pressure	2	2(o)	≤ 1080 psig		x	x	x
(3)	High Drywell Pressure	2	2(o)	≤ 3.5 psig		x	(a)	(a)
(4)	Low Reactor Water Level	2	2(o)	≥ 53 inches (Indicator Scale)		x	x	x
(5)	High Water Level Scram Discharge Volume	2	2(o)	≤ 45 gal.		(b)	x	x

Table 3.6.2a (cont'd)

INSTRUMENTATION THAT INITIATES SCRAM

Limiting Condition for Operation

	Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels per Operable Trip System	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable			
					Shutdown	Refuel	Startup	Run
(6)	Main-Steam-Line Isolation Valve Position	2	4(h)(o)	≤ 10 percent valve closure from full open		(c)	(c)	x
(7)	High Radiation Main-Steam-Line	2	2(o)	≤ 5 times normal background at rated power(n)		x	x	x
(8)	Shutdown Position of Reactor Mode Switch	2	1	---		(k)	x	x
(9)	Neutron Flux (a) IRM (i) Upscale	2	3(d)(o)	≤ 96 percent of full scale		(g)	(g)	(g)

Table 3.6.2a (cont'd)

INSTRUMENTATION THAT INITIATES SCRAM

Limiting Condition for Operation

Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels per Operable Trip System	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable			
				Shutdown	Refuel	Startup	Run
(ii) Inoperative	2	3(d)(o)	---		x	x	
(b) APRM							
(i) Upscale	2	3(e)(o)	Figure 2.1.1		x	x	x
(ii) Inoperative	2	3(e)(o)	---		x	x	x
(iii) Downscale	2	3(e)(o)	≥ 5 percent of full scale		(g)	(g)	(g)
(10) Turbine Stop Valve Closure	2	4(o)	≤ 10% valve closure				(i)
(11) Generator Load Rejection	2	2(o)	(j)				(i)

Table 4.6.2a

INSTRUMENTATION THAT INITIATES SCRAM

Surveillance Requirement

	Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
(1)	Manual Scram	None	Once per week	None
(2)	High Reactor Pressure	None	Once per 3 months (l)	Once per 3 months(l)
(3)	High Drywell Pressure	None	Once per 3 months (l)	Once per 3 months(l)
(4)	Low Reactor Water Level	Once/day	Once per 3 months (l)	Once per 3 months(l)
(5)	High Water Level Scram Discharge Volume	None	Once per 3 months	Once per 3 months
(6)	Main-Steam-Line Isolation Valve Position	None	Once per 3 months	Once per operating cycle
(7)	High Radiation Main-Steam Line	Once/shift	Once per 3 months	Once per 3 months

Table 4.6.2a (cont'd)

INSTRUMENTATION THAT INITIATES SCRAM

Surveillance Requirement

Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
(8) Shutdown Position of Reactor Mode Switch	None	Once during each major refueling outage	None
(9) Neutron Flux			
(a) IRM			
(i) Upscale	(f)	(f)	(f)
(ii) Inoperative	(f)	(f)	(f)
(b) APRM			
(i) Upscale	None	Once per 3 months	Once per week$^{(m)}$ Once per 3 months
(ii) Inoperative	None	Once per 3 months	None
(iii) Downscale	None	Once per 3 months	Once per week$^{(m)}$ Once per 3 months
(10) Turbine Stop Valve Closure	None	Once per 3 months	Once per operating cycle
(11) Generator Load Rejection	None	Once per 3 months	Once per 3 months

(n) Within 24 hours prior to the planned start of the hydrogen injection test with the reactor power at greater than 20% rated power, the normal full-power radiation background level and associated trip and alarm setpoints may be changed based on a calculated value of the radiation level expected during the test. The background radiation level and associated trip and alarm setpoints may be adjusted during the test program based on either calculations or measurements of actual radiation levels resulting from hydrogen injection. The background radiation level shall be determined and associated trip and alarm setpoints shall be reset within 24 hours of re-establishing normal radiation levels after completion of the hydrogen injection or within 12 hours of establishing reactor power levels below 20% rated power, while these functions are required to be operable. At reactor power levels below 20% rated power, hydrogen injection shall be terminated and the injection system secured.

(o) A channel may be placed in an inoperable status for up to 6 hours for required surveillances without placing the Trip System in the tripped condition provided at least one Operable Instrument Channel in the same trip system is monitoring that parameter.

With one channel required by Table 3.6.2a inoperable in one or more Parameters, place the inoperable channel and/or that trip system in the tripped condition* within 12 hours.

With two or more channels required by Table 3.6.2a inoperable in one or more Parameters:

1. Within one hour, verify sufficient channels remain Operable or tripped* to maintain trip capability for the Parameter, and

2. Within 6 hours, place the inoperable channel(s) in one trip system and/or that trip system** in the tripped condition*, and

3. Within 12 hours, restore the inoperable channels in the other trip system to an Operable status or tripped*.

Otherwise, take the ACTION required by Specification 3.6.2a for that Parameter.

* An inoperable channel or trip system need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, if the inoperable channel is not restored to Operable status within the required time, the ACTION required by Specification 3.6.2a for the parameter shall be taken.

** This ACTION applies to that trip system with the most inoperable channels; if both trip systems have the same number of inoperable channels, the ACTION can be applied to either trip system.

Table 3.6.2b

INSTRUMENTATION THAT INITIATES
PRIMARY COOLANT SYSTEM OR CONTAINMENT ISOLATION

Limiting Condition for Operation

Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels per Operable Trip System	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable			
				Shutdown	Refuel	Startup	Run
PRIMARY COOLANT ISOLATION							
(Main Steam, Cleanup, and Shutdown)							
(1) Low-Low Reactor Water Level	2	2(f)	≥ 5 inches (Indicator Scale)			x	x
(2) Manual	2	1	---	x	x	x	x
MAIN-STEAM-LINE ISOLATION							
(3) High Steam Flow Main-Steam Line	2	2(f)	≤ 105 psid			x	x

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Table 3.6.2b (cont'd)

INSTRUMENTATION THAT INITIATES PRIMARY COOLANT SYSTEM OR CONTAINMENT ISOLATION

Limiting Condition for Operation

Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels per Operable Trip System	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable: Shutdown	Refuel	Startup	Run
(4) High Radiation Main Steam Line	2	2(f)	$\leq$ 5 times normal background at rated power[(e)]			x	x
(5) Low Reactor Pressure	2	2(f)	$\geq$ 850 psig				x
(6) Low-Low-Low Condenser Vacuum	2	2(f)	$\geq$ 7 in. mercury vacuum			(a)	x
(7) High Temperature Main Steam Line Tunnel	2	2(f)	$\leq$ 200F			x	x

Table 3.6.2b (cont'd)

INSTRUMENTATION THAT INITIATES PRIMARY COOLANT SYSTEM OR CONTAINMENT ISOLATION

Limiting Condition for Operation

Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels per Operable Trip System	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable: Shutdown	Refuel	Startup	Run
CLEANUP SYSTEM ISOLATION							
(8) High Area Temperature	1	2(g)	≤ 190°F	x	x	x	x
SHUTDOWN COOLING SYSTEM ISOLATION							
(9) High Area Temperature	1	1	≤ 170°F	x	x	x	x
CONTAINMENT ISOLATION							
(10) Low-Low Reactor Water	2	2(f)	≥ 5 inches (Indicator Scale)	(c)		x	x

Table 3.6.2b (cont'd)

INSTRUMENTATION THAT INITIATES
PRIMARY COOLANT SYSTEM OR CONTAINMENT ISOLATION

Limiting Condition for Operation

Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels per Operable Trip System	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable			
				Shutdown	Refuel	Startup	Run
(11) High Drywell Pressure	2	2(f)	≤ 3.5 psig	(c)		(b)	(b)
(12) Manual	2	1	---	x	x	x	x

Table 4.6.2b

INSTRUMENTATION THAT INITIATES
PRIMARY COOLANT SYSTEM OR CONTAINMENT ISOLATION

Surveillance Requirement

Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
PRIMARY COOLANT ISOLATION			
(Main Steam, Cleanup and Shutdown)			
(1) Low-Low Reactor Water Level	Once/day	Once per 3 months(d)	Once per 3 months(d)
(2) Manual	---	Once during each major refueling outage	---
MAIN-STEAM-LINE ISOLATION			
(3) High Steam Flow Main-Steam Line	Once/day	Once per 3 months(d)	Once per 3 months(d)
(4) High Radiation Main-Steam Line	Once/shift	Once per 3 months	Once per 3 months
(5) Low Reactor Pressure	Once/day	Once per 3 months(d)	Once per 3 months(d)

Table 4.6.2b (cont'd)

INSTRUMENTATION THAT INITIATES
PRIMARY COOLANT SYSTEM OR CONTAINMENT ISOLATION

Surveillance Requirement

Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
CONTAINMENT ISOLATION			
(10) Low-Low Reactor Water Level	Once/day	Once per 3 months[d]	Once per 3 months[d]
(11) High Drywell Pressure	Once/day	Once per 3 months[d]	Once per 3 months[d]
(12) Manual	---	Once during each operating cycle	---

NOTES FOR TABLES 3.6.2b and 4.6.2b

(a) May be bypassed in the refuel and startup positions of the reactor mode switch when reactor pressure is less than 600 psi.

(b) May be bypassed when necessary for containment inerting.

(c) May be bypassed in the shutdown mode whenever the reactor coolant system temperature is less than 215°F.

(d) Only the trip circuit will be calibrated and tested at the frequencies specified in Table 4.6.2b, the primary sensor will be calibrated and tested once per operating cycle.

(e) Within 24 hours prior to the planned start of the hydrogen injection test with the reactor power at greater than 20% rated power, the normal full-power radiation background level and associated trip and alarm setpoints may be changed based on a calculated value of the radiation level expected during the test. The background radiation level and associated trip and alarm setpoints may be adjusted during the test program based on either calculations or measurements of actual radiation levels resulting from hydrogen injection. The background radiation level shall be determined and associated trip and alarm setpoints shall be reset within 24 hours of re-establishing normal radiation levels after completion of the hydrogen injection or within 12 hours of establishing reactor power levels below 20% rated power, while these functions are required to be operable. At reactor power levels below 20% rated power hydrogen injection shall be terminated and the injection system secured.

(f) A channel may be placed in an inoperable status for up to 6 hours for required surveillances without placing the Trip System in the tripped condition provided at least one Operable Instrument Channel in the same Trip System is monitoring that Parameter.

With the number of Operable Channels less than required by the Minimum Number of Operable Instrument Channels per Operable Trip System requirement for one trip system, either

1. Place the inoperable channel(s) in the tripped condition within

   a. 12 hours for Parameters common to SCRAM Instrumentation, and
   b. 24 hours for Parameters not common to SCRAM Instrumentation.

or

2. Take the ACTION required by Specification 3.6.2a for that Parameter.

(f) (cont'd)

With the number of Operable Channels less than required by the Minimum Number of Operable Instrument Channels per Operable Trip System requirement for both trip systems,

1. Place the inoperable channel(s) in one trip system in the tripped condition within one hour,

and

2. a. Place the inoperable channel(s) in the remaining trip system in the tripped condition within

   (1) 12 hours for Parameters common to SCRAM Instrumentation, and
   (2) 24 hours for Parameters not common to SCRAM Instrumentation.

   or

   b. take the ACTION required by Specification 3.6.2a for that Parameter.

(g) A channel may be placed in an inoperable status for up to 6 hours for required surveillances without placing the Trip System in the tripped condition provided at least one Operable Instrument Channel in the same Trip System is monitoring that Parameter.

With the number of Operable channels less than required by the Minimum Number of Operable Instrument Channels for the Operable Trip System, either

1. Place the inoperable channel(s) in the tripped condition within 24 hours.

or

2. Take the ACTION required by Specification 3.6.2a for that Parameter.

Table 3.6.2c

INSTRUMENTATION THAT INITIATES OR ISOLATES EMERGENCY COOLING

Limiting Condition for Operation

Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels per Operable Trip System (d)	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable			
				Shutdown	Refuel	Startup	Run
EMERGENCY COOLING INITIATION							
(1) High-Reactor Pressure	2	2(e)	≤ 1080 psig	(b)		x	x
(2) Low-Low Reactor Water Level	2	2(e)	≥ 5 inches (Indicator Scale)	(b)		x	x
EMERGENCY COOLING ISOLATION (for each of two systems)							
(3) High Steam Flow Emergency Cooling System	2	2(a)(f)	≤ 11.5 psid			x	x

Table 4.6.2c

INSTRUMENTATION THAT INITIATES
OR ISOLATES EMERGENCY COOLING

Surveillance Requirement

Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
EMERGENCY COOLING INITIATION			
(1) High Reactor Pressure	None	Once per 3 months[c]	Once per 3 months[c]
(2) Low-Low Reactor Water Level	Once/day	Once per 3 months[c]	Once per 3 months[c]
EMERGENCY COOLING ISOLATION (for each of two systems)			
(3) High Steam Flow Emergency Cooling System	None	Once per 3 months[c]	Once per 3 months[c]

NOTES FOR TABLES 3.6.2c and 4.6.2c

(a) Each of two differential pressure switches provide inputs to one instrument channel in each trip system.

(b) May be bypassed in the cold shutdown condition.

(c) Only the trip circuit will be calibrated and tested at the frequencies specified in Table 4.6.2c, the primary sensor will be calibrated and tested once per operating cycle.

(d) A channel may be placed in an inoperable status for up to 6 hours for required surveillances without placing the Trip System in the tripped condition provided at least one Operable Instrument Channel in the same Trip System is monitoring that parameter.

(e) With the number of Operable channels less than required by the Minimum Number of Operable Instrument Channels per Operable Trip System requirement:

1. For one channel inoperable, place the inoperable channel in the tripped condition within 24 hours or take the action required by Specification 3.6.2a for that Parameter.
2. With more than one channel inoperable, take the ACTION required by Specification 3.6.2a for that Parameter.

(f) With the number of Operable channels less than required by the Minimum Number of Operable Instrument Channels per Operable Trip System requirement for one trip system, either

1. Place the inoperable channel(s) in the tripped condition within 24 hours.

or

2. Take the ACTION required by Specification 3.6.2a for that Parameter.

With the number of Operable channels less than required by the Minimum Number of Operable Instrument Channels per Operable Trip System requirement for both trip systems,

1. Place the inoperable channel(s) in one trip system in the tripped condition within one hour

and

2. a. Place the inoperable channel(s) in the remaining trip system in the tripped condition within 24 hours.

   or

   b. Take the ACTION required by Specification 3.6.2a for that Parameter.

Table 3.6.2d

INSTRUMENTATION THAT INITIATES CORE SPRAY[e]

Limiting Condition for Operation

Parameter		Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels per Operable Trip System (f)	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable: Shutdown	Refuel	Startup	Run
START CORE SPRAY PUMPS								
(1)	High Drywell Pressure	2	2	≤ 3.5 psig	(d)	x	(a)	(a)
(2)	Low-Low Reactor Water Level	2	2	≥ 5 inches (Indicator Scale)	(b)	x	x	x
OPEN CORE SPRAY DISCHARGE VALVES								
(3)	Reactor Pressure and either (1) or (2) above.	2	2	≥ 365 psig	x	x	x	x

Table 4.6.2d

INSTRUMENTATION THAT INITIATES CORE SPRAY

Surveillance Requirement

Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
START CORE SPRAY PUMPS			
(1) High Drywell Pressure	Once/day	Once per 3 months(c)	Once per 3 months(c)
(2) Low-Low Reactor Water Level	Once/day	Once per 3 months(c)	Once per 3 months(c)
OPEN CORE SPRAY DISCHARGE VALVES			
(3) Reactor Pressure and either (1) or (2) above	None	Once per 3 months(c)	Once per 3 months(c)

NOTES FOR TABLES 3.6.2d and 4.6.2d

(a) May be bypassed when necessary for containment inerting.

(b) May be bypassed when necessary for performing major maintenance as specified in Specification 2.1.1.e.

(c) Only the trip circuit will be calibrated and tested at the frequencies specified in Table 4.6.2d, the primary sensor will be calibrated and tested once per operating cycle.

(d) May be bypassed when necessary for integrated leak rate testing.

(e) The instrumentation that initiates the Core Spray System is not required to be operable, if there is no fuel in the reactor vessel.

(f) A channel may be placed in an inoperable status for up to 6 hours for required surveillances without placing the Trip System in the tripped condition provided at least one Operable Instrument Channel in the same Trip System is monitoring that parameter.

With the number of Operable channels less than required by the Minimum Number of Operable Instrument Channels per Operable Trip System requirement:

1. With one channel inoperable, place the inoperable channel in the tripped condition within 24 hours or take the ACTION required by Specification 3.6.2a for that Parameter.
2. With more than one channel inoperable, take the ACTION required by Specification 3.6.2a for that Parameter.

Table 3.6.2e

INSTRUMENTATION THAT INITIATES CONTAINMENT SPRAY

Limiting Condition for Operation

Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels per Operable Trip System (c)	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable			
				Shutdown	Refuel	Startup	Run
(1)a. High Drywell Pressure	2	2	≤ 3.5 psig	(a)		x	x
and							
b. Low-Low Reactor Water Level	2	2	≥ 5 inches (Indicator Scale)	(a)		x	x

Table 4.6.2e

INSTRUMENTATION THAT INITIATES CONTAINMENT SPRAY

Surveillance Requirement

Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
(1)a. High Drywell Pressure	Once/day	Once per 3 months[b]	Once per 3 months[b]
b. Low-Low Reactor Water Level	Once/day	Once per 3 months[b]	Once per 3 months[b]

NOTES FOR TABLES 3.6.2e and 4.6.2e

(a) May be bypassed in the shutdown mode whenever the reactor coolant temperature is less than 215°F.

(b) Only the trip circuit will be calibrated and tested at the frequencies specified in Table 4.6.2e, the primary sensor will be calibrated and tested once per operating cycle.

(c) A channel may be placed in an inoperable status for up to 6 hours for required surveillances without placing the Trip system in the tripped condition provided at least one Operable Instrument Channel in the same Trip System is monitoring that parameter.

With the number of Operable channels less than required by the Minimum Number of Operable Instrument Channels per Operable Trip System requirement:

1. With one channel inoperable, place the inoperable channel in the tripped condition within 24 hours or take the ACTION required by Specification 3.6.2a for that Parameter.

2. With more than one channel inoperable, take the ACTION required by Specification 3.6.2a for that Parameter.

Table 3.6.2f

INSTRUMENTATION THAT INITIATES AUTO DEPRESSURIZATION

Limiting Condition for Operation

Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels per Operable Trip System (d)	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable: Shutdown	Refuel	Startup	Run
INITIATION							
(1)a. Low-Low-Low Reactor Water Level	2(a)	2(a)	≥ -10 inches * (Indicator Scale)	(b)		(b)	x
and							
b. High Drywell Pressure	2(a)	2(a)	≤ 3.5 psig	(b)		(b)	x

* greater than (≥) means less negative

Table 4.6.2f

INSTRUMENTATION THAT INITIATES AUTO DEPRESSURIZATION

Surveillance Requirement

Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
(1)a. Low-Low-Low Reactor Water	None	Once per 3 months(c)	Once per 3 months(c)
and			
b. High Drywell Pressure	Once/day	Once per 3 months(c)	Once per 3 months(c)

NOTES FOR TABLES 3.6.2f and 4.6.2f

(a) Both instrument channels in either trip system are required to be energized to initiate auto depressurization. One trip system is powered from power board 102 and the other trip system from power board 103.

(b) May be bypassed when the reactor pressure is less than 110 psig and the reactor coolant temperature is less than the corresponding saturation temperature.

(c) Only the trip circuit will be calibrated and tested at the frequencies specified in Table 4.6.2f, the primary sensor will be calibrated and tested once per operating cycle.

(d) A channel may be placed in an inoperable status for up to 6 hours for required surveillances without placing the Trip System in the tripped condition provided at least one operable channel in the same Trip System is monitoring that parameter.

With the number of Operable channels less than required by the Minimum Number of Operable Instrument Channels per Operable Trip System requirement:

1. With one channel inoperable, place the inoperable channel in the tripped condition within 24 hours or take the ACTION required by Specification 3.6.2a for that Parameter.

2. With more than one channel inoperable, take the ACTION required by Specification 3.6.2a for that Parameter.

Table 3.6.2g

INSTRUMENTATION THAT INITIATES CONTROL ROD WITHDRAWAL BLOCK

Limiting Condition for Operation

Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels per Operable Trip System (i)	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable: Shutdown	Refuel	Startup	Run
(1) SRM							
a. Detector not in Startup Position	2	2(a)(e)	---		x	x	
b. Inoperative	2	2(a)	---		x	x	
c. Upscale	2	2(a)	$\leq 10^5$ counts/sec		x	x	
(2) IRM							
a. Detector not in Startup Position	2	3(b)	---		x	x	
b. Inoperative	2	3(b)	---		x	x	

Table 3.6.2g (cont'd)

INSTRUMENTATION THAT INITIATES CONTROL ROD WITHDRAWAL BLOCK

Limiting Condition for Operation

Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels per Operable Trip System (i)	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable: Shutdown	Refuel	Startup	Run
c. Downscale	2	3(b)	≤ 5 percent of full scale for each scale		x	x	
d. Upscale	2	3(b)	≤ 88 percent of full scale for each scale		x	x	
(3) APRM							
a. Inoperative	2(h)	3(c)	---		x	x	x
b. Upscale (Biased by Recirculation Flow)	2(h)	3(c)	Figure 2.1.1(h)		x	x	x
c. Downscale	2(h)	3(c)	≥ 2 percent of full scale		(d)	(d)	x

Table 3.6.2g (cont'd)

INSTRUMENTATION THAT INITIATES CONTROL ROD WITHDRAWAL BLOCK

Limiting Condition for Operation

Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels per Operable Trip System	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable: Shutdown	Refuel	Startup	Run
(4) Recirculation Flow							
a. Comparator Off Normal	2	1	---		x	x	x
b. Flow Unit Inoperative	2	1	---		x	x	x
c. Flow Unit Upscale	2	1	Figure 2.1.1		x	x	x
(5) Refuel Platform and Hoists	2(f)	1	---		x		
(6) Mode switch in Shutdown	1	1	---	x			

Table 3.6.2g (cont'd)

INSTRUMENTATION THAT INITIATES CONTROL ROD WITHDRAWAL BLOCK

Limiting Condition for Operation

	Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels per Operable Trip System	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable			
					Shutdown	Refuel	Startup	Run
(7)	Mode Switch in Refuel (Blocks withdrawal of more than 1 rod)	1	1	---		x		
(8)	Scram Dump Volume Water Level Scram Bypass	2	1	---	x	x		

Table 4.6.2g (cont'd)

INSTRUMENTATION THAT INITIATES CONTROL ROD WITHDRAWAL BLOCK

Surveillance Requirement

Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
(3) APRM			
a. Inoperative	None	Once per 3 months	None
b. Upscale (Biased by Recirculation Flow)	None	Once per 3 months	Once per 3 months
c. Downscale	None	Once per 3 months	Once per 3 months
(4) Recirculation Flow			
a. Comparator Off Normal	None	Once per 3 months	Once per 3 months
b. Flow Unit Inoperative	None	Once per 3 months	Once per 3 months
c. Flow Unit Upscale	None	Once per 3 months	Once per 3 months

NOTES FOR TABLES 3.6.2g AND 4.6.2g

(a) No more than one of the four SRM inputs to the single trip system shall be bypassed.

(b) No more than one of the four IRM inputs to each instrument channel shall be bypassed. These signals may be bypassed when the APRM's are onscale.

(c) No more than one of the four APRM inputs to each instrument channel shall be bypassed provided that the APRM in the other instrument channel in the same core quadrant is not bypassed. No more than two C or D level LPRM inputs to an APRM shall be bypassed and only four LPRM inputs to only one APRM shall be bypassed in order for the APRM to be considered operable. In the Run mode of operation, bypass of two chambers from one radial core location in any one APRM shall cause that APRM to be considered inoperative. A Travelling In-Core Probe (TIP) chamber may be used as a substitute APRM input if the TIP is positioned in close proximity to the failed LPRM it is replacing. If one APRM in a quadrant is bypassed and meets all requirements for operability with the exception of the requirement of at least one operable chamber at each radial location, it may be returned to service and the other APRM in that quadrant may be removed from service for test and/or calibration only if no control rod is withdrawn during the calibration and/or test.

(d) May be bypassed in the startup and refuel positions of the reactor mode switch when the IRM's are onscale.

(e) This function may be bypassed when the count rate is ≥ 100 cps.

(f) One sensor provides input to each of two instrument channels. Each instrument channel is in a separate trip system.

(g) Calibrate and/or test prior to startup and normal shutdown. Thereafter test once per week until no longer required.

(h) The actuation of either or both trip systems will result in a rod block.

(i) A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the Trip System in the tripped condition, provided at least one other operable channel in the same Trip System is monitoring that Parameter.

Table 3.6.2h

VACUUM PUMP ISOLATION

Limiting Condition for Operation

Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels per Operable Trip System (b)	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable			
				Shutdown	Refuel	Startup	Run
MECHANICAL VACUUM PUMP							
High Radiation Main Steam Line	2	2	$\leq$ 5 times normal[(a)] background		x	x	x

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Table 4.6.2h

VACUUM PUMP ISOLATION

Surveillance Requirement

Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
MECHANICAL VACUUM PUMP			
High Radiation Main Steam Line	Once/shift	Once per 3 months	Once per 3 months

NOTES FOR TABLES 3.6.2h and 4.6.2h

(a) Within 24 hours prior to the planned start of the hydrogen injection test with the reactor power at greater than 20% rated power, the normal full-power radiation background level and associated trip and alarm setpoints may be changed based on a calculated value of the radiation level expected during the test. The background radiation level and associated trip and alarm setpoints may be adjusted during the test program based on either calculations or measurements of actual radiation levels resulting from hydrogen injection. The background radiation level shall be determined and associated trip and alarm setpoints shall be reset within 24 hours of re-establishing normal radiation levels after completion of the hydrogen injection or within 12 hours of establishing reactor power levels below 20% rated power, while these functions are required to be operable. At reactor power levels below 20% rated power hydrogen injection shall be terminated and the injection system secured.

(b) A channel may be placed in an inoperable status for up to 6 hours for required surveillances without placing the Trip System in the tripped condition provided at least one operable channel in the same Trip System is monitoring that parameter.

With the number of Operable channels less than required by the Minimum Number of Operable Instrument Channels per Operable Trip System requirement for one trip system, either

1. Place the inoperable channel(s) in the tripped condition within 12 hours.
or
2. Take the ACTION required by Specification 3.6.2a for that Parameter.

With the number of Operable channels less than required by the Minimum Number of Operable Instrument Channels per Operable Trip System requirement for both trip systems,

1. Place the inoperable channel(s) in one trip system in the tripped condition within one hour.
and
2. a. Place the inoperable channel(s) in the remaining trip system in the tripped condition within 12 hours.
   or
   b. Take the ACTION required by Specification 3.6.2a for that Parameter.

Table 3.6.2j

EMERGENCY VENTILATION INITIATION

Limiting Condition for Operation

Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels per Operable Trip System	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable: Shutdown	Refuel	Startup	Run
(1) High Radiation Reactor Building Ventilation Duct	1	2(d)	≤ 5mr/hr	x	x	x	x
(2) High Radiation Refueling Platform	1	1	≤ 1000mr/hr	(a)	(a)	(a)	(a)

NOTES FOR TABLES 3.6.2j AND 4.6.2j

(a) This function shall be operable any time that irradiated fuel or the irradiated fuel cask is being handled in the reactor building.

(b) Once per shift whenever this function is required to be operable.

(c) Immediately prior to when function is required and once per week thereafter until function is no longer required.

(d) A channel may be placed in an inoperable status for up to 6 hours for required surveillances without placing the Trip System in the tripped condition provided at least one Operable Instrument Channel in the same Trip system is monitoring that parameter.

With the number of Operable channels less than required by the Minimum Number of Operable Instrument Channels for the Operable Trip System, either

1) Place the inoperable channel(s) in the tripped condition within 24 hours.
or
2) Take the ACTION required by Specification 3.6.2a for that Parameter.

Table 3.6.2k

HIGH PRESSURE COOLANT INJECTION

Limiting Condition for Operation

Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels per Operable Trip System	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable: Shutdown	Refuel	Startup	Run
(1) Low Reactor Water Level	2	2(c)	≥ 53 inches (Indicator Scale)	(a)		(a)	x
(2) Automatic Turbine Trip	1	1	---	(a)		(a)	x

Table 4.6.2k

HIGH PRESSURE COOLANT INJECTION

Surveillance Requirement

	Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
(1)	Low Reactor Water Level	Once per day	Once per 3 months[b]	Once per 3 months[b]
(2)	Automatic Turbine Trip	None	Once during each operating cycle	None

NOTES FOR TABLES 3.6.2k and 4.6.2k

(a) May be bypassed when the reactor pressure is less than 110 psig and the reactor coolant temperature is less than the corresponding saturation temperature.

(b) Only the trip circuit will be calibrated and tested at the frequencies specified in Table 4.6.2k, the primary sensor will be calibrated and tested once per operating cycle.

(c) A channel may be placed in an inoperable status for up to 6 hours for required surveillances without placing the Trip System in the tripped condition provided at least one operable channel in the same Trip System is monitoring that parameter.

With the number of Operable channels less than required by the Minimum Number of Operable Instrument Channels per Operable Trip System requirement:

1. For one channel inoperable, place the inoperable channel in the tripped condition within 24 hours or take the ACTION required by Specification 3.6.2a for that Parameter.

2. With more than one channel inoperable, take the ACTION required by Specification 3.6.2a for that Parameter.

High Flow-Main Steam Line, ±1 psid

High Flow-Emergency Cooling Line, ±1 psid

High Area Temperature-Main Steam Line, ±10F

High Area Temperature-Clean-up and Shutdown, ±6F

High Radiation-Main Steam Line, +100% and -50% of set point value

High Radiation-Emergency Cooling System Vent, +100% and -50% of set point

High Radiation-Reactor Building Vent, +100% and -50% of set point

High Radiation-Refueling Platform, +100% and -50% of set point

High Radiation-Offgas Line, ±50% of set point, (Appendix D)*

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30851P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System," and MDE-77-0485, "Technical Specification Improvement Analysis for Nine Mile Point Nuclear Station, Unit 1."

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30851P-A Suppl2, "Technical Specification Improvement Analyses for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," and with NEDC-31677P-A, "Technical Specification Improvement Analyses for BWR Isolation Actuation Instrumentation." Because of local high radiation, testing instrumentation in the area of the main steam line isolation valves can only be done during periods of Station shutdown. These functions include high area temperature isolation and isolation valve position scram.

* FSAR

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30936P-A, "BWR Owners' Group Technical Specification Improvement Methodology (with Demonstration for BWR ECCS Actuation Instrumentation)," Parts 1 and 2 and RE-003, "Technical Specification Improvement Analysis for the Emergency Core Cooling System Actuation Instrumentation for Nine Mile Point Nuclear Station, Unit 1."

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," as approved by the NRC and documented in the SER (letter to R. D. Binz IV from C. E. Rossi dated July 21, 1992).

Testing of the scram associated with the shutdown position of the mode switch can be done only during periods of Station shutdown since it always involves a scram.

b. The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR is maintained greater than the SLCPR. The trip logic for this function is 1 out of n; e.g., any trip on one of the eight APRM's, eight IRM's or four SRM's will result in a rod block. The minimum instrument channel requirements provide sufficient instrumentation to assure the single failure criteria is met. Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30851P-A Suppl 1, "Technical Specification Improvement Analyses for BWR Control Rod Block Instrumentation," and with GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-Of-Service Times for Selected Instrumentation Technical Specifications," as approved by the NRC and documented in the SER (letter to R. D. Binz IV from C. E. Rossi dated July 21, 1992).

The APRM rod block trip is flow biased and prevents a significant reduction in MCPR especially during operation at reduced flow. The APRM provides gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The trips are set so that MCPR is maintained greater than the SLCPR.

The APRM rod block also provides local protection of the core; i.e., the prevention of critical heat flux in a local region of the core, for a single rod withdrawal error from a limiting control rod pattern. The trip point is flow biased. The worst case single control rod withdrawal error has been analyzed and the results show that with the specified trip settings rod withdrawal is blocked before the MCPR reaches the SLCPR, thus allowing adequate margin. Below ~60% power the worst case withdrawal of a single control rod results in a MCPR > SLCPR without rod block action, thus below this level it is not required.

The IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that trip setting is less than a factor of 10 above the indicated level. Analysis of the worst case accident results in rod block action before MCPR approaches the SLCPR.

A downscale indication on an APRM or IRM is an indication the instrument has failed or the instrument is not sensitive enough. In either case the instrument will not respond to changes in control rod motion and the control rod motion is prevented. The downscale rod blocks are set at 5 percent of full scale for IRM and 2 percent of full scale for APRM (APRM signal is generated by averaging the output signals from eight LPRM flux monitors).

Table 3.6.11-1

ACCIDENT MONITORING INSTRUMENTATION

	Parameter	Total Number of Channels	Minimum Number of Operable Sensors or Channels	Action (See Table 3.6.11-2)
(1)	Relief Valve Position Indication	2/Valve	1/Valve*	1
(2)	Safety Valve Position Indication	2/Valve	1/Valve*	1
(3)	Reactor Vessel Water Level	2	1*	2
(4)	Drywell Pressure Monitor	2	1	4
(5)	Suppression Chamber Water Level	2	1*	4
(6)	Containment Hydrogen Monitor	2	1	4
(7)	Containment High Range Radiation Monitor	2	1	3
(8)	Suppression Chamber Water Temperature	2	1	2

* A channel may be placed in an inoperable status for up to 6 hours for required surveillance provided at least one Operable channel is monitoring that Parameter.

Table 4.6.11

ACCIDENT MONITORING INSTRUMENTATION

Surveillance Requirement

	Parameter	Instrument Channel Test	Instrument Channel Calibration
(1)	Relief Valve Position Indicator (Primary - Acoustic)	Once per quarter	Once during each major refueling outage
	Relief Valve Position Indicator (Backup - Thermocouple)	Once per quarter	Once during each major refueling outage
(2)	Safety Valve Position Indicator (Primary - Acoustic)	Once per quarter	Once during each major refueling outage
	Safety Valve Position Indicator (Backup - Thermocouple)	Once per quarter	Once during each major refueling outage
(3)	Reactor Vessel Water Level	Once per quarter	Once during each major refueling outage
(4)	Drywell Pressure Monitor	Once per month	Once during each major refueling outage
(5)	Suppression Chamber Water Level Monitor	Once per quarter	Once during each major refueling outage
(6)	Containment Hydrogen Monitor	Once per month	Once per quarter
(7)	Containment High Range Radiation Monitor	Once per month	Once during each major refueling outage
(8)	Suppression Chamber Water Temperature	Once per month	Once during each major refueling outage

Accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident. This capability is consistent with the recommendations of NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," and/or NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980 and NUREG 0661, "Safety Evaluation Report Mark I Containment Long Term Program."

The maximum allowable setpoint deviation for the Suppression Chamber Water Level Instrumentation is ± 1.8 inches.

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-Of-Service Times for Selected Instrumentation Technical Specifications," as approved by the NRC and documented in the SER (letter to R. D. Binz IV from C. E. Rossi dated July 21, 1992).

ATTACHMENT B
NIAGARA MOHAWK POWER CORPORATION
LICENSE NO. DPR-63
DOCKET NO. 50-220

SUPPORTING INFORMATION AND NO SIGNIFICANT HAZARDS CONSIDERATION EVALUATION

I. INTRODUCTION

The proposed changes reflect those standard Technical Specification (TS) revisions contained in the GE Topical Reports which, based upon probabilistic analyses, justify the identified time extensions by reducing: i) potential unnecessary plant scrams, ii) excessive equipment test cycles, and iii) the diversion of personnel and resources on unnecessary testing. The NRC staff has reviewed and approved these Licensing Topical Reports in the letters, and accompanying Safety Evaluation Reports (SERs), from A. C. Thadani (NRC) to T. A. Pickens (BWR Owners' Group) dated July 15, 1987; from Charles E. Rossi (NRC) to D. N. Grace (BWR Owners' Group) dated January 6, 1989; from Charles E. Rossi (NRC) to S. D. Floyd (BWR Owners' Group) dated June 18, 1990; from A. C. Thadani (NRC) to D. N. Grace (BWR Owners' Group) dated December 9, 1988; from Charles E. Rossi (NRC) to D. N. Grace (BWR Owners' Group) dated December 9, 1988; from Charles E. Rossi (NRC) to D. N. Grace (BWR Owners' Group) dated September 22, 1988; and from Charles E. Rossi (NRC) to R. D. Binz IV (BWR Owners' Group) dated July 21, 1992.

II. DESCRIPTION OF THE PROPOSED CHANGES

Revise the Reactor Protection System (RPS, Scram), Isolation Actuation, ECCS Actuation, Control Rod Block and Selected Instrumentation Technical Specification (TS) requirements (TS 3.6.2 and TS 3.6.11) regarding the surveillance test intervals (STIs) and allowable out-of-service times (AOTs) in accordance with General Electric Company (GE) Licensing Topical Reports (LTRs) NEDC-30851P-A dated March 1988, NEDC-30851P-A (Suppl 2) dated March 1989, NEDC 31677P-A dated July 1990, NEDC-30936P-A (Parts 1 and 2) dated December 1988, NEDC 30851P-A (Suppl 1) dated October 1988 and GENE-770-06-1 dated February 1991. Also, AOTs are clarified in accordance with most recently approved BWR Owners' Group letters. In summary, changes requested are the following:

NEDC-30851P-A dated March 1988/MDE-77-0485 dated April 1985

1. Revise and clarify[1] note (o), which includes allowable out-of-service times for specified scram Parameters for required surveillance or repair (TS Table 3.6.2a and NOTES FOR TABLES 3.6.2a AND 4.6.2a).

2. Editorial - add "Upscale" to Parameter (9)(a)(i) in Table 3.6.2a.

3. Decrease the Instrument Channel Test interval requirement in TS Table 4.6.2a for Manual Scram from quarterly to weekly. Parameter (1)

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[1] BWROG 92102 letter from C. L. Tully to B. K. Grimes (NRC), "BWR Owners' Group (BWROG) Topical Reports on Technical Specification Improvement Analysis for BWR Reactor Protection Systems - Use For Relay and Solid State Plants (NEDC-30884 and NEDC-30851P)," dated November 4, 1992.

4. Increase the Instrument Channel Test interval requirement in TS Table 4.6.2a from weekly or monthly to quarterly for the following Parameters:

   a. High Reactor Pressure - Parameter (2)
   b. High Drywell Pressure - Parameter (3)
   c. Low Reactor Water Level - Parameter (4)
   d. High Water Level Scram Discharge Volume - Parameter (5)
   e. Main-Steam-Line Isolation Valve Position - Parameter (6)
   f. High Radiation Main-Steam Line - Parameter (7)
   g. APRM Upscale - Parameter (9)(b)(i)
      APRM Inoperative - Parameter (9)(b)(ii)
      APRM Downscale - Parameter (9)(b)(iii)
   h. Generator Load Rejection - Parameter (11)

<u>NEDC-30851P-A Suppl 2 dated March 1989/NEDC-31677P-A dated July 1990</u>

5. Revise/clarify[2] note (f) and add/clarify[2] note (g), which includes allowable out-of-service times for specified Isolation Parameters for required surveillance or repair (TS Table 3.6.2b and NOTES FOR TABLES 3.6.2b AND 4.6.2b).

6. Editorial - add units (°F) to Parameters (8) and (9), "High Area Temperature," in Table 3.6.2b.

7. Increase the Instrument Channel Test interval requirement in TS Table 4.6.2b from weekly or monthly to quarterly for the following Parameters:

   a. <u>PRIMARY COOLANT ISOLATION</u>

      1) Low-Low Reactor Water Level - Parameter (1)

   b. <u>MAIN-STEAM-LINE ISOLATION</u>

      1) High Steam Flow Main Steam Line - Parameter (3)
      2) High Radiation Main Steam Line - Parameter (4)
      3) Low Reactor Pressure - Parameter (5)

   c. <u>CONTAINMENT ISOLATION</u>

      1) Low-Low Reactor Water Level - Parameter (10)
      2) High Drywell Pressure - Parameter (11)

<u>NEDC-30936P-A (Parts 1 and 2) dated December 1988/NEDC-31677P-A dated July 1990/RE-003 dated January 1987</u>

8. Add/clarify[2] notes (e) and (f), which includes allowable out-of-service times for specified <u>INITIATION/ISOLATION EMERGENCY COOLING</u> Parameters for required repair (Table 3.6.2c and NOTES FOR TABLES 3.6.2c AND 4.6.2c).

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[2] 0G90-579-32A, letter to M. L. Wohl (NRC) from W. P. Sullivan and J. F. Klapproth (GE), "Implementation Enhancements to Technical Specification Changes Given in Isolation Actuation Instrumentation Analysis," dated June 25, 1990.

9. Increase the Instrument Channel Test interval requirement in TS Table 4.6.2c from monthly to quarterly for the following Parameters:

    EMERGENCY COOLING INITIATION

    1) High Reactor Pressure - Parameter (1)
    2) Low-Low Reactor Water Level - Parameter (2)

10. Revise note (d), which includes allowable out-of-service times for specified INITIATION/ISOLATION EMERGENCY COOLING Parameters for required surveillance (NOTES FOR TABLES 3.6.2c AND 4.6.2c).

NEDC-30936P-A (Parts 1 and 2) dated December 1988/RE-003 dated January 1987

11. Delete old note (f) which applied to 1986 License condition (TS Table 3.6.2d and NOTES FOR TABLES 3.6.2d AND 4.6.2d).

12. Add new note (f), which includes allowable out-of-service times for specified CORE SPRAY INITIATION Parameters for required surveillance or repair (Table 3.6.2d and NOTES FOR TABLES 3.6.2d AND 4.6.2d).

13. Delete note (g), as note (f) includes surveillance requirements (TS Table 3.6.2d and NOTES FOR TABLES 3.6.2d AND 4.6.2d).

14. Increase the Instrument Channel Test interval requirement in TS Table 4.6.2d from monthly to quarterly for the following Parameters:

    a. START CORE SPRAY PUMPS

        1) High Drywell Pressure - Parameter (1)
        2) Low-Low Reactor Water Level - Parameter (2)

    b. OPEN CORE SPRAY DISCHARGE VALVES

        1) Reactor Pressure and either (1) or (2) above - Parameter (3)

GENE-770-06-1 dated February 1991

15. Increase the Instrument Channel Test interval requirement in TS Table 4.6.2e from monthly to quarterly for the following Parameters:

    INITIATES CONTAINMENT SPRAY

    1) High Drywell Pressure - Parameter (1)a.
    2) Low-Low Reactor Water Level - Parameter (1)b.

16. Revise note (c), which includes allowable out-of-service times for specified CONTAINMENT SPRAY INITIATION Parameters for required surveillance or repair (NOTES FOR TABLES 3.6.2e AND 4.6.2e).

NEDC-30936P-A (Parts 1 and 2) dated December 1988/RE-003 dated January 1987

17. Increase the Instrument Channel Test interval requirement in TS Table 4.6.2f from monthly to quarterly for the following Parameters:

    INITIATES AUTO DEPRESSURIZATION

    1) Low-Low-Low Reactor Water - Parameter (1)a.
    2) High Drywell Pressure - Parameter (1)b.

18. Revise note (d), which includes allowable out-of-service times for specified AUTO DEPRESSIONS INITIATION Parameters for required surveillance and repair (NOTES FOR TABLES 3.6.2f AND 4.6.2f).

NEDC-30851P-A Suppl 1 dated October 1988/GENE-770-06-1 dated February 1991

19. Add note (i), which includes allowable out-of-service times for specified CONTROL ROD WITHDRAWAL BLOCK Parameters for required surveillance (Table 3.6.2g and NOTES FOR TABLES 3.6.2g AND 4.6.2g).

20. Increase the Instrument Channel Test interval requirement in TS Table 4.6.2g from monthly to quarterly for the following Parameters:

    a. APRM CONTROL ROD WITHDRAWAL BLOCK

        1) Inoperative - Parameter (3)a.
        2) Upscale (Biased by Recirculation Flow) - Parameter (3)b.
        3) Downscale - Parameter (3)c.

    b. RECIRCULATION FLOW CONTROL ROD WITHDRAWAL BLOCK

        1) Comparator Off Normal - Parameter (4)a.
        2) Flow Unit Inoperative - Parameter (4)b.
        3) Flow Unit Upscale - Parameter (4)c.

NEDC-31677P-A dated July 1990

21. Increase the Instrument Channel Test interval requirement in TS Table 4.6.2h from weekly to quarterly for MECHANICAL VACUUM PUMP Isolation Parameter, "High Radiation Main Steam Line."

22. Revise/clarify[2] note (b), which includes allowable out-of-service times for the specified VACUUM PUMP ISOLATION Parameter for required surveillance and repair (NOTES FOR TABLES 3.6.2h AND 4.6.2h).

NEDC-30851P-A Suppl 2 dated March 1989

23. Revise/clarify[2] note (b), which includes allowable out-of-service times for the specified EMERGENCY VENTILATION INITIATION Parameters for required surveillance and repair (Table 3.6.2j and NOTES FOR TABLES 3.6.2j AND 4.6.2j).

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[2] 0G90-579-32A, letter to M. L. Wohl (NRC) from W. P. Sullivan and J. F. Klapproth (GE), "Implementation Enhancements to Technical Specification Changes Given in Isolation Actuation Instrumentation Analysis," dated June 25, 1990.

NEDC-30936P-A (Parts 1 and 2) dated December 1988/RE-003 dated January 1987

24. Revise note (c), which includes allowable out-of-service times for the specified HIGH PRESSURE COOLANT INJECTION Parameter for required surveillance and repair (Table 3.6.2k and NOTES FOR TABLES 3.6.2k AND 4.6.2k).

25. Increase the Instrument Channel Test interval requirement in TS Table 4.6.2k HIGH PRESSURE COOLANT INJECTION from monthly to quarterly for Low Reactor Water Level - Parameter (1).

All License Topical Reports

26. Modify BASES 3.6.2 AND 4.6.2 PROTECTIVE INSTRUMENTATION to reference the GE Topical and Plant-Specific Reports which justify the above proposed changes and delete outdated FSAR supplement information.

GENE-770-06-1 dated February 1991

27. Add note *, which includes allowable out-of-service times for specified ACCIDENT MONITORING Parameters for required surveillance (Table 3.6.11-1).

28. Increase Instrument Channel Test interval requirement in TS Table 4.6.11 ACCIDENT MONITORING INSTRUMENTATION from monthly to quarterly for the following Parameters:

    a. Relief Valve Position Indication (Primary and Backup) - Parameter (1)

    b. Safety Valve Position Indication (Primary and Backup) - Parameter (2)

    c. Reactor Vessel Water Level - Parameter (3)

    d. Suppression Chamber Water Level - Parameter (5)

All License Topical Reports

29. Modify BASES 3.6.11 AND 4.6.11 ACCIDENT MONITORING INSTRUMENTATION to reference the GE Topical Reports which justify the above proposed changes.

III. JUSTIFICATION FOR THE PROPOSED CHANGES

Niagara Mohawk has extended the generic analyses completed by the BWR Owners' Group to NMP1 by completing the required plant-specific analyses. The following discussion provides the information requested by the NRC staff in plant-specific submittals while Section V contains the No Significant Hazards Consideration Evaluation completed pursuant to 10 CFR 50.92. As stated within the NRC SERs for the Licensing Topical Reports (LTRs), three issues for NEDC-30851P-A and two issues for the other LTRs must be addressed to justify the applicability of the generic analyses to individual plants when specific facility Technical Specifications are considered for revision. The following is a discussion of those issues:

1. Confirm the applicability of the generic analyses to the specific facility. (This issue applies to all LTRs)

RESPONSE

a. Licensing Topical Report NEDC-30851P-A, Appendix L identifies NMP1, a BWR2, as a participating utility in the development of the RPS (SCRAM) Technical Specification Improvement Analysis. Section 7.4, "Conclusions of Plant Specific Applications," specifies that:

> "The evaluation found various differences between the RPS configuration of various plants and the generic plant. These differences include HFA relays, four scram contactors for BWR/2, sensor differences, scram parameter differences, and SDV sensor diversity differences. The assessment of these differences shows that while the HFA relays and the four scram contactors for BWR/2 would result in a higher overall RPS failure frequency, the improved technical specification intervals and allowable out-of-service times based on the generic plant would result in a net improvement to plant safety for plants with such differences. The effect of other differences on the RPS failure frequency is insignificant. Therefore, the generic results can be applied to plants in the BWROG Technical Specification Improvement Program"

Furthermore, included in this submittal is GE Plant-Specific Report MDE-77-0485, titled "Technical Specification Improvement Analysis for Nine Mile Point Nuclear Station, Unit 1" which concludes in Section 4 that "the generic analysis in Reference 1 (NEDC-30851P-A) is applicable to NMP1." NMPC has reviewed the LTRs and plant-specific report and verified applicability to NMP1. For a discussion of the differences between NMP1 and the generic plant analyzed in NEDC-30851P-A, see the response to Issue 3. below and GE Plant-Specific Report MDE-77-0485.

b. Licensing Topical Report NEDC-30851P-A Suppl 2, Appendix A identifies Niagara Mohawk as a participating utility in the development of the BWR Isolation Instrumentation Common to RPS (Reactor Protection System) and ECCS (Emergency Core Cooling System) Technical Specification Improvement Analysis. Section 3.4 specifically analyzes BWR 2 plants. NMPC has reviewed this LTR and verified applicability to NMP1.

c. Licensing Topical Report NEDC-31677P-A, Appendix E identifies Niagara Mohawk as a participating utility in the development of the BWR Isolation Actuation Instrumentation not common to the RPS or ECCS Technical Specification Improvement Analysis. Section 5.4 and Appendix C3 specifically analyzes BWR 2 plants. NMPC has reviewed this LTR and verified applicability to NMP1.

d. Licensing Topical Report NEDC-30936P-A, Appendix N (Part 1) and Appendix B (Part 2) identifies Niagara Mohawk as a participating utility in the development of the BWR ECCS Actuation Instrumentation Technical Specification Improvement Analysis. Section 5.3 (Part 2) specifically analyzes BWR 2 plants and this submittal contains the NMP1 Plant-Specific General Electric Company Report RE-003, titled "Technical Specification Improvement Analysis for the Emergency Core Cooling System Actuation Instrumentation for Nine Mile Point Nuclear Station,

Unit 1." NMPC has reviewed the LTRs and plant-specific report and verified applicability to NMP1.

e. Licensing Topical Report NEDC-30851P-A Suppl 1, Appendix B identifies Niagara Mohawk as a participating utility in the development of the BWR Control Rod Block Instrumentation Technical Specification Improvement Analysis. Section 4.0 specifically addresses BWR 2 plants. NMPC has reviewed the LTR and verified applicability to NMP1.

f. Licensing Topical Report GENE-770-06-1, identifies application of changes to surveillance test intervals and allowed out-of-service times for Selected Instrumentation Technical Specifications to all BWR plants. NMPC has reviewed the LTR and verified applicability to NMP1.

2. Demonstrate that the drift characteristics for RPS (SCRAM), ECCS, Isolation, Rod Block, and Selected Channel Instrumentation are bounded by the assumptions used in LTRs when the functional test interval is extended from monthly to quarterly. (This issue applies to all LTRs)

RESPONSE

This requirement as stated was difficult to address because the LTRs do not contain quantitative instrument drift assumptions. In order to resolve this concern, the BWR Owners' Group and the NRC staff reviewed the BWR setpoint calculation methodology and decided that additional clarification was in order. As a result, the NRC staff provided additional guidance in a letter from C. E. Rossi (NRC) to R. F. Janecek (BWROG) dated April 27, 1988 which specifically indicates that:

> "...licensees need only confirm that the setpoint drift which could be expected under the extended STIs has been studied and either (1) has been shown to remain within the existing allowance in the RPS and ESFAS instrument setpoint calculation or (2) that the allowance and setpoint have been adjusted to account for the additional expected drift."

Instrument setpoint drift is monitored during channel calibration tests when setpoints are required to be verified, not during the performance of the channel functional tests. A concern exists for plants that have calibration intervals shorter than the proposed quarterly functional tests. For those cases, an extension of the functional test interval would require a change to the channel calibration interval. The change in calibration interval would then require consideration of the effects on setpoint drift. The calibration intervals for the NMP1 instrumentation addressed by these LTRs have been verified to be equal to or longer than once per quarter and are therefore unaffected by the proposed changes.

3. Confirm that the differences between the parts of the RPS that perform the trip functions in the plant and those of the base case plant were included in the specific analysis done using the procedures of Appendix K to NEDC-30851P-A. (This issue applies to LTR NEDC-30851P-A only)

RESPONSE

Included in this submittal is the General Electric Company Report MDE-77-0485, titled "Technical Specifications Improvement Analysis for Nine Mile Point Nuclear Station, Unit 1." In summary, the report extends the generic study completed in Licensing Topical Report NEDC-30851P-A to NMP1 to increase surveillance test intervals (STIs) and add allowable out-of-

service times (AOTs) to the SCRAM (RPS) Instrumentation requirements in the Technical Specifications. This report utilizes the procedures of Licensing Topical Report NEDC-30851P-A, Appendix K to identify and evaluate the differences between the parts of the SCRAM (RPS) that perform the trip functions at NMP1 and those analyzed in the generic study. The results of the analysis indicate that the differences and their impact do not significantly affect the improvement in the Technical Specifications developed by the generic efforts of Licensing Topical Report NEDC-30851P-A. Therefore, the conclusions reached in NEDC-30851P-A apply to NMP1 and the plant-specific changes contained in this request are bounded by both the generic analysis and the NRCs' SER.

IV. CONCLUSION

As discussed in Item III above, Niagara Mohawk Power Corporation has satisfactorily addressed the three issues for NEDC-30851P-A and the two issues for the other GE Licensing Topical Reports (LTRs) which the NRC staff has indicated are necessary to implement the generic Technical Specification changes identified in the LTRs on a plant-specific basis. The first issue, which applies to all LTRs, required confirmation of the applicability of the generic analyses to NMP1. Two required plant-specific reports on SCRAM (RPS) and ECCS Instrumentation for NMP1 concluded that the generic analyses are applicable to NMP1. The information provided in the proprietary reports contained in this submittal address the differences between NMP1 and the generic analyses and, when applied with the conclusions contained in NEDC-30851P-A and NEDC-30936P-A, justifies the proposed changes. NMPC has reviewed the LTRs and plant-specific reports and verified applicability to NMP1. The second issue required demonstrating that instrument setpoints can remain unchanged because drift assumptions remain the same. A concern exists for plants that have calibration intervals shorter than the proposed quarterly functional tests. For those cases, an extension of the functional test interval would then require consideration of the effects on setpoint drift. The calibration intervals for the NMP1 instrumentation addressed by those LTRs have been verified to be equal to or longer than once per quarter and are therefore unaffected by the proposed changes. The third issue, which applies only to LTR NEDC-30851P-A for RPS (SCRAM), required confirmation that the base plant and NMP1 RPS (SCRAM) trip function differences allowed NMP1 to be bounded by the LTR. General Electric Company Report MDE-77-0485, titled "Technical Specification Improvement Analysis for Nine Mile Point Nuclear Station, Unit 1" confirmed that the NMP1 differences and their impact do not significantly affect the improvement in the Technical Specifications developed by the generic efforts of LTR NEDC-30851P-A.

Recent approved BWR Owners' Group clarifications used in the development of NUREG 1433, "Improved Standard Technical Specifications," were implemented in the proposed allowable out-of-service times (AOTs) which enhance AOT wording to ensure a "loss of function" does not occur. Also, two editorial changes are made:

Add "Upscale" to Parameter (9)(a)(i) in Table 3.6.2a;
Add engineering units (°F) to Parameters (8) and (9) in Table 3.6.2b;

Finally, an outdated note, which applied to a 1986 one-time-only license condition, was deleted.

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, about the issue of no significant hazards consideration. Therefore, in accordance with 10 CFR 50.91, the following evaluation has been performed:

V. NO SIGNIFICANT HAZARDS CONSIDERATION EVALUATION

1. <u>The operation of Nine Mile Point Unit 1, in accordance with the proposed amendment, will not involve a significant increase in the probability or consequences of an accident previously evaluated.</u>

The generic analysis contained in Licensing Topical Report NEDC-30851P-A assessed the impact of changing SCRAM (RPS) surveillance test intervals (STIs) and adding allowable out-of-service times (AOTs) on the SCRAM (RPS) failure frequency, the scram frequency and equipment cycling. Specifically, Section 5.7.4, "Significant Hazards Assessment," of NEDC-30851P-A states that:

> "Fewer challenges to the safeguards system, due to less frequent testing of the RPS, conservatively results in a decrease of approximately one percent in core damage frequency. This decrease is based upon the following:
>
> - Based on the plant-specific experience presented in Appendix J, the estimated reduction in scram frequency (0.3 scrams/ yr) represents a 1 to 2 percent decrease in core damage frequency based on the BWR plant-specific Probabilistic Risk Assessments (PRAs) listed in Table 5-8.
>
> - The increase in core damage frequency due to less frequent testing is less than one percent. This increase is even lower (less than 0.01 percent) when the changes resulting from the implementation of the Anticipated Transients Without Scram (ATWS) rule are considered. Therefore, this increase is more than offset by the decrease in CDF due to fewer scrams.
>
> - The effect of reducing unnecessary cycles on RPS equipment, although not easily quantifiable also results in a decrease in core damage frequency.
>
> - The overall impact on core damage frequency of the changes in allowable out-of-service times is negligible."

From this generic analysis, the BWR Owners' Group concluded that the proposed changes do not significantly increase the probability or consequences of an accident previously evaluated, namely the increase in probability of a scram failure due to SCRAM (RPS) unavailability is insignificant, and the overall probability of an accident is actually decreased as the time the SCRAM (RPS) Instrumentation logic operates as designed is increased resulting in less inadvertent scrams during testing and repair. Furthermore, the proprietary plant-specific analysis contained in this submittal demonstrates that although NMP1 differs from the generic model analyzed in Licensing Topical Report NEDC-30851P-A, the net effect of the plant-specific differences do not alter the generic conclusions. Also, the BWROG 92102 letter clarifications used in the development of NUREG 1433, "Improved Standard Technical Specifications," for AOTs which enhanced wording to ensure "loss of function" does not occur, results in no significant increase in the probability or consequences of an accident previously evaluated."

The generic analysis contained in Licensing Topical Reports NEDC-30851P-A Suppl 2/NEDC-31677P-A assessed the impact of changing STIs and AOTs for BWR Isolation Instrumentation common/not common to SCRAM (RPS) and ECCS instrumentation. Specifically, Section 4.0, "Summary of Results," of NEDC-30851P-A Suppl 2 states that:

> "The results indicate that the effects on probability of failure to initiate isolation are very small and the effects on probability or frequency of failure to isolate are

negligible in nearly every case. In addition, the results indicate that increasing the AOT to 24 hours for tests and repairs has a negligible effect on the probability of failure of the isolation function. These combined with changes to the testing intervals and allowed out-of-service times for RPS and ECCS instrumentation provide a net improvement to plant safety and operations."

and Section 5.6, "Assessment of Net Effect of Changes," of NEDC-31677P-A states that:

> "A reduction in core damage frequency (CDF) of at least as much as estimated in the ECCS instrumentation analysis can be expected when the isolation actuation instrumentation STIs are changed from one month to three months. The chief contributor to this reduction is the channel functional tests for the MSIVs. Inadvertent closure of the MSIVs will cause an unnecessary plant scram. This reduction in CDF more than compensates for any small incremental increase (10% or 1.0E-07/year) in calculated isolation function failure frequency when the STI is extended to three months."

From this generic analysis, the BWR Owners' Group concluded that the proposed changes do not significantly increase the consequences of an accident previously evaluated, namely the increase in probability of an isolation failure due to isolation instrumentation unavailability is insignificant, and the overall probability of an accident is actually decreased as the time the SCRAM (RPS) Instrumentation logic operates as designed is increased resulting in less inadvertent scrams during testing and repair. Also, the BWROG 0690-579-32A letter clarifications used in the development of NUREG 1433, "Improved Standard Technical Specifications," for AOTs which enhanced wording to ensure "loss of function" does not occur, results in no significant increase in the probability or consequences of an accident previously evaluated.

The generic analysis contained in Licensing Topical Report NEDC-30936P-A (Parts 1 and 2) assessed the impact of changing STIs and AOTs for all BWR ECCS Actuation Instrumentation. Specifically, Section 4.0, "Technical Assessment of Changes," of NEDC-30936P-A (Part 2) states that:

> "The results indicate an insignificant (less than 5E-7 per year) increase in water injection function failure frequency when STIs are increased from 31 days to 92 days, AOTs for repair of the ECCS actuation instrumentation are increased from one hour to 24 hours, and AOTs for surveillance testing are increased from two to six hours. For all four BWR models the increase represents less than 4% increase in failure frequency. However, when other factors which influence the overall plant safety are considered, the net result is judged to be an improvement in plant safety."

From this generic analysis, the BWR Owners' Group concluded that the proposed changes do not significantly increase the probability or consequences of an accident previously evaluated, namely the increase in probability of a water injection failure due to ECCS instrumentation unavailability is insignificant and the net result is judged to be an improvement in plant safety. Furthermore, the proprietary plant-specific analysis contained in this submittal demonstrates that although NMP1 differs from the generic model analyzed in Licensing Topical Report NEDC-30936P-A, the net affect of the plant-specific differences do not alter the generic conclusions.

The generic analysis contained in Licensing Topical Report NEDC-30851P-A Suppl 1, assessed the impact of changing Rod Block STIs on Rod Block failure frequency. Specifically, Section 5 (BNL's Tech. Eval. Report - Attach. 2 to the NRC SER) of NEDC-30851P-A Suppl 1 states that:

> "The BWR Owners' Group proposed changes to the Technical Specifications concerning the test requirements for BWR control rod block instrumentation. The changes consist of increasing the surveillance test intervals from one to three months. These test interval extensions are consistent with the already approved changes to STIs for the reactor protection system. The technical analysis reviewed and verified as documented herein indicates that there will be no significant changes in the availability of the control rod block function if these changes are implemented. In addition, there will be a negligible impact on the plant core melt frequency due to the decreased testing."

From this generic analysis, the BWR Owners' Group concluded that the proposed changes do not significantly increase the probability of an accident previously evaluated or consequences of an accident previously evaluated.

Bases contained in GE Topical Report GENE-770-06-1 assessed the impact of changing STIs and AOTs on selected systems failure frequency. Specifically, Section 2.0, "Summary," of GENE-770-06-1 states that:

> "Technical bases are provided for selected proposed changes to the instrumentation STIs and AOTs that were identified in the BWROG Improved BWR Technical Specification activity. These STI and AOT changes are consistent with approved changes to the RPS, ECCS, and isolation actuation instrumentation. These proposed changes do not result in a degradation to overall plant safety."

From these Bases, the BWR Owners' Group concluded that the proposed changes do not significantly increase the probability of an accident previously evaluated or consequences of an accident previously evaluated.

Also, NMPC concluded the editorial items do not involve a significant increase in the probability or consequences of an accident previously evaluated. These changes do not alter the meaning or intent of any requirements.

2. <u>The operation of Nine Mile Point Unit 1, in accordance with the proposed amendment, will not create the possibility of a new or different kind of accident from any accident previously evaluated.</u>

The proposed change will not alter the physical characteristics of any plant systems or components and all safety-related systems and components remain within their applicable design limits. Thus, system and component performance is not adversely affected by this change, 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.

The addition of allowable out-of-service times (AOTs) and the increase in surveillance test intervals (STIs) does not alter the function of the SCRAM (RPS), ECCS, Isolation, Rod Block, and Selected Instrument Systems nor involve any type of plant modification and no new modes of plant operation are involved with these changes. Also, the editorial changes contained herein do not alter plant configurations or operating modes. Editorial changes do not, by their nature, create the possibility of a new or different kind of accident. Finally, the approved BWR Owners' Group clarifications are enhancements to allowable out-of-service times (AOTs) and do not create the possibility of a new or different kind of accident. Therefore, operation in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. <u>The operation of Nine Mile Point Unit 1, in accordance with the proposed amendment, will not involve a significant reduction in a margin of safety.</u>

The NRC staff has reviewed and approved the generic studies contained in the GE Topical Reports (LTRs) and has concurred with the BWR Owners' Group that the proposed changes do not significantly affect the availability of the SCRAM (RPS), ECCS, Isolation, Rod Block, or Selected Instrument Systems. The proposed addition of allowable out-of-service times (AOTs) for the instruments addressed in the LTRs provides reasonable time for making repairs and performing tests. The lack of sufficient AOTs in the current Technical Specifications (TS) creates a hurried atmosphere during repairs and tests which could cause an increased risk of error. Also, placing an individual channel in a tripped condition because no AOT exists, as in the current TS, increases the potential of an inadvertent scram. The proposed AOTs provide realistic times to complete the required actions without increasing the overall instrument failure frequency. Therefore, there is no significant reduction in the margin of safety.

The incorporation of extended surveillance test intervals (STIs) does not result in significant changes in the probability of instrument failure, as demonstrated by the LTRs. Also, the calibration frequency has not changed, therefore assurance exists that the setpoints will not be affected by drift. These changes, when coupled with the reduced probability of test-induced plant transients and equipment failures, result in an overall increase in the margin of safety.

Also, the BWR Owners' Group clarification items do not alter the meaning or intent of any requirements, therefore does not affect the margin of safety.

General Electric Company Report
MDE-77-0485 dated April 1985 and RE-003 dated January 1987
Technical Specification Improvement Analyses for the
Reactor Protection System and ECCS Actuation Instrumentation
Nine Mile Point Unit 1 Station

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