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see Draft Tech Specs

SUBJECT: Forwards changes to final draft Tech Specs, transmitted by NRC 860627 ltr, categorized as necessary for certification, needed for operational flexibility or clarification. Expeditious resolution of items requested.

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NIAGARA MOHAWK

NIAGARA MOHAWK POWER CORPORATION/300 ERIE BOULEVARD WEST, SYRACUSE, N.Y. 13202/TELEPHONE (315) 474-1511

August 6, 1986
(NMP2L 0807)

Ms. Elinor G. Adensam, Director
BWR Project Directorate No. 3
U.S. Nuclear Regulatory Commission
7920 Norfolk Avenue
Washington, DC 20555

Dear Ms. Adensam:

Re: Nine Mile Point Unit 2
Docket No. 50-410

We have substantially completed our review of the Final Draft Technical Specifications for Nine Mile Point Unit 2 which you provided by letter dated June 27, 1986. As a result, we have identified a number of necessary changes which are categorized as necessary for certification, editorial, needed for operational flexibility, or clarifying in nature.

The specific changes to the Technical Specifications and their justification are provided in the enclosure to this letter. Where the requested changes and justifications have already been provided to you, reference to the transmittal letter is made. Where the requested changes to the Technical Specifications also affect the Final Safety Analysis Report or the Safety Evaluation, the changes to the appropriate pages are enclosed. A list of the changes to the Technical Specifications, the Final Safety Analysis Report, and the Safety Evaluation Report are included to aid your staff in the review of these changes.

Since certification of the Technical Specifications now appears to be a critical step in the licensing of Nine Mile Point Unit 2, we would appreciate your expeditious resolution of these items. We are continuing our review of the recently received Supplement 3 of the Safety Evaluation Report and will inform you of our comments when it is complete.

Very truly yours,

CVMangan

C. V. Mangan
Senior Vice President

LSL:ja
1889G

Enclosures

xc: W. A. Cook, NRC Resident Inspector
Project File (2)

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1/1

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
Niagara Mohawk Power Corporation	)	Docket No. 50-410
(Nine Mile Point Unit 2)	)	

AFFIDAVIT

C. V. Mangan, being duly sworn, states that he is Senior Vice President of Niagara Mohawk Power Corporation; that he is authorized on the part of said Corporation to sign and file with the Nuclear Regulatory Commission the documents attached hereto; and that all such documents are true and correct to the best of his knowledge, information and belief.

CVMangan

Subscribed and sworn to before me, a Notary Public in and for the State of New York and County of Onondaga, this 6th day of August, 1986.

Janis M Macro
Notary Public in and for
Onondaga County, New York

My Commission expires:

JANIS M. MACRO
Notary Public in the State of New York
Qualified in Onondaga County No. 4784555
My Commission Expires March 30, 1987

LIST OF TECHNICAL SPECIFICATIONS,
FINAL SAFETY ANALYSIS REPORT, AND
SAFETY EVALUATION REPORT PAGES CHANGED

Description	Document[1]	Page	Category
Primary Containment Isolation Valves	T.S.	B3/4 6-5	Clarification
Radioactive Liquid Effluent Monitoring Instrumentation	T.S.	3/4 3-99	Certification
	T.S.	3/4 3-101	
	FSAR	11.5-13	
	SER	11-10	
Safety Relief Valves	T.S.	3/4 4-10	Operational
	T.S.	3/4 5-3	
	T.S.	B3/4 4-3	
	T.S.	B3/4 5-2	
	SER	6-3	
Accident Monitoring Instrumentation	T.S.	3/4 3-85	Clarification
	T.S.	3/4 3-86	
Administrative Controls[2]	T.S.	6-11	Certification
Source Check[2]	T.S.	1-8	Certification
Secondary Containment Integrity	T.S.	1-7	Certification
Main Steam Isolation Valve Leak Rate	T.S.	3/4 6-6	Operational
Fire Protection Program[2]	T.S.	3/4 7-25	Operational
	T.S.	3/4 7-30	
Primary Containment Isolation Valves	T.S.	3/4 6-28	Certification

Description	Document (1)	Page	Category
Rod Worth Minimizer	T.S.	3/4 1-16	Certification
		3/4 10-2	
Other Items	T.S.	2-1	Certification
	T.S.	2-7	
	T.S.	3/4 3-66	
	T.S.	3/4 3-82	
	T.S.	3/4 3-83	
	T.S.	3/4 3-85	
	T.S.	3/4 4-16	
	T.S.	3/4 5-5	
	T.S.	3/4 6-24	
	T.S.	3/4 7-3	
	T.S.	3/4 7-6	
	T.S.	3/4 10-7	
Other Items	T.S.	3/4 3-3	Editorial
	T.S.	3/4 3-45	
	T.S.	3/4 3-79	
	T.S.	3/4 6-28	
	T.S.	3/4 6-42	
	T.S.	3/4 7-26	
	T.S.	3/4 8-2	

Description	Document (1)	Page	Category
Other Items (cont.)	T.S.	3/4 8-3	Editorial
	T.S.	3/4 8-21	
	T.S.	3/4 8-22	
	T.S.	3/4 8-23	
	T.S.	3/4 8-28	
	T.S.	B3/4 4-7	

Notes:

(1) T.S. = Technical Specifications
FSAR = Final Safety Analysis Report
SER = Safety Evaluation Report

(2) This item was previously submitted to the Nuclear Regulatory Commission.

Changes to Technical Specification

Bases 3/4.6.3

"Primary Containment Isolation Valves"

Subject: Justification for change to Technical Specification Bases 3/4.6.3, "Primary Containment Isolation Valves"

The requested change is enclosed. This change will clarify the relationship between isolation system instrumentation response time and isolation valve closing time.

CHANGE REQUESTED FOR CLARIFICATION

FINAL DRAFT

Valve closing times do not include isolation instrumentation response times,

PRIMARY CONTAINMENT

PRIMARY CONTAINMENT ISOLATION VALVES

3/4.6.3 (Continued)

GDC 54 through 57 of Appendix A to 10 CFR 50: Measurement of the closure time of automatic containment isolation valves is performed for the purpose of demonstrating PRIMARY CONTAINMENT INTEGRITY and system OPERABILITY (Specification 3/4.6.1).

The maximum isolation times for primary containment automatic isolation valves listed in this specification are either the analytical times used in the accident analysis as described in the FSAR; or times derived by applying margins to the vendor test data obtained in accordance with industry codes and standards. For non-analytical automatic primary containment isolation valves, the maximum isolation time is derived as follows:

1) Valves with full stroke times less than or equal to 10 seconds, maximum isolation time approximately equals the vendor tested closure time multiplied by 2.0.

2) Valves with full stroke time greater than 10 seconds, maximum isolation time approximately equals the vendor tested closure time multiplied by 1.5.

3/4.6.4 SUPPRESSION CHAMBER - DRYWELL VACUUM BREAKERS

Vacuum relief breakers are provided to equalize the pressure between the suppression chamber and drywell. This system will maintain the structural integrity of the primary containment under conditions of large differential pressures.

The vacuum breakers between the suppression chamber and the drywell must not be inoperable in the open position since this would allow bypassing of the suppression pool in case of an accident. There are four pairs of valves to provide redundancy so that operation may continue for up to 72 hours with no more than one pair of vacuum breakers inoperable in the closed position.

3/4.6.5 SECONDARY CONTAINMENT

Secondary containment is designed to minimize any ground level release of radioactive material which may result from an accident. The reactor building and associated structures provide secondary containment during normal operation when the drywell is sealed and in service. At other times, the drywell may be open and, when required, secondary containment integrity is specified.

Establishing and maintaining a subatmospheric condition in the reactor building with the standby gas treatment system once per 18 months, along with the surveillance of the doors, hatches, dampers, and valves, is adequate to ensure that there are no violations of the integrity of the secondary containment.

JUN 25 1986

Changes to Technical Specifications

Tables 3.3.7.10-1 and 4.3.7.10-1

in the Area of Radioactive Liquid Effluent
Monitoring Instrumentation

Subject: Justification for changes to Technical Specification Tables 3.3.7.10-1 and 4.3.7.10-1 in the area of radioactive liquid effluent monitoring instrumentation

The current Technical Specification Section 3.3.7.10 requires the Liquid Radwaste Monitor to be OPERABLE at all times, whether radwaste discharge is occurring or not. System design provides three valves to prevent inadvertent discharge. These valves must be specifically lined up in the course of making a discharge. Inherent in this design is the isolation of the small section of discharge line from and to which the Liquid Radwaste monitor's sample pump takes supply and return. When in continuous use, the sample pump produces more heat than can be dissipated in the small volume of water contained in this section of pipe. Therefore, it is requested to revise Technical Specification Tables 3.3.7.10-1 and 4.3.7.10-1 to provide:

1. The Liquid Waste Monitor must be OPERABLE at all times during discharge of liquid waste.

2. The CHANNEL CHECK and SOURCE CHECK are to be performed P (prior to discharge).

The requested changes to Technical Specification Tables 3.3.7.10-1 and 4.3.7.10-1 are enclosed. These changes also affect the Final Safety Analysis Report and the Safety Evaluation Report. Changes to the appropriate pages of these reports are also enclosed.

CHANGE REQUESTED FOR CERTIFICATION

FINAL DRAFT

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITIONS FOR OPERATION

3.3.7.10 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3.7.10-1 shall be OPERABLE with their Alarm/Trip Setpoints set to ensure that the limits of Specification 3.11.1.1 are not exceeded. The Alarm/Trip Setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

APPLICABILITY: At all times

ACTION:

a. With a radioactive liquid effluent monitoring instrumentation channel Alarm/Trip Setpoint less conservative than required by the above specification, immediately suspend the release of radioactive liquid effluents monitored by the affected channel, or declare the channel inoperable, or change the setpoint so it is acceptably conservative.

b. With the number of channels OPERABLE less than the Minimum Channels OPERABLE requirement, take the ACTION shown in Table 3.3.7.10-1. Restore the instruments to OPERABLE status within 30 days and, if unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.10 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST at the frequencies shown in Table 4.3.7.10-1.

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FINAL DRAFT

TABLE 3.3.7.10-1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. Radioactivity Monitors Providing Alarm and Automatic Termination of Release		
Liquid Radwaste Effluent Line	1**	128
2. Radioactivity Monitors Providing Alarm But Not Providing Automatic Termination of Release		
a. Service Water Effluent Line A	1	130
b. Service Water Effluent Line B	1	130
c. Cooling Tower Blowdown Line	1	130
3. Flow Rate Measurement Devices		
a. Liquid Radwaste Effluent Line	1	131
b. Service Water Effluent Line A	1	131
c. Service Water Effluent Line B	1	131
d. Cooling Tower Blowdown Line	1	131
4. Tank Level Indicating Devices*	1	132

* Tanks included in this specification are those outdoor tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system, such as temporary tanks.

**This instrumentation need be OPERABLE only during times of discharging.

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TABLE 3.3.7.10-1 (Continued)

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

TABLE NOTATIONS

ACTION 128 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases may continue provided that before initiating a release:

a. At least two independent samples are analyzed in accordance with Specification 4.11.1.1.1, and

b. At least two technically qualified members of the facility staff independently verify the release rate calculations and discharge line valving;

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 129 - Not used.

ACTION 130 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that, at least once per 12 hours, grab samples are collected and analyzed for radioactivity at a limit of detection of at least 5×10^{-7} microcuries/ml.

ACTION 131 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue, provided the flow rate is estimated at least once per 4 hours during actual releases. Pump performance curves generated in place may be used to estimate flow.

ACTION 132 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, liquid additions to this tank may continue provided the tank liquid level is estimated during all liquid additions to the tank.

JUN 25 1986

TABLE 4.3.7.10-1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. Radioactivity Monitors Providing Alarm and Automatic Termination of Release				
Liquid Radwaste Effluent Line	~~D~~ P	~~NA~~ P	R(c)	M(a)(b)
2. Radioactivity Monitors Providing Alarm But Not Providing Automatic Termination of Release				
a. Service Water Effluent Line A	D	M	R(c)	SA(b)
b. Service Water Effluent Line B	D	M	R(c)	SA(b)
c. Cooling Tower Blowdown Line	D	M	R(c)	SA(b)
3. Flow Rate Measurement Devices				
a. Liquid Radwaste Effluent Line	D(d)	NA	R	Q
b. Service Water Effluent Line A	D(d)	NA	R	Q
c. Service Water Effluent Line B	D(d)	NA	R	Q
d. Cooling Tower Blowdown Line	D(d)	NA	R	Q
4. Tank Level Indicating Devices*	D**	NA	R	Q

* Tanks included in this specification are those outdoor tanks that are not surrounded by liners, dikes, or wa[illegible] capable of holding the tank contents and do not have tank overflows and surrounding area drains connected t[illegible] the liquid radwaste treatment system, such as temporary tanks.

** During liquid additions to the tank.

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TABLE 4.3.7.10-1 (Continued)

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATIONS

(a) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if the instrument indicates measured levels above the Alarm/Trip Setpoint.

(b) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists:

(1) Instrument indicates measured levels above the Alarm Setpoint, or
(2) Circuit failure, or
(3) Instrument indicates a downscale failure, or
(4) Instrument controls not set in operate mode.

(c) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards, standards that are traceable to the National Bureau of Standards, or using actual samples of liquid effluents that have been analyzed on a system that has been calibrated with National Bureau of Standards traceable sources. These standards shall permit calibrating the system over its intended range of energy and measurement. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration may be used.

(d) CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once per 24 hours on days on which continuous, periodic, or batch releases are made.

radioactivity alarm in the main control room. Tritium in the plant areas is determined on the basis of representative grab samples collected from the effluent points or ventilation exhaust ducts. Grab samples are obtained from locations indicated in Table 11.5-2. Samples are analyzed in the health physics laboratory, or by contracted laboratories.

11.5.3 Effluent Monitoring and Sampling

All potentially radioactive gaseous and liquid effluent discharge paths are either continuously monitored or routinely sampled for radiation level during discharge (Section 11.5.2). Solid waste shipping containers are monitored with gamma sensitive portable survey instruments. The following gaseous effluent paths are sampled and monitored:

1. Plant main stack exhaust.

2. Combined radwaste/reactor building ventilation exhaust.

The following liquid effluent paths are sampled and monitored:

1. Liquid radwaste system effluent.

2. Circulating water system cooling tower blowdown line.

3. Service water system discharge.

All monitor ranges are listed in Table 11.5-1.

An isotopic analysis is performed periodically on samples obtained from each liquid effluent release path to verify the adequacy of effluent processing to meet the discharge limits to unrestricted areas.

This effluent monitoring and sampling program is comprehensive and provides the information for the effluent measuring and reporting programs required by 10CFR50 Section 36a, Appendix A, General Design Criterion 64, and Appendix I and Regulatory Guide 1.21 in semiannual reports to the NRC. The frequency of the periodic sampling and analysis described in the technical specifications is a minimum and is increased if effluent levels approach technical specification limits. Isotopic content of gaseous effluents is continuously monitored by offline monitors. All

equipment performance, and (4) monitor and control radioactivity levels in plant discharges to the environs.

Table 11.5 provides the proposed locations of continuous monitors. Monitors on certain effluent release lines will automatically terminate discharges in the event that radiation levels exceed a predetermined value. Systems that are not amenable to continuous monitoring, or for which detailed isotopic analyses are required, will be periodically sampled and the samples will be analyzed in the plant laboratory.

The potential airborne radioactivity releases to the environs from NMP-2 from two monitored points: (1) plant main stack, and (2) the combined radwaste and reactor building vent.

The plant main stack receives inputs from standby gas treatment, turbine building ventilation, turbine gland seal exhaust, condenser offgas, and mechanical vacuum pump exhaust. The combined reactor and radwaste building vent receives inputs from radwaste building ventilation and normal reactor building ventilation. Each release point is monitored by an online isotopic gaseous radiation monitoring system that is designed (1) to obtain continuous isokinetic and representative samples; (2) to continuously determine gaseous effluent isotopic content; (3) to store the information on releases of radioactive particulates, iodine, and noble gas; (4) to retrieve the data on command; and (5) to alarm in the main control room in the event that specified rates of release of radioactive material are exceeded.

Each online isotopic gaseous radiation monitor consists of three detectors (one each for iodine, particulate, and noble gas channels) with associated valving, electronics, multichannel analyzer, and computer. The monitor satisfies all requirements of NUREG-0737 with a range that satisfies RG 1.97. The monitor is capable of functioning both during and after an accident, and provides continuous monitoring of high-level postaccident releases of radioactive noble gases from the plant.

The potential radioactive liquid effluent release points to the environment are (1) processed liquid radwaste discharge, (2) circulating water system cooling tower blowdown line, (3) residual heat removal system service water effluent, and (4) service water system discharge to Lake Ontario. All release points are either continuously monitored or routinely sampled for radioactivity ~~before~~ during discharge. The liquid radwaste discharge is automatically terminated in the event that radioactivity concentration reaches a preset value. All other monitors will alarm in the main control room upon detection of radioactivity in the discharge. The staff's review included the locations and types of effluent and process monitoring provided for NMP-2. On the basis of the plant's design, continuous monitoring locations, and intermittent sampling locations, the staff concluded that all normal and potential release pathways are monitored. The staff also determined that the sampling and monitoring provisions are adequate for detecting radioactive material leakage to normally uncontaminated systems and for monitoring plant processes that could affect radioactivity releases. On these bases, the staff considers that the monitoring and sampling provisions satisfy the requirements of GDC 60, 63, and 64 and conform to the guidelines of RG 1.21.

Changes to Technical Specifications

in the Area of Safety/Relief Valves

Subject: Justification for changes to Technical Specifications in the area of safety relief valves

Requested changes to the Technical Specifications are enclosed. These changes support the power ascension test program. Justification for these changes is in Appendix 15C, "Two Safety/Relief Valves Out of Service," of the Final Safety Analysis Report.

Proposed change to Supplement 2 of the Safety Evaluation Report is also enclosed. This change is consistent with Appendix 15C.

CHANGE REQUESTED FOR OPERATIONAL FLEXIBILITY

FINAL DRAFT

3/4.4.2 SAFETY/RELIEF VALVES

LIMITING CONDITIONS FOR OPERATION

3.4.2 The safety valve function of at least ~~17~~ 16 of the following reactor coolant system safety/relief valves shall be OPERABLE with the specified code safety valve function lift settings*; the acoustic monitor for each OPERABLE valve shall be OPERABLE:

2 safety/relief valves @ 1148 psig ±1%
4 safety/relief valves @ 1175 psig ±1%
4 safety/relief valves @ 1185 psig ±1%
4 safety/relief valves @ 1195 psig ±1%
4 safety/relief valves @ 1205 psig ±1%

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With the safety valve function of one ~~or more~~ of the above required 16 safety/relief valves inoperable, restore a valve to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the ~~next~~ following 24 hours.

c. ~~b.~~ With one or more safety/relief valves stuck open, provided that the average water temperature in the suppression pool is less than 110°F, close the stuck-open safety/relief valve(s); if unable to close the open valve(s) within 5 minutes or if the average water temperature in the suppression pool is 110°F or more, place the reactor mode switch in the Shutdown position.

d. ~~c.~~ With one or more safety/relief valve acoustic monitors inoperable, restore the inoperable monitor(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With the safety valve function inoperable for more than one of the above required 16 safety/relief valves, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

* The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.

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FINAL DRAFT

3/4.4.2 SAFETY/RELIEF VALVES

SURVEILLANCE REQUIREMENTS

4.4.2.1 The acoustic monitor for each safety/relief valve shall be demonstrated OPERABLE with the setpoint verified to be 0.25 of the full-open noise level* by performance of a:

a. CHANNEL FUNCTIONAL TEST at least once per 31 days, and a
b. CHANNEL CALIBRATION at least once per 18 months.**

*Initial setting shall be in accordance with the manufacturers recommendation. Adjustment to the valve full-open noise level shall be accomplished during the startup test program.

**The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

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FINAL DRAFT

LIMITING CONDITIONS FOR OPERATION

3.5.1 (Continued)

ACTION:

d. For ECCS Divisions I and II, provided that ECCS Division III is OPERABLE:

1. With LPCI subsystem "A" and either LPCI subsystem "B" or "C" inoperable, restore at least the inoperable LPCI subsystem "A" or the inoperable LPCI subsystem "B" or "C" to OPERABLE status within 72 hours.

2. With the LPCS system inoperable and either LPCI subsystems "B" or "C" inoperable, restore at least the inoperable LPCS system or the inoperable LPCI subsystem "B" or "C" to OPERABLE status within 72 hours.

3. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours*.

e. For ECCS Divisions I and II, provided that ECCS Division III is OPERABLE and Divisions I and II are otherwise OPERABLE:

1. With ~~one~~ up to two of the above required ADS valves inoperable, restore the ~~inoperable ADS~~ valve(s) to OPERABLE status within 14 days of the first valve becoming inoperable or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to less than or equal to 100 psig within the next 24 hours.

2. With ~~two~~ three or more of the above required ADS valves inoperable, be in at least HOT SHUTDOWN within 12 hours and reduce reactor steam dome pressure to less than or equal to 100 psig within the next 24 hours.

f. In the event an ECCS is actuated and injects water into the reactor coolant system, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days, describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

* Whenever two or more RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

FINAL DRAFT

RECIRCULATION SYSTEM

3/4.4.1 (Continued)

recirculation pump and recirculation nozzles. Sudden equalization of a temperature difference $\geq$ 145°F between the reactor vessel bottom head coolant and the coolant in the upper region of the reactor vessel by increasing core flow rate would cause undue stress in the reactor vessel bottom head.

3/4.4.2 SAFETY/RELIEF VALVES

The safety/relief valves operate during a postulated ATWS event to prevent the reactor coolant system from being pressurized above a design allowable value of 1375 psig in accordance with the ASME Code. A total of ~~17~~ 16 OPERABLE safety/relief valves is required to limit local pressure at active components to within ASME III allowable design values (Service Level A). All other appropriate ASME III limits are also bounded by this requirement.

The safety-relief valve lift settings will be demonstrated only during shutdown in accordance with the provisions of Specification 4.0.5.

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.3.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. These detection systems are consistent with the recommendations of RG 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.3.2 OPERATIONAL LEAKAGE

The allowable leakage rates from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The background leakage normally expected to result from equipment design and the detection capability of the instrumentation for determining system leakage were also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for UNIDENTIFIED LEAKAGE, the probability is small that the imperfection or crack associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be PRESSURE BOUNDARY LEAKAGE, the reactor will be shut down to allow further investigation and corrective action.

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity, thereby reducing the probability of gross valve failure and consequent intersystem LOCA.

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EMERGENCY CORE COOLING SYSTEM

BASES

ECCS - OPERATING AND SHUTDOWN

3/4.5.1 & 3/4.5.2 (Continued)

The capacity of the system is selected to provide the required core cooling. The HPCS pump is designed to deliver greater than or equal to 516/1550/6350 gpm at differential pressures of 1160/1130/200 psi, respectively. Initially, water from the condensate storage tank is used instead of water injected from the suppression pool into the reactor, but no credit is taken in the safety analyses for the condensate storage tank water.

With the HPCS system inoperable, adequate core cooling is assured by the OPERABILITY of the redundant and diversified automatic depressurization system and both the LPCS and LPCI systems. In addition, the reactor core isolation cooling (RCIC) system, a system for which no credit is taken in the safety analysis, will automatically provide makeup water at reactor operating pressures on a reactor low water level condition. The HPCS out-of-service period of 14 days is based on the demonstrated OPERABILITY of redundant and diversified low-pressure core cooling systems.

The Surveillance Requirements provide adequate assurance that the HPCS system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test with reactor vessel injection requires the reactor to be shut down. The pump discharge piping is maintained full to prevent water hammer damage.

Upon failure of the HPCS system to function properly after a small-break loss-of-coolant accident, the automatic depressurization system (ADS) automatically causes selected safety/relief valves to open, depressurizing the reactor so that flow from the low-pressure core cooling systems can enter the core in time to limit fuel cladding temperature to less than 2200°F. ADS is conservatively required to be OPERABLE whenever reactor vessel pressure exceeds 100 psig. This pressure is substantially below that for which the low-pressure core cooling systems can provide adequate core cooling for events requiring ADS.

ADS automatically controls seven selected five safety/relief valves although the safety analysis only takes credit for ~~six~~ valves. It is, therefore, appropriate to permit ~~one~~ two valves to be out of service for up to 14 days without materially reducing system reliability.

rapidly propagating fracture will be minimized, and that the requirements of General Design Criterion (GDC) 51 are satisfied.

6.3 Emergency Core Cooling System

In Section 6.3 of the SER the staff discussed the loss-of-coolant accident (LOCA) analyses results for a lead plant that was representative of NMP-2. The applicant had committed to perform the plant-specific LOCA analyses and determine the minimum number of automatic depressurization system (ADS) safety relief valves (SRVs) needed to achieve a rapid depressurization during a small-break LOCA based on plant-specific LOCA analyses.

The applicant provided the plant-specific LOCA analyses in FSAR Amendment 20, dated July 1985. The plant-specific LOCA analyses included a spectrum of large and small postulated pipe breaks and the worst single failures based on the lead plant LOCA analyses. The results of the plant-specific LOCA analyses indicate that the most limiting break is a design-basis break in the recirculation suction piping with a low-pressure core spray (LPCS) diesel generator failure.

The plant-specific LOCA analysis shows that the calculated maximum total hydrogen generation from the chemical reactor of the cladding with water or steam for the most limiting LOCA case is 0.07%, which is well within the 10 CFR 50.46 limit of 1%.

Table 6.2 illustrates how the plant-specific LOCA analysis results meet the requirements of 10 CFR 50.46.

In a small-break emergency core cooling system (ECCS) analysis, a particular ADS flow rate is required to bring the vessel pressure down in a prescribed time to allow the operation of the low-pressure core cooling system following the postulated failure of the high-pressure core spray (HPCS). This ADS flow rate determines the number of ADS valves.

Seven SRVs were needed to perform the ADS function based on a generic BWR/5 calculation. The NMP-2 plant-specific ECCS analysis is in compliance with the requirements of 10 CFR 50, Appendix K, and has confirmed the adequacy of the seven ADS valve design by using only six ADS valves in the small-break ECCS analysis.

The technical specification will state that only one ADS valve is permitted to be out of service during operation.

→ (INSERT ATTACHED PARAGRAPH)

The staff concludes that the plant-specific LOCA analyses and the results for NMP-2 are acceptable. On the basis of the above, confirmatory issues 10, 15, and 16 in the SER are considered to be resolved.

6.4 Control Room Habitability Systems

Section 6.4 of the SER contains the sentence

> "To ensure maintenance of a low-leakage control room, the staff will require periodic surveillance through Technical Specifications to ensure that 1/8-in. water gauge control room pressurization relative

INSERT ------- to page 6-3, SSER-2

A subsequent analysis to support entended operation with two SRVs out of service has been docketed as Appendix 15C of the FSAR. This analysis has confirmed the plant's capability to meet 10 CFR 50, Appendix K criteria by using only five ADS valves in the small break ECCS analysis. The associated technical specification revision, hence, states that two ADS valves are permitted to be out of service during operation for up to 14 days.

Changes to Technical Specifications

Table 4.3.7.5-1

"Accident Monitoring Instrumentation
Surveillance Requirements"

Subject: Justification for changes to Technical Specification Table 4.3.7.5-1, "Accident Monitoring Instrumentation Surveillance Requirements"

The requested changes are enclosed. The changes clarify the intent of the channel check and channel calibration requirements of primary containment isolation valve position indication. Without the requested change, it can be misinterpreted that the valve must be stroked in order to determine operability of the valve position indication.

These changes were discussed with Carl Schulten, of your staff.

CHANGE REQUESTED FOR CLARIFICATION

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MONITORING INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITIONS FOR OPERATION

3.3.7.5 The accident monitoring instrumentation channels shown in Table 3.3.7.5-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3.7.5-1.

ACTION:

With one or more accident monitoring instrumentation channels inoperable, take the ACTION required by Table 3.3.7.5-1.

SURVEILLANCE REQUIREMENTS

4.3.7.5 Each of the above required accident monitoring instrumentation channels shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.7.5-1.

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TABLE 3.3.7.5-1

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. Reactor Vessel Pressure	2	1	1, 2	80
2. Reactor Vessel Water Level				
a. Fuel Zone	2	1	1, 2	80
b. Wide Range	2	1	1, 2	80
3. Suppression Pool Water Level				
a. Narrow Range	2	1	1, 2, 3	83
b. Wide Range	2	1	1, 2, 3	83
4. Suppression Pool Water Temperature	8, 2/Quadrant	4, 1/Quadrant	1, 2	80
5. Suppression Chamber Pressure	2	1	1, 2	80
6. Drywell Pressure	2	1	1, 2	80
7. Drywell Air Temperature	2	1	1, 2	80
8. Drywell Oxygen Concentration	2	1	1, 2	80
9. Drywell Hydrogen Concentration Analyzer and Monitor	2	1	1, 2	80
10. Safety/Relief Valve Position Indicators*	2/Valve	1/Valve	1, 2	80

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TABLE 3.3.7.5-1 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE	APPLICABLE OPERATIONAL CONDITIONS	ACTION
11. Drywell High Range Radiation Monitors	2	1	1, 2, 3	81
12. RHR Heat Exchanger Service Water Radiation Monitor	1/Heat Exchanger	1/Heat Exchanger	1, 2, 3	81
13. Refuel Platform Area Radiation Monitor	1	1	**	82
14. Neutron Flux†				
APRM	2	1	1, 2	80
IRM	2	1	1, 2	80
SRM	2	1	1	80
15. Primary Containment Isolation Valve Position Indication	1	1	1, 2	84

*Acoustic monitoring and tail pipe temperature
**When handling fuel, or components in the fuel pool or reactor cavity.
†Neutron flux indication is sufficient to meet the OPERABILITY requirement of this specification.

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Table 3.3.7.5-1 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

ACTION

ACTION 80 - a. With the number of OPERABLE accident monitoring instrumentation channels less than the Required Number of Channels shown in Table 3.3.7.5-1, restore the inoperable channel(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum Channels OPERABLE requirements of Table 3.3.7.5-1, restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

ACTION 81 - With the number of OPERABLE accident monitoring instrumentation channels less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable channel(s) to OPERABLE status within 72 hours, or:

a. Initiate the preplanned alternate method of monitoring the appropriate parameter(s), and

b. In lieu of another report required by Specification 6.9.2, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

ACTION 82 - With the number of OPERABLE accident monitoring instrumentaiton channels less than the Minimum Channels OPERABLE requirements of Table 3.3.7.5-1, suspend movement of fuel or components in the fuel pool or reactor cavity, or, initiate the preplanned alternate method of monitoring the appropriate parameter(s).

ACTION 83 - a. With the number of OPERABLE accident monitoring instrumentation channels less than the Required Number of Channels shown in Table 3.3.7.5-1, restore the inoperable channel(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum Channels OPERABLE requirement of Table 3.3.7.5-1, restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

Action 84 - Take the ACTION required by Specification 3.6.3.

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TABLE 4.3.7.5-1

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	APPLICABLE OPERATIONAL CONDITIONS
1. Reactor Vessel Pressure	M	R	1, 2
2. Reactor Vessel Water Level			
a. Fuel Zone	M	R	1, 2
b. Wide Range	M	R	1, 2
3. Suppression Pool Water Level			
a. Narrow Range	M	R	1, 2, 3
b. Wide Range	M	R	1, 2, 3
4. Suppression Pool Water Temperature	M	R*	1, 2
5. Suppression Chamber Pressure	M	R*	1, 2
6. Drywell Pressure	M	R	1, 2
7. Drywell Air Temperature	M	R*	1, 2
8. Drywell Oxygen Concentration	M	R	1, 2
9. Drywell Hydrogen Concentration Analyzer and Monitor	M	Q**	1, 2
10. Safety/Relief Valve Position Indicators	M	R	1, 2
11. Drywell High Range Radiation Monitors	M	R†	1, 2, 3
12. RHR Heat Exchanger Service Water Radiation Monitor	M	R	1, 2, 3
13. Refuel Platform Area Radiation Monitor	M	R	††
14. Neutron Flux			
a. APRM	M	R	1, 2
b. IRM	M	R	1, 2
c. SRM	M	R	1
15. Primary Containment Isolation Valve Position Indication	M†††	R***	1, 2

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TABLE 4.3.7.5-1 (Continued)

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATIONS

* Excludes sensors; sensor comparison shall be done in lieu of sensor calibration.

** Using sample gas containing:

a. One volume percent hydrogen, balance nitrogen.
b. Four volume percent hydrogen, balance nitrogen.

† The CHANNEL CALIBRATION shall consist of an electronic calibration of the channel, not including the detector, for range decades above 10 R/hr and a one point calibration check of the detector below 10 R/hr with an installed or portable gamma source.

†† When handling fuel or components in the fuel pool or reactor cavity.

††† Red or green or equivalent indicator shall be verified as indicating valve position.

*** The position indication verification to meet the requirements of ASME Section XI IWV-3300 shall suffice in meeting the requirement when performed at this frequency.

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Changes to Technical Specifications in the
Area of Administrative Controls

[illegible]
[illegible]

Subject: Changes to Technical Specifications in the area of Administrative Controls

The requested changes and justification for these changes were submitted to you in a letter dated July 21, 1986. Enclosed is a copy of that letter for your information.

CHANGE REQUESTED FOR CERTIFICATION