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MANGAN, C. V. Niagara Mohawk Power Corp.
RECIP. NAME RECIPIENT AFFILIATION
ADENSAM, E. G. BWR Project Directorate 3

SUBJECT: Requests change to Tech Spec Table 3.3.9-1 re identification number of reactor water level instrument. Request matches as-built design & correct typo transmitted in util 860819 ltr. Marked-up Tech Spec Table 3.3.9-1 encl.

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NIAGARA MOHAWK

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

NMP2L 0899
October 10, 1986

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

Niagara Mohawk requests a change to the Nine Mile Point Unit 2 Technical Specification Table 3.3.9-1 related to the identification number of a reactor vessel water level instrument.

The instrumentation number 2ISC*LSH162A,B,C should be 2ISC*LSH1624A,B,C. This request matches the as-built design and corrects the typographical error transmitted in our letter dated August 19, 1986 (NMP2L 0827). Enclosed is a marked-up change to Table 3.3.9-1 (page 3/4 3-112) of the Technical Specifications.

This change has been discussed with the Nuclear Regulatory Commission Technical Specifications reviewer.

Very truly yours,

C. V. Mangan
Senior Vice President

LL/pns
2125G
Enclosure

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

B001

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.

C V Mangan

Subscribed and sworn to before me, a Notary Public in and for the State of New York and County of Onondaga, this 10th day of October, 1986.

Christine Austin
Notary Public in and for
Onondaga County, New York

My Commission expires:
CHRISTINE AUSTIN
Notary Public in the State of New York
Qualified in Onondaga Co. No. 4787687
My Commission Expires March 30, 1987

TABLE 3.3.9-1

PLANT SYSTEMS ACTUATION INSTRUMENTATION

TRIP FUNCTION	INSTRUMENT NUMBER	MINIMUM OPERABLE CHANNELS (a)	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. Feedwater System/Main Turbine Trip System				
Reactor Vessel Water Level - High, Level 8	2ISC*~~LSH162A,B,C~~ LSH1624A,B,C	3	1	140
2. Service Water System				
a. Discharge Bay Level	2SWP*LS30A,B	2	1,2,3,4,5	142
b. Intake Tunnel 1 & 2 Water Temperature	2SWP*TSL64A,65A 2SWP*TSL64B,65B	1/Division 1/Division	1,2,3,4,5 1,2,3,4,5	144 144
c. Service Water Bay	2SWP*LS73A,B	2	1,2,3,4,5	143
d. Service Water Pumps Discharge Strainer Differential Pressure - Train "A"	2SWP1*PDSH1A,C,E	1/Strainer	1,2,3,4,5	146
e. Service Water Pumps Discharge Strainer Differential Pressure - Train "B"	2SWP1*PDSH1B,D,F	1/Strainer	1,2,3,4,5	146
f. Service Water Supply Header Discharge Water Temperature	2SWP*TY31A,B	2	1,2,3,4,5	147
g. Service Water Inlet Pressure for EDG*2 (HPCS, Division III)				
1) Division I Supply Header	2SWP*PSL95A	1	1,2,3,4,5	145
2) Division II Supply Header	2SWP*PSL95B	1	1,2,3,4,5	145

(a) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the Trip System in the tripped condition, except for discharge bay level and service water bay level which may be placed in an inoperable status for up to 4 hours without placing the Trip System in a tripped condition.

SEP 0 8 1986

FINAL DRAFT

September 29, 1986

Docket No. 50-410

DISTRIBUTION:
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Mr. C. V. Mangan, Senior Vice President
Niagara Mohawk Power Corporation
300 Erie Boulevard West
Syracuse, New York 13202

Dear Mr. Mangan:

Subject: Corrections to the Final Draft Technical Specifications for Nine Mile Point, Unit 2

On September 10 and 11, 1986, we sent you the revised pages to the Final Draft Technical Specifications which were the result of our review of your requested changes to the Technical Specifications through August 22, 1986. In subsequent discussions with your staff, errors in those pages were identified. The staff therefore reviewed the Final Draft Technical Specifications for additional errors.

The enclosed corrected pages for the Nine Mile Point, Unit 2 Technical Specifications contain changes resulting from either editorial or word processing errors and therefore do not alter the meaning of the specifications as provided in the final draft Technical Specifications provided on June 27, 1986 and September 10 and 11, 1986.

The enclosed corrected pages should replace the corresponding pages in your final draft Technical Specifications and should be included in your recertification of the revised final draft Technical Specifications as requested in our letter of September 10, 1986.

Sincerely,

/S/

Elinor G. Adensam, Director
BWR Project Directorate No. 3
Division of BWR Licensing

Enclosure:
As stated

cc: C. Schulten
D. Vassallo

BWD-3:DBL	LA:BWD-3:DBL	D:BWD-3:DBL
MHaughey/vag	EHylton	EAdensam
09/29/86	09/29/86	09/29/86

Mr. C. V. Mangan
Niagara Mohawk Power Corporation

Nine Mile Point Nuclear Station
Unit 2

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TABLE 4.11.1-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

LIQUID RELEASE TYPE	SAMPLING FREQUENCY	MINIMUM ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	LOWER LIMIT OF DETECTION (LLD)(a) (μCi/ml)
1. Batch Waste Release Tanks(b) a. 2LWS-TK4A b. 2LWS-TK4B c. 2LWS-TK5A d. 2LWS-TK5B	P Each Batch	P Each Batch	Principal Gamma Emitters(c)	5×10^{-7}
			I-131	1×10^{-6}
	P One Batch/M	One Batch/M	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
	P Each Batch	M Composite(d)	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	P Each Batch	Q Composite(d)	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}
2. Continuous Releases	Grab Sample M(e)	Grab Sample M(e)	Principal Gamma Emitters(c)	5×10^{-7}
			I-131	1×10^{-6}
a. Service Water Effluent A			Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
b. Service Water Effluent B			H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
c. Cooling Tower Blowdown	Grab Sample Q(e)	Grab Sample Q(e)	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}
d. Auxiliary Boiler Pump Seal and Sample Cooling Discharge (Service Water)	Grab Sample M(f)	Gram Sample M(f)	Principal Gamma Emitters(c)	5×10^{-7}
	Grab Sample Q(f)	Grab Sample Q(f)	H-3	1×10^{-5}

LIQUID EFFLUENTS

DOSE

3/4.11.1.2 (Continued)

Revision 1, October 1977 and R.G. 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977. This specification applies to the release of radioactive materials in liquid effluents from each unit at the site. For units with shared radwaste treatment systems, the liquid effluents from the shared system are to be proportioned among the units sharing that system.

3/4.11.1.3 LIQUID RADWASTE TREATMENT SYSTEM

The OPERABILITY of the liquid radwaste treatment system ensures that this system will be available for use whenever liquid effluents require treatment before release to the environment. The requirement that the appropriate portions of this system be used when specified provides assurance that the releases of radioactive materials in liquid effluents will be kept as low as is reasonably achievable. This specification implements the requirements of 10 CFR 50.36a, GDC 60 of Appendix A to 10 CFR 50 and the design objective given in Section II.D of Appendix I to 10 CFR 50. The specified limits governing the use of appropriate portions of the liquid radwaste treatment system were specified as a suitable fraction of the dose design objectives set forth in Section II.A of Appendix I to 10 CFR 50 for liquid effluents. This specification applies to the release of radioactive materials in liquid effluents from each unit at the site. For units with shared radwaste treatment systems, the liquid effluents from the shared system are to be proportioned among the units sharing that system.

3/4.11.1.4 LIQUID HOLDUP TANKS

The tanks listed in this specification include all those outdoor radwaste tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system.

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of 10 CFR 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an UNRESTRICTED AREA.

3/4.11.2 GASEOUS EFFLUENTS

3/4.11.2.1 DOSE RATE

This specification is provided to ensure that the dose rate at any time at and beyond the SITE BOUNDARY from gaseous effluents from all units on the site will be within the annual dose limits of 10 CFR 20 to UNRESTRICTED AREAS.

GASEOUS EFFLUENTS

DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIOACTIVE MATERIAL IN PARTICULATE FORM

3/4.11.2.3 (Continued)

milk and meat is assumed), and (4) deposition on the ground with subsequent exposure to man. This specification applies to the release of radioactive materials in gaseous effluents from each unit at the site. For units with shared radwaste treatment systems, the gaseous effluents from the shared system are proportioned among the units sharing that system.

3/4.11.2.4 & 3/4.11.2.5 GASEOUS RADWASTE TREATMENT SYSTEM AND VENTILATION EXHAUST TREATMENT SYSTEM

The OPERABILITY of the GASEOUS RADWASTE TREATMENT SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM ensures that the systems will be available for use whenever gaseous effluents require treatment before release to the environment. The requirement that the appropriate portions of these systems be used, when specified, provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept as low as is reasonably achievable. This specification implements the requirements of 10 CFR 50.36a, GDC 60 of Appendix A to 10 CFR 50, and the design objectives given in Section II.D of Appendix I to 10 CFR 50. Limits governing the use of appropriate portions of the system were specified as a suitable fraction of the dose design objectives set forth in Sections II.B and II.C of Appendix I to 10 CFR 50, for gaseous effluents. This specification applies to the release of radioactive materials in gaseous effluents from each unit at the site. For units with shared radwaste treatment systems, the gaseous effluents from the shared system are proportional among the units sharing that system.

3/4.11.2.6 EXPLOSIVE GAS MIXTURE

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the GASEOUS RADWASTE TREATMENT SYSTEM is maintained below the flammability limits of hydrogen and oxygen. Automatic control features are included in the system to prevent the hydrogen concentrations from reaching these flammability limits. These automatic control features include injection of dilutants to reduce concentrations below flammability limits. Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of GDC 60 of Appendix A to 10 CFR 50.

3/4.11.2.7 MAIN CONDENSER - OFFGAS

Restricting the gross radioactivity rate of noble gases from the main condenser offgas provides reasonable assurance that the total body exposure to an individual at the exclusion area boundary will not exceed a small fraction of the limits of 10 CFR 100 in the event this effluent is inadvertently discharged directly to the environment without treatment. This specification implements the requirements of GDC 60 and 64 of Appendix A to 10 CFR 50.

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The General Superintendent - Nuclear Generation shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during the Superintendent's absence.

6.1.2 The Station Shift Supervisor - Nuclear (or during the Supervisor's absence from the control room, a designated individual) shall be responsible for the control room command function. A management directive to this effect, signed by the Vice President - Nuclear Generation shall be reissued to all station personnel annually.

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for unit management and technical support shall be as shown on Figure 6.2.1-1.

UNIT STAFF

6.2.2 The unit organization shall be as shown on Figure 6.2.2-1 and:

a. Each on-duty shift shall be composed of at least the minimum shift crew shown in Table 6.2.2-1;

b. At least one Licensed Operator shall be in the control room when fuel is in the reactor. In OPERATIONAL CONDITIONS 1, 2, or 3, at least one Licensed Senior Operator or Licensed Operator shall be at the controls of the unit.

c. A Radiation Protection Technician* shall be on site when fuel is in the reactor;

d. At least two Licensed Operators shall be present in the control room during reactor startup, scheduled reactor shutdown, and during recovery from reactor trips.

e. A Licensed Senior Operator shall be required in the control room during OPERATIONAL CONDITIONS 1, 2, and 3 and when the emergency plan is activated. This may be the Station Shift Supervisor - Nuclear, the Assistant Station Shift Supervisor - Nuclear or other individuals with a valid senior operator license. When the emergency plan is activated in OPERATIONAL CONDITIONS 1, 2, or 3 the Assistant Station Shift Supervisor - Nuclear becomes the Shift Technical Advisor and the Station Shift

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* The Radiation Protection Technician and Fire Brigade composition may be less than the minimum requirements for a period of time not to exceed 2 hours, in order to accommodate unexpected absence, provided immediate action is taken to fill the required positions. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming crewman being late or absent.

REPORTING REQUIREMENTS

ROUTINE REPORTS

ANNUAL REPORTS

6.9.1.5 (Continued)

whole-body dose received from external sources should be assigned to specific major work functions.

b. The results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.5. The following information shall be included: (1) Reactor power history starting 48 hours before the first sample in which the limit was exceeded; (2) Results of the last isotopic analysis for radioiodine performed before exceeding the limit, results of analysis while the limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Cleanup system flow history starting 48 hours before the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

c. Documentation of all challenges to safety/relief valves; and

d. Any other unit unique reports required on an annual basis.

MONTHLY OPERATING REPORTS

6.9.1.6 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the main steam system safety/relief valves, shall be submitted monthly to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the Regional Office of the NRC no later than the 15th of each month following the calendar month covered by the report.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.9.1.7 Routine Annual Radiological Environmental Operating Reports covering the operation of the unit during the previous calendar year shall be submitted before May 1 of each year. The initial report shall be submitted before May 1 of the year after the plant achieves initial criticality.

The Annual Radiological Environmental Operating Report shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison, as appropriate, with preoperational studies, operational controls,

* A single submittal may be made for a multiple unit site. The submittal should combine those sections that are common to all units at the site.

REPORTING REQUIREMENTS

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORTS

6.9.1.7 (Continued)

previous environmental surveillance reports, and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of the land use census required by Specification 3.12.2.

The Annual Radiological Environmental Operating Reports shall include the results of analysis of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the OFFSITE DOSE CALCULATION MANUAL, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplemental report.

The reports shall also include the following: a summary description of the Radiological Environmental Monitoring Program; at least two legible maps* covering all sampling locations keyed to a table giving distances and directions from the centerline of one reactor; the results of licensee participation in the Interlaboratory Comparison Program, required by Specification 3.12.3; discussion of all deviations from the Sampling Schedule of Table 3.12.1-1; and discussion of all analyses in which the LLD required by Table 4.12.1-1 was not achievable.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT**

6.9.1.8 Routine Semiannual Radioactive Effluent Release Reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date the plant achieves initial criticality.

The Semiannual Radioactive Effluent Release Reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power

* One map shall cover stations near the SITE BOUNDARY; a second shall include the more distant stations.

** A single submittal may be made for a multiple unit site. The submittal should combine those sections that are common to all units at the site; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

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, 3
b. Wide Range	M	R	1, 2, 3
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. Suppression Chamber Air Temperature	M	R*	1, 2
7. Drywell Pressure			
a. Narrow Range	M	R	1, 2
b. Wide Range	M	R	1, 2
8. Drywell Air Temperature	M	R*	1, 2
9. Drywell Oxygen Concentration	M	R	1, 2
10. Drywell Hydrogen Concentration Analyzer and Monitor	M	Q**	1, 2
11. Safety/Relief Valve Position Indicators	M	R	1, 2
12. Drywell High Range Radiation Monitors	M	R†	1, 2, 3
13. RHR Heat Exchanger Service Water Radiation Monitor	M	R	1, 2, 3
14. Refuel Platform Area Radiation Monitor	M	R	††
15. Neutron Flux			
a. APRM	M	R	1, 2
b. IRM	M	R	1, 2
c. SRM	M	R	1
16. Primary Containment Isolation Valve Position Indication	M†††	R***	1, 2

INSTRUMENTATION

MONITORING INSTRUMENTATION

FIRE DETECTION INSTRUMENTATION

LIMITING CONDITIONS FOR OPERATION

3.3.7.8 As a minimum, the fire detection instrumentation for each fire detection zone shown in Table 3.3.7.8-1 shall be OPERABLE.

APPLICABILITY: Whenever equipment protected by the fire detection instrument is required to be OPERABLE.

ACTION:

a. With any, but not more than one-half the total in any fire zone, Function N* fire detection instruments shown in Table 3.3.7.8-1 inoperable, restore the inoperable Function N* instrument(s) to OPERABLE status within 14 days or within 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour.

b. With more than one-half the Function N* fire detection instruments in any fire zone shown in Table 3.3.7.8-1 inoperable or with any Functions S* or X* instruments shown in Table 3.3.7.8-1 inoperable, or with any two or more adjacent instruments shown in Table 3.3.7.8-1 inoperable, within 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour.

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

SURVEILLANCE REQUIREMENTS

4.3.7.8.1 Each of the above required fire detection instruments which are accessible during unit operation shall be demonstrated OPERABLE at least once per 6 months by performance of a CHANNEL FUNCTIONAL TEST. Fire detectors which are not accessible during unit operation shall be demonstrated OPERABLE by the performance of a CHANNEL FUNCTIONAL TEST during each COLD SHUTDOWN exceeding 24 hours unless performed in the previous 6 months.

4.3.7.8.2 The NFPA Standard 72D supervised circuits supervision associated with the detector alarms of each of the above required fire detection instruments shall be demonstrated OPERABLE at least once per 6 months.

4.3.7.8.3 The non-supervised circuits associated with detector alarms between the instruments and the control room shall be demonstrated OPERABLE at least once per 31 days.

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* These letters are found in the alpha-numeric fire zone designation and are explained in the footnote to Table 3.3.7.8-1

TABLE 4.3.7.11-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
4. Main Stack Effluent					
a. Noble Gas Activity Monitor †	D	M	R(a)	Q(c)	*
b. Iodine Sampler	W	NA	NA	NA	*
c. Particulate Sampler	W	NA	NA	NA	*
d. Flow-Rate Monitor	D	NA	R	Q	*
e. Sampler Flow-Rate Monitor	D	NA	R	Q	*

TABLE 3.3.9-2

PLANT SYSTEMS ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. Feedwater System/Main Turbine Trip System		
a. Reactor Vessel Water Level - High Level 8	≤202.3 in.*	≤209.3 in.
2. Service Water System		
a. Discharge Bay Level	≤275' Elev.	≤275' 2-3/4" Elev.
b. Intake Tunnel 1 & 2 Water Temperature	≥39°F	≥38°F
c. Service Water Bay	≥234' Elev.	≥233' 1-1/4" Elev.
d. Service Water Pumps Discharge Strainer Differential Pressure - Train "A"	≤10 psid	≤14.5 psid
e. Service Water Pumps Discharge Strainer Differential Pressure- Train "B"	≤10 psid	≤14.5 psid
f. Service Water Supply Header Discharge Water Temperature	NA	NA
g. Service Water Inlet Pressure for EDG*2 (HPCS, Division III)		
1) Division I Supply Header	≥25 psig	≥17.5 psig
2) Division II Supply Header	≥22 psig	≥17.5 psig

*See Bases Figure B3/4 3-1.

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

ISOLATION VALVE NO.	VALVE FUNCTION	VALVE GROUP	ISOLATION SIGNAL(a)	MAXIMUM CLOSING TIME (SECONDS)
2RHS*MOV142(j)(m)	RHS Drain to Radwaste Outside IV	4	A,Z,F,RM	30
2RHS*MOV149(j)(m)	RHS Drain to Radwaste Inside IV	4	A,Z,F,RM	25
2RHS*SOV35 A/B (j)(m)	RHS Sample HX Inside IVs	4	A,Z,F,RM	5
2RHS*SOV36 A/B (j)(m)	RHS Sample HX Outside IVs	4	A,Z,F,RM	5
2RDS*AOV124(k)	SCRAM Discharge volume vent	NA		NA
2RDS*AOV132(k)	SCRAM Discharge volume vent	NA		NA
2RDS*AOV123(k)	SCRAM Discharge volume drain	NA		NA
2RDS*AOV130(k)	SCRAM Discharge volume drain	NA		NA
B. Remote Manual				
2RHS*MOV15 A,B	Containment Spray to Drywell Outside IV's	12	RM	NA
2RHS*MOV 1 A,B,C	RHS Pump Suction Outside IVs	12	RM	NA
2RHS*MOV30 A,B	RHS Test Line to SP Outside IVs	12	RM	NA
2RHS*MOV25 A,B(n)	Containment Spray to Drywell Outside IVs	12	RM	NA
2RHS*MOV24 A,B,C	RHS/LPCI to RPV Outside IVs	12	RM	NA
2CSH*MOV118(n)	CSH Suction from SP Outside IV	12	RM	NA
2CSH*MOV105	HPCS Min Flow Bypass Outside IV	12	RM	NA
2CSH*MOV107	CSH to RPV Outside IV	12	RM	NA
2CSL*MOV112	CSL Suction from SP Outside IV	12	RM	NA
2CSL*MOV104	CSL to RPV Outside IV	12	RM	NA
2ICS*MOV136(n)	ICS Suction from SP Outside IV	12	RM	NA
2ICS*MOV143(n)	ICS Min flow to SP Outside IV	12	RM	NA

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

ISOLATION VALVE NO.	VALVE FUNCTION	VALVE GROUP	ISOLATION SIGNAL(a)	MAXIMUM CLOSING TIME (SECONDS)
D. Other				
Safety Relief				
2RHS*RV20 A,B,C(d)	RHS RV disch. to SP Outside IVs			
2RHS*RV61 A,B,C(d)	RHS RV disch. to SP Outside IVs			
2RHS*RV108(d)	RHS RV disch. to SP Outside IVs			
2RHS*RV110(d)	SDC to RHS Pump suction RV			
2RHS*RV139(d)	RHR Hdr. Flush to Radwaste RV			
2RHS*RV152(n)	SDC Supply from RCS RV Inside IV			
2RHS*RV56 A,B(d)	RHS HX shell side RVs			
2RHS*SV34 A,B(d)	RHS HX steam supply Safety valves			
2RHS*SV62 A,B(d)	RHS HX steam supply Safety valves			
2RHS*RVV35 A,B(d)	RHS Vacuum Breakers			
2CSL*RV105(d)	CSL RV Disch. to SP Outside IV			
2CSL*RV123(d)	CSL RV Disch. to SP Outside IV			
2RHS*RVV36 A,B(d)	RHS Vacuum Breakers			
2CCP*RV170(n)	CCP RV Discharge Inside IV			
2CCP*RV171(n)	CCP RV Discharge Inside IV			
2CSH*RV113(d)	CSH RV Disch. to SP Outside IV			
2CSH*RV114(d)	CSH RV Disch. to SP Outside IV			

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

ISOLATION VALVE NO.	VALVE FUNCTION	VALVE GROUP	ISOLATION SIGNAL(a)	MAXIMUM CLOSING TIME (SECONDS)
Excess Flow Check(e) Reactor Instrumentation Lines				
2ISC*EFV1	Inst. Line from MSS			
2ISC*EFV2	Inst. Line from N14,200°			
2ISC*EFV3	Inst. Line from N14,160°			
2ISC*EFV4	Inst. Line from N13,190°			
2ISC*EFV5	Inst. Line from N14,20°			
2ISC*EFV6	Inst. Line from N14,340°			
2ISC*EFV7	Inst. Line from N13,10°			
2ISC*EFV8	Inst. Line from N12,160°			
2ISC*EFV10	Inst. Line from N12,200°			
2ISC*EFV11	To 2ISC*FT47K,FT48B			
2ISC*EFV13	To 2ISC*FT47H			
2ISC*EFV14	Vessel Bottom Tap, loop A Jet Pump			I
2ISC*EFV15	Inst. Line from N12,340°			
2ISC*EFV17	Inst. Line from N12,20°			
2ISC*EFV18	To 2ISC*FT47J,FT48A			
2ISC*EFV20	To 2ISC*FT47E			
2ISC*EFV21	Vessel Bottom Tap for CSH, RDS			
2ISC*EFV22	Vessel Bottom Tap for WCS and Loop B J.P.			I
2ISC*EFV23	To 2ISC*FT48C and Postaccident Sampling			
2ISC*EFV24	To 2ISC*FT48D and Postaccident Sampling			
2ISC*EFV25	To 2ISC*FT47L			
2ISC*EFV26	To 2ISC*FT47C			
2ISC*EFV27	To 2ISC*FT47A			
2ISC*EFV28	To 2ISC*FT47R			
2ISC*EFV29	To 2ISC*FT47G			
2ISC*EFV30	To 2ISC*FT47N			
2ISC*EFV31	To 2ISC*FT48A			
2ISC*EFV32	To 2ISC*FT47T			
2ISC*EFV33	To 2ISC*FT47V,FT48C			

ELECTRICAL POWER SYSTEMS

ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITIONS FOR OPERATION

3.8.4.2 All primary containment penetration conductor overcurrent protective devices* shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one or more of the primary containment penetration conductor overcurrent protective devices* inoperable, declare the affected system or component inoperable and apply the appropriate ACTION statement for the affected system and:

   1. For 13.8-kV circuit breakers, deenergize the 13.8-kV circuits by tripping the associated redundant circuit breaker(s) within 72 hours and verify the redundant circuit breaker(s) to be tripped at least once every 7 days thereafter.

   2. For 600 volt MCC circuit breakers, remove the inoperable circuit breaker(s) from service by opening the breaker within 72 hours and verify the inoperable breaker(s) to be in the open position at least once every 7 days thereafter.

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

b. The provisions of Specification 3.0.4 are not applicable to overcurrent devices in 13.8-kV circuits which have their redundant circuit breakers tripped or to 600-volt circuits which have the inoperable circuit breaker disconnected.

SURVEILLANCE REQUIREMENTS

4.8.4.2 Each of the primary containment penetration conductor overcurrent protective devices* shall be demonstrated OPERABLE:

a. At least once per 18 months:

   1. By verifying that the medium voltage 13.8-kV circuit breakers are OPERABLE by selecting, on a rotating basis, at least 10% of the circuit breakers of each voltage level and performing:

* Excluded from this specification are those penetration assemblies that are capable of withstanding the maximum current available because of an electrical fault inside containment.

SPECIAL TEST EXCEPTIONS

3/4.10.7 SPECIAL INSTRUMENTATION - INITIAL CORE LOADING

LIMITING CONDITIONS FOR OPERATION

3.10.7 During initial core loading within the Startup Test Program the provisions of Specification 3/4.9.2 may be suspended provided that at least two source range monitor (SRM) channels with detectors inserted to the normal operating level are OPERABLE with:

a. One of the required SRM channels continuously indicating* in the control room,

b. One of the required SRM detectors located in the quadrant where CORE ALTERATIONS are being performed and the other required SRM detector located in an adjacent quadrant,**

c. The RPS "shorting links" shall be removed prior to and during fuel loading,

d. The reactor mode switch is OPERABLE and locked in the Refuel position.

APPLICABILITY: OPERATIONAL CONDITION 5

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving initial core loading.

SURVEILLANCE REQUIREMENTS

4.10.7.1 Within one hour prior to and at least once per 12 hours during the initial core loading verify that:

a. The above required SRM channels are OPERABLE by:

1. Performance of a CHANNEL CHECK***

2. Confirming that the above required SRM detectors are at the normal operating level and located in the quadrants required by Specification 3.10.7.

*Up to 16 fuel bundles may be loaded without a visual indication of count rate.

**The use of special movable detectors during CORE ALTERATIONS in place of the normal SRM nuclear detectors is permissible as long as these special detectors are connected to the normal SRM circuits.

***May be performed by use of movable neutron source.