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SUBJECT: Forwards marked-up FSAR pages, per verification of FSAR changes will be included in next FSAR update.

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

August 19, 1986
NMP2L 0825

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

As a result of Niagara Mohawk's verification of the Final Safety Analysis Report for Nine Mile Point Unit 2, we are submitting Final Safety Analysis Report pages marked to show the necessary changes.

Generally, these changes are minor and do not affect the Safety Evaluation Report. We have provided a summary of the changes and an explanation of each change to aid review of this material. Niagara Mohawk would appreciate your expeditious review of these items.

Typed versions of these changed pages will be provided as soon as possible. These changes will be included in a subsequent Final Safety Analysis Report update.

Very truly yours,

T. E. Lempges
Vice President
Nuclear Generation

WHB/TEL:saa
1948G

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

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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

T. E. Lempges, being duly sworn, states that he is 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.

T E Lempges

Subscribed and sworn to before me, a Notary Public in and for the State of New York and County of Onondaga, this 19th day of August, 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

SUMMARY OF FSAR CHANGES

Legend

R = Response to NRC Question or request, or SER Item.

E = Editorial or typographical change that has not affected basis of FSAR.

N = Nonsafety-related change in design, schedule, and/or procedure.

SN = Change to a safety-related item that has no effect on SER.

SS = Change to safety-related item that could affect the SER.

TS = Change that affects the Technical Specification.

SUMMARY OF FSAR CHANGES

Page	Change Description	Justification	Change Code	SER Impact	Tech. Spec. Impact
Tables 2.1-2 and 2.1.3	Remove information on Granby; insert information on Richland.	Grandby is not within 10 miles of Unit 2, and Richland is.	E	No	No
Pages 2.5-5 and 2.5-128	Revise statement that the NYSERDA and ESEERCO project reports will be available in the future.	These reports are now available.	E	No	No
Table 3.5-5	Change overspeed failure probability from "zero" to 3.140×10^{-6}.	This value was inadvertently left out.	E	No	No
Table 3.5-11	Change the overspeed failure probability from 1.591×10^{-13} to 2.206×10^{-13} and 1.313×10^{-13} to 1.308×10^{-13}, also change 3.521×10^{-13} to 3.523×10^{-13}.	These changes correct typographical errors.	E	No	No
Table 3.5-14	Change the overspeed failure probability from 6.524×15^{-11} to 6.524×10^{-11} and from 8.902×10^{-6} to 8.962×10^{-6}.	These changes correct typographical errors.	E	No	No
Appendix 3C	Change various pages from 3C-1 through 3C-20 to correct the as-built pipe whip study.	The as-built pipe whip study resulted in some changes to the FSAR. The actual study was submitted under separate cover.	E	No	No

SUMMARY OF FSAR CHANGES

Page	Change Description	Justification	Change Code	SER Impact	Tech. Spec. Impact
Table 6.2-32	Change the recirculation suction subcompartment line break values.	These values reflect as-built calculations and correction of typographical errors.	E	No	No
Table 6.2-45B	Change the total effective break area from 3.8043 to 5.072.	These changes are correction of errors in the table.	E	No	No
Tables 6.2-48 and 6.2-49	Change certain feedwater line break RPV annulus values.	These values reflect as-built calculations and correction of typographical errors.	E	No	No
Table 6.2-59D	Change aluminum weight from "41,500" to "40,300", change galvanized steel from "58,540" to "58,500" and "6,968" to "6,970".	These are based upon as-built calculations. Actual values are very close or less than previous values.	E	No	No
Tables 9.4-11 and 9.4-12	Change the chilled water and glycol heating equipment design data.	These changes reflect final as-built specification and and vendor data.	E	No	No
Pages 9.5-9, 9.5-10 9.5-12, 9.5-14, and 9.5-16	Remove code call paging from the dial telephone system.	Nine Mile Point Unit 2 does not have a code call paging system.	E	No	No

SUMMARY OF FSAR CHANGES

Page	Change Description	Justification	Change Code	SER Impact	Tech. Spec. Impact
Table 13.1-4	Change titles to match latest organization charts.	These changes are made to match the latest organization shown in our letter dated July 24, 1986, and Technical Specifications	N	No	No
Table 14.2-41	Change scope of preoperational test to remove hot sample testing.	A hot sample cannot be generated during preoperational testing. This will be verified during startup testing.	E	No	No

TABLE 2.1-2

1980 POPULATION AND POPULATION DENSITIES
FOR TOWNS AND CITIES WITHIN 16 KM (10 MI) OF UNIT 2*

	1980 Population	Population Density (people/sq km)
City of Oswego	19,793	1,029.0
Oswego (town)	7,865	116.9
Granby	6,341	55.2
Scriba	5,455	52.9
Volney	5,358	46.0
Mexico	4,790	41.8
Palermo	3,253	31.6
New Haven	2,421	31.7
Minetto	1,905	125.5
Richland	5,594	40.2

*These numbers are based on the entire town and not just the portion within the 16-km (10-mi) radius.

TABLE 2.1-3

1970-1980 POPULATION GROWTH FOR TOWNS AND CITIES
WITHIN 16 KM (10 MI) OF UNIT 2* 7

	1970	1980	1970-1980 Percent Change	
City of Oswego	20,913	19,793	-5.4	
Oswego (town)	6,514	7,865	20.7	
Granby	4,718	6,341	34.4	7
Scriba	3,619	5,455	50.7	
Volney	4,520	5,358	18.5	
Mexico	4,174	4,790	14.8	
Palermo	2,321	3,253	40.2	7
New Haven	1,845	2,421	31.2	
Minetto	1,688	1,905	12.9	
Richland	5,324	5,594	5.1	

*Based on total town population.

A geothermal test well drilled in the Spring 1982, penetrated the basement near Auburn, NY. Preliminary results ~~will be reviewed when available,~~ and basement composition compared to information reported herein.

Deep well data from western New York State[11,12] indicate extensive areas of granite and marble. Locally, gabbroic intrusives, surrounded by areas of metavolcanics are known[12].

The Central Metasedimentary Belt of the Canadian Shield in Ontario consists of rock types similar to those extracted from deep wells in central and western New York State. In the belt, local mafic intrusives and extensive felsic intrusive bodies are surrounded by highly deformed gneisses, marbles, and schists[6]. This information suggests that prominent local density and magnetic contrasts may exist in the basement rocks. Moreover, the pattern of regional Bouguer gravity anomalies (Figure 2.5-8)[13,14] and regional aeromagnetic anomalies (Figure 2.5-9)[15] confirms such contrasts. From these observations, it appears reasonable to suggest that the Central Metasedimentary Belt extends southward from the Canadian Shield into central and western New York. It is uncertain, however, how far south this belt continues.

Paleozoic Sedimentary Rocks

The Paleozoic formations in northern central New York State form a wedge that thickens to the south, away from the Canadian Shield. The strata are relatively flat-lying but have been rotated slightly, exhibiting a gentle, regional gradient to the south (approximately 9.5 m/km, 50 ft/mi). In the vicinity of the Nine Mile Point site, Late Ordovician formations have been exposed by erosion that removed younger units (Figure 2.5-3). Farther south, younger Silurian, Devonian, and Carboniferous formations are still preserved in the central part of the Appalachian Basin.

The basal units of this sedimentary wedge consist of an Early Cambrian clastic sequence (the Potsdam Sandstone) and an Ordovician carbonate sequence (the Beekmantown, Black River, and Trenton Groups) both of which were deposited in a relatively stable shelf environment (Figures 2.5-7 and 2.5-10). These strata are overlain by a Late Ordovician-Early Silurian clastic sequence, which constitutes the Utica Shale and the Lorraine Group (Whetstone Gulf and Pulaski Formations), the Oswego Sandstone, the Queenston Formation, and the Grimsby Formation[16]. This sedimentary sequence represents a transition from a shelf to a terrestrial

The results of this test well are presented in NYSERDA Report 84-18 - Final Report dated December 1983, prepared by Donohue Anstey and Morrill, Boston, Massachusetts.

[illegible]

2.5.4.1.4 Unrelieved Residual Stresses

Regional Stress Conditions

A review of stress determinations in the Lake Ontario region was undertaken in 1978[94] to assist in evaluating the stress conditions at Nine Mile Point. Five groups of data were considered with the intention of identifying regional characteristics of rock stress. Of these five groups, only the results of overcoring and hydrofracturing are field measurements of in situ stress and even these measurements are based on simple models that result in approximations of the in situ stress. Additionally, analyses of earthquake seismic records, surface strain-relief observations and trends of postglacial buckles were compiled and assessed. These measurements and indicators of stress provide a profile of stress conditions from the surface and near surface (surface strain relief, postglacial buckles, and overcoring) through intermediate depths (hydrofracturing) to depths of up to 15,240 m (50,000 ft) (earthquake focal mechanisms).

The available hydrofracture test data in western New York State are presented in Volume III of Reference 94. Maximum horizontal stresses of 105 to 160 kg/sq cm (1,495 to 2,275 psi) are reported from tests ranging from 152 to 528 m (500 to 1,700 ft) below the ground surface. The maximum principal stresses, as determined by hydrofracturing, generally trend east-northeast. Additionally, hydrofracturing tests were completed by the USGS in a well near Auburn, NY, in the spring of 1982. These tests, conducted as part of a NYSERDA and ESEERCO-sponsored geothermal project, were performed in the Paleozoic cover rock. but test results were not available at the time of the writing of this document.

Table 2.5-3 is an updated summary of near-surface overcore test sites and results in the region. An outstanding characteristic of the data is the variability of both the maximum and minimum horizontal stress magnitudes, stress differences, and stress orientations. The reported magnitude of maximum horizontal stress at depths of less than 26 m (85 ft) varies from -49 to 302 kg/sq cm (-700 to 4,300 psi) and averages approximately 84 kg/sq cm (1,200 psi). Despite the variability of stress magnitude and orientation, the regional data indicate a horizontal stress that is higher than the value that could be attributed to a simple gravitational loading by the present overburden.

The result of these tests are presented in Research Report EP82-11 - Final Report, dated April 1984, prepared by United States Geological Survey at Menlo Park, California and Schlumberger-Doll Research, Ridgefield, Connecticut.

TABLE 3.5-5

DAMAGE PROBABILITY DUE TO LOW TRAJECTORY TURBINE MISSILES FROM JAMES A. FITZPATRICK POWER STATION STRIKING PLANT REGIONS AT UNIT 2

Manufacturer's Probability

Safety-Related Region	Design Overspeed Failure			Destructive Overspeed Failure			
	P_1	P_2XP_3	$P_1XP_2XP_3$	P_1	P_2XP_3	$P_1XP_2XP_3$	
Reactor building	1.37×10^{-8}	5.807×10^{-4}	7.956×10^{-12}	1.37×10^{-8}	5.446×10^{-5}	7.461×10^{-13}	
Control building	1.37×10^{-8}	1.712×10^{-4}	2.345×10^{-12}	1.37×10^{-8}	1.508×10^{-5}	2.066×10^{-13}	
Diesel generator building	1.37×10^{-8}	5.453×10^{-5}	7.471×10^{-13}	1.37×10^{-8}	5.099×10^{-6}	6.986×10^{-14}	
Screenwell building - service water pump room	1.37×10^{-8}	3.140×10^{-6}	0	1.37×10^{-8}	0	0	
Standby gas treatment and RB access lock	1.37×10^{-8}	1.553×10^{-4}	2.128×10^{-12}	1.37×10^{-8}	1.457×10^{-5}	1.996×10^{-13}	
Radwaste building	1.37×10^{-8}	2.397×10^{-4}	3.288×10^{-12}	1.37×10^{-8}	2.243×10^{-5}	3.073×10^{-13}	26
Auxiliary service building and north and south auxiliary bays	1.37×10^{-8}	4.616×10^{-5}	6.324×10^{-13}	1.37×10^{-8}	4.319×10^{-6}	5.917×10^{-14}	
Intake and discharge shaft area	1.37×10^{-8}	7.182×10^{-5}	9.839×10^{-13}	1.37×10^{-8}	6.688×10^{-6}	9.163×10^{-14}	
Main steam tunnel	1.37×10^{-8}	3.830×10^{-5}	5.247×10^{-13}	1.37×10^{-8}	3.582×10^{-6}	4.907×10^{-14}	

NOTE: Manufacturer's probability: $P_1 = 1.37 \times 10^{-8}$

TABLE 3.5-11

DAMAGE PROBABILITY DUE TO HIGH TRAJECTORY TURBINE MISSILES FROM JAMES A. FITZPATRICK POWER STATION STRIKING PLANT REGIONS AT UNIT 2

Manufacturer's Probability

Safety-Related Region	Design Overspeed Failure			Destructive Overspeed Failure			
	P_1	$P_2 \times P_3$	$P_1 \times P_2 \times P_3$	P_1	$P_2 \times P_3$	$P_1 \times P_2 \times P_3$	
Reactor building	1.37×10^{-8}	1.161×10^{-5}	1.591×10^{-13} 2.206	1.37×10^{-8}	4.839×10^{-6}	6.629×10^{-14}	26
Control building	1.37×10^{-8}	8.779×10^{-6}	1.203×10^{-13}	1.37×10^{-8}	2.641×10^{-5}	3.618×10^{-13}	
Diesel generator building	1.37×10^{-8}	4.800×10^{-6}	6.576×10^{-14}	1.37×10^{-8}	2.695×10^{-5}	3.692×10^{-13}	
Screenwell building - service water pump room	1.37×10^{-8}	3.760×10^{-6}	5.151×10^{-14}	1.37×10^{-8}	4.449×10^{-7}	6.095×10^{-15}	
Standby gas treatment and RS access lock	1.37×10^{-8}	4.272×10^{-5}	5.853×10^{-13}	1.37×10^{-8}	9.526×10^{-6}	1.305×10^{-13}	26
Radwaste building	1.37×10^{-8}	9.584×10^{-6}	1.313×10^{-13} 08	1.37×10^{-8}	8.962×10^{-6}	1.228×10^{-13}	26
Auxiliary service building and north and south auxiliary bays	1.37×10^{-8}	1.449×10^{-4}	1.985×10^{-12}	1.37×10^{-8}	2.570×10^{-5}	3.521×10^{-13} 3.523	
Intake and discharge shaft area	1.37×10^{-8}	3.010×10^{-6}	4.124×10^{-14}	1.37×10^{-8}	2.400×10^{-6}	3.288×10^{-14}	26
Main steam tunnel	1.37×10^{-8}	6.524×10^{-7}	8.938×10^{-15}	1.37×10^{-8}	1.267×10^{-6}	1.736×10^{-14}	26

NOTE: Manufacturer's probability: $P_1 = 1.37 \times 10^{-8}$

TABLE 3.5-14

DAMAGE PROBABILITY DUE TO HIGH TRAJECTORY TURBINE MISSILES FROM JAMES A. FITZPATRICK POWER STATION STRIKING PLANT REGIONS AT UNIT 2

NRC Probability

Safety-Related Region	Design Overspeed Failure			Destructive Overspeed Failure			
	p_1	$p_2 X p_3$	$p_1 X p_2 X p_3$	p_1	$p_2 X p_3$	$p_1 X p_2 X p_3$	
Reactor building	1×10^{-4}	1.161×10^{-5}	1.161×10^{-9}	1×10^{-4}	4.839×10^{-6}	4.839×10^{-10}	26
Control building	1×10^{-4}	8.779×10^{-6}	8.779×10^{-10}	1×10^{-4}	2.641×10^{-5}	2.641×10^{-9}	
Diesel generator building	1×10^{-4}	4.800×10^{-6}	4.800×10^{-10}	1×10^{-4}	2.695×10^{-5}	2.695×10^{-9}	
Screenwell building	1×10^{-4}	3.760×10^{-6}	3.760×10^{-10}	1×10^{-4}	4.449×10^{-7}	4.449×10^{-11}	
Standby gas treatment and RR access lock	1×10^{-4}	4.272×10^{-5}	4.272×10^{-9}	1×10^{-4}	9.526×10^{-6}	9.526×10^{-10}	
Radwaste building	1×10^{-4}	9.584×10^{-6}	9.584×10^{-10}	1×10^{-4}	8.902×10^{-6} 62	8.902×10^{-10} 62	
Auxiliary service building and north and south auxiliary bays	1×10^{-4}	1.449×10^{-4}	1.449×10^{-8}	1×10^{-4}	2.570×10^{-5}	2.570×10^{-9}	26
Intake and discharge shaft area	1×10^{-4}	3.010×10^{-6}	3.010×10^{-10}	1×10^{-4}	2.400×10^{-6}	2.400×10^{-10}	26
Main steam tunnel	1×10^{-4}	6.524×10^{-7}	6.524×15^{-11} 10	1×10^{-4}	1.267×10^{-6}	1.267×10^{-10}	

NOTE: NRC probability: $p_1 = 1 \times 10^{-4}$

APPENDIX 3C

FAILURE MODE ANALYSIS FOR PIPE BREAKS AND CRACKS

3C.1 INTRODUCTION

This appendix describes the specific pipe failure protection provided to satisfy the requirements of Section 3.6.1A and demonstrates that the essential systems, components, and equipment are not adversely affected by pipe breaks or cracks.

The information provided by this appendix is in four sections: 3C.2, a discussion of high energy pipe breaks and the effects of pipe whip; 3C.3, a discussion of the effects of jet impingement; 3C.4, a discussion of moderate energy pipe cracks and the effects of spraying; and 3C.5, a discussion of flooding as a result of breaks or cracks.

Subcompartment pressurization is discussed in detail in Appendix 3B.

This appendix does not address the specific protection of field-routed essential instrument tubing or electrical conduit. However, these items are protected in accordance with the requirements of Section 3.6.1A.

For a detailed discussion of break/crack locations and types, break exclusion areas, guard pipes, and whip restraints that are frequently mentioned in this appendix, refer to Sections 3.6.1A and 3.6.2A.

3C.2 HIGH ENERGY PIPE BREAKS AND EFFECTS OF PIPE WHIP

The following systems are described in the noted sections:

Main steam piping system	3C.2.1
Feedwater piping system	3C.2.2
Reactor recirculation	3C.2.3
RCIC system	3C.2.4
LPCS/HPCS system	3C.2.5
RHR system (LPCI mode)	3C.2.6

12 RHR system (shutdown mode) 3C.2.7

RWCU system 3C.2.8

RPV vent line/RCIC head spray 3C.2.9

Main steam safety relief valve (SRV) piping 3C.2.10

3-inch and smaller high energy piping 3C.2.11

Each section references appropriate isometric drawings with break location and restraints. In addition, composite drawings showing pipe/equipment/room configurations are
12 provided in Figures 3.6A-52 through 3.6A-60 but are not specifically referenced.

The only pipe breaks of concern in the non-Category I turbine building were those having a potential to impact safety-related equipment in adjacent buildings. Although the reactor protection sensors for turbine stop and control valve scram initiation are located in the turbine building, they are not considered essential in the evaluation of low probability events such as pipe breaks. Other scram signals, not related to line breaks in the turbine building, are available as backup.

12 3C.2.1 Main Steam Piping System (MSS)

The locations of postulated pipe breaks and pipe whip
12 restraints are shown on Figures 3.6A-12 through 3.6A-14, ~~and Tables 3.6A-28 and 3.6A-29. The results of the associated stress calculations are summarized in~~
12 ~~Tables 3.6A-8 through 3.6A-9~~.

Each of the four main steam lines is welded to the appropriate reactor nozzle (el 321 ft 11 7/8 in) above the top of the shield wall. After the first elbow, each line runs downward to approximate el 293 ft 9 in, and then horizontally to the third elbow where it runs downward to approximate el 251 ft 6 in. It then passes through the containment penetrations, the reactor building steam tunnel, and into the turbine building.

The break exclusion zone, as described in Section 3.6.2.1.5A, High-Energy Fluid Systems, Item 2,
12 starts inboard of the zero-gap restraint adjacent to the inboard isolation valve and extends just beyond the jet impingement wall of the reactor building steam tunnel which

[illegible]

contains the outboard zero-gap restraint. ~~The safety classifications of the various portions of the main steam piping are given in Tables 3.6A-2 through 3.6A-9~~. 12

A total of 18 safety/relief valves are mounted on the horizontal runs between the reactor and the first isolation valves inside the containment. The discharge piping from these valves is normally unpressurized; therefore, there is no potential for pipe whip or similar hazards. (Pipe whip prevention of the relief valve discharge lines is discussed in Section 3C.2.10.)

In addition, a 10-in line branching from main steam line 2MSS-026-44-1 supplies steam to the RCIC turbine and to the RHR heat exchanger. This line passes through the RCIC pipe chase and is discussed in the analysis of the RCIC system.

Inside Containment

The main steam piping, if allowed to whip, can impact targets such as the 12-in LPCI piping, feedwater piping, structural steel at various elevations, the containment liner, and the biological shield wall.

To preclude any likelihood of loss of an essential system, or structural integrity of the containment or Category I walls, a total of 36 restraints have been installed inside the containment for the main steam system.

For example, LPCI line 2RHS-012-8-1 is protected by restraints 2MSS*PRR021A, 2MSS*PRR024, and 2MSS*PRR031A. LPCI line 2RHS-012-125-1 is protected by restraints 2MSS*PRR004 and 2MSS*PRR016. ~~Table 3.6A-28 identifies all targets inside the containment and the protecting restraints~~. 12

Restraints 2MSS-PRR005, 2MSS-PRR006, 2MSS-PRR017, 2MSS-PRR018, 2MSS-PRR025, 2MSS-PRR026, 2MSS-PRR036, and 2MSS-PRR-037 protect the MSIV for breaks inside containment and ensure that stress allowables are within the limits defined for the break exclusion region.

Inside Steam Tunnels

The main steam piping, from the zero-gap restraints inboard of the first isolation valve (inside the containment) to and including zero-gap restraints 2MSS*PRR101, 2MSS*PRR111, 2MSS*PRR121 and 2MSS*PRR131 at the jet impingement wall, meets the stress criteria for no postulated breaks, as

[illegible]

discussed in Section 3.6A and therefore is defined as a break exclusion zone.

From the jet wall, the four 28-in main steam lines (north inner and outer loops, south inner and outer loops) run straight for approximately 8 ft and then run vertically upward to el 295 ft 1 in before making another turn above the 12-line wall, where they connect to the main steam headers.

Pipe whip of the main steam lines in the steam tunnel area has been precluded by the placement of 20 restraints shown
12 on Figure 3.6A-14 and Table 3.6A-29.

Turbine Building

There are no breaks postulated in the turbine building because there are no essential targets in the turbine building.

Conclusions

Using very conservative assumptions and criteria, no postulated failure of the main steam lines can cause additional damage that could impair the ability to safely shut down the reactor or that could increase the offsite radiation effects beyond the limits of 10CFR100.

3C.2.2 Feedwater Piping System (FWS)

The locations of postulated pipe breaks and pipe whip restraints are shown on Figures 3.6A-19 through 3.6A-21~~, and Tables 3.6A-34 and 3.6A-35~~. ~~The results of the associated stress calculations are summarized in Tables 3.6A-13 through 3.6A-16~~.

12 The two feedwater loops each consist of three 12-in discharge lines that connect to the RPV at el 309 ft 1 in. The lines pass through the BSW openings and drop down to the 24-in/18-in/12-in header line at el 292 ft 8 in. The header lines then drop down to el 257 ft 0 in and penetrate the primary containment into the main steam tunnel. The lines run straight through the tunnel, then turn up and loop to the turbine building.

The feedwater piping, from the zero-gap restraints inboard of the first isolation valve (inside containment) to and including zero-gap restraints at the jet impingement wall, meets the criteria for no postulated breaks, as discussed in Section 3.6A, and therefore, is defined as a break exclusion zone.

Inside Containment

The feedwater piping, if allowed to whip, can impact targets such as the 12-in LPCI piping, HPCS piping, structural steel at various elevations, the containment liner, and the BSW.

To preclude any likelihood of loss of an essential system or structural integrity of the containment, restraints have been installed inside the containment for the feedwater system.

For example, LPCI line 2RHS-012-125-1 is protected by restraints 2FWS*PRR010, 2FWS*PRR012, 2FWS*PRR009, and 2FWS*PRR014. The HPCS piping is protected by restraint 2FWS*PRR001.

Restraints 2FWS*PRR017, 018, 035, and 036 protect the FWS check valves (2FWS*V12A and V12B) for breaks inside the containment and ensure that the stresses in the break exclusion region are within allowable limits. ~~Table 3.6A-24 identifies targets that would be affected by uncontrolled pipe whip and protection requirements~~.

Main Steam Tunnel

The break exclusion zone starts inboard of the zero-gap restraint adjacent to the inboard check valve and extends just beyond the jet impingement wall of the reactor building steam tunnel which contains the outboard zero-gap restraint. 12
These zero-gap restraints also are used to prevent overstressing the penetration or disabling the outboard isolation valves (2FWS*AOV23A and B) following postulated breaks.

From the jet wall, the two 24-in feedwater lines run straight until passing to the west side of 12-line wall. Pipe whip of the feedwater lines between 11-line and 12-line walls has been precluded by the placement of ~~eight~~ restraints shown on Figure 3.6A-21~~, and Table 3.6A-25.~~

Conclusions

Using very conservative assumptions and criteria, no postulated failure of the feedwater lines can cause additional damage that could impair the ability to safely shut down the reactor or that could increase the offsite radiation effects beyond the limits of 10CFR100.

[illegible]

3C.2.3 Reactor Recirculation System (RCS)

The locations of the postulated pipe breaks and pipe whip restraints are shown on Figures 3.6A-29 ~~and 3.6A-30, and Table 3.6A-43~~. The ~~stress results~~ pipe whip restraints, break locations, and types are the responsibility of GE and are described in Section 3.6B.

The two recirculation loops consist of a 24-in suction line that exits the RPV at el 282 ft 4 3/8 in. The line penetrates the BSW opening (which contains a flow diverter to minimize annulus pressurization) and drops to el 249 ft 5 3/4 in. The lines then run horizontally to the RCS pump. On the discharge side of the pump the lines run horizontally at el 250 ft and turn up at 90 deg and 270 deg. At el 275 ft 0 7/8 in, 16-in headers loop around the RPV, and 12-in risers branch off the headers at 30-deg increments and discharge into the RPV.

The recirculation piping, if allowed to whip, can impact targets such as structural steel platforms, SRV lines, and the RCS pump supports.

12 To preclude any likelihood of loss of an essential system or structural integrity of the containment, ~~30 GE and 6 SWEC~~ restraints have been installed for the recirculation system. Additionally, the BSW doors have been designed for pipe rupture loads.

For example, pipe whip due to nozzle breaks on the 12-in discharge lines will be prevented by the GE strap-type restraints just after the first elbow. The BSW doors also will be impacted as the pipe whips away from the RPV and will provide additional control. ~~Table 3.6A-43 identifies targets that would be affected by uncontrolled pipe whip and protection requirements~~.

<u>Conclusions</u>

Using very conservative assumptions and criteria, no postulated failure of the RCS lines can cause additional damage that could impair the ability to safely shut down the reactor or that could increase the offsite radiation effects beyond the limits of 10CFR100.

3C.2.4 Reactor Core Isolation Cooling System (RCIC)

The location of postulated pipe breaks and pipe whip restraints are shown on Figures 3.6A-22 and 3.6A-24. ~~and Tables 3.6A-36 and 3.6A-38. The results of the associated~~

s~~tress calculations are summarized in Tables 3.6A-17 and 3.6A-18~~.

The RCIC 10-in line branches off the main steam line (2MSS-026-44-1) at el 302 ft 2 in. The line runs through the primary containment and drops to el 263 ft 7 1/2 in, passing through its penetration at azimuth 185 deg. Once outside the primary containment, the line drops into the RCIC pipe chase. At el 234 ft there is an interconnection between the RCIC and RHR systems, where the RCIC line reduces in size from 10 in to 4 in and leads into the RCIC turbine room at el 175 ft. The high-energy portion of the line stops at normally closed valves 2ICS*MOV120 and 2RHS*MOV22A. The RCIC head spray 6-in line attaching to the RPV head is high energy up to normally closed valve 2ICS*AOV157 (Section 3C.2.9).

<u>Inside Containment</u>

The RCIC piping, if allowed to whip, can impact targets such as the containment liner, low-pressure core spray system, and structural steel.

To preclude any likelihood of loss of an essential system or structural integrity of the containment, ~~seven~~ restraints have been installed inside the containment for the RCIC system. 12

For example, pipe whip due to a break at the connection to the main steam line will be prevented by 2ICS*PRR001. ~~Table 3.6A-36 identifies targets that would be affected~~ by ~~uncontrolled pipe whip and protection requirements~~.

<u>Outside Containment</u>

To preclude target impacts and pipe whip of the RCIC piping outside containment, ~~eight~~ restraints have been installed, as shown on Figure 3.6A-24. T~~able 3.6A-39 identifies targets that would be affected by uncontrolled pipe whip and protection requirements~~.

<u>Conclusions</u>

Using very conservative assumptions and criteria, no postulated failure of the RCIC lines can cause additional damage that could impair the ability to safely shut down the reactor or that could increase the offsite radiation effects beyond the limits of 10CFR100.

3C.2.5 LPCS/HPCS System

3C.2.5.1 Low-Pressure Core Spray (LPCS)

The locations of postulated pipe breaks and pipe whip restraints are shown on Figure 3.6A-26. ~~and Table 3.6A-40.~~ The ~~results of the associated stress calculations are summarized in Table 3.6A-20 and~~ The system description is detailed in Section 6.3.2.2.3.

The LPCS 10-in line is welded to the reactor nozzle (el ~~307 ft 11 1/8 in~~ 308) and passes through the BSW. In the primary containment the line expands to 12 in and runs to 2CSL*AOV101, which is the end of the high-energy portion of the line. After the valve, the line drops to el 295 ft where it passes through its primary containment penetration.

The LPCS piping, if allowed the to whip, can impact targets such as ~~structural steel~~, containment liner, and biological shield wall. To preclude any likelihood of loss of structural integrity of the containment, a total of three restraints are installed for the LPCS system as shown on Figure 3.6A-26. ~~and Table 3.6A-40~~.

12

3C.2.5.2 High-Pressure Core Spray (HPCS)

The locations of postulated pipe breaks and pipe whip restraints are shown on Figure 3.6A-25. ~~and Table 3.6A-39.~~ ~~The results of the associated stress calculations are summarized in Table 3.6A-19, and~~ The system description is detailed in Section 6.3.2.2.1.

The HPCS 10-in line is welded to the reactor nozzle (el 307 ft 11 3/16 in) and passes through the BSW. In the primary containment the line expands to 12 in and runs to 2CSH*AOV108, which is the end of the high-energy portion of the line. After the valve, the line drops to el 291 ft 11 11/16 in where it passes through its primary containment penetration.

The HPCS piping, if allowed to whip, can impact targets such as the containment liner and biological shield wall. To preclude any likelihood of loss of structural integrity of the containment, a total of five restraints are installed inside containment for HPCS.

For example, pipe whip due to a break at AOV108 will be prevented by restraint 2CSH*PRR001, and 2CSH*PRR003A at the BSW will provide additional (2CSH*PRR001) control. ~~Table 3.6A-39~~

~~identifies all targets that would be affected~~ by ~~uncontrolled pipe whip and protection requirements~~.

Conclusions

Using very conservative assumptions and criteria, no postulated failure of the LPCS/HPCS lines can cause additional damage that could impair the ability to safely shut down the reactor or that could increase the offsite radiation effects beyond the limits of 10CFR100.

3C.2.6 RHR System (RHR-LPCI Mode)

The low-pressure coolant injection subsystem is an operating mode of the RHR system. The locations of postulated pipe breaks and pipe whip restraints are shown on Figure 3.6A-28 ~~and Table 3.6A-12~~. T~~he results of the associated stress calculations are summarized in Table 3.6A-22, and~~ the system description is detailed in Section 6.3.2.2.4.

The three high-energy lines for the LPCI mode are 12-in lines welded to the reactor nozzles at el 299 ft 0 3/8 in. The lines pass through the BSW openings and loop to their isolation valves, which end the high-energy portion of this system. After the valves, the lines drop to their primary containment penetrations.

The RHR, if allowed to whip, can impact targets such as the containment liner, BSW, and star truss. To preclude any likelihood of loss of structural integrity of the containment, the restraints as shown on Figure 3.6A-28 are installed for the LPCI mode of the residual heat removal system. 12

3C.2.7 Residual Heat Removal (Shutdown Mode)

The locations of postulated pipe breaks and pipe whip restraints are shown on Figure 3.6A-27 ~~and Table 3.6A-41~~. Th~~e results of the associated stress calculations are~~ su~~mmarized in Table 3.6A-21~~.

The three high-energy portions of this system branch off the reactor recirculation piping at el 266 ft 10 7/8 in and el 271 ft 1 7/8 in and run to their isolation valve inside the primary containment. The suction line is 20 in, and the two discharge lines are 12 in. After isolation valves 2RHS*AOV39A, 39B, and 2RHS*MOV112, these lines drop to el 249 ft 6 in and el 247 ft where they penetrate the primary containment wall.

To preclude target impacts and pipe whip of the RHR piping, six restraints, two per line, have been installed. ~~Table 3.6A-27 identifies targets that would be affected by uncontrolled pipe whip and protection requirements~~. The two discharge RHR lines have jet impingement source shields around the first elbow after they branch off the RCS pipes. These source shields protect the CRD from jet impingement loads (Section 3C.3).

Conclusions

Using very conservative assumptions and criteria, no postulated failure of the RHR lines can cause additional damage that could impair the ability to safely shut down the reactor or that could increase the offsite radiation effects beyond the limits of 10CFR100.

3C.2.8 Reactor Water Cleanup System (RWCU)

The locations of postulated pipe breaks and pipe whip restraints are shown on Figures 3.6A-31 through 3.6A-41. ~~and Tables 3.6A-44 and 3.6A-45~~. ~~The results of the associated stress calculations are summarized in Tables 3.6A-24 and 3.6A-25~~.

This system has three suction lines: two 4-in lines
12 branching from the recirculation piping and a 2-in line from the RPV bottom head. The 2-in line expands to 2.5 in and passes through the reactor pedestal at el 263 ft; this line expands again to 4 in inside the primary containment. The three 4-in suction lines lead into an 8-in header line which passes through the primary containment wall at azimuth 185 deg. Outside containment the RWCU lines branch into various process equipment and discharge into the feedwater piping, as shown on Figures 3.6A-31 through 3.6A-41.

Inside Containment

The RWCU piping, if allowed to whip, can impact targets such as reactor pedestal, CRD, and main steam safety relief valve piping.

To preclude any likelihood of loss of an essential system or structural integrity of containment, 20 restraints have been installed inside the containment for the reactor water cleanup system. ~~Table 3.6A-44 identifies targets that would be affected by uncontrolled pipe whip and protection requirements~~.

establish a break exclusion zone and protect the floor at elevation 240 ft.

Outside Containment

Most breaks in the RWCU outside containment are isolated with nonsafety-related targets such as walls, ceilings, floors, and the failed piping system itself. A total of 10 restraints are provided to protect safety-related targets such as the primary containment wall and penetrations. Table 3.6A-45 identifies targets and protection requirements. A study was done to determine the pipe whip and jet impingement consequences for a hypothetical break at the terminal end of the reactor water cleanup line connection to the feedwater thermal tee. The conclusions of the study are discussed below:

1. The reactor water cleanup pipe would impact a maintenance platform at el 251 ft. This is acceptable, since the platform's failure causes no secondary damage to other safety-related equipment and does not impair plant shutdown.

2. The main steam penetration Z1A and the main steam line (spool piece 2MSS-026-154-1) would be struck by the whipping reactor water cleanup pipe. In accordance with Section 3.6.1.1.a, this will not result in the rupture of the main steam line or penetration, since the whipping pipe is smaller in diameter and thickness than the pipes it strikes. Additionally, the function of the outside main steam isolation valve will not be impaired. 18

3. The feedwater line is the only jet impingement target affected. Stresses resulting in the feedwater line are acceptable and within allowable limits. In addition, the feedwater isolation valves, penetration, and jet impingement wall will not be overloaded due to the effects of jet impingement. Since the stresses induced by the jet are less than the design allowable, this condition is acceptable.

In summary, plant safety would not be jeopardized for the hypothetical reactor water cleanup break. No additional pipe whip restraints are required in this area.

Conclusions

Using very conservative assumptions and criteria, no postulated failure of the RWCU lines can cause additional damage that could impair the ability to safely shut down the

reactor or that could increase the offsite radiation effects beyond the limits of 10CFR100.

3C.2.9 RPV Vent Line/RCIC Head Spray

3C.2.9.1 RCIC Head Spray

The locations of postulated pipe breaks are shown on Figure 3.6A-23, ~~and are tabulated in Table 3.6A-3~~7.

The RCIC head spray (2ICS-006-67-1) is a 6-in line which originates in the secondary containment from the ICS pumps. The pipe runs through the primary containment at el 292 ft 0 in through penetration Z-22. Once inside the primary containment it rises to el 330 ft 7/8 in after penetrating through the refueling bulkhead. From that elevation the pipe continues as 2ICS-006-33-1. This pipe continues to rise and has valve 2ICS*AOV157 mounted on it at el 342 ft 10 5/8 in and finally leads into the RPV head. The section of the pipe attaching the RPV head to normally closed valve 2ICS*AOV157 is considered a high-energy line.

The ICS piping, if allowed to whip, can impact targets such as the drywell head and line 2MSS-002-106-1.

To preclude any likelihood of loss of an essential system or structural integrity of the drywell head, structural steel framing is erected as the barrier between the ruptured pipe and the drywell head.

<u>Conclusions</u>

Using very conservative assumptions and criteria, no postulated failure of the ICS line can cause additional damage that could impair the ability to safely shut down the

reactor or that could increase the offsite radiation beyond the limits of 10CFR100.

3C.2.9.2 RPV Steam Vent Line

The locations of postulated pipe breaks are shown on Figure 3.6A-15 and Table 3.6A-30. The results of the associated stress calculations are summarized in Table 3.6A-10.

The main steam 2-in vent line branches off the main steam line (2MSS-026-43-1) at el 314 ft 10 in. It then loops its way up to el 343 ft 2 in and drops vertically into the RPV head.

The MSS piping if allowed to whip could impact targets such as drywell head, vessel dome, refueling bulkhead, star truss, primary containment liner and rupture restraint 2RHS*PRR014. Since the piping is a 2-in diameter line, the impacting forces of the piping system are relatively small to cause detrimental results on the targets mentioned above. The pipe whip due to a break in this line will not cause any loss of structural integrity to the targets mentioned above.

Conclusions

12 Using very conservative assumptions and criteria, no postulated failure of the MSS vent line can cause damage that could impair the ability to safely shut down the reactor or that could increase the offsite radiation effects beyond the limits of 10CFR100.

3C.2.10 Main Steam Safety Relief Valve Piping (SVV)

The location of pipe breaks are shown on Figures 3.6A-16 and 3.6A-17. Tables 3.6A-10 and 3.6A-31 give results of pipe stress calculations and break protection requirements.

The main steam safety relief valve piping for each of the 18 SRVs consists of an 8-in connection off the main steam line to the SRV and a 10-in discharge line from the SRV to the suppression pool. The high-energy portion of the system is the 1-ft length for the connection from main steam pipe to the normally closed SRV. Two breaks are postulated for each SRV; they are terminal point, circumferential breaks for the 8-in connection.

Targets from pipe whip include the primary containment wall and 12-in RHS lines. Impact analysis shows failure of SVV piping cannot cause additional damage that could impair the

ability to safely shut down the reactor or that could increase the offsite radiation beyond the limits of 10CFR100.

3C.2.11 3-Inch and Smaller High-Energy Piping

Included in 3-in and smaller high-energy piping are the main steam drains, standby liquid control system, and the CRD.

3C.2.11.1 Control Rod Drive System (CRD)

The locations of postulated pipe breaks are shown on Figure 3.6A-43 through 49. ~~Tables 3.6A-27 and 3.6A-47 give results of pipe stress calculations and break protecti~~on r~~equirement~~s. | 26

Inside the primary containment this system has no postulated breaks from the penetration to the reactor interface, since all the lines are less than or equal to 1-in diameter pipe size. | 26

3C.2.11.2 Standby Liquid Control System (SLC)

The locations of postulated pipe breaks are shown on Figure 3.6A-42. ~~Tables 3.6A-26 and 3.6A-46 give results of pipe stress calculations and break protection requirements~~.

The SLC ~~1-1/2~~ 2-in line branches off the high-pressure core spray line in the primary containment at el 307 ft 11 3/16 in.

The high-energy portion of this system runs straight to normally closed check valve 2SLS*V10.

There is no pipe whip associated with the 2 circumferential breaks in this straight section of ~~1-1/2~~-in piping; therefore, there are no targets or protection requirements.

3C.2.11.3 Main Steam Drain Line

The locations of postulated pipe breaks and pipe whip restraints are shown on Figure 3.6A-18, an~~d Tables 3.6A-32 and 3.6A-33~~.

Th~~e results of the associated stress calculations are~~ su~~mmarized in Table 3.6A-12~~.

The main steam drain piping inside containment tap off the four 26-in main steam lines. The 10-in ICS line is normally pressurized and ASME Code Class 1.

250 ft

The four 2-in drain lines branching out from the main steam lines connect into the 6-in horizontal line at el ~~249 ft 9 7/16 in~~. This 6-in line (2MSS-006-150-1) runs through the primary containment at el 246 ft 3 in passing through its penetration at Z-2.

The main steam drain pipings, if allowed to whip, can impact targets such as ~~the inboard isolation valves (2MSS*HFV6A, B, C, and D),~~ the platforms at el 247 ft 6 in, and ~~some main steam systems like 2MSS*PRR006, PRR010, PRR026, and PRR037~~.
el. 261 ft-0 in, and various structural steel.

12 To preclude any likelihood of loss of an essential system or structural integrity ~~of the containment,~~ restraints 2MSS*PRR201, ~~PRR202~~ ~~and~~ ~~PRR203~~ are installed for the system.
PRR204A, PRR204B and PRR203

Conclusions

Using very conservative assumptions and criteria, no postulated failure of the main steam drain lines can cause additional damage that could impair the ability to safely shut down the reactor or that could increase the offsite radiation effects beyond the limits of 10CFR100.

[illegible]

3C.3 EFFECTS OF JET IMPINGEMENT

3C.3.1 Jet Impingement Analysis

3C.3.1.1 Introduction

To ensure the integrity of structures, systems, and components whose failure could impact safety and which are potential targets of jet impingement, it is necessary to calculate the jet intensity and the impingement loadings on targets. The analytical tools required for making these estimates are provided in this section. Jet impingement loadings are determined as follows:

1. The jet force at the exit plane of a pipe break is calculated as discussed in Section 3.6.2.2.2A. This jet force is dependent on the fluid condition in the system, which varies with time. For jet impingement analysis, only the peak force is used unless a complete jet time history is required to reduce conservatism. A rise time of 1 millisecond is used.

2. The jet expands as it travels along its path. The jet shape is assumed to be conical at a 10-deg half-angle expansion for subcooled water. Moody's 12
asymptotic expansion model[1] is adopted for saturated water and saturated steam (Figure 3C.3-1).

3. The impinging jet proceeds along a straight path that is normal to the break area.

   a. Circumferential breaks result in pipe severance, the break area is circular in shape, and the centerline of the jet is coincident with the pipe centerline (Figure 3C.3-1).

   b. Longitudinal breaks result in an axial split without pipe severance. The break area is circular in shape and equal to the effective cross-sectional flow area of the pipe at the break location. The jet centerline is normal to the opening and the pipe centerline (Figure 3C.3-1).

4. If the ruptured pipe is not restrained, the path of jet impingement is defined by the trajectory of the whipping pipe.

RCIC Head Spray - Main Steam Safety Relief Valve Piping (Between Main Steam Line and First Relief Valve) 3C.2.10

3-In and Smaller High Energy Piping 3C.2.11

The above systems have been described and evaluated for the effects of pipe breaks in the corresponding 3C.2 sections. In many cases, pipe whip restraints have been designed to minimize the effects of pipe whip and resulting jet impingement.

This section evaluates the effects of jet impingement considering single active failure, loss of offsite power (if loss of normal onsite power is a direct consequence of the event), and other appropriate assumptions, criteria, etc., as explained in Section 3.6.1.1A.

26 ~~Targets affected by jet impingement loads are listed in Tables 3.6A-53 through 58, 3.6A-70 and 71 for each postulated break location.~~ For each target, a section is referenced to explain the protective measure. In cases where more than one target can fail, the protective measure reference will address the complete event.

In certain cases, protective measures can be grouped to apply to all events. The following deal with protective measures common throughout this section.

3C.3.2.1 Jet Impingement Targets and Protection Measures

Targets Designed to Maintain Containment Integrity

The primary containment, drywell floor, and drywell head have been designed for all applicable jet loads, including the effects of temperature. These structures are required to maintain containment integrity following a LOCA inside containment.

Targets Designed to Prevent Generation of Missiles

Items located in the primary containment, although not directly essential for safe shutdown and containment integrity, have been designed for jet loads since their failure could lead to unacceptable consequences. These targets are as follows:

Structural Steel - All structural steel identified as targets has been designed for jet loads since it supports piping and equipment essential for safe shutdown.

TABLE 6.2-32

SUBCOMPARTMENT NODAL DESCRIPTION

24-Inch Recirculation Suction Line Break
Drywell Head Subcompartment

Volume No.	Volume (ft³)	Initial Conditions Temp (°F)	Pressure (psia)	Humidity (%)	DBA Break Conditions % Break in Vol.	Break Line	Break Area (ft²)	Break Type	Calculated Peak Pressure Difference(1) (psid)	Design Peak Pressure Difference(1) (psid)	Design Margin(2) (%)
1	8,620	150	14.2	0	0				0.00	7.22	6.2
2	276,000	150	14.2	0	100	Recirc. Suction	2.58 26	DER	6.02 6.77	7.22	16.6 [illegible]

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(1) Peak pressure difference [(P2-P1) peak] is shown on Figure 6.2-33B.
(2) Design margin: 1-(calculated Δpeak/design Δpeak).

TABLE 6.2-45B

BLOWDOWN DATA

24-Inch Recirculation Suction Line Break
RPV-BSW Annulus

Time (sec)	Blowdown Mass Flow Rate (lbm/sec)*	Blowdown Enthalpy (Btu/lbm)	Blowdown Energy Release Rate (Btu/sec)*	Total Effective Break Area (ft^2)
0.0000	17,247	529.0	9.124×10^6	5.072 3.8043
1.5400	17,247	529.0	9.124×10^6	5.072 3.8043
1.5401	13,108	529.0	6.934×10^6	2.891
2.0000	13,108	529.0	6.934×10^6	2.891

*Due to symmetry in the nodalization, the tabulated blowdown represents one-half of the total blowdown. Of the tabulated blowdown, 85 percent is directed to Node 19 and 15 percent to Node 20 by the flow diverter.

TABLE 6.2-48

PROJECTED AREAS AND MOMENT ARMS FOR FORCE AND MOMENT CALCULATIONS
12-INCH FEEDWATER LINE BREAK
21 NODE MODEL
RPV-BSW ANNULUS

Node No.	Nodal Height (Ft)	Boundary Azimuths(°)	Span(°)	Projected X-Area (Sq Ft)	Projected Z-Area (Sq Ft)	Moment Arm (Ft)
1	16.54	180.00 - 210.00	30.00	116.29	-31.16	8.269
2	16.54	150.00 - 180.00	30.00	85.13	-85.13	8.269
3	16.54	120.00 - 150.00	30.00	31.16	-116.29	8.269
4	16.54	90.00 - 120.00	30.00	-31.16	-116.29	8.269
5	16.54	60.00 - 90.00	30.00	-85.13	-85.13	8.269
6	16.54	30.00 - 60.00	30.00	-116.29	-31.16	8.269
7	15.96	180.00 - 210.00	30.00	112.22	-30.07	24.518
8	15.96	150.00 - 180.00	30.00	82.15	-82.15	24.518
9	15.96	120.00 - 150.00	30.00	30.07	-112.22	24.518
10	15.96	90.00 - 120.00	30.00	-30.07	-112.22	24.518
11	15.96	60.00 - 90.00	30.00	-82.15	-82.15	24.518
12	15.96	30.00 - 60.00	30.00	-112.22	-30.07	24.518
13	7.06	180.00 - 210.00	30.00	49.64	-13.30	36.028
14	15.08	150.00 - 180.00	30.00	77.65	-77.65	40.041
15	15.08	120.00 - 150.00	30.00	28.42	-106.07	40.041
16	15.08	90.00 - 120.00	30.00	-28.42	-106.07	40.041
17	15.08	60.00 - 90.00	30.00	-77.65	-77.65	40.041
18	15.08	30.00 - 60.00	30.00	-106.07	-28.42	40.041
19	6.00	180.00 - 210.00	30.00	42.19	-11.30	42.558
20	2.02	180.00 - 210.00	30.00	14.24	-3.82	46.571
21	0.0	0.00 - 0.00	0.00	0.00	0.00	0.00

11

TABLE 6.2-49

PROJECTED AREAS AND MOMENT ARMS FOR FORCE AND MOMENT CALCULATIONS 12-INCH FEEDWATER LINE BREAK 37 NODE MODEL RPV-BSW ANNULUS

Node No.	Nodal Height (Ft)	Boundary Azimuths(°)	Span(°)	Projected X-Area (Sq Ft)	Projected Z-Area (Sq Ft)	Moment Arm (Ft)
1	15.82	180.00 - 210.00	30.00	111.26	-29.81	7.911
2	15.82	150.00 - 180.00	30.00	81.45	-81.45	7.911
3	16.53	120.00 - 150.00	30.00	31.15	-116.24	8.265
4	2.42	105.00 - 120.00	15.00	-1.16	-8.69	15.323
5	14.11	105.00 - 120.00	15.00	-6.76	-51.37	7.057
6	2.42	90.00 - 105.00	15.00	-3.40	- 8.20	15.323
7	14.11	90.00 - 105.00	15.00	-19.83	-47.87	7.057
8	16.53	60.00 - 90.00	30.00	-85.09	-85.09	8.265
9	16.53	30.00 - 60.00	30.00	-116.24	-31.15	8.265
10	16.13	200.00 - 210.00	10.00	39.38	-3.45	23.885
11	16.13	180.00 - 200.00	20.00	74.01	-26.94	23.885
12	16.13	160.00 - 180.00	20.00	60.33	-50.62	23.885
13	16.13	150.00 - 160.00	10.00	22.67	-32.63	23.885
14	15.96	135.00 - 150.00	15.00	22.42	-54.13	24.510
15	15.96	120.00 - 135.00	15.00	7.65	-59.08	24.510
16	24.85	90.00 - 120.00	30.00	-46.83	-174.76	28.958
17	26.02	60.00 - 90.00	30.00	-133.94	-133.94	29.542
18	15.96	45.00 - 60.00	15.00	-54.13	-22.42	24.510
19	15.96	30.00 - 45.00	15.00	-58.08	-7.65	24.510
20	6.88	200.00 - 210.00	10.00	16.79	-1.47	35.385
21	6.88	180.00 - 200.00	20.00	31.55	-11.48	35.385
22	6.88	160.00 - 180.00	20.00	25.72	-21.58	35.385
23	6.88	150.00 - 160.00	10.00	9.67	-13.81	35.385
24	8.90	135.00 - 150.00	15.00	12.50	-30.17	36.937
25	8.90	120.00 - 135.00	15.00	4.26	-32.38	36.937
26	10.06	45.00 - 60.00	15.00	-34.13	-14.14	37.521
27	10.06	30.00 - 45.00	15.00	-36.63	-4.82	37.521
28	3.05	190.00 - 210.00	20.00	14.68	-2.59	46.057
29	5.71	190.00 - 210.00	20.00	27.45	-4.84	41.677
30	3.05	180.00 - 190.00	10.00	6.78	-3.16	46.057
31	5.71	180.00 - 190.00	10.00	12.68	-5.91	41.677
32	5.03	150.00 - 180.00	30.00	25.90	-25.90	45.067
33	3.73	150.00 - 180.00	30.00	19.19	-19.19	40.688
34	6.20	120.00 - 150.00	30.00	11.68	-43.58	44.404
35	6.20	90.00 - 120.00	30.00	-11.68	-43.58	44.484
36	5.03	30.00 - 90.00	60.00	-61.27	-35.38	45.067
37	0.00	0.00 - 0.00	0.00	0.00	0.00	0.00

TABLE 6.2-59D

ALUMINUM AND ZINC INVENTORY EXPOSED TO SPRAYS

Material Type	Surface Area (ft^2)	Weight (lbm)	
Aluminum	650	~~41,500~~ 40,300	15
Galvanized steel	~~58,540~~ 58,500	~~6,968~~ 6,970	
Zinc primer	2,400	230	

TABLE 9.4-11

DESIGN DATA OF PRINCIPAL EQUIPMENT
PLANT CHILLED WATER SYSTEM

Control Building Chilled Water System

1. Liquid Chillers:

Equipment Mark No.	2HVK*CHL1A, 1B
Quantity	2 (100% capacity each)
Type	Centrifugal
Refrigerant	R-11
Capacity, each, tons	145
Entering chilled water temperature, °F	55
Leaving chilled water temperature, °F	45
Chilled water flow rate, each, gpm	340
Condenser water flow rate, each, gpm	340

2. Chilled Water Pumps:

Equipment Mark No.	2HVK*P1A, 1B
Quantity	2 (100% capacity each)
Type	Centrifugal
Capacity, each, gpm	350
Total head, each, ft	~~95~~ 85
Motor, each, hp	15

3. Expansion Tanks:

Equipment Mark No.	2HVK*TK1A, 1B
Quantity	2 (one per system)
Capacity, each, gal	~~32~~ 30
Design pressure, psig	~~150~~ 125

Ventilation Chilled Water System

1. Liquid Chillers:

Equipment Mark No.	2HVN-CHL1A, 1B, 1C
Quantity	3 (50% capacity each)
Type	Hot water absorption
Capacity, each, tons	400
Hot water flow rate, each, gpm	~~540~~ 576
Entering hot water temperature, °F	250
Leaving hot water temperature, °F	225

TABLE 9.4-11 (Cont)

	Entering chilled water temperature, °F	58
	Leaving chilled water temperature, °F	48
	Chilled water flow rate, each, gpm	960
	Condenser water flow rate, each, gpm	1,550
2.	Chilled Water Pumps:	
	Equipment Mark No.	2HVN-P1A, 1B
	Quantity	2 (100% capacity each)
	Type	Centrifugal
	Capacity, each, gpm	1,920
	Total head, each, ft	~~120~~ 150
	Motor, each, hp	~~100~~ 125
3.	Service Water Pumps:[1]	
	Equipment Mark No.	2HVN-P2A, 2B
	Quantity	2 (100% capacity each)
	Type	Centrifugal
	Capacity, each, gpm	3,100
	Total head, each, ft	90
	Motor, each, hp	125
4.	Expansion Tank:	
	Equipment Mark No.	2HVN-TK1
	Capacity, gal	400
	Design pressure, psig	150

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[1] These pumps are booster pumps, delivering service water for use as absorption chillers condenser water.

TABLE 9.4-12

DESIGN DATA FOR THE PLANT GLYCOL AND HOT WATER HEATING SYSTEMS

Turbine Building

1. Glycol Heating Pumps:		
Equipment Mark No.	2HVG-P1A, 1B	
Quantity	2	
Type	Horizontal centrifugal	
Capacity, each, gpm	470	
Total head, each, ft	~~65~~ 75	
Motor, each, hp	15	
2. Glycol Heat Exchanger:		
Equipment Mark No.	2HVG-E1	
Quantity	1	
Type	Two-pass tube; one-pass shell; fixed tubesheet at both ends	
Design pressure, psig	Shell: 350	Tube: 125
Design temperature, °F	Shell: 300	Tube: 275
Total duty, mbh	Shell: 8,812	Tube: 8,812
3. Air Separator:		
Equipment Mark No.	2HVG-ASP1	
Quantity	1	
Type	Rolairtrol	
Capacity, gpm	850	
Maximum working pressure, psig	125	
Maximum operating temperature, °F	~~185~~ 350	
4. Heating Coils:		
Equipment Mark No.	2HVT-CH1	
Quantity	4 units right hand	
Type	Aerofin Type C	
Capacity, total, cfm	80,000	
Load, total, mbh	7,412	
5. Glycol Addition Tank:		
Equipment Mark No.	2HVG-TK4	
Quantity	1	
Type	Horizontal	
6. Glycol Loop Drainage Tank:		
Equipment Mark No.	2HVG-TK7	
Quantity	1	
Type	Horizontal	
Capacity, gal	1,100	

TABLE 9.4-12 (Cont)

7. Expansion Tank:		
Equipment Mark No.	2HVG-TK1	
Quantity	1	
Type	Vertical	
Capacity, gal	184	
8. Hot Water Heating Pumps:		
Equipment Mark No.	2HVH-P1A, 1B	
Quantity	2	
Type	Horizontal centrifugal	
Capacity, each, gpm	1,086	
Total head, each, ft	~~150~~ 170	
Motor, each, hp	75	
9. Building Heating Intermediate Heat Exchangers:		
Equipment Mark No.	2HVH-E1A, 1B	
Quantity	2	
Type	CEU - Horizontal	
Design pressure, psig	Shell: 270	Tube: 350
Design temperature, °F	Shell: 418	Tube: 400
Total duty, mbh	Shell: 17,184	Tube: 17,184
10. Building Heating Auxiliary Heat Exchangers:		
Equipment Mark No.	2HVH-E3A, 3B	
Quantity	2	
Type	CEU - Horizontal	
Design pressure, psig	Shell: ~~210~~ 175	Tube: 350
Design temperature, °F	Shell: ~~410~~ 377	Tube: 400
Total duty, mbh	Shell: 17,184	Tube: 17,184
11. Air Separator:		
Equipment Mark No.	2HVH-ASP1	
Quantity	1	
Type	Rolairtrol	
Capacity, gpm	1,900	
Maximum working pressure, psig	350	
Maximum operating temperature, °F	650	

TABLE 9.4-12 (Cont)

12. Makeup Water Pumps:	
Equipment Mark No.	2HVH-P2A, 2B
Quantity	2
Type	Metering
Capacity, each, gpm	40
Pressure, each, psig	300
Motor, each, hp	1/2
13. Expansion Tank:	
Equipment Mark No.	2HVH-TK1
Quantity	1
Type	Horizontal
Capacity, gal	430
<u>Reactor Building (Standby Gas Treatment Building)</u>	
1. Glycol Heating Pumps:	
Equipment Mark No.	2HVG-P2A, 2B
Quantity	2
Type	Horizontal centrifugal
Capacity, each, gpm	822
Total head, each, ft	~~140~~ 115
Motor, each, hp	~~50~~ 40
2. Glycol Heat Exchanger:	
Equipment Mark No.	2HVG-E2
Quantity	1
Type	Two-pass tube; one-pass shell; fixed tubesheet at both ends
Design pressure, psig	Shell: 350 Tube: 125
Design temperature, °F	Shell: 375 Tube: 275
Total duty, mbh	Shell: 15,422 Tube: 15,422
3. Air Separator:	
Equipment Mark No.	2HVG-ASP2
Quantity	1
Type	Rolairtrol
Capacity, gpm	850
Maximum working pressure, psig	125
Maximum operating temperature, °F	~~185~~ 350

TABLE 9.4-12 (Cont)

4. Heating Coils:	
Equipment Mark No.	2HVR-CH1
Quantity	4 units left hand; 4 units right hand
Type	Aerofin Type C
Capacity, total, cfm	140,000
Load, total, mbh	12,968
5. Glycol Addition Tank:	
Equipment Mark No.	2HVG-TK5
Quantity	1
Type	Horizontal
6. Glycol Loop Drainage Tank:	
Equipment Mark No.	2HVG-TK8
Quantity	1
Type	Horizontal
Capacity, gal	1,300
7. Expansion Tank:	
Equipment Mark No.	2HVG-TK2
Quantity	1
Type	Vertical
Capacity, gal	184
<u>Radwaste Building</u>	
1. Glycol Heating Pumps:	
Equipment Mark No.	2HVG-P3A, 3B
Quantity	2
Type	Horizontal centrifugal
Capacity, each, gpm	235
Total head, each, ft	~~80~~ 90
Motor, each, hp	7 1/2
2. Glycol Heat Exchanger:	
Equipment Mark No.	2HVG-E3
Quantity	1
Type	Two-pass tube; one-pass shell; fixed tubesheet at both ends
Design pressure, psig	Shell: 350 Tube: 125
Design temperature, °F	Shell: 375 Tube: 275
Total duty, mbh	Shell: 4,410 Tube: 4,410

TABLE 9.4-12 (Cont)

3. Air Separator:	
Equipment Mark No.	2HVG-ASP3
Quantity	1
Type	Rolairtrol
Capacity, gpm	300
Maximum working pressure, psig	125
Maximum operating pressure, °F	~~185~~ 350
4. Heating Coils:	
Equipment Mark No.	2HVW-CH1
Quantity	2 units right hand
Type	Aerofin Type C
Capacity, total, cfm	~~40,000~~ 47,800
Load, total, mbh	~~3,706~~ 4,410
5. Glycol Addition Tank:	
Equipment Mark No.	2HVG-TK6
Quantity	1
Type	Horizontal
6. Glycol Loop Drainage Tank:	
Equipment Mark No.	2HVG-TK9
Quantity	1
Type	Horizontal
Capacity, gal	500
7. Expansion Tank:	
Equipment Mark No.	2HVG-TK3
Quantity	1
Type	Vertical
Capacity, gal	50

monitors. Additional details of ventilating and smoke removal systems are included in Section 9.4.

9.5.1.3 Safety Evaluation (Fire Hazards Analysis)

The fire protection program allows the plant to maintain the ability to perform safe shutdown functions and minimize radioactive releases to the environment in the event of a fire. The safe shutdown analysis is described in Appendix 9B, and the radioactive release analysis is contained in the fire hazards analysis described in Appendix 9A. 23

Potential fire hazards throughout the plant and the effect of postulated fires on safety-related plant areas are analyzed. The evaluation of analysis includes the consideration of fire loading, maximum fire intensity, and automatic and manual firefighting activities.

9.5.1.4 Inspection and Testing Requirements

Periodic operational checks, inspections, and servicing required to maintain fire protection systems, including the alarm detection system, conforms to the requirement of the NFPA standards.

9.5.1.5 Personnel Qualification and Testing

Qualification and testing for NMPC personnel are described in Chapter 13.

9.5.2 Communication Systems

The plant communication systems are designed to provide reliable inplant communication, communication with Nine Mile Point Unit 1, and plant-to-offsite communication during normal conditions and under maximum potential noise levels. The communication systems also provide for emergency alarms and evacuation signals. The systems for inplant and Unit 2 to Unit 1 communications consist of:

1. A dial telephone system with code call paging.

2. A portable radio communication system.

3. A page party/public address communication (PP/PA) system with emergency evacuation signals and other emergency alarms.

4. A maintenance and calibration communication (M/CC) system.

5. A sound-powered communication (SPC) system.

Plant-to-offsite communication is provided through a commercial telephone system, interconnected with the plant telephone system, and portable radios. Plant-to-offsite communication capabilities during emergency conditions are described in the Site Emergency Plan.

9.5.2.1 Design Bases

9.5.2.1.1 Dial Telephone System

A dial telephone system with an integrated code call system
23 is provided for communication between selected office areas and selected locations inside and outside the station. The dial telephone system is connected to the NMPC telephone tie system for offsite communication including communication with the local law enforcement authority, local fire department, and others. The telephone system main equipment is powered from the plant normal uninterruptible power supply (UPS) system.

23 9.5.2.1.2 Radio Communication Systems

A hand-held portable radio communication system is provided for communication between station personnel within the
23 plant. A plant-to-offsite radio console is provided for communication between station personnel and NMPC personnel located outside the station in case the dial telephone system between the station and the points outside the station becomes inoperable.

23 9.5.2.1.3 Page Party/Public Address System

A PP/PA system with five party channels and one page channel is provided for communication between various station buildings and locations. The system has the following characteristics:

1. Satisfactory voice communications are possible even under extremely noisy conditions.

2. Simultaneous conversations may take place on the page and all party channels without any interference.

3. Plant emergency alarm, evacuation signals, and code call are provided by this system.

selected area telephones provide direct dialing of the outgoing calls. A direct dialing telephone link is also provided solely for load dispatching purposes. Special emergency communication systems are described in Appendix 13B.

A code call system is provided with the telephone system for calling supervisory and management level personnel when they are away from their normal locations. The code call system utilizes a tone generator, telephone sets, and the page channel of the PP/PA system. Each of the supervisory and management level personnel is assigned a two-digit code number. To call a particular person to the telephone, the person originating the call must go to a telephone set and dial an access number followed by a two-digit code number for the person being called. A series of tones closely approximating the sound of a single stroke gong is generated and sounded over the station PP/PA loudspeakers located throughout the station. The code call signal continues to sound until the originator terminates the call by hanging up his telephone or the person being called goes to the nearest telephone and dials a call-back number. When this is done, the code call signal will stop and he will be in direct communication with the originator. The code call signal can be originated from any telephone set located in the plant and the administration building.

The telephone system main equipment is powered from the 120-V ac plant normal UPS system.

9.5.2.2.2 Radio Communication Systems

The portable radio communication system utilizes hand-held portable radios operated on VHF band frequencies. The radios are powered by rechargeable batteries. The system utilizes a leaky wire antenna system throughout the plant with repeaters which are fed from an uninterruptible power supply. The radios are used for maintenance, operating communications, and communication among the plant security force.

A plant-to-offsite radio communication system is provided by a console located in the control room. This system provides communication to Oswego Fire, Offsite Administration, Offside Radiation Teams, and Power Control. Under emergency conditions or in case the dial telephone system between the station and the points outside the station becomes inoperable, this serves as the alternate means of communication.

physically separated to provide isolation of one paging line from the other. This improves the operating reliability of the paging aspects of the plant communication system making it less susceptible to common mode failure hazards.

The page lines of the communication system are electrically supervised for continuity. Each of the two paging systems has a supervisory signal generator, signal detecting and relaying equipment, and an annunciator for monitoring and alarming the page system lines. The main purpose of this monitoring system is to check the integrity of the page lines for fire alarm purposes. Selected branches of each of the page lines are independently alarmed at the control and remote shutdown consoles so that an operator can readily identify the affected paging line. The control room and remote shutdown room consoles each have a switch to merge/isolate the independent page lines via tie relay cabinets that are located at three different elevations in the turbine building.

The code call tone generator provided with the telephone system also provides the supervisory signal to backup either multitone generator. This code call tone is transmitted over the Unit 2 page line but may be blocked by a voice-operated relay (VOX) when a page is in progress.

Handset stations are located so that they will meet the minimum requirements for manual fire alarm pull stations.

The foregoing features have been incorporated in the plant communication system to meet the intent of NFPA72D code requirements for fire protection signaling systems.

The PP/PA system is powered from two 120-V ac normal UPS systems, 2VBB-UPS1C and 2VBB-UPS1D. The PP/PA relay and control cabinets with tone generators receive power supply from both UPS systems through a transfer switch. The handsets and the speaker amplifiers are fed from either UPS system.

The PP/PA system is capable of operating merged or isolated with the existing Nine Mile Point Unit 1 communication system. The merge/isolation operation is controlled from the communication console in Unit 1 or Unit 2 by selector switches with indicating lamps. The PP/PA system has muting facilities that operate directly or indirectly from the hook switches and silence one or more nearby loudspeakers to prevent acoustical feedback when the associated handset is lifted.

9.5.2.3 Inspection and Testing Requirements

The design of the communication systems permits routine surveillance and testing without disrupting normal communication facilities. The page line of the PP/PA system is electrically supervised permitting immediate corrective action to be taken if the page line becomes faulted. The evacuation warning signal, fire alarms, and other emergency alarms are tested in accordance with the Emergency Plan. Functional tests are performed under conditions that approximate the maximum plant noise levels being generated 24 during the various operating conditions, including fire, to demonstrate the system capabilities. Power sources are monitored by indicating instruments. Battery cells are checked periodically for voltage and specific gravity.

9.5.2.4 System Evaluation

The PP/PA system with evacuation and fire alarms, the M/CC 24 system, the dial telephone system with code call paging, and the portable radio communication systems provide adequate communication facilities for communication during normal conditions under maximum potential noise levels. The paging system with the multitone generator provides emergency evacuation signals and other alarms. This system is electrically supervised so that any fault developed in the system could be detected and rectified immediately. The SPC system provides voice communication in case of total loss of electric power to the PP/PA system and M/CC system. The dial telephone system, interconnected with the NMPC telephone tie system, provides plant-to-offsite communication including communication with local law enforcement authority, fire department, and others, during normal and emergency conditions. The system works even under extremely noisy conditions. The telephones in extremely noisy areas are provided with acoustical booths. Under emergency conditions, or in case the dial telephone system between the station and the points outside the station becomes 24 inoperable, the plant-to-offsite radio communication system serves as the alternate means of communication.

The dial telephone system and the PP/PA system are powered from two normal UPS systems, 2VBB-UPS1C and 2VBB-UPS1D. The UPS systems are fed from three sources: normal ac, bypass ac, and dc. System 2VBB-UPS1C is fed from 600-V load centers 2NJS-US3 (normal), 2NJS-US5 (bypass), and 125-V dc battery 2BYS-BAT1A (backup) via dc switchgear 2BYS-SWG001A. System 2VBB-UPS1D is fed from 600-V load centers 2NJS-US1 (normal), 2NJS-US6 (bypass), and 125-V dc battery 2BYS-BAT1B

TABLE 13.1-4

NUMBER AND QUALIFICATIONS OF SITE PERSONNEL

Title	No. Site Personnel	Section of ANSI/ANS 3.1-1978 Containing Qualifications	
General Superintendent	1	4.2.1	
Nuclear Generation Administrative Assistant for Administration and Planning	1	NA	
Station Superintendent	2	4.2.1	
Site Maintenance Superintendent Nuclear	1	4.2.3	
Technical Superintendent Nuclear	1	4.2.4	
Superintendent Training Nuclear	1	4.2.2	
Superintendent Chemistry and Radiation Management	1	4.4.3 or 4.4.4	
Superintendent Operations Nuclear	2	4.2.2	23
Assistant Superintendent Operations Nuclear	4	4.2.2	23
Supervisor Radwaste Operations	2	4.3.2	
Assistant Supervisor Radwaste Operations	2	4.3.2	
Superintendent Training Nuclear	1	4.2.2	
Supervisors Training Nuclear	~~8~~ 7	4.3.1 or 4.3.2	
Assistant Supervisors Training Nuclear	~~14~~ 40	4.3.1 or 4.3.2	

Generation Specalists Nuclear (Series)

TABLE 13.1-4 (Cont)

Title	No. Site Personnel	Section of ANSI/ANS 3.1-1978 Containing Qualifications	
			16
Assistant Superintendent Training Nuclear	~~1~~ 3	4.2.2	
Supervisors Training Nuclear	6	4.3.2	16
Assistant Supervisors Training Nuclear	5	4.3.2	
Training Specialist Nuclear	11	NA	
Emergency Coordinator	1	NA	
Supervisor Chemistry and Radiation Protection	1	4.4.3* or 4.4.4	
Unit Radiation Protection Supervisor	2	4.4.4	
Unit Chemistry Supervisor	2	4.4.3	
Supervisor Instrument Support	1	4.7.2	
Supervisor Radiological Support	1	4.6.1*	
Dosimetry Coordinator	1	4.7.2	
ALARA Coordinator	1	4.7.2	
Radiation Protection Technicians	As needed	4.5.2	
Chemistry and Radio-chemistry Technicians	As needed	4.5.2	
Environmental Protection Coordinator	1	4.7.2	

TABLE 14.2-41

PROCESS SAMPLING SYSTEM

System 17

Test Objectives

1. To demonstrate the operation of the turbine, reactor, and radwaste buildings sampling systems and components.

2. To ensure the systems are properly designed and constructed.

Safety Precaution

Follow NMPC safety rules and proper procedures during testing.

Prerequisites

1. All applicable preliminary tests are completed and the system turned over to NMPC.

2. All applicable power sources to supply electric power to motors, control circuits, and instrumentation are available. | 24

3. Valve lineups are completed.

4. The turbine and reactor building closed loop cooling systems are available to support testing.

Test Procedure

1. The test procedure verifies proper system instrumentation response by simulated signals or actual parameter variation.

2. All applicable controls, interlocks, and valves are verified for proper operation to ensure performance within system specifications.

Note: Temperature Reduction Equipment will be tested during the startup testing phase when hot samples are available

3. All applicable alarms and annunciators are verified for proper operation in conjunction with the tests performed.

TABLE 14.2-41 (Cont)

Acceptance Criteria

22 1. All air-operated sample system isolation valves operate from their respective sample panels.

22 2. All applicable system instrumentation, interlocks, and trips function as designed in accordance with Section 9.3.2.

22 3. The system functions as described in Section 9.3.2.

2. With the exception of the equipment used to reduce hot sample temperature, All applicable System instrumentation, interlocks, and trips function as designed in accordance with Section 9.3.2

3. The ability to take a grab sample from each System, will be verified.