REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR: 8609290033 DOC. DATE: 86/09/23 NOTARIZED: YES DOCKET #
FACIL: 50-410 Nine Mile Point Nuclear Station, Unit 2, Niagara Moha 05000410

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

SUBJECT: Forwards marked-up FSAR pages, indicating changes & justification for changes. Affect on SER noted.

DISTRIBUTION CODE: B001D COPIES RECEIVED: LTR 1 ENCL 10 SIZE: 46
TITLE: Licensing Submittal: PSAR/FSAR Amdts & Related Correspondence

NOTES:

	RECIPIENT ID CODE/NAME		COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME		COPIES LTTR	COPIES ENCL
	BWR EB		1	~~1~~	BWR EICSB		2	~~2~~ *
	BWR FOB		1	~~1~~	BWR PD3 LA		1	~~1~~
	BWR PD3 PD		1	1 *	HAUGHEY, M	01	2	2 *
	BWR PSB		1	~~1~~	BWR RSB		1	~~1~~
INTERNAL:	ACRS	41	6	~~6~~ *	ADM/LFMB		1	0
	ELD/HDS3		1	0	IE FILE		1	1 *
	IE/DEPER/EPB	36	1	~~1~~	IE/DQAVT/QAB	21	1	~~1~~
	NRR BWR ADTS		1	~~1~~	NRR PWR-B ADTS		1	~~1~~
	NRR ROE, M. L		1	~~1~~	NRR/DHFT/MTB		1	~~1~~
	~~REG FILE~~	04	1	1 *	RGN1		3	~~3~~ *
	RM/DDAMI/MIB		1	0				
EXTERNAL:	BNL (AMDTS ONLY)		1	~~1~~	DMB/DSS (AMDTS)		1	~~1~~
	LPDR	03	1	1 *	NRC PDR	02	1	1 *
	NSIC	05	1	1 *	PNL GRUEL, R		1	~~1~~

LIMITED DISTRIBUTION per PM
M. HAUGHEY

*- w/ Encl

TOTAL NUMBER OF COPIES REQUIRED: LTTR 36 ENCL 31

NIAGARA MOHAWK

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

September 23, 1986
(NMP2L 0884)

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 our August 22, 1986 letter, NMP2L-0851, we have identified a number of additional Final Safety Analysis Report changes. Generally, these changes are minor and do not affect the Safety Evaluation Report. However, where a change may affect the Safety Evaluation Report, this has been noted.

Attached are marked-up Final Safety Analysis Report pages. To aid the Nuclear Regulatory Commission in their review of this material, Niagara Mohawk also has included justification for these changes. These changes will be included in a subsequent Final Safety Analysis Report update.

Very truly yours,

C V Mangan

C. V. Mangan
Senior Vice President

WHB:meg
2066G
Attachment

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

Limited Distribution per PM M. Haughey

B001
1/10

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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

AFFIDAVIT

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

CVMangan

Subscribed and sworn to before me, a Notary Public in and for the State of New York and County of Onondaga, this 23rd day of September, 1986.

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

My Commission expires:
JANIS M. MACRO
Notary Public in the State of New York
Qualified in Onondaga County No. 4784558
My Commission Expires March 30, 19 87

SUMMARY OF FSAR CHANGES

Page	Change Description	Justification	Change Code	SER Impact	Tech. Spec. Impact
Table 3.2-1 Page 26a of 26	Remove Items 2, 3 and 4 from the examples of footnote 24, Quality Group Classification of the recirculation system seal cooling.	This change reflects the declassification of the recirculation pump seal cooling and is consistent with previous changes made to FSAR Sections 9.2.2 and 1.10 (Item II.K.3.25). The service water interconnection was removed. Reactor Building closed loop cooling is used to cool the seals. The line was Quality Group C due to the connection of service water.	E	No	No
Table 3.4-7	Change the expected performance of the waterstops to a higher temperature (100 to 325F).	The expected performance of the waterstop material is up to 325F.	E	No	No
Pages 3.9A-24a and 24b	Update ASME Section Code references for certain code cases. Also correct code case numbering problem.	This change is made to be consistent with design data.	SN	No	No
Table 3.9B-1	Change the number of startups (100°F/heat rate) from 117 to 120, and change cycles to events.	This change is made to make Table 3.9B-1 consistent with the design documents, and Table 5.7.1-1 of the Technical Specification.	E	No	No
Table 3.9b-2b, 2m (Sht. 2), 2n (Sht. 1), 2v (Sht. 2)	Correct typographical errors and editorial errors on these tables.	These changes are made to clarify the tables.	E	No	No

SUMMARY OF FSAR CHANGES

Page	Change Description	Justification	Change Code	SER Impact	Tech. Spec. Impact
Page 6.2-46a	Change the calculated minimum drywell spray flow rate from 6880 to 6450.	This change is made to reflect an as-built calculation of the spray flow, and reflects the final plant design. See comment on Table 6.2-52 below.	SS	SSER 1 Pg. 6-2	No
Table 6.2-30	Change the table to reflect a new footnote.	The footnote was added to minimize the changes to the FSAR due to as-built conditions. The actual head loss coefficient and vent area values changed, but the values used in the analysis are conservative.	E	No	No
Table 6.2-52	Change the footnote from "3" to "9.7" percent.	The actual as-built spray flow values differ from the values used in the analysis (as shown in the table) due to the installation of the spray nozzles in the plant. Certain spray nozzles were blocked due to physical obstructions (such as pipe supports) in the field, resulting in lower spray flow rate in the drywell. However the lower spray flow rate does not impact the results of the safety analysis of the plant.	SS	SSER 1 Pg. 6-2	No
Figure 6.2-70	Change the title of the Figure from penetration "Z-9-" to "Z-90."	This change is made to make the Figure title consistent with the information on the Figure.	E	No	No

SUMMARY OF FSAR CHANGES

Page	Change Description	Justification	Change Code	SER Impact	Tech. Spec. Impact
Page 6.3-3	Change IEEE 308 1974 to ANSI Std. N195-1976.	The IEEE 308 standard does not address fuel oil storage and ANSI-N195 does as discussed in FSAR Section 9.5.4. This change is needed for FSAR consistency.	E	No	No
Page 6.5-4	Remove the commitment to ERDA 76-21	The actual field installations have access panels on the SGTS housing, rather than doors as required by ERDA 76-21. However, this is not required by Regulatory Guide 1.52.	SN	No	No
Table 6A.4-4	Change the Time Interval column for Test run numbers 25 and 26.	The actual 4TCO test data time intervals are slightly different from that listed. These minor changes are needed for consistency with the source references.	E	No	No
Page 7.3-18	Change the referenced section from 6.5.2.3.1 to 6.2.2.3.1.	This change corrects the references.	E	No	No
Page 8.3-18a and 8.3-18b	Change service water load sequencing to match Tables 8.3-1, 8.3-2 and 8.3-5.	The actual load sequencing was updated previously in our letter dated August 22, 1986. This change reflects those values previously provided.	E	No	No

SUMMARY OF FSAR CHANGES

Page	Change Description	Justification	Change Code	SER Impact	Tech. Spec. Impact
Pages 9A.3-56 and 9A.3-58	Add a clarification to the text regarding the 600V switch gear room and the RCIC pump room.	These changes are provided to clarify the FSAR text and make it consistent with Figure 9A.3-6 for the 600V switchgear room. The change to the RCIC pump room is to clearly identify the relief panel as a non-rated device in a three-hour barrier.	SN	No	No
Page 10.4-4	Remove the description of the condenser tube-side operational leak test and replace it with the correct description.	The actual operational leak test is described in the change pages. The description now matches plant testing. This has no safety impact.	N	No	No
Page 11.4-3a	Change the words "Auxiliary Condensate" to condensate makeup and drawoff.	The as-built design uses condensate makeup and draw-off system to provide flushing water. This has no safety impact.	N	No	No
Page 13.1-6	Change ANSI N18.1-1978 to ANSI/ANS 3.1-1978.	This corrects the FSAR reference. The change makes the commitment consistent with other FSAR sections and the Technical Specifications.	E	No	No
Table 15.7-12	Change the Fuel Handling Radiological Effects Results.	This change reflects revised calculations of the radiological impact of a fuel handling accident. Airborne activity in the reactor building is now assumed to be released instantaneously, rather than over a 30-second period.	SN	No	No

SUMMARY OF FSAR CHANGES

Page	Change Description	Justification	Change Code	SER Impact	Tech. Spec. Impact
Table 15.7-12 (Continued)		The results are still within the allowable value specified in the regulations. The staff has independently checked the analysis in the Safety Evaluation Report and confirmed the acceptability of the design.			
FSAR Q/R Heavy Loads Table 3-3, 4-1 and Figure 5-2	The changes to Table 3-3 reflect final design lift weights and provides additional loads which are added to the table. The deletion to Table 4-1 reflects the shifting of loads.	A load drop evaluation has not been performed for the loads previously listed in Table 4-1, Sheet 1 of 10 (25-ton auxiliary hoist). The subject loads may be handled by the No. 3 125-ton main hoist.	SS	SER Appendix G	No
FSAR Q/R Table 421.36-1 Page 1 of 18	The lower end of the instrument range of the Intermediate Range Monitor detector (C51-N002A-H) is changed to 4.0×10^{-4} percent power. The unit of counts per seconds (cps) is added to the instrument range of the Source Range Monitor detector (C51-N001A-D). Seismic and environmental qualification of the Control Rod Position are deleted.	The change in the lower end instrument range of the Intermediate Range Monitor detector is made to be consistent with FSAR Figure 7.6-2. The unit of cps is missing and is now added for clarification purpose. As discussed in FSAR Section 7.7, the control rod position circuitry performs no safety-related function, and is not required to be seismically or environmentally qualified. This is consistent with the Category 3 classification assigned to the control rod position parameter in Regulatory Guide 1.97, Rev. 3, and is also consistent with other	E	No	No

Page	Change Description	Justification	Change Code	SER Impact	Tech. Spec. Impact
FSAR Q/R Table 421.36-1 Page 1 of 18 (Continued)		Category 3 variables listed in Table 421.36-1 which are also not seismically or environmentally qualified. This change supercedes the change previously submitted by our letter dated August 22, 1986 (NMP2L 0851). This change also corrects the changes previously submitted by our letter dated January 20, 1986 (NMP2L 0589).			

TABLE 3.2-1 (Cont)

1. Valves F001, F002, F009, F013, F014, and F017 for pump seal purge line (inside containment) to recirculation pump.

2. Valves F019, F020, F021, F022, and F059 for sample line from recirculation loops.

3. Vent valves F025, F026, F068, and F069 for remote operated valves.

4. Valve F079 for pump seal staging line.

An Example~~s~~ of the Quality Group C essential piping and valves in the recirculation system are the following: 9

1. Pump seal leak detection piping up to and including valve F086.

DELETE

2. Recirculation motor cooling water piping inside containment including valves F007, F085, and F082.

3. Recirculation motor bearing cooling water piping inside containment.

4. Pump seal cooling water piping inside containment including valves F081 and F087.

[25] Examples of Quality Group D nonessential piping and valves in the recirculation system are the following:

1. Pump seal purge piping (outside containment) to recirculation pump including valves F008, F016, and F015.

2. Recirculation pump seal staging piping including valves F084 and F088.

3. Pump seal leak detection piping beyond valve F086.

[26] This equipment conforms to ANSI Standard B31.1 and IEEE 344-71 seismic requirements. To qualify as equivalent to ASME Section III, Class 3 standards, the equipment will be pressure tested at above normal operating pressures.

TABLE 3.4-7

PERFORMANCE OF WATER STOP MATERIAL IN EXPECTED ENVIRONMENT

	Expected Environment				Expected Performance of Material			
Material	Temperature Range(1)	Chemicals	Radiation Level	Aging	Temperature Range	Chemicals	Radiation Level	Aging
Styrene-Butadiene synthetic rubber waterstops	-20°F to +325°F	Unit 2 site has average pH -8.0-8.4. No acidic environment expected within the walls below grade area.	Below $1.6x10^7$ rads	40 yr at normal operating temperature (109°F)	-35°F to +100°F +325°F	Unaffected by acidic or alkaline soils or soil bacteria.	$2x10^6$ rads before threshhold damage. $1x10^7$ rads before 25% damage. $6.0x10^7$ rads before 50% damage.	40 yr at 109°F.

(1)Temperature range varies from -26°F minimum outside at Site, 109°F normal operating inside secondary containment, to 325°F maximum accident inside secondary containment.

NF-2121 Permitted Material Specifications

1974

The Summer 1974 Addenda dated June 30, 1975, is invoked to permit the use of SA672 material.

NF-2121 Permitted Material Specifications

The Winter 1974 Addenda dated December 31, 1974, is invoked to permit the use of increased allowable stress for SA515 G65.

NF-2121 Permitted Material Specifications

The Summer 1976 Addenda dated June 30, 1976, is invoked to include the new subparagraph NF-2121(c) to permit the exclusion of certain shim stock from the requirements of Article NF-2120.

NF-2121 Permitted Material Specifications

The 1977 Edition dated July 1, 1977, is invoked to permit the use of SA36 material.

NF-2121 Permitted Material Specifications

The 1980 Edition dated July 1, 1980, is invoked to permit the use of SA564, Type 630 material.

NF-2121 Permitted Material Specifications

1981

The 1981 Edition dated July 1, 1980, is invoked to permit the use of SA-194-2H nuts.

NF-2130 Certification by Material Manufacturer

The Summer 1982 Addenda dated June 30, 1982, is invoked for material certification.

NF-2610 Documentation and Maintenance of Quality Systems Programs

The 1977 Edition dated July 1, 1977, is invoked to revise the material manufacturers and material suppliers responsibilities for materials defined as small products or materials permitted to be supplied with Certificates of Compliance.

NF-3274 Snubbers

The Summer 1976 Addenda dated June 30, 1976, is invoked for NF-3134.6 to permit the use of mechanical snubbers.

32
NF-2626 Special Stress Limits
NF-3321.1 Design Conditions
XVII-2211 Stress in Tension

The Winter 1978 Addenda dated December 31, 1978, is invoked for these paragraph sections which in effect delete the code methods for consideration of through thickness stresses in plates and elements of rolled shapes.

NF-3391.1 Allowable Stress Limits
NF-3392.1 Allowable Stress Limits

23 Winter December 31,
The Summer 1979 Addenda dated June 30, 1979, is invoked for these paragraph sections which in effect delete the code methods for consideration of through thickness stresses in plates and elements of rolled shapes.

XVII-2454 Butt and Groove Welds

The 1980 Edition dated July 1, 1980, is invoked to redefine the throat thickness of partial penetration groove welds in accordance with Table XVII-2452.1-1.

In the case that material cannot be purchased to meet the specified ASME III Code, then material that meets subsequent ASME III Code Editions/ Addenda up to and including the 1980 Edition/Summer 1982 Addenda may be substituted after a review and reconcilation of related requirements of the ASME III Code are performed and documented.

Table 3.9A-14 lists the load conditions, load combinations, and allowable stresses. Loads are applied in whatever manner is necessary to attain the worst possible stress levels for all support elements. Component standard supports are qualified either by analysis or by a combination of analysis and load rating. All other supports are qualified by analysis.

The design criteria and dynamic testing requirements for component and pipe supports listed in the following paragraphs are applicable under all plant operating conditions.

<u>Instrument Lines</u> The requirements for instrument lines are listed in Table 3.9A-15.

TABLE 3.9B-1

PLANT EVENTS

Normal, Upset, and Testing Condition	No. of Cycles
1. Bolt up[1]	123
2. Design hydrostatic test	130
3. Startup (100°F/hr heatup rate)[2]	117 120
4. Daily reduction to 75% power[1]	10,000
5. Weekly reduction 50% power[1]	2,000
6. Control rod pattern change[1]	400
7. Loss of feedwater heaters (80 cycles total)	80
8. 50% SSE event at rated operating conditions (OBE)	10/50[3]
9. Scram:	
a. Turbine generator trip, feedwater on, isolation valves stay open	40
b. Other scrams	140
c. Loss of feedwater pumps, isolation valves closed	10
d. Single safety or relief valve blowdown	8
10. Reduction to 0% power, hot standby, shutdown (100°F/hr cooldown rate)[2]	111
11. Unbolt	123

TABLE 3.9B-1 (Cont)

Emergency Condition	No. of Cycles events
12. Scram:	
a. Reactor overpressure with delayed scram, feedwater stays on, isolation valves stay open	1[4]
b. Automatic blowdown	1[4]
13. Improper start of cold recirculation loop	1[4]
14. Sudden start of pump in cold recirculation loop	1[4]
15. Hot standby, RPV drain shutoff, recirculation pumps restart	1[4]
Faulted Condition	
16. Pipe rupture and blowdown	1[4]
17. Safe shutdown earthquake at rated operating conditions	1[4]

---

(1) Applies to reactor pressure vessel only.
(2) Bulk average vessel coolant temperature change in any 1-hr period. events
(3) 50 peak OBE cycles events for NSSS piping; 10 peak OBE cycles events for other NSSS equipment and components.
(4) Annual encounter probability of the one-cycle events is $<10^{-2}$ for emergency and $<10^{-4}$ for faulted events.

TABLE 3.9B-2b

CONTROL ROD GUIDE TUBE

Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
Primary Stress Limit The allowable primary membrane stress plus bending stress is based on ASME Section III, for Type 304 stainless steel tubing and SA351 type CF8 casting (base)				
For Service Levels A & B (normal and upset) conditions: $1.5\ S_m = 1.5 \times 16{,}000$ $= 24{,}000$ psi	1. Dead weight 2. External pressure 3. Lateral flow impingement 4. OBE + SRV	Primary membrane plus bending	24,000	15,056
For Service Level C (emergency) condition: $S_{limit} = 2.25\ S_m$ $= 2.25 \times 16{,}000$ $= 36{,}000$ psi	1. Dead weight 2. External pressure 3. Lateral flow impingement 4. OBE + SRV	Primary membrane plus bending	36,000	15,056
For Service Level D (faulted) condition: $S_{limit} = 3.6\ S_m$ $= 3.6 \times 16{,}000$ $= 57{,}600$ psi	1. Dead weight 2. External pressure 3. Lateral flow impingement 4. SSE + AP/F + JR 5. Annulus pressurization	Primary membrane plus bending	57,600	27,770

|26

TABLE 3.9B-2m (Cont)

Component/Load Type	Highest Calculated Load ~~(lb)~~	Allowable Load ~~(lb)~~	Ratio Calculated/ Allowable	Loading	Identification of Equipment with Highest Loads	
Snubber Load (lb)						
~~Snubber/~~Service Level B	68,035	100,000	0.68	1. OBE 2. SRV	Snubber SB10	26
~~Snubber/~~Service Level C	13,477	66,500	0.20	1. Chugging 2. SRV	Snubber SB24	26
~~Snubber/~~Service Level D	94,978	150,000	0.63	1. SSE 2. AP	Snubber SB10	26
Flange Moment (in-lb)						
Level B	1,275,770	1,527,140	0.84	1. Weight 2. Thermal 3. OBE 4. SRV	Discharge valve (Loop A)	
Level C	701,623	1,527,140	0.46	1. Weight 2. Thermal 3. Chugging 4. SRV	Discharge valve (Loop A)	
Level D	1,501,271	1,527,140	0.98	1. Weight 2. Thermal 3. SSE 4. CO 5. SRV	Discharge valve (Loop A)	26
Acceleration (g)						
Horizontal	2.93	9.0	0.33	1. SSE 2. Chugging 3. SRV	Flow control valve (Loop B)	
Vertical	1.80	6.0	0.30	1. SSE 2. AP	Flow control valve (Loop B)	

TABLE 3.9B-2n

REACTOR REFUELING AND SERVICING EQUIPMENT

(i) Equipment Storage Racks 26

Acceptance Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
The allowable primary bending stress is based on ASME Section III for type ASTM B221 or 308 6061T6 aluminum alloy				
F_u = 38,000 psi				
F_y = 35,000 psi				
For normal condition: $S_{limit} = 0.66\ F_y$	For normal condition: 1. Normal operating loads	Bending	23,100	16,887 26
For emergency condition: $S_{limit} = 0.88\ F_y$	For emergency condition: 1. Normal operating loads 2. OBE 3. SRV discharge 4. LOCA*	Bending	30,800(1)	26,130 26
For faulted condition: $S_{limit} = 0.88\ F_y$	For faulted condition: 1. Normal operating loads 2. SSE 3. SRV discharge 4. LOCA	Bending	30,800(1)	26,415

*Used for conservatism 26

TABLE 3.9B-2v (Cont)

Criteria/Loading	Component	Limiting Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
	8. Cylinder head studs	Tensile	25,000	
	9. Stuffing box studs	Tensile	25,000	
Emergency Condition:				
1. Design pressure	1. Fluid cylinder	General membrane	21,360	4,450
2. Design temperature	2. Discharge valve stop	General membrane	21,360	13,600
3. Deadweight	3. Cylinder head extension	General membrane	21,360	13,600
4. Thermal expansion	4. Discharge valve cover	General membrane	21,360	8,150
5. Nozzle loads	5. Cylinder head	General membrane	21,360	8,150
6. Safety relief valve discharge	6. Stuffing box flange plate	General membrane	21,360	10,390
7. LOCA	7. Stuffing box gland	General membrane	42,000	11,420
Faulted Condition:				
1. Design pressure	1. Cylinder head studs	Tensile	25,000	18,820
2. Design temperature	2. Stuffing box studs	Tensile	25,000	24,750
3. Nozzle loads	3. Dowel pins[2]	Shear only[2]	23,400	19,430
4. Safety relief valve discharge	4. Studs, cylinder tie	Tensile[2]	25,000	8,685
5. LOCA	5. Pump holddown bolts	Shear	30,000	11,350
6. SSE	6. Pump holddown bolts	Tensile	37,500	17,680
	7. Power frame-foot area	Shear	15,000	1,850
	8. Power frame-foot area	Tensile	15,000	11,390
	9. Motor holddown bolts	Shear	12,000	3,470
	10. Motor holddown bolts	Tensile	15,500	5,660
	11. Motor frame-foot area	Shear	7,500	2,550
	12. Motor frame-foot area	Tensile	7,500	5,100

15,000

containment spray system is in accordance with Category I and Safety Class 2 requirements.

6.2.2.3.1.2 System Design

The containment spray system consists of two subsystems, the drywell spray and the suppression chamber spray. The drywell spray consists of two independent loops and spray headers (Figure 6.2-39). The suppression chamber spray consists of one spray header supplied from two otherwise independent loops. Since the water source for all containment sprays is the suppression pool, the system is a closed loop. The spray water is cooled by the RHR heat exchangers. The rated flows for the drywell and suppression chamber sprays are 6,880 gpm/loop and 420 gpm/loop, respectively. Containment spray is an operational mode of the RHR system (Section 5.4.7). 12

CALCULATED MINIMUM 6450

The containment spray isolation valves are electrically interlocked to allow actuation of the drywell spray only when 1) there is a LOCA signal or a system-level LPCI manual initiation signal, and 2) there is a high drywell pressure signal present. A second electrical interlock prevents actuation of either the drywell or suppression chamber spray lines until the corresponding LPCI injection valve is shut.

The containment spray system is safety related and, in case of loss of offsite power, supplied with a redundant onsite standby power source.

The system is designed to operate under the conditions indicated in Table 6.2-6.

A procedural restriction prohibits the operators, during the first 10 min following a LOCA, from closing an LPCI injection valve and interrupting core cooling (Section 6.2.2.2). Containment spray must be initiated and secured by operator action.

Distribution of spray in the air space is made as complete and uniform as practical with minimal direct impingement on wall and component surfaces. The sizes, types, number, and location of the spray nozzles are suitable for delivering the required quantity of water in the proper spray pattern and particle size. The expected spray pattern of the spray nozzles is hollow cone. Figures 6.2-40 through 6.2-42 show expected spray coverage several feet below the spray nozzles. Figure 6.2-43 shows the extent of the volume coverage by the sprays. Spray drops 27

| 11

TABLE 6.2-30

SUBCOMPARTMENT VENT PATH DESCRIPTION

6-Inch RCIC Head Spray Line Break
Drywell Head Subcompartment

Vent Path No.	From Volume Node No.	To Volume Node No.	Description of Vent Path Flow (Choked/Unchoked)	Vent Area (ft^2)	L/A (ft/ft^2)	Head Loss Coefficient Friction	Turning	Expansion	Contraction	Total
1	1	2	Unchoked	2.405	0.618	0.0017	1.245	0.9694	0.4816	2.70
2*	1	2	Unchoked	1.009	0.714	0.0072	1.085	0.9873	0.4923	2.57
3	1	2	Unchoked	2.584	0.613	0.0060	1.099	0.9676	0.4802	2.55
4*	1	2	Unchoked	1.009	0.714	0.0072	1.085	0.9873	0.4923	2.57
5	1	2	Unchoked	2.405	0.618	0.0017	1.245	0.9694	0.4816	2.70
6*	1	2	Unchoked	1.009	0.714	0.0072	1.085	0.9873	0.4923	2.57

* Actual values for head loss coefficient and vent area differ from the values shown for this junction; however, they are conservative for modeling vent path flow.

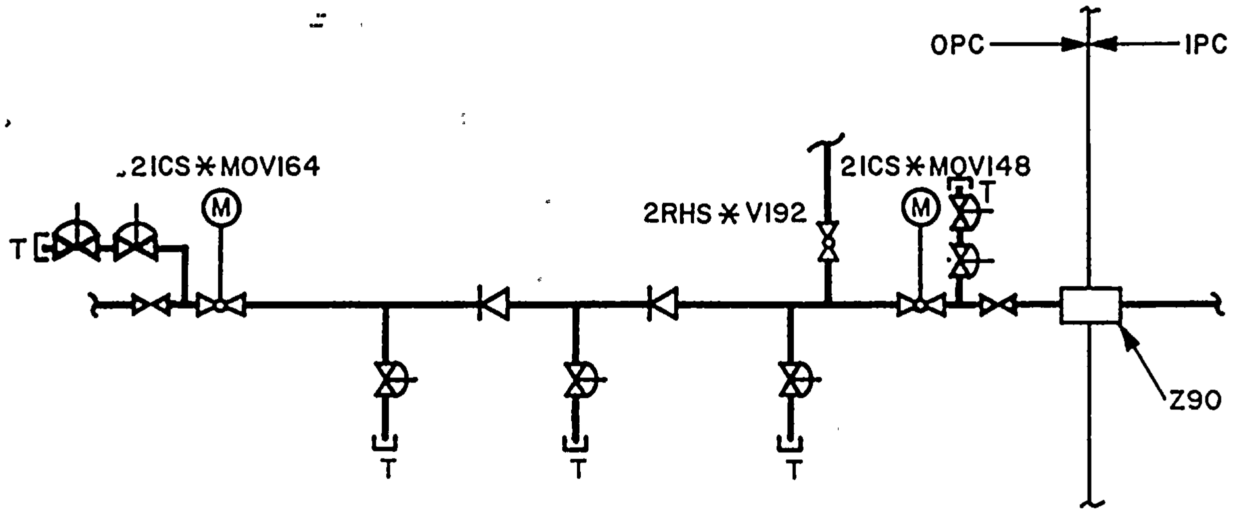
TABLE 6.2-52

ACCIDENT ANALYSIS PARAMETERS USED
FOR DBA OF CONTAINMENT HEAT REMOVAL

1.	Design basis accident (for containment sprays)	Steam line break area of 0.4 ft^2
2.	Steam bypass factor	0.05 ft^2 ($A/\sqrt{K}$ factor)
3.	Containment spray initiation	a. Manual action b. Spray operation within 30 min after break
4.	Containment parameters	Tables 6.2-1, 6.2-2, 6.2-3, and 6.2-6
5.	Spray rate, gpm	
	Drywell	7,078*
	Suppression chamber	372*
6.	Heat exchanger K factor	199.24 Btu/sec/°F
7.	No. of downcomers	121
8.	Spray drop efficiency	90%

27

*Actual values differ in the drywell by 3 9.7 percent (less) and in the suppression chamber by 13 percent (more). These changes do not impact the safety analysis of the plant. 2



Z-90 RCIC TURBINE EXHAUST VACUUM BREAKER

FIGURE 6.2-70

ISOLATION VALVE ARRANGEMENT
FOR PENETRATION Z-9 Z-90
SHEET 36 OF 43

NIAGARA MOHAWK POWER CORPORATION
NINE MILE POINT-UNIT 2
FINAL SAFETY ANALYSIS REPORT

b. 1 LPCI loop, the LPCS, and the ADS; or

c. 1 LPCI loop, the HPCS, and the ADS; or

d. The LPCS, the HPCS, and ADS.

These are the minimum ECCS combinations which result after assuming any failure (from 4 above) and assuming that the ECCS line break disables the affected system.

6. Long term (10 min after initiation signal) cooling requirements call for the removal of decay heat via the service water system. In addition to the break that initiated the loss-of-coolant event, the system must be able to sustain one failure, either active or passive, and still have at least one ECCS pump (LPCI, HPCS, or LPCS) operating with a heat exchanger and 100-percent service water flow.

7. Offsite power is the preferred source of power for the ECCS network and every reasonable precaution is made to assure its high availability. However, onsite emergency power is provided with sufficient redundancy and capacity so that all the above requirements can be met even if offsite power is not available.

8. The onsite diesel fuel reserve is in accordance with IEEE-308-1974 criteria.
ANSI STD N195-1976

9. The diesel load configuration is as follows:

a. 1 LPCI loop (with heat exchanger) and the LPCS connected to a single diesel generator (Division I).

b. 2 additional LPCI loops (1 loop with heat exchanger) connected to a single diesel generator (Division II).

c. The HPCS connected to a single diesel generator (Division III).

All these diesel generators are physically isolated from and electrically independent of each other.

10. Systems that interface with, but are not part of, the ECCS are designed and operated in such a way that failure(s) in the interfacing systems do not

26 A 4,000-cfm capacity centrifugal fan is provided downstream
of each SGTS filter train. This fan is a direct-drive type
with a single-speed motor powered from Class 1E buses. The
decay heat produced by the radioactive particles in the
26 inactive charcoal filter train is removed by passing 500 cfm
of air from the equipment room through the inactive filter
train. The air is then exhausted to the main stack by the
fan of the active filter train. A missile-protected opening
26 with a backdraft-type tornado damper located in the
equipment room allows outside air to be induced into the
room when makeup air for decay heat cooling is required.

The SGTS charcoal filter trains are located in the standby gas treatment building at el 261 ft.

Airtight ~~Access~~ Access doors are provided to give complete accessibility to all components for servicing. The doors are airtight, fitted with locking devices, and have provisions for opening inside the housing, as recommended in ERDA-76-21, Section 4.5.

6.5.1.3 Design Evaluation

The SGTS is designed to preclude direct release of fission products from the reactor building to the environment during all modes of operation by the following features:

1. The SGTS is housed in a Category I structure. All surrounding equipment, components, and supports are designed to pertinent safety class and Category I requirements.

2. The SGTS consists of two 100-percent capacity, physically separated filter trains. Should any component in one train fail, filtration can be performed by the redundant train.

3. The SGTS component design and qualification testing are in accordance with the recommendations of Regulatory Guide 1.52 to the extent discussed in Section 1.8.

4. During loss of offsite power, all active components such as motors, damper operators, controls, and instrumentation operate from their respective independent standby power supplies.

Should a loss-of-coolant accident occur during primary containment purge with the SGTS operating in the pressure

TABLE 6A.4-4

4TCO TIME PERIODS FOR BASIC CO LOAD
AND CO LOAD WITH ADS

	Run Number	Time Segment Specified for Basic CO Load* (sec)	Time Interval of CO Observed in Test (sec)
Basic CO Load	3	13 to 15	5 to 29
	4	10 to 12	5 to 29
	5	19 to 21	5 to 38
	8	5 to 7	5 to 28
	9	10 to 23	5 to 38
	10	28 to 30	5 to 41
	12	21 to 25	5 to 48
	14	25 to 31	5 to 59
	15	31 to 48	5 to 50
	22	13 to 21	5 to 32
	23	5 to 7	5 to 28
	24	12 to 14	5 to 30
	25	32 to 42	5 to 29
			30 to 42 32 to 44
	26	16 to 24,	5 to 26 24
		32 to 36	29 to 38 37
	27	16 to 34	5 to 40
	28	17 to 19	5 to 36
CO Load with ADS	13	50 to 59	5 to 60
	14	50 to 59	5 to 59

*These segments contain the maximum PSD between 0 and 60 Hz in applicable test conditions. Selection criteria are discussed in Sections 6A.4.5.3.1 and 6A.4.5.3.2.

the RHR shutdown cooling subsystem injection and suction lines.

Four redundant pressure transmitters, two for each set of valves, monitor reactor vessel pressure. Each of the pressure monitoring sensors provides a signal to one of the two (inside or outside) logic trip channels.

Testability

Refer to Section 7.3.2.1.3, Conformance to Regulatory Guide 1.22.

7.3.1.1.3 RHR Containment Spray Cooling Mode (RCSCM) - Instrumentation and Controls

System Function

The RCSCM is an operating mode of the RHR system. It is designed to condense steam in the suppression chamber air volume and/or the drywell atmosphere following a LOCA (Section 6.5.2.3.1).
2

System Operation

Schematic arrangements of system mechanical equipment are shown on Figure 5.4-13. RHR system component control logic is shown on Figure 7.3-6. Instrument specifications are listed in Table 7.3-6. Elementary diagrams are identified in Section 1.7. Operator information displays are shown on Figures 5.4-13 and 7.3-6.

The RCSCM is initiated by the control room operator by diverting LPCI flow to the suppression pool via valves MO F027A (MOV33A) or F027B (MOV33B). The LPCI flow can also be diverted to the drywell via MO F016A (MOV15A) and F017A (MOV25A) or F016B (MOV15B) and F017B (MOV25B).

The following permissive conditions must exist before the operator can initiate a containment spray cooling loop:

1. The automatic LOCA signal or manual pushbutton signal that initiated the LPCI must still exist.

2. Drywell high pressure is monitored by two redundant pressure transmitters. One of the two transmitters must indicate high pressure. (Applies only to valves F016A, B and F017A, B.)

3. The operator must close the LPCI injection valves MO F042A (MOV24A), F042B (MOV24B).

events is similar to those under LOCA (Condition 2) followed by tripping of the offsite power supply breakers and shedding of selected loads on emergency buses and then sequencing loads. The first group of loads is sequenced after the residual voltage on the largest motor in this group has decayed to approximately 25 percent of its rated voltage, which takes 1 sec. The sequence of events occurring in case of LOOP and subsequent LOCA is similar to those under LOOP (Condition 1).

Tables 8.3-1 and 8.3-2 give details of sequencing loads on the standby diesel generators under the conditions of simultaneous LOCA and LOOP, LOOP with delayed LOCA, LOCA with delayed LOOP, and LOOP with unit trip.

In these tables, the time, T, is measured from the instant the diesel generator attains its rated speed, voltage, and frequency, and is connected to the bus by closing the diesel generator breaker.

T=0 indicates that these loads are not shed from their buses and are energized as soon as the diesel generator breaker is closed.

T=1, 6, etc, indicates the time of the equipment start after the closing of the diesel generator breaker (T=0).

Sequencing of the service water pumps under the first three conditions described above is as follows. (Sequencing for Division I is explained. Division II is similar and is indicated in parentheses.)

At T=32 sec, service water pump 2SWP*P1A (2SWP*P1B) 23
receives an autostart signal if it was previously in service. The other two Division I (Division II) service water pumps are blocked from manual
starting until T=55 sec. 23
70
37
At T=39 sec, service water pump 2SWP*P1C (2SWP*P1D) 23
receives an autostart signal if it was previously in service and none of the other Division I (Division II) service water pumps are running. Pumps 2SWP*P1E and 2SWP*P1A (2SWP*P1F) are blocked
from manual starting until T=55 sec. 23
70

42

23 At T=46 sec, service water pump 2SWP*P1E (2SWP*P1F) receives an autostart signal if it was previously in service and if none of the other Division I (Division II) service water pumps are running.
23 Pumps 2SWP*P1A and 2SWP*P1C are blocked from manual starting until T=55 sec.

nonaccident 70

23 For condition IV (LOOP with unit trip, noncoincident loading), the sequencing of the service water pumps is similar, except that the sequencing starts earlier (T=32, 39, and 46 secs), since there are no RHR or LPCS pumps sequenced to start prior to service water. 37 42

under this condition

Tables 8.3-5 and 8.3-6 give total starting and running loads under different load blocks totaled from Tables 8.3-1 and 8.3-2, respectively.

Assumptions used in these calculations are as follows:

a. For 600-volt motor loads, a conservative value of 1 kVA per hp has been assumed.

b. For 4-kV motors, brake hp driven equipment requirements are used to calculate motor kVA.

c. MOVs are assumed to be energized intermittently and all Class 1E MOVs are assumed to operate for the initial 2-min post-accident period. MOVs are assumed to have a starting kVA of 10 times the nameplate hp rating. If actual MOV constant load operating torques are not known, load torque is assumed to be 200 percent of nameplate hp rating. The higher opening or closing torque usually required at the beginning or end of the valve operation is assumed to occur over the initial 2-min diesel generator loading sequence and does not appear as a significant additional load to the diesel generator. Twenty percent of the MOV running loads are assumed after the initial 2 min of diesel generator loading sequence. These assumptions are very conservative.

d. Transformer magnetizing inrush current is assumed to be 12 times full load nameplate current with a 0.1 sec duration. Transformer losses for units smaller than 75 kVA are

9A.3.7.4 Plant Computer Room

The plant computer is not safety related; it is not required to shut down the reactor or limit the release of radioactivity to the environment. The computer room is located within the control room pressure boundary on the relay room elevation. The computer room walls have glass panels and are not fire rated. The relay room elevation of the control room pressure boundary is separated from all other areas by 3-hr barriers. A smoke detection system that alarms in the control room and an automatic total-flooding Halon suppression system have been provided in the computer room. Manual CO_2 and water hose reels and portable extinguishers are available for backup.

9A.3.7.5 Switchgear Rooms

Switchgear rooms throughout the plant are protected with automatic total-flooding CO_2 extinguishing systems. Manual CO_2 and water hose reels and portable extinguishers are available for backup protection. Detection systems annunciate in the control room and alarm locally. Three-hour walls separate switchgear rooms from each other and
6 from the remainder of the plant except for the 600 volt switchgear room which is seperated by 1 1/2 hour barrier. Manually actuated smoke removal capability is provided for normal switchgear building and the control building switchgear rooms. For further details, refer to Section 9.4. Smoke removal capability would be provided through the use of portable fans if required for other switchgear rooms.

9A.3.7.6 Remote Safety-Related Panels

Unit 2 has two remote shutdown rooms separated from each other and other plant areas by 3-hr barriers. An automatic fire detection system annunciates in the control room and alarms locally. Manual CO_2 and water hose reels and portable extinguishers have been provided outside this room.

9A.3.7.7 Safety-Related Battery Rooms

Battery rooms are separated from each other and from the remainder of the plant by 3-hr fire walls, which are in excess of the fire load calculated in the fire hazards analysis. All penetrations have a 3-hr rating.

Separate ventilation systems have been provided for battery rooms to limit hydrogen buildup to 2 percent maximum. Fan failure is annunciated in the control room. Loss of flow initiates the redundant fan and prints out on the computer.

9A.3.7.11 Safety-Related Pumps

(the RCIC pump room has non-UL rated relief panels.)

All safety-related pumps are separated from each other and from the remainder of the plant by 3-hr fire walls. The RHR and HPCS pump rooms do not have automatic sprinkler protection for the general area, because the fire hazards analysis demonstrates that this is not required. However, other than the pumps and motors themselves, the only other combustible materials in these rooms are the cable trays, which have automatic sprinkler systems. Early warning detection systems, which alarm locally and in the control room, and manual water hose reels and portable extinguishers have been provided. The RCIC pump room has a water sprinkler system which is automatically actuated by smoke detectors. Floor drains are included in these areas. The ventilation system in the reactor building is a 100-percent outside air supply and exhaust system that will remove smoke in the event of a fire.

9A.3.7.12 New Fuel Area

Smoke detection, water hose reels, and portable extinguishers are provided throughout the reactor building, which includes the new fuel area. The detection system 6 annunciates in the control room and alarms locally. The new fuel storage area has a drainage system to preclude the accumulation of water; however, the storage configuration is such that criticality would not occur if water accumulated.

9A.3.7.13 Spent Fuel Pool Area

Detection, water hose reels, and portable extinguishers are provided throughout the reactor building, which includes the spent fuel pool area. The detection system annunciates in the control room and alarms locally.

9A.3.7.14 Radwaste Building and Decontamination Areas

The radwaste building is separated from other areas of the plant by 3-hr fire walls. Automatic wet-pipe sprinklers are provided in the asphalt storage and process areas, liquid radwaste fiberglass storage tank areas, above certain cable tray runs in aisleways (four or more levels), and the truck loading area.

Preaction sprinklers are provided in the dry compacted waste area, and deluge protection is provided for the asphalt/radwaste process areas. Manual water hose reels and portable extinguishers are provided. An early warning detection system with local and control room annunciation is

the turbine building. Loss of main condenser vacuum, at a pressure greater than 23 in Hg abs, will close the turbine bypass valves (Section 10.4.4) and the MSIVs. The amount of steam exhaust to the turbine building is within technical limits and monitored at its release point (the main stack).

6 Circulating water inleakage is monitored by two methods. The first method is by grab samples as stated in Section 9.3.2. High sample conductivity will initiate a manual repair if the limits stated in Section 10.4.1.2 are exceeded. The second method is checked in the condensate polishing system as addressed in Section 10.4.6.

10.4.1.4 Tests and Inspections

The condenser shell receives a field hydrostatic test prior to initial operation. This test consists of filling the condenser shell with water and inspecting the entire tube sheet and shell welds and surfaces for visible leakage and/or excessive deflection. The tube side of the condenser is hydrostatically field tested at 5 psi above design pressure. The acceptance criteria are no visible leaks or loss of pressure within 30 min.

Insert next page

6 There are no inservice inspection attributes associated with the condenser. The condenser is monitored for air inleakage as stated in Section 10.4.1.3 and circulating water inleakage as stated in Section 10.4.1.2.

10.4.1.5 Instrumentation Requirements

Indicators for condenser vacuum are located in the main control room and are shown on Figure 10.4-1. For a description of condenser hotwell level controls and monitors refer to Section 9.2.6.5.

10.4.2 Main Condenser Air Removal System

10.4.2.1 Design Bases

10.4.2.1.1 Safety Design Bases

The main condenser air removal system is not required to effect or support the safe shutdown of the reactor or to perform in the operation of reactor safety systems. The system is designed to maintain hydrogen concentration below the lower explosive limit in the condenser.

All piping and components from the main condenser to the offgas system are designed to be capable of withstanding the

Insert for page 10.4-4

The tube side of the condenser is given an operational leak test as part of the circulating water system. The acceptance criterion is no visible leaks.

11. Radwaste sampling system (SSW) provides process sampling.

Condensate Makeup and Drawoff System

12. Auxiliary condensate system (CNA) provides flushing water for various piping sections. CNS

13. Radwaste seal water system (SWR) provides seal water for pumps with double mechanical seals. Details are given in Section 11.2.

A compactor is provided to compress dry wastes such as paper, rags, and plastic for packaging in plywood low specific activity (LSA) boxes. Incompressible solid wastes are packaged in LSA boxes, 55-gal drums, or encapsulated in
27 liners ranging in size from 50 to 200 cu ft.

Design of the waste solidification system and equipment is in accordance with Topical Report No. WPC-VRS-001, with the following differences:

1. One waste sludge tank is used to receive both spent resin and filter sludge in lieu of a tank for each feed stream (WPC-VRS-001, Drawing No. SK-VRS-2-R1).

2. The waste sludge tank is designed to ASME Section VIII Division I in compliance with Regulatory Guide 1.143 instead of API 620 (WPC-VRS-001, Amendment I, Attachment A, Table I) (see FSAR Section 1.8).

3. Asphalt storage temperature has been increased to 325°F from 250°F to facilitate pumping (WPC-VRS-001, Section II.B.2).

4. Asphalt storage tank overflow and drain are directed into the tank cubicle instead of outdoors (WPC-VRS-001, Amendment II, Response 1.7). The tank cubicle is designed to American Nuclear Insurers (ANI) requirements. It has a 3-hr fire rating and is capable of retaining the tank contents and potential fire suppression water.

5. The ventilation hood at the container fill station is not fitted with HEPA and charcoal filters (WPC-VRS-001, Amendment I, Response 2). Instead, ventilation air from the fill station is processed through the radwaste building ventilation system, which is consistent with the gaseous radwaste management design interior discussed in FSAR Section 11.3.

13.1.2.3 Operating Shift Crews

Table 13.1-2 shows the position titles, applicable operating licensing requirements, and minimum numbers of personnel planned for each shift for the various reactor operating modes. Unique requirements for additional personnel for the refueling modes are listed in Table 13.1-2. Round-the-clock chemistry and radiation protection coverage is met by qualified technicians. Technicians are qualified in accord-
9 ance with the requirements of ANSI/ANS-3.1-1978 as outlined in Section 13.2. Round-the-clock fire brigade coverage is provided by shift fire brigades. Fire brigades are qualified in accordance with 10CFR50 Appendix R as described in Section 13.2.

There are six operating shift crews. The schedule for filling all operating shift crews is listed in Table 13.1-5.

13.1.3 Qualifications of Personnel

13.1.3.1 Supervisory Personnel

The essential technical qualifications required for each position in the onsite supervisory staff meet the intent of the requirements of ANSI N18.1-1978 and Regulatory Guide 1.8. Table 13.1-4 shows the qualifications of site personnel in accordance with ANSI N18.1-1978.

ANSI/ANS-3.1 ANSI/ANS-3.1

TABLE 15.7-12

FUEL HANDLING ACCIDENT (DESIGN BASIS ANALYSIS)
RADIOLOGICAL EFFECTS

	Whole Body Dose (Rem)	Thyroid Dose (Rem)	Beta Dose (Rem)	
	6.46-01	4.40-01	7.10-01	
Exclusion area boundary (2 hr)	1.11-01	8.46-01	1.20-01	
	2.27-01	9.25+00	1.79-01	
Low population zone (2 hr)	3.59-02	9.63-02	3.85-02	12
	2.11-02	2.74+01	7.08-01	
Control room*	1.14-02	6.69-03	2.79-01	

*Control room doses listed are the maximum values calculated for time periods for and beyond the release duration of 2 hr. 12

NOTE: 1.11-01 = 1.11×10^{-1}

Info Only

TABLE 3-3

2MHR-CRN1/POLAR CRANE LOADS

Crane Load	Weight (tons)	Designated Lifting Device	Governing Handling Procedure	Frequency Handled
Fuel Transfer Shielding Bridge	34	Main Strongbacks and Sling Assemblies	*	Twice each refueling operation
Drywell Shield Plug A	90	Main Strongbacks and Sling Assemblies	*	Twice each refueling operation
Drywell Shield Plug B	102	Main Strongbacks and Sling Assemblies	*	Twice each refueling operation
Drywell Shield Plug C (with rigging)	119**	Main Strongbacks and Sling Assemblies	*	Twice each refueling operation
Drywell Shield Plug D	90	Main Strongbacks and Sling Assemblies	*	Twice each refueling operation
Drywell Shield Plug E	82	Main Strongbacks and Sling Assemblies	*	Twice each refueling operation
Drywell Shield Plug F	93	Main Strongbacks and Sling Assemblies	*	Twice each refueling operation
Drywell Shield Plug G	93	Main Strongbacks and Sling Assemblies	*	Twice each refueling operation
Drywell Shield Plug H	82	Main Strongbacks and Sling Assemblies	*	Twice each refueling operation
Drywell Head	55	Main Strongbacks and Sling Assemblies	*	Twice each refueling operation

TABLE 3-3 (Cont)

Crane Load	Weight (tons)	Designated Lifting Device	Governing Handling Procedure	Frequency Handled
Reactor Vessel Head	92	Main Strongbacks and Sling Assemblies	*	Twice each refueling operation
Steam Dryer	50	Main Strongbacks, Sling Assemblies, and Spreader Beams	*	Twice each refueling operation
Steam Separator	80	Main Strongbacks, Sling Assemblies, and Spreader Beams	*	Twice each refueling operation
Reactor Vessel Head Insulation and Support Frame	40 8	Main Strongbacks and Sling Assemblies	*	Twice each refueling operation
Spent Fuel Shipping Cask	100	Cask Lifting Yoke	*	As needed over the life of plant
Reactor Head Stud Rack	21 4	Sling Assemblies	*	Twice each refueling operation
Reactor Stud Tensioner	5	Sling Assemblies	*	Twice each refueling operation
Refueling Canal Plugs	16 (max)	Main Strongbacks and Sling Assemblies	*	Twice each refueling operation
WCS Filter Demineralizer Removal Plugs	15	Main Strongbacks and Sling Assemblies	*	As needed over the life of plant
SFC Filter Removal Plugs	10	Main Strongbacks and Sling Assemblies	*	As needed over the life of plant

TABLE 3-3 (Cont)

Crane Load	Weight (tons)	Designated Lifting Device	Governing Handling Procedure	Frequency Handled
SFC Filter Demineralizer Removal Plugs	4	Sling Assemblies	*	As needed over the life of plant
Reactor Service Platform	5	Main Strongbacks and Sling Assemblies	*	Twice each refueling operation
Storage Pool Gate	50	Main Strongbacks and Sling Assemblies	*	Twice each refueling operation
Recirculation Pump Motor	33.5	Main Strongbacks and Sling Assemblies	*	As needed over the life of plant

Insert next page

*Load-handling procedures will be developed to cover load-handling operations for heavy loads that are handled over or in proximity to spent fuel or safe shutdown equipment.
**Maximum load.

Insert for Table 3-3, 3 of 3

Crane Load	Weight (tons)	Designated Lifting Device	Governing Handling Procedure	Frequency Handled
Storage pool plug	82	Main Strongbacks and Sling Assemblies	*	Twice each refueling operation
Fuel Pool Gate	1.4	Main Strongbacks and Sling Assemblies	*	Twice each refueling operation
JIB Cranes	3.3	Main Strongbacks and Sling Assemblies		As needed over the life of plant
Decontamination Platform Boom extension	1.2	Main Strongbacks and Sling Assemblies		As needed over the life of plant
Removal hoist plate	3.6	Main Strongbacks and Sling Assemblies		As needed over the life of plant

(2066G)

TABLE 4-1

HAZARD ELIMINATION

MARK NO. 2MHR-CRN1
(25-Ton Auxiliary Hoist)

Impact Area: Reactor Building - Refueling Floor
Location: At El 387 ft 4 in, Azimuth 0° to 359°

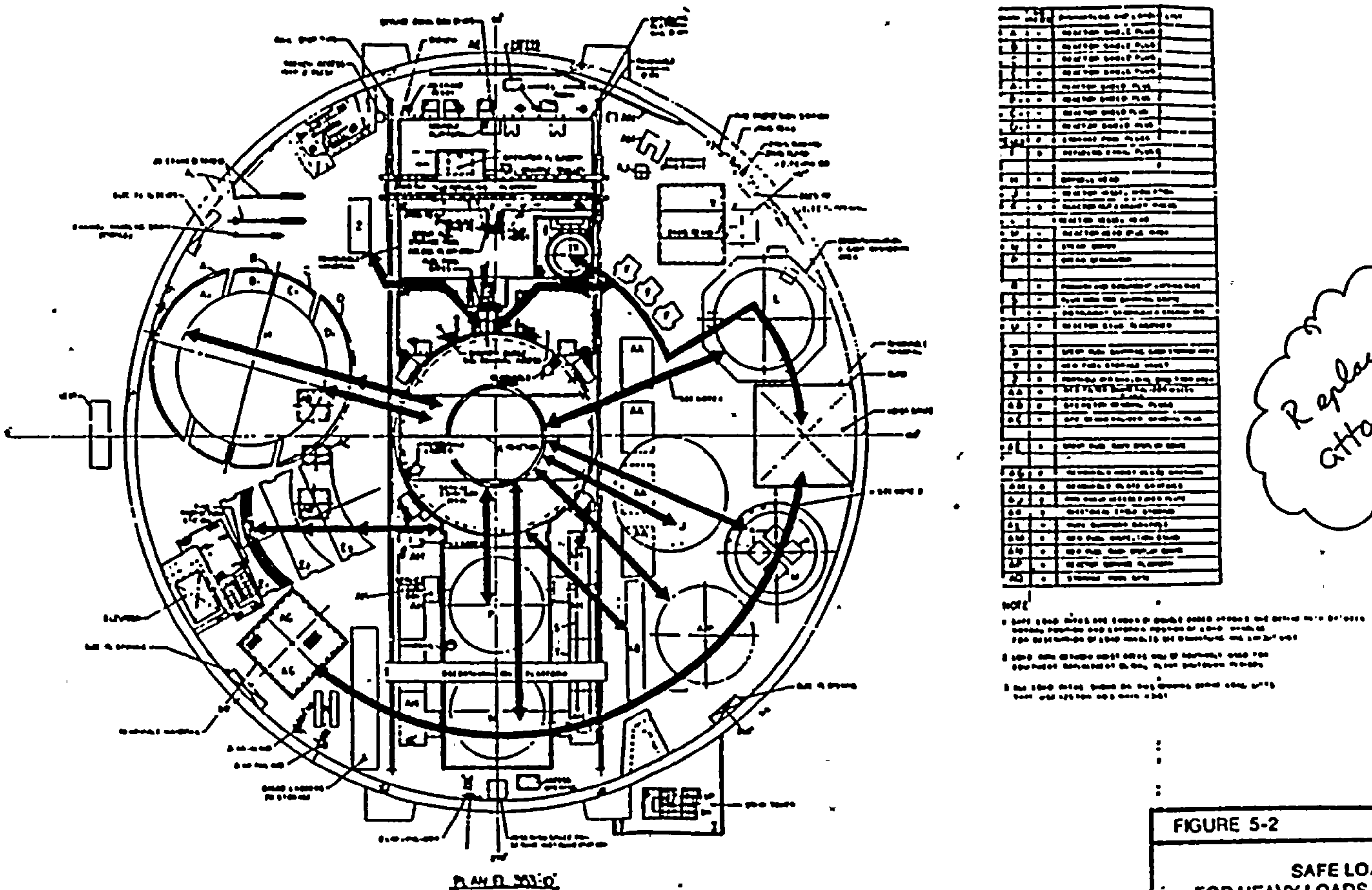
Load	Elevation	Safety-Related Equipment	Hazard Elimination Category
WCS Filter Demineralizer Removal Plugs	353 ft, 10 in	Reactor and Spent Fuel Pool	A*
SFC Filter Removal Plugs	353 ft, 10 in	Reactor and Spent Fuel Pool	A*
SFC Filter Demineralizer Removal Plugs	353 ft, 10 in	Reactor and Spent Fuel Pool	A*
Removal Plate Gratings	353 ft, 10 in	Reactor and Spent Fuel Pool	A*
Removal Hoist Plate Gratings	353 ft, 10 in	Reactor and Spent Fuel Pool	A*
Radioactive Tunnel Access Plug	353 ft, 10 in	Reactor and Spent Fuel Pool	A*

*Electrical interlocks and limit switches will prevent the carrying of any of these loads over spent fuel or safety-related equipment. See crane restricted area diagram (Figure 5-4).

Replace WITH Attachment A

ATTACHMENT A

When handling loads with the 25-ton auxiliary hoist, slings specified for use on safety-related equipment must be used. These loads shall not be permitted over safety-related equipment. When handling loads greater than 1/2-ton, lifts must be done in accordance NUREG-0612, Control of Heavy Loads at Nuclear Power Plants.

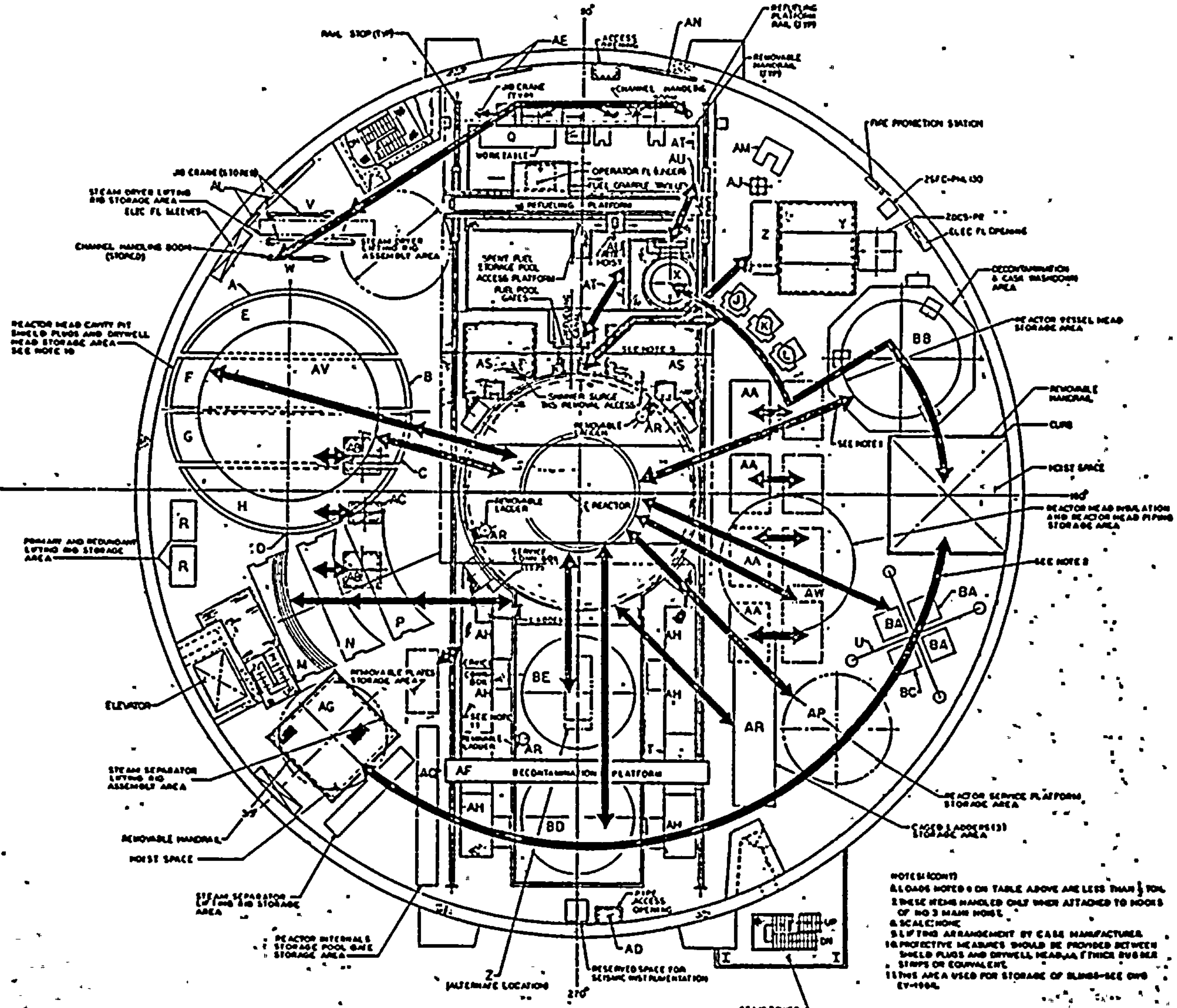


Replace with attached

FIGURE 5-2

SAFE LOAD PATHS
FOR HEAVY LOADS REFUELING FLOOR

NIAGARA MOHAWK POWER CORPORATION
NINE MILE POINT-UNIT 2
THE CONTROL OF HEAVY LOADS AT NMP2



PLAN EL 353'-10"

MARK	NO. REQD	DISMANTLING & LAYDOWN LIST	WEIGHT (TONS)	HOIST HOOK	REMARKS	REF. DWGS.
A	1	REACTOR SHIELD PLUG	90	NO. 3 (MAIN)	SEE NOTE 3	EV-190M
B	1	REACTOR SHIELD PLUG	102	NO. 3 (MAIN)	SEE NOTE 3	EV-190M
C	1	REACTOR SHIELD PLUG	102	NO. 3 (MAIN)	SEE NOTE 3	EV-190M
D	1	REACTOR SHIELD PLUG	90	NO. 3 (MAIN)	SEE NOTE 3	EV-190M
E	1	REACTOR SHIELD PLUG	82	NO. 3 (MAIN)	SEE NOTE 3	EV-190M
F	1	REACTOR SHIELD PLUG	93	NO. 3 (MAIN)	SEE NOTE 3	EV-190M
G	1	REACTOR SHIELD PLUG	93	NO. 3 (MAIN)	SEE NOTE 3	EV-190M
H	1	REACTOR SHIELD PLUG	82	NO. 3 (MAIN)	SEE NOTE 3	EV-190M
M	1	STORAGE POOL PLUG	82	NO. 3 (MAIN)	SEE NOTE 3	EV-190M
N	1	STORAGE POOL PLUG	78	NO. 3 (MAIN)	SEE NOTE 3	EV-190M
P	1	STORAGE POOL PLUG	76	NO. 3 (MAIN)	SEE NOTE 3	EV-190M
J	1	REFUELING CANAL PLUGS	13.5	NO. 3 (MAIN)	SEE NOTE 3	EV-190M
K	1	REFUELING CANAL PLUGS	13.5	NO. 3 (MAIN)	SEE NOTE 3	EV-190M
L	1	REFUELING CANAL PLUGS	16	NO. 3 (MAIN)	SEE NOTE 3	EV-190M
AT	2	FUEL POOL GATES-OUTER (2=TOTAL)	14	NO. 3 (MAIN)	SEE NOTE 3	EV-190M
AU	2	FUEL POOL GATES-INNER	0.9	NO. 3 (MAIN)	SEE NOTE 3	EV-190M
AV	1	DRYWELL HEAD	55	NO. 3 (MAIN)	SEE NOTE 3	EV-190J
AW	1	REACTOR VESSEL INSULATION FRAME	40	NO. 3 (MAIN)	SEE NOTE 3	EV-190K
BA	3	REACTOR NUT & WASHER RACKS	*	NO. 1 (AUX.)	SEE NOTE 5	———
BB	1	REACTOR VESSEL HEAD	92	NO. 3 (MAIN)	SEE NOTE 3	EV-190J
BC	1	REACTOR HEAD STUD RACK	28	NO. 3 (MAIN)	SEE NOTE 3	EV-190AA
BD	1	STEAM DRYER	50	NO. 3 (MAIN)	SEE NOTE 3	EV-190L
BE	1	STEAM SEPARATOR	80	NO. 3 (MAIN)	SEE NOTE 3	EV-190P
Q	1	WORK TABLE	*	NO. 1 (AUX.)	SEE NOTE 5	EV-190AA
R	2	PRIMARY AND REDUNDANT LIFTING RIGS	SEE NOTE 7	———	———	EV-190A
T	1	INSTRUMENT STRONGBACK STORAGE PIT	*	NO. 1 (AUX.)	SEE NOTE 5	EV-190AA
U	1	REACTOR STUD TENSIONER	5	NO. 3 (MAIN)	SEE NOTE 3	EV-190AP
V	2	JIB CRANES	3.3 EA	NO. 3 (MAIN)	SEE NOTE 3	EV-190V
W	1	CHANNEL HANDLING BOOM	0.5	NO. 3 (MAIN)	SEE NOTE 3	EV-190V
X	1	SPENT FUEL SHIPPING CASK	100	NO. 3 (MAIN)	———	SEE NOTE 9
Y	1	NEW FUEL VAULT GRATING	*	NO. 1 (AUX.)	SEE NOTE 5	EV-190AA
Z	1	FUEL TRANSFER SHIELDING BRIDGE	34	NO. 3 (MAIN)	SEE NOTE 3	EV-190N
AA	4	RCS FILTER DEMINERALIZER ACCESS PLUGS	15 EA	NO. 3 (MAIN)	SEE NOTE 3	EV-190U
AB	2	SFC FILTER REMOVAL PLUGS	10 EA	NO. 3 (MAIN)	SEE NOTE 3	EV-190U
AC	1	SFC FILTER PUMP REMOVAL PLUG	*	NO. 3 (MAIN)	SEE NOTE 3	EV-190U
AD	1	PIPE ACCESS COVER PLATE	*	NO. 1 (AUX.)	SEE NOTE 5	EV-190AA
AE	1	SPENT FUEL RACK DISPLAY BOARD	*	NO. 1 (AUX.)	SEE NOTE 5	EV-190AA
AG	1	REMOVABLE HOIST PLATE	3.6	NO. 3 (MAIN)	SEE NOTE 3	———
AH	8	REMOVABLE PLATES	1.2 EA	NO. 3 (MAIN)	SEE NOTE 3	———
AJ	1	PIPE CHASE ACCESS COVER PLATE	*	NO. 1 (AUX.)	SEE NOTE 5	EV-190AA
AF	1	DECON. PLATFORM BOOM EXTENSION	1.2	NO. 3 (MAIN)	SEE NOTE 3	EV-190AA
AL	1*	FUEL SUPPORT GRAPPLE	*	NO. 1 (AUX.)	SEE NOTE 5	———
AM	1	NEW FUEL INSPECTION STAND	1.5	NO. 3 (MAIN)	SEE NOTE 3	———
AN	1	NEW FUEL RACK DISPLAY BOARD	1.6	NO. 1 (AUX.)	SEE NOTE 5	———
AP	1	REACTOR SERVICE PLATFORM	5	NO. 3 (MAIN)	SEE NOTE 3	EV-190D
AQ	1	STORAGE POOL GATE	30	NO. 3 (MAIN)	SEE NOTE 3	EV-190N
AR	3	REMOVABLE CAGED LADDERS	*	NO. 1 (AUX.)	SEE NOTE 5	———
AS	4	SKIMMER TANKS GRATINGS	*	NO. 1 (AUX.)	SEE NOTE 5	———

NOTES:
1. SAFE LOAD PATHS ARE SHOWN BY DOUBLE ENDED ARROWS AND DEFINE PATH BETWEEN NORMAL POSITION AND LAYDOWN POSITION OF LOAD HANDLED. FOR DESCRIPTION OF LOAD HANDLED SEE DISMANTLING AND LAYDOWN LIST.
2. LOAD PATH BETWEEN HOIST AREAS WILL BE ROUTINELY USED FOR EQUIPMENT REPLACEMENT DURING PLANT SHUTDOWN PERIODS.
3. ALL LOAD PATHS SHOWN ON THIS DRAWING DEFINE LOAD LIFTS/PATHS USING THE 125 TON NO. 3 MAIN HOIST.
4. LOAD TRAVEL OUTSIDE OF DESIGNATED SAFE LOAD PATHS REQUIRES ADMINISTRATIVE CONTROL.
5. WHEN HANDLING LOADS WITH AUXILIARY HOOKS NO. 1 & NO. 2, SPECIAL CARE MUST BE USED INCLUDING USING SLINGS SPECIFIED FOR USE ON SAFETY RELATED EQUIPMENT. THESE LOADS SHALL NOT BE PERMITTED OVER SAFETY RELATED EQUIPMENT. WHEN HANDLING LOADS GREATER THAN ½ TON WITH AUXILIARY HOOK NO 2 AN EVALUATION MUST BE DONE IN ACCORDANCE WITH HAZARD ELIMINATION TABLES IN SECTION 4 OF THE CONTROL OF HEAVY LOADS AT NMP2 FSAR, BEFORE ANY LIFTS ARE MADE. AREA ENCLOSED BY ——— SHOWS LOCATION OF SAFETY RELATED EQUIPMENT.

NOTES (CONT)
6. LOADS NOTED * ON TABLE ABOVE ARE LESS THAN ½ TON.
7. THESE ITEMS HANDLED ONLY WHEN ATTACHED TO HOOKS OF NO 3 MAIN HOIST.
8. SCALE: NONE
9. LIFTING ARRANGEMENT BY CASK MANUFACTURER.
10. PROTECTIVE MEASURES SHOULD BE PROVIDED BETWEEN SHIELD PLUGS AND DRYWELL HEAD, AA" THICK RUBBER STRIPS OR EQUIVALENT.
11. THIS AREA USED FOR STORAGE OF BLINDS-SEE DWG EV-190AL.

REFERENCE DRAWINGS:
1. DISMANTLING & LAYDOWN - OPERATING FLOOR — EM-45A
2. CRANE RESTRICTED AREA DIAGRAM — S&W NO. 2410-044-064
3. BRIDGE & TROLLEY TRAVEL - RESTRICTED PATH - L.S. LOCATIONS — S&W NO 2410-044-007

NUCLEAR SAFETY RELATED QA CAT I

SAFE LOAD PATHS FOR HEAVY LOADS REFUELING FLOOR

NINE MILE POINT NUCLEAR STATION — UNIT 2
NIAGARA MOHAWK POWER CORPORATION
STONE & WEBSTER ENGINEERING CORPORATION

SEE RECORD OF DRAWING CHANGES 12177-ECF-22A-3

FOR INFORMATION ONLY

CH 12177-EF-22A-3

TABLE 421.36-1

CONFORMANCE TO REGULATORY GUIDE 1.97

SWEC/	Parameter	Reg. Guide 1.97, Rev. 3 Parameter		Sensor		Qualification		QA	Power	Display	
GE-NED I.D.#	Description	Variable	Classification	Location	Instr. Range	Seismic	Environmental	Class	Supply	Location	Notes
B13-D193	Power Rng Flux Level	B1a.	1	Core	0.5-125% pwr	Yes	Yes	II	Non-1E	P603	1
	Average Pwr Rng Flux Lvl	B1b.	1	N/A	0-125% pwr	No	No	II	Non-1E	P603	2
C51-N002A-H	Inter-mediate Rng Flux Level	B1c.	1	Core	4.0x10^{-4} 4.0x10^{-5} - 12.6% Pwr	Yes	Yes	II	Non-1E	P603	-
C51-N001A-D	Source Rng Flux Level	B1d.	1	Core	0.1-1x10^{6} cps	~~Yes~~ No	~~Yes~~ No	II	Non-1E	P603	-
	Control Rod Position	B2	3	Core	Withdrawn or Scram	~~Yes~~ No	~~Yes~~ No	II	Non-1E	P603	-
	Rx Coolant Boron Conc	B3	3	Unit 1 N.P. Lab.	50-2,000 ppm	N/A	N/A	II	Non-1E	-	4
2ISC*LT13A/ B22-N044A	Reactor Vsl Level - A (Fuel Zone)	B4a.	1	Rx Bldg (Sec Contmt)	230.64-430.69"	Yes	Yes	I	Div. 1	P601	5,41
2ISC*LT13B/ B22-N044B	Reactor Vsl Level - B (Fuel Zone)	B4b.	1	Rx Bldg (Sec Contmt)	230.69-430.69"	Yes	Yes	I	Div. 2	P601	5,41
2ISC*LT9C/ B22-N091E, 2ISC*LT9A/ B22-N091A	Reactor Vsl Level - A (Wide Range)	B4a	1	Rx Bldg (Sec Contmt)	375.70-585.70	Yes	Yes	I	Div. 1	P601	5,41
2ISC*LT9D/ B22-N091F, 2ISC*LT9B/ B22-N091B	Reactor Vsl Level - B (Wide Range)	B4b	1	Rx Bldg (Sec Contmt)	375.70-585.70	Yes	Yes	I	Div. 2	P601	5,41
N/A N/A	Core Temperature	B5	1	-	-	-	-	-	-	-	6