NRC FORM 195

U.S. NUCLEAR REGULATORY COMMISSION

NRC DISTRIBUTION FOR PART 50 DOCKET MATERIAL

DOCKET NUMBER	[redacted] 50-410
FILE NUMBER	
DATE OF DOCUMENT	5-4-76
DATE RECEIVED	5-8-76
NUMBER OF COPIES RECEIVED	1 signed

TO: Mr. J. F. Stolz

FROM: Niagara Mohawk Power Corp.
Syracuse, N.Y. 13202
Gearald K. Rhode

☒LETTER ☒ORIGINAL ☐COPY ☐NOTORIZED ☒UNCLASSIFIED

PROP

INPUT FORM

DESCRIPTION Ltr trans the following:

DISTRIBUTION PER MR. KANE 3-13-76

PLANT NAME: Nine Mile Pt. Unit 2 Plant

ENCLOSURE ATTACHMENT A entitled "Main Steam Line Islation Valve Water Seal Leakage Analysis Abstract & Results for Nine Mile Pt. Unit 2"...

ATTACHMENT B entited "Change from Inerted to Air Containment Atomsphere(De-Inerted) for Nine Mile Pt. Unit 2 Plant"....

(1 cy ea encl rec'd)

ACKNOWLEDGED

Do Not Remove

FOR ACTION/INFORMATION

SAFETY			ENVIRO	DHL 5-13-76
	ASSIGNED AD :		ASSIGNED AD :	
✓	BRANCH CHIEF :	Stolz	BRANCH CHIEF :	
✓	PROJECT MANAGER:	Kane	PROJECT MANAGER :	
✓	LIC. ASST. :	Smith	LIC. ASST. :	

INTERNAL DISTRIBUTION

✓	REG FILE		SYSTEMS SAFETY		PLANT SYSTEMS		ENVIRO TECH
✓	NRC PDR		HEINEMAN		TEDESCO		ERNST
✓	I & E (2)		SCHROEDER	✓	BENAROYA		BALLARD
	OELD			✓	LAINAS		SPANGLER
	GOSSICK & STAFF		ENGINEERING		IPPOLITO		
	MIPC		MACCARY				SITE TECH
	CASE	✓	KNIGHT		OPERATING REACTORS		GAMMILL
	HANAUER		SIHWEIL		STELLO		STEPP
	HARLESS		PAWLICKI				HULMAN
					OPERATING TECH		
	PROJECT MANAGEMENT		REACTOR SAFETY		EISENHUT		SITE ANALYSIS
	BOYD		ROSS		SHAO		VOLLMER
	P. COLLINS		NOVAK		BAER		BUNCH
	HOUSTON		ROSZTOCZY		SCHWENCER		J. COLLINS
	PETERSON		CHECK		GRIMES		KREGER
	MELTZ						
	HELTEMES		AT & I		SITE SAFETY & ENVIRO		
	SKOVHOLT		SALTZMAN		ANALYSIS		N
			RUTBERG		DENTON & MULLER		

EXTERNAL DISTRIBUTION

				CONTROL NUMBER
✓	LPDR: OSWEGO, N.Y.	NATL LAB	BROOKHAVEN NATL LAB	
	TIC	REG. V-IE	ULRIKSON(ORNL)	
	NSIC	LA PDR		4580
	ASLB	CONSULTANTS		
✓	ACRS 16 HOLDING [redacted]			

NIAGARA MOHAWK POWER CORPORATION

NIAGARA MOHAWK

300 ERIE BOULEVARD, WEST
SYRACUSE, N.Y. 13202

Regulatory Docket File

May 4, 1976

RECEIVED
MAY 8 1976
U.S. NUCLEAR REGULATORY COMMISSION
Mail Section

Director of Reactor Regulation
Attn: Mr. J. F. Stolz, Chief
Branch 2-1
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

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

DOCKETED
USNRC
MAY 8 1976
Mail Section
Docket Clerk
DL

Dear Sir:

Attached for your review and approval are three copies each of the "Main Steam Line Isolation Valve Water Seal Leakage Analysis Abstract and Results" (Attachment A) and "Change from Inerted to Air Containment Atmosphere (De-Inerted)" (Attachment B) for Nine Mile Point Unit 2. Attachment A is submitted in response to Section 9.2.4 of the Commission's Safety Evaluation Report which required certain additional information and analyses. Also, Attachment B provides justification for changing to a de-inerted containment in accordance with Draft Regulatory Guide 1.7, August 1974.

Very truly yours,

NIAGARA MOHAWK POWER CORPORATION

GERALD K. RHODE
Vice President - Engineering

NLR/sz

Attachments

4580

Attachment A

MAIN STEAM LINE ISOLATION VALVE WATER SEAL LEAKAGE ANALYSIS ABSTRACT AND RESULTS

NINE MILE POINT NUCLEAR STATION - UNIT 2
NIAGARA MOHAWK POWER CORPORATION

Docket No. 50-410

April, 1976

Table of Contents

LIST OF FIGURES

1.0 INTRODUCTION

The purpose of this report is to provide a description of the leakage analysis and orientation of the main steam isolation valves to be installed at Nine Mile Point Nuclear Station - Unit 2. This analysis is submitted in response to Section 9.2.4 of the Commission's Safety Evaluation Report.[1,2] The results of the full analysis of the valve and water seal system will be contained in the Final Safety Analysis Report.

2.0 DISCUSSION

An analysis was performed to determine the water leakage rate and thermal stress levels of a typical main steam isolation valve (shown in Figure 1) considering valve component temperature distribution and actuator forces during operation of the water seal system.[3]

The valve used in this report is a 26-inch diameter, Y-pattern globe valve[4] with its cylindrical poppet and disc moving on a centerline 45 degrees to the steam flow direction. All main steam line isolation valves will be mounted in 45 degree pipe runs (see Figure 1 and 3) so that the valve stem is vertical. This is an improvement from the orientation described in the Preliminary Safety Analysis Report.[5] The improved orientation meets all code requirements and the stress and fatigue limits for piping between isolation valves described in Mechanical Engineering Branch Technical Position 3-1".[6] The vertical valve stem orientation results in the following advantages:

1. Reduced Personnel Exposure

   With a vertical stem and operator, the main steam line isolation valve is significantly easier to maintain, thus reducing personnel exposure during maintenance.

2. Improved Operability

   With the stem movement vertical, friction forces will be reduced, and, consequently, valve operability will be improved.

3. Reduced Maintenance Costs

   Reduced maintenance time will result in cost savings over the life of the plant.

This analysis of the typical valve is conservative in that the assumed clearances around the poppet (between the poppet and the body) are small at $\theta_1 = 0$ degrees (see Figure 1). The assumed clearance results in a severe flow restriction of seal water to the downstream side of the valve poppet and, therefore, a higher temperature differential across the poppet.

3.0 ANALYTICAL METHOD

The major assumptions used in the analysis are:

Seal water at 100 F is injected into the MSLIV through the warm-up drain connection with the valve at an equibilbrium temperature of 546 F. Actual seal water temperature will be somewhat lower (70-100 F), and the lower temperature will be factored into the final analysis. However, the effect on the results is expected to be insignificant.

The maximum valve actuator force is 30,000 lbs.

During seal water injection, the differential pressure across the valve disc remains at 50 psi.

The hard-faced sealing surfaces of the valve disc and seat have only point contact at the location $\theta_1 = 0$ degrees for all cases of thermal distortion and application of the actuator load.

The valve is closed with essentially zero leakage occuring before the 100 F water injection.

The inner isolation valve is analyzed for leakage with the outer isolation valve considered to have failed in the open position.

An elastic analysis is applicable.

Seal water injection flow rate is 70 to 80 gpm.

A heat transfer analysis of the valve body and disc was performed to determine temperature distribution as a function of time during seal water injection into the hot valve. During this period, one side of the valve body, poppet, and disc ($\theta_2 = 180$ degrees) is cooled by a larger pool of water than the opposite ($\theta_1 = 0$ degrees) side. The major heat transfer mechanism for the side cooled by the larger pool of water is nucleate boiling during most of the thermal shock transient. The heat transfer mechanism for the opposite side is stable film boiling due to the more restrictive circulation path for seal fluid. Heat transfer coefficients were calculated based on the references listed in footnotes 10 and 11.

Temperature distributions in the body and poppet were obtained by the "LION" heat transfer computer code[7] in the form,

$$T(r,z,\theta,t) = TF1(r,z,t) + TF2(r,z,t)\cos\theta.$$

Where
- r = Radial location
- z = Axial location
- θ = Circumferential location
- t = Time

At any time $T(r,z,0,t)$ gives the r-z plane temperature distribution due to stable film boiling. $T(r,z,180,t)$ gives the r-z plane temperature distribution due to convective nucleate boiling on the opposite side of the valve. These temperature distributions versus time were input into separate finite element models of the valve body and disc. Thermal distortions were determined with the ASAAS computer program.[8] The ASAAS program was also used in performing thermal stress analyses. Displacements of the valve disc and body, due to actuator seating forces applied to the valve stem, were calculated using ASAAS and SHELL 1[8] computer codes.

The leakage area due to differential thermal growth and the actuator seating force is shown in Figure 2. The maximum gap width, G_{max}, was obtained by considering the radial and axial relative motions of the valve disc and body at $\theta_1 = 0$ and $\theta_2 = 180$ degrees. The gap width variation around the circumference is expressed as $G(\theta) = G_{max} \sin \theta/2$ as shown in Figure 2. The total leakage of injection water from the main steam line isolation valve was calculated based on the area of this gap between the valve disc and the body.

Using the methods described above, the peak valve leakage was determined to be 26.4 gpm, 7 minutes following seal water injection. Leakage rate as a function of time is shown in Figure 3. Valve stresses are within the allowable limits.[9]

4.0 CONCLUSIONS

Since the peak leakage rate of 26.4 gpm is lower than the injection rate (70 - 80 gpm), a water seal will be maintained. Valve leakage will be steam produced due to the flashing of spent fuel pool water rather than steam from the reactor vessel.

Footnotes:

1. Safety Evaluation of the Nine Mile Point Nuclear Station - Unit 2 (SER), June 15,1973.

2. SER, Supplement 1, August 7, 1973.

3. Nine Mile Point Nuclear Station - Unit 2 Preliminary Safety Analysis Report (PSAR) Chapter 10 Section 10.16.

4. "Design and Performance of GE Boiling Water Reactor Main Steam Isolation Valve" Report No. APED-5750, General Electric Co., March, 1969.

5. PSAR Chapter 4 Section 4.5.3.

6. Branch Technical Position MEB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment."

7. "LION" Computer Code "Temperature Distribution for Arbitrary Shapes and Arbitrary Spaces and Complicated Boundary Conditions," J. R. Schmid, et al, GE-KAPL, KAPL-M-6532 (EC-57,) July 27, 1966.

8. PSAR Appendix C Response C7.6.

9. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, Div. I, Nuclear Power Plant Components, 1974, paragraph NB 3222.2.

10. Heat, Mass and Momentum Transfer, Rohsenow & Choi, Prentice-Hall, 1961.

11. Heat Transmission, W. H. McAdams, McGraw-Hill, 1954.

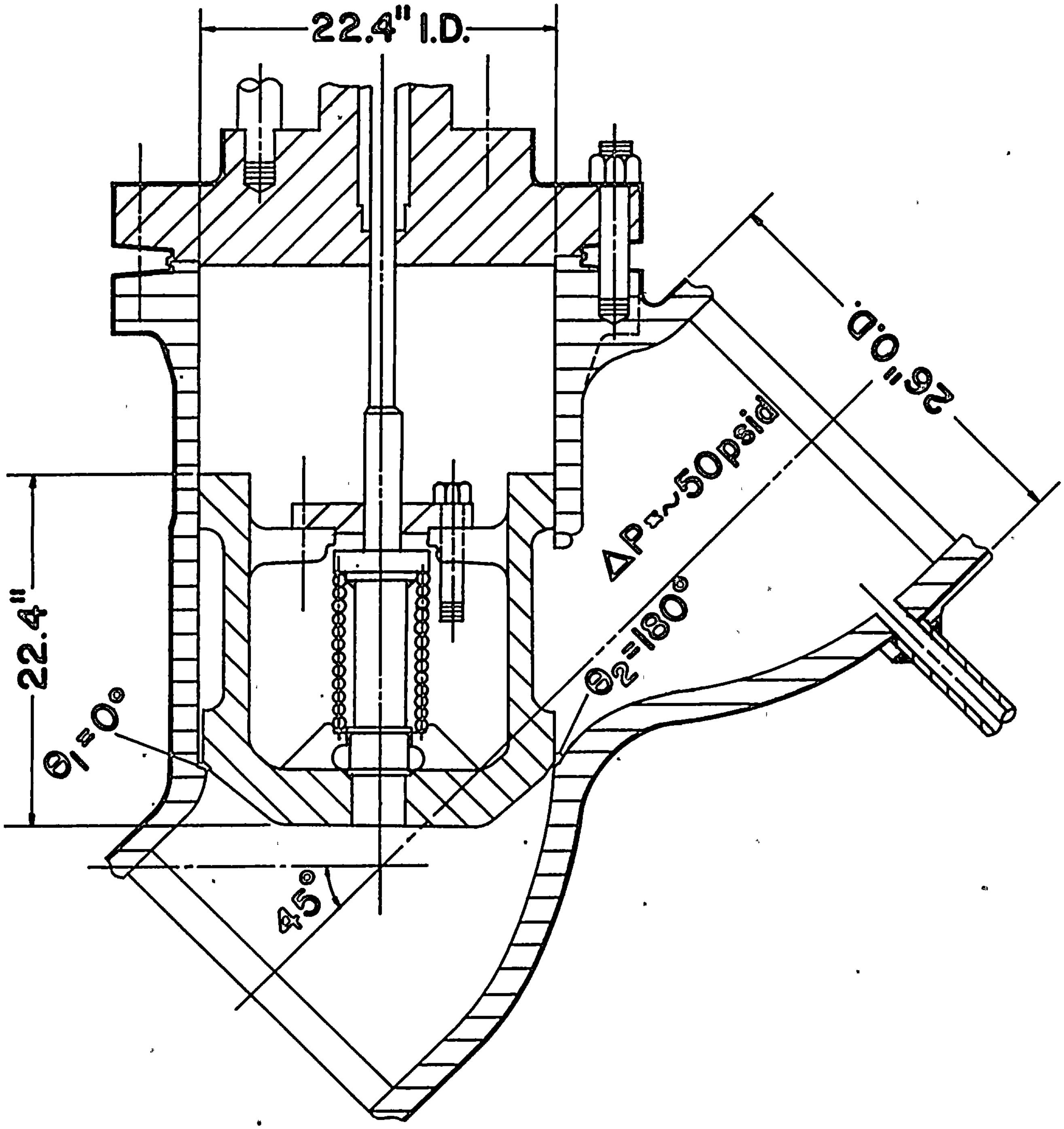


Figure 1
TYPICAL MAIN STEAM ISOLATION VALVE

NINE MILE POINT NUCLEAR STATION, UNIT 2
NIAGARA MOHAWK POWER CORPORATION

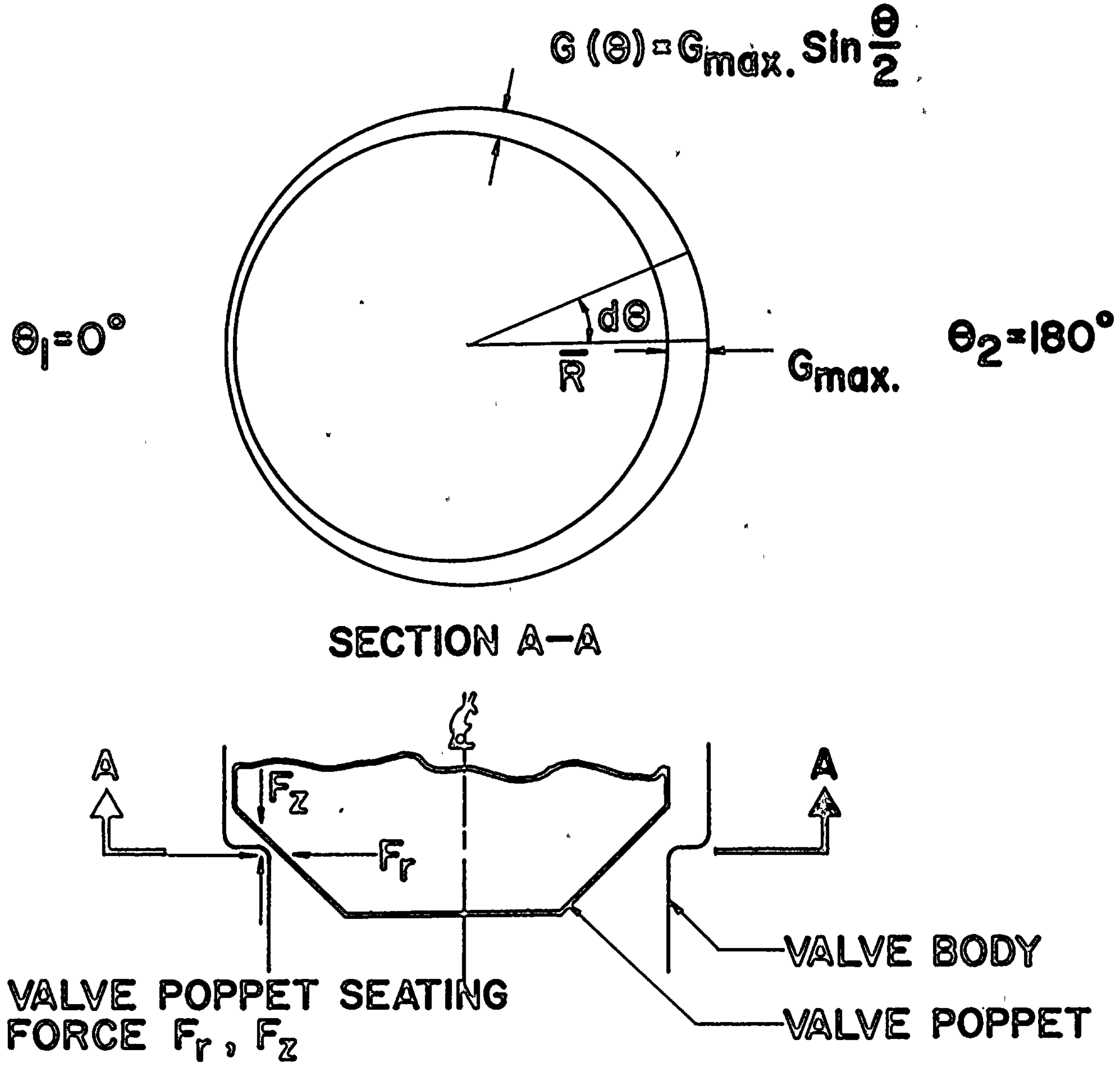


Figure 2
LEAKAGE AREA AND GAP WIDTH VARIATION

NINE MILE POINT NUCLEAR STATION, UNIT 2
NIAGARA MOHAWK POWER CORPORATION

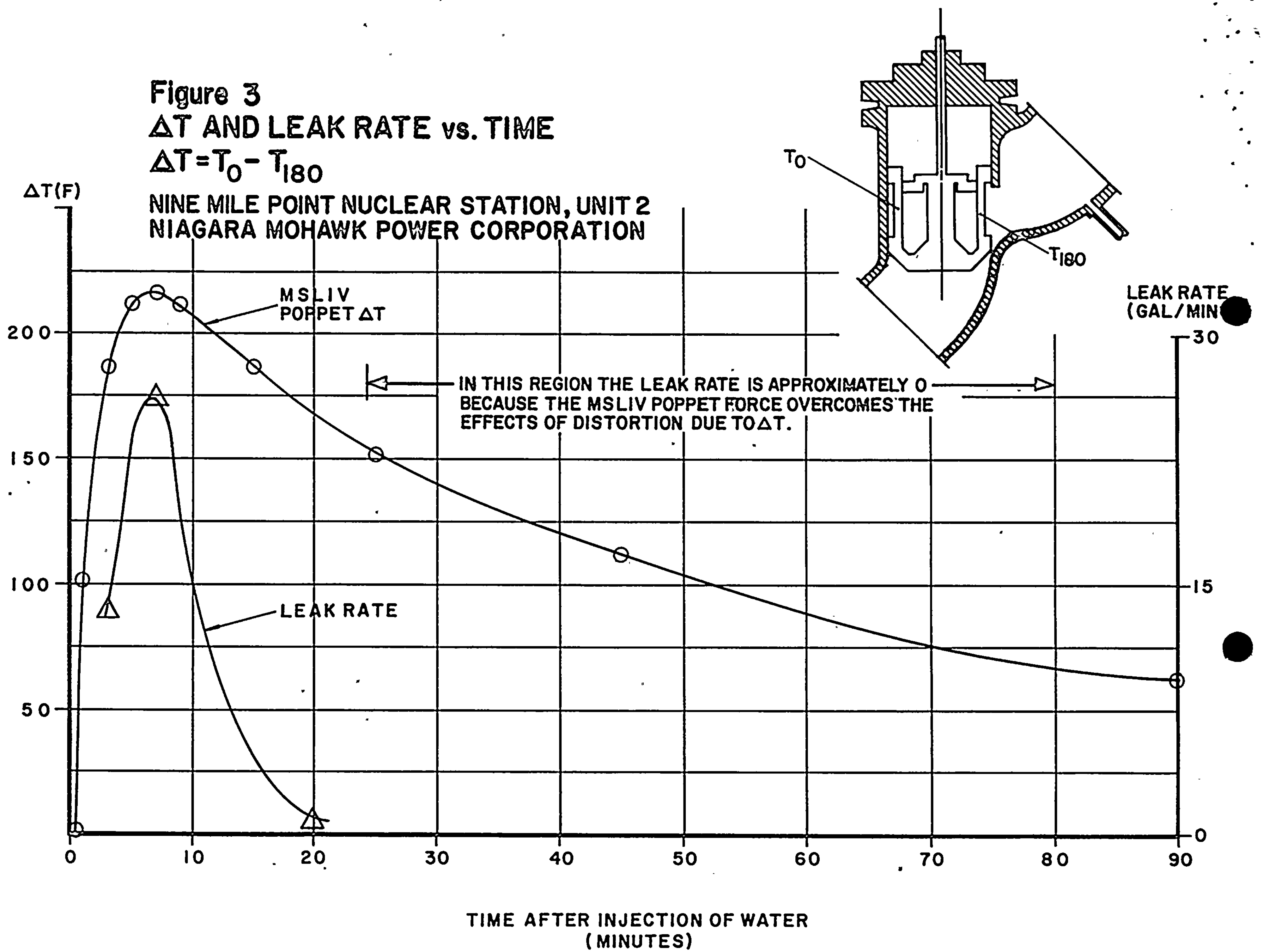


Figure 3
ΔT AND LEAK RATE vs. TIME
$\Delta T = T_0 - T_{180}$
NINE MILE POINT NUCLEAR STATION, UNIT 2
NIAGARA MOHAWK POWER CORPORATION
ΔT(F)
T_0
T_{180}
MSLIV POPPET ΔT
LEAK RATE (GAL/MIN)
IN THIS REGION THE LEAK RATE IS APPROXIMATELY 0 BECAUSE THE MSLIV POPPET FORCE OVERCOMES THE EFFECTS OF DISTORTION DUE TO ΔT.
LEAK RATE
200
150
100
50
30
15
0
0
10
20
30
40
50
60
70
80
90
TIME AFTER INJECTION OF WATER
(MINUTES)

Attachment B

CHANGE FROM INERTED TO AIR CONTAINMENT
ATOMSPHERE (DE-INERTED)

NINE MILE POINT NUCLEAR STATION - UNIT 2
NIAGARA MOHAWK POWER CORPORATION

Docket No. 50-410

April, 1976

LIST OF FIGURES

1.0 INTRODUCTION

This report describes the analysis of hydrogen generation due to a loss of coolant accident for Nine Mile Point - Unit 2. The report compares hydrogen control utilizing a hydrogen recombiner and an air purge system for both inerted and noninerted containments. This submittal is provided to justify changing from an inerted primary containment to a noninerted containment. This change does not compromise safe control of the flammable mixtures of hydrogen and oxygen produced during a loss of coolant accident. A noninerted containment is desirable because it simplifies the control of concentrations of flammable mixtures and eliminates the safety hazards associated with residual nitrogen in the containment.

Several disadvantages of an inerted containment may be summarized as follows:

1. Entry of working personnel into an inerted conatainment requires special breathing apparatus and may be hazardous.

2. Reduced inspection capability due to time spent in determining hazardous residual concentrations of nitrogen after deinerting the containment.

3. Additional radiation exposure resulting from the recognized need to locate possible nitrogen pockets.

4. An inerted containment may require a recombiner with a recycle stream and provision to add oxygen at the recombiner inlet. This involves more equipment and controls, increasing maintenance, and reducing reliability.

Regulatory Guide 1.7 (1) originally specified that, following a loss of coolant accident, a five percent metal cladding/water reaction was assumed to occur. Draft 2 of Regulatory Guide 1.7 (2) and Branch Technical Position CSB 6-2 (3) have altered the basis for determination of hydrogen generation. This results in substantially lower calculated hydrogen generation following a loss of coolant accident.

2.0 DISCUSSION

The present hydrogen control system consists of redundant Seismic Category I recombiner systems with a non-safety related standby air purge system to be used as a backup system. Each recombiner system takes suction from either the drywell or suppression chamber or both and returns hydrogen-free air to the suppression chamber.

The air for the standby purge system is introduced into the containment from the service air system. Initially, air is added which reduces hydrogen concentration and allows containment pressure to rise without releasing to the atmosphere. In this manner, radioactive contaminants are held as long as possible. When the pressure rises to approximately 38 psig, air will be released through a filtration system to the atmosphere. Air will also be added to the primary containment to keep hydrogen concentration below four percent while not exceeding 38 psig.

Of the two alternative criteria used by the Branch Technical Position for determining the performance of the hydrogen control system, it is expected that the one percent metal cladding/water reaction will govern. However, hydrogen is continued to be generated after the initial metal/water reaction due to the radiolytic decomposition of water and is shown as a function of time after loss of coolant accident in Figure 1.

Calculation under Code of Federal Regulations, Title 10, Part 50, Section 50.46 (4) should result in a reaction of less than 0.2 percent of the cladding mass. When this mass is multiplied by five (3) the resultant mass of reaction should not exceed one percent. Verification of the assumption of a one percent reaction for this analysis will be provided in the Final Safety Analysis Report, when calculations under Code of Federal Regulations, Title 10, Part 50, Section 50.46 are available.

The analysis, the results of which are summarized below, determines that one one-hundred standard cubic foot per minute hydrogen recombiner, started within twenty-four hours after loss of coolant accident will hold the hydrogen concentration below four percent. The calculations analyze hydrogen generation rate and recombiner performance and tabulate the gaseous composition of the containment versus time.

3.0 SUMMARY OF ANALYSIS

Parameters

The calculations were based upon all parameters in Regulatory Guide 1.7 (1) as well as the specific project parameters listed below:

1.	Reactor power (105 percent of rated)	3489 Mwt
2.	Drywell volume	340,900 ft^3

3.	Suppression chamber volume (at high water level)	200,600 ft^3
4.	Initial drywell pressure	15.45 psia
5.	Initial drywell temperature	135 F
6.	Initial drywell dew point	80 F
7.	Initial suppression chamber pressure	15.45 psia
8.	Initial suppression chamber temperature	90 F
9	Initial suppression chamber dew point	90 F
10.	Weight of zirconium in outermost 23 mils of active fuel cladding (not including plenum space).	56,486 lbs

Assumptions

1. Recombiner draws from the drywell and returns hydrogen-free gas to the suppression chamber.

2. Water vapor partial pressure is neglected in the calculation of percent volume of H_2, O_2, N_2 (percent volume dry).

3. Containment atmosphere is 21 percent oxygen initially if non-inerted and four percent initially if inerted.

Results

1. 100 scfm recombiner; noninerted containment.

	Composition at start (24 hrs)		Composition at peak H2 conc. (56 hrs)	
	Volume percent		Volume percent	
	Dry	Wet	Dry	Wet
H_2	3.2	2.7	3.9	3.5
O_2	21.0	17.8	21.3	18.8
N_2	75.8	64.3	74.8	65.8
H_2O	—	15.2	—	11.9

See Figures 2 and 3.

2. 100 scfm recombiner; inerted containment.

	Composition at start (24 hrs)		Composition at peak H_2 Conc. (56 hrs)	
	Volume percent		Volume percent	
	Dry	Wet	Dry	Wet
H_2	3.2	2.7	3.9	3.5
O_2	4.7	4.0	5.2	4.6
N_2	92.1	78.1	90.9	79.9
H_2O	--	15.2	--	12.0

3. A purge system adding air to noninerted containment.

   Start - 40 hrs after LOCA
   Maximum pressure - 38 psig @ 18 days
   Maximum addition rate - 84 scfm
   Total air added - 960,000 SCF

   See Figures 4 and 5.

The above results indicate a margin of conservatism in the control of hydrogen:

1. The analysis was performed to limit the hydrogen concentration to below four volume percent, calculated neglecting water vapor. If the percent volume is computed considering water vapor, the hydrogen concentration is held below 3.5 percent.

2. If the recombiners were started before 24 hours, a lower maximum value of hydrogen concentration would result.

3. The flow rate through the recombiner could be increased to reduce hydrogen concentration further.

   The effect of various recombiner capacities can be determined for both the inerted and noninerted cases. The results show that the hydrogen concentration versus time is unaffected by whether or not the containment is initially inerted. For the noninerted case, Figures 2 and 3 plot hydrogen concentration versus time after a loss of coolant accident with and without the recombiner system. Additionally, the effect of a

nonsafety-related air purge system as a backup to the recombiner system was analyzed, and it was determined that the hydrogen concentration could be held below four percent if purging is begun within forty hours after a loss of coolant accident. Figure 4 shows the air addition rate versus time necessary to maintain the hydrogen concentration below four percent.

4.0 CONCLUSIONS

The analysis indicates that the requirements of Branch Technical Position Section 7.d. [3] and Regulatory Guide 1.7 specifying the criteria for hydrogen generation and the allowable limits of hydrogen concentration, are met by employing a redundant hydrogen recombiner and a noninerted atmosphere and by monitoring and controlling hydrogen concentration while allowing the oxygen level to remain at normal air concentration. Elimination of the need to control oxygen results in less equipment and increased system reliability. The noninerted containment eliminates the potential safety hazards associated with the presence of residual nitrogen when the containment is entered.

Footnotes

(1) Nuclear Regulatory Commission, Regulatory Guide 1.7 (Safety Guide 7), March, 1971

(2) Nuclear Regulatory Commission, Regulatory Guide 1.7 "Control of Combusitble Gas Concentration in Containment Following a Loss of Coolant Accident" - Draft 2, August, 1974

(3) Nuclear Regulatory Commission, Branch Technical Position CSB 6-2, "Control of Combustible Gas Concentrations in Containment following a Loss of Coolant Accident", March, 1975

(4) Code of Federal Regulations, Title 10, Part 50, Section 50.46, "Acceptance Criteria for Emergency Core Cooling System for Light Water Nuclear Power Reactors."

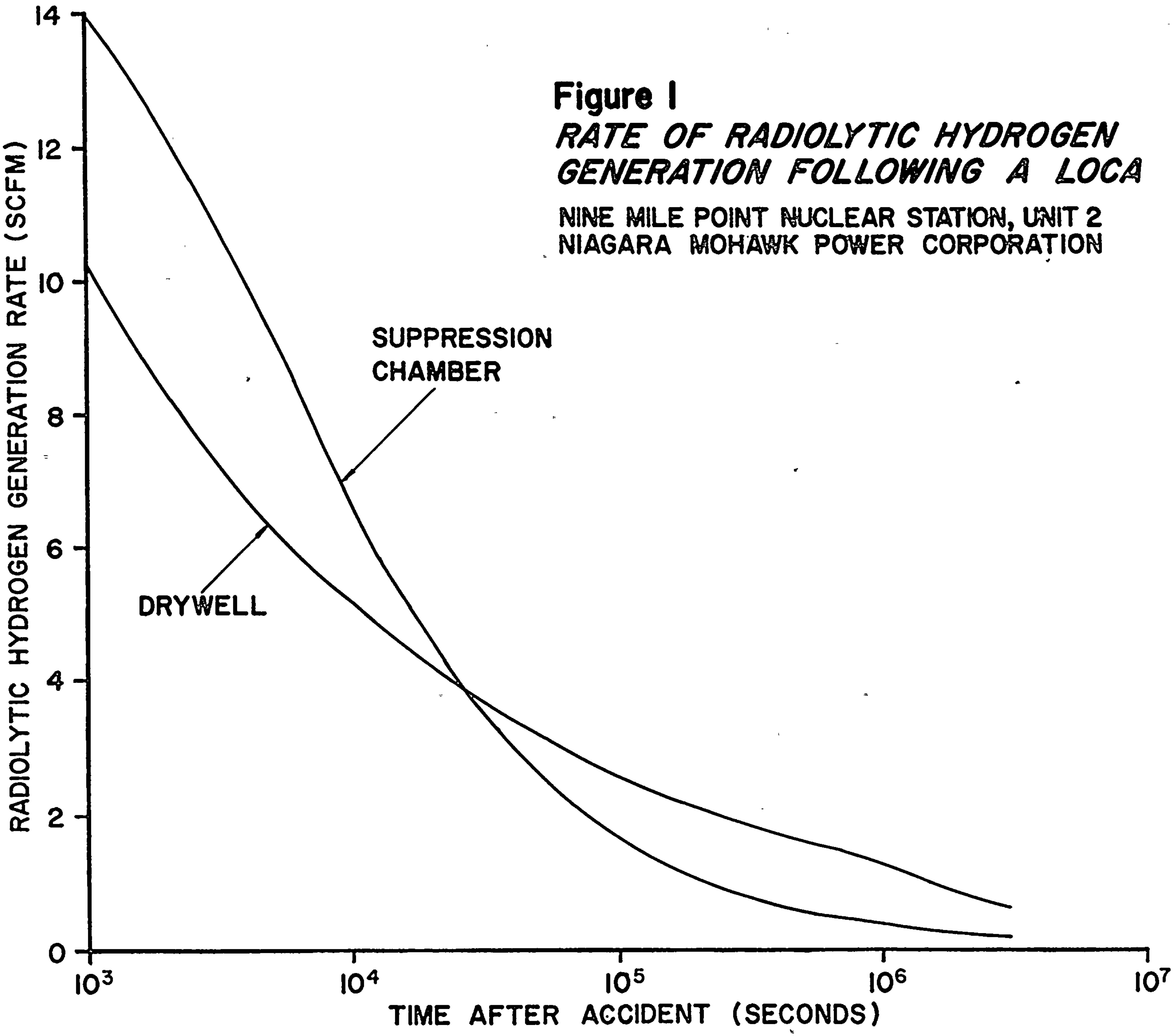


Figure I
RATE OF RADIOLYTIC HYDROGEN GENERATION FOLLOWING A LOCA
NINE MILE POINT NUCLEAR STATION, UNIT 2
NIAGARA MOHAWK POWER CORPORATION
SUPPRESSION CHAMBER
DRYWELL
RADIOLYTIC HYDROGEN GENERATION RATE (SCFM)
14
12
10
8
6
4
2
0
10³
10⁴
10⁵
10⁶
10⁷
TIME AFTER ACCIDENT (SECONDS)

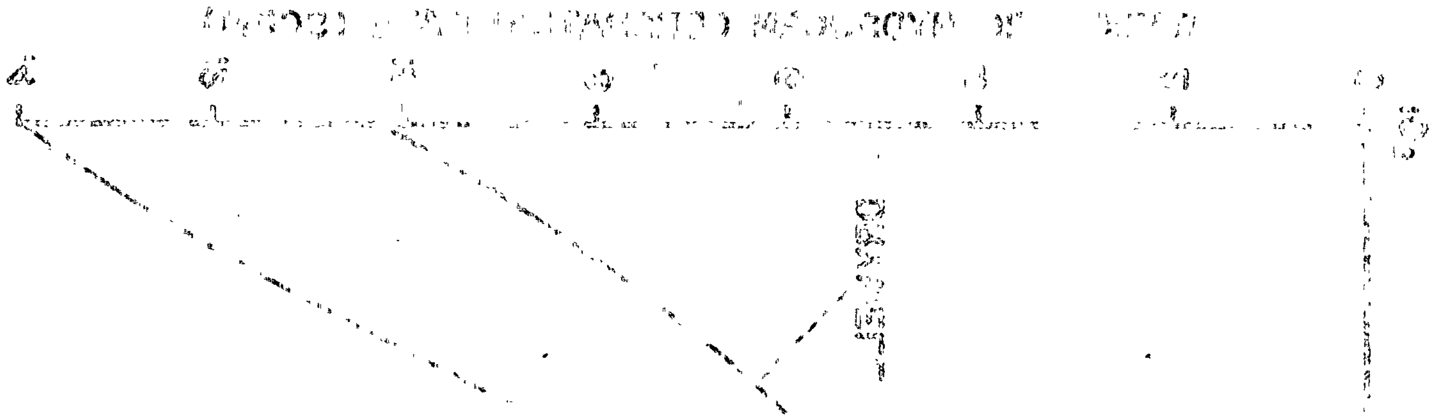


Figure 2

HYDROGEN CONCENTRATION IN PRIMARY CONTAINMENT FOLLOWING A LOCA–SHORT TERM WITH 100 SCFM RECOMBINER

NINE MILE POINT NUCLEAR STATION, UNIT 2
NIAGARA MOHAWK POWER CORPORATION

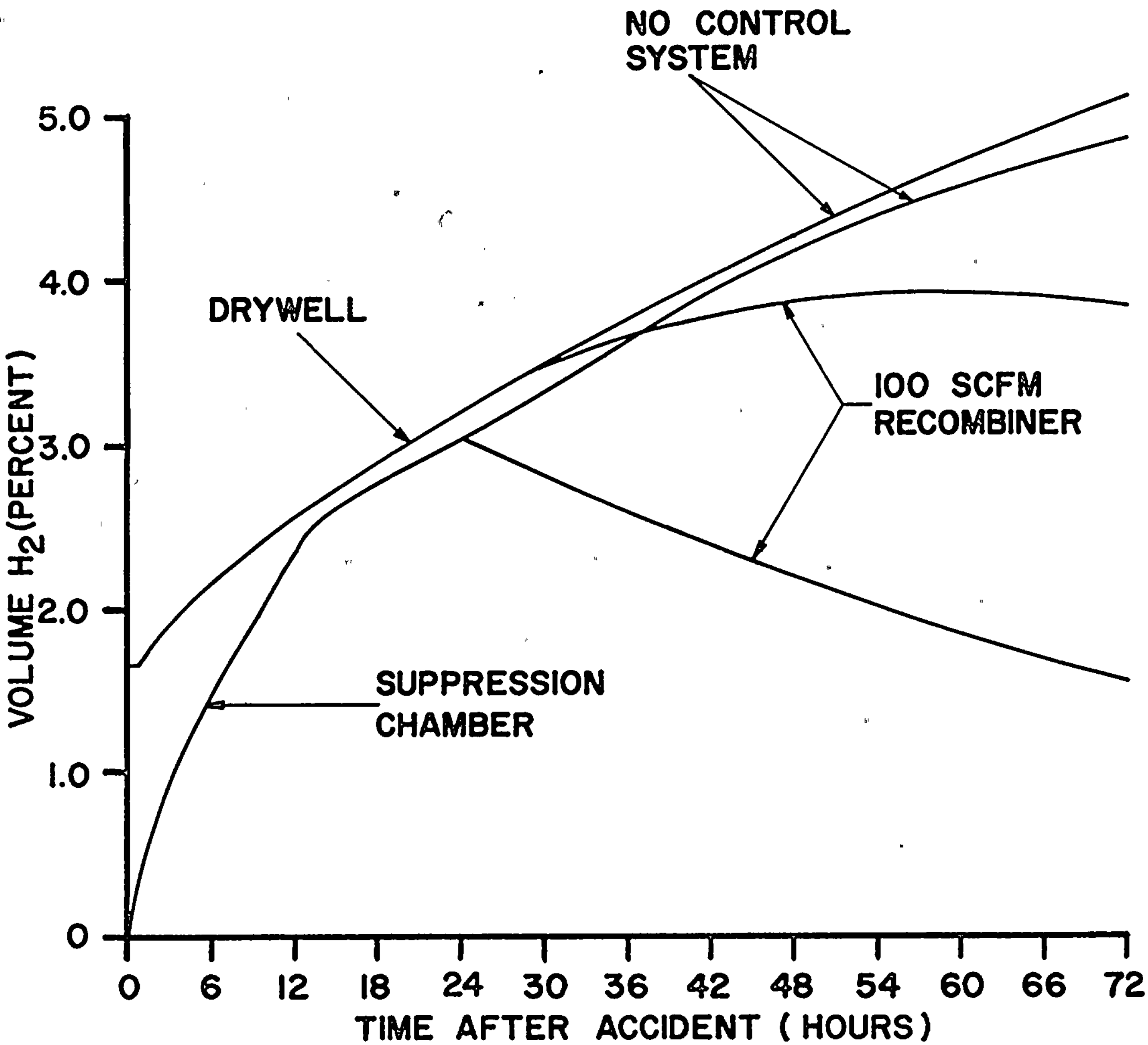
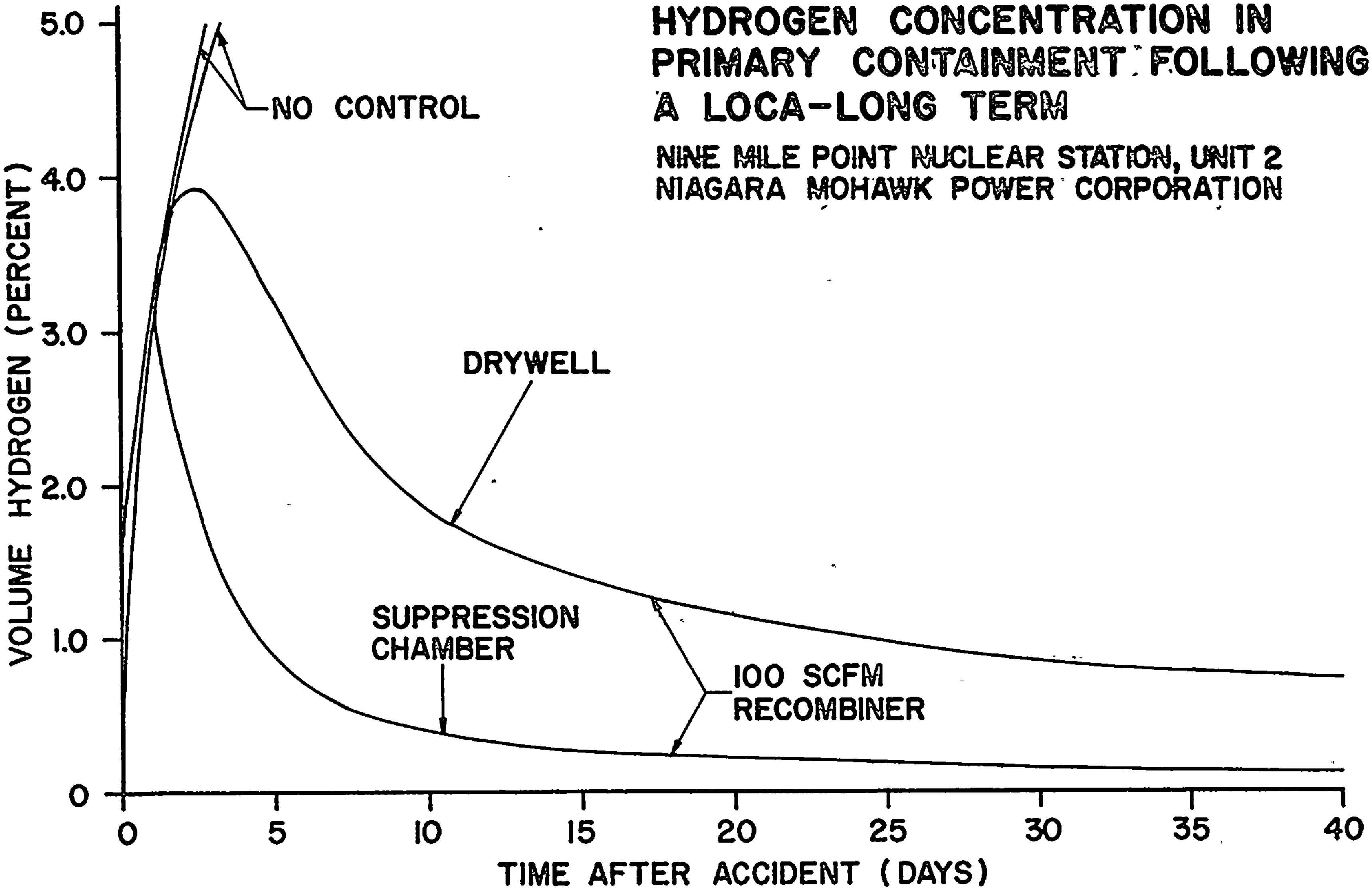


Figure 3

HYDROGEN CONCENTRATION IN PRIMARY CONTAINMENT FOLLOWING A LOCA-LONG TERM

NINE MILE POINT NUCLEAR STATION, UNIT 2
NIAGARA MOHAWK POWER CORPORATION



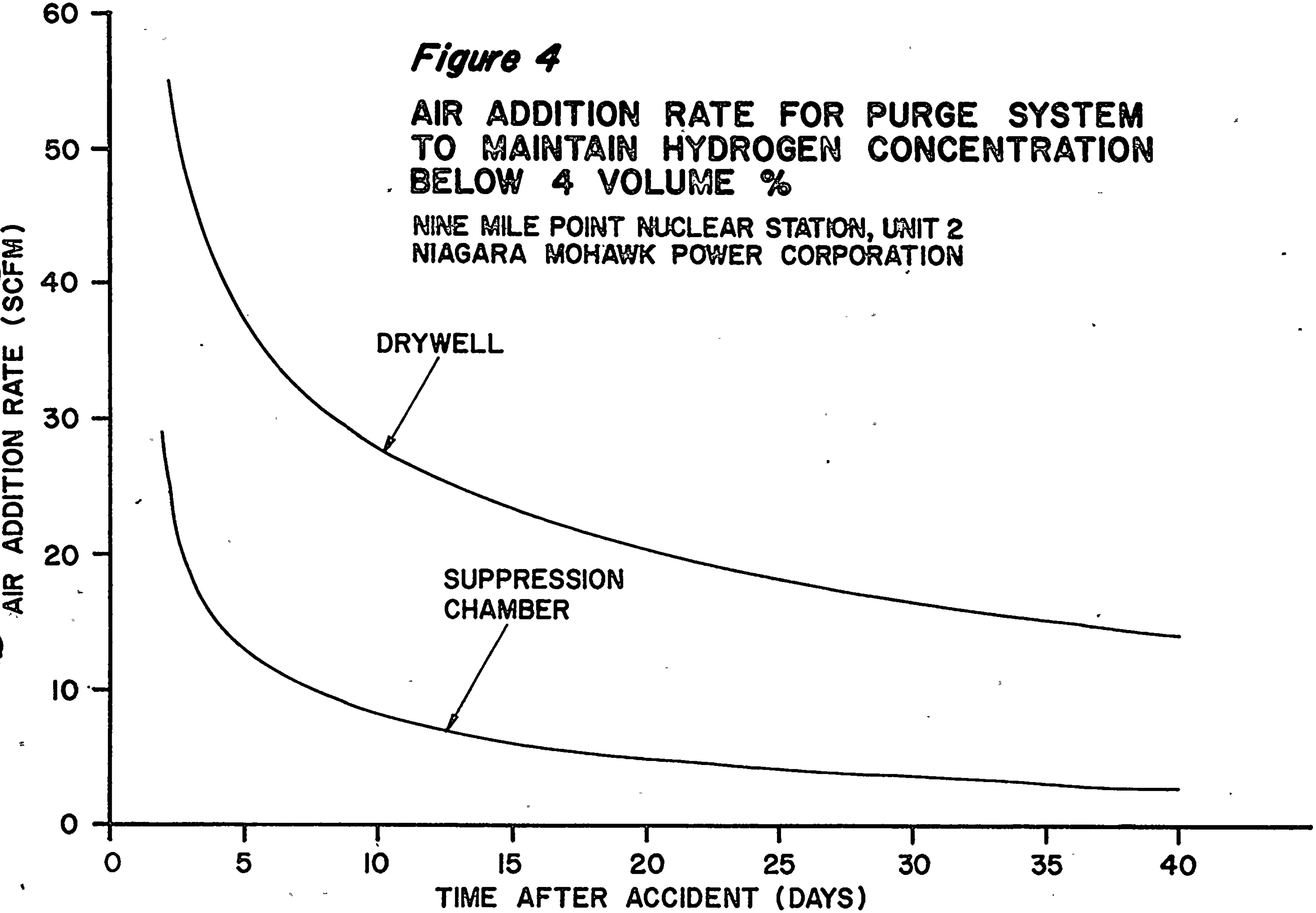


Figure 4
AIR ADDITION RATE FOR PURGE SYSTEM
TO MAINTAIN HYDROGEN CONCENTRATION
BELOW 4 VOLUME %
NINE MILE POINT NUCLEAR STATION, UNIT 2
NIAGARA MOHAWK POWER CORPORATION
AIR ADDITION RATE (SCFM)
60
50
40
30
20
10
0
DRYWELL
SUPPRESSION
CHAMBER
0
5
10
15
20
25
30
35
40
TIME AFTER ACCIDENT (DAYS)

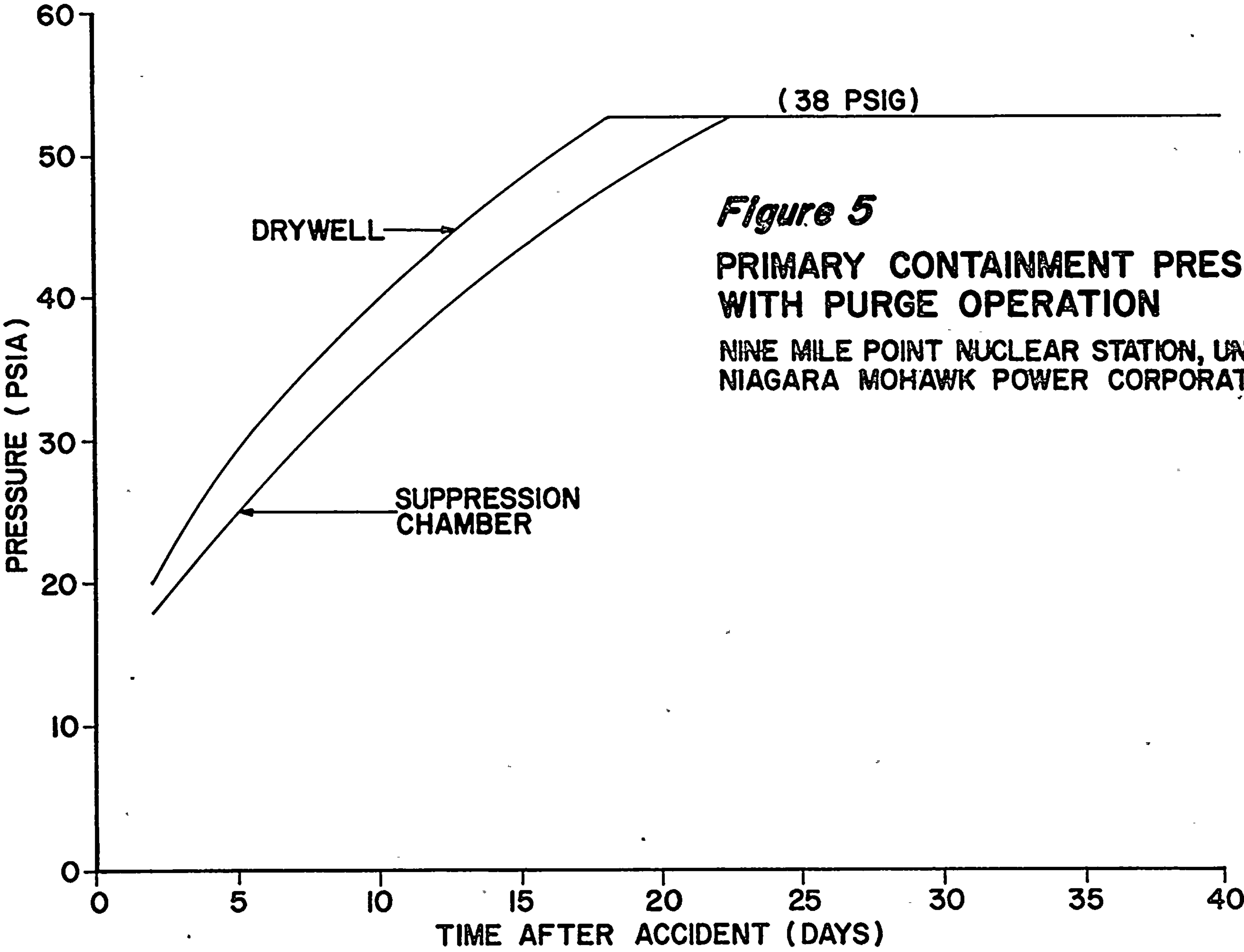


Figure 5

PRIMARY CONTAINMENT PRESSURE WITH PURGE OPERATION

NINE MILE POINT NUCLEAR STATION, UNIT 2
NIAGARA MOHAWK POWER CORPORATION