UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

May 7, 1991

Docket No. 50-220

LICENSEE: Niagara Mohawk Power Corporation

FACILITY: Nine Mile Point Nuclear Station Unit No. 1

SUBJECT: MEETING MINUTES REGARDING THE APRIL 24, 1991, MEETING TO DISCUSS THINNING OF THE NINE MILE POINT NUCLEAR STATION UNIT 1 TORUS WALL MATERIAL

A meeting was held in the NRC One White Flint North Office in Rockville, Maryland, with Niagara Mohawk Power Corporation (NMPC) and NRC staff representatives to discuss thinning of the Nine Mile Point Unit 1 torus wall material. The licensee had requested this meeting. Enclosure 1 is a list of the meeting attendees. The handout material used by the licensee during the meeting is attached as Enclosure 2.

During the meeting the licensee stated that it will be making a submittal by May 15, 1991, to justify continued use of the torus (without modifications) until at least the 1994 refueling outage rather than making modifications during the 1992 refueling outage as originally committed to by the licensee. The licensee requested that, if possible, the NRC staff complete its review of this submittal by December 1991, so that the licensee may make appropriate scheduling plans for the 1992 refueling outage. The staff agreed to begin reviewing the submittal as soon as possible but did not commit to completing the review by December 1991.

Prior to the meeting, the NRC staff had requested that the licensee include in its presentation, discussions regarding the overall performance capabilities of the containment to withstand severe accidents as well as a discussion of the capability of the torus to continue meeting its minimum wall thickness requirements. The licensee included discussions of the requested topics; however, the NRC staff requested the following additional information be submitted.

1. Detailed drawings of the bellows connection to the vent line.

2. Details of the torus saddle connections that were added in the early 1980's.

3. Details of the drywell head section, including bolting used for head closure.

4. Details of penetrations in the cylindrical section of the drywell.

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May 7, 1991

5. Details of stiffeners installed on downcomers to prevent movement of the downcomers.

6. Individual data point (65 points in 1' x 3' area) results of 6-month measurements of torus wall thickness.

7. Coating study reference material.

8. Consideration of a surveillance program to monitor water leakage from the torus.

9. Various modification options, schedules, and resource impacts that Niagara Mohawk has considered. These options, schedules, and resource impacts should consider contingencies for staff review of the May 15, 1991, submittal not being completed by December 1991.

The licensee agreed to supply the requested additional information. This additional information will either be included with the May 15, 1991, submittal, if possible, or will be provided later as an additional submittal.

Donald S. Brinkman

Donald S. Brinkman, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. List of Attendees
2. Licensee Handout Material

cc w/enclosures:
See next page

Mr. B. Ralph Sylvia
Niagara Mohawk Power Corporation

Nine Mile Point Nuclear Station
Unit No. 1

cc:

Mr. Mark J. Wetterhahn, Esquire
Winston & Strawn
1400 L Street, NW.
Washington, D.C. 20005-3502

Supervisor
Town of Scriba
R. D. #4
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Vice President - Nuclear Generation
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Resident Inspector
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Regional Administrator, Region I
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Mr. Kim Dahlberg
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Charlie Donaldson, Esquire
Assistant Attorney General
New York Department of Law
120 Broadway
New York, New York 10271

Mr. Paul D. Eddy
State of New York
Department of Public Service
Power Division, System Operations
3 Empire State Plaza
Albany, New York 12223

ENCLOSURE 1

ATTENDANCE LIST

April 24, 1991 Meeting to Discuss Thinning of Nine Mile Point Unit 1 Torus Wall Material

Name	Position	Organization
Donald S. Brinkman	Senior Project Manager	NRC/PDI-1
Robert A. Capra	Director, PDI-1	NRC/PDI-1
Goutam Bagchi	Branch Chief	NRC/DET
Alan J. Bilanin	Senior Associate	Continuum Dynamics, Inc.
Richard H. Berks	Senior Engineer	Teledyne/NMPC
Andre Drozd	Engineer	NRR/SPLB
Ralph Architzel	Branch Chief (Act)	NRR/SPLB
George Johnson	Section Leader	NRR/DET/EMCB
Nick Spagnoletti	Licensing	Niagara Mohawk
Mark Wetterhahn	Attorney	Winston & Strawn
Gary Wilson	Managing Attorney	Niagara Mohawk
Jose A. Calvo	AD Region I	NRC/NRR
Carl Terry	VP Nuclear Engineer	NMPC
R. F. Oleck	Mgr. Des Basis Recon-Unit 1	NMPC
K. D. Samuelson	Unit 1 Proj. Manager	NMPC
P. B. George	Supr/Civil/Stru Design	NMPC
Gill Yaeger	Mgr of Engineering-NMP1	NMPC
Larry McNeer	Sr. Nuclear Engineer-NMP1	NMPC
Paul Czaya	Licensing Engineer	GPU Nuclear
A. Dromerick	Proj. Manager	NRR-PDI-4
Chen P. Tan	Civil Engineer	NRR/DET
Patrick Harris	Licensing Staff Engineer	Search Lic/Bechtel
Frank J. Witt	Chemical Engineer	NRR/DET/EMCB
S. Lee	Materials Engineer	NRR/DAR/PDLR
Deborah Jackson	Mechanical Engineer	NRR/DAR/PDLR
C. Y. Cheng	Chief	NRR/DET/EMCB
Stephen Koscielny	Chemical Engineer	NRR/DET/EMCB
E. Harold Gray	Sect. Chief - Materials	Region I

ENCLOSURE 2

NIAGARA MOHAWK POWER CORPORATION

TORUS PRESENTATION
AGENDA

APRIL 24, 1991

I. INTRODUCTION/PURPOSE - C. D. Terry

II. CONTAINMENT OVERVIEW - R. F. Oleck

III. STATUS OF CORROSION - L. M. McNeer

IV. STRUCTURAL INSPECTIONS - L. M. McNeer

V. VENTING RELATED ISSUES - L. M. McNeer

VI. REEXAMINATION OF TORUS SHELL CO LOAD - P. B. George

VII. CLOSING - C. D. Terry

LUNCH

VIII. NRC FEEDBACK

NIAGARA MOHAWK

NINE MILE POINT UNIT # 1

CONTAINMENT OVERVIEW

- DESCRIPTION OF DRYWELL/TORUS

- ORIGINAL DESIGN BASIS
  - Design Codes
  - Materials
  - Peak Pressures & Temperatures
  - Stress Analysis & Results

- CONTAINMENT ULTIMATE STRENGTH
  - CB&I Studies (BWROG)
  - Venting (L. McNeer)

- MARK I PROGRAM
  - Decoupling short-term hydrodynamics
  - Governing pool dynamic loads
  - Summary of Mark I modifications

- CONTAINMENT INSPECTION HISTORY
  - Torus inspections from 1975
  - 1979 RCT Torus Corrosion Estimate
  - RAP Torus Issue & Results
  - 1986 Drywell Sand Cushion Inspection

- PRESENT PROGRAM = L. McNEER

DRYWELL AND SUPPRESSION CHAMBER

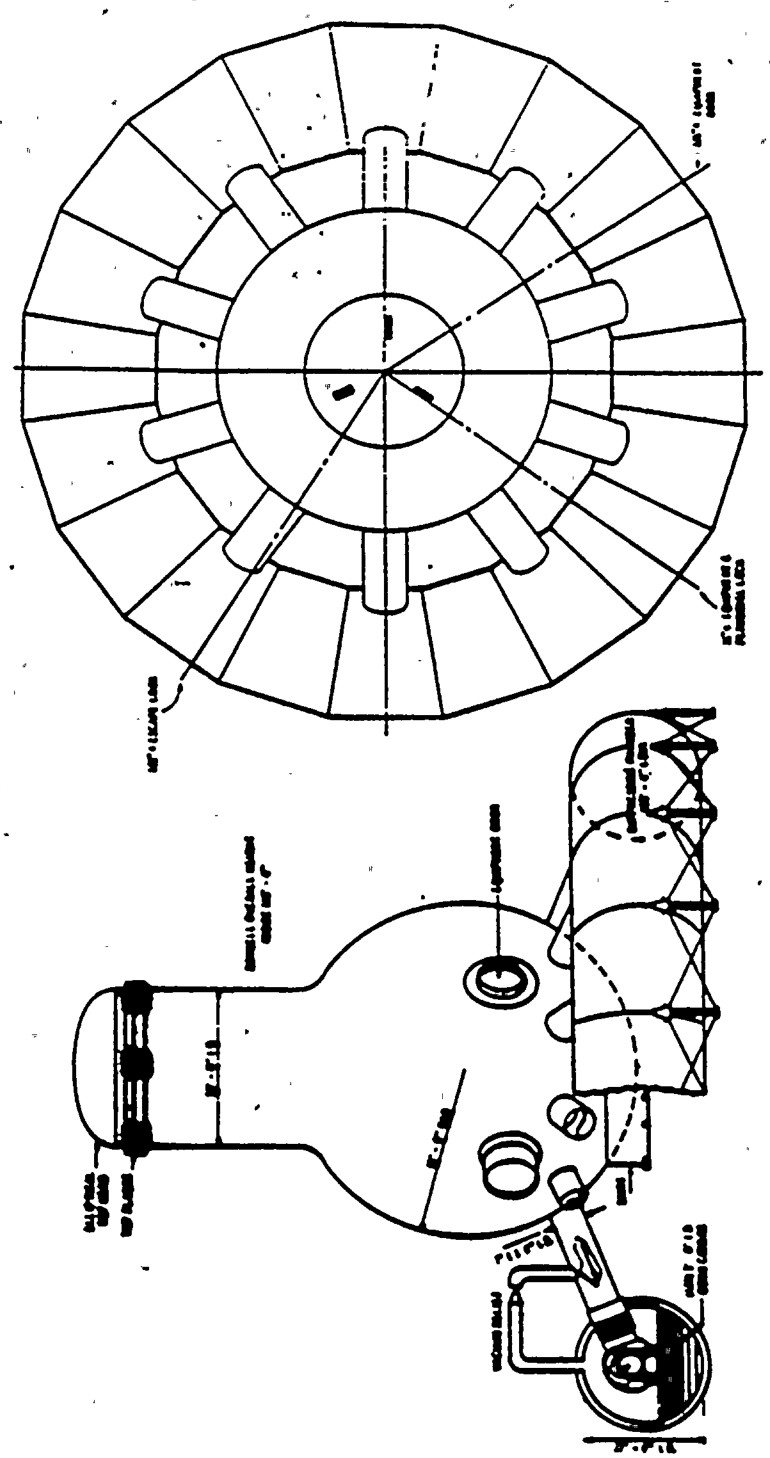
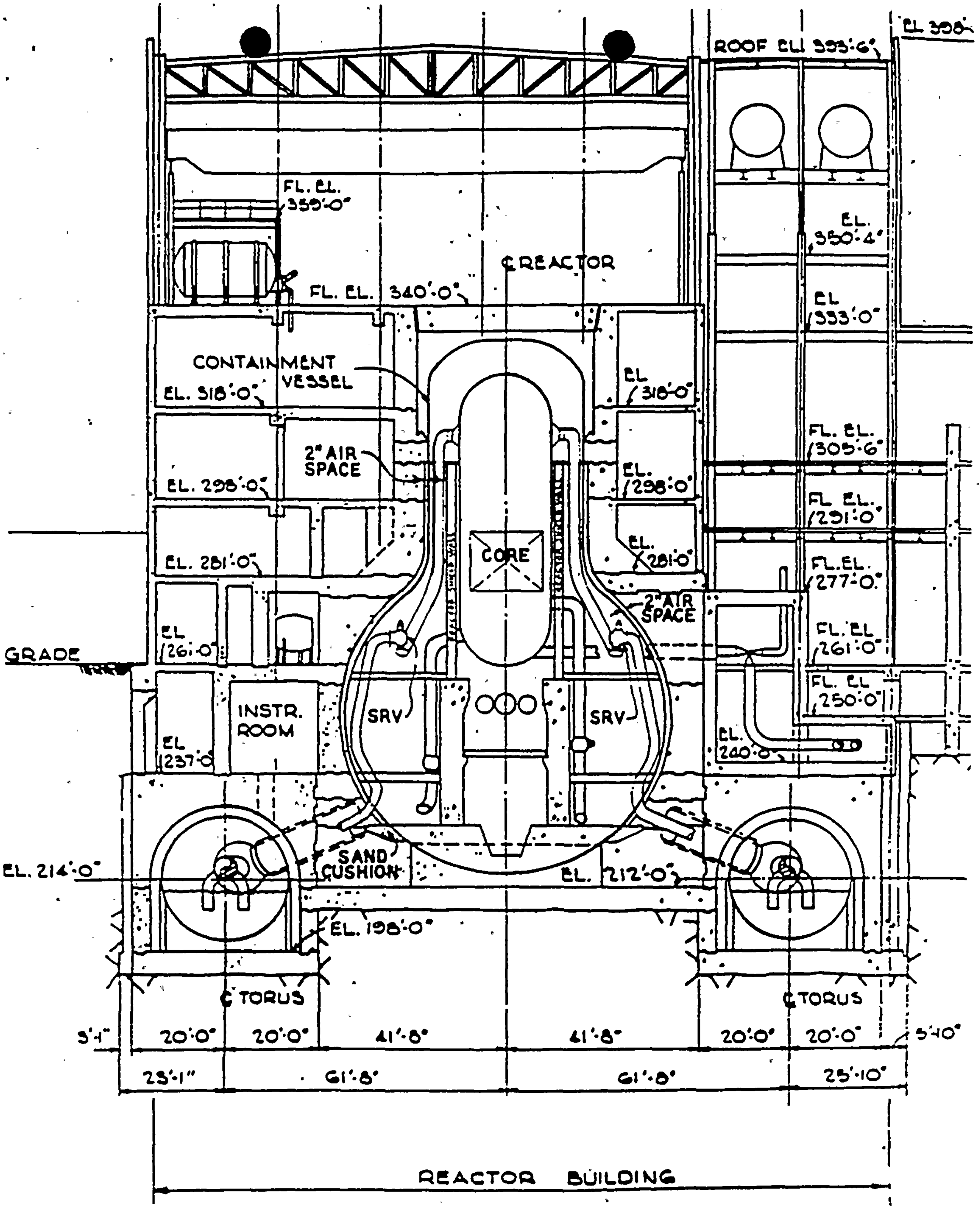


FIGURE V



EL 396
ROOF EL 393'-6"
FL. EL. 359'-0"
EL. 350'-4"
℄ REACTOR
FL. EL. 340'-0"
EL 333'-0"
CONTAINMENT VESSEL
EL. 318'-0"
EL 318'-0"
FL. EL. 305'-6"
2" AIR SPACE
EL. 298'-0"
EL. 298'-0"
FL. EL. 291'-0"
CORE
EL. 281'-0"
EL. 281'-0"
FL. EL. 277'-0"
2" AIR SPACE
EL 261'-0"
FL. EL 261'-0"
GRADE
FL. EL 250'-0"
INSTR. ROOM
SRV
SRV
EL 237'-0"
EL. 240'-0
SAND CUSHION
EL. 214'-0"
EL. 212'-0"
EL. 198'-0"
℄ TORUS
℄ TORUS
3'-1"
20'-0"
20'-0"
41'-8"
41'-8"
20'-0"
20'-0"
5'-0"
23'-1"
61'-8"
61'-8"
25'-10"
REACTOR BUILDING

ORIGINAL DESIGN BASIS

- DESIGN & CONSTRUCTION CODES
  - ASME SECTION III, CLASS B (Version prior to 1964)

- MATERIALS OF CONSTRUCTION

Drywell:	A212 Gr. B FBX to A-300 (Same Ultimate as SA516 Gr. 70)
Torus:	A-201 Gr. B FBX to A-300 (Comparable to SA516 Gr. 60)

NINE MILE #1 CONTAINMENT DESIGN BASIS

	Drywell and Vents	Suppression Chamber
Total Volume (No Equipment)	242,700 cu ft	209,000 cu ft
Approximate Free Volume	180,000 cu ft	120,000 cu ft
Internal Design Pressure	62 psig	35 psig
Internal Design Temperature (Maximum)	310F	205F
Design Leakage Rate at Design Pressure	0.5 w/o* per day	0.5 w/o* per day
External Design Pressure	2 psig	1 psig

* Weight percent

LOSS OF COOLANT ACCIDENT
SUPPRESSION CHAMBER PRESSURE
STRETCH POWER

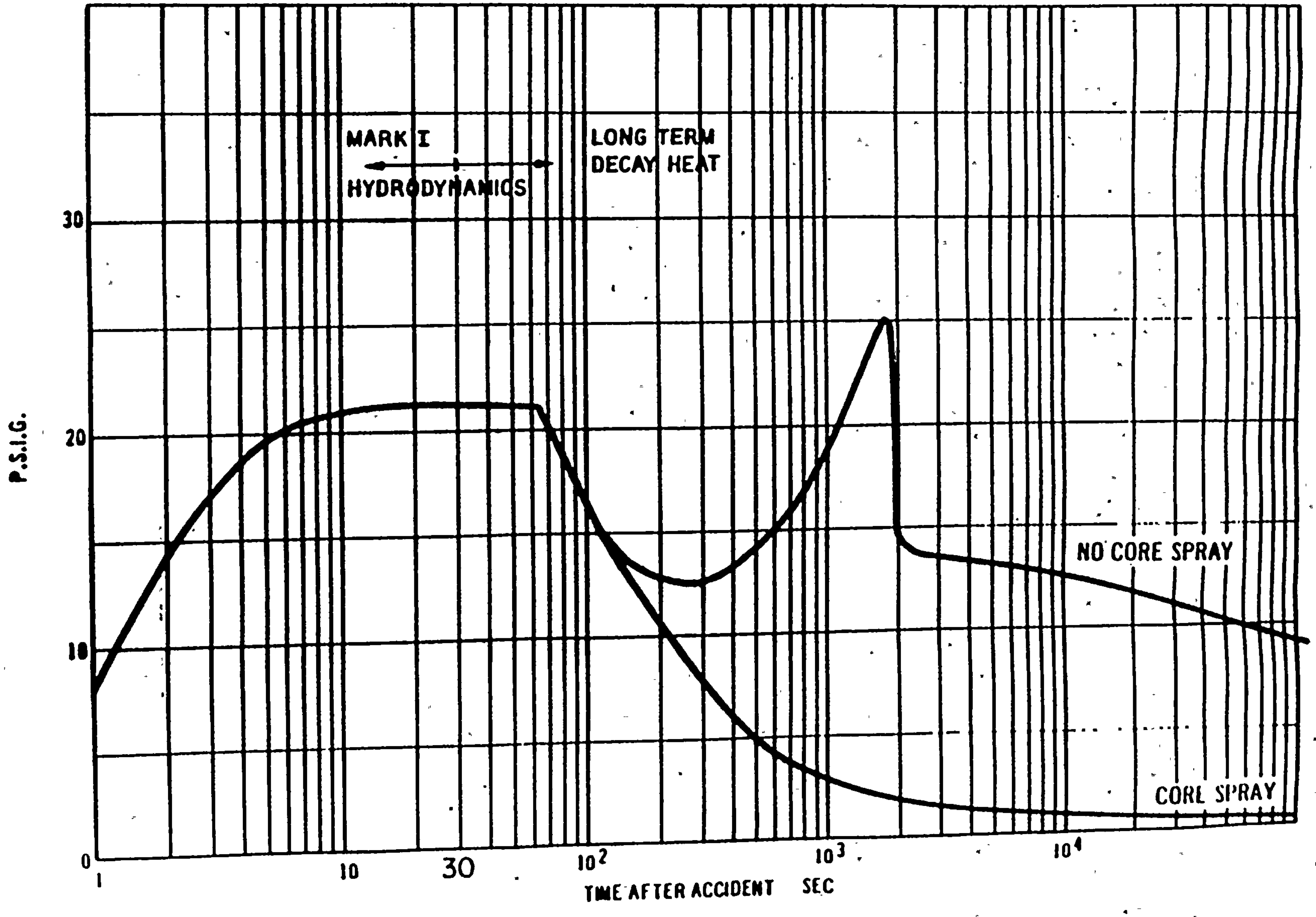


FIGURE E-3

LOSS OF COOLANT ACCIDENT
DRYWELL PRESSURE
STRETCH POWER

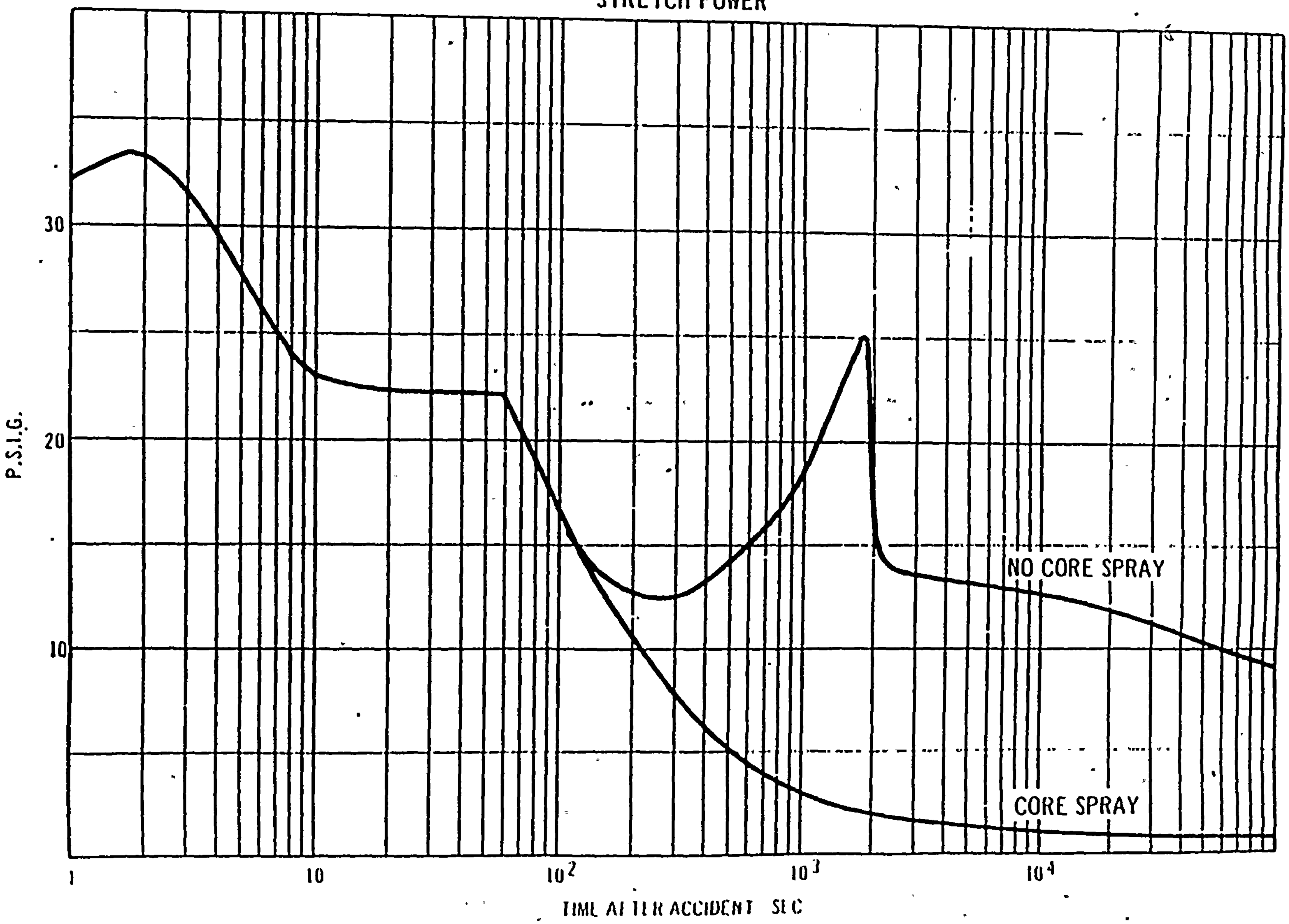


FIGURE [illegible]

LOSS OF COOLANT ACCIDENT
CONTAINMENT TEMPERATURE – WITH CORE SPRAY
STRETCH POWER

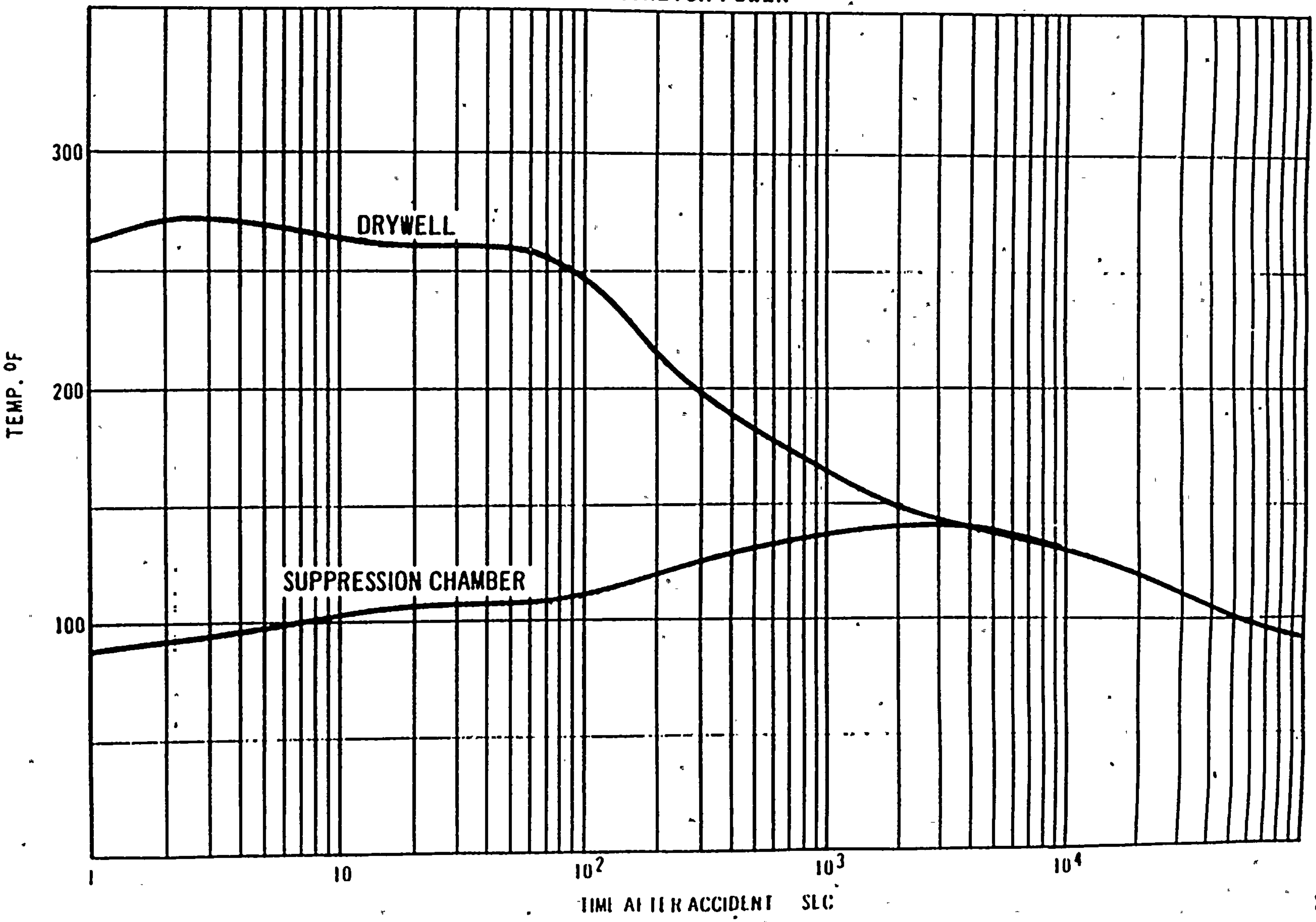
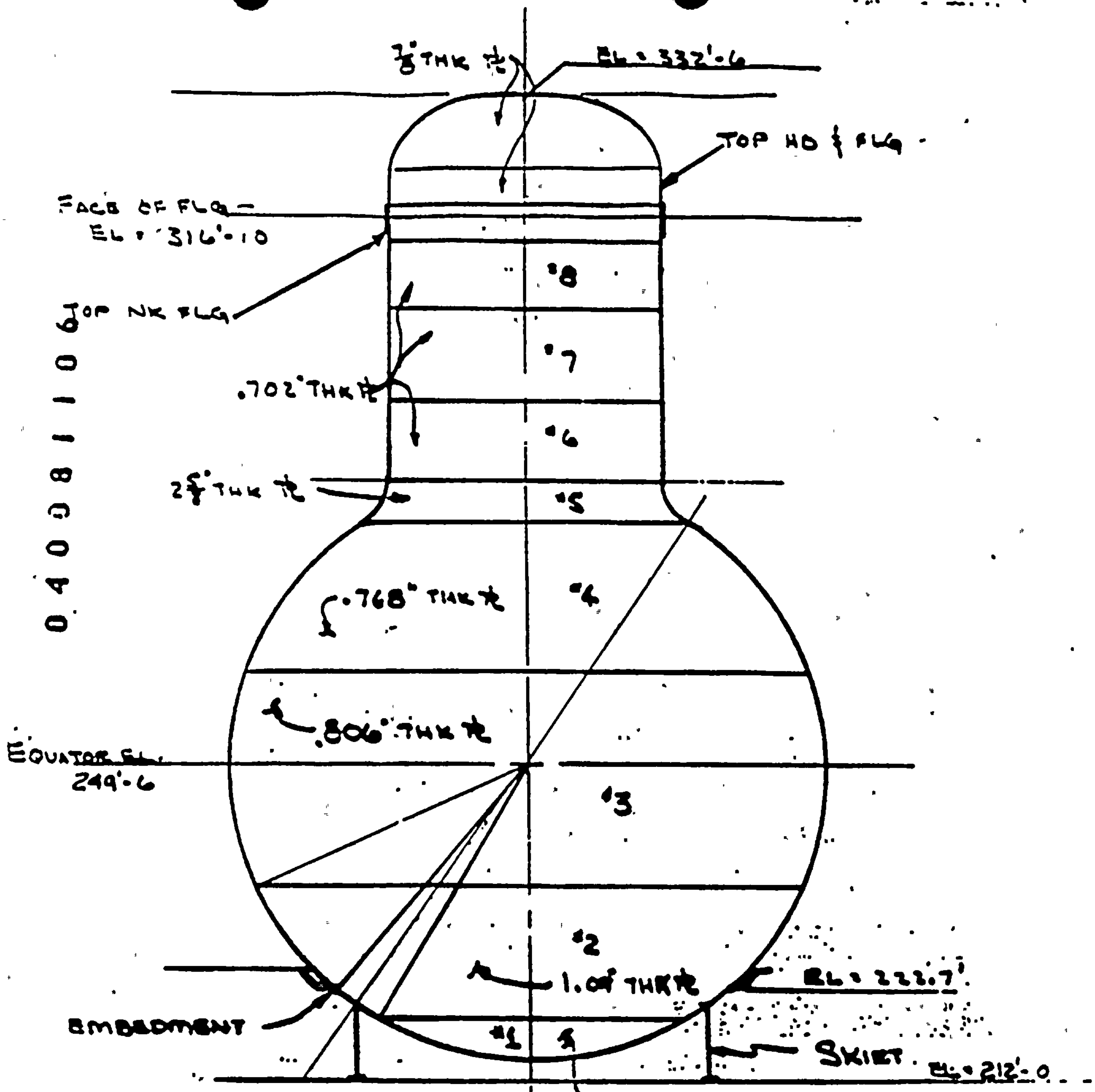


FIGURE III-33



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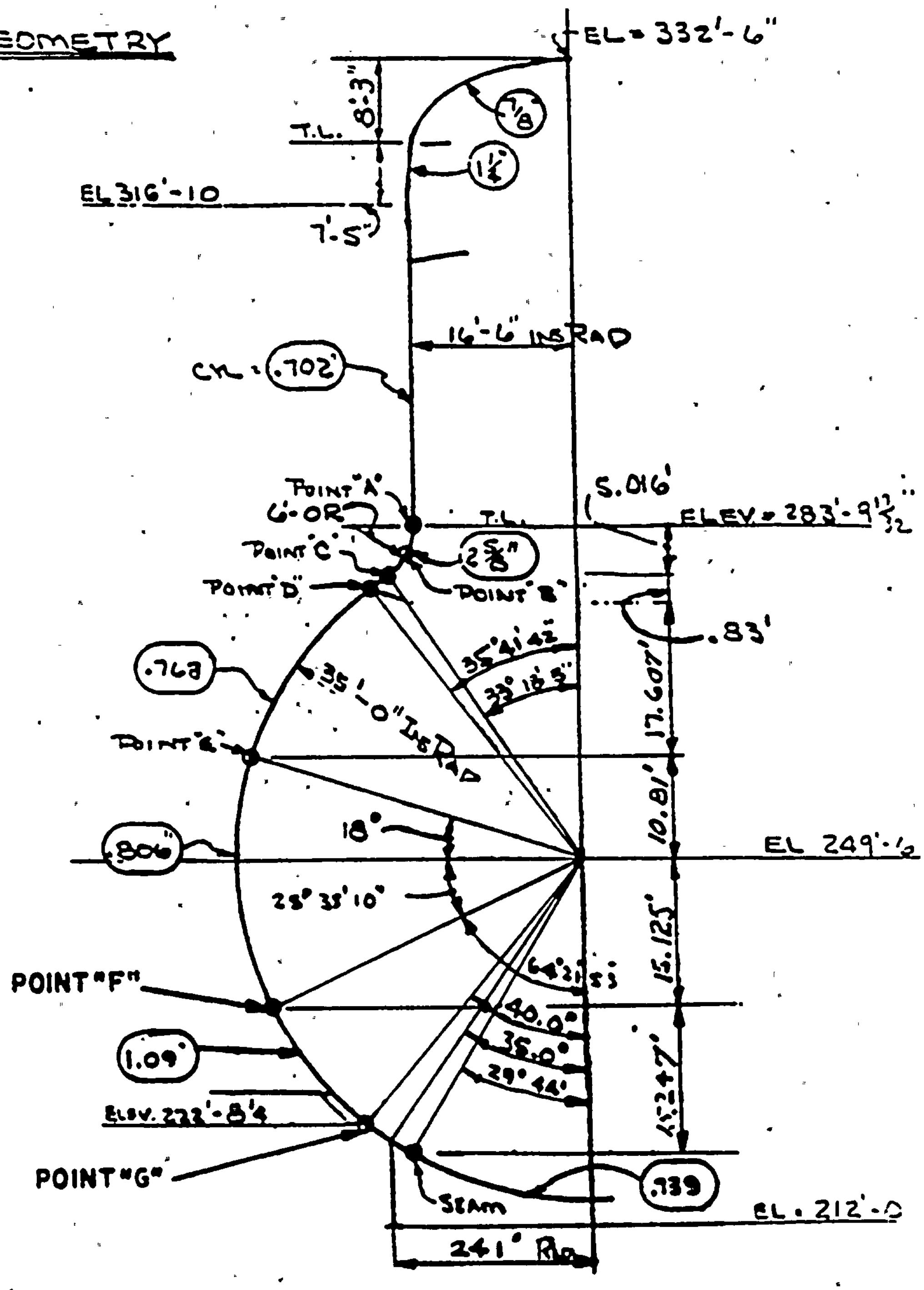
1 11/17/64

2 2/22/68

SECTION OF DRYWELL

DRYWELL SHELL DESIGN ... A1

DRYWELL GEOMETRY



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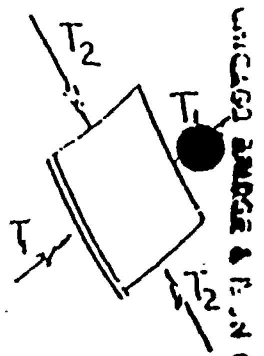
ALL EARTHQUAKE STRESSES ARE FOR "SSE" CASE

Subject SUMMARY STRESSES POINT "F" Cont. 9-1370 Date 3/2/75 By LFR

	CASE 1 T1	CASE 1 T2	CASE 2 T1	CASE 2 T2	CASE 3 T1	CASE 3 T2	CASE 4 T1	CASE 4 T2	CASE 5 (CASE 1) T1	CASE 5 (CASE 1) T2	CASE 6 (CASE 2) T1	CASE 6 (CASE 2) T2
STEEL DEAD LOAD	+497	-457	+497	-457	+497	-457	+497	-457	+497	-457	+497	-457
INTERNAL PRESSURE	—	—	—	—	—	—	—	—	+13020	+13020	+13020	+13020
EXTERNAL PRESSURE	-420	-420	-420	-420	-420	-420	-420	-420	—	—	—	—
WATER SEAL	—	—	—	—	+278	-278	+278	-278	—	—	—	—
LIVE LOADS	+417	-417	+417	-417	+417	-417	+417	-417	+417	-417	+417	-417
AIR												
HORZ EQ STEEL	—	—	∓353	±353	—	—	—	—	—	—	∓353	±353
VERT EQ STEEL	—	—	∓25	±25	—	—	—	—	—	—	∓25	±25
HORZ EQ STL (REFUEL)	—	—	—	—	—	—	∓271	±271	—	—	—	—
VERT EQ STL (REFUEL)	—	—	—	—	—	—	∓22	±22	—	—	—	—
VERT EQ WATER SEAL	—	—	—	—	—	—	∓18	±18	—	—	—	—
HORZ EQ LIVE LOADS	—	—	∓101	±101	—	—	∓101	±101	—	—	∓101	±101
VERT EQ LIVE LOADS	—	—	∓23	±23	—	—	∓23	±23	—	—	∓23	±23
TOTAL (#/in)	+494	-1294	-8 +996	-792 -1796	+772	-1572	+337 1207	-1137 -2007	13934	12146	14436	11644
EFFICIENCY	—	.546	— .338	.755	—	.664	—	.847	—	—	—	—
STRESS (#/in²)	+664	-1740	-11 +1340	-1065 -2416	+1038	-2114	453 1623	-1529 -2699	18741	16336	19416	15661

• ALLOW. TENSILE STRESS = 38000 PSI



(+) IS TENSION

SEE SHEET 182 FOR ALLOWABLE STRESSES

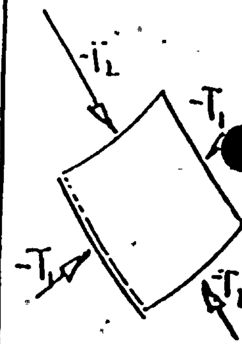
t = .7435"

Subject SUMMARY STRESSES TANK G — Contract 9-1570

ALL EARTHQUAKE STRESSES ARE FOR SSE CASE

	CASE 1 T1	CASE 1 T2	CASE 2 T1	CASE 2 T2	CASE 3 T1	CASE 3 T2	CASE 4 T1	CASE 4 T2	CASE 5 T1	CASE 5 T2	CASE 6 T1	CASE 6 T2
DEAD LOAD	+1201	-1202	+1201	-1202	+1201	-1202	+1201	-1202	+1201	-1202	+1201	-1202
INTERNAL PRESSURE	—	—	—	—	—	—	—	—	+13020	+13020	+13020	+13020
EXTERNAL PRESSURE	-420	-420	-420	-420	-420	-420	-420	-420	—	—	—	—
WATER SEAL	—	—	—	—	+547	-547	+547	-547	—	—	—	—
LIVE LOADS	+821	-821	+821	-821	+821	-821	+821	-821	+821	-821	+821	-821
AIR	—	—	—	—	—	—	—	—	+15	-15	+15	-15
HORZ EQ STEEL	—	—	∓1307	±1307	—	—	—	—	—	—	∓1307	±1307
VERT EQ STEEL	—	—	∓66	±66	—	—	—	—	—	—	∓66	±66
HORZ EQ STL (REFUEL)	—	—	—	—	—	—	∓1052	±1052	—	—	—	—
VERT EQ STL (REFUEL)	—	—	—	—	—	—	∓61	±61	—	—	—	—
VERT EQ WATER SEAL	—	—	—	—	—	—	∓35	±35	—	—	—	—
HORZ EQ LIVE LOADS	—	—	∓535	±535	—	—	∓535	±535	—	—	∓535	±535
VERT EQ LIVE LOADS	—	—	∓45	±45	—	—	∓45	±45	—	—	∓45	±45
TOTAL (#/in)	+1702	-2444	-251 +3655	-411 -4297	+2249	-2991	+521 +3977	-1263 -4719	+15157	+10981	+17110	+9025
STABILITY	—	.540	.164 —	.972	—	.661	—	1.042	—	—	—	—
STRESS (#/in²)	+1656	-2374	-244 +3657	-476 -4279	+2184	-2911	+507 +3871	-1229 -4593	+14751	+10687	+16652	+8786

ALLOW. TENSILE STRESS = 38000 PSI.



(+) IS TENSION

SEE SHT 182 FOR ALLOWABLE STRESSES

t = 1.0275"

CHICAGO BRIDGE & IRON COMPANY

GREENVILLE ENGINEERING DEPT.

RECORDS N...

KFR

ORIGINAL DESIGN BASIS SUMMARY

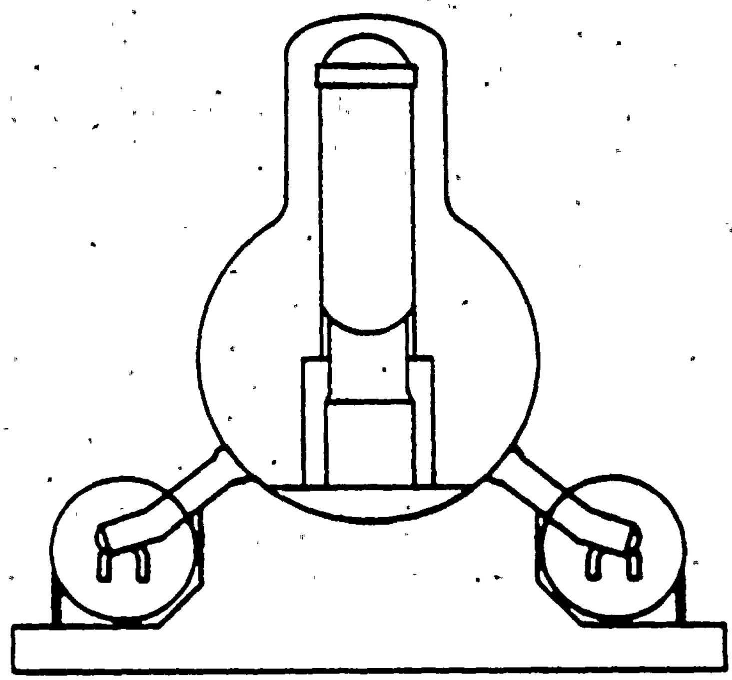
- Original drywell design pressure and temperature consistent with Mark I Analysis

- Drywell design margin is more than adequate

- Corrosion allowance of 1/16" is still usable in drywell

- Short term hydrodynamics in wetwell can be decoupled from long term decay loading in drywell

Scoping Study for Phase 2 Analysis of Mark I Containment Vessels



Nine Mile Point Nuclear Station – Unit 1
Niagara Mohawk Power Company
Scriba, New York

Prepared By:
CBI Na-Con, Inc.
Oak Brook, Illinois
October, 1988

CBI
CBI Na-Con

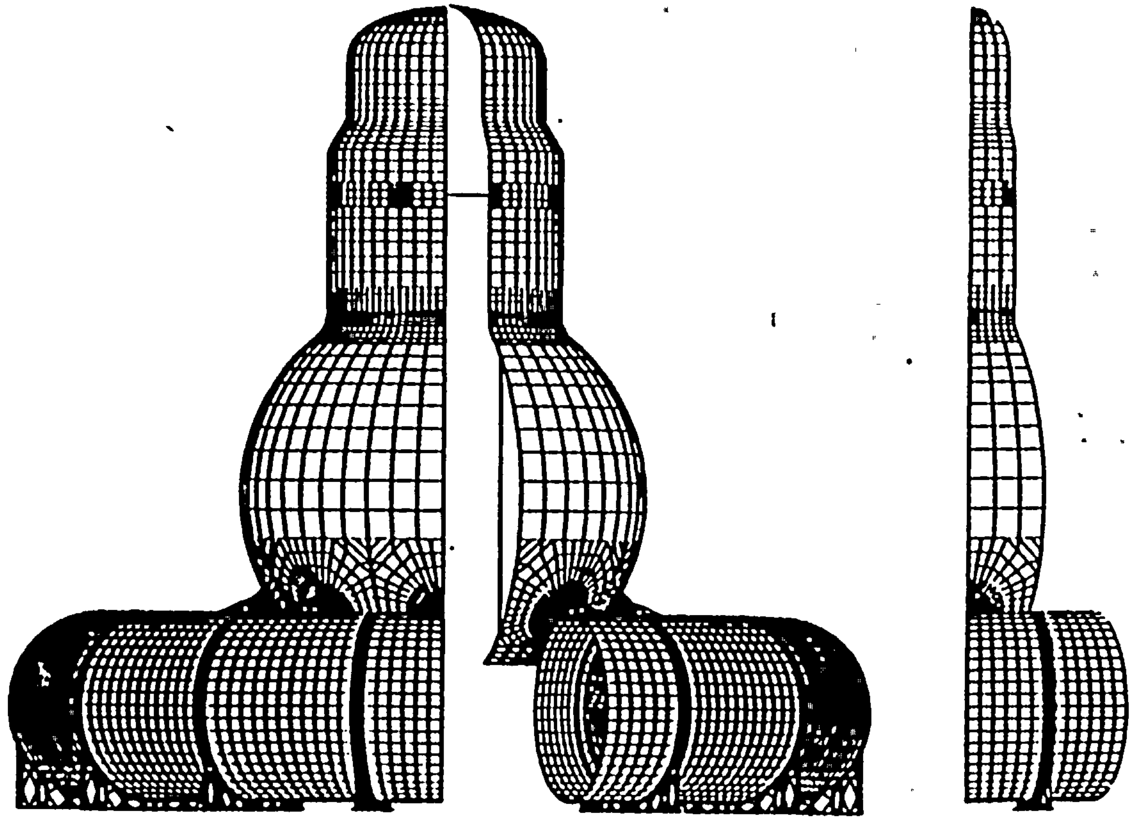


Figure B.3-6
· Elevation of the 22.5° slice of the MARK I Contaiment
Actually modeled in the F.E. Analysis

NIAGARA MOHAWK

CONTAINMENT ULTIMATE STRENGTH STUDY

FAILURE CRITERIA

Membrane Strain	=	1%	(governs)
Surface Strain	=	2%	
Local Peak Strain	=	5%	

RESULTS

Peach Bottom	=	159 psig @ torus shell plate
Nine Mile #1	=	70% - 159 psig = 111 psig (approx.)

NIAGARA MOHAWK

TORUS SHELL ANALYSIS

MARK I PROGRAM EVENT COMBINATIONS

- 27 Mark I event combinations reduced to 4 bounding event combinations

- Bounding event combinations

Number	Title
14	Chug, O.B.E., S.B.A., S.R.V. (C.O.)
18	Pool Swell, O.B.E., D.B.A.
20	C.O., O.B.E., D.B.A. (Chug)
25	Pool Swell, S.S.E., D.B.A., S.R.V.

CO	=	Condensation Oscillation Loads
Chug	=	Chugging Loads
SBA	=	Small Break Accident (Small Diameter Pipe Break)
DBA	=	Design Basis Accident
SRV	=	Safety Relief Valve Actuation
OBE	=	Operating Basis Earthquake
SSE	=	Safe Shutdown Earthquake

NIAGARA MOHAWK

PRESSURE LOAD DEFINITION FOR MARK I PROGRAM

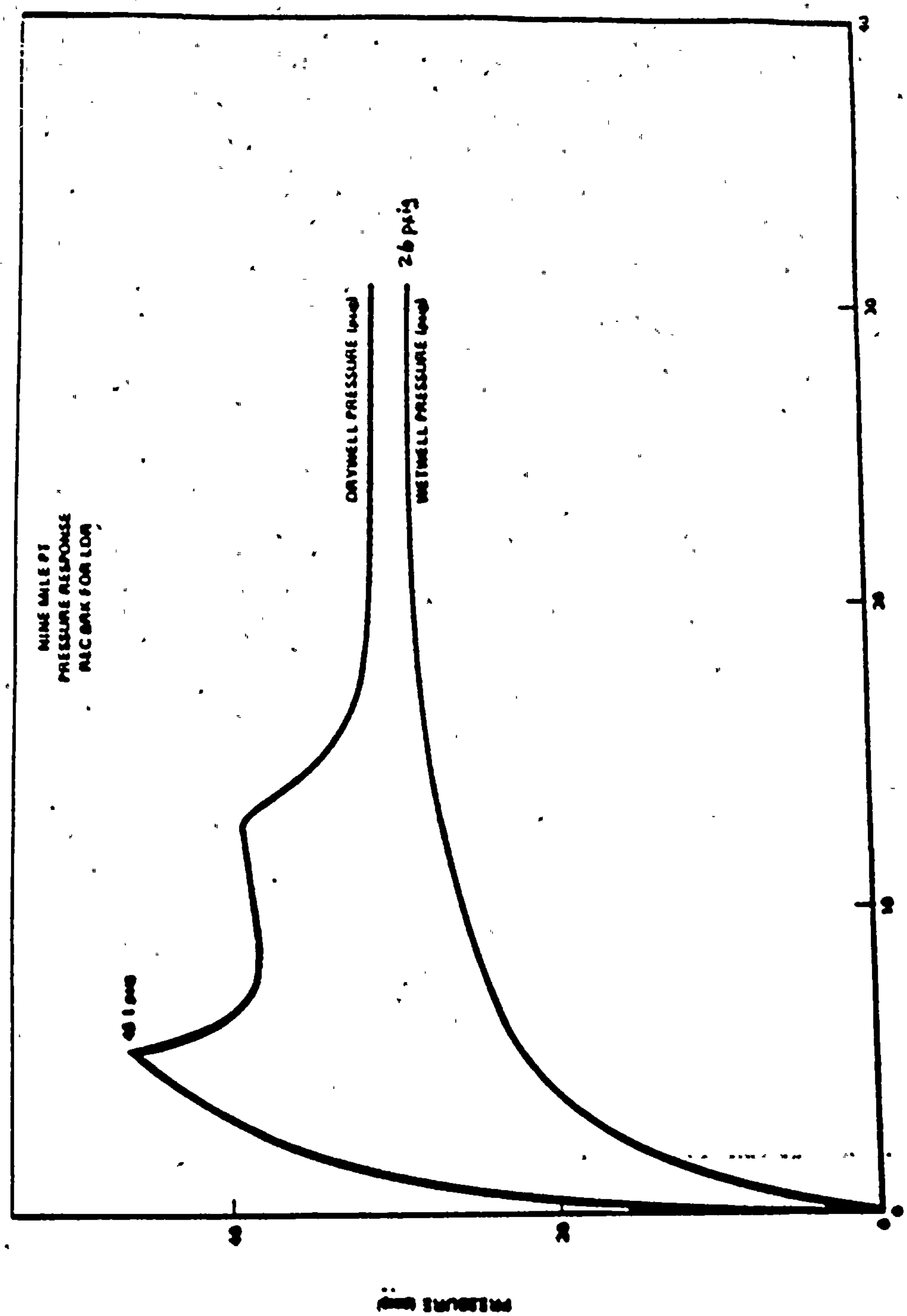


Figure NM 4.1.1-1. DBA Containment Pressure Response

TEMPERATURE LOAD DEFINITION FOR MARK I PROGRAM

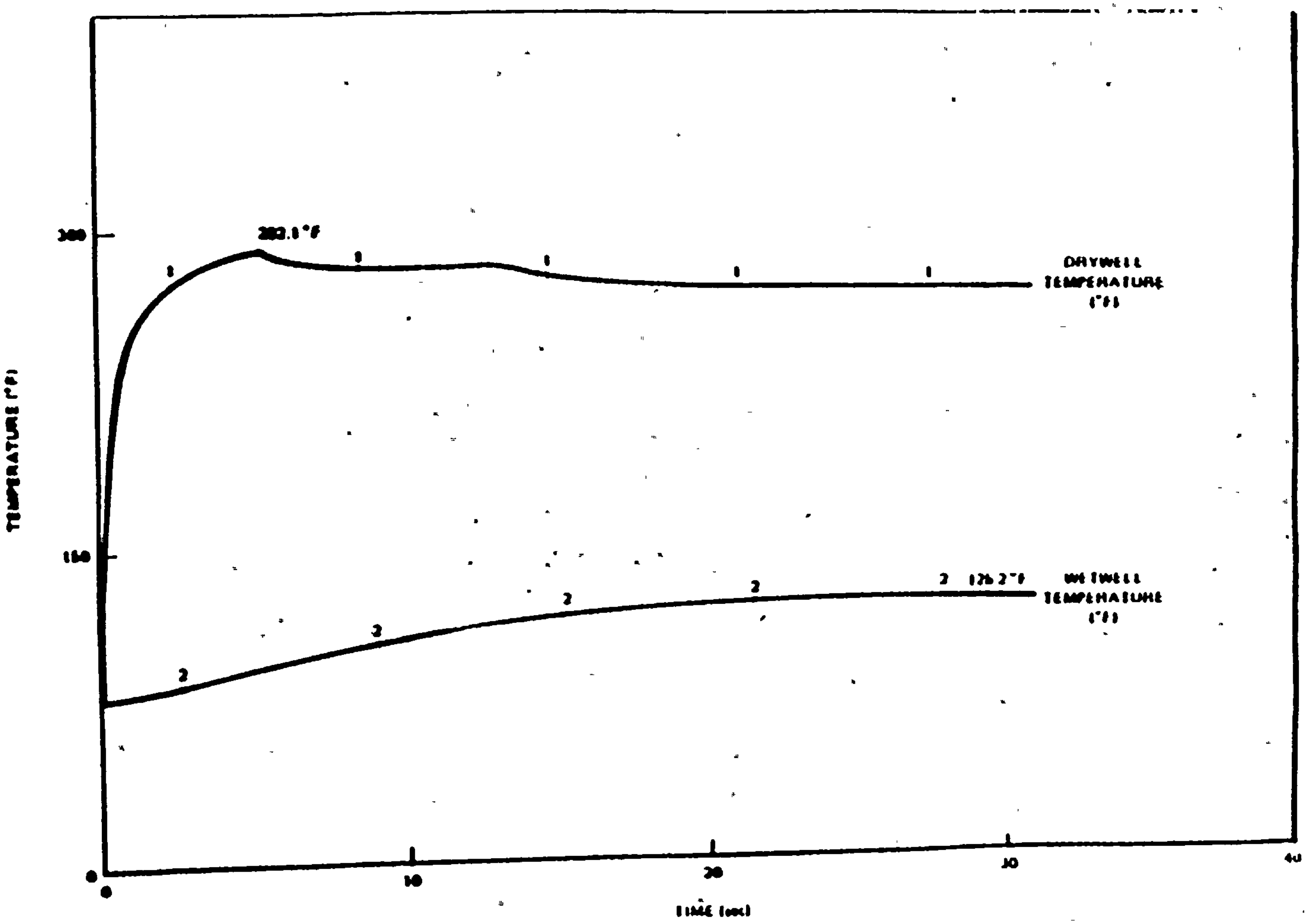


Figure NM 4.1.1-2. DBA Containment Temperature Response

NIAGARA MOHAWK

NIAGARA MOHAWK

SUMMARY

MARK I MODIFICATIONS
@ NINE MILE POINT UNIT #1:

Y - Quenchers

Vent Head Deflectors

Downcomer Tie Straps

Saddles

Catwalk Removal

Relief Valve Vacuum Breakers

Torus Attached Piping

Resupport Of Relief Valve Discharge Lines

NIAGARA MOHAWK

TORUS WALL THICKNESS PROGRAM

I. HISTORY/BACKGROUND

- C.B.I. (1/16" Corrosion)

- TORUS: ISI 1975-1988
  6x6 12x12

- MARK I Program 1975-1984

- Water Quality (Iron-Eating Bacteria) 1979/1980 RCT Report

- Containment Coating Study 1984

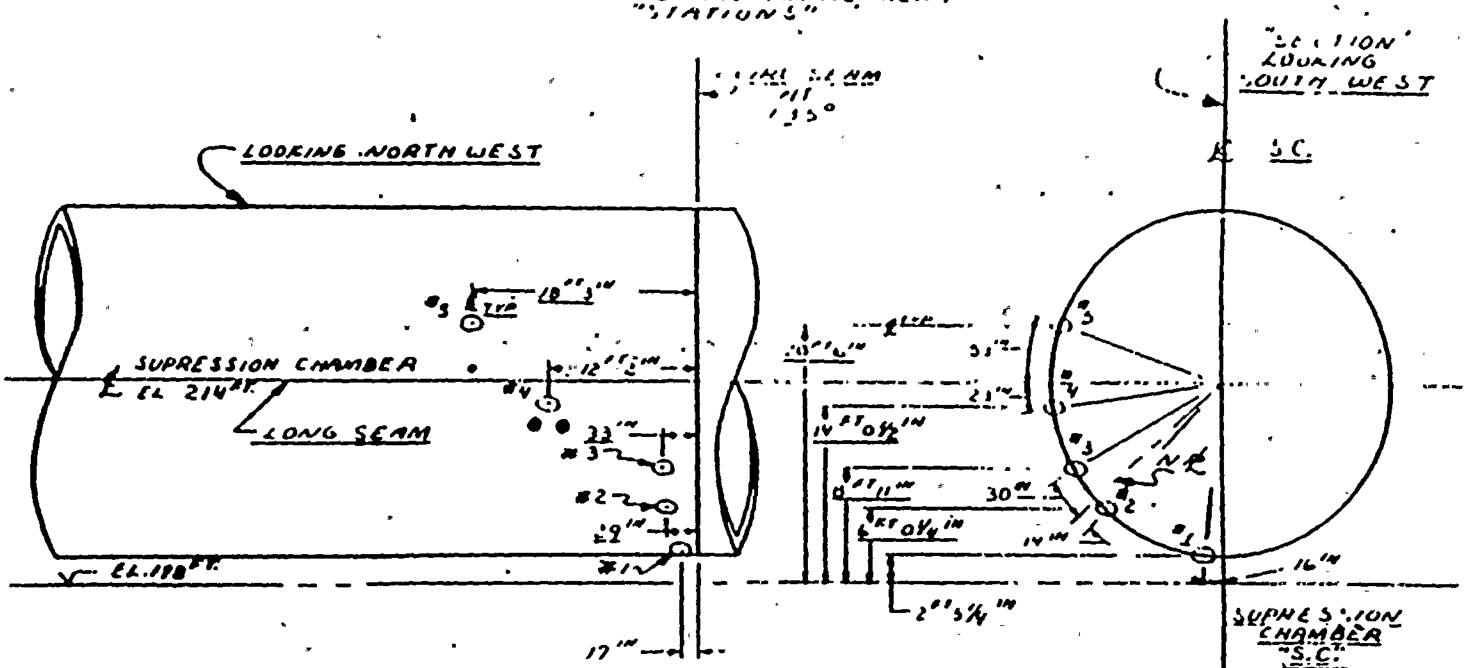
- 1987 - Internal - Evaluation

- T.E.S. ---> Report 9/87 -1/88

- Further Evaluations - Weld Repair, Proposal For Long Term 2/5/88

- 1988 ISI/NRC Readings (UT) - April 1988

LOCATION OF
NINE MILE PT
UNIT #1
SUPRESSION CHAMBER
THICKNESS MEASUREMENT
"STATIONS"



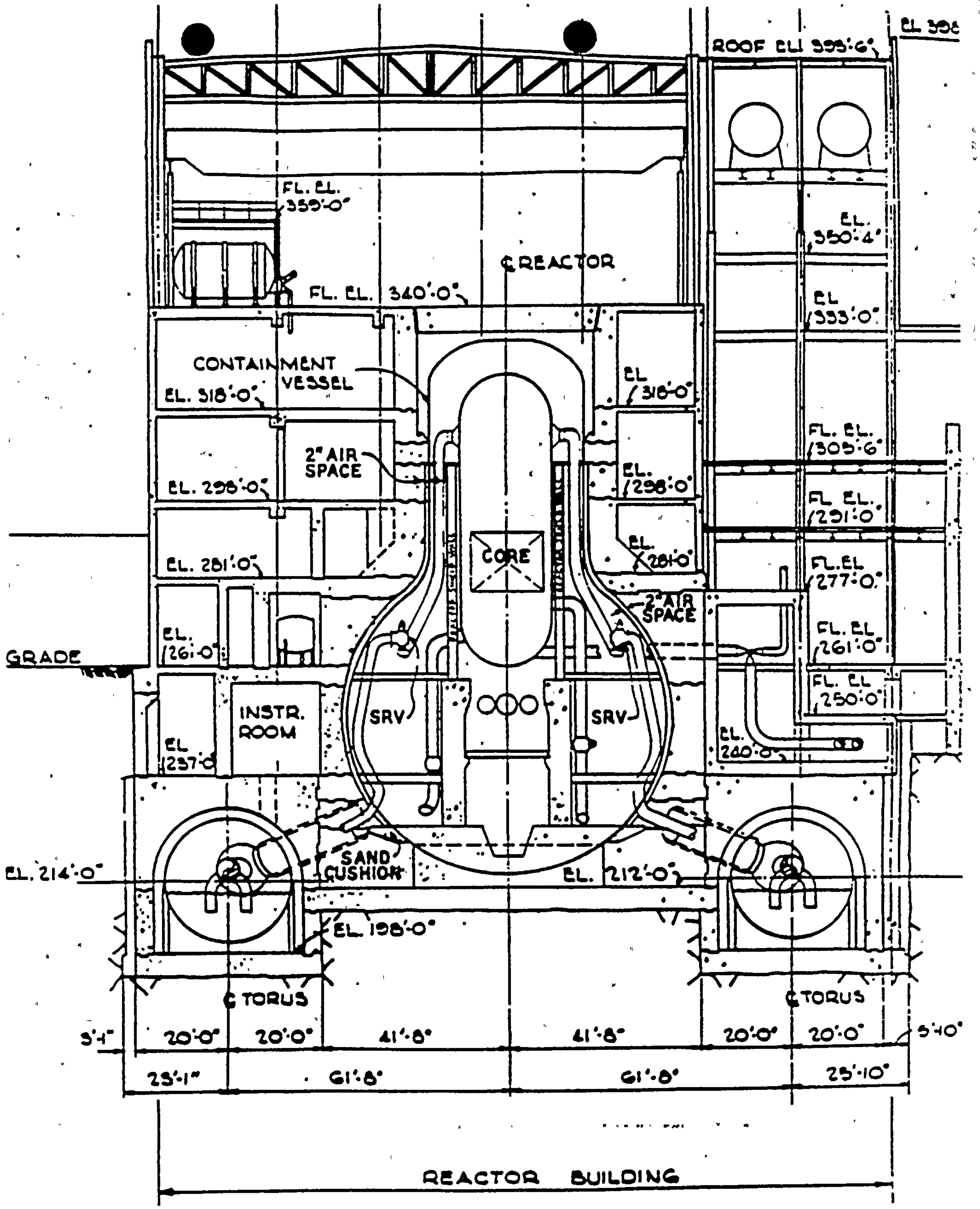
NOTES:

1. SCALE 1/8 IN = 1 FT 0 IN
2. • = 6 IN Ø CORE SPRAY BYPASS
   ● = 12 IN Ø CORE SPRAY

NIAGARA MOHAWK POWER CORP.

4 26 79

SE Corner



ROOF EL 355'-6"
EL 358
FL. EL. 359'-0"
EL. 350'-4"
℄ REACTOR
FL. EL. 340'-0"
EL 333'-0"
CONTAINMENT VESSEL
EL. 318'-0"
EL 318'-0"
FL. EL. 305'-6"
2" AIR SPACE
EL. 298'-0"
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FL. EL. 291'-0"
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2" AIR SPACE
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GRADE
FL. EL. 250'-0"
INSTR. ROOM
SRV
SRV
EL 237'-0"
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SAND CUSHION
EL. 214'-0"
EL. 212'-0"
EL. 198'-0"
℄ TORUS
℄ TORUS
3'-1"
20'-0"
20'-0"
41'-8"
41'-8"
20'-0"
20'-0"
5'-10"
23'-1"
61'-8"
61'-8"
25'-10"
REACTOR BUILDING

NIAGARA MOHAWK

CONTAINMENT PROGRAM SUMMARY

- Long history of Torus Monitoring
- Trending shows very low corrosion rate
- Drywell shell has more than adequate design margin
- Torus shell stress is near allowable
- Continued monitoring & evaluation -- L. McNeer.

NINE MILE POINT UNIT 1

STATUS OF CORROSION

NIAGARA MOHAWK POWER CORPORATION

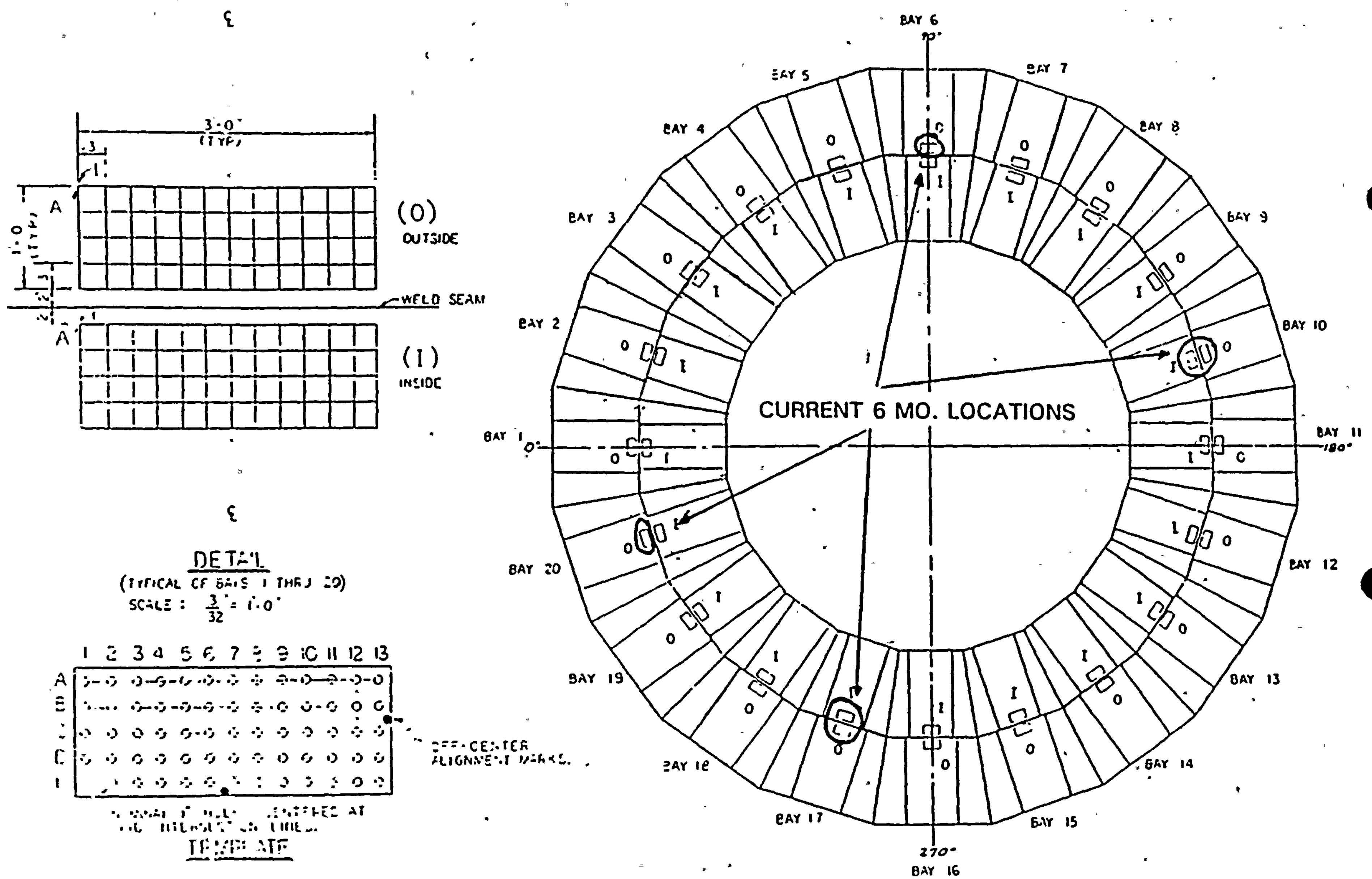
APRIL 24, 1991

BACKGROUND

RAP ISSUE: 1988 - 1989

- NMPC INTERNAL REVIEW OF TORUS INSPECTION PROGRAM
  - ESTABLISHED ONE-TIME SAMPLE OF ALL BOTTOM MID BAY AREAS - AUG 1989
  - CORROSION RATE RE-EVALUATED ON PLATE-BY-PLATE BASIS
  - ESTABLISHED THINNEST ONE-TIME SAMPLE AREAS TO REVISIT EACH 6 MO.

UT MEASUREMENT LOCATIONS



PLAN OF BOTTOM PLATES FOR SUPPRESSION CHAMBER

TYPE OF CORROSION - PITTING OR GENERAL

- CORROSION IS GENERAL AND UNIFORM

  - CONFIRMED BY AUGUST 1989 UT AND VISUAL

  - METALLURGIST INSPECTION AUGUST 1989 CONFIRMED CORROSION UNIFORM THROUGHOUT TORUS

  - UT INSPECTION OF ALL BAYS AT BOTTOM INDICATES VERY SMALL VARIANCE IN THICKNESS

  - ONLY EXCEPTION IS SMALL BAND OF ~ 6" ABOVE NORMAL WATER LEVEL

  - PITS OF 20 MIL UP TO 30 MIL EXIST IN THAT BAND

  - PITS IN THIS AREA NOT A PROBLEM - 150 MILS MARGIN

  - AVERAGE LOSS OF METAL IS 16 MILS IN OVER 20 YEARS OPERATION

THICKNESS MEASUREMENTS, METHOD, NUMBER AND SPATIAL DISTRIBUTION

- MEASUREMENTS ARE UT FROM OUTSIDE SURFACE OF SHELL - PAINT REMOVED

- PLAN INCLUDES SIX GRIDS WITH LOWEST AVG. THICKNESS FROM ONE TIME SAMPLE OF ALL 20 BAYS

- SPATIAL DISTRIBUTION OF MEASUREMENTS IS EVERY 3", TOTAL OF 65 PER GRID

DETAILS OF CORROSION DETECTION PROGRAM AND LICENSEE COMMITMENTS

- CURRENT PROGRAM INCLUDES SIX 1' X 3' GRIDS SELECTED FROM ONE-TIME MEASUREMENT PROGRAM

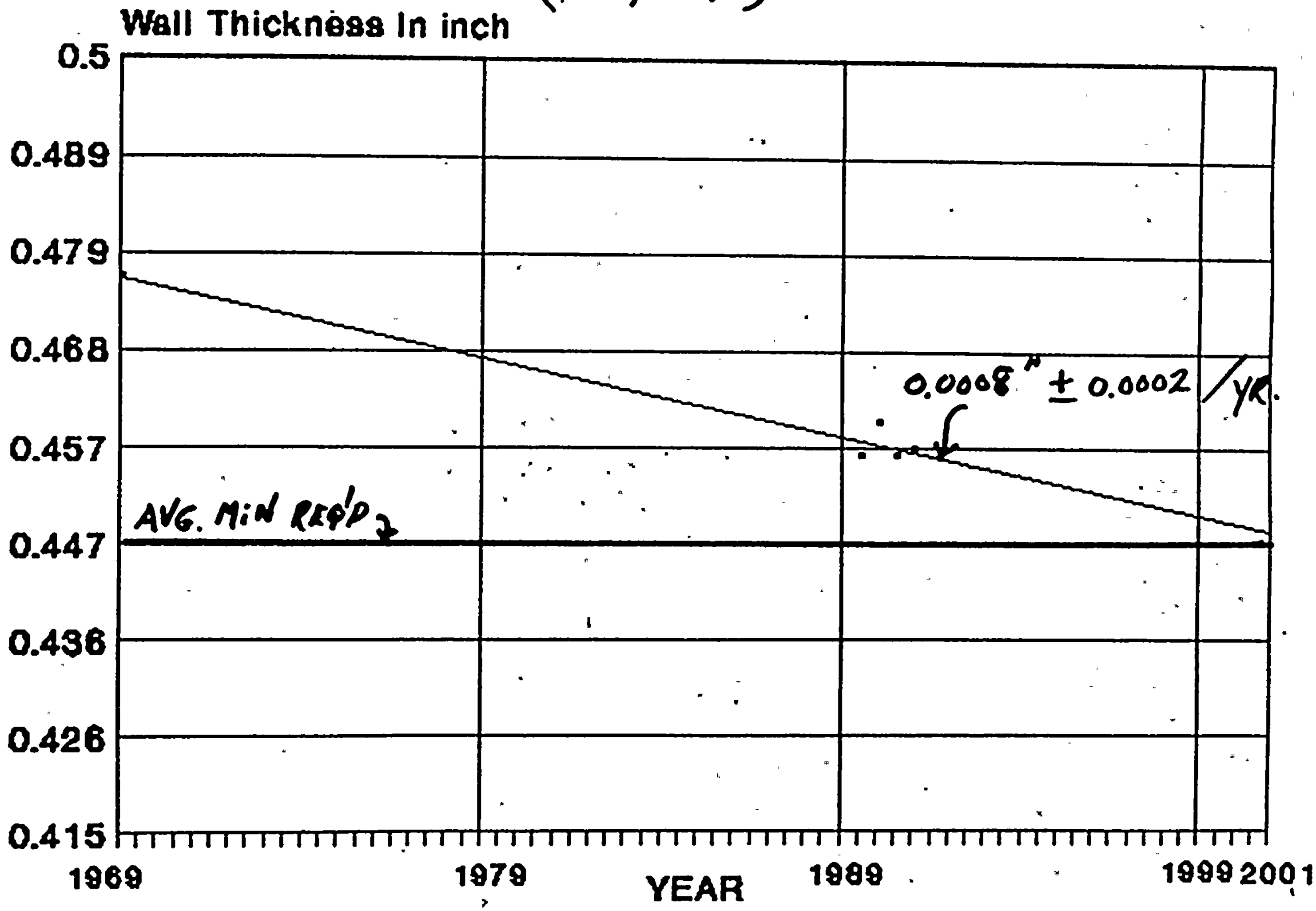
  - TEMPLATE USED WITH MARKINGS TO ASSURE POSITIVE, CONSISTENT LINEUP

  - SAME PROCEDURE, EQUIPMENT USED FOR CONSISTENCY

  - RESULTS CORRECTED FOR CALIBRATION AND TRENDED FOR INDICATED CORROSION RATE

- COMMITMENT IS TO TAKE MEASUREMENTS APPROX. EVERY SIX MONTHS

Torus Wall Thickness (Nine Mile 1)
Average of 6 Thinnest Plates
(COMPOSITE)



RGS 4.23.1991

TRENDING OF CORROSION INDICATORS

- TREND INDICATES AVG. CORROSION RATE IS 0.8 MILS PER YR. FROM UT MEASUREMENTS
- THIS AVG. RATE (0.8) + ONE STD. DEVIATION (0.2) IS 1 MIL PER YEAR
- CONSISTENT RESULTS OBTAINED FROM 1979 SLUDGE SAMPLES
- GRAPHS INDICATE DATA TREND LINE ON <u>WORST</u> PLATE AND COMPOSITE OF 6 THINNEST WITH ACTUAL DATA EXTRAPOLATED TO ORIG. CALC. THICKNESS

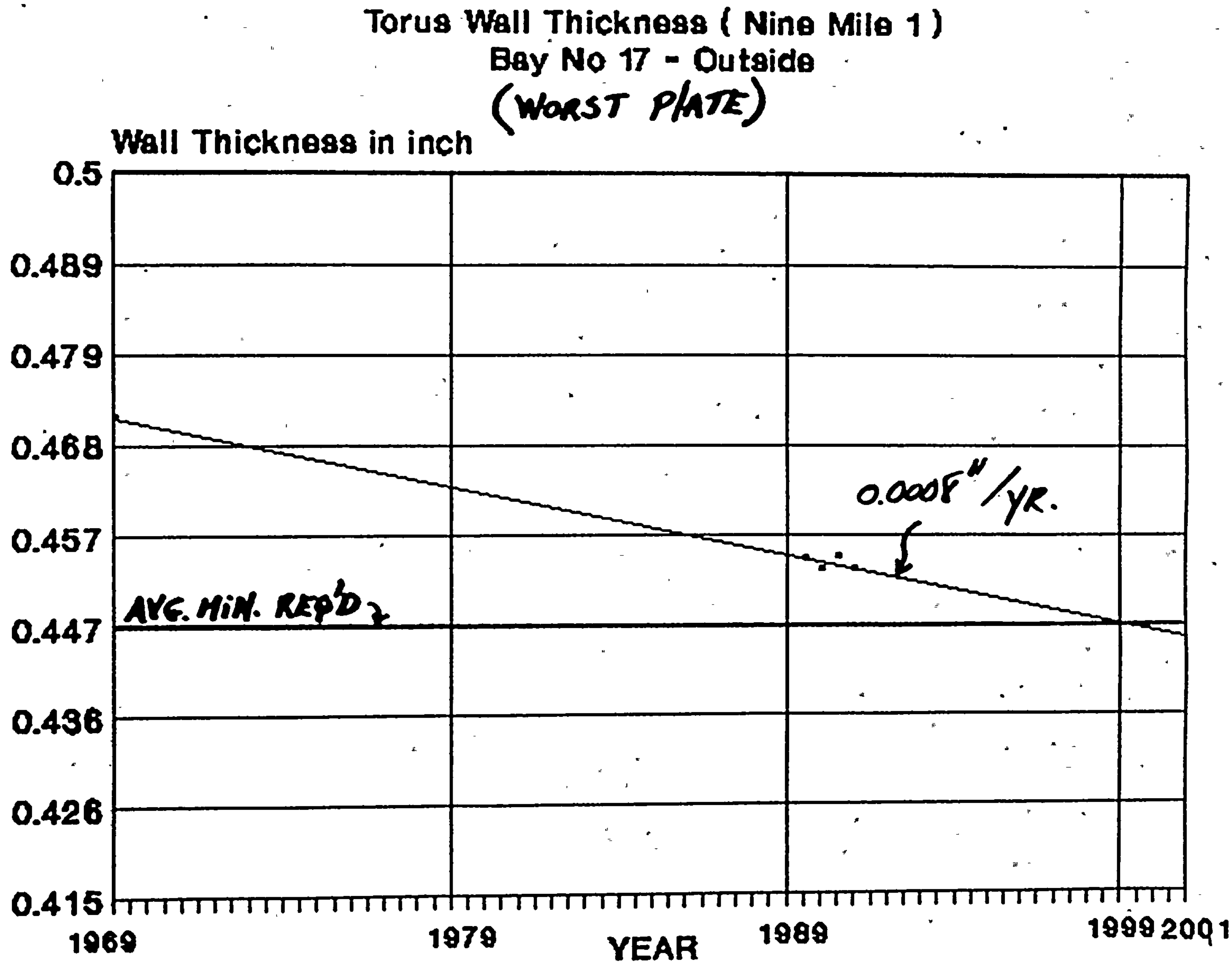
LEVEL OF CONFIDENCE IN STATE OF KNOWLEDGE WITH RESPECT TO MEASUREMENT OF THICKNESS IN AFFECTED AREAS OF TORUS VS ENTIRE TORUS IN TERMS OF STATISTICAL SAMPLING TECHNIQUES AND PREDICTED CORROSION RATES

- HIGH CONFIDENCE IN SAMPLING REPRESENTATION OF AFFECTED AREAS VS ENTIRE TORUS

  - AUG 1989 ONE TIME SAMPLE INCLUDED 1' X 3' GRIDS ON ALL MID BAY BOTTOM PLATES

  - CURRENT PLAN INCLUDES GRIDS ON SIX THINNEST MEASURED PLATES FROM AUG 1989 SAMPLE

  - PLAN IS JUSTIFIABLE BECAUSE OF CONFIRMED UNIFORM CORROSION RATE AND ALL OTHER PLATES IN CRITICAL AREAS ARE THICKER


Torus Wall Thickness (Nine Mile 1)
Bay No 17 - Outside
(WORST PLATE)
Wall Thickness in inch
0.5
0.489
0.479
0.468
0.457
0.447
0.436
0.426
0.415
1969
1979
1989
1999
2001
YEAR
0.0008"/YR.
AVG. MIN. REQ'D

RGS 4.23.1991

- HIGH CONFIDENCE IN PREDICTED CORROSION RATE

  - AVG. CORROSION RATE BASED ON UT MEASUREMENTS ON PLATES IDENTIFIABLE TO ORIG. MIL CERTS

  - INCLUDES 22 YRS. OF CORROSION FROM INITIAL TORUS FILL

  - CORRELATES TO PREDICTED RATE FROM 1979 SLUDGE SAMPLES

  - PREDICTED RATE IS AVG. RATE (0.8 MILS/YR.) + STD. DEVIATION (0.2 MILS/YR.)

TORUS WATER ENVIRONMENT - INHIBITORS, PH, CHLORIDE, SULPHATE, CONDUCTIVITY, AMOUNT AND TYPE OF CORROSION PRODUCTS

- NO INHIBITORS USED, ONLY N_2 ATMOS. DURING OPERATION
- PH: 6.0 TO 7.0; AVG 6.5
- CHLORIDE <10 TO 35 PPB; <10 PPB - 87% OF TIME
- SULPHATE: 2 TO 15 PPB; AVG. 10 PPB
- CONDUCTIVITY RANGES FROM 0.6 TO 2.8 UMHO/CM; AVG. 1.5 UMHO/CM
- CORROSION PRODUCT IS IRON OXIDE
  - EXISTS AS FILM ON SHELL SURFACE AND SLUDGE ON BOTTOM
  - ESTIMATED DEPTH OF SLUDGE ~1/8"

DECISION NOT TO USE COATING

- BASED ON FOLLOWING:
  - o EFFECTIVENESS OF COATINGS IN INDUSTRY (I.E., USEFUL LIFE, PROBLEMS)
  - o ALARA IMPACT
  - o MAINTENANCE (INSPECTION, REPAIRS)
  - o OUTAGE IMPACT (~ 140 DAYS CRITICAL PATH)
  - o COST 8-10M + OUTAGE CRITICAL PATH IMPACT (~ 20M)
- K-T ANALYSIS CONSIDERED ABOVE AND EVALUATED AS LESS DESIRABLE THAN STIFFENING RINGS.

NINE MILE POINT UNIT 1

STRUCTURAL INSPECTIONS RELATED TO DEGRADATION BY CORROSION

NIAGARA MOHAWK POWER CORPORATION

APRIL 24, 1991

Condition of Vent Pipes and Downcomers

- Visual Inspection Each Refuel Outage
  - Covers vent pipe, bellows, downcomer support structures, brackets and bolting
  - Latest inspection 3/7 - 9/91
  - No discontinuities, defects, or accelerated corrosion
- Visual inspection of entire torus interior - August 1989
  - Performed by metallurgist
  - Vent pipes, downcomers in excellent condition
  - Original red lead primer intact

- UT Inspection of Vent Spheres

  - Two inspected in April 1988
  - Two additional inspected in September 1989
  - Most critical location is bottom of sphere
  - UT shows bottom thickness is very close to original thickness
  - Actual UT measured thickness is 150 mils above required thickness

Bellows Connection of Vent Pipes

- Visually inspected each refuel outage (Not to exceed 2 yrs.)
  - Latest inspection 3/91
  - No observed defects
  - Metallurgist inspected in August 1989
  - Found in excellent condition
  - ILRT 5/90 indicates no meas. leakage from vent penetrations

Saddle Support and Important Structural Connections

- Saddle support inspection 3/89 for anchor bolt problems of I&E 89-06

  ° Attachments do not reflect configuration described in 89-06

  ° Anchor bolts did not exhibit any distortions indicated in 89-06.

- Saddle supports inspected in 1983 to verify conformance to Mark I Mod

  ° Conforms to Mark I Mod and Const. Dwgs.

- Region I inspected saddles, supports tie downs in August 1988

  - Insp. Report 88-28 confirmed conformance to NUREG-0661, licensee commitments
  - Compared to Const. Dwgs., confirmed quality and location

  - No deficiencies or violations

Integrity of Welds and Anchors

- Saddle support welds inspected 10/88 and 3/89

  - Determined weld attachments more than adequate to meet Mark I Design loads

- Walkdown and video tape of torus room 1990

  - Shows physical condition is good

- Torus penetrations inspected each refuel outage for discontinuities

  - Last inspection 3/91 - no discontinuities

- Drywell inspected each cycle for penetration discontinuities, support attachments and brackets for defects

  - Last inspection 3/91 - no discontinuities, defects

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NINE MILE POINT UNIT 1

VENTING RELATED ISSUES

NIAGARA MOHAWK POWER CORPORATION

APRIL 24, 1991

PCPL LIMIT, SIZE OF VENT AND PATHWAY, DEGREE OF HARDENED VENT PATHWAY AND IMPLEMENTATION DATE FOR HARDENED VENT

- PCPL LIMIT IS 40.25 PSIG
- SIZE OF VENT IS 20"
- PATHWAY IS AS SHOWN ON DIAGRAM
- VENT IS HARDENED FROM TORUS TO WITHIN 12' OF STACK BREECHING
- DESIGN PRESSURE IS SAME AS WETWELL
- IMPLEMENTATION DATE TO HARDEN LAST 12' IS 1992 REFUEL OUTAGE
    - LAST 12' TO BE REPLACED WITH 30" PIPE
    - MOD. RESULTED FROM REVIEW OF EXIST VENT VS GL89-16 CRITERIA

CURRENT SPECIFIC VENTING PROCEDURE

- VENT PROCEDURES ARE EOP'S 4.0 AND 4.1

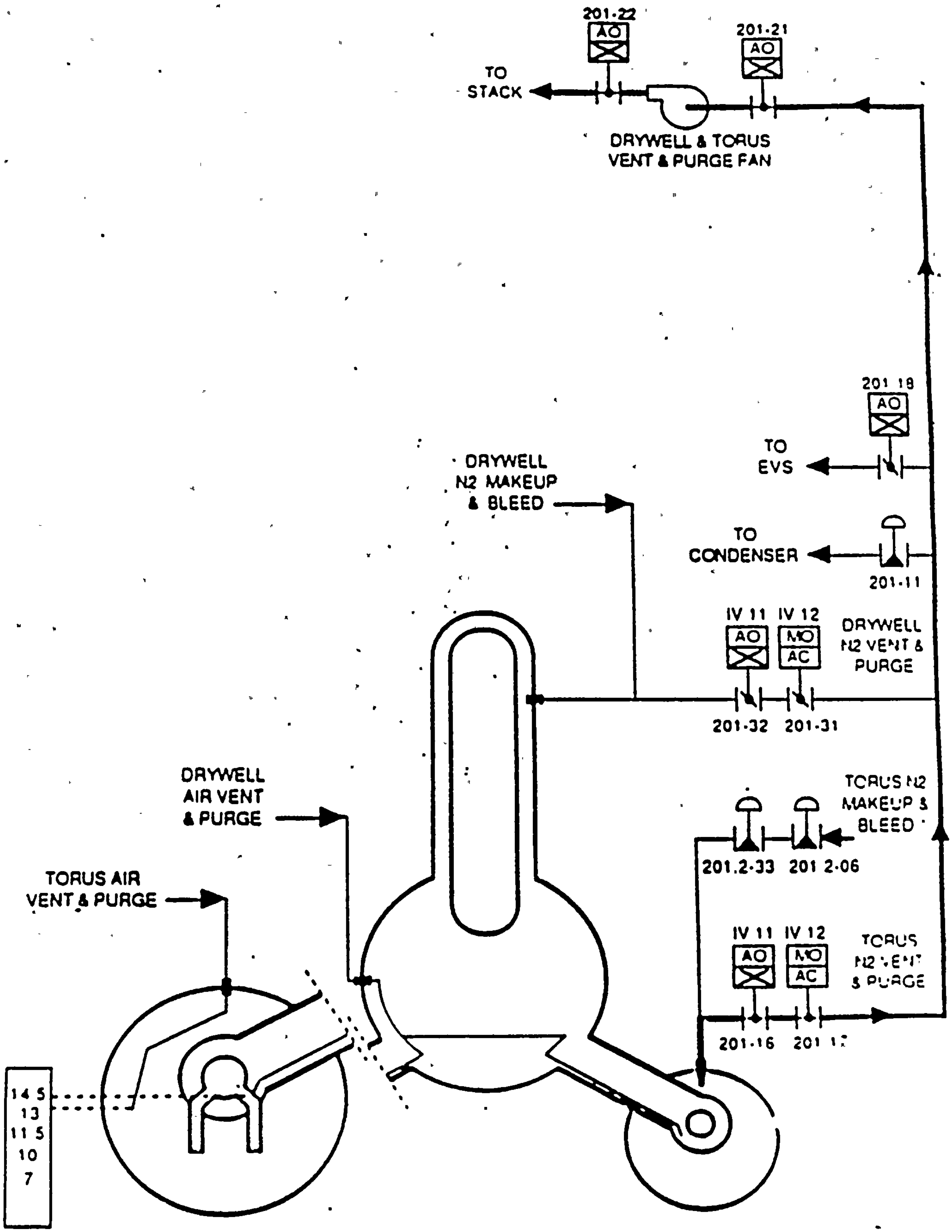
  - o PROCEDURES SPECIFY OPENING INBOARD ISO VALVE, THROTTLING WITH OUTBOARD ISO VALVE

  - o VENT AND PURGE FAN PLACED IN OPERATION

  - o OUTBOARD ISO VALVE THROTTLED AS NECESSARY TO MAINTAIN PRESS < PCPL

  - o VENT LIMIT IS ESTABLISHED ON PRESS CAPABILITY OF VENT VALVE OPENING, NOT ON TORUS SHELL THICKNESS

HARDENED VENT PATHWAY



NINE MILE POINT UNIT 1

RE-EXAMINATION OF TORUS SHELL CONDENSATION OSCILLATION HYDRODYNAMIC LOADS

NIAGARA MOHAWK POWER CORPORATION
APRIL 24, 1991

TORUS CO LOAD REDUCTION

OBJECTIVE

- TO RE-EXAMINE THE MARK I TORUS SHELL CONDENSATION OSCILLATION HYDRODYNAMIC LOAD BY USING A MULTI-BAY HYDRODYNAMIC MODEL THAT TAKES INTO ACCOUNT
  - UNCORRELATED STEAM CONDENSATION
  - NINE MILE POINT UNIT 1 DOWNCOMER CONFIGURATION
- TO SHOW HOW THIS PROVIDES FOR
  - A MORE REALISTIC TORUS SHELL LOAD
  - AN INCREASE IN CORROSION ALLOWANCE
  - POSTPONEMENT OF MODIFICATIONS

TORUS CO LOAD REDUCTION

OUTLINE

- BACKGROUND/FSTF
- NINE MILE POINT UNIT 1 TORUS
- CONSERVATISMS IN THE TORUS CO SHELL LOAD
- OUTLINE OF A MULTI-BAY HYDRODYNAMIC MODEL USED TO MODIFY THE MARK I TORUS CO SHELL LOAD
- UTILIZATION OF RESULTS
- ADDITIONAL CONSERVATISMS IN TORUS SHELL

TORUS CO LOAD REDUCTION

BACKGROUND

MARK I PROGRAM

THE PURPOSE OF THE MARK I TORUS PROGRAM WAS TO EVALUATE THE EFFECTS OF HYDRODYNAMIC LOADS RESULTING FROM A LOSS OF COOLANT ACCIDENT (LOCA) AND AN SRV DISCHARGE ON THE TORUS STRUCTURE.

THE CONTROLLING MARK I LOAD CASE FOR NINE MILE POINT UNIT I INCLUDES THE COMBINATION OF DEAD-WEIGHT, SEISMIC, DBA PRESSURE AND DBA CO LOADS.

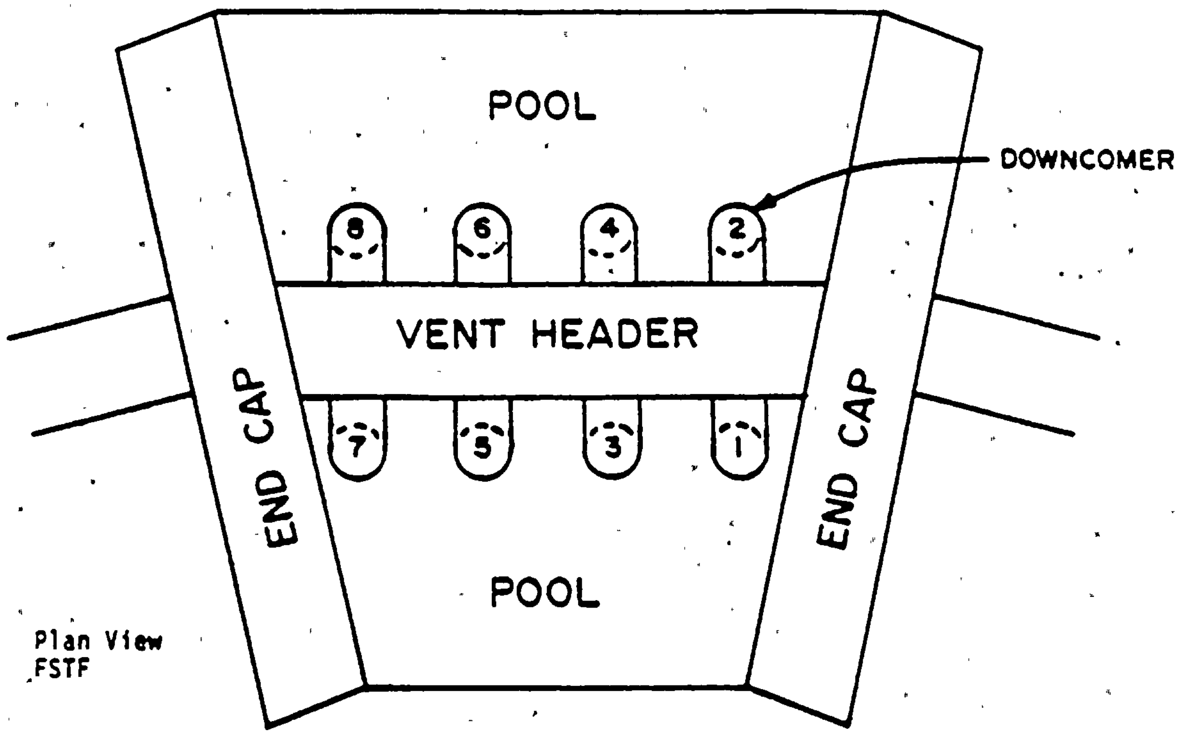
TORUS CO LOAD REDUCTION

BACKGROUND (CONT'D)

- THE MARK I OWNERS GROUP, UNDER GE MANAGEMENT UNDERTOOK A FULL SCALE TEST PROGRAM TO MEASURE CONTAINMENT LOADS DURING LOCA'S
- THE FULL SCALE TEST FACILITY (FSTF) WAS A 22.5° SECTOR (BAY) OF A MARK I SUPPRESSION POOL TORUS.
- THE BAY REPRODUCED AT FULL SCALE WAS ONE THAT CONTAINED EIGHT DOWNCOMERS (CHOSEN TO MAXIMIZE CONTAINMENT LOADS).
- SINCE ONLY A SECTOR WAS MODELED, END CAPS WERE REQUIRED TO END THE BAY, ALLOWING PRESSURIZATION OF THE BAY AND CONTAINMENT OF SUPPRESSION POOL WATER. THESE END CAPS WERE VERY RIGID BY DESIGN.

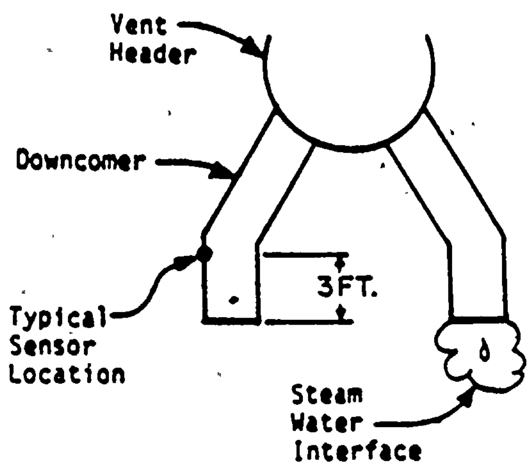
TORUS CO LOAD REDUCTION

FULL SCALE TEST FACILITY (FSTF)



Plan View
FSTF

VENT	TRANSDUCER
1	P5123
2	P5243
3	P5323
4	P5443
5	P5523
6	P5643
7	P5723
8	P5843



Downcomer exit pressure transducers in FSTF.

TORUS CO LOAD REDUCTION

FSTF TEST RESULTS

- ANALYSIS RESULTS BASED ON THE FSTF TESTS HAVE SHOWN THAT DURING CONDENSATION OSCILLATION
  - THE PULSATING CONDENSATION AT EACH EXIT IS RANDOM (UNCORRELATED) IN THE FREQUENCY DOMAIN EXCEPT AT TWO FREQUENCY RANGES
  - THE PULSATING CONDENSATION AT THE DOWN-COMER EXITS ARE STRONGLY CORRELATED BETWEEN DOWNCOMERS AT 4-6 HZ AND WEAKLY CORRELATED AT 8-12 HZ.
  - THESE FINDINGS WERE PRESENTED TO THE NRC ON MARCH 4, 1981
- THE CONSEQUENCE OF THIS RANDOMNESS AND THE GEOMETRY OF THE FULL SCALE TEST FACILITY IS A MEASURED CONDENSATION OSCILLATION TORUS LOAD WHICH IS VERY CONSERVATIVE.

TORUS CO LOAD REDUCTION
CORRELATION OF PRESSURE SOURCES-FSTF RESULTS

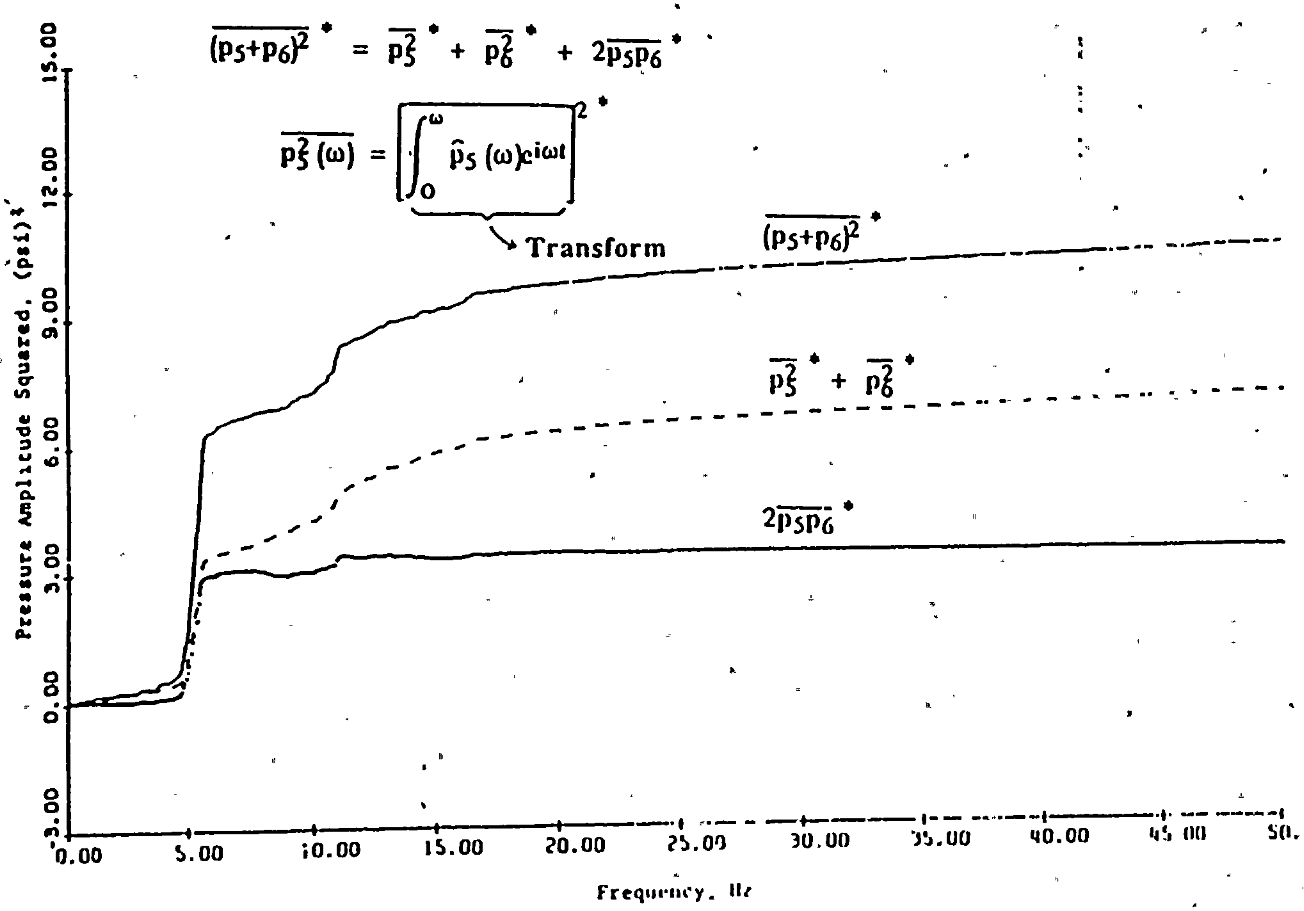


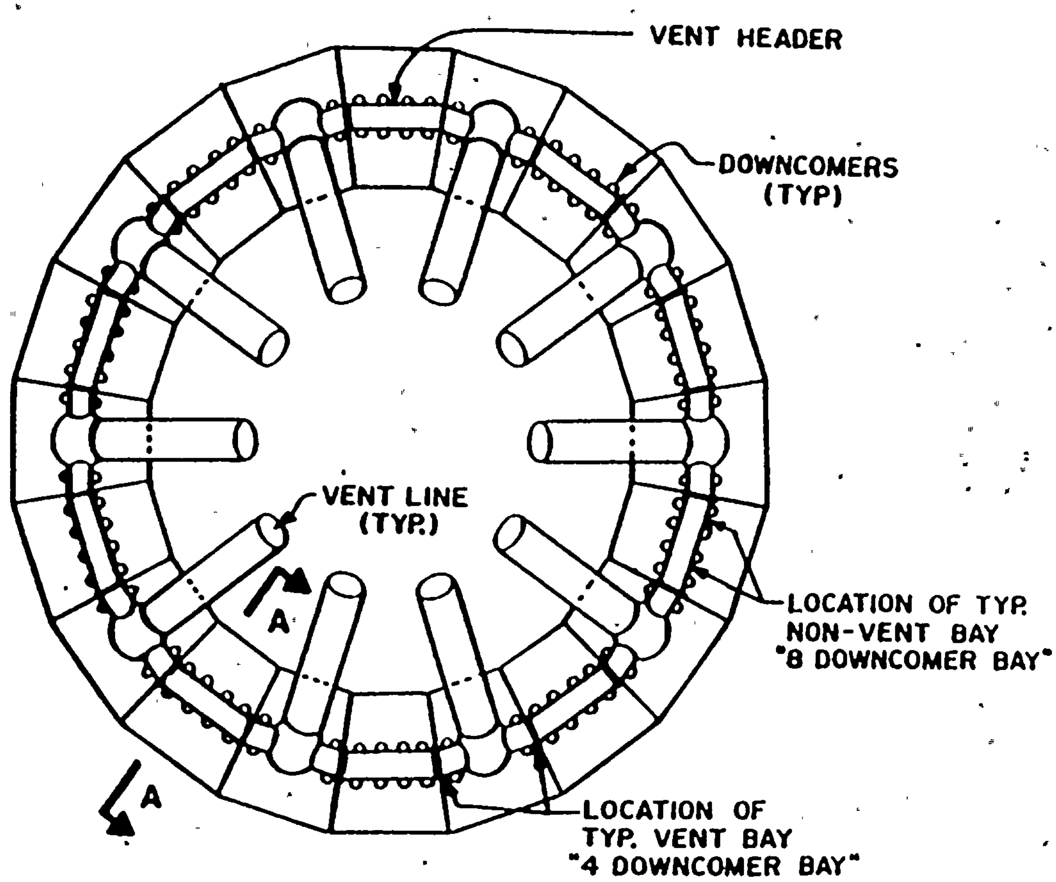
Figure 2 Mean Square pressure signals between downcomers 5 and 6, FSTF Run M8, 20 - 35 seconds during condensation oscillation as a function of frequency (measured from zero frequency)

TORUS CO LOAD REDUCTION
OUTLINE

- BACKGROUND/FSTF
- **NINE MILE POINT UNIT 1 TORUS**
- CONSERVATISMS IN THE TORUS CO SHELL LOAD
- OUTLINE OF A MULTI-BAY HYDRO-DYNAMIC MODEL USED TO MODIFY THE MARK I TORUS CO SHELL LOAD
- UTILIZATION OF RESULTS
- ADDITIONAL CONSERVATISMS IN TORUS SHELL

TORUS CO LOAD REDUCTION

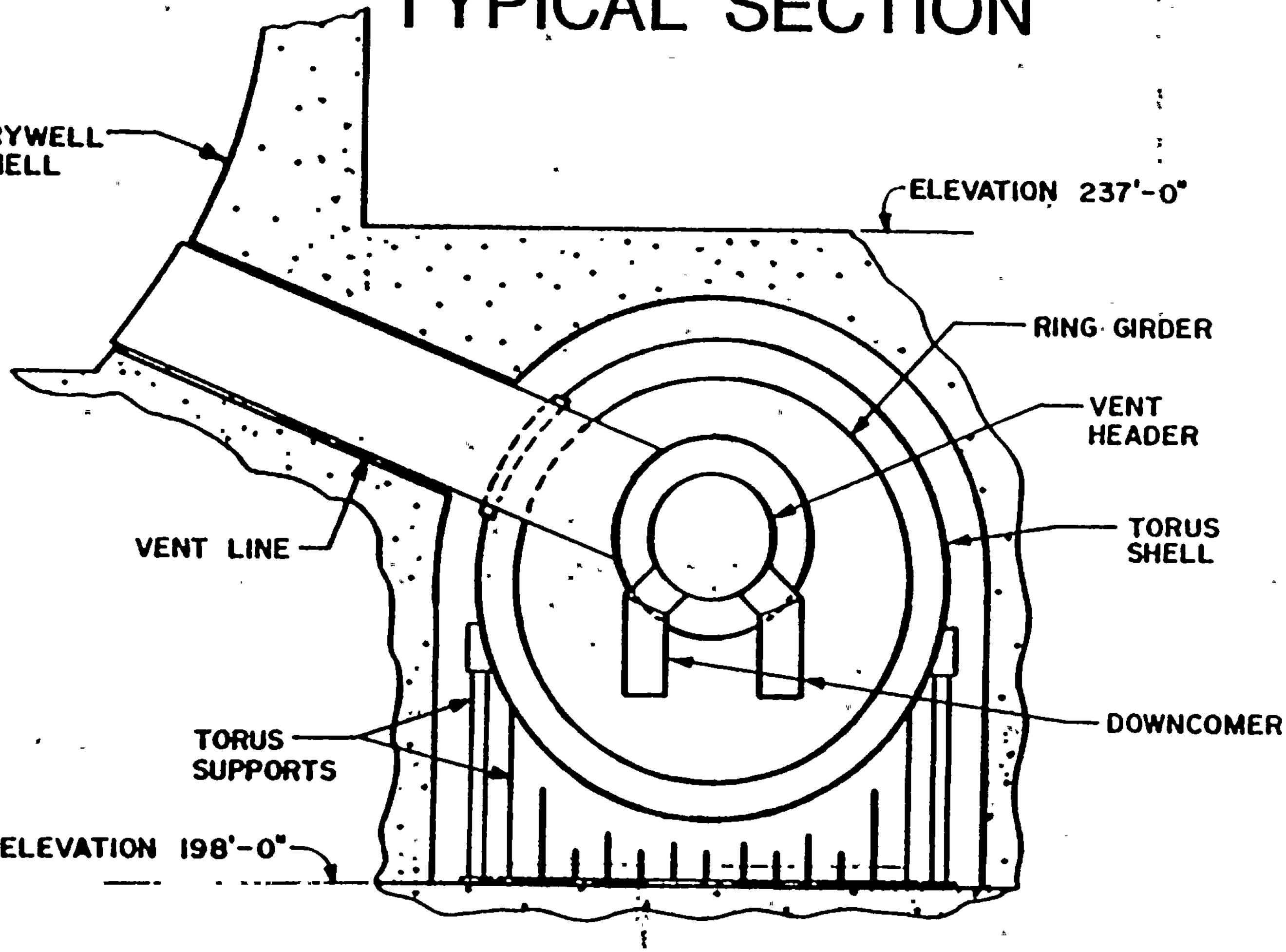
NINE MILE POINT UNIT 1 TORUS



PLAN VIEW OF TORUS (NTS)

TORUS CO LOAD REDUCTION

TYPICAL SECTION



SECTION A-A (NTS)

TORUS CO LOAD REDUCTION

COMPARISON OF NINE MILE POINT & FSTF

	NMP U1	FSTF
NUMBER OF BAYS	20	1 (16)
NUMBER OF DOWNCOMERS/BAY	8 4	8
POOL AREA/DOWNCOMER AREA	28.3	21.2
(8 DOWNCOMER BAY)	21.5	
(4 DOWNCOMER BAY)	42.0	

TORUS CO LOAD REDUCTION
OUTLINE

- BACKGROUND/FSTF

- NINE MILE POINT UNIT 1 TORUS

-CONSERVATISMS IN THE TORUS CO SHELL LOAD

- OUTLINE OF A MULTI-BAY HYDRODYNAMIC MODEL USED TO MODIFY THE MARK I TORUS CO SHELL LOAD

- UTILIZATION OF RESULTS

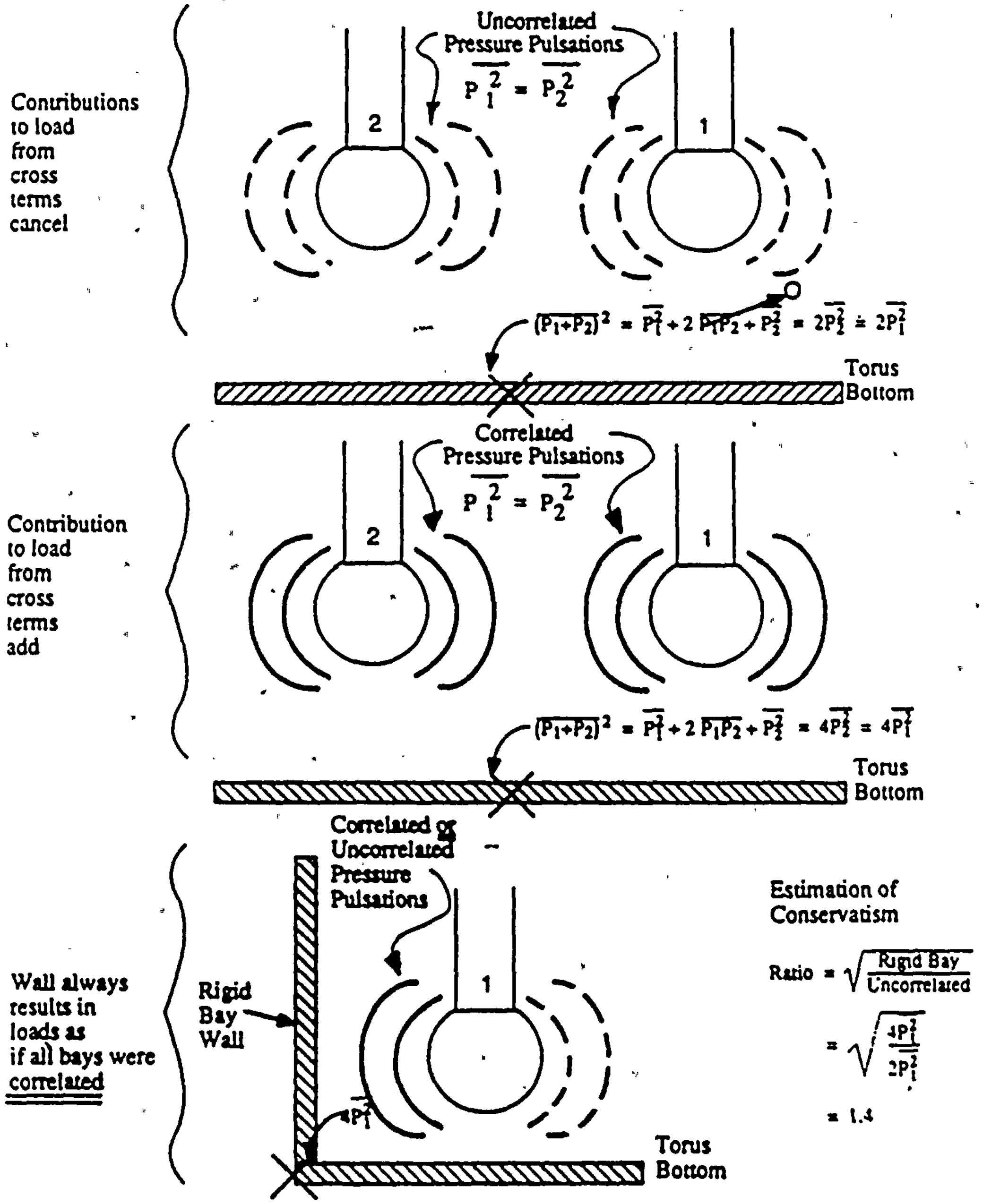
- ADDITIONAL CONSERVATISMS IN TORUS SHELL

TORUS CO LOAD REDUCTION

CONSERVATISMS-END CAP EFFECT

- INTERNAL REPORT PREPARED BY CONTINUUM DYNAMICS ON END CAP EFFECT
  - PRESENTED TO MARK I OWNERS GROUP IN 1983
- CONCLUDED THAT:
  - END CAPS SIMULATED CONDITIONS THAT ALL BAYS ARE CORRELATED
  - THE TORUS BOTTOM CENTER PRESSURE MEASURED IN THE FSTF MAY BE CONSERVATIVE BY UP TO 27%

TORUS CO LOAD REDUCTION
SIMPLIFIED MODELS



TORUS CO LOAD REDUCTION

END CAP EFFECT (CONT.)

- REVIEWED WITH MARK I OWNERS GROUP - INFORMAL CONCLUSIONS
  - END CAPS INTRODUCE CO CONSERVATISMS BUT NRC HAS ALREADY ACCEPTED GENERIC LOAD DEFINITION
  - TOO LATE TO INFLUENCE MOST LONG TERM CONTAINMENT DECISIONS
  - ADDITIONAL CO LOAD REDUCTIONS NOT NEEDED AT THAT TIME
  - THEREFORE DO NOT USE AT THIS TIME (POSSIBLE USE BY INDIVIDUAL UTILITIES FOR FUTURE LOAD REDUCTIONS)

TORUS CO LOAD REDUCTION

OUTLINE

- BACKGROUND/FSTF
- NINE MILE POINT UNIT 1 TORUS
- CONSERVATISMS IN THE TORUS CO SHELL LOAD
- **OUTLINE OF A MULTI-BAY HYDRO-DYNAMIC MODEL USED TO MODIFY THE MARK I TORUS CO SHELL LOAD**
- UTILIZATION OF RESULTS
- ADDITIONAL CONSERVATISMS IN TORUS SHELL

TORUS CO LOAD REDUCTION

NMP1 CURRENT ANALYSIS PLAN

- UTILIZE MULTI-BAY HYDRODYNAMIC MODEL AND APPLY SPECIFICALLY TO NINE MILE POINT UNIT 1 TO PROVIDE A MORE REALISTIC CONDENSATION OSCILLATION TORUS SHELL LOAD
- THIS MULTI-BAY HYDRODYNAMIC MODEL TAKES INTO ACCOUNT
  - UNCORRELATED STEAM CONDENSATION
  - ALTERNATING 8 AND 4 DOWNCOMER BAYS

APRIL 24, 1991

TORUS CO LOAD REDUCTION

NMP1 TORUS PLAN VIEW

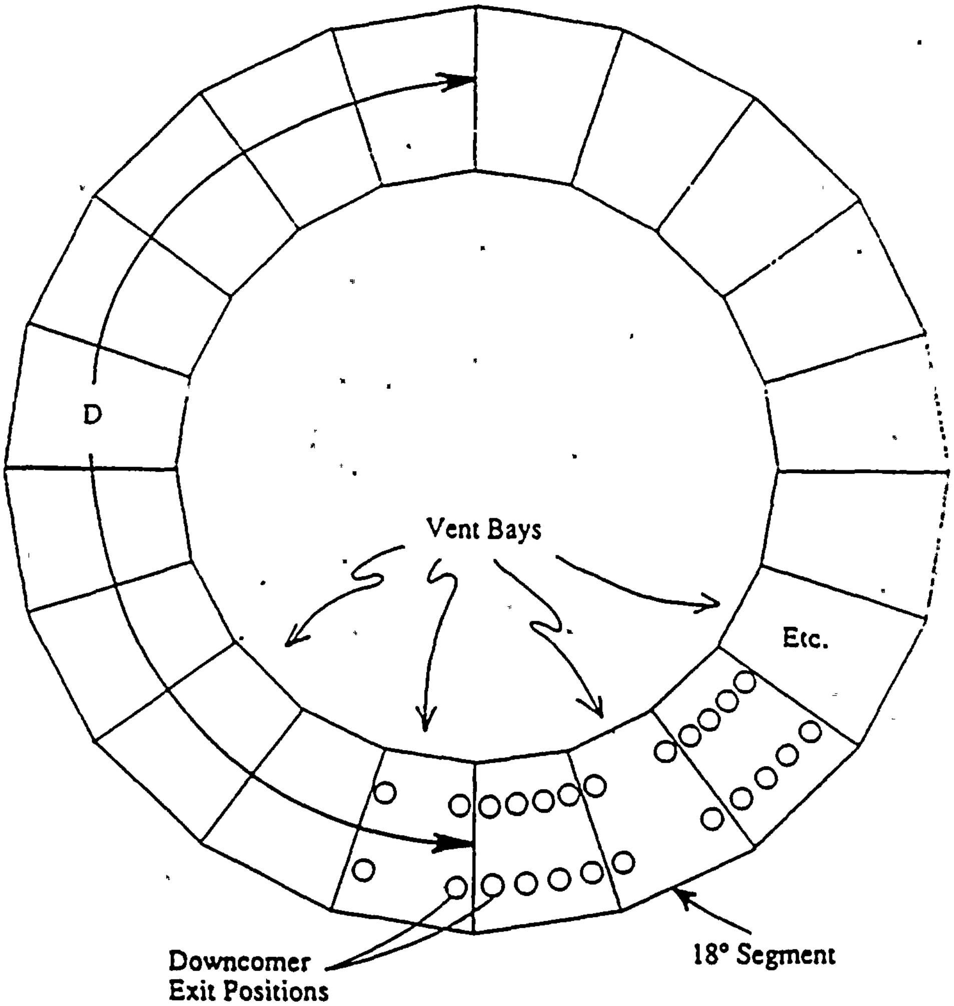


Figure 3. Plan view of Nine Mile Point suppression pool showing 8-4-8-4 downcomer/bay geometry. (Not to Scale)

TORUS CO LOAD REDUCTION

COMPUTATION OF LOAD REDUCTION FACTORS

FOR CORRELATED CONDENSATION:

1) DETERMINE THE SOURCE STRENGTH AT EACH DOWNCOMER EXIT SUCH THAT Pav = 1 UNIT ON THE TORUS BOTTOM FOR FSTF PHYSICAL DIMENSIONS.

2) SUMMING THE PRESSURE FOR EACH FREQUENCY (ω) AND LOCATION (Z) OF EACH DOWNCOMER IN NINE MILE POINT ACCORDING TO:

$$\text{LOAD REDUCTION FACTOR} = \sum_{d=1}^{120} \overline{Pav}(z, \omega)$$

TORUS CO LOAD REDUCTION

COMPUTATION OF LOAD REDUCTION FACTORS

FOR UNCORRELATED CONDENSATION:

1) DETERMINE THE UNCORRELATED SOURCE STRENGTH WITH RESPECT TO THE CORRELATED SOURCE STRENGTH WITH Pav = 1 UNIT ON THE TORUS BOTTOM

UNCORRELATED SOURCE STRENGTH =

$$\sqrt{8}\ \text{(CORRELATED SOURCE STRENGTH) IN FSTF}$$

2) SUMMING THE PRESSURE FOR EACH FREQUENCY (ω) AND LOCATION (z) OF EACH DOWNCOMER IN NINE MILE POINT ACCORDING TO:

$$\text{LOAD REDUCTION FACTOR} = \left(\sum_{d=1}^{120} P_{av}^{\,2}(z,\omega) \right)^{1/2}$$

TORUS CO LOAD REDUCTION

PRESSURE AMPLITUDE LOAD REDUCTION

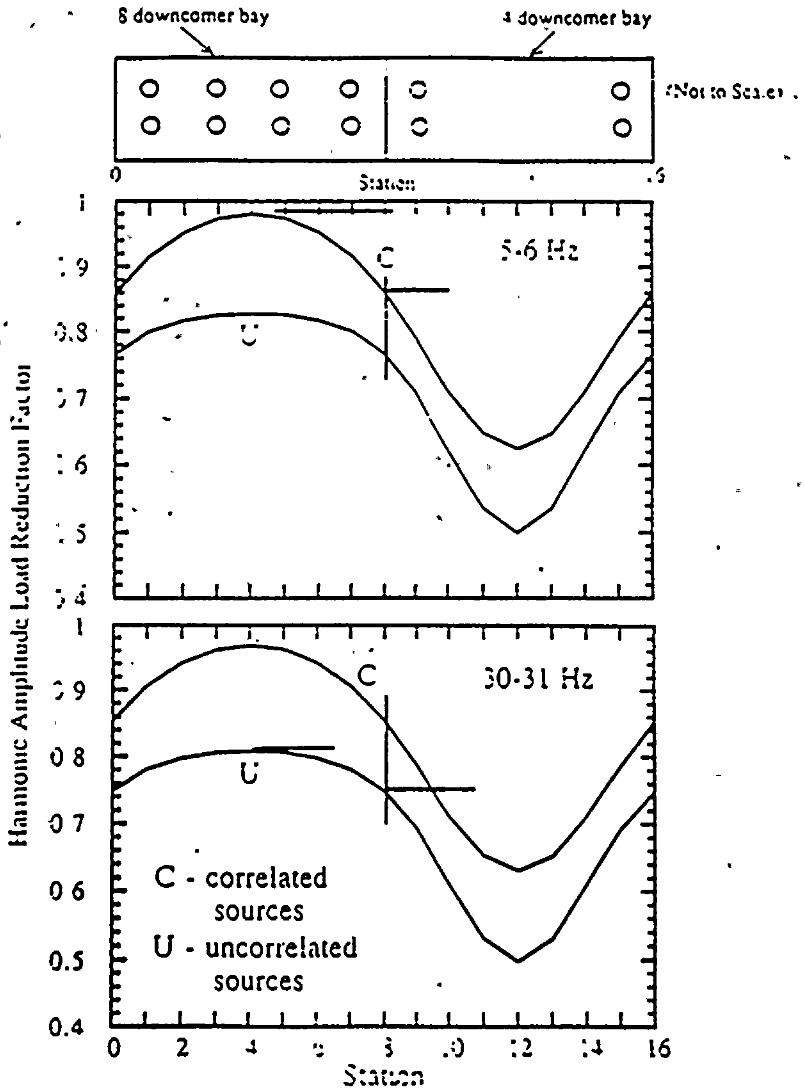


Figure 6. Harmonic amplitude load reduction factor as a function of frequency Acoustic speed = 5000 ft/sec for Nine Mile Point.

TORUS CO LOAD REDUCTION

CONDENSATION OSCILLATION RIGID WALL

PRESSURE AMPLITUDE REDUCTION FACTORS

FOR NMP1

FREQ. RANGE (Hz)	REDUCTION FACTOR - 8 DOWNCOMER BAY AVERAGE VALUE	REDUCTION FACTOR - 4 DOWNCOMER BAYS AVERAGE VALUE
0-1	0.81	0.62
1-2	0.81	0.62
2-3	0.81	0.62
3-4	0.81	0.62
4-5	0.81	0.62
5-6*	0.94	0.72
6-7	0.81	0.62
7-8	0.81	0.62
8-9	0.81	0.62
9-10	0.81	0.62
10-11	0.81	0.62
11-12	0.81	0.62
12-13	0.81	0.62
13-14	0.81	0.62
14-15	0.81	0.62
15-16	0.81	0.62
16-17	0.80	0.62
17-18	0.80	0.62
18-19	0.80	0.62
19-20	0.80	0.62
20-21	0.80	0.62
21-22	0.80	0.62
22-23	0.80	0.62
23-24	0.80	0.62
24-25	0.80	0.62
25-26	0.80	0.62
26-27	0.80	0.62
27-28	0.79	0.62
28-29	0.79	0.61
30-31	0.79	0.61
31-32	0.79	0.61
32-33	0.79	0.61
33-34	0.79	0.61
34-35	0.79	0.61
35-36	0.78	0.61
36-37	0.78	0.61

* VALUES ARE FOR CORRELATED CASE

TABLE 1

TORUS CO LOAD REDUCTION

CONDENSATION OSCILLATION RIGID WALL

PRESSURE AMPLITUDE REDUCTION FACTORS

FOR NMP1 (continued)

FREQ. RANGE (Hz)	REDUCTION FACTOR - 1 DOWNCOMER BAY AVERAGE VALUE	REDUCTION FACTOR - 4 DOWNCOMER BAYS AVERAGE VALUE
37-38	0.78	0.61
38-39	0.78	0.61
39-40	0.78	0.61
40-41	0.77	0.61
41-42	0.77	0.60
42-43	0.77	0.60
43-44	0.77	0.60
44-45	0.77	0.60
45-46	0.76	0.60
46-47	0.76	0.60
47-48	0.76	0.60
48-49	0.76	0.60
49-50	0.76	0.59

TABLE 1

TORUS CO LOAD REDUCTION

OUTLINE

- BACKGROUND/FSTF
- NINE MILE POINT UNIT 1 TORUS
- CONSERVATISMS IN THE TORUS CO SHELL LOAD
- OUTLINE OF A MULTI-BAY HYDRO-DYNAMIC MODEL USED TO MODIFY THE MARK I TORUS CO SHELL LOAD
- **UTILIZATION OF RESULTS**
- ADDITIONAL CONSERVATISMS IN TORUS SHELL

TORUS CO LOAD REDUCTION

USE OF LOAD REDUCTION FACTORS IN TORUS STRUCTURAL ANALYSIS

- LOAD REDUCTION FACTOR FROM TABLE 1 ARE MULTIPLIED TO THE CONDENSATION OSCILLATION BASELINE RIGID WALL PRESSURE AMPLITUDE IN TORUS SHELL BOTTOM DEAD CENTER AS GIVEN IN TABLE 4.4.1-2 IN THE MARK I LOAD DEFINITION REPORT

- LOAD NOW DIFFERS FOR 8 DOWNCOMER & 4 DOWNCOMER BAYS

- TAKES CREDIT FOR UNCORRELATED STEAM CONDENSATION AT APPLICABLE FREQUENCIES

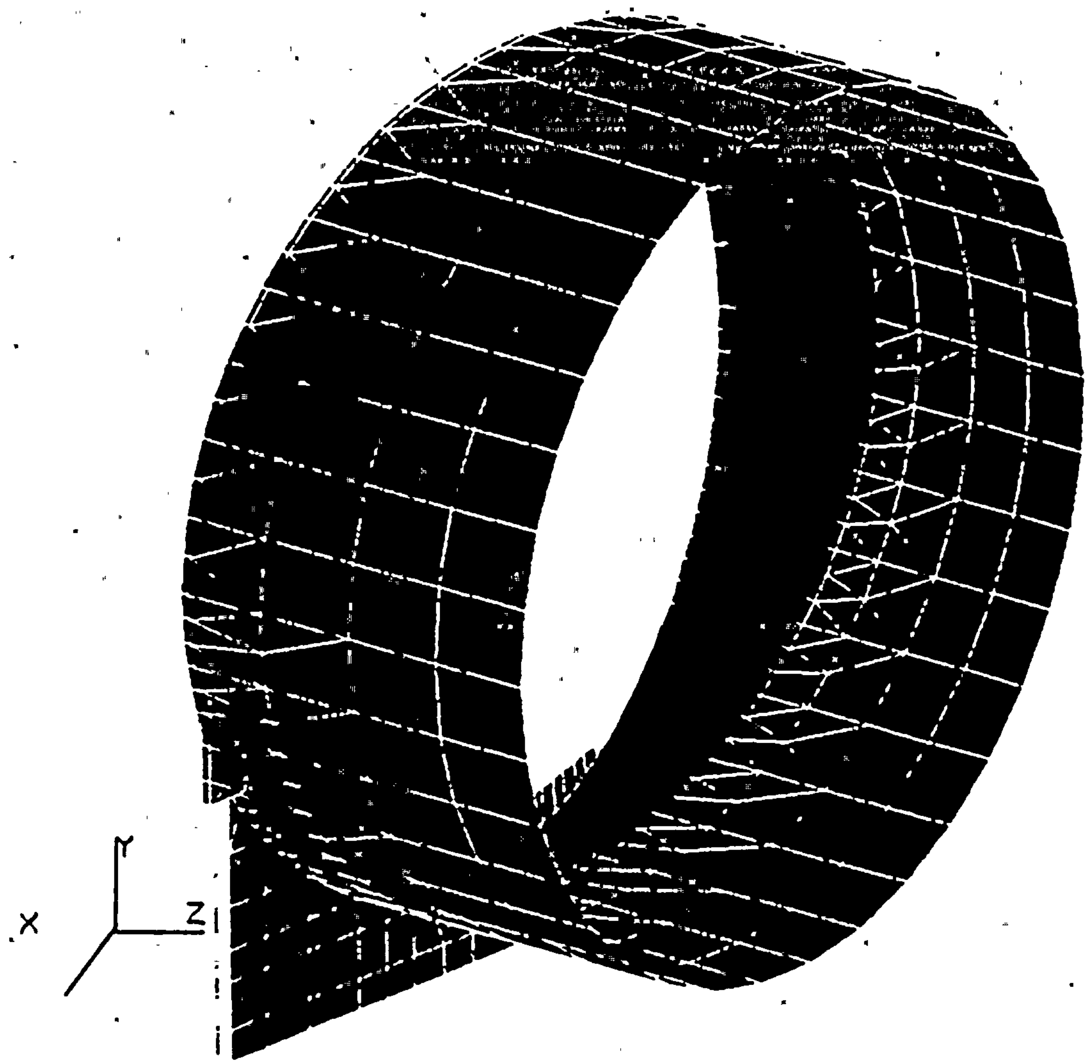
- COMPLETE ANALYSIS AS PER LOAD DEFINITION REPORT

TORUS CO LOAD REDUCTION
COMPUTER MODEL

- PLANT UNIQUE ANALYSIS REPORT FOLLOWED FOR STRUCTURAL ANALYSIS
  - REVIEWED AND ACCEPTED BY THE NRC
  - USE SAME COMPUTER MODEL AS USED IN THE ORIGINAL TORUS ANALYSIS
  - USES COMPUTER CODE "STARDYNE" FOR FINITE ELEMENT ANALYSIS
  - FINITE ELEMENT MODEL IS A 9° SECTOR OF THE TORUS

TORUS CO LOAD REDUCTION

DETAILED SHELL MODEL



TORUS CO LOAD REDUCTION

OUTLINE

- BACKGROUND/FSTF
- NINE MILE POINT UNIT 1 TORUS
- CONSERVATISMS IN THE TORUS CO SHELL LOAD
- OUTLINE OF A MULTI-BAY HYDRO-DYNAMIC MODEL USED TO MODIFY THE MARK I TORUS CO SHELL LOAD
- UTILIZATION OF RESULTS
- **ADDITIONAL CONSERVATISMS IN TORUS SHELL**

TORUS CO LOAD REDUCTION

ADDITIONAL CONSERVATISMS

ADDITIONAL CONSERVATISMS IN TORUS STRUCTURAL ANALYSIS THAT HAVE NOT BEEN TAKEN CREDIT FOR ARE AS FOLLOWS:

- MILL CERTIFICATIONS
- ACOUSTIC SPEED
- POOL DAMPING

TORUS CO LOAD REDUCTION

MILL CERTIFICATIONS

- MILL CERTIFICATIONS FOR NMP1 TORUS WOULD PERMIT A HIGHER ALLOWABLE

S = 17600 psi BASED ON MILL CERTIFICATIONS (FOR A-201 STEEL)*

S = 16500 psi BASED ON ASME CODE

* REF TELEDYNE REPORT TR-6801-2 "MARK I TORUS SHELL REQUIREMENTS, NINE MILE UNIT I NUCLEAR STATION"

TORUS CO LOAD REDUCTION

ACOUSTIC SPEED

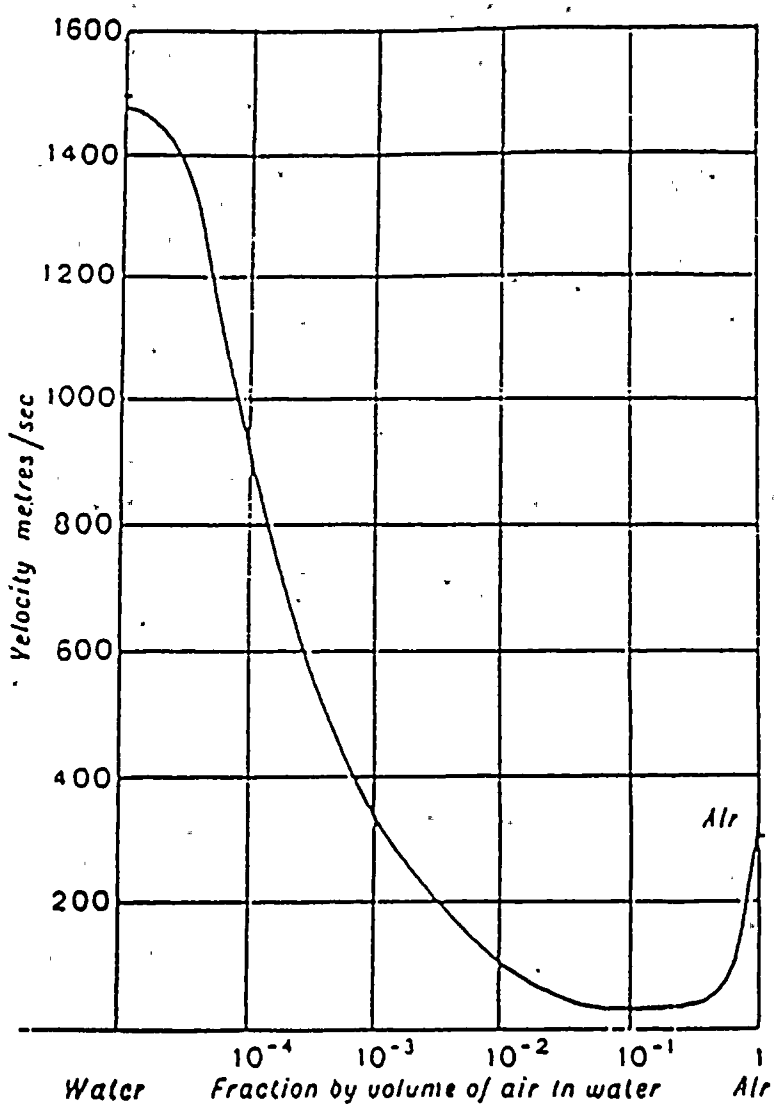


FIG. 100—Velocity of Sound in Air-Water Mixtures

TORUS CO LOAD REDUCTION

HARMONIC AMPLITUDE LOAD REDUCTION FACTOR

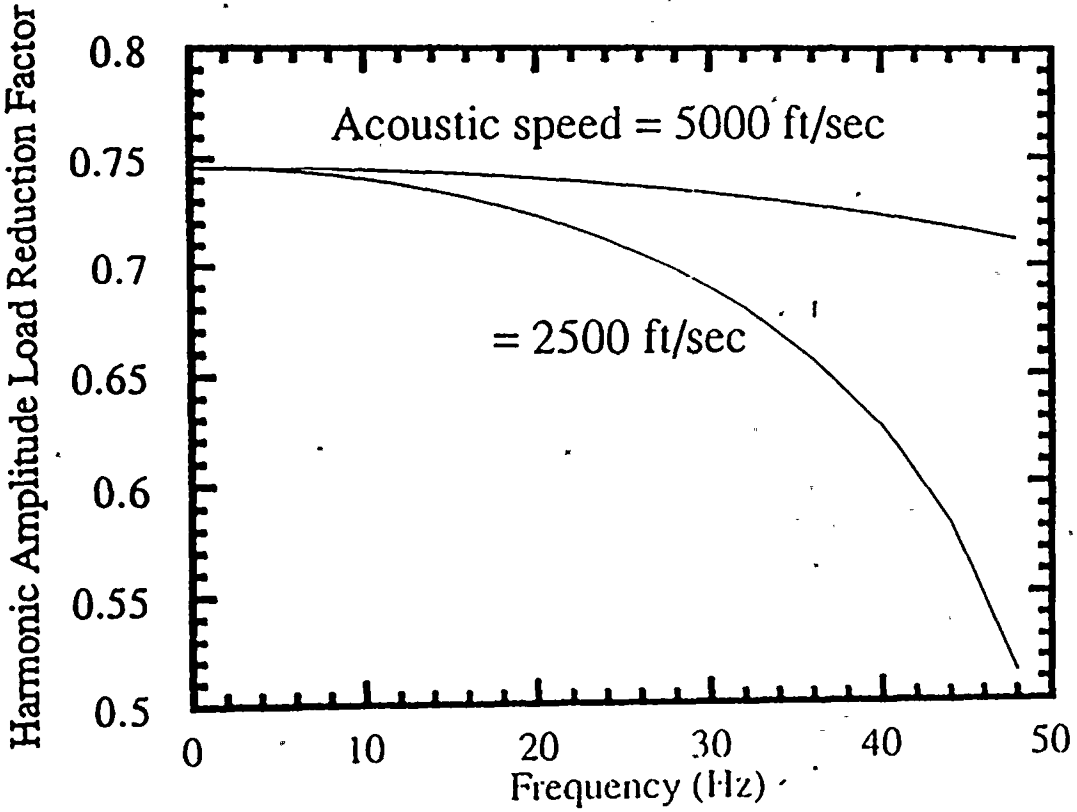


Figure 5 Harmonic amplitude load reduction factor (uncorrelated sources) for FSTF

TORUS CO LOAD REDUCTION
POOL DAMPING CONSERVATISMS

- CHARACTERISTICS OF THE BUBBLY POOL WATER WOULD REDUCE THE LOADS FURTHER THROUGH DAMPING

- THIS EFFECT WAS NOT QUANTIFIED IN FSTF

(NOT CONSIDERED IN ANALYSIS)

TORUS CO LOAD REDUCTION

CONCLUSION

- TAKE CREDIT FOR UNCORRELATED STEAM CONDENSATION
- TAKE CREDIT FOR PLANT SPECIFIC DOWNCOMER CONFIGURATION (i.e. 4 DOWNCOMER BAYS)
- THIS PROVIDES REDUCTION IN THE CONDENSATION OSCILLATION TORUS SHELL LOAD BASED ON A MORE REALISTIC HYDRODYNAMIC MODEL
- LOAD IS STILL CONSERVATIVE BASED ON OTHER CONSERVATISMS

TORUS CO LOAD REDUCTION
CONCLUSION (CONT'D)

CORROSION ALLOWANCE

CONDITION	CORROSION ALLOWANCE, IN.	APPROXIMATE YEAR AVERAGE CORROSION ALLOWANCE WILL BE CONSUMED*
ORIGINAL ANALYSIS	.0132	1994
REDUCED C.O. 8 D.C. BAY	.0292	$\frac{.0292 - .0132}{.00126} + 1994 = 2007$
REDUCED C.O. 4 D.C. BAY	.0569	$\frac{.0569 - .0132}{.00126} + 1994 = 2029$

***AT A CORROSION RATE OF .00126" PER YEAR**

SUMMARY

SUMMARY

- SIGNIFICANT MARGIN IN ORIGINAL DESIGN BASIS OF DRYWELL
- DRYWELL VENTING BELOW DESIGN PRESSURE
- CONTAINMENT ULTIMATE STRENGTH (TORUS) ADEQUATE FOR LONG TERM LOADS
- LONG TERM CONTAINMENT LOADS CAN BE DECOUPLED FROM SHORT TERM HYDRODYNAMIC LOADS

SUMMARY

- CONSERVATISMS IN TORUS SHELL
  - MILL CERTIFICATIONS
  - ACOUSTIC SPEED
  - DAMPING
- CONSERVATISMS IN CORROSION PROGRAM
  - EXTENSIVE DATA COLLECTION
  - LONG TERM MONITORING AND TRENDING PROGRAM ESTABLISHED
  - AVERAGE CORROSION RATE .83 MIL/YR
  - CONSERVATIVE PREDICTION OF CORROSION RATE 1.26 MIL/YR
  - ESTIMATED STRESS IN MOST LIMITING PLATE - 9/94

    16381 PSI - BASED ON .83 MIL/YR
    16528 PSI - BASED ON 1.26 MIL/YR

NMPC PROPOSAL

- DEFER MODIFICATION OF 8 DOWNCOMER BAYS UNTIL 1994 REFUELING OUTAGE
- FORMAL SUBMITTAL OF CO PROGRAM DETAILS BY MAY 15, 1991
- REQUEST RESPONSE ON THESE ISSUES FROM NRC BY DECEMBER, 1991

5. Details of stiffeners installed on downcomers to prevent movement of the downcomers.

6. Individual data point (65 points in 1' x 3' area) results of 6-month measurements of torus wall thickness.

7. Coating study reference material.

8. Consideration of a surveillance program to monitor water leakage from the torus.

9. Various modification options, schedules, and resource impacts that Niagara Mohawk has considered. These options, schedules, and resource impacts should consider contingencies for staff review of the May 15, 1991, submittal not being completed by December 1991.

The licensee agreed to supply the requested additional information. This additional information will either be included with the May 15, 1991, submittal, if possible, or will be provided later as an additional submittal.

ORIGINAL SIGNED BY:

Donald S. Brinkman, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. List of Attendees
2. Licensee Handout Material

cc w/enclosures:
See next page

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Docket No. 50-220

APR 2 5 1991

Central Files

P 137

MEMORANDUM FOR: Donald Brinkman, Project Manager
Project Directorate I-1
Division of Reactor Projects I/II

FROM: James A. Norberg, Chief
Mechanical Engineering Branch
Division of Engineering Technology

SUBJECT: NINE MILE POINT UNIT 1 (CTS-RR-2 AND VG-2) IST RELIEF REQUESTS (TAC NO. 79447)

The Mechanical Engineering Branch has completed a review of relief requests CTS-RR-2 AND VG-2 proposed by Niagara Mohawk Power Corporation for the Nine Mile Point Unit 1 in a letter dated November 11, 1990.

Our Safety Evaluation is provided in Enclosure 1. SALP input is provided in Enclosure 2. This completes the action requested under TAC No. 79447.

/s/

James A. Norberg, Chief
Mechanical Engineering Branch
Division of Engineering Technology

Enclosures:
As stated

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memo 4/40

ENCLOSURE 1

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO INSERVICE TESTING PROGRAM RELIEF REQUESTS FOR
NINE MILE POINT UNIT 1
DOCKET NO. 50-220
TAC NO. 79447

Relief Request CTS-RR-2

The licensee has requested relief from the check valve full-stroke exercising requirement of Section XI, Paragraph IWV-3522 for the containment spray pump discharge valves 80-05, 80-06, 80-25 and 80-26. The licensee has proposed to conduct partial flow tests on an interim basis until modifications can be made to the system to permit access for disassembly or use of non-intrusive testing to demonstrate that the check valves will swing to a full open position under partial flow conditions.

Licensee Basis for Requesting Relief

These check valves are the pump discharge check valves. They are split body (flange) tilting disc check valves with the valve-to-pipe joint welded into the discharge line. These valves are tested quarterly during the surveillance test of the containment spray pump. The flow path during the quarterly test uses a downstream branch line that returns flow to the torus. The test flow rate is limited to approximately 2900 gpm (two loops achieve almost 3000 gpm due to the piping configuration of the cross connect header and the single test line to the torus). ASME Section XI requires forward flow opening be verified at full flow conditions.

Testing and subsequent analysis performed during late 1989 and early 1990 determined that an increase of flow rate from 3000 to 3300 gpm is necessary to assure adequate post-accident cooling of the suppression pool (torus) water at elevated lake temperatures (above 76 F). The normal operating system flow path is from the torus to the containment spray headers which is not available for inservice testing (e.g., spraying the drywell could damage equipment and require extensive cleanup and testing to be performed). Therefore, testing is conducted utilizing the test line at a flow rate of 2900 gpm versus the full flow rate of 3300 gpm.

Relief from the ASME XI requirement to perform full flow testing on these check valves is based on the following; 1) the manufacturer has indicated the valves will be fully open at a flow rate of 2200 GPM, 2) near full flow rate is achieved with the torus to torus test.

Alternate Testing

On an interim basis, the near full flow test (e.g., the quarterly torus-to-torus pump test) will be used to satisfy the forward flow opening.

By the 1992 refueling outage, an alternate arrangement (e.g., a modification to permit access for disassembly and examination, non-intrusive examination techniques, etc.) will be implemented as the long term solution for forward flow opening. A followup relief request, if required, will be submitted once an alternate arrangement has been implemented.

Evaluation

These check valves are not equipped with removable bonnets, inspection ports, position indication devices, or other means to verify their full stroke capability. A flow rate of 2900 gpm (approximately 85% of required flow) can be established through these valves by pumping from the suppression pool and returning the water back to the torus during quarterly pump testing. The licensee has indicated a system modification will be performed during their 1992 refueling outage. This will allow access to these check valves in order to perform disassembly and inspection, or to permit the use of non-intrusive testing to verify full check valve exercising with partial flow. The licensee has not specified which method it will employ.

Significant system modifications would be needed to pass the required design-basis flow rate through these valves. Since the required system flow rate of 3300 gpm cannot be passed through these valves with the existing piping configuration, credit cannot be taken for a full-stroke exercise. Passing the minimum flow rate that should fully open the valve disk as indicated by the valve manufacturer is not a valid method of full-stroke exercising valves. The manufacturer's information is based on valves in good condition and does not apply to valves that might be degraded or fouled by foreign materials. If the flow rate achieved through the valve during quarterly testing opens the valve to the back stop or to the position needed to pass the required system flow rate of 3300 gpm, and this can be verified using non-intrusive techniques, such as ultrasonic, magnetic, or acoustic, this would constitute a full-stroke exercise of the valve. If this can be performed, the licensee should ensure that the techniques used are qualified using the guidance described in NRC GL 89-04, Position 1.

Disassembly and inspection on a sampling basis may be an acceptable method to use to assess valve condition when individually exercising valves with system flow cannot be verified. However, the NRC staff considers valve disassembly and inspection to be a maintenance procedure that is not equivalent to the exercising produced by fluid flow. This procedure has risks which make its routine use as a substitute for testing unacceptable when some method of testing is possible. The NRC staff positions regarding valve disassembly and inspection are explained in detail in Generic Letter 89-04, Attachment 1, Item 2. The minutes from the public meetings on Generic Letter 89-04 regarding Item 2 further stipulate that a partial-stroke exercise test using flow is expected to be performed after disassembly and inspection is completed but before the valve is returned to service.

In order to satisfy the exercise requirement with full flow, the licensee would have to design and install a larger capacity containment spray test line. This requirement would be an excessive burden on the licensee because of the costs involved. Also, this type of system modification would likely decrease the reliability of the containment spray system. The licensee has proposed to use the partial flow test until alternate methods to verify check valve position can be examined. This current testing should adequately demonstrate operational readiness for an interim period of time because a large percentage of the design-basis flow is being passed through the check valves with the partial flow test.

Based on the impracticality of full stroke exercising these valves with the existing piping configuration and test methods, the burden on the licensee if the Code requirements were imposed, and the acceptability of the licensee's proposed alternatives, relief may be granted pursuant to 10 CFR 50.55a(g)(6)(i) for an interim period of one year or until the next refueling outage, whichever is longer. During this interim period, the licensee should evaluate alternate methods to verify check valve full stroke capability. A relief request should be submitted once their alternate testing has been selected.

General Relief Request VG-2

The licensee has requested relief from the trending requirements of Section XI, paragraph IWV-3427(b) for containment isolation valves designated LJ and LA and relief from the leak rate testing requirements of paragraphs IWV-3421 through 3425 as well as the trending requirements of IWV-3427(b) for pressure isolation valves designated LK. The licensee has proposed testing containment isolation valves designated LJ in accordance with Appendix J in lieu of IWV-3421 through 3425 and proposed testing pressure isolation valves designated LK in accordance with Nine Mile Point Unit 1 (NMP1) Technical Specification (TS) Section 3.2.7.1.

Licensee Basis for Requesting Relief

There are three types of leakage tests performed at NMP1. These tests are designated as either LA, LJ, or LK in the test requirement column of the Valve Tables. A description of each test is contained in the following paragraphs.

Containment isolation valves (CIVs) are required to be leakage rate tested in accordance with 10 CFR 50, Appendix J. These valves are designated as LJ valves in the test requirement column of the Valve Tables. The leakage rate requirement is based on a total allowable leakage rate for all valves instead of an individual valve leakage rate. IWV-2200(a) defines Category A as "valves for which seat leakage is limited to a specific maximum amount in the closed position of fulfillment of their function." Although, leakage rates for containment isolation valves are not limited on an individual basis, they have been determined to be Category A valves. Since containment isolation valves are Category A, the leakage rate testing requirements of IWV-3420 must be satisfied. The leakage rate testing performed per Appendix J satisfies the intent of IWV-3421 through 3425. However, it does not satisfy the individual valve leakage rate analysis and corrective actions specified in IWV-3426 and IWV-3427, respectively. In order to prevent duplicate leakage testing of these valves, individual leakage rates will be obtained during Appendix J testing and the requirements of IWV-3426 and 3427(a) will be applied via separate procedure.

The second type of leakage tests are valves that have primarily been included in the IST Program as a result of NMP1 10 CFR 50, Appendix J, testing commitments. These valves, which are designated as LA valves in the test requirement column of the Valve Tables, are containment isolation valves that are tested with water in accordance the IWV-3421 through IWV-3427(a) rather than with air in accordance with 10 CFR 50, Appendix J.

The third type of leakage tests are pressure isolation valves. These valves are designated as LK valves in the test requirement column of the Valve Tables. They are leakage tested in accordance with NMP1 TS Section 3.2.7.1 rather than IWV-3420. This is permitted by Generic Letter 89-04, Position 4, which states that pressure isolation valve testing should be performed in accordance with Plant TS and referenced as such in the IST Program.

As outlined in Generic Letter 89-04, Position 10, the usefulness of IWV-3427 "Corrective Action" part (b) requirement does not justify the burden of compliance with this requirement for valves tested in accordance with 10 CFR 50, Appendix J (air leakage tests for CIVs). Relief is requested from the requirements of IWV-3427 (b) for NMP1-LJ valves based on position 10 of GL 89-04. Similarly, based on a review of NMP1 historical water leakage test results, the usefulness of IWV-3427(b) does not justify the burden of complying with this requirement for LA and LK valves.

Alternate Testing

The NMP1 leakage test program will be conducted as follows:

1. 10 CFR 50, Appendix J containment isolation valves (LJ).

LJ containment isolation valves will be leak rate tested in accordance with the 10 CFR 50, Appendix J, testing program. In addition, individual valve leakage rates will be obtained by test or analysis and the requirements of IWV-3426 and 3427(a) will be applied via a separate procedure for those valves that are Appendix J, Type C, tested. The trending required by IWV-3427(b) will not be performed.

2. NMP1/NRC 10 CFR 50, Appendix J commitments (LA).

LA containment isolation valves will be leakage rate tested with water in accordance with ASME Code Section XI, IWV-3420. The trending required by IWV-3427(b) will not be performed.

3. Pressure Isolation Valves (LK).

LK pressure isolation valves will be leakage rate tested and will have corrective action taken in accordance with NMP1 TS Section 3.2.7.1 versus IWV-3420. The trending required by IWV-3427(b) will not be performed.

Evaluation

LJ Valves: The 10 CFR 50, Appendix J, Type C, leak rate testing requirements essentially meet the ASME Code Section XI, leak rate requirements of paragraphs IWV-3421 through 3425 since these Appendix J requirements incorporate all of the major elements of these paragraphs. The licensee's proposal to comply with the leak test procedures and requirements identified in 10 CFR 50, Appendix J, for containment isolation valves in lieu of the requirements of Section XI, Paragraphs IWV-3421 through 3425, provides an acceptable level of quality and safety. Further, the licensee will comply with the "Analysis of Leakage Rates" and "Corrective Action" requirements of paragraphs IWV-3426 and 3427(a). Industry experience has demonstrated that the corrective actions of IWV-3427(b) are not meaningful for containment isolation valves because valve leakage rates vary widely from test to test due primarily to the valves seating differently; therefore, variations in valve leakage rates may not be due to valve degradation and the Code criteria could require corrective actions on valves that are in good condition. Additionally, the licensee's proposal is in accordance with the NRC staff position as stated in GL 89-04, Position 10, which provides a reasonable alternative to the Code requirements.

Based on the determination that the licensee's proposal provides an acceptable level of safety and is in accordance with GL 89-04, Position 10, relief should be granted as requested per 10 CFR 50.55a(a)(3)(i).

LA Valves: These containment isolation valves are torus suction check valves and are classified as Category A as defined by IWV-2200(a). The licensee has proposed testing these valves under IWV-3520 using water and has requested relief from these requirements of IWV-3427(b). In a telecon with the licensee on April 17, 1991, representatives of Niagara Mohawk Power Corporation explained that the leak rate acceptance criteria for these valves at NMP1 is ½ gpm per inch of pipe diameter up to 5 gpm. It was also explained that procedures require that if the tested valve leakage exceeds the acceptable criteria, the valve is repaired prior to being returned to service. No trending of leakage rates is performed.

The licensee's leakage criteria for these valves are judged by the staff to be conservative due to the volume of the water in the torus and the plant's capabilities to makeup to the torus. Since repair of valves is performed whenever the acceptance criteria are exceeded, trending per IWV-3427(b) could result in unnecessary additional testing and is not considered essential.

Based on the conclusion that the licensee's alternative testing requirements provide an acceptable level of quality and safety, relief may be granted pursuant to 10 CFR 50.55(a)(3)(i).

LK Valves: Paragraph IWV-3427(b) requires that if the valve leakage rate trending shows the valve will exceed the 5 gpm leakage rate limit on the next test, the valve shall be replaced or repaired. Also, if the leakage rate test results reduce the margin between the previously measured leakage and the limiting leakage rate by 50%, the testing frequency shall be doubled. The licensee's proposal to use their plant TS results in testing virtually identical to the requirements for IWV-3527(b). The only exception is the licensee's TS exclude leakage rates below 1.0 gpm from trending.

Based on the conclusion that the licensee's alternative testing is almost the same as the Code requirements and provides an acceptable level of safety, relief may be granted pursuant to 10 CFR 50.55(a)(3)(i).

ENCLOSURE 2

Docket No.: 50-220

SALP REPORT

LICENSEE: Niagara Mohawk Power Corporation

REVIEWER: J. Colaccino

FUNCTIONAL ACTIVITY: IST Relief Requests - TAC No. 79447

FACILITY NAME: Nine Mile Point Unit 1

SUMMARY OF REVIEW/INSPECTION ACTIVITIES

This SE involves two relief requests. Relief request CTS-RR-2 involves verification of the containment spray pump discharge check valve exercising. Relief request VG-2 involves trending leakage rates for containment isolation valves and pressure isolation valves.

NARRATIVE DISCUSSION OF LICENSEES PERFORMANCE - FUNCTIONAL AREA

Relief request VG-2 was written poorly and required additional information from the licensee. Items that were deficient or completely omitted from the relief request included: basis for requesting relief, complete description of alternate testing methods, direct comparison between alternate test methods and Code requirements, and explanation of how alternative test method provides equal protection to that of the Code requirements.

When representatives of Nine Mile Point 1 were questioned on their basis for requesting relief, their response was still weak. Although the staff worked to have the licensee focus on their basis for relief, the licensee continued to emphasize primarily the burden of meeting the Code requirements. The licensee is not completely competent in writing relief requests according to 10 CFR 50.55a and should review the regulations to understand what is required to grant relief from the Code requirements.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

April 18, 1991

Docket Nos. 50-220
and 50-410

MEMORANDUM FOR: Donald R. Haverkamp, Section Chief
Reactor Projects Section 1A
Projects Branch No. 1
Division of Reactor Projects, Region I

THRU: Robert A. Capra, Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

FROM: Donald S. Brinkman, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects
Office of Nuclear Reactor Regulation

SUBJECT: INPUT TO SAFETY ASSESSMENT/QUALITY VERIFICATION FUNCTIONAL AREA FOR NINE MILE POINT NUCLEAR STATION, UNITS 1 AND 2 SALP REPORT

In accordance with James Linville's March 6, 1991, memorandum, the attached information is being forwarded to you as input to the Safety Assessment/Quality Verification functional area of the SALP Report for Nine Mile Point Nuclear Station, Units 1 and 2, for the assessment period which ended March 31, 1991.

Donald S. Brinkman

Donald S. Brinkman, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosure:
SALP Report Input

cc w/enclosure:
W. Cook, Nine Mile Point, SRI

CONTACT:
D. Brinkman
49-21402

memo 4 w

III.G Safety Assessment/Quality Verification

1. Analysis

During the previous assessment period, this functional area was rated as Category 3 with an improving trend. Niagara Mohawk's performance demonstrated some inconsistency, but an overall improving trend was observed. The previous assessment noted an apparent turning point in Niagara Mohawk's approach to assuring quality. The Restart Action Plan was responsible for the better problem identification, more critical problem evaluation and self-assessment, and the establishment of programs and standards to promote and sustain good performance. The approach appeared to have enabled improved results noted in the engineering and surveillance areas and the generally improving direction in most other areas. However, the performance in several areas remained at minimally acceptable levels providing a challenge for Niagára Mohawk management to utilize this better approach to produce improved results on a consistent basis in all aspects of plant operations.

Niagara Mohawk implemented several management changes during this assessment period. These changes included the appointment of a new Executive Vice President, Nuclear; a new Vice President, Nuclear Generation; and a new Plant Manager - Unit 2. A reorganization of the site staff was also implemented. This reorganization provides unitized control of each unit. The transition activities associated with these changes have been successful in sustaining previously initiated performance improvements. The reorganization of the site staff has been successful in improving the accountability of personnel.

During this assessment period, Niagara Mohawk demonstrated continued improvement in this area. Increased management oversight, a conservative attitude, and a good safety perspective in the areas of plant operations and maintenance/surveillance were evident during both routine activities and special evolutions. For example, preparations for the Unit 1 Power Ascension Test Program and the Unit 2 startup following the first refueling outage were comprehensive and thorough. Niagara Mohawk thoroughly reviewed and implemented appropriate lessons learned from another licensee before initiating the Unit 2 turbine torsion test. The operators received special training before initiating this test. Good management oversight was evident during the turbine torsional test and during troubleshooting of the Unit 1 main turbine pressure oscillations.

Generally effective implementation of Niagara Mohawk's standards of performance were observed; however, some isolated inconsistences were present. Implementation of these standards of performance and their reinforcement by accountability meetings were an effective tool in reinforcing individual responsibilities and accountability. An overall improving trend was seen in the area of adherence to these higher performance standards, especially towards the later part of the assessment period. However, occasional lapses were noted in procedural adherence and proper problem identification and resolution. Examples of these lapses include control of Blue Markups at Unit 1, I&C

surveillance testing at Unit 2, implementation of an Appendix J exemption at Unit 1, maintenance of hydraulic control units at Unit 2, and a radwaste building spill at Unit 1.

In response to a noted weakness during the previous assessment period, Niagara Mohawk initiated a good program to ensure the operability of effluent monitors. Although the operability of these monitors has improved, their periods of inoperability were still excessive. Management commitment to improving performance in this area was evident by the assignment of personnel to this program with good technical knowledge of the effluent monitors and by the use of trending analyses to better maintain the operability of effluent monitors.

Procedures for ensuring that plant design changes and modifications were performed in a controlled manner have been established and were effectively implemented by Niagara Mohawk. Sufficient measures were provided for proper technical reviews, independent verifications, appropriate levels of approvals, proper installation, and post-modification testing. The review and approval process ensured that plant changes were evaluated as required by 10 CFR 50.59 to determine if an unreviewed safety question was involved.

Analyses to determine root causes of most events have been thoroughly and effectively performed. A noted exception to this good performance was the evaluation of the Unit 1 feedwater pump Blue Markup. The analysis for this event was not thorough in that management did not consider this to be a programmatic issue but originally chose to focus only on the personnel performance aspect of the event.

Niagara Mohawk's outage management capabilities have been enhanced. Shortcomings in this area have been self-identified and acted upon. During this assessment period, both units instituted a permanent outage group and assigned an outage manager. These groups gained experience and improved their performance throughout the assessment period. They demonstrated an effective outage organization and good overall planning, coordination, and work control during the Unit 1 mid-cycle outage in February-March 1991. This outage was completed ahead of schedule and under the projected ALARA goals.

Niagara Mohawk effectively utilized its self-assessment programs as a management tool. Self-assessments performed during the Unit 1 Power Ascension Test Program were comprehensive and critical. Management was not driven by schedule or capacity factor in ensuring adherence to its standards of performances. Implementation of the self-assessment process became more effective as the Power Ascension Test Program progressed and as Niagara Mohawk made appropriate modifications to improve the process. Self-assessments have continued to function as an effective management tool since completion of the Power Ascension Test Program. Comprehensive and performance-based self-assessments were also effective in providing management oversight in the maintenance of high levels of performance in the functional areas of Security and Emergency Preparedness.

The safety oversight committees (SORC and SRAB) provided a positive performance impact on the station. These committees focused on the safety issues and efficiently reviewed station activities to maintain that focus. The ISEG provided timely and effective reviews of plant and industry events.

The Quality Assurance (QA) organization was strengthened by adding to its staff individuals with an operations background. A reorganization of the QA organization resulted in the Vice President, Quality Assurance, relinquishing all non-nuclear responsibilities and moving his office and staff onsite. The impact and effectiveness of this reorganization has not yet been assessed. The QA audits were indepth and performance-based. Quality Control (QC) surveillances were also performance-based. QC was responsive to station management and provided independent assessments, by special request, of suspected problem areas.

Licensing action submittals have been generally technically sound and thorough. The submittals usually demonstrated sufficient management involvement and oversight so that resolution of the issues was accomplished without requiring additional information, thereby demonstrating a thorough understanding of the issues. License amendment requests have almost always been submitted in a timely manner. However, occasional problems have occurred with some licensing actions. The significant hazards consideration analysis for the Hydrogen Water Chemistry license amendment did not consider the hazards associated with handling and storage of hydrogen. The request for a temporary waiver of compliance for placing an instrument channel in an inoperable status for a limited period of time before placing the channel in a tripped condition was not adequately reviewed by Niagara Mohawk before proposing it to the NRC staff. The initial response to Generic Letter 89-13 did not include sufficient detail to enable the NRC staff to identify that specific actions or assessments would be undertaken for the areas of concern identified in the generic letter on a defined schedule.

Licensee Event Reports (LERs) were well-written and described the major aspects of each event, the system and components involved, and the significant actions taken or planned to be taken to prevent recurrence. An effective method for identifying revisions to previously submitted LERs was implemented. Telephone notifications made pursuant to 10 CFR 50.72 permitted the NRC Operations Officer to accurately describe the events and were provided on a timely basis. A conservative approach in reporting was evident in that events were reported even though reports may not have been specifically required by regulation.

In summary, Niagara Mohawk demonstrated an improved approach to assuring quality and assessing the safety significances of issues affecting plant operations. The self-assessment programs became more effective during the later portions of the assessment period. The new standards of performance and their methods of implementation are effective in articulating management expectations and requirements. Licensing actions were generally technically adequate and timely; however, some submittals did not receive an adequate evaluation before submittal.

2. Performance Rating

Category Trend:

3. Recommendations

Docket Nos. 50-220
and 50-410

April 18, 1991

MEMORANDUM FOR: Donald R. Haverkamp, Section Chief
Reactor Projects Section 1A
Projects Branch No. 1
Division of Reactor Projects, Region I

THRU: Robert A. Capra, Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

FROM: Donald S. Brinkman, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects
Office of Nuclear Reactor Regulation

SUBJECT: INPUT TO SAFETY ASSESSMENT/QUALITY VERIFICATION FUNCTIONAL AREA FOR NINE MILE POINT NUCLEAR STATION, UNITS 1 AND 2 SALP REPORT

In accordance with James Linville's March 6, 1991, memorandum, the attached information is being forwarded to you as input to the Safety Assessment/Quality Verification functional area of the SALP Report for Nine Mile Point Station, Units 1 and 2, for the assessment period which ended March 31, 1991.

ORIGINAL SIGNED BY:

Donald S. Brinkman, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosure:
SALP Report Input

cc w/enclosure:
W. Cook, Nine Mile Point, SRI

CONTACT:
D. Brinkman 49-21402

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

April 1, 1991

Docket Nos. 50-220
and 50-410

MEMORANDUM FOR: Robert A. Capra, Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

FROM: Donald S. Brinkman, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

SUBJECT: FORTHCOMING MEETING WITH NIAGARA MOHAWK POWER CORPORATION

DATE & TIME: Wednesday, April 10, 1991
10:00 a.m. - 2:00 p.m.

LOCATION: One White Flint North
11555 Rockville Pike
Rockville, Maryland
Room 13 B 13

PURPOSE: To discuss current licensing issues for Nine Mile Point.

*PARTICIPANTS:

NRC	Utility
D. Brinkman	D. Greene
D. Oudinot	N. Spagnoletti
	D. Baker

Donald S. Brinkman

Donald S. Brinkman, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

cc: See next page

*Meetings between NRC technical staff and applicants or licensees are open for interested members of the public, petitioners, intervenors, or other parties to attend as observers pursuant to "Open Meeting Statement of NRC Staff Policy," 43 Federal Register 28058, 6/28/78.

memo 4

April 1, 1991

Docket Nos. 50-220
and 50-410

MEMORANDUM FOR: Robert A. Capra, Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

FROM: Donald S. Brinkman, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

SUBJECT: FORTHCOMING MEETING WITH NIAGARA MOHAWK POWER CORPORATION

DATE & TIME: Wednesday, April 10, 1991
10:00 a.m. - 2:00 p.m.

LOCATION: One White Flint North
11555 Rockville Pike
Rockville, Maryland
Room 13 B 13

PURPOSE: To discuss current licensing issues for Nine Mile Point.

*PARTICIPANTS:

NRC	Utility
D. Brinkman	D. Greene
D. Oudinot	N. Spagnoletti
	D. Baker

Donald S. Brinkman, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

cc: See next page

*Meetings between NRC technical staff and applicants or licensees are open for interested members of the public, petitioners, intervenors, or other parties to attend as observers pursuant to "Open Meeting Statement of NRC Staff Policy," 43 Federal Register 28058, 6/28/78.

OFC	:PDI-1:LA	:PDI-1:PE	:PDI-1:PM	:PDI-1:D	:
NAME	:CVogan CV	:DOudinot:avl DO	:DBrinkman DSB	:RACapra DSB for	:
DATE	:4/1/91	:4/1/91	:4/1/91	:4/1/91	:

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Document Name: NMP MEETING NOTICE

Mr. B. Ralph Sylvia
Niagara Mohawk Power Corporation

Nine Mile Point Nuclear Station
Unit Nos. 1 and 2

cc:

Mr. Mark J. Wetterhahn, Esquire
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State Consumer Protection Board
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N. Green, Jr.
L. Plisco
J. Caldwell (Region I Plants) MS 17G21
K. Abraham, Region I
C. Vogan
J. Linville

cc: Licensee/Applicant & Service List

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555
March 28, 1991

Docket No. 50-220

MEMORANDUM FOR: Robert A. Capra, Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

FROM: Donald S. Brinkman, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

SUBJECT: FORTHCOMING MEETING WITH NIAGARA MOHAWK POWER CORPORATION

DATE & TIME: Wednesday April 24, 1991
9:00 a.m. - 2:00 p.m.

LOCATION: One White Flint North
11555 Rockville Pike
Rockville, Maryland
Room 2 F 21

PURPOSE: To discuss thinning of the Nine Mile Point Unit 1 containment torus.

*PARTICIPANTS:

NRC		Utility
W. Russell	G. Bagchi	C. Terry
A. Thadani	D. Brinkman	P. George
C. McCracken	R. Capra	K. Samulson
J. Richardson	H. Kaplan	W. Yeager
W. Lanning	E. Grey, et. al	L. McNeer
		N. Spagnoletti, et. al

Donald S. Brinkman

Donald S. Brinkman, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

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Unit 1 Station Superintendent
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K. Abraham, Region I
C. Vogan
J. Linville
cc: Licensee/Applicant & Service List

Docket No. 50-220

March 28, 1991

MEMORANDUM FOR: Robert A. Capra, Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

FROM: Donald S. Brinkman, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

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J. Richardson	H. Kaplan	W. Yeager
W. Lanning	E. Grey, et. al	L. McNeer
		N. Spagnoletti, et. al

ORIGINAL SIGNED BY:

Donald S. Brinkman, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

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DATE	:3/28/91	:3/28/91	:3/28/91	:3/28/91	:

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PRE-DECISIONAL TRANSMITTAL

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

March 28, 1991

Docket Nos. 50-220
and 50-410

MEMORANDUM FOR: Donald R. Haverkamp, Section Chief
Reactor Projects Section 1B
Projects Branch No. 1
Division of Reactor Projects, Region I

THRU: RAC Robert A. Capra, Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

FROM: Donald S. Brinkman, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

SUBJECTS: INPUT TO SALP FUNCTIONAL AREA FOR NINE MILE POINT NUCLEAR STATION, UNITS 1 AND 2

By copy of this memorandum, the attached information is forwarded to the SALP coordinator as input to the indicated functional area relating to the upcoming Nine Mile Point Nuclear Station SALP report.

In accordance with J. Linville's March 8, 1991, memorandum regarding the Nine Mile Point SALP Report, NRR's writeup of the Safety Assessment/Quality Varification functional area will be submitted to you by 4/20/91.

Donald S. Brinkman

Donald S. Brinkman, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosure:
As stated

cc w/enclosure:
R. Mathew, RI

CONTACT:
D. Brinkman
49-21402

PRE-DECISIONAL TRANSMITTAL

memo 4

ENCLOSURE

A. FUNCTIONAL AREA: Engineering/Technical Support

Niagara Mohawk's preparation for the Unit 1 Power Ascension Test Program was comprehensive and thorough. The test program was conducted in a methodical manner with emphasis being placed upon completing individual tasks carefully, competently and safely rather than being driven by program schedules.

Niagara Mohawk thoroughly reviewed and implemented appropriate lessons learned from Quad Cities before initiating the turbine torsional test at Unit 2. Niagara Mohawk conducted special training for its operators before initiating the test and provided good management control during the test. Good procedural adherence was also evident during the test.

However, in contrast to the above noted examples of good performance, some of Niagara Mohawk's initial responses to generic letters were not fully responsive. The initial response to Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment" did not include sufficient detail or format to enable the staff to identify that specific actions or assessments would be undertaken for each of the five areas recommended in the generic letter on a defined schedule. This inadequate response required the staff to seek additional information in order to conclude that Niagara Mohawk had adequately addressed the recommendations of the generic letter.

PRE-DECISIONAL TRANSMITTAL

Docket Nos. 50-220
and 50-410

March 28, 1991

MEMORANDUM FOR: Donald R. Haverkamp, Section Chief
Reactor Projects Section 1B
Projects Branch No. 1
Division of Reactor Projects, Region I

THRU: Robert A. Capra, Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

FROM: Donald S. Brinkman, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

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ORIGINAL SIGNED BY:

Donald S. Brinkman, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosure:
As stated

cc w/enclosure
R. Mathew, RI

CONTACT:
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PRE-DECISIONAL TRANSMITTAL

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

March 28, 1991

Docket No. 50-220

LICENSEE: Niagara Mohawk Power Corporation

FACILITY: Nine Mile Point Nuclear Station Unit No. 1

SUBJECT: MEETING MINUTES REGARDING THE MARCH 25, 1991, MEETING TO DISCUSS THE LICENSEE'S DESIGN BASIS RECONSTITUTION PROGRAM FOR NINE MILE POINT NUCLEAR STATION UNIT NO. 1.

The meeting was held in the NRC One White Flint North Office in Rockville, Maryland, with Niagara Mohawk Power Corporation (NMPC) and NRC staff representatives. A list of attendees is attached as Enclosure 1. The reference material supplied by the licensee is attached as Enclosure 2. All topics listed in the PRESENTATION OUTLINE were addressed during the meeting.

The purpose of the meeting was to discuss the licensee's Design Basis Reconstitution Program for Nine Mile Point Unit 1. This program is voluntary on the part of the licensee. The staff found the meeting very informative.

Donald S. Brinkman

Donald S. Brinkman, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor regulation

Enclosures:
1. List of Attendees
2. Reference Material

cc w/enclosures:
See next page

97-036

11

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Niagara Mohawk Power Corporation

Nine Mile Point Nuclear Station
Unit No. 1

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ENCLOSURE 1

LIST OF ATTENDEES

PARTICIPANTS	ORGANIZATION
J. Partlow	NRC
J. Calvo	NRC
R. Capra	NRC
D. Brinkman	NRC
W. Lanning	NRC
G. Imbro	NRC
M. McBrearty	NRC
P. Gill	NRC
O. Chopra	NRC
J. Craig	NRC
J. Chan	NRC
C. Cheng	NRC
G. Bagchi	NRC
J. Caldwell	NRC
A. Pal	NRC
D. Oudinot	NRC
K. Eccleston	NRC
N. Wagner	NRC
R. Parkhill	NRC
C. Plows	NUS
C. Terry	NMPC
N. Spagnoletti	NMPC
R. Oleck	NMPC
L. Klosowski	NMPC
G. Wilson	NMPC
M. Montgomery	Bechtel
P. Harris	Bechtel
E. Wenzinger	MPR Assoc.
W. Schmidt	MPR Assoc.

ENCLOSURE 2

NIAGARA MOHAWK

NIAGARA MOHAWK POWER CORPORATION

NINE MILE POINT UNIT 1

DESIGN BASIS RECONSTITUTION/CONFIGURATION MANAGEMENT

PRESENTATION TO NRC

MARCH 25, 1991

NIAGARA MOHAWK

PRESENTATION OUTLINE

I.	INTRODUCTION	C. D. TERRY
II.	DESIGN BASIS ACTIVITIES FOR UNIT 1 RESTART	L. A. KLOSOWSKI
III.	DESIGN BASIS RECONSTITUTION PROGRAM STATUS UPDATE	R. F. OLECK, JR.
IV.	DISCREPANCY RESOLUTION METHODOLOGY	R. F. OLECK, JR.
V.	CLOSING REMARKS	C. D. TERRY

NIAGARA MOHAWK

DESIGN BASIS ACTIVITIES

FOR UNIT 1 RESTART

PRESENTED BY

L. A. KLOSOWSKI

NIAGARA MOHAWK

OVERVIEW

PROVIDE BASIS FOR NMP-1 RESTART CONSIDERING THE LIMITATION IN THE DETAILED DESIGN BASIS AND CONFIGURATION MANAGEMENT AGREEMENT

METHOD

REVIEWED RESULTS FROM RESTART ACTIVITIES AND OTHER COMPLETED ENGINEERING PROGRAMS

FOCUSED REVIEW ON 13 SELECTED SAFETY SYSTEMS

NIAGARA MOHAWK

RESULTS:

- SUFFICIENT BASIS EXISTED TO CONFIRM SAFE OPERATION OF 13 SELECTED SAFETY SYSTEMS AND PERMIT PLANT RESTART
- RESTART AND ENGINEERING PROGRAMS IDENTIFIED SPECIFIC DISCREPANCIES
- SPECIFIC DISCREPANCIES RESOLVED
- ROOT CAUSES FOR DISCREPANCIES DETERMINED AND PROGRAMS EXPANDED WHERE APPROPRIATE
- IDENTIFIED NEED FOR IMPROVED DISCREPANCY RESOLUTION PROCESS

APPROACH FOR REVIEW

4 KEY CRITERIA CONSIDERED TO SUPPORT DESIGN BASIS CONCLUSIONS:

1) COMPLETENESS AND CONSISTENCY OF DESIGN BASIS DOCUMENTATION

2) ACCURACY OF AS-BUILT CONFIGURATION DOCUMENTATION

3) DEMONSTRATION OF SYSTEM FUNCTIONALITY

4) DEMONSTRATION OF STRUCTURAL ADEQUACY

APPROACH FOR REVIEW (continued)

- 40 ACTIVITIES SELECTED AND REVIEWED
  - INCLUDED RESTART ACTIVITIES AND COMPLETED ENGINEERING PROGRAMS
  - DETERMINED EACH PROGRAM OR ACTIVITY CONTRIBUTION TO CONFIRMING:
    1) Design Basis Documentation
    2) As-Built Configuration Documentation
    3) Demonstration of System Functionality
    4) Demonstration of Structural Adequacy

- REVIEW FOCUSED ON 13 SAFETY SYSTEMS
  - ACHIEVE & MAINTAIN SAFE SHUTDOWN
  - REACTOR COOLANT PRESSURE BOUNDARY INTEGRITY
  - CONTROL/MITIGATE OFF-SITE RELEASE

NIAGARA MOHAWK

SUMMARY

- 40 ACTIVITIES REVIEWED CONTRIBUTED SUFFICIENT INFORMATION TO CONCLUDE DESIGN BASIS ADEQUACY FOR THE 13 SELECTED SAFETY SYSTEMS

- SPECIFIC DISCREPANCIES ON EACH PROGRAM RESOLVED

- ROOT CAUSES FOR DISCREPANCIES DETERMINED AND EXPANDED AS APPROPRIATE (e.g., OTHER SYSTEMS)

- EXPANSION TO OTHER SYSTEMS SUFFICIENT TO CONCLUDE DESIGN BASIS ADEQUATE FOR REMAINDER OF PLANT

- CONCLUSIONS REVIEWED BY RESTART ASSESSMENT PANEL, SORC, SRAB

NIAGARA MOHAWK

FOLLOW-ON ACTIVITIES

- REVIEW SUPPORTED THE NEED FOR CONTINUED DESIGN BASIS RECONSTITUTION

- REVIEW IDENTIFIED THE NEED FOR IMPROVED DISCREPANCY RESOLUTION PROCESS

NIAGARA MOHAWK

NIAGARA MOHAWK POWER CORPORATION

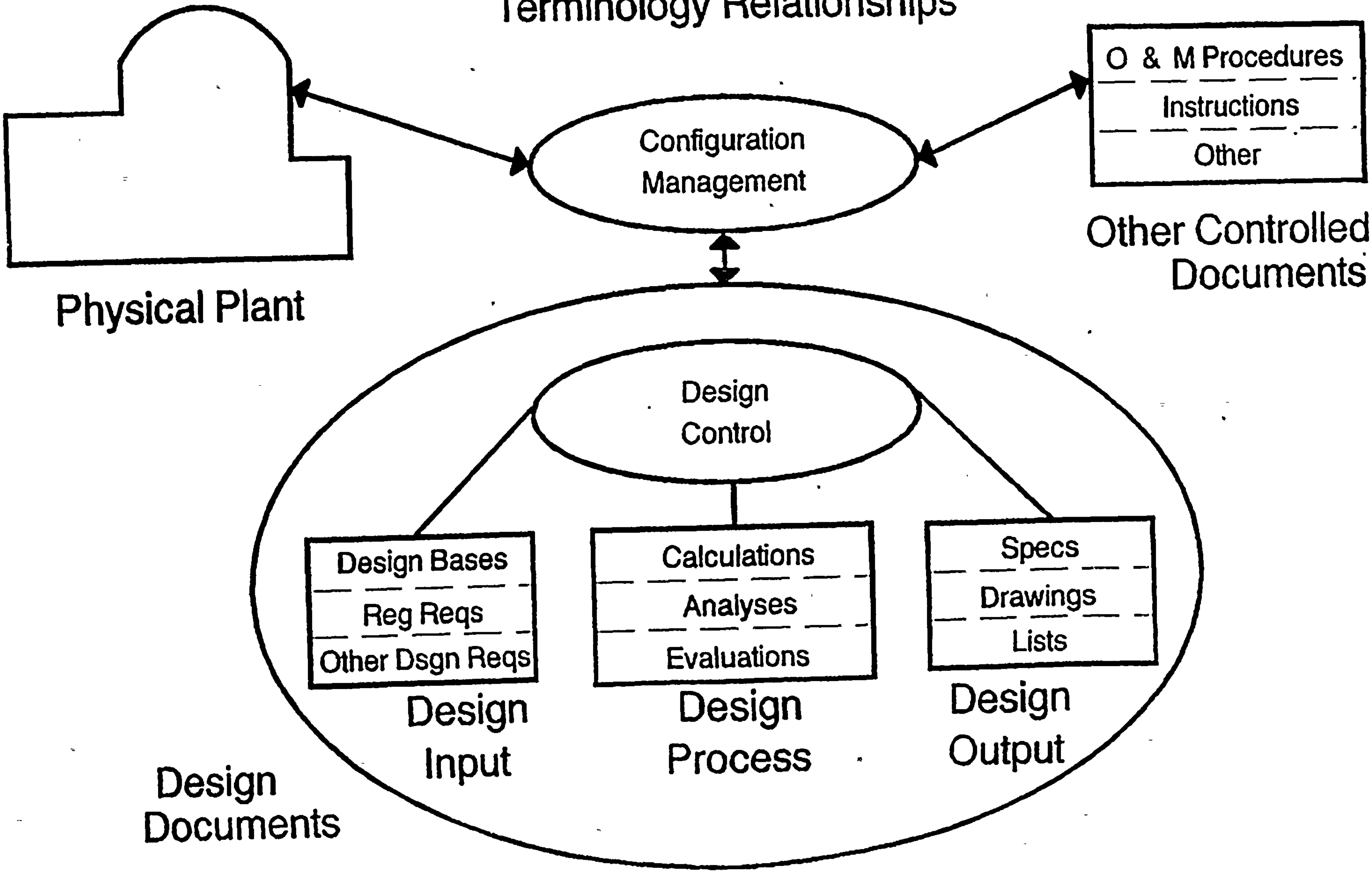
NINE MILE POINT, UNIT 1

DESIGN BASIS RECONSTITUTION PROGRAM STATUS UPDATE

PRESENTED BY

R. F. OLECK, JR.

Figure 1
Terminology Relationships



MOTIVATIONS: FOR DESIGN BASIS RECONSTITUTION/ CONFIGURATION MANAGEMENT UPGRADE

- MORE EFFICIENT
  - SAFETY EVALUATIONS
  - CODE RECONCILIATION
  - MOD. DESIGN/ANALYSIS PROCESS
  - EMERGENCY RESPONSE
  - PLANT OPERATION
- TRAIN ENGINEERS
- PLANT LIFE EXTENSION

NIAGARA MOHAWK

DEVELOPMENT PHILOSOPHY

- MAKE THE NMPC LINE ORGANIZATION PART OF THE TEAM
- EVALUATE RESULTS OF EXISTING PROGRAMS AND UPGRADE AS NECESSARY
- DIVIDE INTO MANAGEABLE SUBTASKS
- MAKE PRODUCTS OF DESIGN BASIS PROGRAMS "USER FRIENDLY"
- PROCEED ONE STEP AT A TIME: TEST PLAN ON PROTOTYPE SYSTEMS
- MODIFY PLAN AS NEEDED AND IMPLEMENT FULL SCOPE
- INTEGRATE INTO EXISTING NMPC ENGINEERING ORGANIZATION AND MAKE PART OF THE WAY NMPC DOES WORK

NIAGARA MOHAWK

PROGRAM INTEGRATION STRATEGY

- IN THE LONG-TERM, INTEGRATE PROGRAMS UNDER FOUR MAJOR GROUPS:

GROUP 1: DESIGN BASIS RECONSTITUTION/CONFIGURATION MANAGEMENT UPGRADE

GROUP 2: EVALUATION, MONITORING, MAINTENANCE OF PLANT CONDITION

GROUP 3: PLANT REEVALUATIONS AND UPGRADES

GROUP 4: ENGINEERING RESOURCES/TRAINING/ METHODOLOGIES

NIAGARA MOHAWK

ENGINEERING PROGRAMS INCLUDED IN DESIGN BASIS RECONSTITUTION

1. Material Condition Study
2. Electrical System Documentation
3. Seismic Upgrade Program
4. Seismic Qualification of Equipment
5. Individual Plant Evaluation
6. As Builts - Electrical
7. As-Builts - Mechanical
8. As-Builts - Structural
9. Vendor Manual
10. Configuration Management
11. FSAR Verification
12. Erosion/Corrosion
13. In-Service Inspection
14. Materials Engineering
15. Equipment Qualification
16. Appendix R - Fire Protection
17. Human Factors
18. Problem Reports
19. Root Cause
20. Prob. Risk Assess (PRA)
21. System Assessment
22. Engineering Excellence
23. In-Service Test
24. Plant Productivity

25. Design Basis Reconstitution
26. Q-List
27. Engineering Procedures
28. Computer Systems
29. Advanced Methodology

GROUP I <u>DESIGN BASIS RECONSTITUTION/CONFIGURATION MANAGEMENT PROGRAMS</u>

- PROGRAM APPROACH

  - COORDINATE ACTIVITIES OF ALL UNIT 1 NMPC DESIGN BASIS RECONSTITUTION AND CONFIGURATION MANAGEMENT ACTIVITIES (I.E., INTEGRATE)

  - DEVELOP "SYSTEM DESIGN BASIS DOCUMENTS" FOR EACH MAJOR SYSTEM OF THE PLANT

  - DEVELOP "TOPICAL DESIGN CRITERIA DOCUMENTS" FOR THE PLANT

  - VERIFY THE AS-BUILT CONFIGURATION OF THE PLANT

  - RECONSTITUTE, EVALUATE AND RECONCILE DESIGN BASIS DOCUMENTS

NIAGARA MOHAWK

SYSTEM DESIGN BASIS DOCUMENTS

TABLE OF CONTENTS

SYSTEM DESIGN BASIS DOCUMENTS

TABLE OF CONTENTS (CONTINUED)

NIAGARA MOHAWK

SYSTEM DESIGN BASIS DOCUMENTS

TABLE OF CONTENTS (CONTINUED)

NIAGARA MOHAWK

DESIGN BASIS RECONSTITUTION
SYSTEM DESIGN BASIS DOCUMENT DEVELOPMENT
SDBD Development Priority

SYSTEM DESIGN BASIS DOCUMENT TITLE	SDBD NUMBER	DBR PROGRAM YEAR(1)
Core Spray System	SDBD-201	1
Service Water System	SDBD-502	1
125V DC Electrical Distribution System	SDBD-806	1
Containment Spray System	SDBD-203	1
Containment Systems	SDBD-202	1
Reactor Building Closed Loop Cooling System	SDBD-503	1
Turbine Building Closed Loop Cooling System	SDBD-504	1
Emergency Diesel Generator System	SDBD-804	1
Automatic Depressurization System	SDBD-301	2
115KV, 4.16KV, 600V & 480V AC Electrical Distribution Systems	SDBD-803	2
24V DC Electrical Distribution System	SDBD-807	3
Control Rod Drive & ATWS Systems	SDBD-303	3
Reactor Protection System	SDBD-302	3
Reactor Vessel Instrumentation System	SDBD-305	3
Motor Generator Sets & 120V AC Elec Distribution System	SDBD-805	3
Emergency Cooling System	SDBD-204	3
Remote Shutdown System	SDBD-304	3
Reactor Recirculation System	SDBD-102	3
Neutron Monitoring System	SDBD-306	4
Service, Instrument & Breathing Air Systems	SDBD-506	4
Shutdown Cooling & Head Spray Systems	SDBD-206	4
Liquid Poison System	SDBD-205	4
Reactor Building HVAC System	SDBD-601	4
Control Room HVAC Systems	SDBD-602	4
Feedwater/HPCI System	SDBD-402	4

DESIGN BASIS RECONSTITUTION
SYSTEM DESIGN BASIS DOCUMENT DEVELOPMENT

SDBD Development Priority

SYSTEM DESIGN BASIS DOCUMENT TITLE	SDBD NUMBER	DBR PROGRAM YEAR(1)
Main & Reheat Steam Systems	SDBD-401	5
Condensate & Condensate Transfer System	SDBD-403	5
Condenser Air-Removal & Off-Gas Systems	SDBD-404	5
Reactor Water Cleanup System	SDBD-103	5
Area Radiation Monitoring System	SDBD-702	5
Process Radiation Monitoring System	SDBD-701	5
Reactor Pressure Vessel & Internals	SDBD-101	6
Spent Fuel Pool Filtering & Cooling System	SDBD-505	6
Safety Parameter Display System	SDBD-307	6
Sampling and Post Accident Sampling System	SDBD-703	6
Circulating Water System	SDBD-501	6
345KV Electrical Distribution System	SDBD-802	6

NOTE: 1. This schedule will be periodically reviewed during the DBR program and additional SDBDs may be written as resources permit.

NIAGARA MOHAWK

DESIGN CRITERIA DOCUMENT (DCD)
DOCUMENT IDENTIFIERS & PLAN FOR DEVELOPMENT
UNDER DBR PROGRAM

DCD IDENTIFIER	DESIGN CRITERIA DOCUMENT TITLE	PROGRAM YEAR
NMP-1 Structural Topics		
DCD-101	Piping Support Design Requirements	Prototype
DCD-102	Classification of Structures & Components for Seismic Design	2
DCD-103	General Architectural Design Requirements	4
DCD-104	Component Structural Design Criteria	2
DCD-105	Anchorage Requirements	2
DCD-106	Missile Loadings	2
DCD-107	Control of Heavy Loads	3
DCD-108	Building Crane Systems	4
DCD-109	Containment Design Requirements	Prototype
DCD-110	Reinforced Concrete Structures	3
DCD-111	Masonry Block Structures	3
DCD-112	Structural Materials	4
DCD-113	Containment Internal Structures	1
DCD-114	Steel Structures	4
DCD-115	Criteria for Seismic Analysis	2
DCD-116	Onsite Seismic Measurement Requirements	4
DCD-117	Fire Protection Criteria	2
DCD-118	Component Support Design Requirements	3
NMP-1 Mechanical Topics		
DCD-201	(Not Assigned)	
DCD-202	Component Functional Design	2
DCD-203	Pipe Break Loadings	1
DCD-204	Hydraulic Design Requirements	2
DCD-205	Heat Transfer Design Requirements	2
DCD-206	Installation Design Requirements	4
DCD-207	System Operating Transients	2
DCD-208	Piping Design Requirements	Prototype
DCD-209	(Not Assigned)	
DCD-210	Equipment Operation Loadings	2
DCD-211	Insulation Material	2
DCD-212	Insulation Materials	3
DCD-213	HVAC General Design Requirements	3
DCD-214	Reactor Vessel Materials	2
DCD-215	Protective Coating Materials	3

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DESIGN CRITERIA DOCUMENT (DCD)
DOCUMENT IDENTIFIERS & PLAN FOR DEVELOPMENT UNDER DBR PROGRAM

DCD IDENTIFIER	DESIGN CRITERIA DOCUMENT TITLE	PROGRAM YEAR
NMP-1 Electrical Topics		
DCD-301	DC Load and Power Distribution	Prototype
DCD-302	Transformer Criteria	4
DCD-303	Electrical Motor Criteria	3
DCD-304	Component Control	2
DCD-305	Nuclear & Process	2
DCD-306	Electrical Cable Design	1
DCD-307	(Not Assigned)	
DCD-308	Environmental Qualification of Instrumentation & Electrical Components	3
DCD-309	Control Panel & Control Display Arrangement Design	2
DCD-310	(Not Assigned)	
DCD-311	AC Load and Power Distribution	1
DCD-312	(Not Assigned)	
DCD-313	Radiation Monitor Setpoints	3
DCD-314	Cathodic Protection Systems Design	4
DCD-315	Switchgear & Interruption Capabilities	2
DCD-316	Heat Tracing System Design	4
DCD-317	Electrical Isolation	1
DCD-318	I&C Setpoint Design Criteria	1
NMP-1 Nuclear & Other Topics		
DCD-401	Accident Loadings	2
DCD-402	Reactor Operational Requirements	2
DCD-403	Fuel Assembly Design	1
DCD-404	Shielding Design/ALARA	2
DCD-405	Vital Area Access/Habitability Analysis	1
DCD-406	Radioactive Design Source Terms	4
DCD-407	External Events	1

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DESIGN BASIS RECONSTITUTION
PROGRAM REVIEWS & PLAN FOR DEVELOPMENT
UNDER DBR PROGRAM

PROGRAM TITLE	PLAN PHASE
Fire Protection and Appendix R Analysis	Prototype
Mark I Containment Program	Prototype
Regulatory Guide 1.97 Review	Prototype
Accident Re-evaluation	1
Core Reload Parameter Verification	1
Design Basis for 125 VDC	1
Electrical System Documentation	1
Erosion/Corrosion Review Program	1
Fire Barrier Adequacy	1
I&C Setpoints	1
Pipe Support Re-evaluation	1
Plant Condition Assessment	1
(PCA)/Plant Life Extension (PLEX)	1
Power Ascension Testing	1
Safety System Functional Inspection (SSFI)	1
Station Blackout Program	1
Support Structural Integrity	1
As-Builts - Electrical	2
As-Builts - Mechanical	2
Channel Functional Testing	2
Environmental Qualification (EQ) of Electrical Components	2
High Pressure Feedwater System	2
IGSCC Review. Program	2
Leak Before Break	2
Materials Engineering Program	2
NUREG-0737: TMI Action Resolution Program	2
Pipe Whip and Jet Thrust Program	2
Probabilistic Testing	2
Root Cause/Trending Program	2
System Descriptions	2

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DESIGN BASIS RECONSTITUTION
PROGRAM REVIEWS & PLAN FOR DEVELOPMENT
UNDER DBR PROGRAM

PROGRAM TITLE	PLAN PHASE
Alarm System Review	3
As Builts - Structural	3
As-Installed Verification	3
Drywell Corrosion Investigation	3
Human Factors Design	3
Integrated Safety Assessment	3
Program (ISAP)/System Evaluation Program (SEP)	3
Recirculation Piping Replacement Program	3
Seismic Qualification Utility Group (SQUG)	3
Seismic Qualification of Equipment	3
Seismic Upgrade Program (SUP)	3
Suppression Chamber Corrosion Evaluation	3
System Assessment	3
Updated FSAR	3
Block Wall Review	4
Configuration Management	4
Control Room Design Review	4
Control of Commercial Grade Items	4
Design Basis Reconstruction	4
Emergency Operating Procedures	4
(EOP) Requalification	4
Individual Plant Evaluation (IPE)	4
Inservice Inspection (ISI)	4
Inservice Testing (IST) - Pump & Valve Validation Program	4
Masonry Wall Cracking	4
Requirements for Walkdown Prior to Core Reload	4
Vendor Technical Manual	4

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SCHEDULE & COST FOR DESIGN BASIS RECONSTITUTION

- OVERALL 5 TO 6 - YEAR EFFORT

- TOTAL EXPECTED FUNDING ABOUT $50 MILLION

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STATUS OF DESIGN BASIS RECONSTITUTION

• DESIGN BASIS RECONSTITUTION PLAN WITH 6 YEAR FUNDING NEEDS DEFINED	COMPLETE
• UTILITY EXPERIENCE SURVEY	COMPLETE
• ENGINEERING PROGRAM REVIEWS	
- VENDOR MANUAL	COMPLETE
- CONFIGURATION MANAGEMENT	IN PROCESS
- MOD PROCESS: OTHERS BEING DEVELOPED	IN PROCESS
• DESIGN BASIS RECONSTITUTION	
- CORE SPRAY SDBD WITH WALKDOWN VERIFICATION	COMPLETE
- 125V DC SDBD	DRAFT IN REVIEW
- TWO DCD'S IN PROCESS (MECH. AND ELEC.)	DRAFT IN REVIEW
• CONFIGURATION MANAGEMENT UPGRADE PLAN	COMPLETE

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SUMMARY

- NMPC UNDERTAKING A SIGNIFICANT INITIATIVE TO RECOVER/RECONSTITUTE THE DESIGN BASIS OF NMP-1

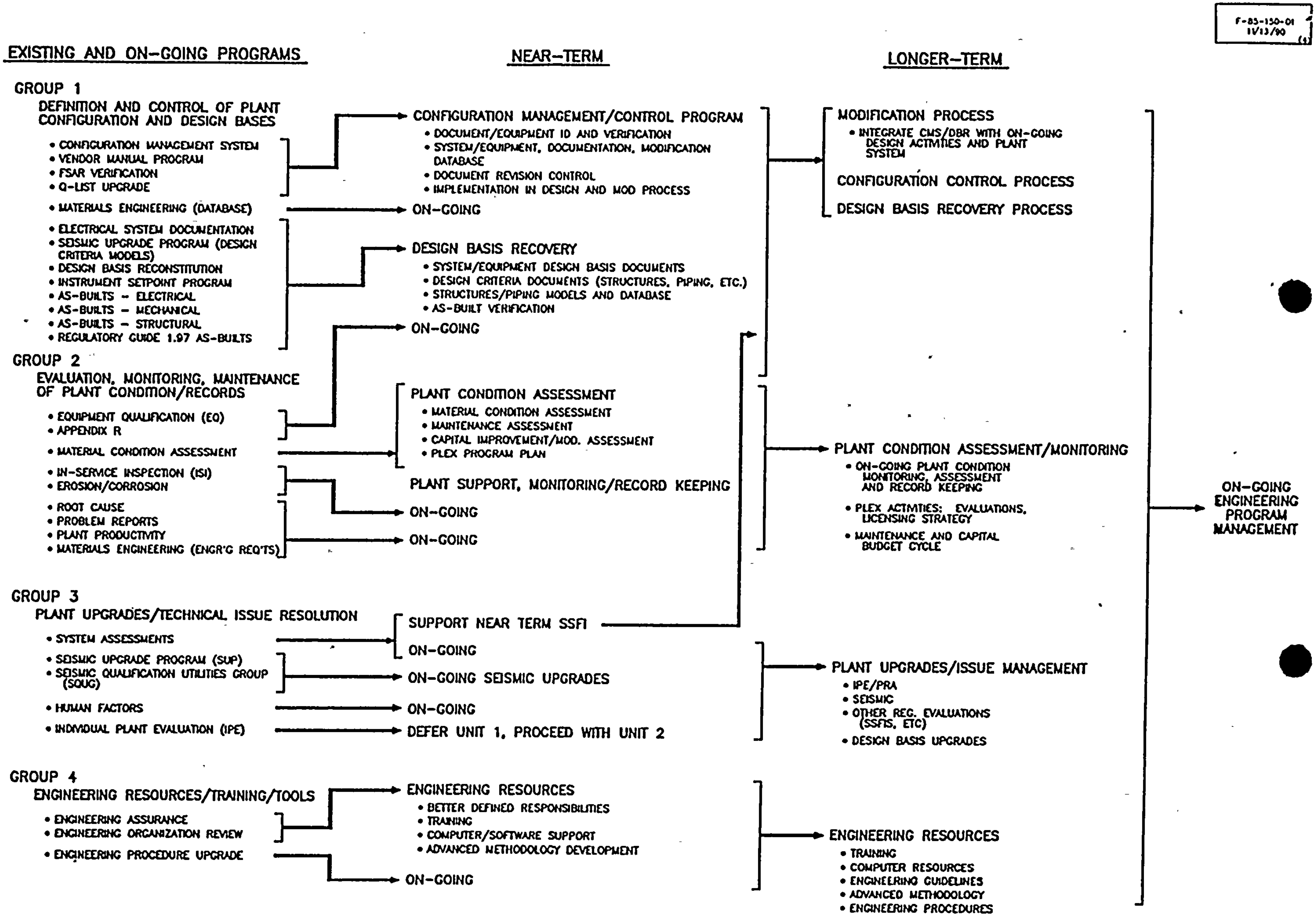
- PROGRAM BEING DEVELOPED AS A DIVISION AND PLANT-WIDE ACTIVITY BUILDING ON RESULTS OF PRIOR PROGRAMS

- SIGNIFICANT RESOURCES HAVE BEEN BUDGETED AND PLANS ARE IN PLACE

- NMPC WILL KEEP NRC APPRISED OF DEVELOPMENTS AND PROGRESS

- SIGNIFICANT BENEFIT OBTAINED FROM INTEGRATION OF EFFORTS FROM EXISTING PROGRAMS

- HELPFUL IN-HOUSE ENGINEERING TOOL/RESOURCE



NIAGARA MOHAWK POWER CORPORATION
DESIGN BASIS RECONSTITUTION

FIGURE 3-1

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DESIGN BASIS RECONSTITUTION

DISCREPANCY RESOLUTION METHODOLOGY

PRESENTED BY

R.F. OLECK, JR.

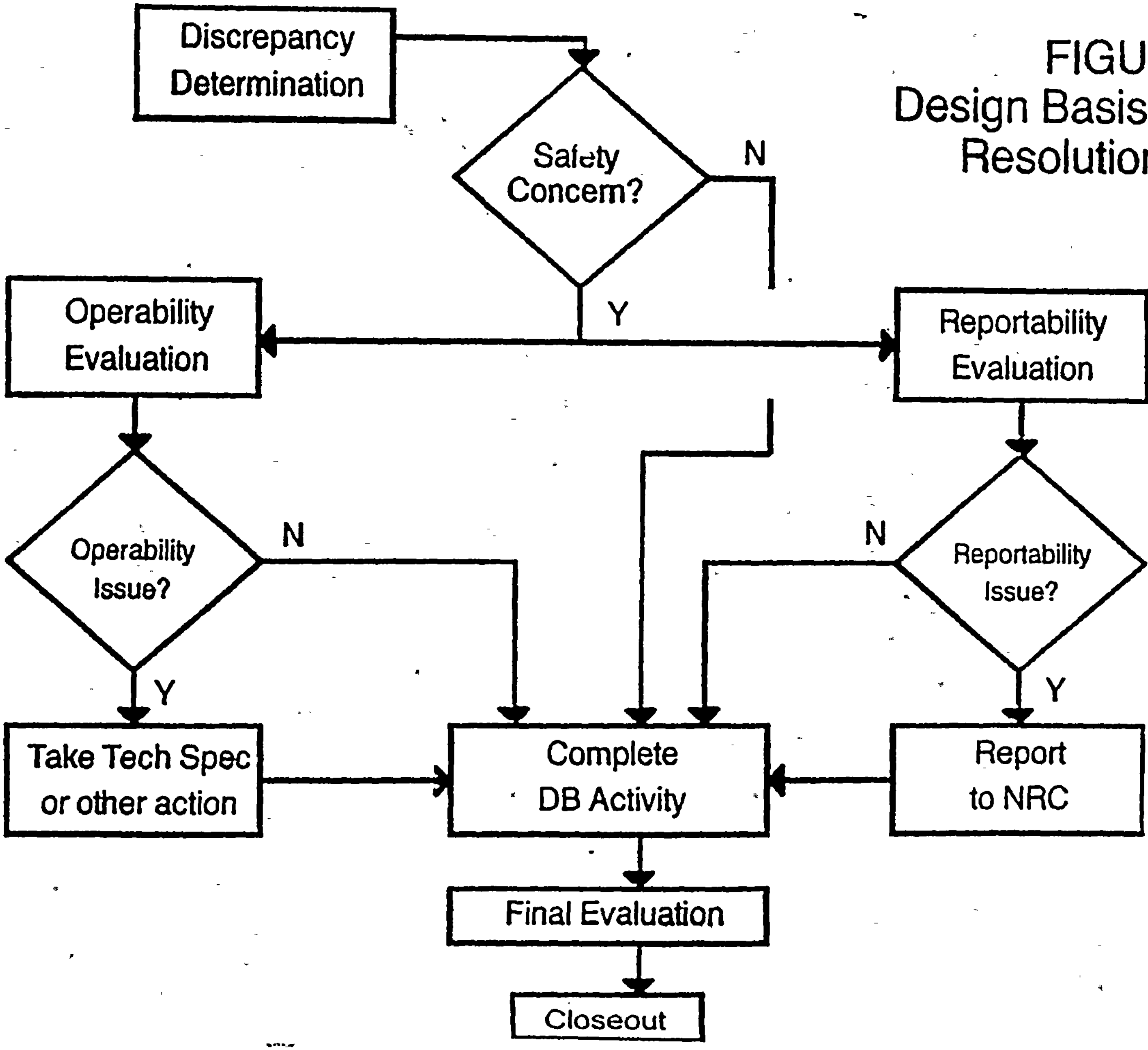

FIGURE 2
Design Basis Discrepancy Resolution Process

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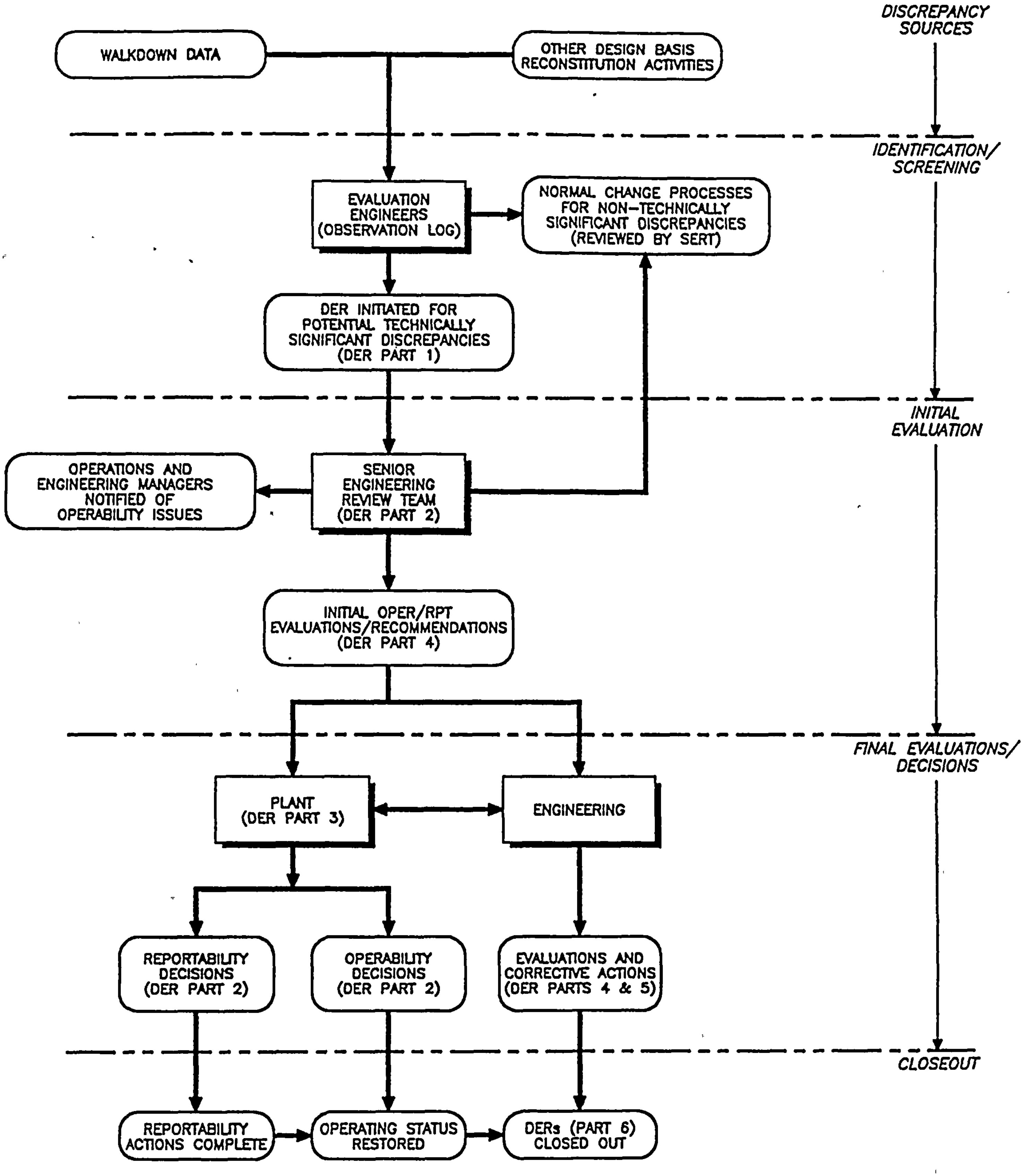
DESIGN BASIS DISCREPANCY RESOLUTION

- FOLLOW INDUSTRY DEVELOPED GUIDELINES AND NMPC DIRECTIVES AND PROCEDURES
- USE A TEAM OF HIGHLY QUALIFIED ENGINEERING AND OPERATIONS PERSONNEL TO ASSESS THE OPERABILITY SIGNIFICANCE OF DISCREPANCIES
- EVALUATE DISCREPANCIES FOR OPERABILITY AND REPORTABILITY IMPACT WITHIN A TIME PERIOD APPROPRIATE FOR THE SIGNIFICANCE
- SCREEN OUT DESIGN BASIS OR CONFIGURATION DISCREPANCIES THAT DO NOT AFFECT COMPONENT/SYSTEM OPERABILITY OR PLANT SAFETY
- COMMUNICATE WITH NRC PERSONNEL ON VARIOUS LEVELS REGARDING PROGRESS OF DISCREPANCY ASSESSMENTS
- RESOLVE/CLOSEOUT DESIGN BASIS PROGRAM RELATED DISCREPANCIES USING APPLICABLE NMPC PROCEDURES

SOURCES OF POTENTIAL DISCREPANCY IDENTIFICATION

- DEVELOPMENT OF DESIGN BASIS DOCUMENTS
- RECONCILIATION OF DATABASES
- IN-PLANT AS-BUILT VERIFICATION WALKDOWNS



OVERALL DISCREPANCY ASSESSMENT METHODOLOGY

EXAMPLE - ELECTRICAL DISCREPANCY ASSESSMENT

WALKDOWN OBSERVATION:

DURING OUTAGE UNFUSED LOAD FOUND ON RPS BUS NOT SHOWN ON ELECT ONE-LINE DWG OR INTER-CONNECTION WIRING DIAGRAM

EVALUATION ENGINEER SCREENING:

- OUTSTANDING NCRs, PRs, & DCRs REVIEWED FOR EXISTING REPORT OF PROBLEM
  - DISCREPANCY NOT PREVIOUSLY ADDRESSED
- OPER & TEST PROCEDURES CHECKED FOR LOAD
  - NEUTRON DETECTION INSTRUMENT SPECIFIED IN APPROVED TEST PROCEDURE
- ITEM CLASSIFIED AS POTENTIAL TECHNICALLY SIGNIFICANT DISCREPANCY
- DISCREPANCY/EVENT REPORT (DER) INITIATED AND FORWARDED TO SENIOR ENGINEERING REVIEW TEAM

SENIOR ENGINEERING REVIEW TEAM EVALUATION:

- DISCREPANCY CONFIRMED AS TECHNICALLY SIGNIFICANT
  - LOAD COULD CAUSE LOSS OF NON REDUNDANT INSTRUMENTS POWERED BY THE CIRCUIT DUE TO POSTULATED FAULT
- OPERABILITY EVALUATION INITIATED
  - ENGINEERING AND OPERATIONS MANAGERS NOTIFIED

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EXAMPLE - ELECTRICAL DISCREPANCY ASSESSMENT (CONTINUED)

- LOAD VERIFIED PROPERLY WIRED BY ELECT. MAINT.
- CONSIDERED NON-GENERIC PROBLEM SINCE OTHER CIRCUITS WALKED DOWN DID NOT EXHIBIT THIS PROBLEM
- SYSTEM JUDGED OPERABLE, BUT FURTHER EVALUATION REQUIRED FOR FINAL DECISION
    - LICENSING COMMITMENT MAY BE COMPROMISED
    - ENGINEERING AND OPERATIONS MANAGERS NOTIFIED OF EVALUATION RESULTS
- DER DISPOSITIONED AND FORWARDED FOR PLANT AND ENGINEERING FOR FURTHER ACTIONS

ENGINEERING EVALUATIONS AND CORRECTIVE ACTIONS:

- DOCUMENT CHANGES INITIATED AGAINST ELECT ONE-LINE DWG, INTER-CONNECTION DIAGRAM, MAINT. PROCEDURE(S), AND RPS LOAD LIST TO REFLECT AS-BUILT CONFIG
- DER EVALUATIONS RESULT IN FAST TRACK MOD DEVELOPED FOR FUSE INSTALLATION DURING NEXT OUTAGE
- SYSTEM DETERMINED TO BE OPERABLE FOR INTERIM
- DISCREPANCY DETERMINED TO BE NON-REPORTABLE

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EXAMPLE - MECHANICAL DISCREPANCY ASSESSMENT

WALKDOWN OBSERVATION:

ISOLATION VALVE FOUND ON INSTRUMENT SENSING LINE, BUT IS NOT SHOWN ON P&ID

EVALUATION ENGINEER SCREENING:

- OUTSTANDING NCRs, PRs, & DCRs REVIEWED FOR EXISTING REPORT OF PROBLEM
  - DISCREPANCY NOT PREVIOUSLY ADDRESSED
- OPER. AND MAINT. PROCEDURES CHECKED
  - VALVE NOT COVERED BY PROCEDURES
- ITEM CLASSIFIED AS POTENTIAL TECHNICALLY SIGNIFICANT DISCREPANCY
- DISCREPANCY/EVENT REPORT (DER) INITIATED AND FORWARDED TO SENIOR ENGINEERING REVIEW TEAM

SENIOR ENGINEERING REVIEW TEAM EVALUATION:

- DISCREPANCY CONFIRMED AS TECHNICALLY SIGNIFICANT
- STRUCTURAL ADEQUACY CHECKED AND FOUND TO BE ACCEPTABLE
- OPERABILITY EVALUATION INITIATED SINCE VALVE COULD INADVERTENTLY BE LEFT IN A CLOSED POSITION
  - ENGINEERING AND OPERATIONS MANAGERS NOTIFIED

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EXAMPLE - MECHANICAL DISCREPANCY ASSESSMENT (CONTINUED)

- VALVE POSITION VERIFIED BY OPERATIONS TO BE OPEN
- OTHER INSTRUMENT LINES CHECKED FOR SAME PROBLEM AND DISCREPANCY DETERMINED NOT TO BE A GENERIC CONCERN
- AFFECTED SYSTEM DETERMINED TO BE OPERABLE FROM AN ENGINEERING STANDPOINT
- ENGINEERING AND OPERATIONS MANAGERS NOTIFIED OF EVALUATION RESULTS

• DER DISPOSITIONED AND FORWARDED FOR PLANT AND ENGINEERING FOR FURTHER ACTIONS

ENGINEERING EVALUATIONS AND CORRECTIVE ACTIONS:

• DOCUMENT CHANGES INITIATED AGAINST P&ID DWG, MAINT. AND OPERATION PROCEDURES

• DISCREPANCY DETERMINED TO BE NON-REPORTABLE

CONCLUSIONS

- THE DBR DISCREPANCY EVALUATION METHODOLOGY IS CONSISTENT WITH:
  - NUMARC DESIGN BASIS PROGRAM GUIDELINES
  - NRC REQUIREMENTS AND EVALUATION GUIDANCE
  - NMPC DIRECTIVES AND PROCEDURES
- THE DBR APPROACH IS FORMALIZED AND WELL DOCUMENTED
- USING THE DBR APPROACH, APPROPRIATELY QUALIFIED PERSONNEL WILL PROMPTLY AND CONSISTENTLY IDENTIFY, EVALUATE AND CORRECT SIGNIFICANT DISCREPANCIES

March 28, 1991

Docket No. 50-220

LICENSEE: Niagara Mohawk Power Corporation

FACILITY: Nine Mile Point Nuclear Station Unit No. 1

SUBJECT: MEETING MINUTES REGARDING THE MARCH 25, 1991, MEETING TO DISCUSS THE LICENSEE'S DESIGN BASIS RECONSTITUTION PROGRAM FOR NINE MILE POINT NUCLEAR STATION UNIT NO. 1.

The meeting was held in the NRC One White Flint North Office in Rockville, Maryland, with Niagara Mohawk Power Corporation (NMPC) and NRC staff representatives. A list of attendees is attached as Enclosure 1. The reference material supplied by the licensee is attached as Enclosure 2. All topics listed in the PRESENTATION OUTLINE were addressed during the meeting.

The purpose of the meeting was to discuss the licensee's Design Basis Reconstitution Program for Nine Mile Point Unit 1. This program is voluntary on the part of the licensee. The staff found the meeting very informative.

ORIGINAL SIGNED BY:

Donald S. Brinkman, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. List of Attendees
2. Reference Material

cc w/enclosures:
See next page

Distribution

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CVogan	JLinville
DOudinot	

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Docket No. 50-220

March 22, 1991

LICENSEE: Niagara Mohawk Power Corporation

FACILITY: Nine Mile Point Nuclear Station Unit No. 1

SUBJECT: MEETING MINUTES REGARDING MARCH 5, 1991, MEETING TO DISCUSS UPDATES TO REACTOR COOLANT AND PRIMARY CONTAINMENT ISOLATION VALVES TABLES - NINE MILE POINT NUCLEAR STATION UNIT NO. 1

A meeting was held in the NRC One White Flint North Office in Rockville, Maryland, with Niagara Mohawk Power Corporation (NMPC) and NRC staff representatives to discuss a proposed license amendment for the reactor coolant and primary containment isolation valves tables for Nine Mile Point Unit 1. The NRC staff requested this meeting. Enclosure 1 is a list of the meeting attendees.

The meeting attendees discussed the proposed Technical Specification changes item by item. The staff representatives obtained clarification on various points. In summary, the licensee stated the proposed amendment was submitted to update the reactor coolant system and primary containment isolation valve tables and to conform to the requirements of 10 CFR Part 50, Appendix J, and the NRC's SE dated May 6, 1988. Several administrative changes were also discussed.

ORIGINAL SIGNED BY:

Donald S. Brinkman, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosure:
As stated

cc w/enclosure:
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ACRS (10)
RACapra
PDI-1 Reading
WCook, SRI at NMP

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NAME	:CVogan CV	:DOudinot:cs	:DBrinkman DB	:RACapra RC	:
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MEMO 4

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

March 22, 1991

Docket No. 50-220

LICENSEE: Niagara Mohawk Power Corporation

FACILITY: Nine Mile Point Nuclear Station Unit No. 1

SUBJECT: MEETING MINUTES REGARDING MARCH 5, 1991, MEETING TO DISCUSS UPDATES TO REACTOR COOLANT AND PRIMARY CONTAINMENT ISOLATION VALVES TABLES - NINE MILE POINT NUCLEAR STATION UNIT NO. 1

A meeting was held in the NRC One White Flint North Office in Rockville, Maryland, with Niagara Mohawk Power Corporation (NMPC) and NRC staff representatives to discuss a proposed license amendment for the reactor coolant and primary containment isolation valves tables for Nine Mile Point Unit 1. The NRC staff requested this meeting. Enclosure 1 is a list of the meeting attendees.

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Donald S. Brinkman

Donald S. Brinkman, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosure:
As stated

cc w/enclosure:
See next page

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475 Allendale Road
King of Prussia, Pennsylvania 19406

Ms. Donna Ross
New York State Energy Office
2 Empire State Plaza
16th Floor
Albany, New York 12223

ENCLOSURE 1

Attendance List
March 5, 1991 Meeting to Discuss
Updates to Reactor Coolant
and Primary Containment
Isolation Valves Tables

Name	Position	Organization
D. Oudinot	Licensing Engineer	NRR/PDI-1
C. R. Nichols	Reactor Systems Engineer	NRC/SPLB
J. C. Pulsipher	Reactor Systems Engineer	NRC/SPLB
P. Francisco	Manager Nuclear Licensing	NMPC
W. Drews	Manager Tech Support	NMPC
J. Beres	Licensing Engineer	NMPC