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MANGAN,C.V. Niagara Mohawk Power Corp.
RECIP.NAME RECIPIENT AFFILIATION
BUTLER,W. Licensing Branch 2

SUBJECT: Forwards info re safety relief valve pool loads,bulk to local pool temp differences & Mark III containment,closing SER Confirmatory Items 6,13F & 13I.FSAR Pages 6A.3-17-17a & figures withheld (ref 10CFR2.790).

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NIAGARA MOHAWK

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

September 30, 1985
(NMP2L 0503)

Mr. Walter Butler, Chief
Licensing Branch No. 2
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Butler:

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

Enclosed is information discussing Safety Relief Valve Pool Loads and submerged structure loads, bulk to local pool temperature differences and Mark III containment concerns for Nine Mile Point Unit 2. This information is provided to close Safety Evaluation Report confirmatory items 6, 13F, and 13I. Amendment 22 of the Final Safety Evaluation Report will incorporate these descriptions.

In accordance with the provisions of Section 2.790(a)(4) of the Commission's Regulations [10CFR2.790(a)(4)], it is hereby requested pages 6A.3-17-17a and drawings 6A.3-43-44 be treated by the Commission as confidential and proprietary and be withheld from public disclosure. As required by Section 2.790(b)(1), an affidavit from Pennsylvania Power and Light Company, requesting withholding, was submitted with our January 26, 1983 letter to Mr. Harold Denton which supports our request. A copy of said affidavit is enclosed.

Very truly yours,

CVMangan

C. V. Mangan
Senior Vice President

CVM/NLR/dd
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0965G

xc: R. A. Gramm, NRC Resident Inspector
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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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

AFFIDAVIT

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

C. V. Mangan

Subscribed and sworn to before me, a Notary Public in and for the State of New York and County of Onondaga, this 30 day of September, 1985.

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

My Commission expires:
JANIS M. MACRO
Notary Public in the State of New York
Qualified in Onondaga County No. 4784555
My Commission Expires March 30, 1987

COMMONWEALTH OF PENNSYLVANIA)
: SS
COUNTY OF LEHIGH)

I, NORMAN W. CURTIS, being duly sworn according to law depose and say that I am Vice President, Engineering & Construction - Nuclear of Pennsylvania Power & Light Company (PP&L), and that:

1. For the reasons listed below, portions of Appendix 6A of the Niagara Mohawk Power Corporation Nine Mile Point Unit Two Final Safety Analysis Report (Appendix 6A of the Nine Mile Point Unit Two FSAR), which is being filed with the U.S. Nuclear Regulatory Commission (NRC) by Niagara Mohawk Power Corporation (NMPC) in connection with its License Application at Docket No. 50-410, contains information considered by PP&L to be confidential information containing trade secrets and should be withheld from public disclosure.

2. In support of its averment that the above-mentioned information is confidential, PP&L provides the following reasons:

a. Kraftwerk Union Aktiengesellschaft (KWU) developed a quencher technology as a means of pressure suppression in a wetwell containment system. This technology was developed by KWU during a research program which cost considerable amounts of time and money. The information developed during this research program is not available to the public. PP&L entered into a contract with KWU for, <u>inter alia</u>, the purchase of an information package that was developed by KWU during its research program. PP&L sent copies of the information package to the NRC. This information was accompanied by an affidavit signed by KWU requesting the NRC to withhold it from public disclosure.

b. Part of the contractual arrangement between KWU and PP&L involved KWU performing a full scale test simulating the parameters that exist at PP&L's Susquehanna Steam Electric Station (Susquehanna). The purpose of this test

was to verify the design analyses used to engineer the quencher device. PP&L submitted to the NRC its Susquehanna DAR (See Docket Nos. 50-387 and 50-388) which explained the response of the containment system to safety relief valve (SRV) discharge and loss of coolant accident (LOCA) loads, and contained certain information considered by PP&L as proprietary. These proprietary portions of the Susquehanna DAR concerned the full scale test portion of the PP&L and KWU efforts and the design technique for assessing the containment system margins. The document is of considerable importance because it enables one to define loads associated with SRV discharge. The document, therefore, is a valuable tool in applying the KWU information.

c. Although the Susquehanna DAR and the KWU information was prepared specifically for application to Susquehanna, it is of value to anyone desiring to apply it to any plant employing a similar pressure suppression system. Because of the value of this information and its market potential, PP&L considers it to be the type that is customarily held in confidence. As a result, PP&L requested and the NRC agreed to withhold this information from public disclosure pursuant to 10 CFR §2.790.

d. Appendix 6A of the Nine Mile Point Unit Two FSAR is based on the Susquehanna DAR and the KWU information. It was developed as a result of significant effort and expense by PP&L. PP&L has spent considerable time and money in obtaining a contract with KWU, and developing the test plan and test matrix. PP&L thus far has paid KWU approximately $4.5 million in connection with the KWU contract.

e. Pursuant to an agreement dated August 10, 1978, PP&L sold to NMPC the right to use the KWU information described herein for

NMPC's Nine Mile Point Unit Two Nuclear Generating Station. Pursuant to the agreement, NMPC is required to maintain the proprietary and confidential nature of the information contained in Appendix 6A of the Nine Mile Point Unit Two FSAR.

f. The information contained in Appendix 6A of the Nine Mile Point Unit Two FSAR has been and will continue to be held in strict confidence. External distribution of this information is restricted. Each of the recipients of this information has promised to honor the proprietary value of the information and is precluded from using this information for any purpose other than certain specific enumerated applications. These recipients limit the access of the information to those persons in their employ who have a need to use the information in the performance of their duties.

g. None of the information provided therein will be found in any public sources.

h. Because of the expense it has incurred in the development of this information, the marketability of this information, which will enable it to recoup a portion of these expenses, and the economic harm that PP&L would suffer from its inability to sell the information, the need for protecting PP&L's interest outweighs the interest involved in public disclosure.

N W Curtis

Norman W. Curtis
Vice President,
Engineering & Construction - Nuclear
Pennsylvania Power & Light Company

Sworn and subscribed before me this 26th day of January, 1983

Jean A. Smolick

JEAN A. SMOLICK, Notary Public
Allentown, Lehigh County, Pa.
My Commission Expires May 14, 1984

SER CONFIRMATORY ITEM 6:

Reevaluation of the capability of the containment internal structures to the newly identified SRV pool loads.

INTRODUCTION:

The evaluations of capability of reactor internal structures for SRV and LOCA loadings along with combinations of latest system loads, have been completed. The effects of SRV loads, which were compared with the original design loadings without SRV, are presented in the data summary.

SRV & LOCA ANALYSIS:

Using loading data supplied by vendors (KWU and GE for SRV and LOCA loads respectively) as referenced in FSAR Appendix 6A.3.6, Reference 17 and Appendix 6A.4.10, References 9 and 18 based on NUREG 0802 and 0808, dynamic analyses of the reactor building including primary containment and reactor internal structures were performed to obtain response spectra and forces and moments for the design.

COMPARISON OF RESULTS:

Tables 1-5 summarize the comparison of results for the various internal structures. Each structure has been evaluated for all appropriate original design loading combinations without SRV and these were compared with the results of latest SRV and LOCA loads. For steel floor framing, an estimate of the average effect of SRV and LOCA loadings is provided. This simplification is presented because the stresses resulting from these loadings are all different for particular beams.

CONCLUSION:

All reactor internal structures were evaluated and found to be acceptable for all appropriate combinations of latest loads, including SRV and LOCA loads. This occurs because the original designs typically had design margins provided to permit acceptance of loads not available at the time of original design and final system loads. The structures outside primary containment were affected insignificantly by SRV pool loads.

REACTOR INTERNAL STRUCTURES

TABLE - 1

(1) DRYWELL FLOOR - CONCRETE STRUCTURE

	(a) For Radial Rebar	Original Loading without SRV	SRV/LOCA Loading	% Increase
Radius 45'6"	Moment K.Ft/Ft.	587	32.7	5.6
	Normal Force K/Ft.	259.7	12.1	4.7
Radius 15'3"	Moment K.Ft/Ft.	876	56.7	6.5
	Normal Force K/Ft.	267	17.3	6.5
	(b) For Hoop Rebar			
Radius 45'6"	Moment K.Ft/Ft.	249.3	6.4	2.6
	Normal Force K/Ft.	190.2	13.5	7.1
Radius 15'3"	Moment K.Ft/Ft.	294.2	11.3	3.8
	Normal Force K/Ft.	209.9	17.3	8.2

TABLE - 2

(2) REACTOR VESSEL PEDESTAL - CONCRETE STRUCTURE

	(a) For Vertical Rebar	Original Loading Without SRV	SRV/LOCA Loading	% Increase
El. 175'0"	Moment KFt./Ft.	945.2	111.0	11.7
	Normal Force K/Ft.	110.03	23.0	20.9
El. 232'0"	Moment KFt./Ft.	1287.6	13.0	10.1
	Normal Force K/Ft.	58.4	21.0	36.0
	(b) For Hoop Rebar			
El. 175'0"	Moment KFt./Ft.	210.2	17.0	8.1
	Normal Force K/Ft.	973.0	9.0	0.9
El. 232'0"	Moment KFt./Ft.	448.1	3.0	0.7
	Normal Force K/Ft.	96.4	6.0	6.2

TABLE - 3

(3) BIOLOGICAL SHIELD WALL - STEEL STRUCTURE

	Stresses in KSI	Original Loading Without SRV	SRV/LOCA Loading	% Increase
El. 272'0"	Hoop Stress	22.7	0.45	2.0
	Vertical Stress	14.8	0.90	6.1
El. 310'0"	Hoop Stress	26.0	0.79	3.0
	Vertical Stress	5.4	0.89	16.5

NOTE: THE SHEAR STRESS VALUES ARE INSIGNIFICANT.

TABLE - 4

(4) Startruss

COMPARISON OF APPLIED LOADS (KIPS) IN STARTRUSS DESIGN
(See NOTE below)

LOADING	DESIGN LOADS	LATEST LOADS
SSE	1752.6	1440.0
OBE	1218.3	925.0
MSS (Long.)	679.0	ELIMINATED
MSS (Cir.)	475.0	111.0
SRV	155.4	109.0
AP	389.0	1020.0
LOCA (Chug/CO)	64.6	28.9
FAULTED	2651.6	2571.0
NORMAL	1218.3	925.0

NOTE: Conceptual loads were used in the design of Startruss. An adequate built-in margin was left in to allow for any changes in loads. A comparison of the latest loads including hydrodynamic loads and design loads indicate that the new loads are enveloped by the original design loads.

TABLE - 5

(5) FLOORS - PRIMARY CONTAINMENT STEEL FRAMING:

	SRV & LOCA LOADS	AVG. INCREASE IN STRESS DUE TO SRV/LOCA LOADING ON STEEL MEMBERS
El. 222'6"	Horizontal Vertical	30% 25%
El. 247/249"	Horizontal Vertical	20% 20%
El. 261'0"	Horizontal Vertical	15% 15%
El. 278'6"	Horizontal Vertical	15% 15%
El. 288'0"	Horizontal Vertical	15% 15%
El. 305'9"	Horizontal Vertical	10% 10%

NOTE: Hypothetical loads were used in the design of steel framing. An adequate built-in margin was left in to allow for any change in loads. A minor modification was required for certain beams as a result of final system loads and not by addition of SRV loadings.

2. SRV discharge following isolation/scram:

3. SRV discharge following small break accident (SBA):

6A.10.2.3 Analysis Method

The analysis uses the computer code CONSBA[3], which is designed to analyze the reactor and primary containment system transient during normal and abnormal shutdown of the reactor system. The program uses a finite difference technique using input-specified time steps to solve the transient equations. In each time step, the program determines the mass and energy flow across all control volumes and performs thermodynamic state calculations for the reactor vessel, drywell, suppression chamber, and suppression pool, assuming thermodynamic equilibrium conditions.

The reactor coolant is represented by volumes of steam and liquid in thermal equilibrium. The total volume of the coolant (steam and liquid) in the reactor system is assumed constant. The reactor water level is maintained by feedwater, ECCS systems, and CRD flow throughout the transient. Heat is added to the reactor coolant from thermal mixing with ECCS/feedwater/CRD flow, decay heat, fuel sensible heat, fission energy, and the reactor vessel and internals metal mass. At the beginning of the transient, the reactor vessel, internals, and coolant are assumed to be in thermal equilibrium. With the depressurization of the reactor vessel, the coolant temperature decreases, establishing a heat flow from the reactor vessel and internals to the coolant.

Steam can flow from the reactor coolant steam volume to the drywell, main condenser, or through SRVs into the suppression pool. For the small break accident (SBA), steam is added directly to the drywell from the break and to the
22 suppression pool through the SRVs. SRV and break flows are calculated using the frictionless Moody flow model.

1. For steam mass flux (through the quencher) greater than 94 lbm/sq ft-sec, the suppression pool local temperature shall not exceed 200°F.

2. For steam mass flux less than 42 lbm/sq ft-sec, the suppression pool temperature shall be at least 20°F subcooled with respect to the local saturation temperature at the quencher elevation.

3. For steam mass flux greater than 42 lbm/sq ft-sec but less than 94 lbm/sq ft-sec, the suppression pool temperature can be established by linearly interpolating the local temperatures established under Items 1 and 2.

For Unit 2, the minimum quencher submergence is 19.9 ft. Assuming the minimum suppression chamber pressure of 14.2 psia, the local boiling temperature would be 234°F. Considering 20°F of subcooling, the maximum local suppression pool temperature limit would be 214°F.

6A.10.2.7 Results/Conclusions

Table 6A.10-1 summarizes the results of the suppression pool temperature transients. Figures 6A.10-2 through 6A.10-9 show the bulk pool temperature and reactor pressure transients for NUREG-0783 transients. It is observed that the peak pool temperature remains below the 212°F design limit in all cases.

Figures 6A.10-10 through 6A.10-13 compare bulk pool temperature as a function of quencher mass flux (reactor vessel pressure) to the local pool temperature limit. The minimum temperature difference between allowable local temperature and the pool bulk temperature ranges from 66°F to 13°F.

The worst transient is the isolation/scram (Case 2). The peak bulk temperature of 208.6°F is within the design limit of 212°F.

A maximum local to bulk temperature difference on the order of 10°F has been demonstrated by in-plant tests at Caorso[6] and LaSalle[7]. On this basis, it is expected that the Unit 2 local to bulk difference will be less than the minimum available temperature difference, predicted by this analysis to be 13.4°F, for the isolation/scram event (Case 2). This conclusion is supported by the pool thermal mixing analysis described in the following section. 22

6A.10.3 Suppression Pool Thermal Mixing

6A.10.3.1 Discussion

To preclude the occurrence of condensation instability dynamic loads during safety/relief valve (SRV) discharge events, limitations have been placed on the local temperature of suppression pool water feeding the condensation zone. These limits were discussed in the previous section and are shown on Figures 6A.10-10 through 6A.10-13. However, the bulk or mass average pool temperature is used in plant transient analyses to characterize pool heat-up. Accordingly, the difference between the bulk and local valves must be specified so that the analysis can demonstrate operation within the prescribed limits.

This analysis demonstrates that the local to bulk temperature difference for Unit 2 will be less than 10°F considering a single SRV discharging and no residual heat removal (RHR) system flow.

22

6A.10.3.2 Methodology

The three-dimensional hydrothermal analysis code TEMPEST[8] was used to predict the local temperature transient following a single SRV discharge for Unit 2. The computer code and analytical method were initially benchmarked against the LaSalle inplant test data[7]. Code parameters such as the condensing zone entrainment factor were adjusted parametrically until the code prediction of near field, far field, and surface temperatures agreed with the test data. The model, benchmarked in this fashion, was then applied to the Unit 2 geometry.

6A.10.3.3 Nodalization

The LaSalle extended blowdown test identified in Reference 7 as "SRV-E" was utilized for the comparison. The LaSalle pool geometry is shown schematically in Figure 6A.10-14. The TEMPEST model developed for this pool consists of 390 fluid cells or nodes derived by dividing the annular pool geometry into 13 sectors with each sector containing 30 nodes arranged 6 vertically by 5 radially. The nodal size was established such that the 10-ft tee-quencher is entirely within one cell.

Figure 6A.10-15 shows the unit 2 pool geometry. By comparison with Figure 6A.10-14, it is observed that the Unit 2 and LaSalle geometries are very similar. Therefore,

the same 390 node model was applied in the Unit 2 analysis with minor changes to reflect the Unit 2 dimensions.

6A.10.3.4 Results

Temperature response data obtained at four sensor locations in the LaSalle SRV-E test are compared with the analytical predictions of the TEMPEST model for the cells representing the sensor locations. These comparisons are presented in Figures 6A.10-16 through 6A.10-19. The sensor locations are shown on Figure 6A.10-14. 22

Figure 6A.10-20 shows the local-to-bulk temperature difference predicted for Unit 2 as a result of a single SRV blowdown. In this case the local temperature is the predicted temperature for the fluid cell immediately above the quencher while the bulk temperature is the mass average temperature. These results indicate that the local-to-bulk temperature difference applicable to Unit 2 is approximately 10°F.

6A.10.4 REFERENCES

1. Suppression Pool Temperature Monitoring Recommendations, GE Specification 22A4379, February 15, 1977.

2. Assumptions for Use in Analyzing Mark II BWR Suppression Pool Temperature Response to Plant Transients Involving Safety/Relief Valve Discharge, Rev. 1, December 1980 (White Paper, Rev. 1).

3. Stone & Webster Engineering Corporation, Containment Small Break Accident Code (CONSBA), Nu-169.

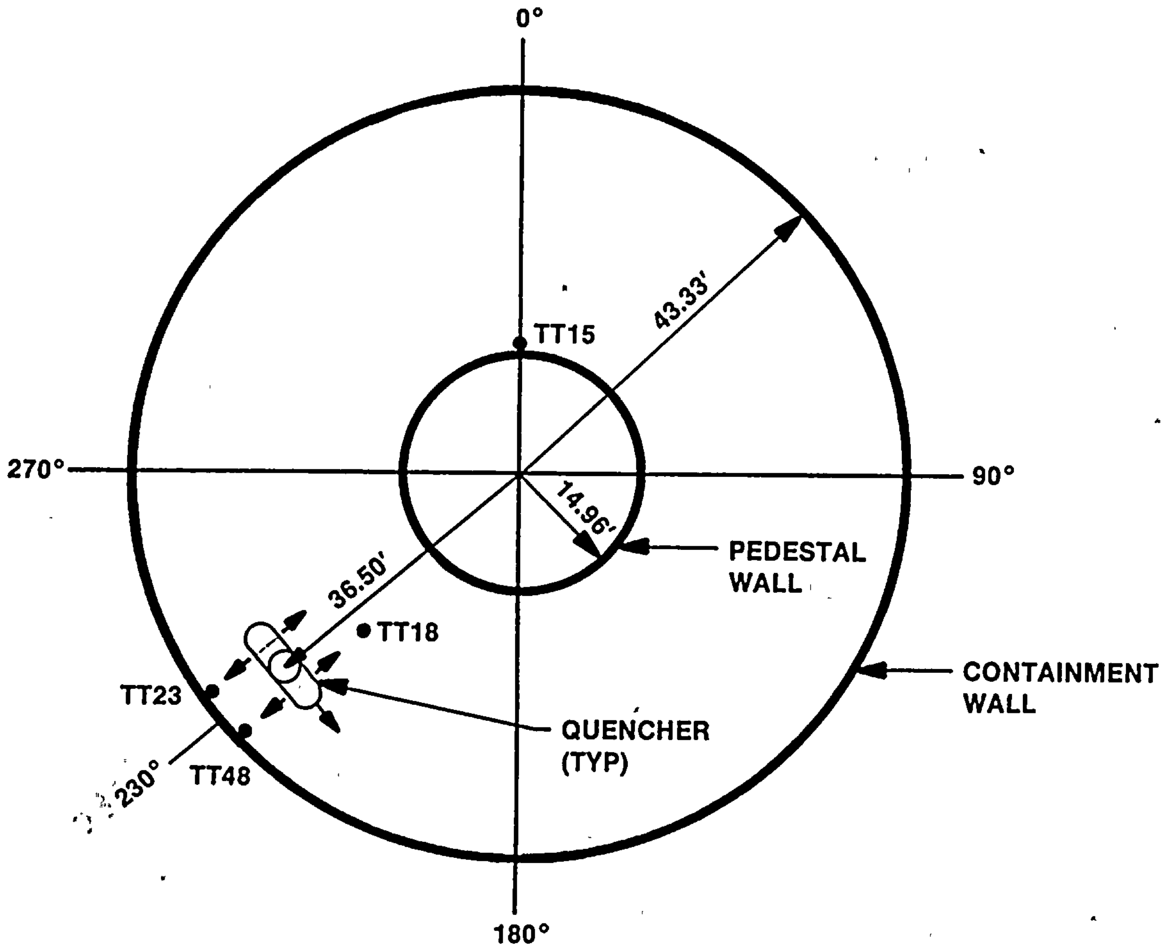
4. Mark II Containment Lead Plant Program Load Evaluation and Acceptance Criteria, USNRC NUREG-0487, October 1978.

5. Suppression Pool Temperature Limits for BWR Containment, USNRC NUREG-0783, November 1981.

6. NEDE-25127, "CAORSO Thermal Mixing Test Final Report," August 1979.

7. Niagara Mohawk Letter From C. D. Terry, Manager, Project Engineering, to C. C. Zappile, Project Engineer, Nine Mile Point Nuclear Station - Unit 2, Stone & Webster Engineering Corporation, Transmitting Temperature Data From the Commonwealth Edison Company LaSalle County Station SRV In-Plant Test, March 4, 1983.

22 8. Pacific Northwest Laboratory. A Three-Dimensional Time Dependent Computer Program for Hypothermal Analysis (TEMPEST), PNL-4348.



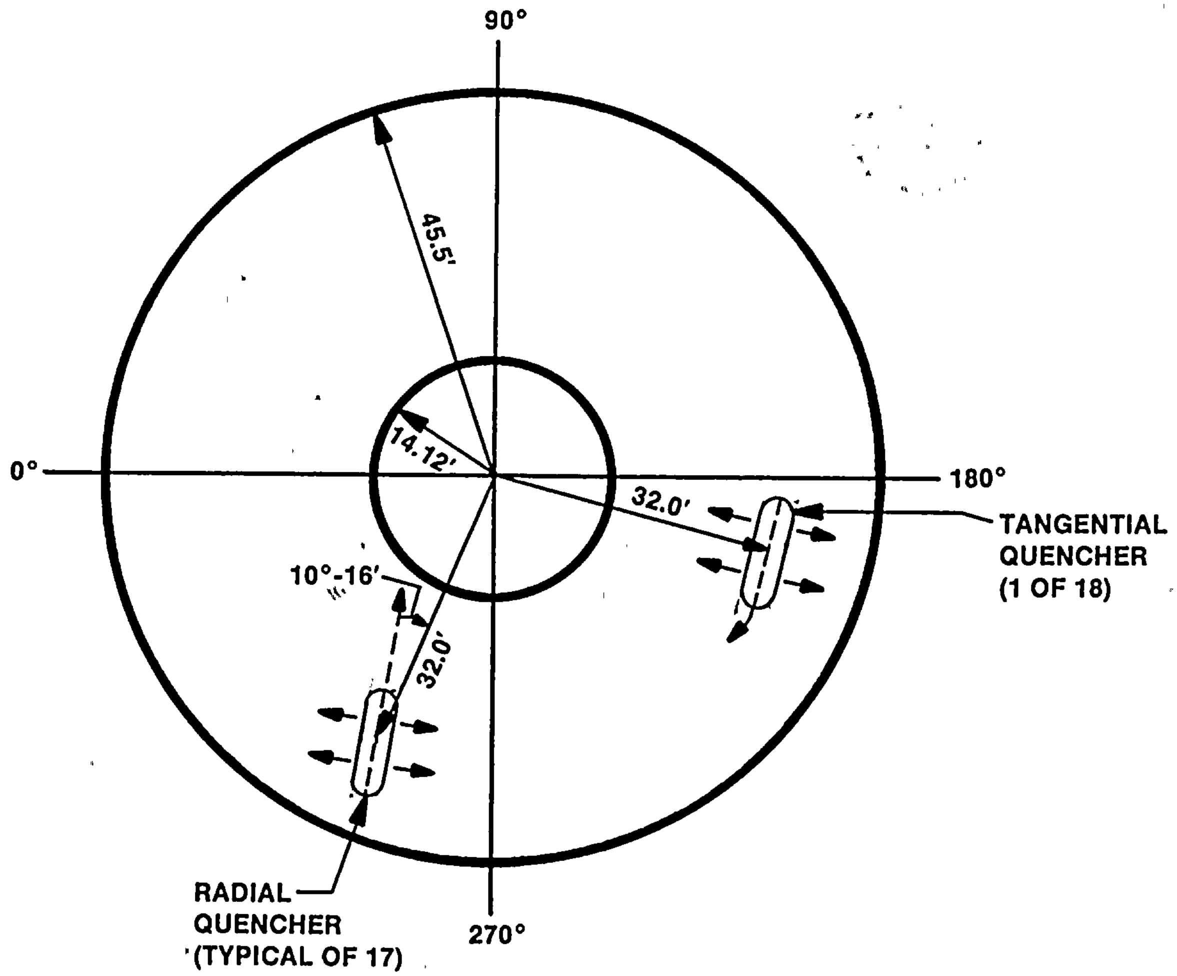
ELEVATIONS

WATER SURFACE		— 26.6'
QUENCHER ℄		— 3.5'
BOTTOM OF POOL		— 0.0'
TEMPERATURE SENSOR	TT15	— 3.5'
	TT18	— 3.5'
	TT23	— 11.4'
	TT48	— 25.7'

FIGURE 6A.10-14

LA SALLE SUPPRESSION POOL SCHEMATIC

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



ELEVATIONS

WATER SURFACE — 24.00'

QUENCHER ℄ — 3.58'

BOTTOM OF POOL — 0.0'

FIGURE 6A.10-15

NMP2 SUPPRESSION POOL SCHEMATIC

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

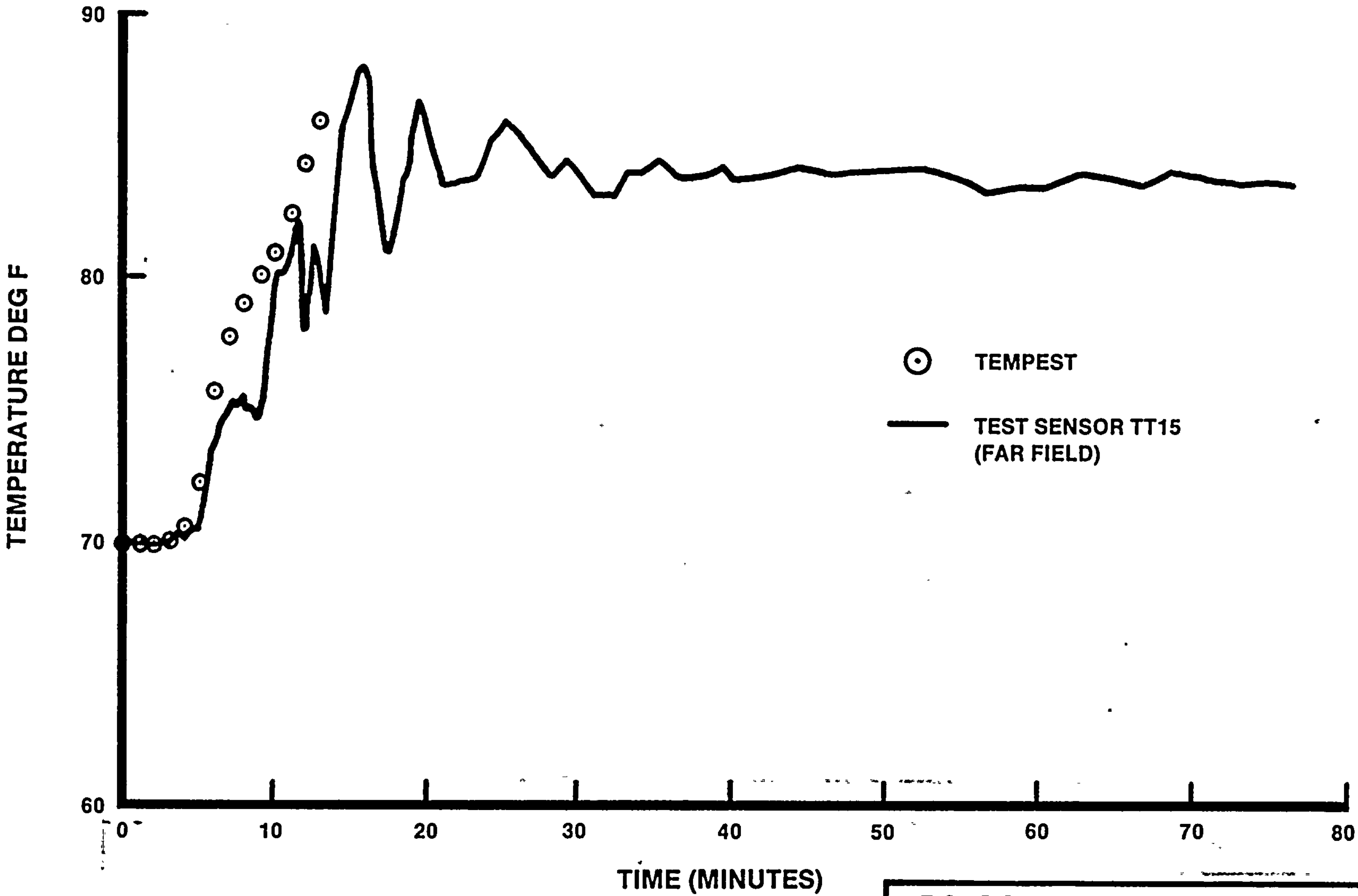


FIGURE 6A.10-16

TEMPEST BENCHMARK COMPARISON
WITH LA SALLE POOL TEMPERATURE
TEST DATA

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

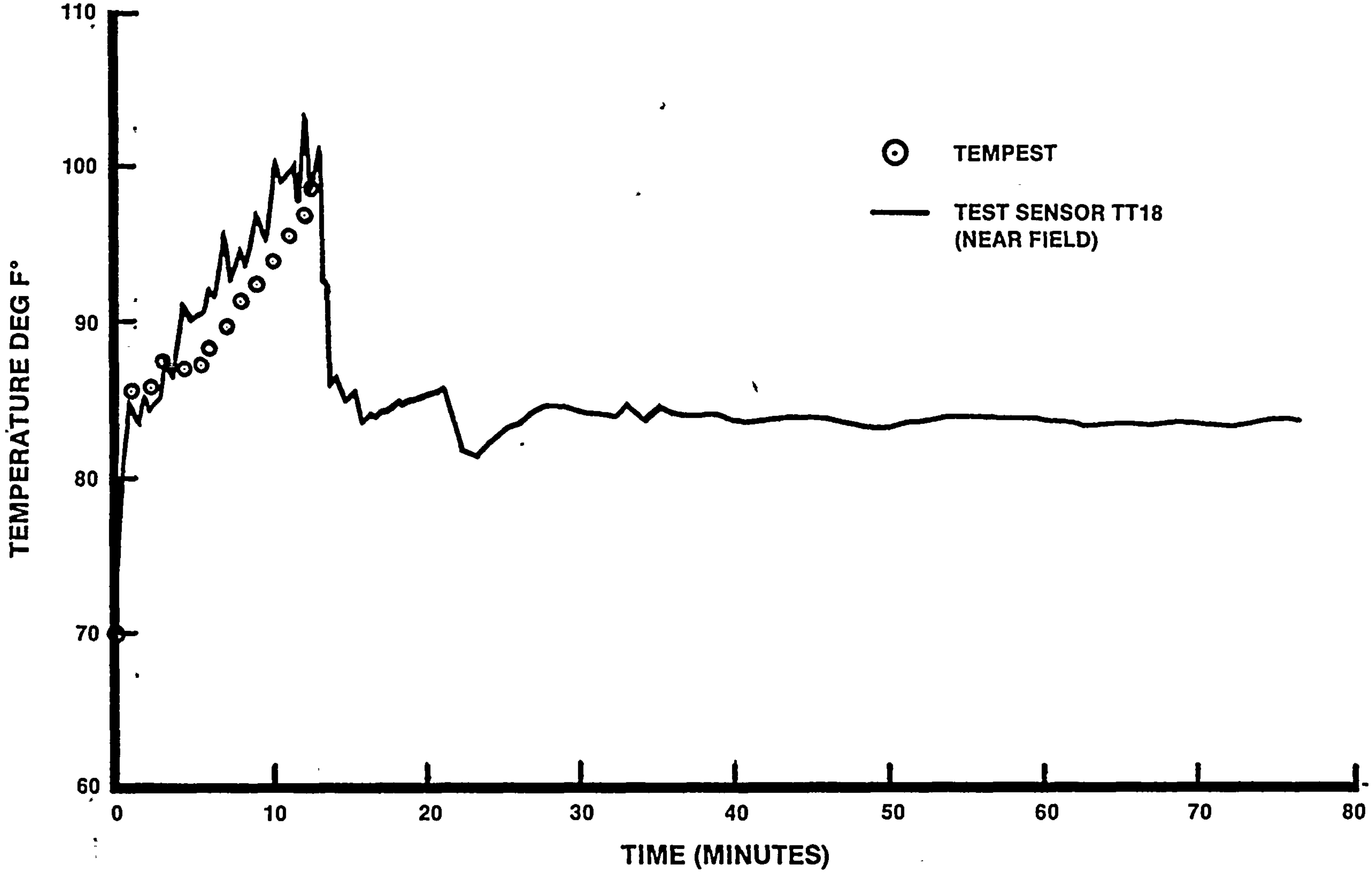


FIGURE 6A.10-17

TEMPEST BENCHMARK COMPARISON WITH LA SALLE POOL TEMPERATURE TEST DATA

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

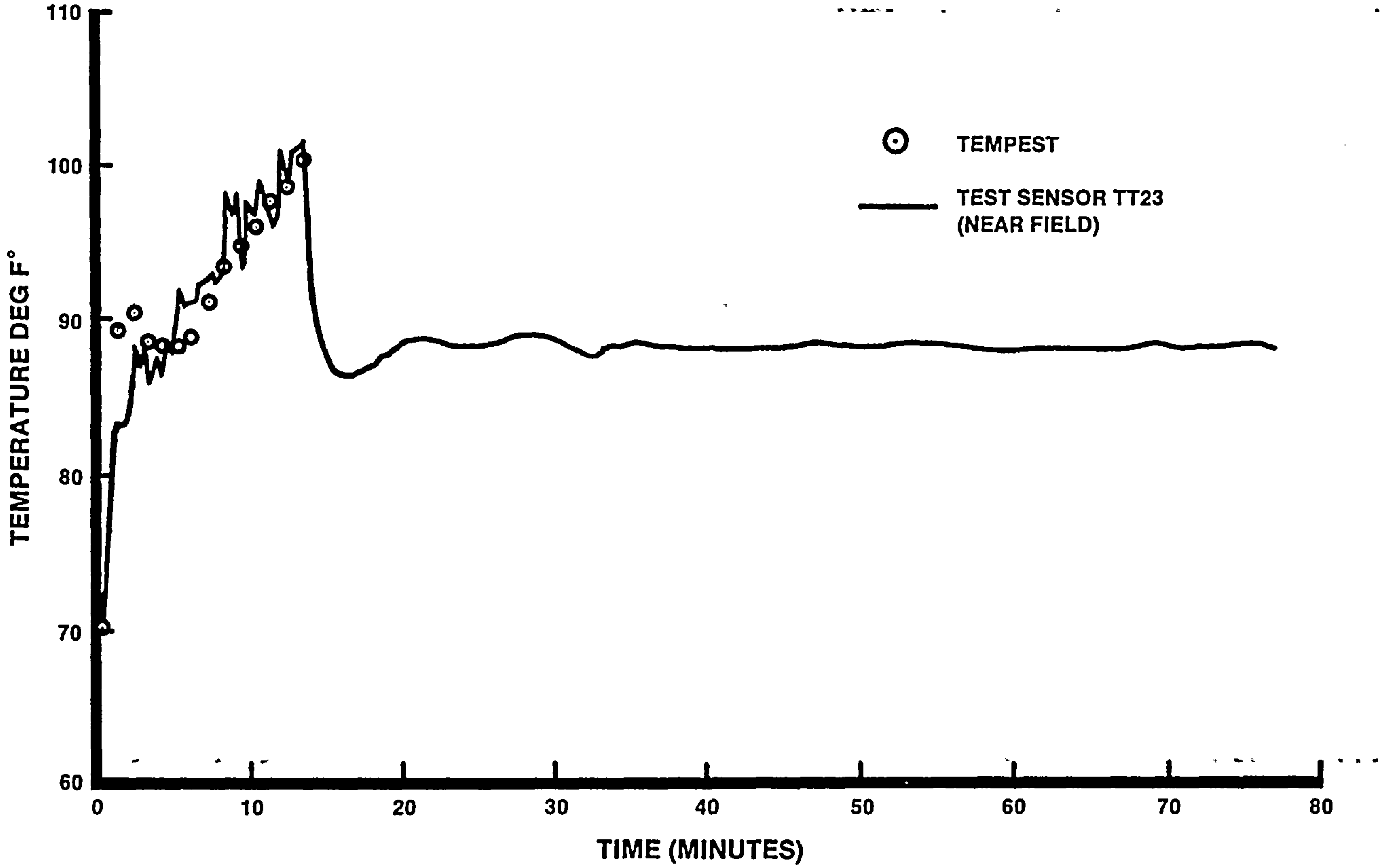


FIGURE 6A.10-18

TEMPEST BENCHMARK COMPARISON WITH LA SALLE POOL TEMPERATURE TEST DATA

NIAGARA MOHAWK POWER CORPORATION

NINE MILE POINT-UNIT 2

FINAL SAFETY ANALYSIS REPORT

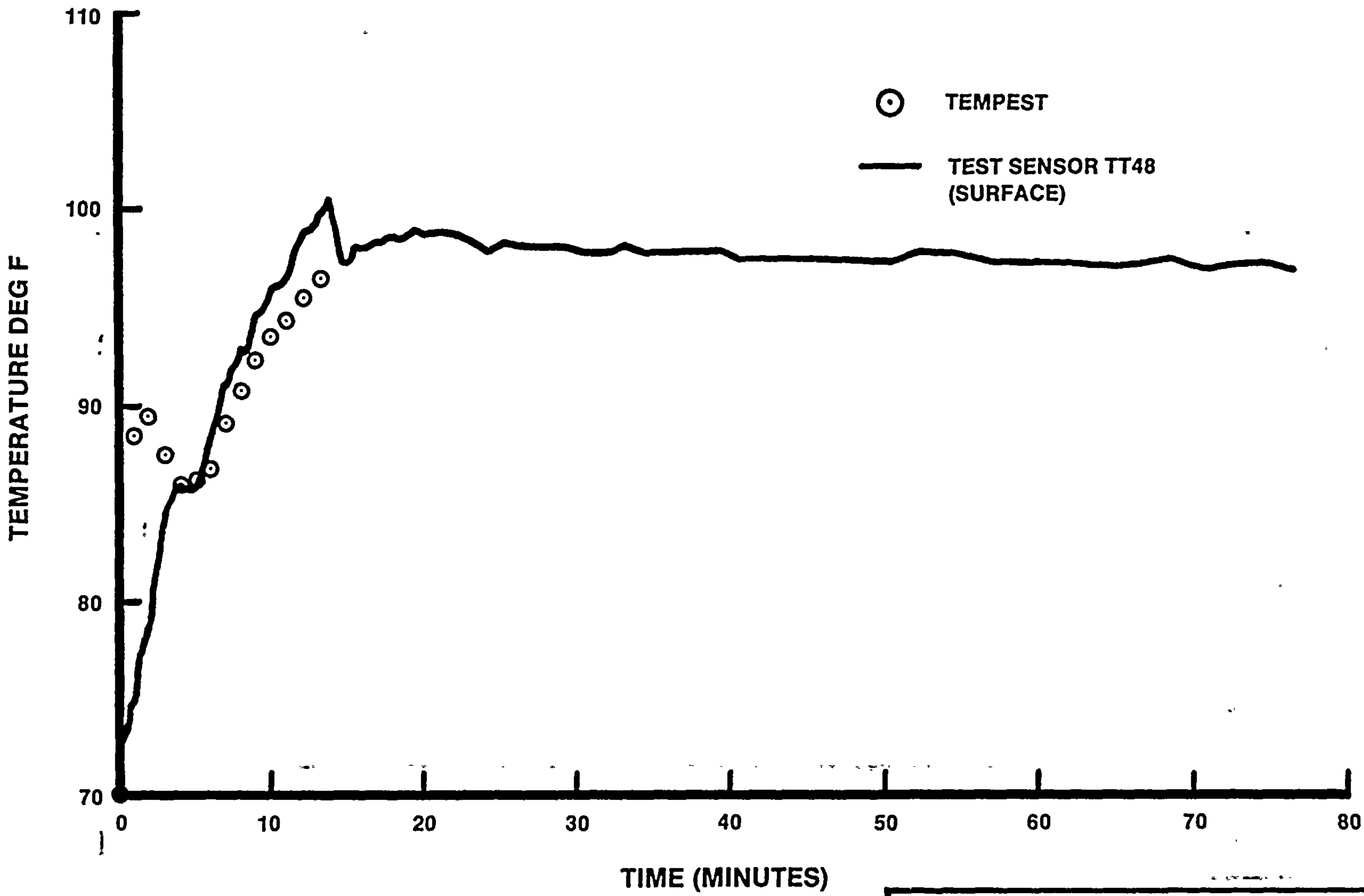


FIGURE 6A.10-19

TEMPEST BENCHMARK COMPARISON
WITH LA SALLE POOL TEMPERATURE
TEST DATA

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NINE MILE POINT-UNIT 2
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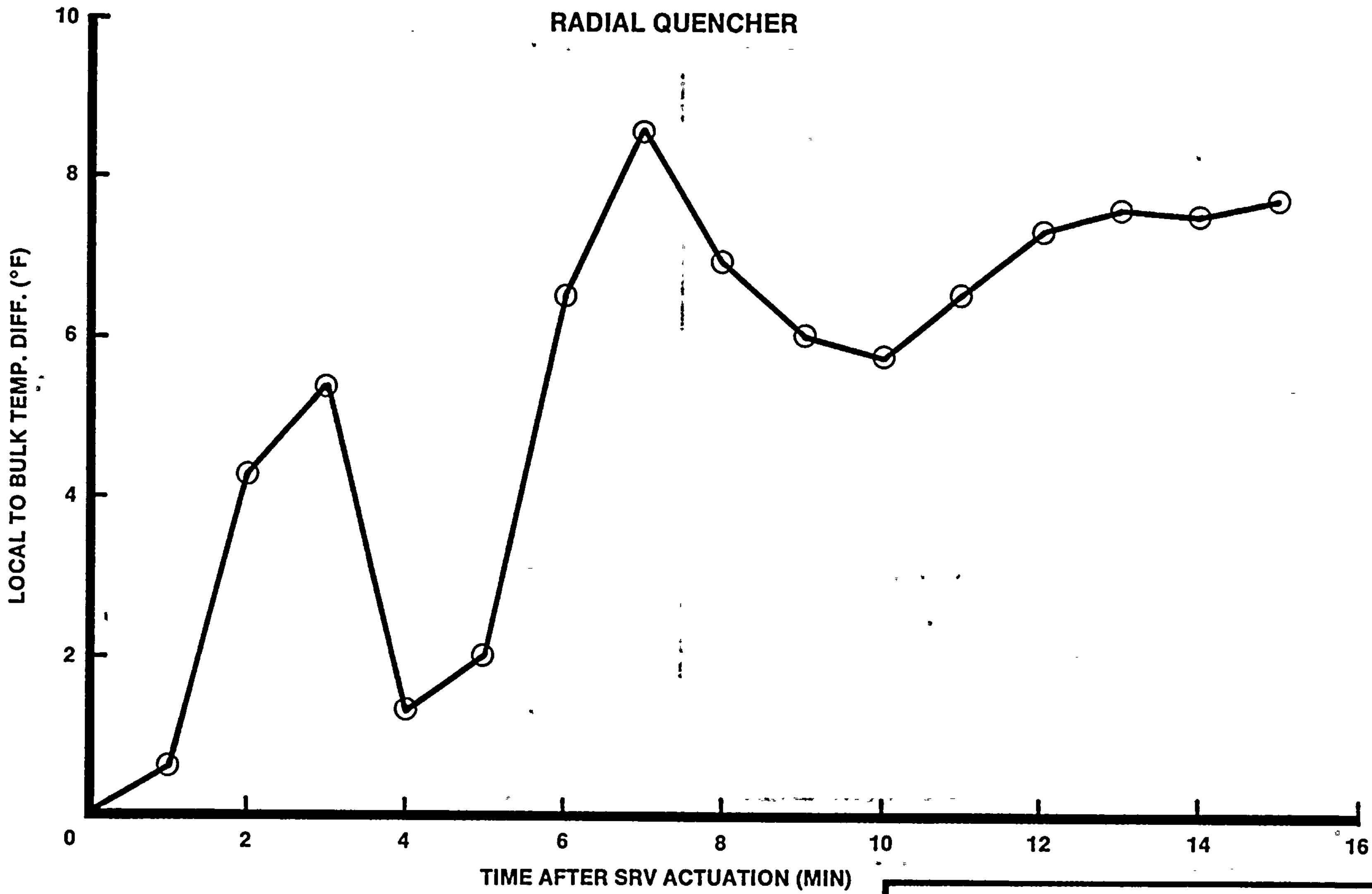

FIGURE 6A.10-20

SUPPRESSION POOL LOCAL TO BULK TEMPERATURE DIFFERENCE

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