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March 15, 1983
4410-83-L-0052

TMI Program Office
Attn: Mr. L. H. Barrett, Deputy Program Director
US Nuclear Regulatory Commission
c/o Three Mile Island Nuclear Station
Middletown, PA 17057-0191

Dear Sir:

Three Mile Island Nuclear Station, Unit 2 (TMI-2)
Operating License No. DPR-73
Docket No. 50-320
Polar Crane SER - Additional Information

Attached for your information is additional information on the refurbished Polar Crane Safety Evaluation Report. This information is provided as a result of questions by your staff. NRC staff comments are listed by SER section number followed by GPUNC's response.

If you have further questions, please contact Mr. J. J. Byrne of my staff.

Sincerely,

B. K. Kanga

B. K. Kanga
Director, TMI-2

BKK/RBS/jep

Attachment

CC: Dr. B. J. Snyder, Program Director - TMI Program Office

U.S. NUCLEAR REGULATORY COMMISSION
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POLAR CRANE LOAD TEST SER

(NRC Comments and Responses Thereto)
(Comments listed by SER section number)

Comment 2.4: Provide stress analysis of the D-ring wall at point of contact with stacked missile shields.

Response 2.4 A stress analysis is not specifically needed since the missile shields have previously been stacked on the D-rings safety using wood blocking to elevate them above interferences. The D-ring walls are designed for this function. The present plan substitutes steel for the wood blocks to avoid bringing combustibles into containment. For further information on design of the D-rings, see Section 3.8.3 of the FSAR.

Comment 3.1 Provide a list of any exception taken to ANSI Standards for the crane and lift rigging.

Response 3.1 Exceptions taken to ANSI standards for the crane are described in the Polar Crane Functional Description, 2-M72-MH02, submitted as Revision 0 to the NRC via 4410-82-L-0021 dated October 8, 1982. Exceptions to ANSI standards for lift rigging were provided via GPUN letter 4410-83-L-0004 dated January 4, 1983.

Comment 3.2 Provide explanation of the statement, "Two individual though not redundant rope systems."

Response 3.2 The main hoist rope system includes two lengths of rope tied to two hoist drums and through a tandem reeved load block. The two ropes acting together are sized for the original 500 ton design rating of the crane.

Comment 3.2 Provide a clarification statement to the effect that the crane was originally designed to 500 tons.

Response 3.2 The original design capacity of the crane is stated in Section 9.1.4.3 of the FSAR.

Comment Figure 3.3-1

Provide an explanation of the load rating on the Dillon load cell, including the factor of safety to ultimate (breaking) strength. Clarify the "pull to 220 tons."

Response Figure 3.3-1

The factor of safety on the Dillon load cell is a 3 to yield and 5 to ultimate based upon a 200 ton load.

The 0-200 ton Dillon load cells have been recalibrated to measure the range 20-220 tons. The wording "pull to 220 tons" refers to the manner in which the calibration laboratory performed this task. Further detail of the calibration sequence would show that routine calibration techniques were used.

Comment Figures 3.3-1 and 3.3-2

Provide an analysis of load-bearing members in the lift rigging to be used. List design ratings and breaking strength.

Response Figures 3.3-1 and 3.3-2

Attachment 1 provides a comparison of design stress versus allowable stress or a factor of safety to yield and to failure of all the load-bearing members, excluding the head lift rig. The head lift rig was designed for a 170 ton load and tested to 255 tons. This lift rig has been used to remove the TMI-2 head twice.

Attachment 2 provides an evaluation of design loads and stresses for the RV head and internals handling equipment.

Comment 4.0 General comment. Do not use the word "probability" without quantification. Either quantify the various probabilities mentioned or provide a clarification to the effect that the term "probability" should be interpreted to mean likelihood.

Response 4.0 The interpretation stated above is correct. The word "probability" is not meant to be interpreted in the mathematical sense but rather is synonomous with "likelihood."

Comment 4.1.3.1 Last paragraph. Quantify any postulated release of radioactivity.

Response 4.1.3.1 An excerpt from the head removal safety evaluation which quantifies the postulated releases of radioactivity is repeated below:

> During head removal activities, there is a remote possibility that the krypton-85 which is assumed to be in the reactor core may be released. This remaining krypton-85 may be securely trapped in the grain boundaries of fuel pellets or in intact fuel rods. An analysis of the potential release was performed, based on the following assumptions.
>
> - Krypton-85 inventory at shutdown (March 28, 1979) is 9.6×10^4 curies.
> - Known releases of krypton-85 inventory are 44,600 curies (reference 2). This is the quantity released during the June-July 1980 reactor building purge. All other releases are negligible.

- The remaining krypton-85 is decayed to January 1, 1983.
- The offsite doses are based on an instantaneous release of the remaining krypton-85.
- An accident X/Q of 6.1×10^{-4} sec/m^3 is used.

These assumptions yield a maximum release of 37,400 curies of krypton-85. Using Regulatory Guide 1.109 methodology and curies to dose conversion tables, the maximum site boundary total body dose is 12 mrem.

These calculations are considered conservative since accident X/Q is used, when in reality, the purge will be isolated. Therefore, release meteorology can be controlled reducing doses by as much as a factor of 100. Also, the assumption of Krypton remaining is conservative since RCS venting activities have not shown significant quantities of Krypton.

Comment 4.2.4 (B) State the quantity of unborated water that could be delivered to the sump as a result of load drops.

Response 4.2.4 (B) Attachment 3 provides the water volume by system of those systems that could be affected by load drop.

Comment (Table 4.2-1): Clarify the intent of using hazard elimination Category E.

Response: Hazard elimination Category E was used with the intent to state that analyses of postulated damage to safety-related equipment have shown that the consequences of this damage are not significant with respect to maintaining required safety functions as stated in the SER.

Comment 5.2 (2) Provide summary of or reference to recent decay heat analyses regarding losses to ambient at reduced RV water level.

Response 5.2 (2) A summary of the referenced analysis is provided on Attachment 4.

Comment 7.0 Include a description of the NDE Program on the main hook.

Response 7.0 The NDE of the main hook will be performed in accordance with GPUNC procedure MTIS-004 Revision 3 and work package M0043. These have been provided to the NRC.

ADDITIONAL COMMENTS:

Comment A Clarify whether or not each indivual set of redundant main hoist brakes is capable of holding the test load.

Response A Each individual set of redundant main hoist brakes is capable of handling the test load.

Comment B — Will the purge be isolated during the actual lifting times of the load test?

Response B — The purge system will be isolated during those times when a missile shield or the test load is actually being lifted by the polar crane.

Comment C — Provide a summary of laboratory test report on wire rope sample.

Response C — A summary of the laboratory test is provided as Attachment 5. The complete laboratory test is available for inspection at your request.

Comment D — Clarify what loads are served by the out-of-containment power supply breaker that would be used to interrupt power to the polar crane.

Response D — MCC-2-32A, the 480V supply breaker in question supplies power only to the polar crane. No other loads are supplied by this breaker.

Comment E — Explain the selection basis for the chosen load path as shown on the figures in Section 2 for movement of the missile shields from the test load frame to their storage location on the "B" D-ring. Refer to figures 2.4-2 and 2.4-3. Specifically, state the reasoning by which clockwise rotation was selected over counter-clockwise rotation.

Response E — Load path selection was based on maximizing the simplicity of movements and minimizing the number of manipulations required to transport the subject loads, as well as considering the load drop analysis as described in the SER.

Comment F: Provide an analysis of load-bearing members of the load test frame. List design ratings and breaking strengths.

Response F: Attachment 2 of the first response to verbal comments provides a comparison of computed stress versus allowable stresses for load-bearing members of the test frame.

Comment G: Verify that rigging components associated with the load test have been certified to appropriate ANSI standards.

Response G: Newly-designed and/or procured slings associated with the load test have been certified to meet appropriate ANSI standards. Other portions of the lifting assembly will be verified by a combination of testing and inspections as described in Section 3.3 of the SER.

POLAR CRANE LOAD TEST ASSEMBLY
(Shown on Drawing 2-COP-1301)

Component and Drawing Detail	Type of Stess	Computed Stress (KSI) (See Note 1)	Allowable Stress (KSI) (See Note 2)
Extension Bar (1), DET. 6	Tension	7.4	16.2
	Shear	7.4	14.5
5½" ϕ Pins (3), ELEV. A	Shear	6.8	14.5
	Bending	14.9	27.0
Lifting Plates (6), DET. 2	Tension	7.3	16.2
	Shear	7.3	14.5
Base Plates for Lifting Plates (3), DET. 2	Bending	17.6	27.0
	Shear	3.5	14.5
Lifting Plates (4), DET. 3	Tension	6.4	16.2
	Shear	5.7	14.5
Base Plates for Lifting Plates (4), DET. 3	Bending	8.2	27.0
	Shear	2.3	14.5
Load Spreading Frame, Partial Plan B			
– W27 x 146	Bending	11.9	22.0
	Shear	8.9	14.5
– W27 x 178	Bending	14.0	22.0
	Shear	3.6	14.5
– W30 x 173 (2)	Bending	14.1	22.0
	Shear	5.5	14.5
1 3/4" ϕ Wire Ropes (8) and Fittings, ELEV. A	Tension	(See Note 3)	(See Note 3)
Lifting Plates (8), DET. 1	Tension	7.1	16.2
	Shear	7.2	14.5

Component and Drawing Detail	Type of Stress	Computed Stress (KSI)	Allowable Stress (KSI)
Load Frame, Partial Plan C			
– W24 x 104 (4)	Bending	12.8	24.0
	Shear	4.6	14.5
– L'S 4 x 4 x ½	Compression	3.4	17.8
<u>Bolted Connections</u> Load Spreading Frame 7/8" ϕ Bolts (20), SECT. L	Shear	12.2	30
Load Spreading Frame 7/8" ϕ Bolts (24), SECT. K	Shear	20.4	30
Base Plate for Lifting Plates 1½" ϕ Bolts (18), DET. 2	Tension	19.9	44
Base Plate for Lifting Plates 1½" ϕ Bolts (16), SECT. H	Tension	22.8	44
Load Frame 3/4" ϕ Bolts (32), SECT. D	Shear	14.8	30
<u>Welded Connections</u> Load Spreading Frame 3/8" Fillet Weld, SECT. L	Shear	10.5	21
Load Spreading Frame 7/16" Fillet Weld, SECT. K	Shear	14.1	21
Lifting Plates 3/4" Parial Pen. Weld, DET. 2	Tension	4.7	21

Component and Drawing Detail	Type of Stress	Computed Stress (KSI)	Allowable Stress (KSI)
Lifting Plates - 3/4" Partial Pen. Weld, DET. 3	Tension	6.5	21

Notes:

1. Computed stresses are based on a vertical design load of 220 tons.

2. Allowable stresses are based on the latest edition of the AISC "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings."

3. Wire ropes and fittings have a factor of safety greater than 5 with respect to their ultimate strength.

4. Any load test assembly components not listed in the above table are secondary members provided for lateral support only.

5. The existing head and internals handling fixture assembly and turnbuckle pendant assemblies shown on drawing 2-COP-1301 were furnished by B&W. These components were designed for a lifted load of 170 tons and were load-tested to 255 tons by B&W.

LOAD CELL RIGGING
(Shown on Drawing 2-COP-1302)

Component and Drawing Detail	Yield Safety Factor (Design Load, See Note 1)	Ultimate Safety Factor (Design Load, See Note 1)	Yield Safety Factor (220 Tons)	Ultimate Safety Factor (220 Tons)
10" ϕ Cylinders (2), DET. 1	3.8	5.1	3.5	4.6
Attachment PL (2), DET. 2	4.1	6.6	3.7	6.0
7" ϕ Pin (2), DET. 3	4.0	5.4	3.6	4.9
10½" ϕ Pin (1), B&W Mark 228	7.9	9.9	7.2	9.0
Bore Plates (2), B&W Mark 223	4.1	6.6	3.7	6.0

Notes:

1. The design load for the load cell rigging components listed above is 200 tons.

2. The additional components shown in elevation with the load cell rigging but not listed in the table above are parts of the existing internals handling extension furnished by B&W. These components were designed for a lifted load of 176 tons and were load-tested to 264 tons by B&W.

MISSILE SHIELD RIGGING
(Shown on Drawing 2-COP-1301)

Component and Drawing Detail	Yield Safety Factor (Design Load)	Ultimate Safety Factor (Design Load)
Attachment Plate (1) Shown in DET. 5	6.8	10.9
1¼" ϕ Wire Ropes (4) Shown in Elev.	N/A	5.2
75 Ton "Wide Body" Shackles (2) Shown in Elev.	N/A	15.8
1 3/4" Screw Pin Anchor Shackles (4) Shown in Elev.	N/A	12.7
Missile Shield Lifting Lugs (4/panel) shown on B&R Dwg. 4156	3.0	4.9

Notes:

1. Safety factors are based on a vertical design load of 41 tons.

TMI-2 RECOVERY PROGRAM

HEAD & INTERNALS HANDLING EQUIPMENT
DESIGN LOADS & STRESS
EVALUATION SUMMARY

Compiled By

J. G. Sparks
Mechanical Engineering

BABCOCK & WILCOX
UTILITY POWER GENERATION DIVISION
LYNCHBURG, VA 24505

FEBRUARY 1983

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HEAD & INTERNALS HANDLING FIXTURE AND INTERNALS HANDLING EXTENSION

1. The design load for the Fixture and Extension was as follows:

   (360 KIPS) X (3G's - Allowance for shock or impact effect)

2. The allowable stresses were as follows:

   A. Yield strength (From no particular code) - Where tensile, bending, bearing and shear stresses were concerned.

   B. AISC Manual - Where compressive and weld stresses were concerned.

   C. AISC Manual - Where combined (Compressive and bending or tensile and bending) stresses were concerned except that in the equations the yield strength (Fy) was used for F_b and .6Fy.

TURNBUCKLE PENDANT

1. The design loads were 120 KIPS for the turnbuckle and 360 KIPS for the remaining part numbers.

2. The allowable stress for all part numbers except the turnbuckle was the yield strength (from no particular code).

3. The proof (roughly, yield) and ultimate loads for the turnbuckle were 244.4 KIPS (S.F.2 x 122.2 KIP safe working load) and 611 KIPS (S.F.5 x 122.2 KIP safe working load) respectively.

HANDLING FIXTURE SLING

1. The design load was 360 KIPS

2. The ultimate load for the wire rope was 818 KIPS.

HEAD AND INTERNALS HANDLING FIXTURE ACTUAL VS. ALLOWABLE STRESSES (KSI)

Part No.	Tensile		Bending		Combined Stress Factor	Compressive		Shear		Weld Stress		Bearing		Yield Strength Sy	Ultimate Strength Su
	ft	Ft	fb	Fb		fa	Fa	fv	Fv	Act.	All.	fp	Fp		
247&248	20.3	36	11.6	36	.89 < 1.0	-	-	-	-	-	-	-	-	36	58
249	-	-	21.3	36	.98 < 1.0	11.6	34.8	-	-	-	-	-	-	36	58
245	-	-	-	-	-	10.3	20.1	-	-	-	-	-	-	40	70
241	9.6	36	6.6	36	-	-	-	6.6 (Tearout by P/N 320)	36	-	-	11.4 (P/N 320 Against P/N 241)	36	36	58
320	-	-	18.2	66	-	-	-	8.3	66	-	-	12.5	66	66	100
241 to 247 and 249 weld	-	-	-	-	-	-	-	-	-	11.4	18	-	-	36	58
251	1.5	40	-	-	-	-	-	-	-	-	-	-	-	40	70
242	-	-	8.4	36	-	-	-	-	-	-	-	-	-	36	58
249 to 243 weld	-	-	-	-	-	-	-	-	-	13.2	18	-	-	36	58
240	20.7	36	-	-	-	-	-	18.7 (Tearout)	36	-	-	28.7 (Pin Bearing)	36	36	58
247 or 248 to 240 weld	-	-	-	-	-	-	-	-	-	12	18	-	-	36	58

INTERNALS HANDLING EXTENSION ACTUAL VS. ALLOWABLE STRESSES (KSI)

Part No.	Tensile ft	Tensile Ft	Shear fv	Shear Fv	Bearing fp	Bearing Fp	Bending fb	Bending Fb	Yield Strength Sy	Ultimate Strength Su
223	24	36	21.1 (Tearout)	36	23.6 (P/N 229 Against P/N 223)	36	-	-	36	58
224	23.3	36	20.9 (Tearout)	36	28.7 (P/N 230 Against P/N 224)	36	-	-	36	58
222	34.4 *	50	30.5 (Tearout) *	50	38.2 (P/N 230 Against P/N 222) *	50	-	-	50	70
228	-	-	6.2	40	15.7	40	12	40	40	80
229	-	-	14	66	23.7	66	41.1	66	66	100
230	-	-	17.6	66	28.8	66	33.8	66	66	100

*Effect of part numbers 225 and 226 and associated welding was neglected in the calculation of this stress.

TURNBUCKLE PENDANT ACTUAL VS. ALLOWABLE STRESSES (KSI)

Part No.	Tensile		Shear Tearout		Thread Shear		Pin Bearing		Yield Strength Sy	Ultimate Strength Su
	ft	Ft	fv	Fv	fv	Fv	fp	Fp		
375	9.9	36	8.9	36	8	36	12.2	36	36	58
376/377	43.2	52	-	-	10.6	52	-	-	52	90

*HANDLING FIXTURE SLING END FITTING ACTUAL VS. ALLOWABLE STRESSES (KSI)

Part Name	Shear		Tensile		Bearing		Bending		Yield Strength	Ultimate Strength
	fv	Fv	ft	Ft	fp	Fp	fb	Fb	Sy	Su
Open Socket	8.9 (Tearout)	50	16	50	13.7	50	-	-	50	80
Closed Socket	9.2 (Tearout)	50	15.9	50	12.5	50	-	-	50	80
5¼" Dia Pin	8.3	40	-	-	13.7	40	18.2	40	40	80

*Assumptions concerning socket geometry and socket and pin materials were made; therefore, the actual stresses in this table are subject to verification.

Water Volume for Unborated Water System Inside Containment

1) Reactor Building Fire Protection: FS-V639-FS-V637, Penetration No. R-543, 1,800 gal.

2) Reactor Building Demineralized Water: Penetration No. R-535, DW-V31-DW-V141, 100 gal.

3) Reactor Building Nuclear Services Closed Cooling Water: Penetration No. R-557 and No. R-558, 900 gal.

4) Reactor Building Intermediate Closed Cooling Water Penetration No. R-563 and No. R-567, 500 gal.

5) Reactor Building Normal Cooling:

   A. Inside the Reactor Building - Penetration No. R-577, R-579, R-580, R-584, R-587, R-588, R-585, R-581, R-582 and R-578 (normal cooling coils), 6,400 gal.

   B. Total Reactor Building Normal Cooling - Volumes consisting of Part A plus the volume outside the Reactor Building, 12,200 gal.

6) Secondary Side of SG Including Steam and FW Piping, 41,000 gal/loop. B Generator drained and A Generator at head removal level and MS and FW lines drained, 12,488 gal.

ADDENDUM

TO THE

TMI-2 DECAY HEAT REMOVAL REPORT

OF APRIL 1982

Prepared by

G. A. Hipp
L. L. Losh
E. R. Miller

Prepared for

GPU Nuclear Corporation
Under Master Services Contract 595-3174

by

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ADDENDUM

to the

TMI-2 DECAY HEAT REMOVAL REPORT

of April 1982

Contents

ABSTRACT

A conservative calculation of equilibrium temperatures and heatup rates for the reactor coolant system (RCS) draindown to head removal level (321.5 ft. elevation) has determined that the draindown with no supplemental heat removal can be accomplished after December 1, 1982 without exceeding the temperature criterion. A similar conservative analysis for RCS draindown to the bottom of the reactor vessel nozzles (314 ft. elevation) supports draindown after January 1, 1984 without exceeding the temperature criterion. The criterion is that fluid temperatures do not exceed 170°F.

These conservative calculations were made with models originally developed in the TMI-2 Decay Heat Removal Analysis of April 1982. In addition, best estimate models, benchmarked to temperatures measured following the partial draindown for the Quick Look inspection, were developed and used to predict the expected reactor coolant system heatup following the draindown to head removal level and draindown to the bottom of the reactor vessel nozzles. The best estimate models predict that draindown with no supplemental heat removal can be accomplished after December 1, 1982 for both draindown levels without exceeding the 170°F temperature criterion.

INTRODUCTION

The TMI-2 reactor has been in the decay heat natural circulation cooling mode for the past several years. In July 1982, the reactor coolant system (RCS) was partially drained to permit access for the Quick Look inspection. The next step in the recovery process called for draining down the RCS further to allow removal of the reactor vessel head. An analysis was performed to determine whether the TMI-2 decay heat loss to containment is sufficient to support the RCS draindown to head removal level (321.5 Ft. elevation) without exceeding the temperature criterion. It was concluded that the draindown to reactor head removal level can be accomplished after December 1, 1982 without exceeding the 170°F criterion.

An additional analysis was performed to determine whether the TMI-2 reactor decay heat loss to containment is sufficient to support the RCS draindown to the bottom of the reactor vessel nozzles (314 Ft. elevation) without exceeding the temperature criterion. The temperatures predicted with the April 1982 conservative models for December 1, 1982 and July 1, 1983 draindown dates exceed the 170°F criterion. This is the result of the large degree of conservatism in the decay heat generation, heat transfer, and heat capacity models.

Two best estimate models have been developed for draindown to the reactor vessel nozzle level; one including the hot legs' and steam generators' heat transfer areas and heat capacities and the other not including them. The reason for two best estimate models is uncertainty whether the steam generators would be in effective thermal communication with the core with the cold legs no longer full. Both best estimate models, however, yield temperature prediction well below the 170°F temperature limit.

SIGNIFICANCE OF QUICK LOOK DATA

During the month following the draindown for the Quick Look inspection, the RCS water temperature and reactor building ambient temperature were monitored daily. These data differ significantly from the data upon which the TMI-2 Decay Heat Removal Analysis Report of April 1982 was based in that the new data depict the dynamic temperature response of the RCS rather than "snapshots" of equilibrium temperatures. The advantage of the dynamic data is that it provides an indication of effective system heat capacity which steady-state data cannot. The system heat capacity in turn provides an indication of how much of the RCS is involved in the heat transfer process. Thus the new data provides an opportunity to further refine the existing analytical models and increases confidence in analytical predictions.

ASSESSMENT OF EXISTING MODELS

As a first step, the Quick Look draindown was simulated with the analytical models from the April 1982 analysis. The RCS heatup thus calculated was then compared to the measured RCS temperatures to assess the degree of conservatism in the existing analytical models. The comparison of the predicted and measured temperature trends is shown on Figure 1. As expected, the existing models predict higher RCS temperatures than actually measured. Thus it can be concluded that the models developed in the April 1982 analysis are conservative. Each model will be described briefly.

The existing models to be used in this assessment, decay heat generation, heat transfer, and heat capacity, were those developed in the April 1982 analysis to predict RCS temperatures after partial draindown. The decay heat model provides a conservative calculation of core power based on ANSI/ANS 5.1 - 1979 standard methodology. The decay heat power values for the time frame of interest are shown on Figure 2.

The heat transfer model assumes heat to be transferred only through the reactor vessel walls, lower dome, closure head, and hot legs. This model does not allow any heat transfer through the steam generators or cold legs to assure conservative results. Thus only the reactor building ambient air temperature is needed to predict RCS bulk water temperatures. The reactor building ambient temperatures used in both the previous and the current analyses are from the TMI-2 daily logsheets. Since only one reactor building temperature was recorded, it was necessarily assumed that the ambient air temperature is constant throughout the reactor building. (The April 1982 analysis further assumed that water in the reactor building sump was 60°F and that both RCS loops contribute to heat transfer, and also averaged temperatures for nodes between measured temperatures.) The conservative heat transfer model used in the current analysis is summarized in Table 1.

The heat capacity model includes only the [illegible] of the RCS consistent with the heat transfer model, i.e., only the reactor vessel and the water contained in it. This produces a conservatively small system heat capacity which results in a fast RCS heatup. The heat capacity model is summarized in Table 2.

Having shown the existing models to be conservative, the drain down to reactor vessel head removal level can be simulated. The equilibrium temperatures and heatup rates thus calculated should be appropriate for licensing submittals.

CONSERVATIVE ANALYSIS OF DRAINDOWN TO HEAD REMOVAL LEVEL

Using the models as developed in the April 1982 analysis and as described in the preceding section with slight modification, the further RCS draindown to reactor vessel head removal level was simulated. The modifications to reflect the further draindown were a reduction in heat transfer area and a reduction in system heat capacity. It was assumed that no heat would be transferred through the reactor vessel head when drained down. This assumption reduces the reactor heat transfer area by 170 square feet. In addition, the lowering of the RCS water level reduces the system heat capacity. The resulting models are shown in Table 3.

The objective of this analysis was two-fold:

1) To determine the equilibrium RCS bulk water temperature on several specific dates given the reactor building ambient temperature.

2) To determine the RCS heatup rate starting at specified initial temperatures on specified dates.

The dates in question are December 1, 1982, July 1, 1983, and January 1, 1984. The reactor building ambient temperatures are 70°F in winter and 85°F in summer, and the initial RCS temperatures are 100 and 130°F.

The method used to calculate the equilibrium RCS temperatures is based upon the equation: $Q = \Sigma UA(t_{RCS}-t_{AMB})$

where: Q is decay heat

U is the air side film coefficient (since it is dominant)

A is the surface area

t_{RCS} is the reactor vessel bulk water temperature

t_{AMB} is the reactor building ambient temperature

This equation can be solved for t_{RCS} since the values of all the other terms are known:

$$t_{RCS}=t_{AMB}+Q/\Sigma UA.$$

The results of the equilibrium RCS temperature analysis with the RCS drained down to head removal level are as follows:

Date	Equilibrium RCS Water Temperature
December 1, 1982	$165.3^{o}F$
July 1, 1983	$158.0^{o}F$
January 1, 1984	$130.3^{o}F$

It is significant to note that all of these equilibrium temperatures are less than the $170^{o}F$ criterion adopted to maintain a positive margin to boiling.

The method used to calculate the RCS heatup rates following draindown is based upon the equation: $Q_{NET}(t)=\Sigma mc_p(t_{NEXT}-t_{NOW})$

where: $Q_{NET}(t)$ is the difference between decay heat generated and heat transferred out, discretized by time

Σmc_p is the system heat capacity

t_{NOW} is the RCS water temperature in the current timestep

t_{NEXT} is the RCS water temperature for the next timestep

This equation can be rearranged and solved for sequential timesteps to calculate the RCS heatup rate starting at a given initial RCS temperature:

$$t_{NEXT}=t_{NOW} + Q_{NET}(t)/\Sigma mc_p$$

The results of this analysis of heatup rates are shown on Figure 3. The temperature traces are asymptotically approaching the equilibrium temperatures calculated above.

The results reported in this section support the conclusion that the RCS can be drained down to reactor vessel head removal level without exceeding the temperature criterion after December 1, 1982. These results, however, are conservative and are not expected to be observed during the actual RCS draindown. The next section identifies the sources of the conservatism in these results in preparation for a best-estimate calculation of temperatures that are expected to be observed during the RCS draindown.

IDENTIFICATION OF CONSERVATISM

In order to quantify the degree of conservatism in the results reported in the preceding section, best-estimate models for decay heat generation, system heat capacity, and heat transfer were generated. The Quick Look temperature data was used for benchmarking best-estimate type models. Once these models were developed, temperatures resulting from the RCS draindown to reactor vessel head removal level were calculated.

I. Decay Heat

The ANSI decay heat prediction method is believed to be very conservative in its treatment of the neutron absorption factor (G factor) which causes high decay heat predictions during the time frame of interest. The TMI-2 decay heat analysis based upon the LOR-2 code (the B&W version of ORIGIN) is estimated to provide a more realistic prediction or best-estimate of the decay heat power levels. A comparison of the LOR-2 and ANSI based decay heat power levels is shown in Figure 4. The LOR-2 based decay heat power levels were used for best-estimate purposes.

II. System Heat Capacity

The system heat capacity was expanded significantly to reproduce the shape of the measured Quick Look temperatures. The physical description of the best-estimate system heat capacity is shown on Table 4. Minor core and reactor vessel internals contributions were added along with hot leg piping. The major new contributors, however, were the steam generators and the primary and secondary side water in them. Only 50% of the total available steam generator/water heat capacity, however, was needed to reproduce the measured temperature trace. This magnitude of effective contribution to system heat capacity appears credible and was thus assumed for best-estimate purposes.

III. Heat Transfer

The heat transfer model was expanded to remain consistent with the system heat capacity model. In addition to the reactor vessel, the new heat transfer model included all of the hot legs and the steam generators. Since the hot leg to ambient and steam generator to ambient temperature difference is not known, a factor was determined which could be applied to the core to ambient temperature difference to estimate the effective hot leg or steam generator to ambient temperature difference. This factor, .27, balances the heat transfer to produce the measured terminal temperature at the end of the RCS heatup. One other refinement was made to the heat transfer coefficients. The constant value coefficients developed in the April 1982 analysis were replaced by temperature difference dependent air film correlations from the ASHRAE handbook. The best-estimate heat transfer model thus developed is summarized in Table 5.

The Quick Look draindown temperatures calculated with these best-estimate models are compared to the measured heatup temperatures in Figure 5. The agreement between measured and calculated temperatures is excellent. These best-estimate models were used to simulate the RCS draindown to reactor vessel head removal level.

BEST ESTIMATE ANALYSIS OF DRAINDOWN TO HEAD REMOVAL LEVEL

Using the best-estimate models developed in the preceding section, the RCS draindown to head removal level can be simulated. Only a few modifications were needed to reflect the further draindown. The system heat capacity was decreased both by the lowered water level on the primary side and the assumed complete draining of the steam generator secondary side water. The total system heat capacity was reduced to 584,132 BTU/^{o}F. The only change to the heat transfer model was to assume that no heat is transferred through the closure head dome.

Using the same methods and reactor building ambient temperatures as before, but with the best-estimate models, the following equilibrium temperatures were determined:

Date	Equilibrium RCS Water Temperature
December 1, 1982	111.6^{o}F
July 1, 1983	120.1^{o}F

The July 1983 equilibrium temperature is higher than the December 1982 temperature (when more decay heat is being generated) because the ambient temperature assumed for July is 15^{o}F higher (85^{o}F versus 70^{o}F). The heatup rates calculated with the best-estimate models assuming an initial RCS temperature of 100^{o}F are shown on Figure 6. Again the temperature traces asymptotically approach the calculated equilibrium temperatures.

CONSERVATIVE ANALYSIS OF DRAINDOWN TO BOTTOM OF REACTOR VESSEL NOZZLES

Several modifications to the existing conservative models developed for draindown to head removal level were made to reflect further draindown to the bottom of the reactor vessel nozzles. It was again assumed that no heat would be transferred through the reactor vessel head when drained down. In addition, because of the even lower water level, no heat was assumed transferred through the upper shell, head support and closure flange. This reduced the reactor heat transfer area an additional 390 square feet and also lowered the system heat capacity. (See Table 6)

The equilibrium RCS bulk water temperatures were determined for the previously specified dates and are as follows:

Date	Equilibrium RCS Water Temperature
December 1, 1982	198.0°F
July 1, 1983	183.1°F
January 1, 1984	151.1°F

As the results above show, the existing conservative models do not predict RCS temperatures within the 170°F temperature criterion until January 1, 1984. The temperatures for earlier dates exceed the 170°F criterion. As noted on page 9, however, these conservative values are not expected to be observed during the actual RCS draindown.

The results of heatup rate calculations based on this conservative model are shown in Figure 7. The temperature traces asymptotically approach the above calculated equilibrium temperatures.

BEST ESTIMATE ANALYSIS OF DRAINDOWN TO BOTTOM OF REACTOR VESSEL NOZZLES

Modifications to the best-estimate models previously developed were made to reflect the further draindown. The system heat capacity was again decreased both by the lowered water level on the primary side and the assumed complete draining of the steam generator secondary side water. Uncertainty as to whether the steam generators would be in effective thermal communication with the core now that the cold legs were no longer full resulted in the development of two best estimate models: one including the heat transfer areas and heat capacities of the hot legs and steam generators and the other not including them. The uncertainty as to whether or not to include the hot legs and steam generators stems from uncertainty as to which of two possible heat transfer mechanisms accounted for the conbributions the hot legs and steam generators made to heat transfer and heat capacity that were deduced from heatup data following the draindown for Quick Look Inspection. One possible heat transfer mechanism is the convection of heated vapor up the hog legs to the steam generators. This mechanism would still function with the RCS water level at the bottom of the reactor vessel nozzles. The other possible mechanism is a stratified convective circulation through the cold legs to the steam generators. This mechanism would be interrupted by the reduced water level. Since the validity of each of the two possible heat transfer mechanisms is unknown. Two best estimate models are postulated. (See Table 7-10).

Using the same methods and reactor building ambient temperatures as before, but with the best-estimate models, the following equilibrium temperatures were determined:

Date	Equilibrium RCS Water Temperature	
	w/hot legs & Steam Gen.	w/o hot legs & Steam Gen.
December 1, 1982	116.8°F	148.7°F
July 1, 1983	124.6°F	151.5°F
January 1, 1984	104.8°F	128.5°F

Again, the July, 1983 equilibrium temperatures are higher than the December, 1982 temperatures because of the higher July ambient temperature (85°F versus 70°F). The heatup rates calculated with the best-estimate models are shown in Figures 8 and 9. These temperature traces also asymptotically approach the calculated equilibrium temperatures.

SUMMARY AND CONCLUSIONS

Both conservative and best-estimate equilibrium temperatures and heatup rates have been determined for draindown to the head removal level and to the bottom of the reactor vessel nozzles. The equilibrium temperatures and heatup rates calculated with the best-estimate models are predictably lower than those calculated with the models from the April 1982 analysis. For draindown to the head removal level (321.5 Ft. elevation), the conservative temperatures and heatup rates show that RCS temperatures do not exceed the 170°F criterion after December 1, 1982. The best-estimate temperatures and heatup rates are felt to be more representative of the expected RCS temperature response to the draindown to head removal level and are in the 110-120°F range. The conservative temperatures and heatup rates for draindown to the bottom of the reactor vessel nozzles (314 Ft. elevation) do exceed the 170°F criterion for December 1, 1982 and July 1, 1983. The best-estimate temperatures and heatup rates for this water level, however, are well below the criterion for all specified dates for the models both with and without hot leg/steam generator heat transfer areas.

It is the conclusion of these analyses that, based on the conservative models from the April, 1982 analysis, the RCS draindown to reactor vessel head removal level can be accomplished without exceeding the temperature criterion after December 1, 1982. Draindown to the bottom of the reactor vessel nozzles is supported by the conservative models from the April 1982 analysis after January 1, 1984. Based on the best-estimate models, however, RCS draindown to the bottom of the reactor vessel nozzles can be accomplished without exceeding the temperature criterion after December 1, 1982. The criterion is that RCS bulk water temperature does not exceed 170°F to insure adequate margin to boiling.

TABLE 1

Original Heat Transfer Model

Component	Heat Transfer Coefficient (BTU/hr-ft^2-^{0}F)	Surface Area (ft^2)
Bottom Head	.34	330
Lower & Intermediate Shells	.72	960
Upper Shells & Flanges	.71	390
Closure Head Dome	.89	170
Hot Leg Piping	.48	900

TABLE 2

Original Heat Capacity Model

Component	Mass (LBM)	Specific Heat (BTU/LBM-^{0}F)		Heat Capacity (BTU/^{0}F)
Reactor Vessel, Head & Studs	881,200	.115		101300
Water	248,500	1.00		248500
			Total	349800

TABLE 3

Conservative Model For Draindown to Reactor Vessel Head Removal Level

1. Heat Transfer Model

Component	Heat Transfer Coefficient (BTU/HR-FT^2-^{o}F)	Surface Area (FT^2)
Bottom Head	.34	330
Lower & Intermediate Shells	.72	960
Upper Shells & Flanges	.71	390

2. Heat Capacity Model

Component	Mass (LBM)	Specific Heat (BTU/LBM-^{o}F)	Heat Capacity (BTU/^{o}F)
Reactor Vessel, Head, & Studs	881,200	.115	101,300
Water	213,487	1.00	213,500
		Total	314,800

TABLE 4

Best Estimate Heat Capacity Model
For Draindown To Reactor Vessel
Head Removal Level

Component	Heat Capacity (BTU/°F)
Reactor Vessel, Head, & Studs	101,300
Core Support Assembly	27,600
Plenum Assembly	11,500
Core	19,100
Hot Leg Piping	18,900
Water (Reactor vessel & hot legs)	270,500
Steam Generators	131,600*
Water (Primary side of stm. gen.)	82,700*
(Secondary side of stm. gen.)	123,700*
Total	786,900

* - Only 50% of the available steam generator and water heat capacity assumed to contribute to system heat capacity to match Quick Look measured temperature trend.

TABL[illegible]

Best Estimate Hea[illegible] For Draindown
To Reactor V[illegible] [illegible] Level

Component	Heat Trans[illegible]r Correlation (Reference)	Surface Area (FT^2)
1) Reactor Vessel		
A. Bottom Head	1*	330
B. Lower & Intermediate Shells	2	960
C. Upper Shell & Flanges	2	390
D. Closure Head Dome	3	170
2) Hot Legs		
A. Reactor Vessel to Thermocouple	4	910
B. Candy Cane	4	530
3) Steam Generators	4	5050

* - Heat Transfer Correlations: ($BTU/HR\text{-}FT^2\text{-}^{o}F$)

1 $U=.10(\Delta t)^{.33}$

2 $U=.18(\Delta t)^{.33}$

3 $U=.22(\Delta t)^{.33}$

4 $U=.22(.27\Delta t)^{.33}$ Hot leg/ambient and stm. gen./ambient Δt estimated to be .27 of core/ambient Δt to match Quick Look measured temperatures.

TABLE 6

Conservative Models for Draindown to Bottom of Reactor Vessel Nozzles

1. **Heat Transfer Model**

Component	Heat Transfer Coefficient (BTU/HR-FT°- F)	Surface Area (FT°)
Bottom Head	.34	330
Lower & Intermediate Shells	.72	960

2. **Heat Capacity Model**

Component	Mass (LBM)	Specific Heat (BTU/LBM-°F)	Heat Capacity (BTU/°F)
Reactor Vessel, Head, & Studs	881,200	.115	101,300
Water	162,300	1.00	162,300
		Total	263,600

TABLE 7

Best Estimate Heat Capacity Model for Draindown to Bottom of Reactor Vessel Nozzles With Hot Leg & Steam Generator Areas

Component		Heat Capacity (BTU/°F)
Reactor Vessel, Head, & Studs		92,600
Core Support Assembly		27,600
Plenum Assembly		11,500
Core		19,100
Hot Leg Piping		18,900
Water (Reactor Vessel)		162,300
Steam Generators		131,600
Water (Steam Generators)		46,200
	Total	509,800

TABLE 8

Best Estimate Heat Capacity Model for Draindown to Bottom of Reactor Vessel Nozzles Without Hot Leg & Steam Generator Areas

Component		Heat Capacity (BTU/°F)
Reactor Vessel, Head, & Studs		92,600
Core Support Assembly		27,600
Plenum Assembly		11,500
Core		19,100
Water (Reactor Vessel)		162,300
	Total	313,100

TABLE 9

Best Estimate Heat Transfer Model for Draindown to Bottom of Reactor Vessel Nozzles With Hot Leg & Steam Generator Areas

Component	Heat Transfer Correlation* (Reference)	Surface Area (FT^2)
1) Reactor Vessel		
A. Bottom Head	1	330
B. Lower & Intermediate Shells	2	960
2) Hot Legs		
A. Reactor Vessel to Thermocouple	3	910
B. Candy Cane	3	530
3) Steam Generators	3	5050

TABLE 10

Best Estimate Heat Transfer Model for Draindown to Bottom of Reactor Vessel Nozzles Without Hot Leg & Steam Generator Areas

Component	Heat Transfer Correlation* (Reference)	Surface Area (FT^2)
1) Reactor Vessel		
A. Bottom Head	1	330
B. Lower & Intermediate Shells	2	960

* - Heat Transfer Correlations: ($BTU/HR\text{-}FT^2\text{-}{}^oF$)

1 $U=.10(\Delta t)^{.33}$
2 $U=.18(\Delta t)^{.33}$
3 $U=.22(.27\Delta t)^{.33}$ · Hot leg/ambient and stm. gen./ambient Δt estimated to be .27 of core/ambient Δt to match Quick Look measured temperatures.

Figure 1
Comparison of Measured
and Predicted Quick Look
Heatup – Original Models

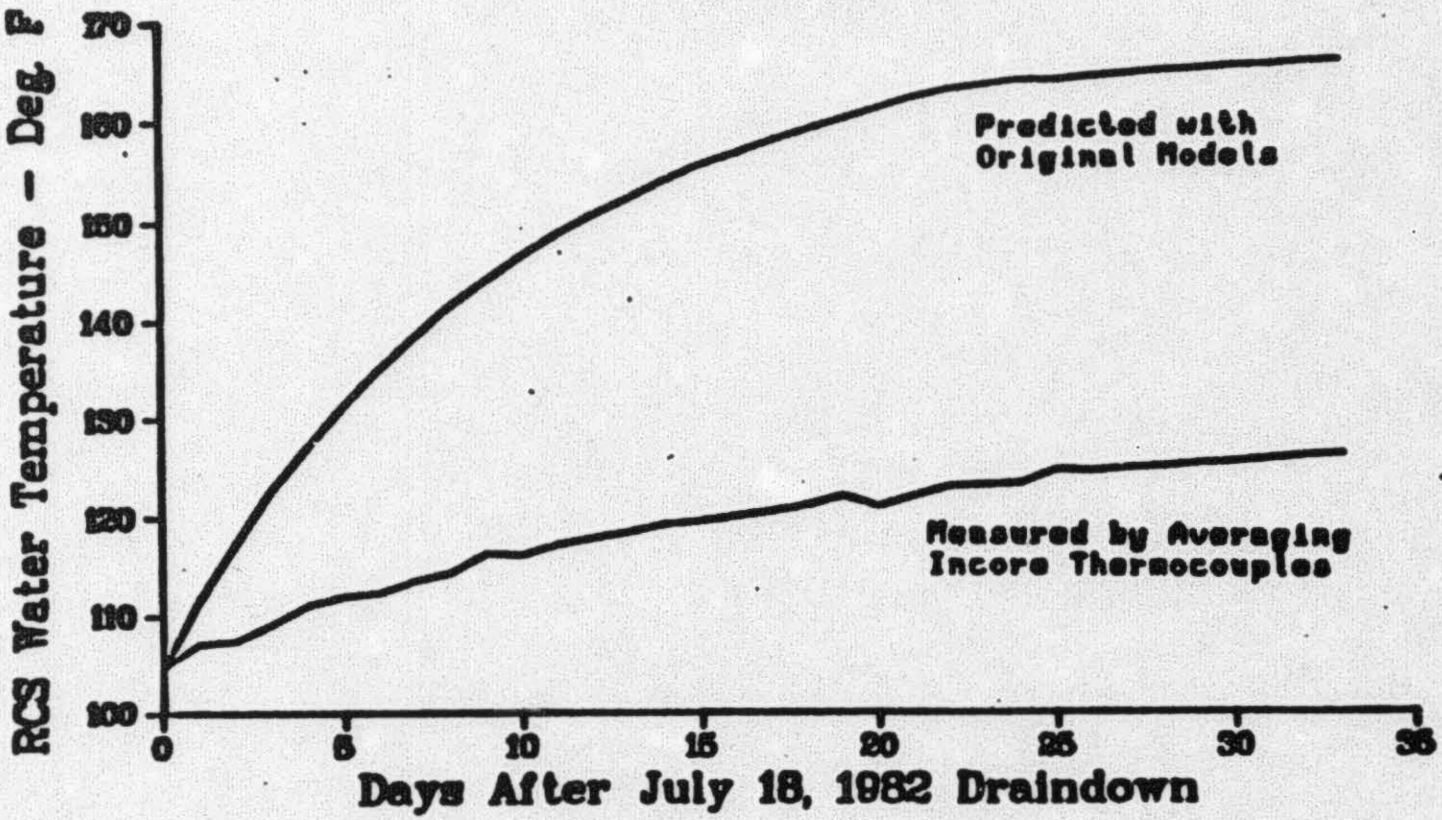


Figure 2
Conservative ANSI
Decay Heat Model
from April 82 Report

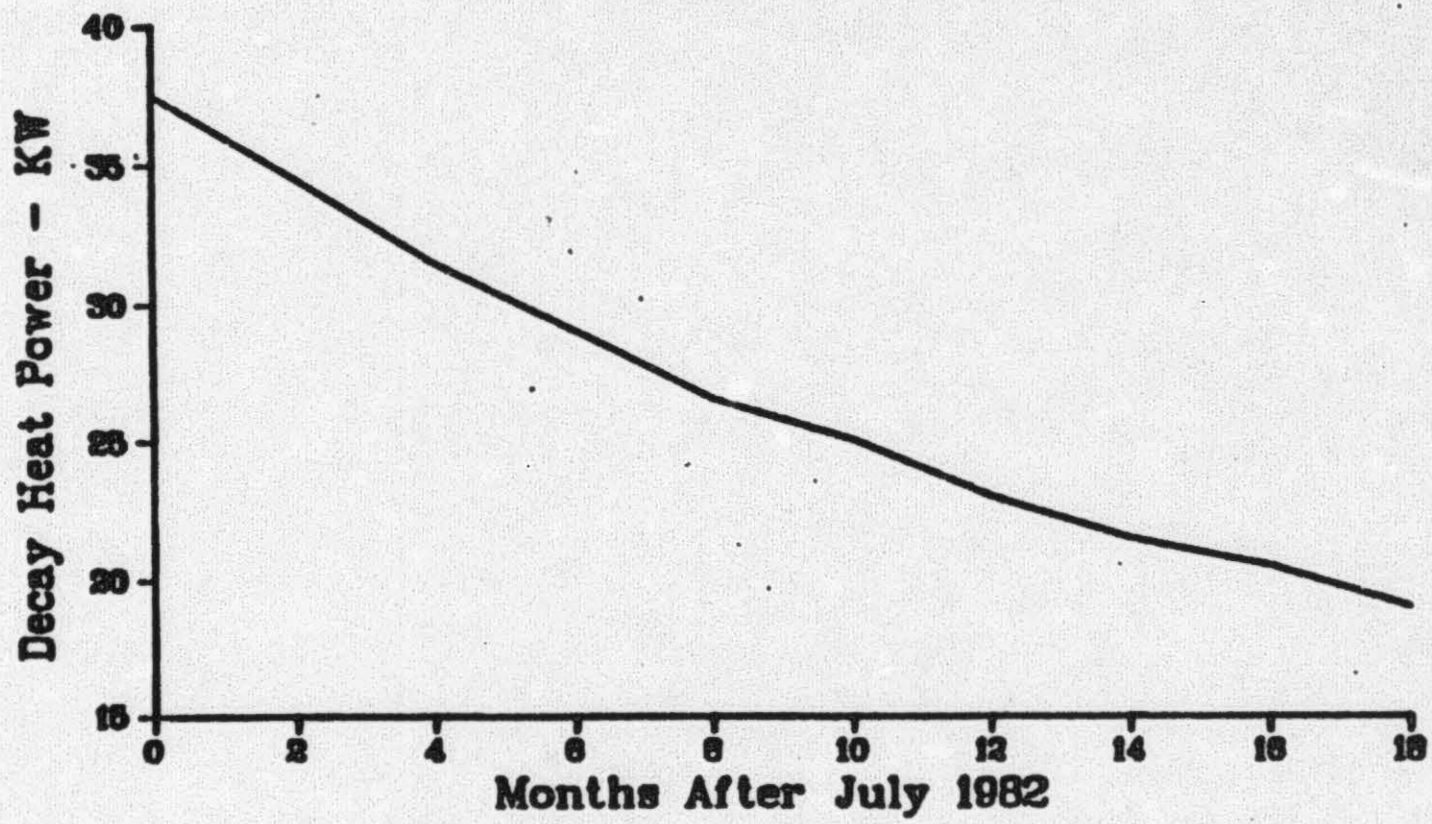


Figure 3
RCS Heatup Rates Following Head Removal Draindown Using Conservative Models

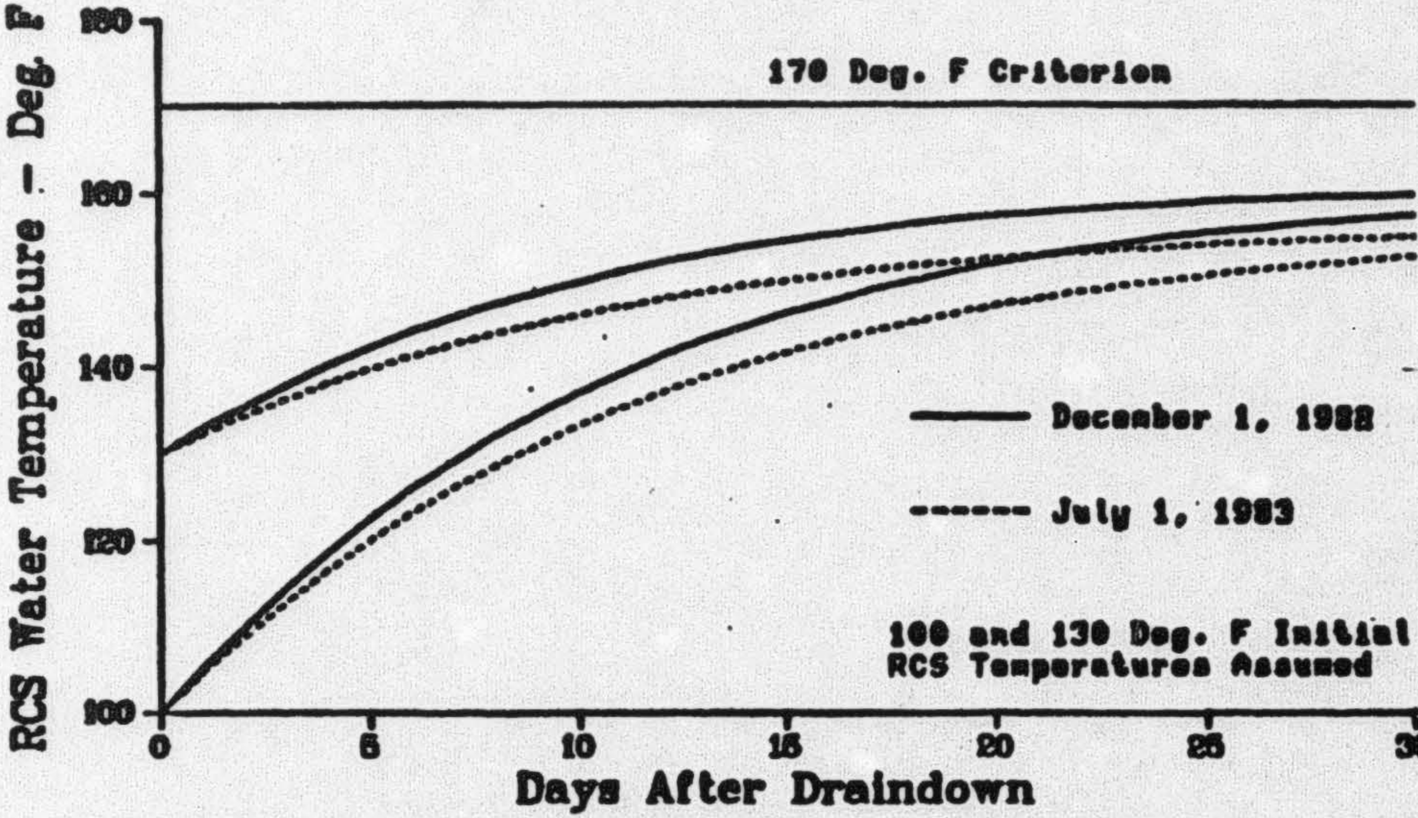


Figure 4
Comparison of ANSI and LOR-2 Based Decay Heat Curves

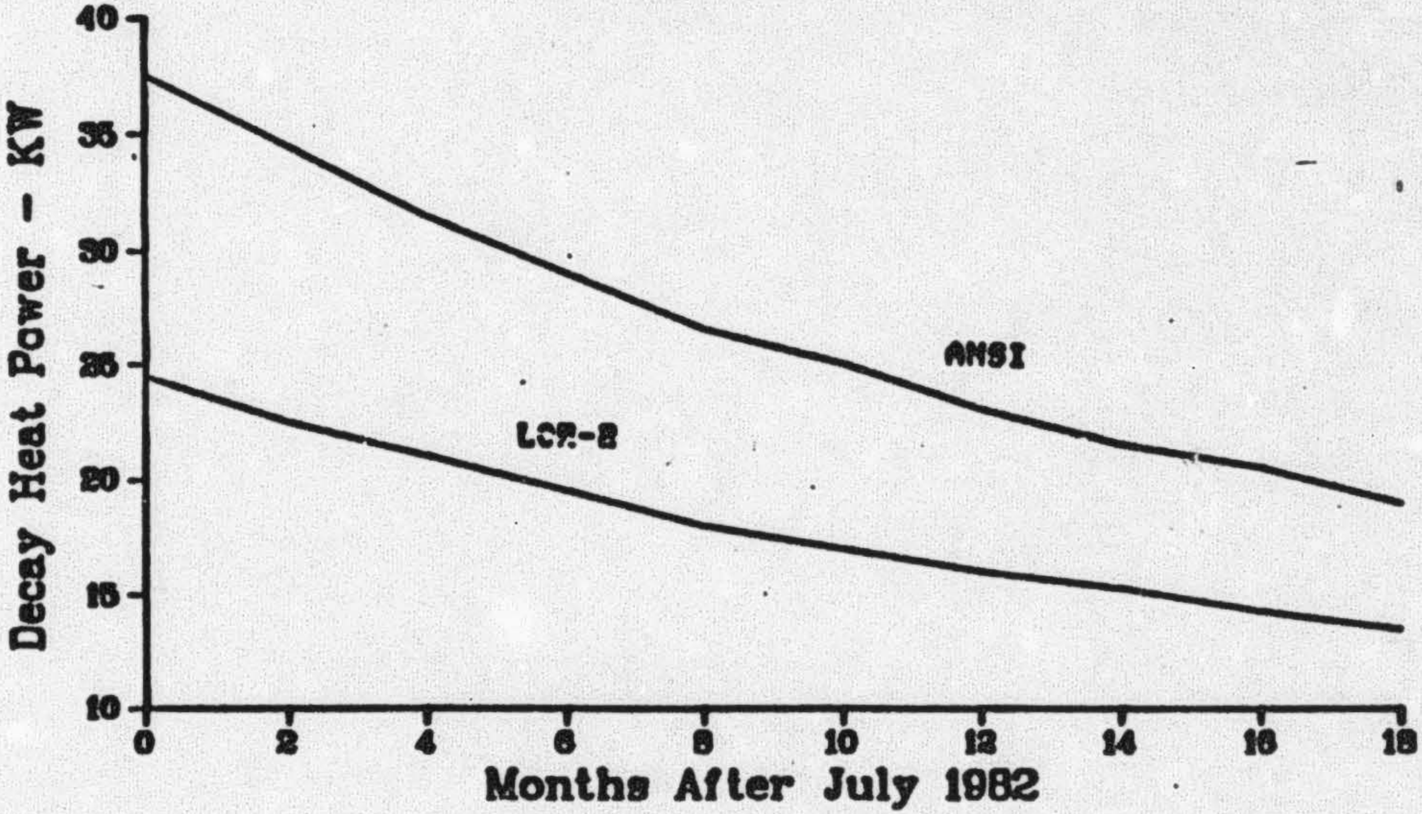


Figure 6
Comparison of Measured and
Predicted Quick Look Heatup
with Best Estimate Models

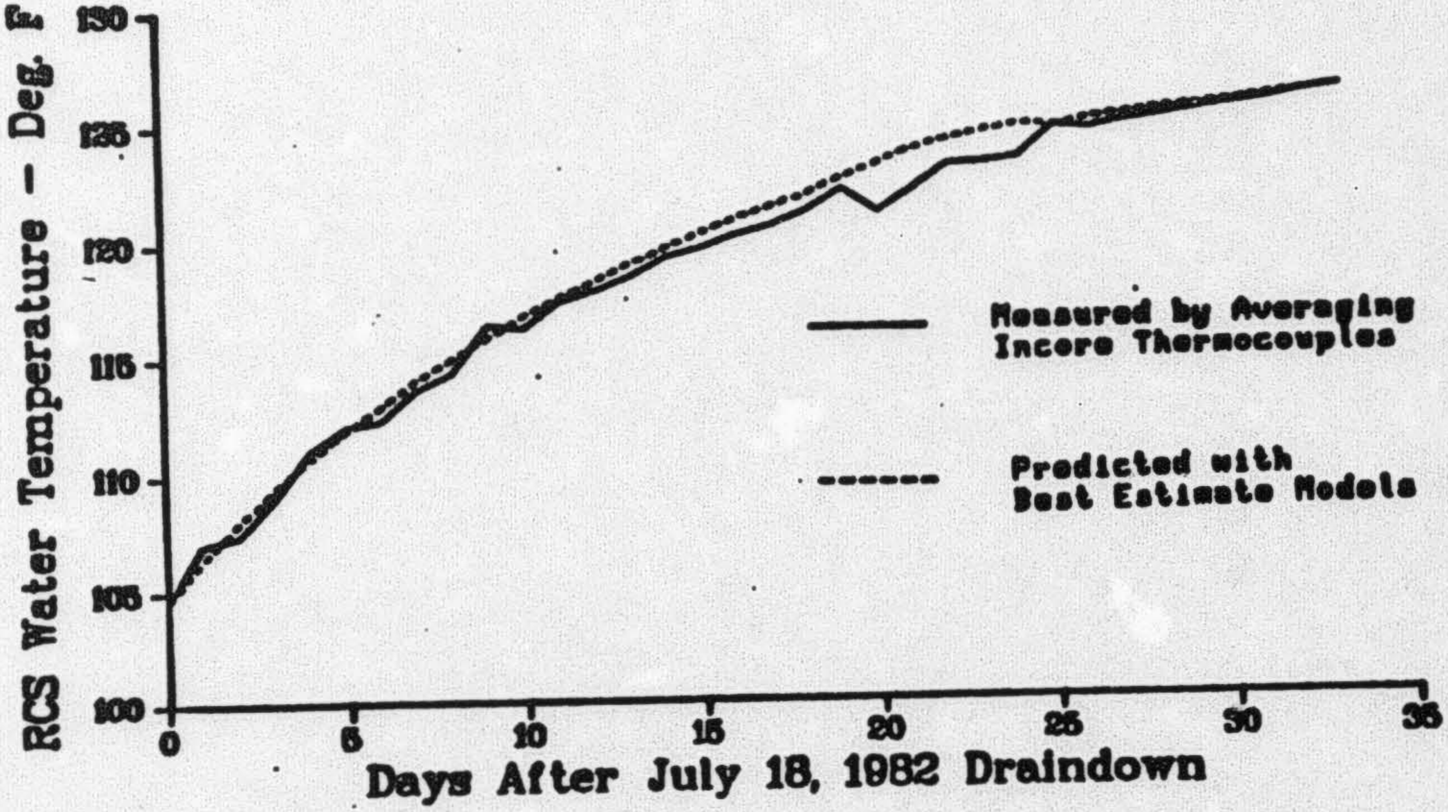


Figure 6
Best Estimate Heatup Rates
from 100 Deg. F Initial RCS Temp
Following Head Removal Draindown

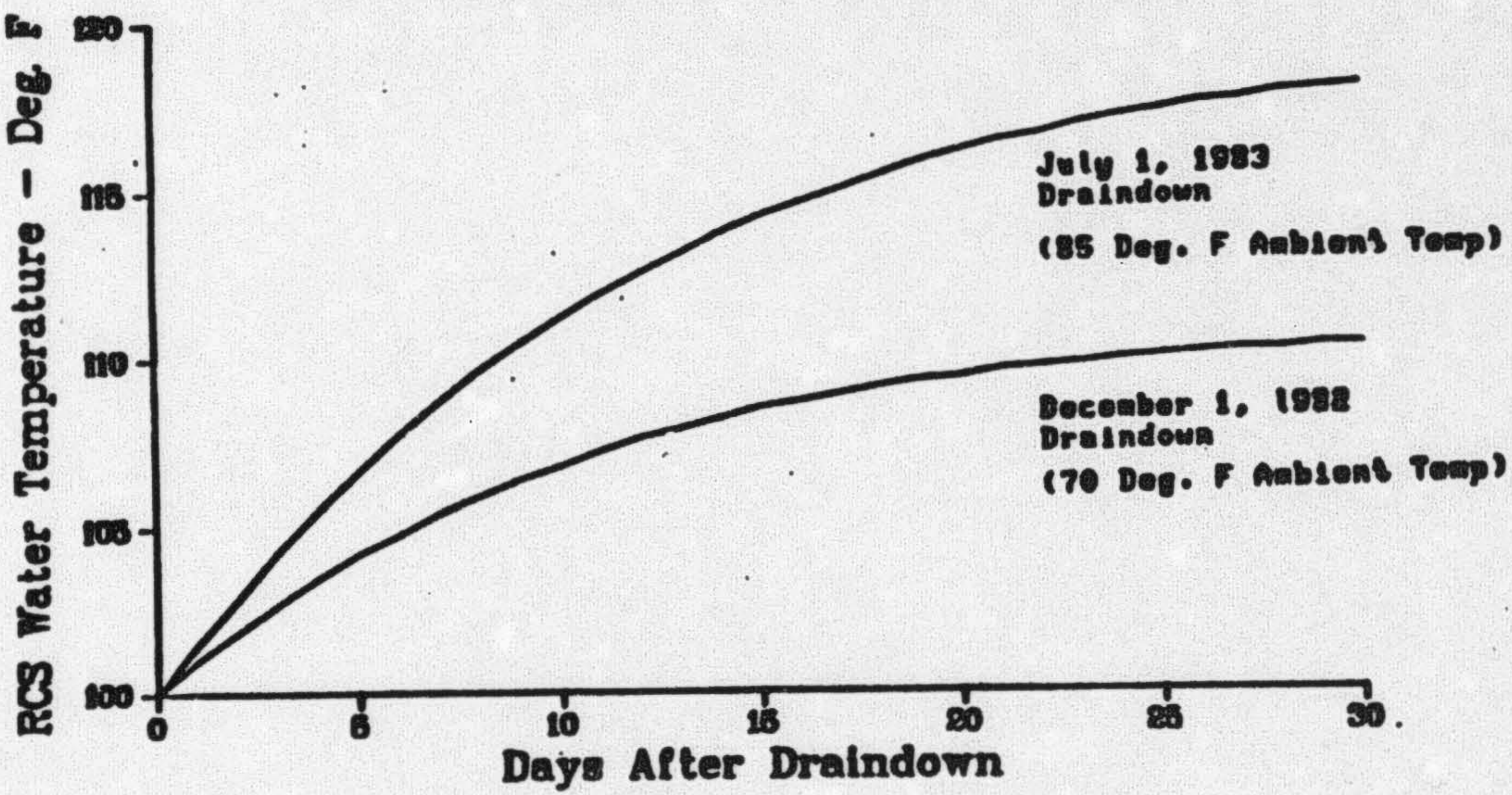


Figure 7
HEATUP FOLLOWING DRAINDOWN TO
BOTTOM OF REACTOR VESSEL NOZZLES
Conservative Model

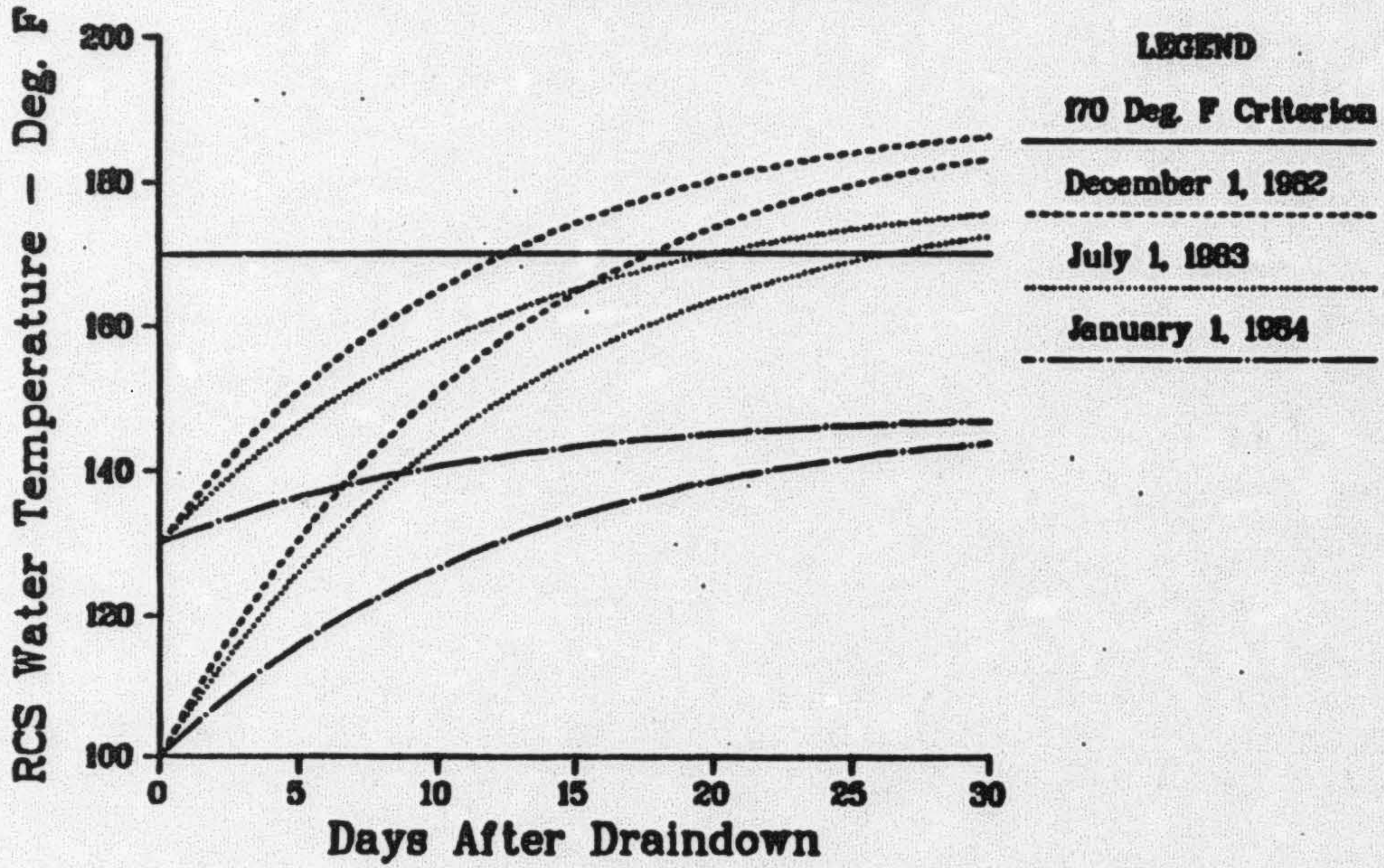


Figure 8
HEATUP FOLLOWING DRAINDOWN TO BOTTOM OF REACTOR VESSEL NOZZLES
Best Est. with Hot Leg and Steam Gen. Areas

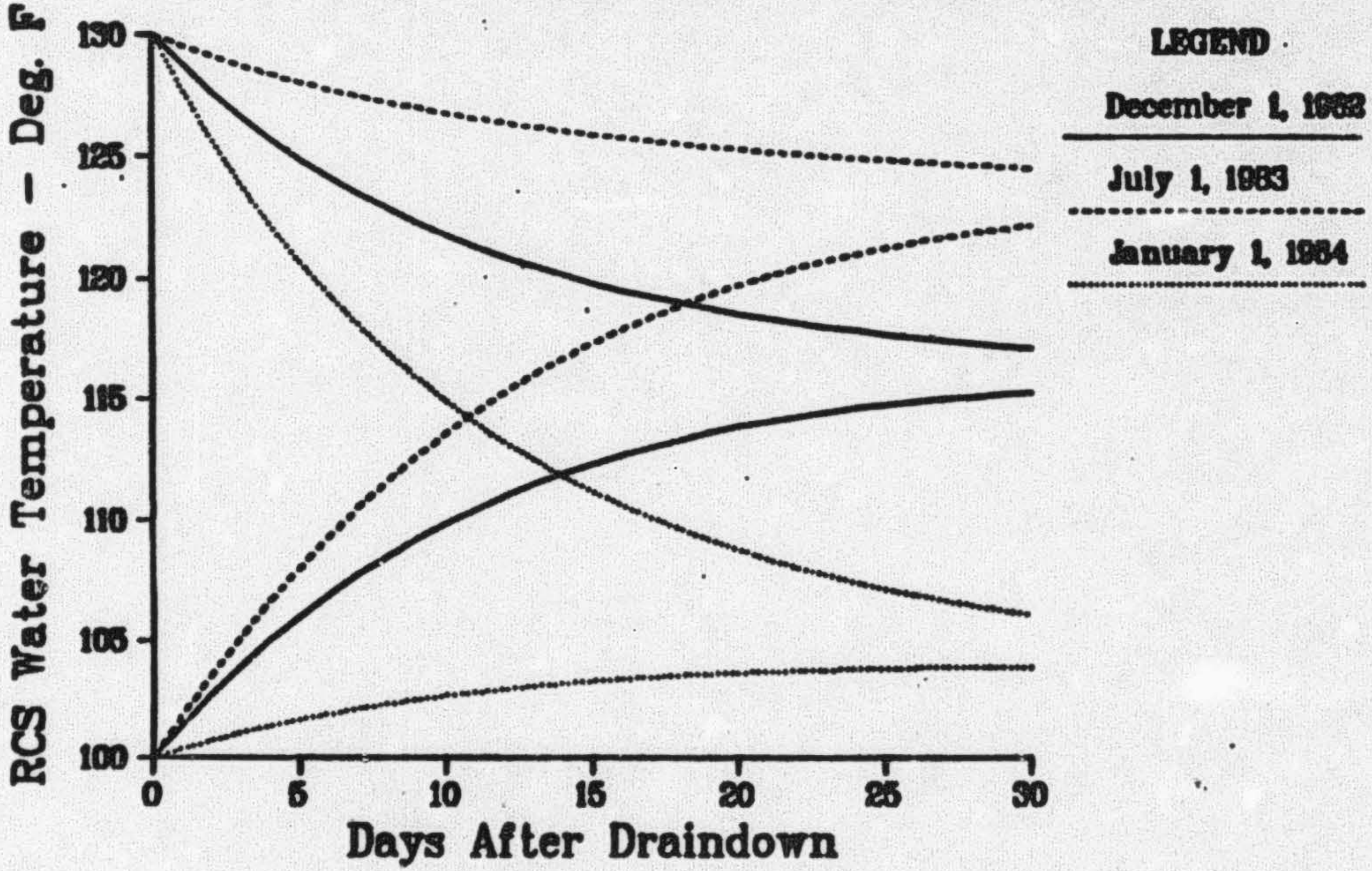
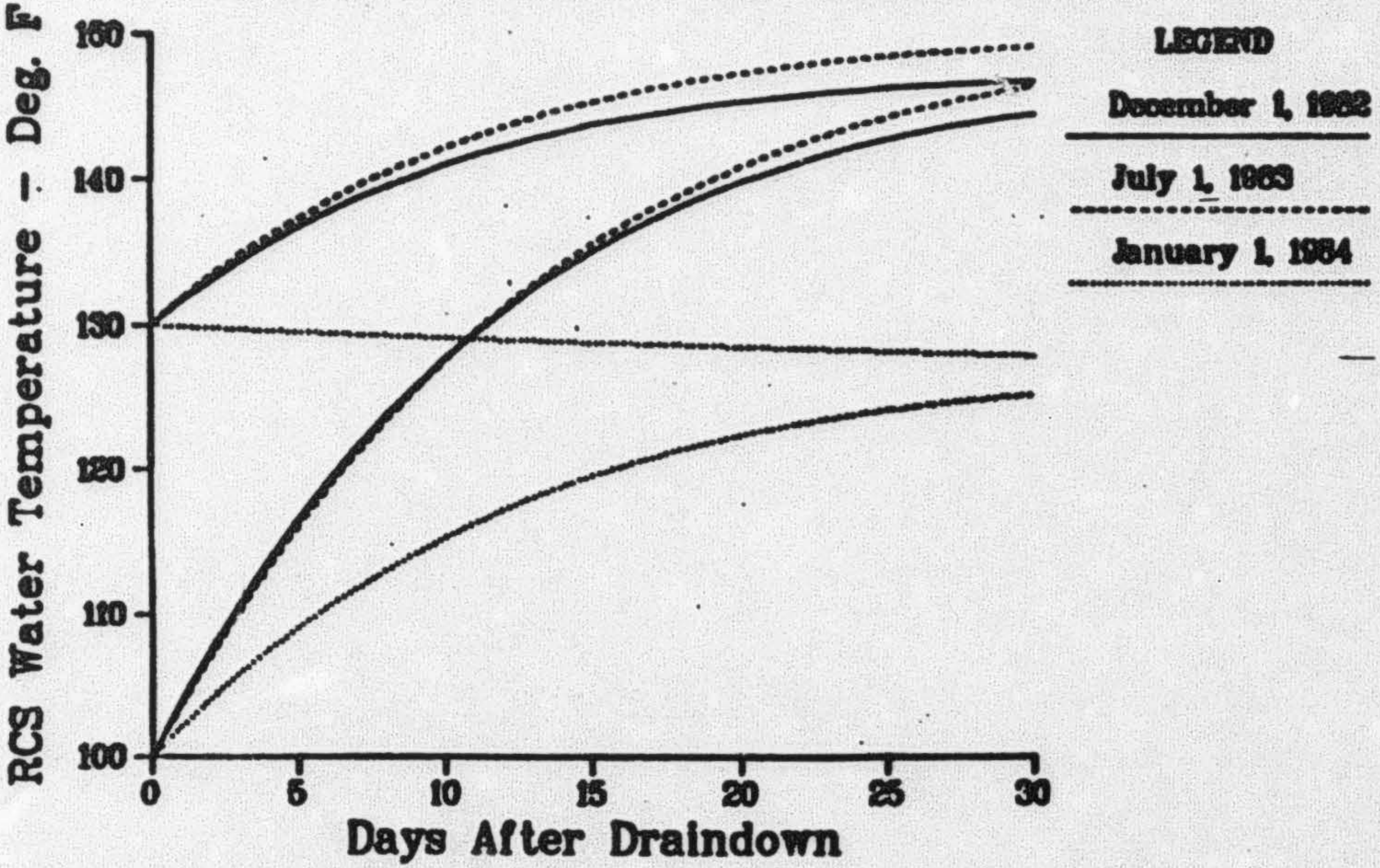


Figure 9

HEATUP FOLLOWING DRAINDOWN TO BOTTOM OF REACTOR VESSEL NOZZLES

Best Est. without Hot Leg and Steam Gen. Areas



Summary

1. The tensile test data of the subject wires tested compared very similarly to that of the new unused wire that was tested. The significance of this is that the tensile strength for these wires appears to have been unaffected by service conditions.
2. SEM analysis shows that the new and used wires are similar except for the presence of surface residue (persumably from in-service and decontamination conditions).
3. The EDAX analysis shows that the surface residue contains chlorine, potassium and silicon.
4. The microstructure examinations show no evidence of hydrogen damage. Both new and used wire show an identical microstructure - cold worked high strength steel normal for the application. No evidence of corrosion was observed in any of the samples; hence, the presence of chlorine has not caused significant corrosion to date.
5. Microhardness data for the used and the control sample were similar. This indicates similar hardness, expected strength and lack of decarburized surface that would weaken the wire.

As a general assessment of the above data,the subject wires and the new control sample appear to have equivalent properties in terms of lift considerations.