Department of Energy
Washington, D.C. 20545
Docket No. 50-537
HQ:S:82:149

DEC 20 1982

Mr. Paul S. Check, Director
CRBR Program Office
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Check:

MEETING SUMMARY FOR THE SHUTDOWN HEAT REMOVAL MEETING, DECEMBER 16, 1982

The purpose of this letter is to summarize the resolution of items discussed between the Nuclear Regulatory Commission and the Clinch River Breeder Reactor Plant project on December 16, 1982.

Enclosure 1 is the agenda used for the meeting, Enclosure 2 is the agreements and commitments from the meeting with item resolution status, and Enclosure 3 is the list of attendees.

Any questions regarding the information provided or further activities can be addressed to A. Meller (FTS 626-6355) or D. Florek (FTS 626-6188) of the Project Office Oak Ridge staff.

Sincerely,

John R. Longenecker

John R. Longenecker
Acting Director, Office of
Breeder Demonstration Projects
Office of Nuclear Energy

3 Enclosures

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Enclosure 1

BRIEFING ON
CRBRP SHUTDOWN HEAT REMOVAL CONCERNS

FOR THE
NUCLEAR REGULATORY COMMISSION
CRBRP PROGRAM OFFICE
BETHESDA, MARYLAND
DECEMBER 16, 1982

AGENDA

DIRECT HEAT REMOVAL SERVICE .... R. E. HOTTEL
- SPECIFIC CHANGES TO INCORPORATE SINGLE FAILURE CAPABILITY
- RESULTS OF LATEST SINGLE FAILURE ANALYSIS
- DETAILS OF BASE-CASE F-2 ANALYSIS

NATURAL CIRCULATION .... R. R. LOWRIE/R. L. MARKLEY/R. E. HOTTEL
- RESULTS OF LONG TERM NC ANALYSIS
- UNCERTAINTY FACTOR APPLICATION TO NC (4.4-9)(4.4-7)
- NATURAL CIRCULATION TESTING (5.3-19)(4.4-1)

DISCUSSION OF ACRS QUESTIONS .... R. E. HOTTEL
- DHRS OVERFLOW LINE
- PWST CAPABILITY
- CONDENSATE STORAGE TANK DETAILS (5.3-17)
- PRIMARY CHECK VALVE

DISCUSSION OF ADDITIONAL OPEN ITEMS .... R. E. HOTTEL
- CLARIFICATION OF AFWS/SWRPRS (5.3-13 & 5.5-1)
- PACC ANALYSIS (5.3-14)
- AFWS RELIABILITY EVALUATION (5.3-18)
- LOSS OF BULK AC POWER (15.1-7)
- PIPE BREAK ANALYSIS (15.3-2)

DISCUSSION OF REMAINING 760 SERIES QUESTIONS .... BROOKHAVEN NATIONAL LAB

SUMMARY .... R. E. HOTTEL

3152 2

Enclosure 2

Shutdown Heat Removal Meeting
December 16, 1982
Agreements and Commitments

1. Direct Heat Removal Service

The specific changes to incorporate single failure capability, the results of latest single failure analysis, and the details of the base case F-2 analysis were discussed. (see attached)

The peak temperatures for the single failure assessment slightly exceed (by 17°F) the structural peak temperature analyzed for the F-2 transient. This is judged acceptable based on a scoping analysis since the capability of the primary boundary to maintain structural integrity, based on creep rupture considerations, at temperatures of approximately 1200°F is approximately 100 hours.

Action: The updated (performance) design case and the single failure evaluation case will be placed in the PSAR in addition to the description currently in the PSAR concerning the structural design case.

This item is resolved for the SER.

2. Natural Circulation Long Term Analyses

The results of long term natural circulation analysis were presented (see attached).

This item is considered resolved for the SER.

Uncertainity Factor Application to Natural Circulation - The uncertainty factors applied to natural circulation were discussed (see attached). This item is considered resolved for the SER.

Natural Circulation Testing - The project commits to conduct a whole plant test program of CRBR natural circulation and DHRS performance during plant startup testing. The objective of the proposed test program is to confirm the thermal-hydraulic computer codes which have been used to predict both the plant natural circulation behavior as well as the plant thermal-hydraulic behavior under DHRS flow conditions.

The natural circulation and DHRS test program will provide information to:

1. Ensure adequate prediction of nat'l cir. capability (transition as well as steady state flow and

temperatures) and DHRS Decay Heat Removal capability.

2. Ensure adequate prediction of the effect on temp. and flow caused by variations in the heat sinks available (e.g. venting, PACC's, 2-HTS loops).

   This item is resolved for the SER.

3. Dicussion of ACRS Questions

   The DHRS overflow line, PWST capability, condensate storage tank details, and primary check valve were discussed (see attached).

   The information in the PSAR and meeting discussion was adequate, this item is considered resolved.

4. Discussion of Additional Open Items

   Clarification of AFWS/SWRPRS - The discussion of the operation of the isolation valves under AFWS and SWRPRS initiation resolved this item (see attached).

   PACC Analysis - The meeting discussion and recent PSAR amendment resolved this item (see attached). NRC is reviewing the PSAR amendment.

   AFWS Reliability Evaluation - The meeting discussion and PSAR information resolved this item (see attached).

   Loss of Bulk AC Power - The meeting discussion resolved this item (see attached).

   Pipe Break Analysis (see attached)

   Action: The DEMO code output and assumptions used for hot leg pipe break analysis will be provided to NRC.

1-1

5.6.2 Direct Heat Removal Service (DHRS)

5.6.2.1 Design Bases

5.6.2.1.1 Performance Objectives

41 The Reliability Program, discussed in Appendix C, will provide verification that SGAHRS removes residual heat following a reactor shutdown with a high level of reliability. Hence it is judged that only steam and feedwater trains backed by SGAHRS are necessary to safely and reliably remove residual heat following shutdown from three loop, full power operation. To enhance the reliability of decay heat removal, the DHRS provides a fourth redundant heat removal path and heat sink. The impact on overall shutdown heat removal reliability by inclusion of the
41 DHRS is being determined by the Reliability Program described in Appendix C. The DHRS provides this supplementary capability by satisfying the following objectives:

a. Function to remove reactor decay heat following reactor shutdown from three loop rated power operation, assuming loss of all heat transfer through the IHX's at the time of reactor trip. Operation of three primary pump pony motors and maximum reactor decay power is assumed. Under these conditions, the DHRS is designed to provide sufficient cooling to ensure primary coolant boundary integrity and prevent loss of in-place coolable geometry of the core.

   To meet this objective, DHRS components will be sized such that, under these conditions, the average bulk primary sodium temperature will be limited to approximately 1140°F. Capability will be provided to permit remote manual initiation of DHRS from the Control Room. The overflow and makeup circuit and the spent fuel cooling system will be able to be cross connected in a manner which permits both EVST NaK cooling trains to be used when DHRS is removing full capacity heat load.

b. Accommodate the thermal transients resulting from normal, upset, emergency and faulted plant events in which continued performance of its function is not impaired.

c. Accommodate floods (Section 3.4), tornadoes (Section 3.3), missiles (Section 3.5), and earthquakes (Section 3.7), in which continued performance of its function is not impaired.

d. Function in a manner which will not significantly reduce the reliability and availability of the EVST heat removal chain. This objective requires the EVST NaK cooling circuits to be designed to remove concurrently the heat generated by the spent fuel and the reactor decay heat.

e. Accommodate the failure of a single active component. To meet this objective, the equipment performing the DHRS function will be redundant to the extent that failure of a single active component will not result in complete loss of DHRS heat removal capability (i.e. adequate heat removal capability to maintain coolable core geometry will remain following failure of a single active component) based on analyses where conservatisms are selectively applied.

TABLE 5.6-10

26 DHRS VALVE CLASSIFICATION

	LOCATION	VALVE NO.**	ACTIVE/INACTIVE	NORMALLY* OPEN/CLOSED	OPERATING MODE
	Makeup Pump Suction	~~1~~ 116	Inactive	Open	Isolation
		~~2~~ 132	Inactive	Open	Isolation
26		~~11, 12~~ 114, 133	Inactive	Closed	Isolation
46 26	Overflow VSL Drain/Sample Return	~~15~~ 153	~~In~~active ~~(1)~~	Open	Isolation
	Makeup Pump Dicharges	~~4~~ 104	~~In~~active	Open	Isolation
26		~~3~~ 118	~~In~~active	Open+	Isolation
		~~5~~ 102	Active	Open+	Isolation
		~~6~~ 109	Active	Open+	Flow Control
		~~7~~ 103, 107	Active	Closed+	Isolation
26	Overflow Ht. Exch. Sodium Out.	~~8~~ 150	Inactive	Open	Isolation
		~~9~~ 151	Inactive	Open	Check
46		~~10~~ 149	Inactive ~~(1)~~	Open	Isolation
	Overflow Ht. Exch. to Airblast (hot leg)	~~31~~	~~Inactive~~	~~Open~~	~~Isolation~~
		~~24, 25~~ 416, 357	Active	Closed+	Isolation
		~~22, 23~~ 359, 420	Active	Open+	Isolation
	Airblast to Overflow Ht. Exch. (cold leg)	~~30~~	~~Inactive~~	~~Open~~	~~Isolation~~
26		~~20, 21~~ 358, 415	Active	Closed+	Isolation

26 * The position indictated is the valve position during normal plant operation. Valves with + are those which are manually opened, or closed, prior to initiation of DHRS operating mode. All active valves are remote -manual operable from the control room.

** Valve numbers are those shown on Figure 5.1-7.

46 ~~(1) Considered "Inactive" for DHRS purposes but "Active" for leak mitigation.~~

~~Amend. 46~~
~~Aug. 1978~~

1-3

TABLE 5.6-10

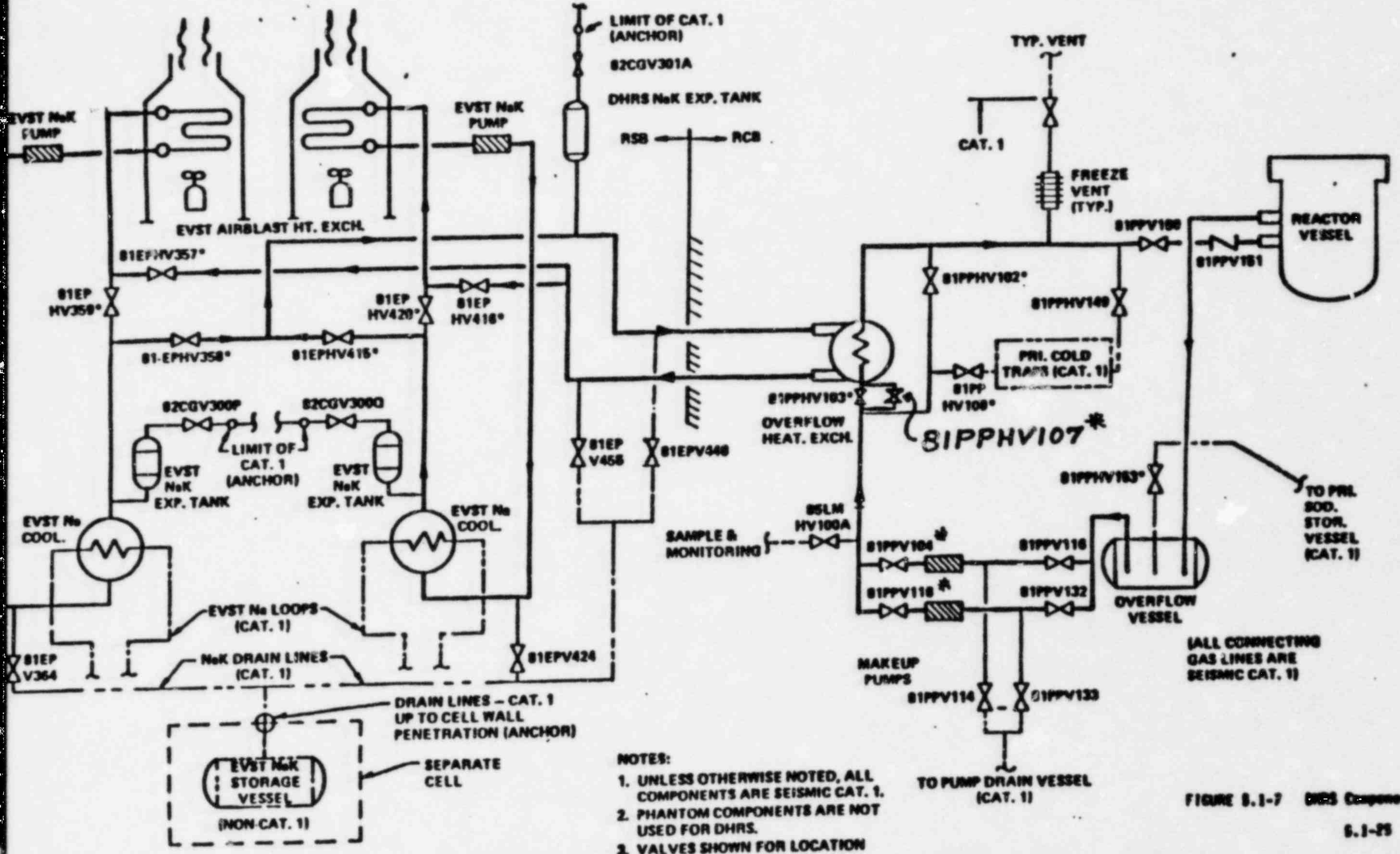
(Continued)
Page 2

DHRS VALVE CLASSIFICATION

LOCATION	VALVE NO.**	ACTIVE/INACTIVE	NORMALLY* OPEN/CLOSED	OPERATING MODE
NaK Drain Lines	364, 424, 446, ~~26, 27, 28,~~ ~~29~~ 455	Inactive	Closed	Isolation
EVST NaK Exp. Tank	~~32, 33, 37~~ 300P, 300Q	Inactive	Open	Isolation
Cover Gas	~~36~~ 301A	~~Active~~ Inactive	Closed+	Isolation
Sodium Vent (Typ.)	~~34~~	Inactive	Closed	Isolation
Sodium Sample	~~35~~ 100A	Inactive	Open	Isolation

* The position indicated is the valve position during normal plan operation. Valves with + are those which are manually opened or closed prior to initiation of DHRS operating mode. All active valves are remote-manual operable from the control room.

** Value numbers are those shown on Figure 5.1-7.



NOTES:

1. UNLESS OTHERWISE NOTED, ALL COMPONENTS ARE SEISMIC CAT. 1.
2. PHANTOM COMPONENTS ARE NOT USED FOR DHRS.
3. VALVES SHOWN FOR LOCATION ONLY.
4. VALVE CLASSIFICATION – SEE TABLE 5.6-10.

*ACTIVE

FIGURE 5.1-7 DHRS Components

5.1-25

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5.6.2.1.2 Applicable Code Criteria and Cases

The components of the DHRS shall be designed, fabricated, erected, constructed, tested and inspected to the standards of Section III of the ASME Code, 1974 edition through the summer 1974 Addenda, Class 1 or 2~~, as~~
26 ~~listed in Table 5.6-12.~~

Applicable code cases will be used to supplement the design analysis required by the ASME Code.

5.6.2.1.3 Surveillance Requirements

The need for surveillance of the DHRS piping and components will
26 be determined as the system design progresses and as the need to monitor austenitic stainless steel is determined by ongoing programs. If a requirement is identified, a surveillance program will be designed in accordance with the philosophy of 10 CFR 50, Appendix H.

5.6.2.1.4 Material Considerations

High Temperature Design Criteria

High temperature components in the DHRS will be analyzed in
26 accordance with the requirements specified in ASME Boiler and Pressure Vessel Code, Section III, as supplemented by the applicable code cases and RDT standards.

Material Specifications

Stainless steel materials which satisfy the requirements of the ASME Code will be specified for use in the DHRS system, as noted in Table
26 5.6-13.

49 5.6.2.1.5 Leak Detection Requirements

The DHRS will be monitored for sodium and NaK leaks and leak
26 indication will be provided in the Control Room by the leak detection system described in Section 7.5.5.

5.6.2.1.6 Instrumentation Requirements

DHRS is remote manually activated and controlled from the Control Room. Instrumentation required to monitored the condition of the DHRS consists of thermocouples on the EVST sodium outlet lines (3 loops) and level indicators in the EVST and the Reactor Vessel (RV). These instruments confirm that the sodium levels in the RV remain above the loop outlet nozzles
50 46 and that temperatures remain below design limits. Other DHRS diagnostic instrumentation is not essential for DHRS operation as the pumps and air blast heat exchanger are being operated at maximum design rates. When the reactor decay heat load has dropped sufficiently, the cooling capacity of the system may manually be reduced by lowering flow rates or fan speed, or by shutting down one one of the EVST cooling trains.

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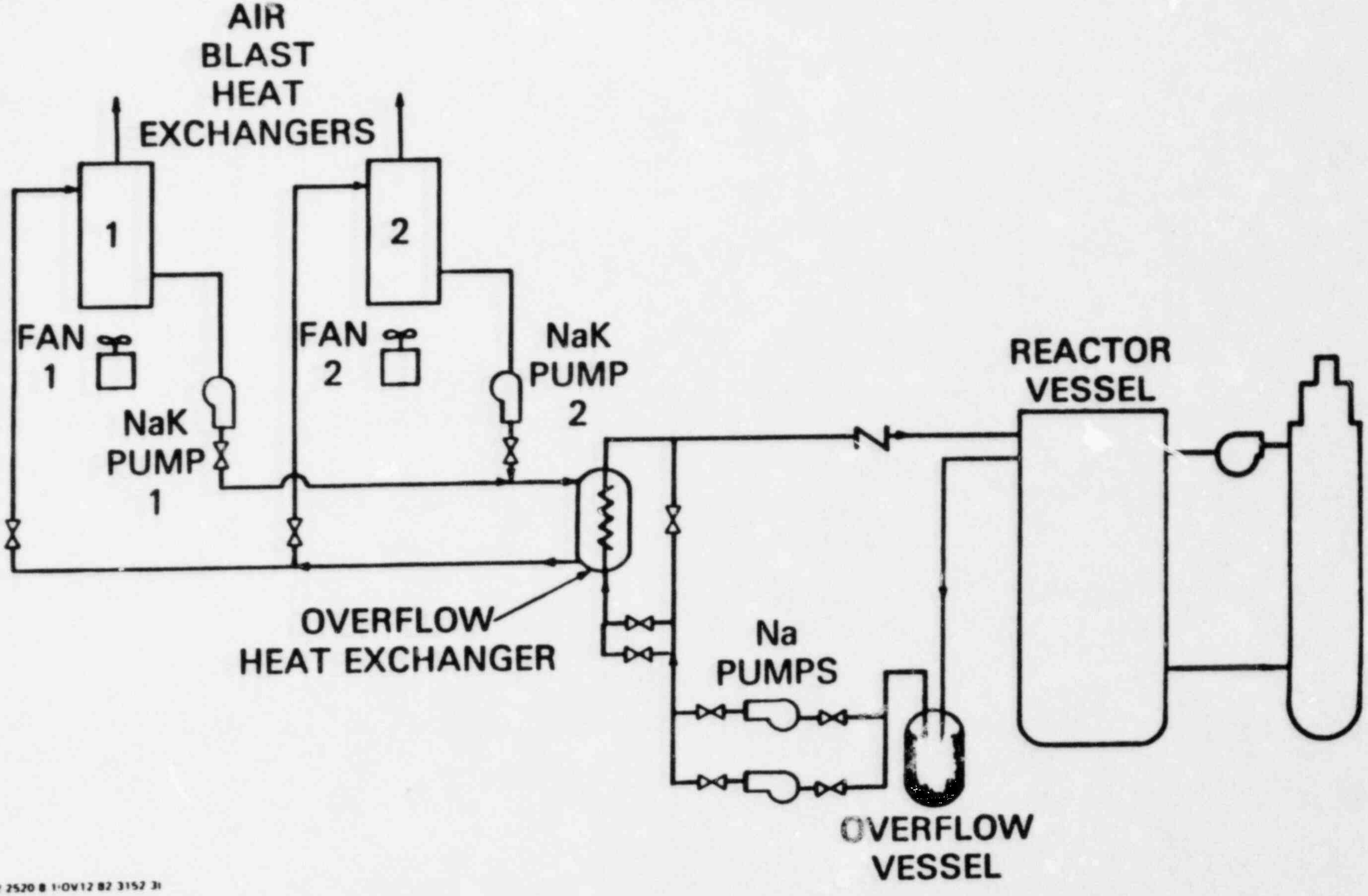
A REDUNDANT VALVE AT THE OHX INLET PROVIDES THE CAPABILITY TO WITHSTAND A SINGLE ACTIVE FAILURE WITHIN THE DHRS



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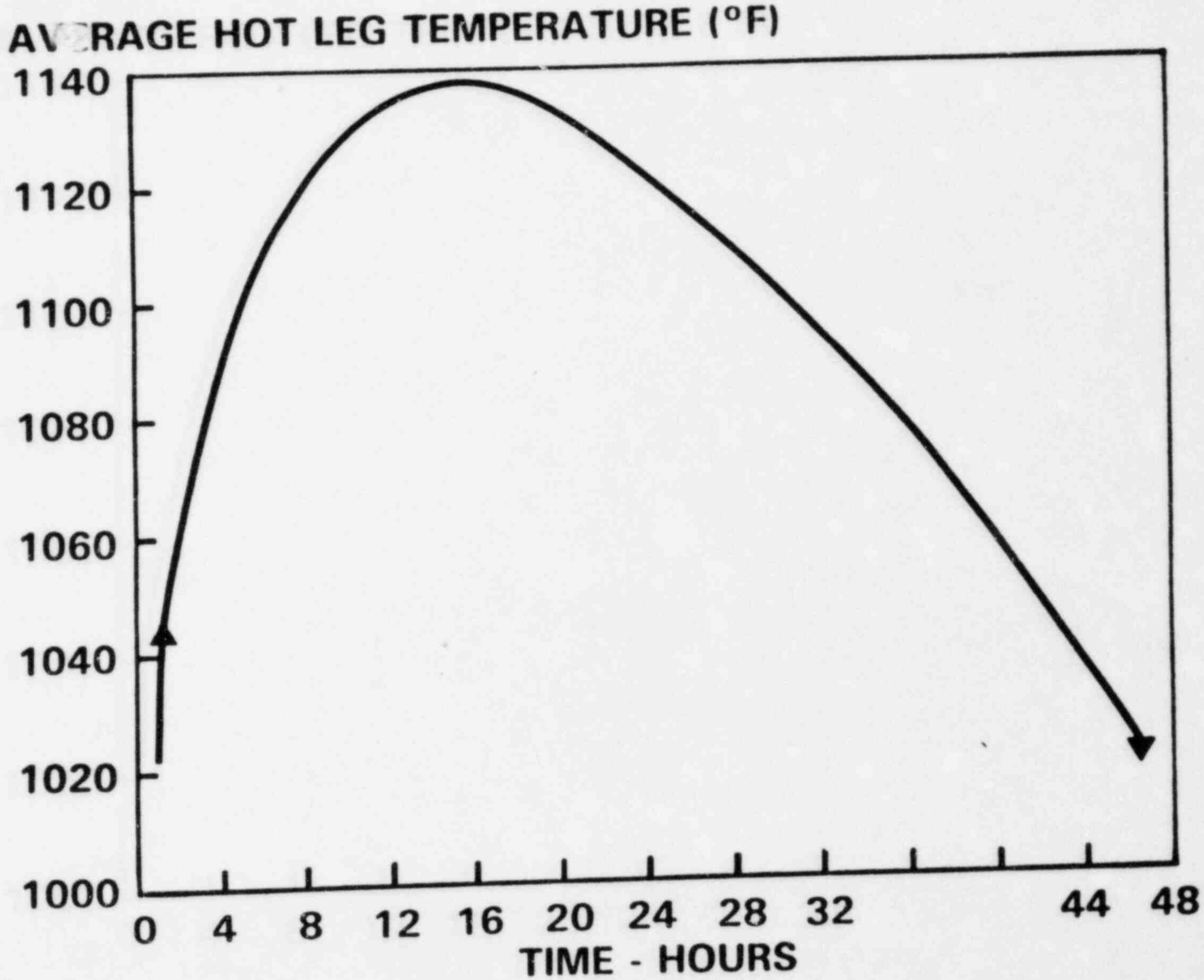
THE INPUTS FOR THE UPDATED DHRS SINGLE FAILURE ANALYSIS INCLUDE:

- TWO PHTS LOOPS OPERATING
- NO IHTS HEAT CAPACITY
- NO SGS HEAT CAPACITY
- ONE Na MAKEUP PUMP AT 600 GPM
- ONE NaK PUMP AT 600 GPM
- ONE AIR BLAST HEAT EXCHANGER
- CONSERVATIVE DECAY HEAT USED
- NO EVST HEAT LOAD
- HEAT LOSSES THROUGH INSULATION INCLUDED

UPDATED DHRS SINGLE FAILURE ANALYSIS



Long Term Natural Circulation Event
(Two Hour Station Blackout)

The staff requested natural circulation analyses extending beyond the 600 second transient time cases previously submitted by the applicant. The objective of the extended analyses was to demonstrate continuing stable natural sodium circulation and acceptable transition in heat sink configuration from steam venting to PACC's only operation.

The following series of attachments provides the long term results of natural circulation. The curves demonstrate that long term natural circulation provides adequate core cooling and continually removes decay heat from the core. The analytical assumptions and explanation of these curves follows:

The CRBRP-ARD-0308 analysis is based on maximum decay heats and conservative heat sink capability. Therefore it is appropriate for use in evaluating peak core temperature as well as the transition from forced circulation to natural circulation.

The progress of any long term natural circulation decay heat removal event is controlled by the heat balance at the water-sodium interface (steam generator system). The heat inputs are the monotonically decreasing reactor decay power and the sensible heat from the sodium systems. The heat sinks include the natural draft PACC's, the turbine driven auxiliary feed pump and the SGAHRS vents.

The long term natural circulation analysis was performed to address the transition from steam venting as the primary heat sink to the ACC's as the only heat sink that is required for plant cooling. This analysis assumed 1) nominal decay heat to assure that the transition to the PACC's as the sole heat sink would occur within the analysis run time and 2) a conservative HTS pump coastdown of 100 sec which is less than both water and sodium testing has demonstrated. The PACC minimum capability is 4/4 Mu/loop.

As shown in the following curves, inherent behavior of the present control system assures safety for well over 2 hours of station blackout. Design optimization by the OL stage will enhance these conclusions. The time scale affords ample opportunity for operator control to improve response and instrumentation will support any required decisions. Three redundant HTS loops further assure safety even if failure occurs in any one loop.

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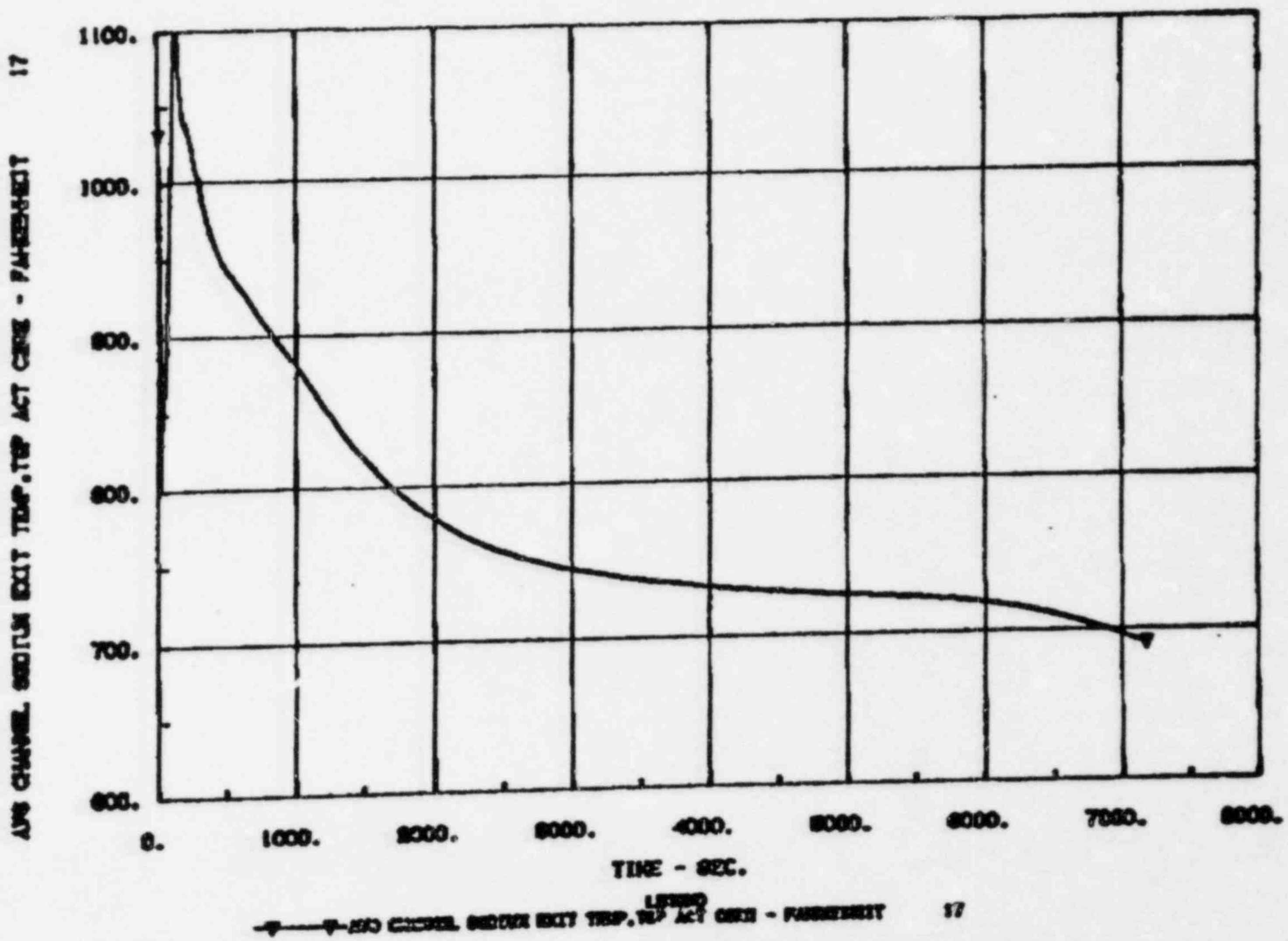
FRAME 04

NAT CIRC LØNG



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FRAME 10

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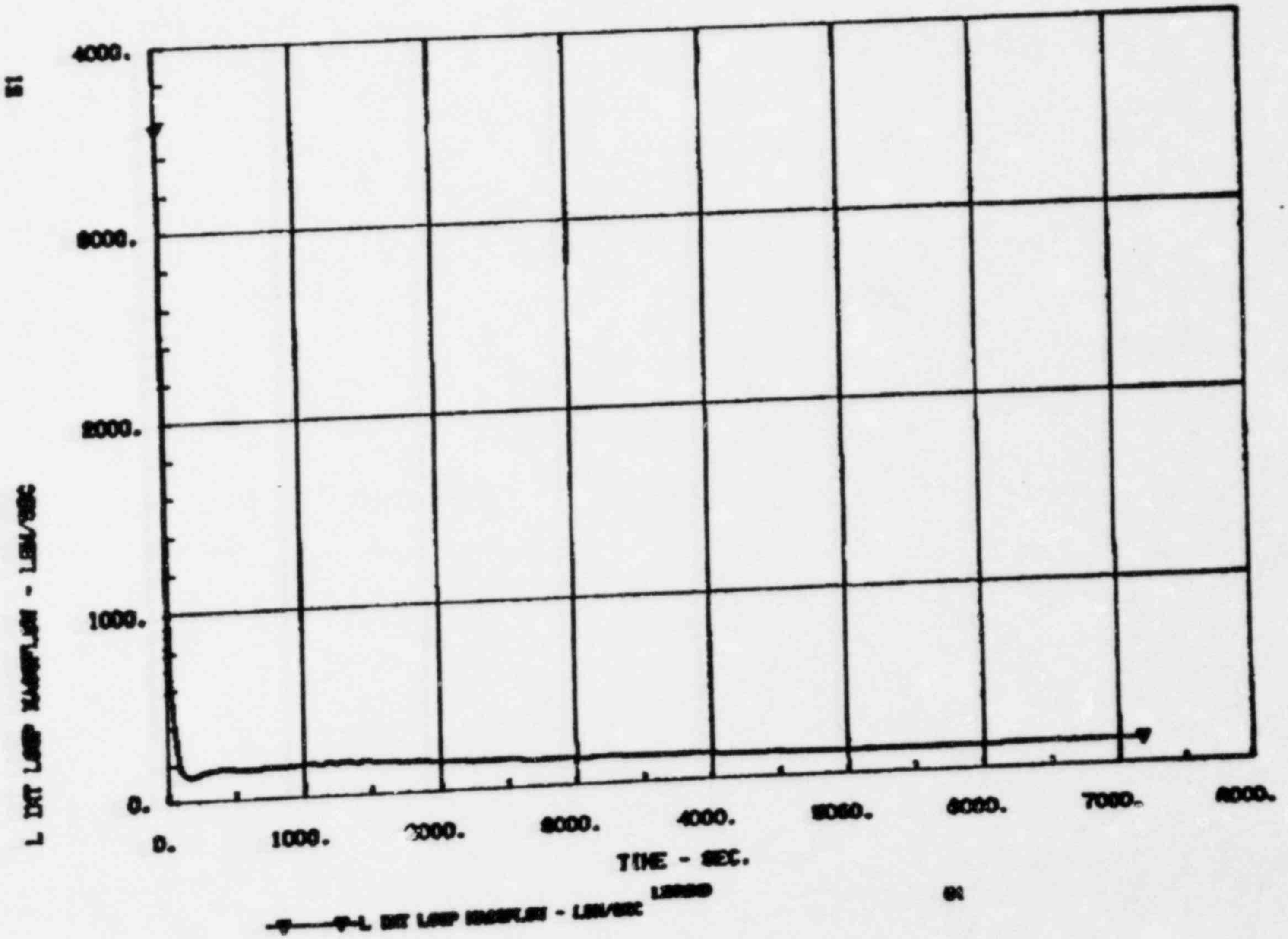

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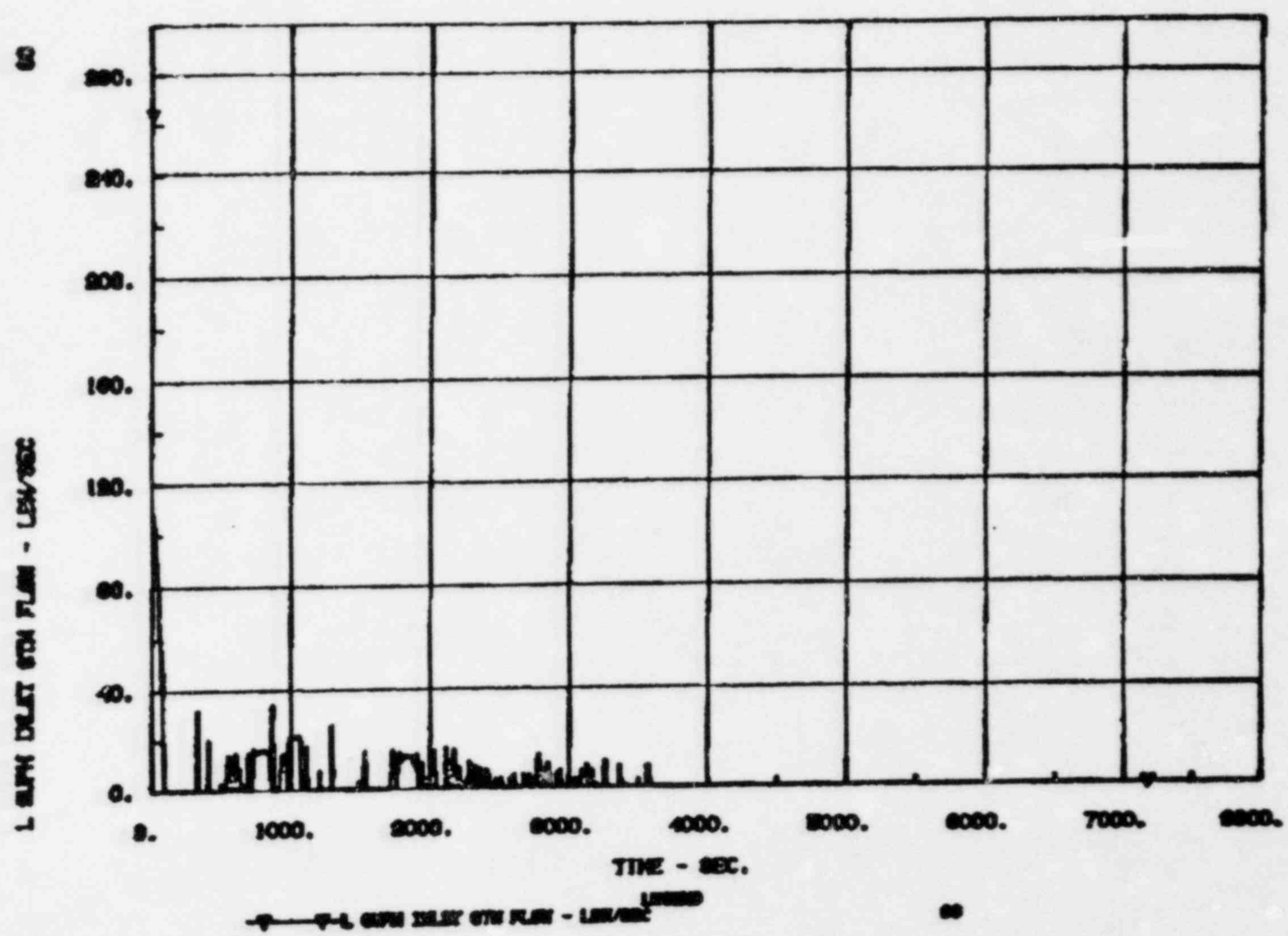
FRAME 24

NAT CIRC LØNG



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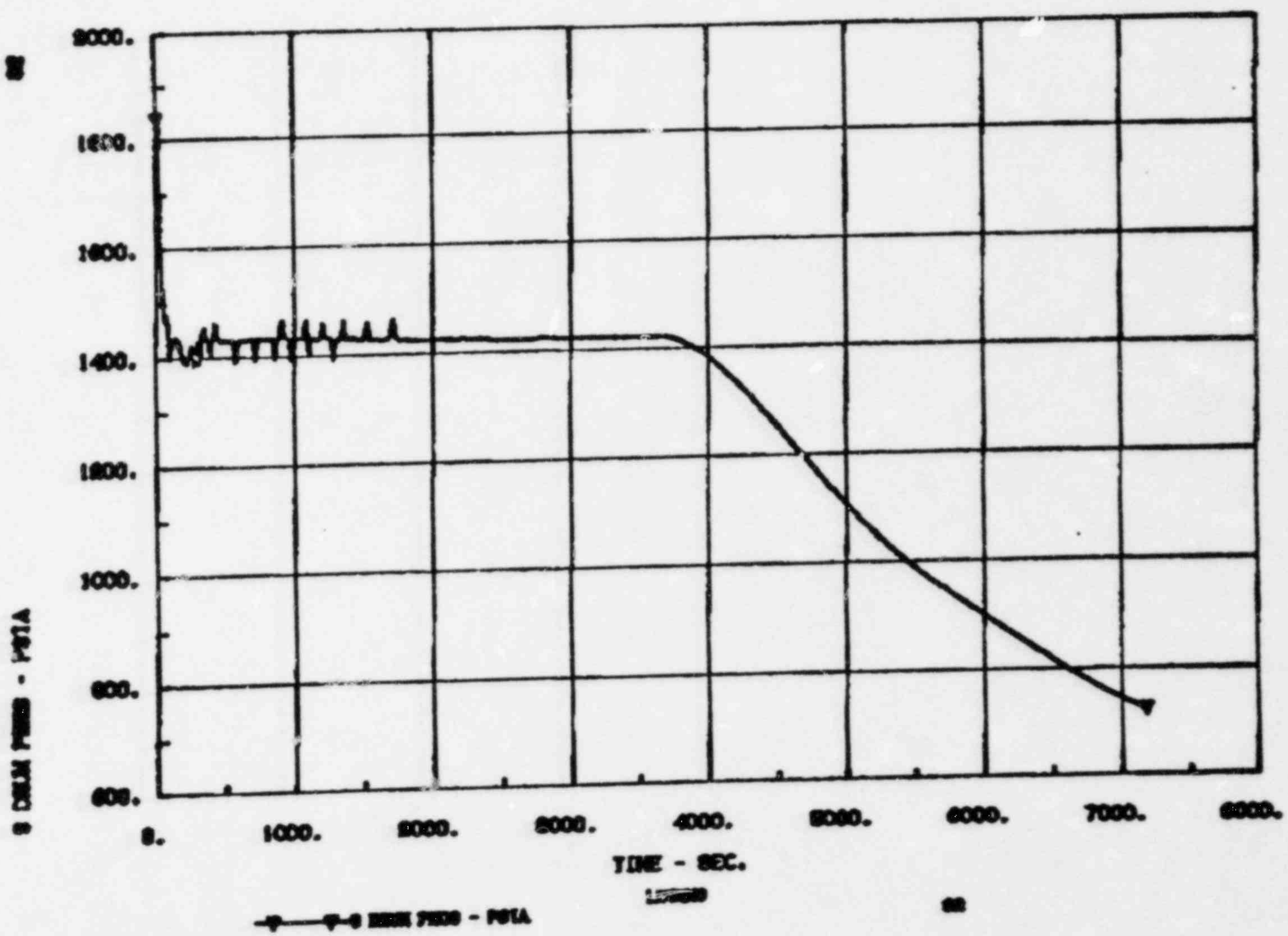
FRAME 25

NAT CIRC LØNG



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COOLANT HCF = POWER UNCERTAINTIES + FLOW UNCERTAINTIES AT LOW FLOW/POWER

I. POWER UNCERTAINTIES

- Full Power Uncertainties at 3σ Plus 3σ Decay Heat Uncertainty

II. FLOW UNCERTAINTIES

- Inlet Flow Maldistribution
  - -- Full Flow Uncertainties Plus 30% Uncertainty on ΔP Applied
- Flow Distribution Calculational Uncertainty
  - -- Full Flow Uncertainties Used Plus Full Power/Flow Peak to Average ΔT Conservatively Used at Low Flows (Intra-Assembly Flow and Heat Redistribution)
- Subchannel Flow Area
  - -- No Change at Low Flow
- Coolant Properties
  - -- No Change at Low Flow

III. Data and Analyses Substantiate Significant Conservatism in Not Taking Credit for Intra- and Inter-Assembly Flow and Heat Redistribution, i.e., Self-Compensating Buoyancy Plus Heat Transfer Effects

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SUMMARY

UNCERTAINTY FACTORS APPLICABLE TO NATURAL CIRCULATION PREDICTIONS

- CRBRP Core Natural Circulation Analyses Very Conservative Using 100% HCFs Plus:
  - -- Not Taking Credit for Inter- and Intra-Assembly Flow and Heat Redistribution
  - -- Using Large Uncertainty for Core ΔP
- Many Conservative, Worst Case Assumptions Used in Analyses - Still Large Margin-to-Boiling
- Conservatism of Approach for Calculating Maximum Core Temperatures Exemplified by Comparisons to Prototypic FFTF Natural Circulation Test Data
- Heterogeneous Core Configuration and Long Term Operation Effects Enveloped by Analyses in CRBRP-ARD-0308
- FSAR Will Include Refined Modeling on Inter- and Intra-Assembly Flow and Heat Redistribution Using a Verified System of Three Computer Codes (DEMO, COBRA-WC and FORE-2M)
- Acceptance Test Phase Experiments Will be Performed for Final Confirmation of CRBRP Natural Circulation Capability

MAJOR ASSUMPTIONS USED IN NATURAL CIRCULATION HOT ROD ANALYSIS

- Conservative Plant THDV +20°F Initial Conditions (e.g., 750°F Reactor Inlet)
- Worst Case Doppler Coefficient Including Uncertainties (±30%)
- Minimum Control Rod Shutdown Worth (One Stuck Rod)
- Conservative Flow Coastdown of Primary Pumps
- 3σ Hot Channel/Spot Factors (ID Temperatures)
- Maximum Core Pressure Drop
- Highest Power and Temperature Hot Rods at Worst Time in Life
- Worst End of Uncertainty Range Used for Fuel Properties and Fuel/Clad Gap Conductance for Both Power and Temperature Calculations
- Maximum Decay Heat Loads Including 3σ Uncertainties and Time in Life Effects
- No Credit Taken for Inter- and Intra-Assembly Flow and Heat Redistribution
- Negative Reactivity Feedbacks Neglected (e.g., Core Radial Expansion, Bowing, Axial Expansion of Fuel and Cladding)
- All Above Assumed to Occur Simultaneously

8-2

DHRS OVERFLOW LINE

- OVERFLOW LINE CAPACITY INCLUDES ALL DHRS OPERATING CONDITIONS.
- FULL SCALE MODEL TEST CONFIRMS CAPABILITY OF OVERFLOW LINE
- DHRS OPERATING PARAMETERS AND OVERFLOW LINE DESIGN PREVENT FREEZING DURING DHRS OPERATION.

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PROTECTED WATER STORAGE TANK CAPACITY

- PWST INVENTORY ACCOMMODATES ALL DESIGN BASE EVENTS
- WITH PACC OPERATING ON AIR SIDE NATURAL CIRCULATION (BEYOND DESIGN BASIS), PWST INVENTORY ACCOMMODATES A THIRTY DAY MISSION
- WITH DECAY HEAT REMOVAL PERFORMED ONLY BY VENTING (BEYOND DESIGN BASE), PWST INVENTORY ACCOMMODATES A MINIMUM OF NINE HOURS OF HEAT REMOVAL

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CONDENSATE STORAGE TANK

- AVAILABLE WATER SUPPLY TO PWST AND TO AFW PUMP SUCTION (250,000 GALLON CAPACITY)
- GRAVITY DRAIN TO BOTH PWST AND AFW PUMPS
- SAFETY-RELATED ISOLATION VALVES
- NON-SEISMIC CATEGORY TANK
- NOT CONSIDERED IN SAFETY ANALYSIS

PRIMARY CHECK VALVE

- SCALE-UP OF FFTF CHECK VALVE WITH DASHPOT ADDED
- MINIMAL PRESSURE DROP UNDER NATURAL CIRCULATION FLOW (~0.03 PSIG)
- 13 1/2° FREEHANGING ANGLE ON SEAT
- DASHPOT HAS BEEN PERFORMANCE TESTED AND THERMAL-CYCLE TESTED
- FAILURE MODES AND EFFECTS ANALYSIS PERFORMED
- FAILURE MODE EVALUATION AND PROBABILISTIC EVALUATIONS PERFORMED

3-4

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CLARIFICATION OF AFWS/SWRPRS USE

- RESPONSE PROVIDED IN NRC QUESTION RESPONSES CS 421.28 AND CS 760.115
- SWRPRS ACTUATION INVOLVES ONLY AFW ISOLATION VALVES IN THE **AFFECTED** LOOP
- DECAY HEAT REMOVAL CONTINUES VIA REMAINING TWO HEAT TRANSPORT LOOPS

4-1

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PACC SYSTEM ANALYSIS

- PSAR SECTION 5.6.1.2.3 AND 5.6.1.3.2 UPDATED TO PROVIDE DETAILS FOR SYSTEM DESIGN AND ANALYSIS
- PROJECT IS COMMITTED TO VERIFICATION TESTING

4-2

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AUXILIARY FEEDWATER SYSTEM RELIABILITY EVALUATION

- ADDRESSED IN APPENDIX H, SECTION II.E.1.1 (AMENDMENT 66)
- EVENT TREE AND FAULT TREE ANALYSIS DONE UNDER DHR BY KEY SYSTEM REVIEW
- FMEA AND CCFA BEING DONE UNDER QUALITATIVE RELIABILITY PROGRAM
- QUANTITATIVE ANALYSIS BEING DONE USING FAILURE STATE MODEL
- PSAR SECTION 5.6.1.3.12 DETAILS COMPLIANCE WITH SRP SECTION 10.4.9 AND BTP ASB 10-1

4-3

LOSS OF BULK AC POWER (PLANT BLACKOUT) FOR TWO HOURS

- CURRENTLY ASSUMED AS A CONDITION FOR NATURAL CIRCULATION EVALUATIONS
- NATURAL CIRCULATION DESCRIBED PREVIOUSLY AND IN CRBRP-ARD-0308 PRESENTS LONG TERM AND SHORT TERM ANALYTICAL RESULTS

4-4

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Task A-44 Station Blackout

A loss of all ac power is not a design-basis event for the CRBRP facility. Nonetheless, requirements have been imposed to ensure that CRBRP will have substantial resistance to a loss of all ac power and that, even if a loss of all ac power should occur, there is reasonable assurance the core will be cooled and the health and safety of the public assured. These design, requirements are discussed below.

A loss of offsite ac power involves a loss of both the preferred and backup sources of offsite power.

If offsite ac power is lost, three diesel generators and their associated distribution systems will deliver emergency power to safety-related equipment.

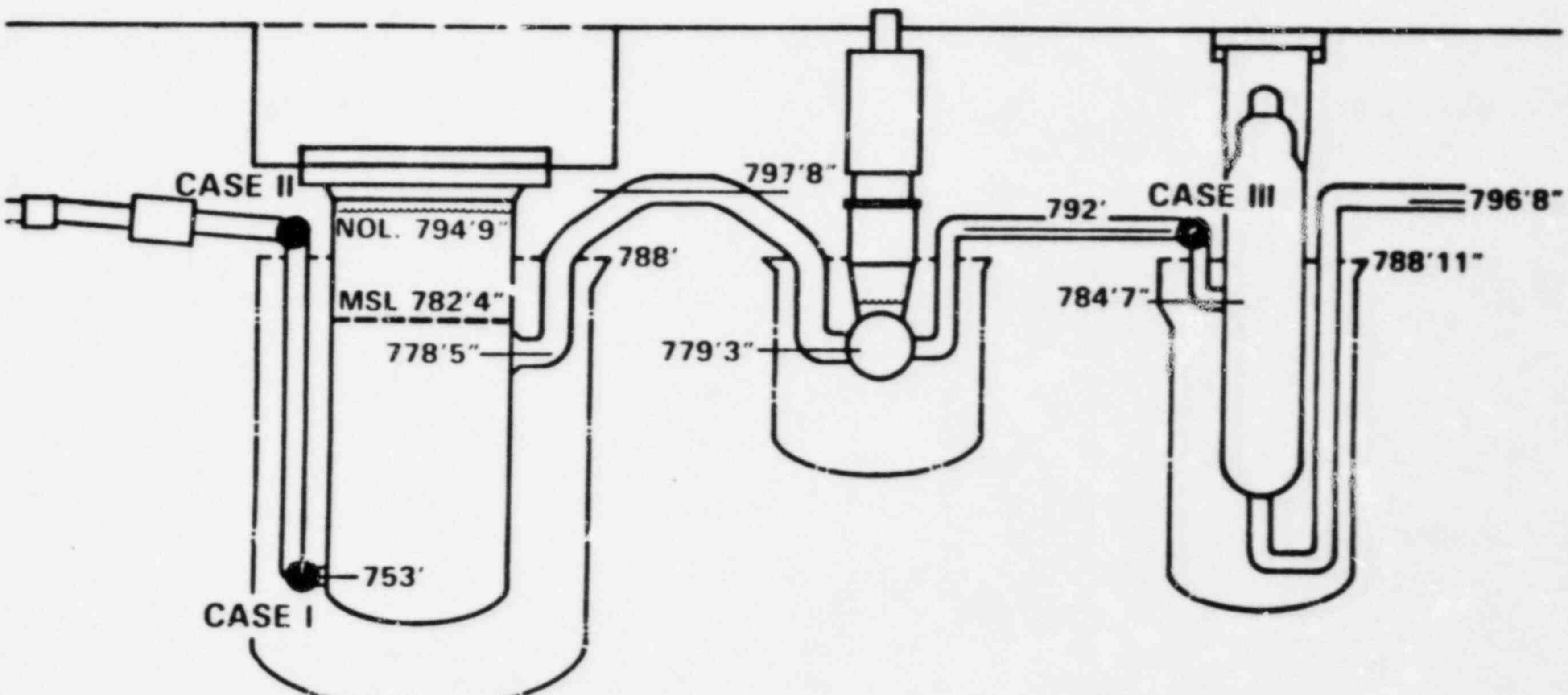
If both offsite and onsite ac power are lost, CRBRP is designed to remove reactor generated decay heat on natural circulation with the heat sink provided by the steam generator auxiliary heat removal system. This capability ensures that adequate cooling can be maintained for at least 2 hours, which allows time for restoration of ac power from either offsite or onsite sources. This capability has been described in PSAR Chapter 5.6 and CRBRP-ARD-0308.

The decay heat generated in the spent fuel in the Ex-Vessel Storage Tank (EVST) is also capable of being removed by natural circulation. This is provided by the third EVST cooling loop which is designed to remove all decay heat produced in the EVST during natural circulation. This capability is described in PSAR Chapter 9.1.

The Ex-Vessel Transfer Machine is designed to assure that cladding temperature is maintained within limits by a natural convection cooling system. This assures cooling of a fuel assembly in transit between the reactor and EVST. This capability is discussed in PSAR Chapter 9.1.

A two hour station blackout while handling a bare fuel assembly during normal Fuel Handling Cell (FHC) operations could result in release of fission products to the environment. The potential radiation doses at the site boundary resulting from such a release are below established limits. The results of such releases are discussed in PSAR Chapter 9.1. Based on the above considerations, the Project concludes that there is reasonable assurance that CRBRP can proceed before the ultimate resolution of this generic issue without endangering the health and safety of the public.

FAILURE LOCATION FOR GUILLOTINE RUPTURE CASES EVALUATED



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ASSUMPTIONS

- DOUBLE ENDED GUILLOTINE RUPTURES
- TRIP ON FLUX $-\sqrt{\text{PRESSURE}}$
- PLANT EXPECTED OPERATING CONDITIONS
- NOMINAL PEAK CORE TEMPERATURES
- MINIMUM SATURATION TEMPERATURE

4-7

DEMO DOUBLE ENDED PIPE RUPTURE RESULTS

(BASED ON PLANT BEST ESTIMATE CONDITIONS)

	FUEL ASSEMBLY	INNER BLANKET ASSEMBLY
• CASE I	1210°F	1265°F
• CASE II	1185°F	1185°F
• CASE III	1110°F	1145°F

CONCLUSION: CASE I IS THE WORST CASE.

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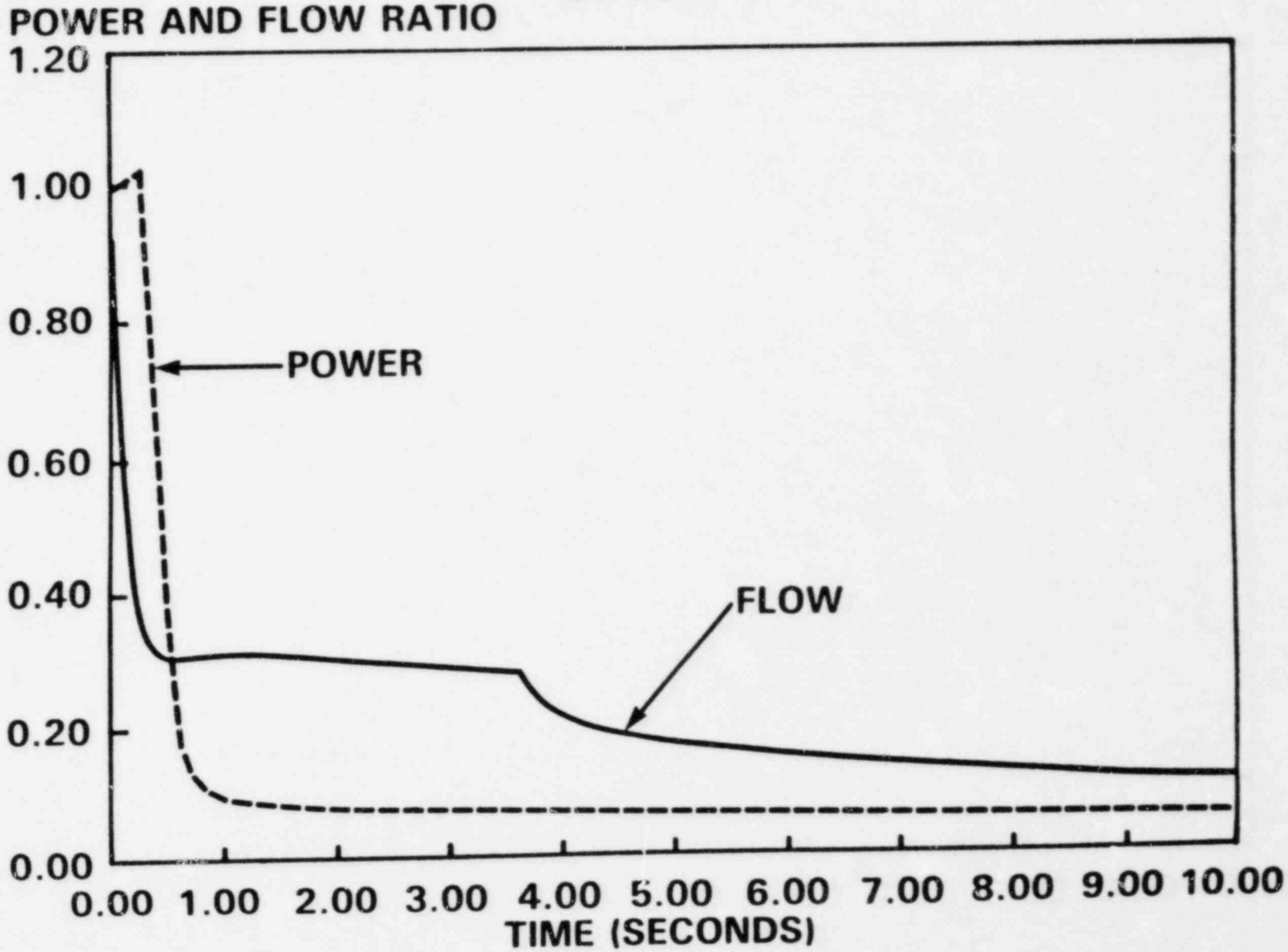
PIPE RUPTURE, CASE I POWER AND FLOW COASTDOWN

VARIATION IN POWER AND FLOW FOR CASE I



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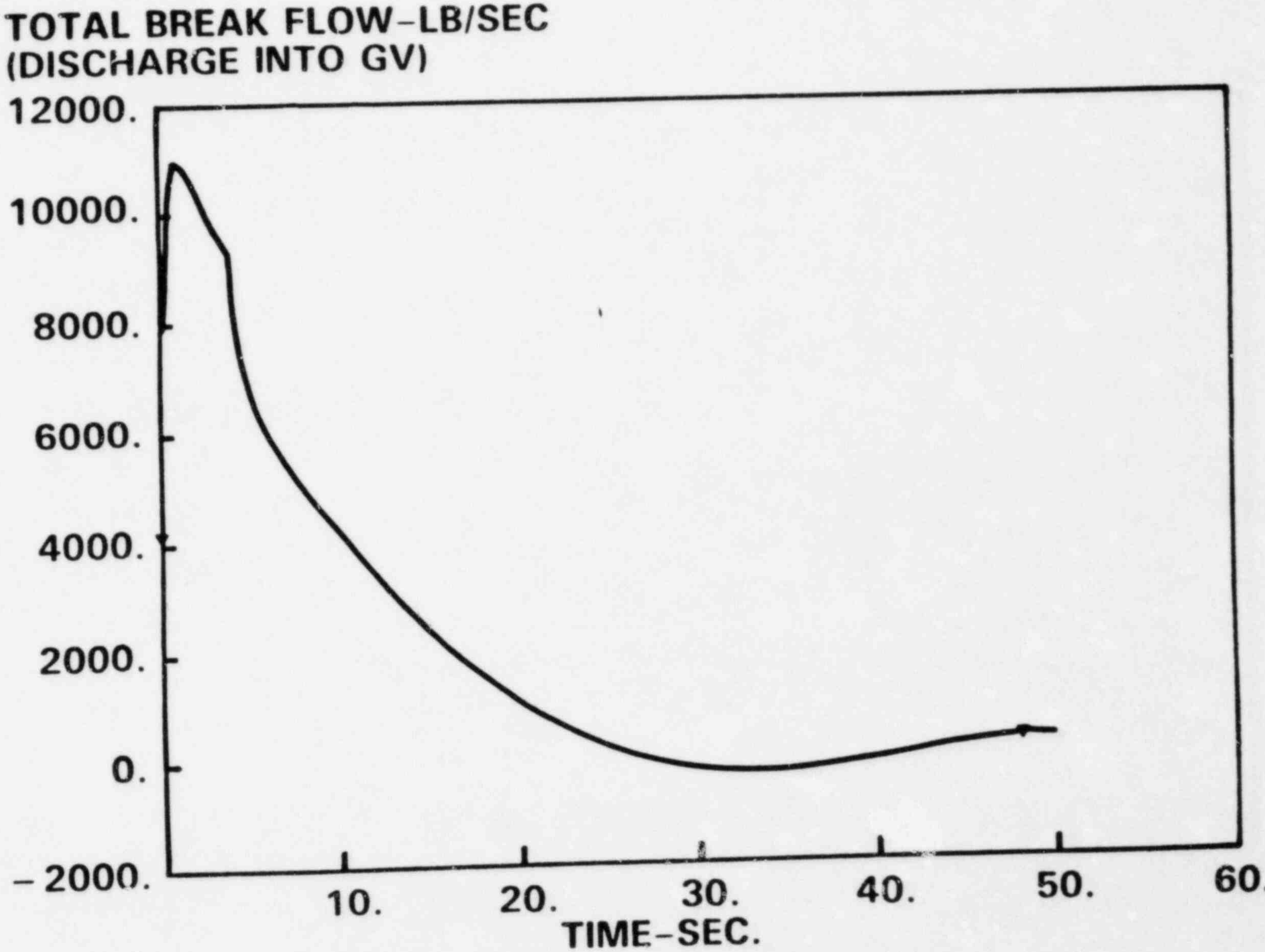
DER AT RV INLET NOZZLE



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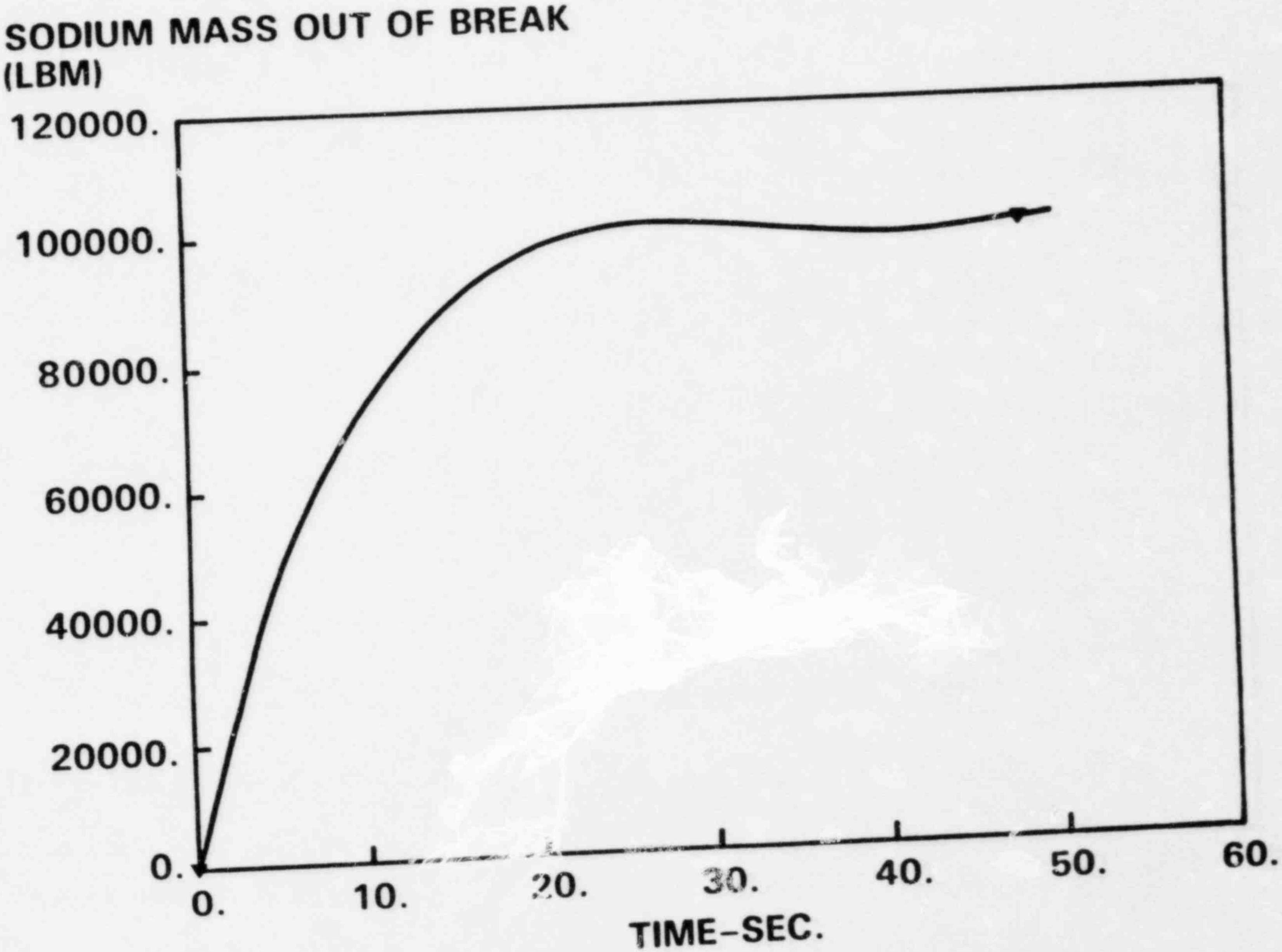
DER AT RV INLET NOZZLE



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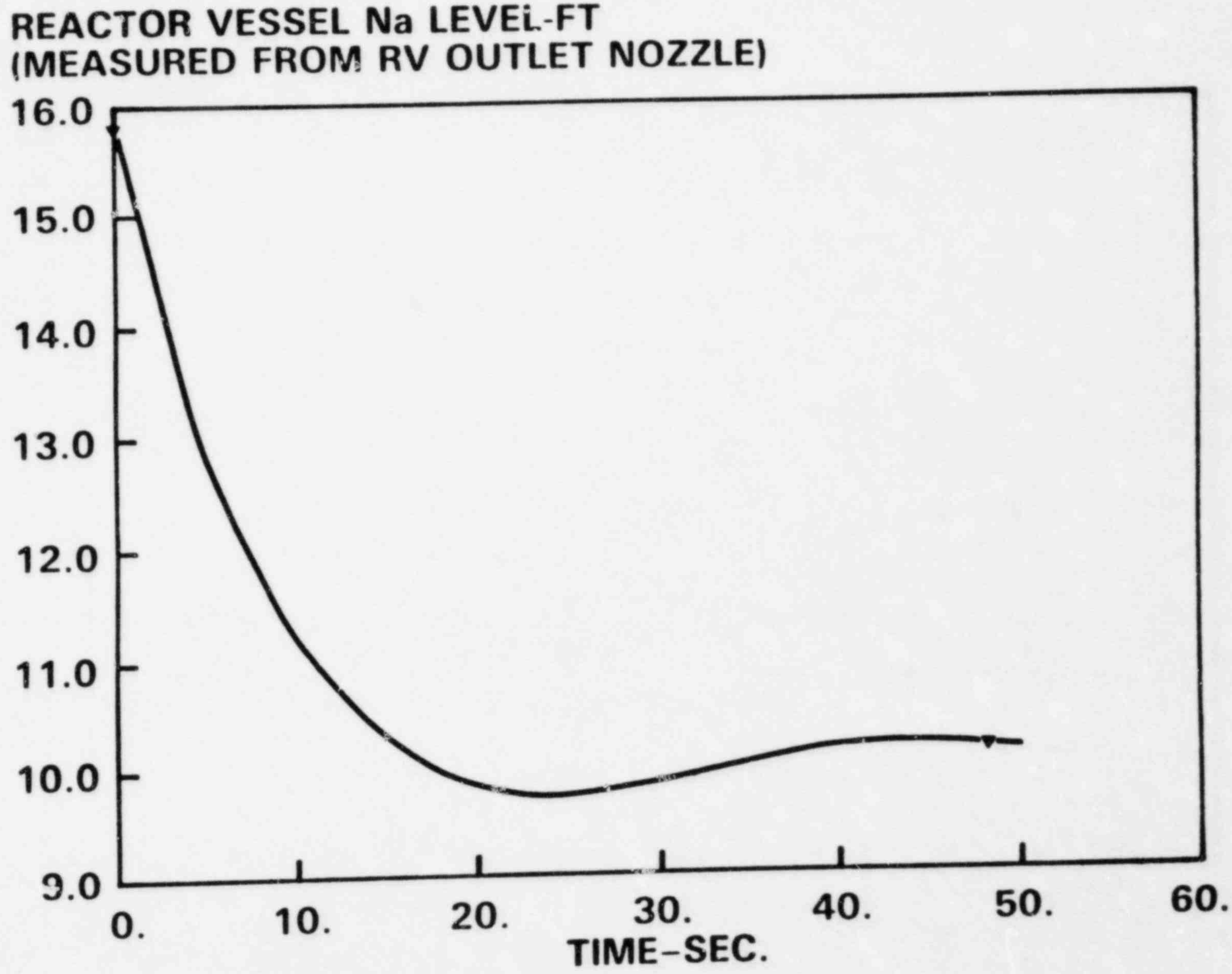
DER AT RV INLET NOZZLE



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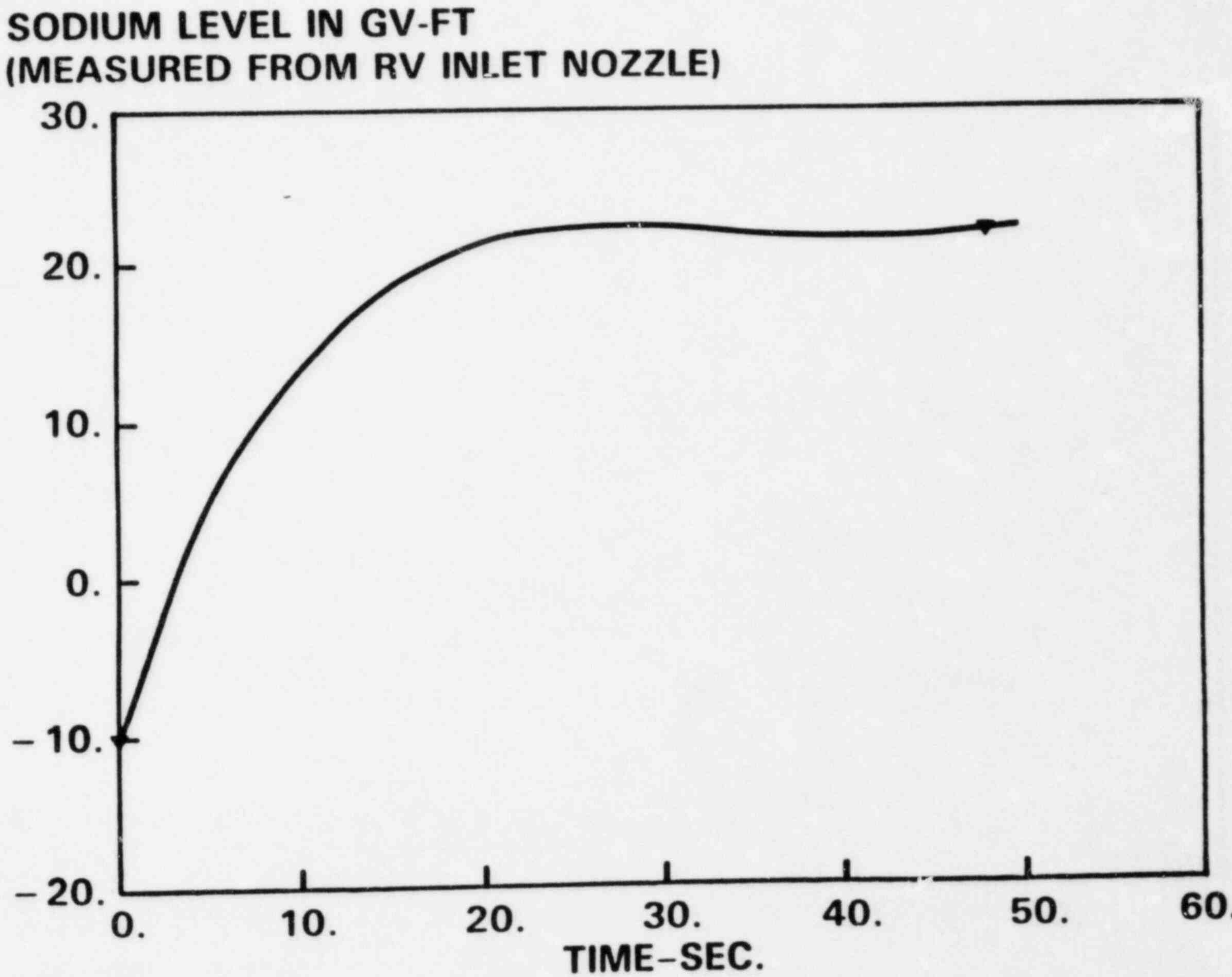
DER AT RV INLET NOZZLE



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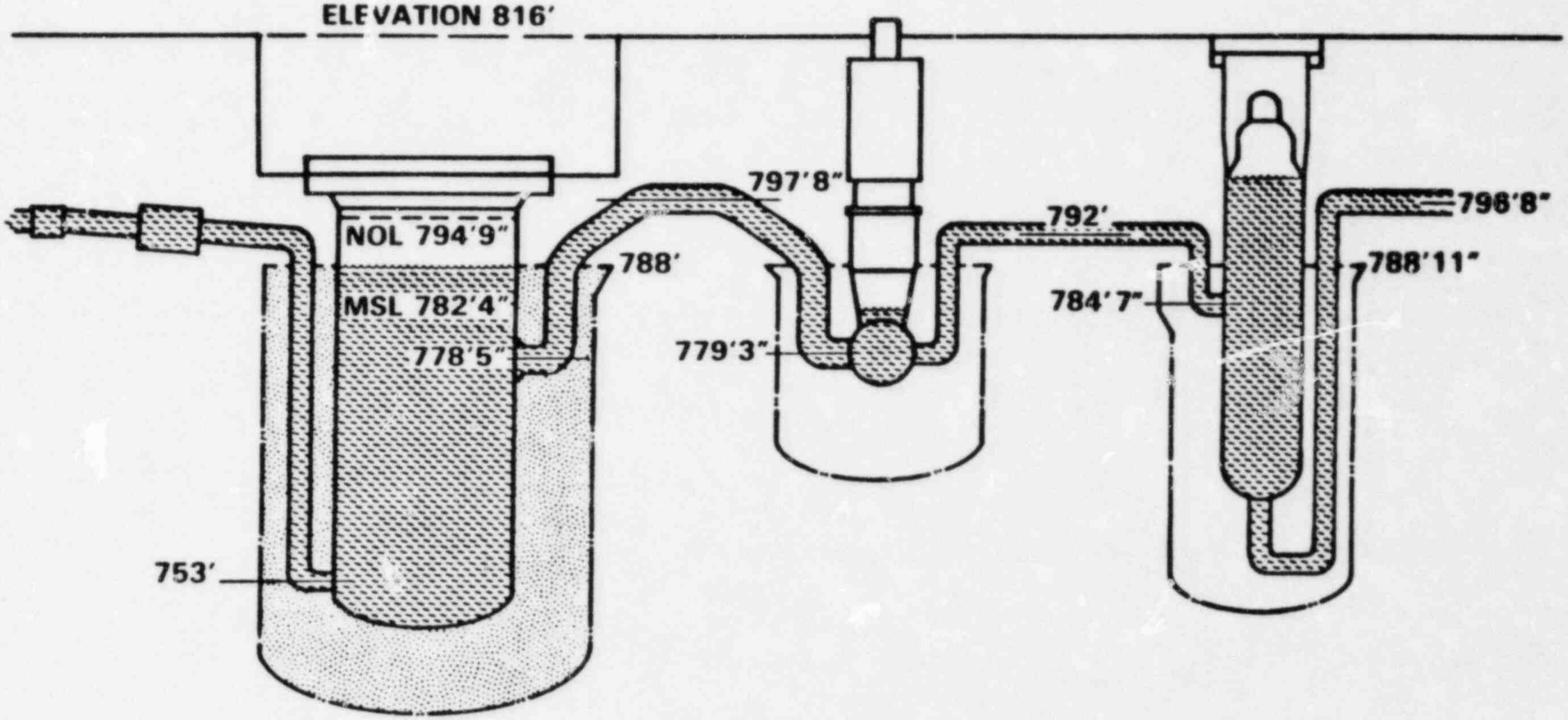
SIMPLIFIED HYDRAULIC PROFILE—PHTS
REACTOR INLET INTO GUARD VESSEL



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F-14

FORE-2M MAXIMUM TEMPERATURES FOR THE WORST CASE DOUBLE ENDED PIPE RUPTURE AS CALCULATED BY DEMO (CASE I)

(BASED ON PLANT BEST ESTIMATE CONDITIONS)

	MAXIMUM COOLANT TEMPERATURE	SATURATION TEMPERATURE	MARGIN TO BOILING
• FUEL ASSEMBLY	1399°F	1586°F	187°F
• INNER BLANKET ASSEMBLY	1380°F	1586°F	206°F
• OUTER BLANKET ASSEMBLY	1345°F	1586°F	240°F

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Enclosure 3

SHUTDOWN HEAT REMOVAL MEETING
December 16, 1982

Attendees

Name	Organization
Al Meller	CRBRP-PO/PMC
J. G. Guppy	Brookhaven National Lab
R. A. Markley	W-ARD
Vipin L. Shah	ANL
Mohsen Khetib-Rahbar	BNL
F. M. Heck	ARD
Ron Coffield	W-ARD
Stephen Additon	Westinghouse (WLLCO)
Richard E. Ireland	NRC-Idaho
L. N. Rib	NRC - Consultant
M. B. Holz	NRC CRBR-PO
T. L. Kinnaman	EG&G Idaho
W. P. Barthold	Barthold & Associates, Inc.
D. L. DeMott	W-OR
Bob Capp	EG&G Idaho
K. R. Perkins	BNL
G. J. Van Tuyle	BNL
Stephen Sands	NRC
Tom King	NRC
Rich Stark	NRC
Don Florek	CRBRP-PO-DOE
R. Lowrie	W-ARD
R. Hottel	W-OR
Bill Murphie	CRBRP-DOE-HQ