CRBR LOOSE PARTS MONITORING SYSTEM

STAFF PRESENTATION TO ACRS - 3/16/83

T. KING

TR04
delete B. White

SUMMARY OF STAFF REVIEW

- APPLICANTS INITIALLY PROPOSED NO LOOSE PARTS MONITORING SYSTEM.

- STAFF DID NOT FIND ANY COMPELLING REASON TO EXCLUDE SUCH A SYSTEM FROM CRBR AND; THEREFORE, PER SRP SECTION 4.4 SUCH A SYSTEM IS CONSIDERED APPLICABLE TO CRBR.

- APPLICANTS HAVE NOW COMMITTED TO THE DESIGN INSTALLATION AND OPERATION OF A LPMS PER R.G. 1.133 GUIDELINES.

MAJOR DESIGN CRITERIA

R.G. 1.133 GUIDELINES
- LOCATE TWO SENSORS AT EACH NATURAL COLLECTION POINT OF PRIMARY SYSTEM.
- BE SENSITIVE ENOUGH TO DETECT LOOSE PARTS LARGE ENOUGH TO CAUSE DAMAGE
- DESIGN SYSTEM FOR CHANNEL SEPARATION AND OBE
- PROVIDE AUDIBLE ALERT TO OPERATORS
- PROVIDE FOR TESTABILITY AND CALIBRATION

CRBR SENSOR LOCATIONS
- REACTOR VESSEL
- PHTS & IHTS PUMPS
- IHXS
- SGS
- NATURAL COLLECTION POINTS OF SYSTEM

COMPONENT NOISE AND VIBRATION MEASUREMENTS TO LOOK FOR DEGRADATION.

EXPERIENCE BASE

EBR-II - SUCCESSFUL TESTING OF HIGH TEMPERATURE IN-SODIUM MICROPHONES.

FFTF - INSTALLATION AND OPERATION OF LPMS FOR REACTOR VESSEL.

- EXPERIENCE WITH HIGH TEMPERATURE ACCELEROMETERS IN-VESSEL.

CONCLUSION

COMMITMENT BY APPLICANTS TO DESIGN, INSTALL AND OPERATE A LPMS FOR CRBR IN ACCORDANCE WITH R.G. 1.133 IS ACCEPTABLE FOR CP.

OBJECTIVES OF NRC REVIEW OF PSI/ISI PLAN

- FABRICATION EXAMINATIONS AND PSI ARE PERFORMED WITH BEST AVAILABLE TECHNOLOGY
- NDE REQUIRED FOR MAINTENANCE, REPAIR OR MODIFICATION AND IS CONSIDERED IN PLANT DESIGN
- ACCESS PROVIDED FOR PERIODIC VOLUMETRIC ISI
- SPECIALIZED EQUIPMENT DESIGNED FOR PLANNED ISI

CP REVIEW CONSIDERATIONS

FABRICATION/PSI

- DOUBLE ANGLE RT WILL BE PERFORMED ON WELDS IN VESSELS AND PIPING
- UT OF PIPE WELDS ($\geq$ 1/2 INCH WALL) SHOULD BE PERFORMED USING EXISTING TECHNOLOGY
- UT OF VESSEL WELD HAZ AND ADJACENT BASE METAL SHOULD BE PERFORMED WHERE TECHNICALLY FEASIBLE

CP REVIEW CONSIDERATIONS

ISI

- EXAMINATIONS SHOULD BE PERFORMED DURING THE REQUIRED MAINTENANCE, THUS REDUCING OCCUPATIONAL EXPOSURE AND PLANT OUTAGE TIME
- SELECTED VOLUMETRIC EXAMINATIONS SHOULD BE PERFORMED TO DETECT GENERIC, UNANTICIPATED DEGRADATION MECHANISMS
- INTEGRATED LEAKAGE DETECTION SYSTEM WILL BE DEMONSTRATED UNDER PLANT ENVIRONMENT AND TESTED PER TECHNICAL SPECIFICATIONS TO MAINTAIN ITS EFFECTIVENESS

TOPICS STILL BEING EVALUATED BY THE APPLICANT

- SURVEILLANCE PROCEDURES FOR THE REACTOR INTERNALS
- INSPECTION TECHNIQUES FOR THE IHX TUBE BUNDLE IN THE EVENT THAT MAINTENANCE PROVIDES ACCESS TO THE TUBES

OL REVIEW CONSIDERATIONS

- EVALUATION OF EXISTING NDE TECHNOLOGY FOR ISI
- DESIGNATE SPECIFIC LOCATIONS, METHODS AND FREQUENCY OF PERIODIC INSERVICE INSPECTION

SUMMARY

- ADEQUATE EXAMS WILL BE PERFORMED DURING FABRICATION/PSI TO PROVIDE ASSURANCE THAT NO SIGNIFICANT FLAWS ESCAPE DETECTION
- BASELINE DATA WILL BE AVAILABLE IN THE EVENT ISI IS REQUIRED
- NDE TECHNIQUES WILL BE AVAILABLE TO SUPPORT INSERVICE MAINTENANCE, REPAIR AND MODIFICATION
- PERIODIC ISI WILL BE PERFORMED TO DETECT GENERIC DEGRADATION

LOOSE PARTS MONITORING SYSTEM
CONCLUSIONS

- CRBRP PROJECT HAS DETERMINED THAT THERE ARE NO POTENTIAL LOOSE PARTS THAT COULD DEGRADE THE ABILITY OF CRBRP SYSTEMS TO PERFORM THEIR INTENDED SAFETY FUNCTION.
- LPMS WILL BE A VALUABLE DIAGNOSTIC TOOL FOR THE CRBRP.

ACRS SUBCOMMITTEE MEETING
ACCOMMODATION OF LOCAL FUEL FAULTS
AND ROLE OF INSTRUMENTATION

March 16, 1983

Westinghouse Electric Co.
Advanced Energy Systems Division
Madison, PA 15663

R. A. Markley
A. L. Schwallie
L. E. Strawbridge
R. W. Tilbrook
R. J. Tinder

INTRODUCTION

INTRODUCTION

- Summarize Local Faults Technical Information
  - Provide Description of Fuel Design Characteristics
  - Review Pertinent Data Base and Conclusions on Fuel Failure Propagation
  - Provide Description of Fuel Failure Monitoring System Response
  - Intended Use and Utility of Core Outlet Thermocouples
  - Foreign and Domestic Experience on PPS Instrumentation Relative to Local Faults

TECHNICAL SUMMARY ON LOCAL FAULTS

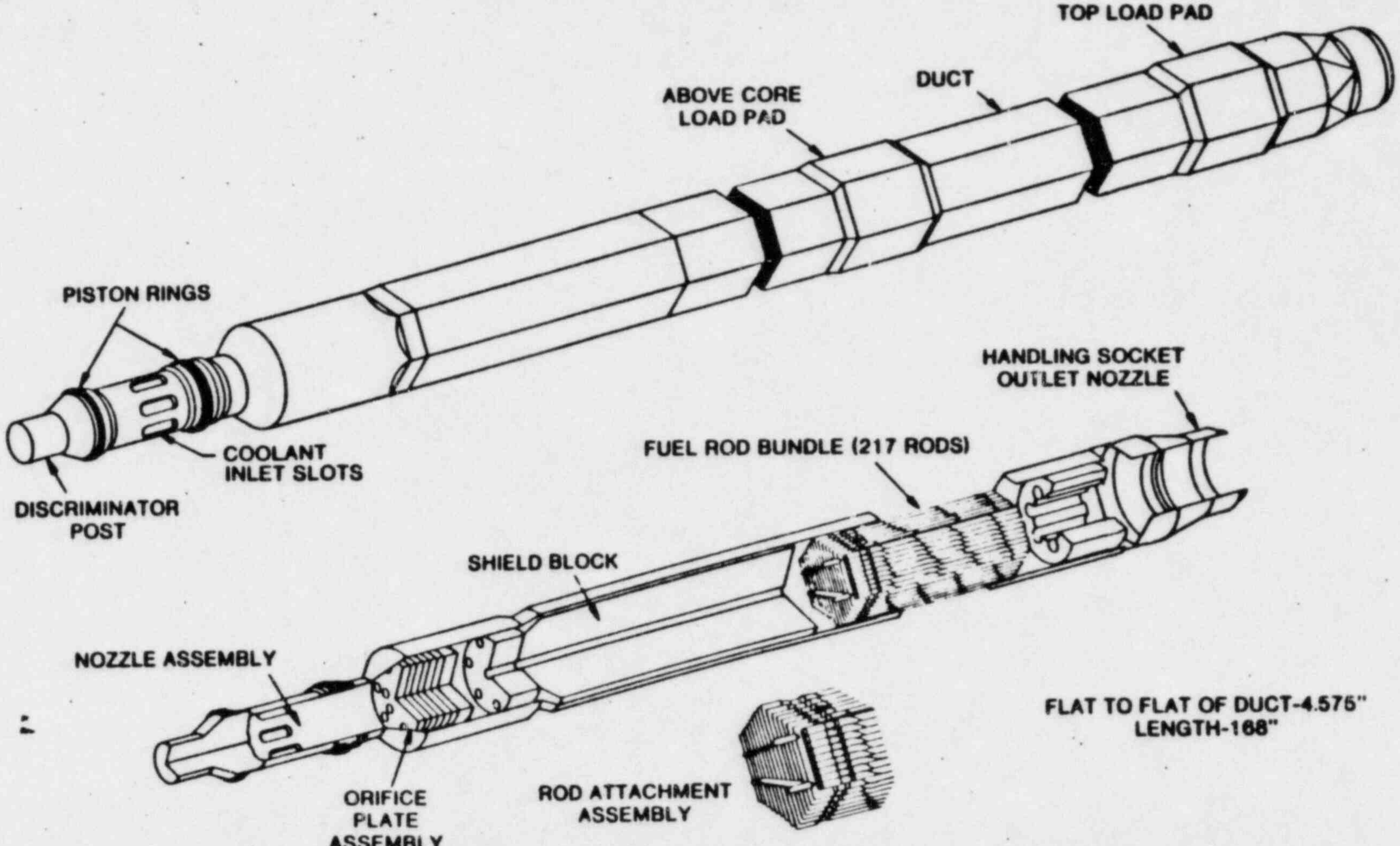
- Rapid Rod-to-Rod Failure Propagation Due to Local Faults Leading to a Condition of Loss of Coolable Geometry is Incredible. It has not Occurred in Run Beyond Cladding Breach (RBCB) Testing nor in Operating Reactors with Failed Fuel.

- Instrumentation to Alert the Operator to Local Fault Conditions is Available in CRBRP, and Considering the Long Time Intervals Necessary for Rod-to-Rod Propagation, Provides Sufficient Warning for Operator Corrective Action.

- Since Rapid Rod-to-Rod Propagation is Incredible, Requirements for Local Fault Detection and Protection Against Fuel Failure Propagation in the PPS are not Considered Necessary.

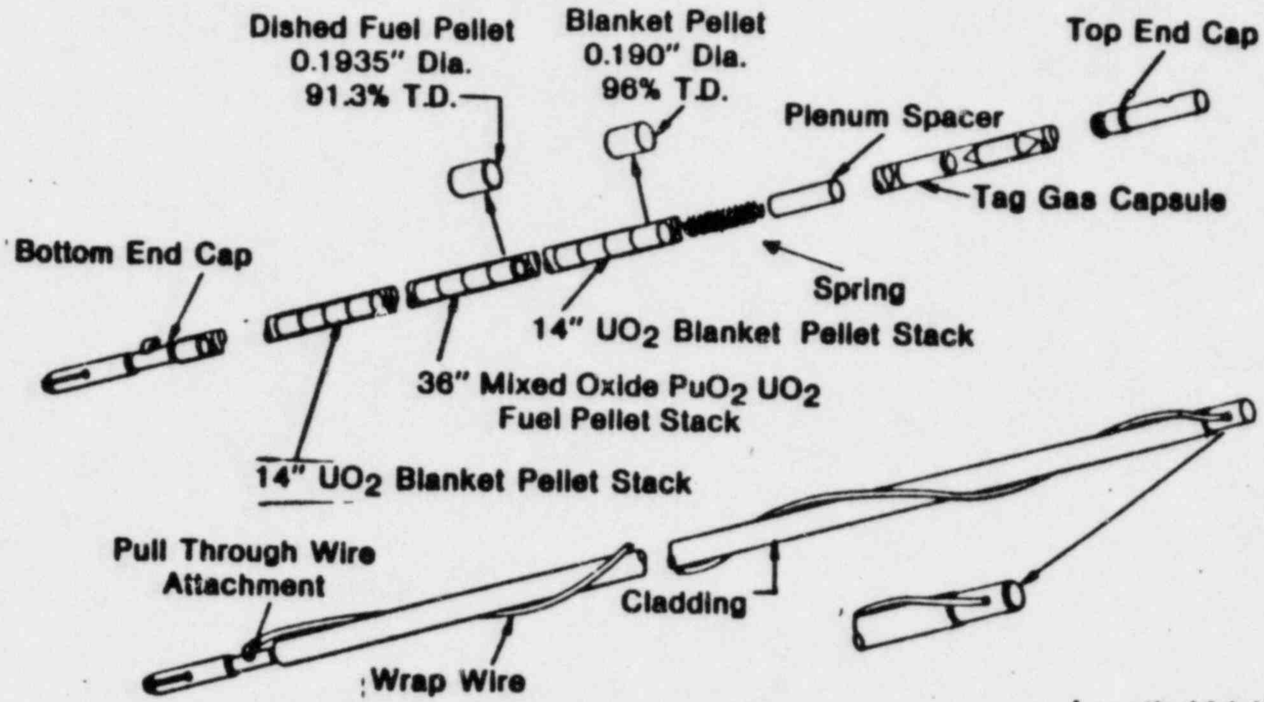
FUEL DESIGN OVERVIEW

CRBRP FUEL ASSEMBLY



FLAT TO FLAT OF DUCT-4.575"
LENGTH-168"

CRBRP FUEL ROD



Length 114.4"

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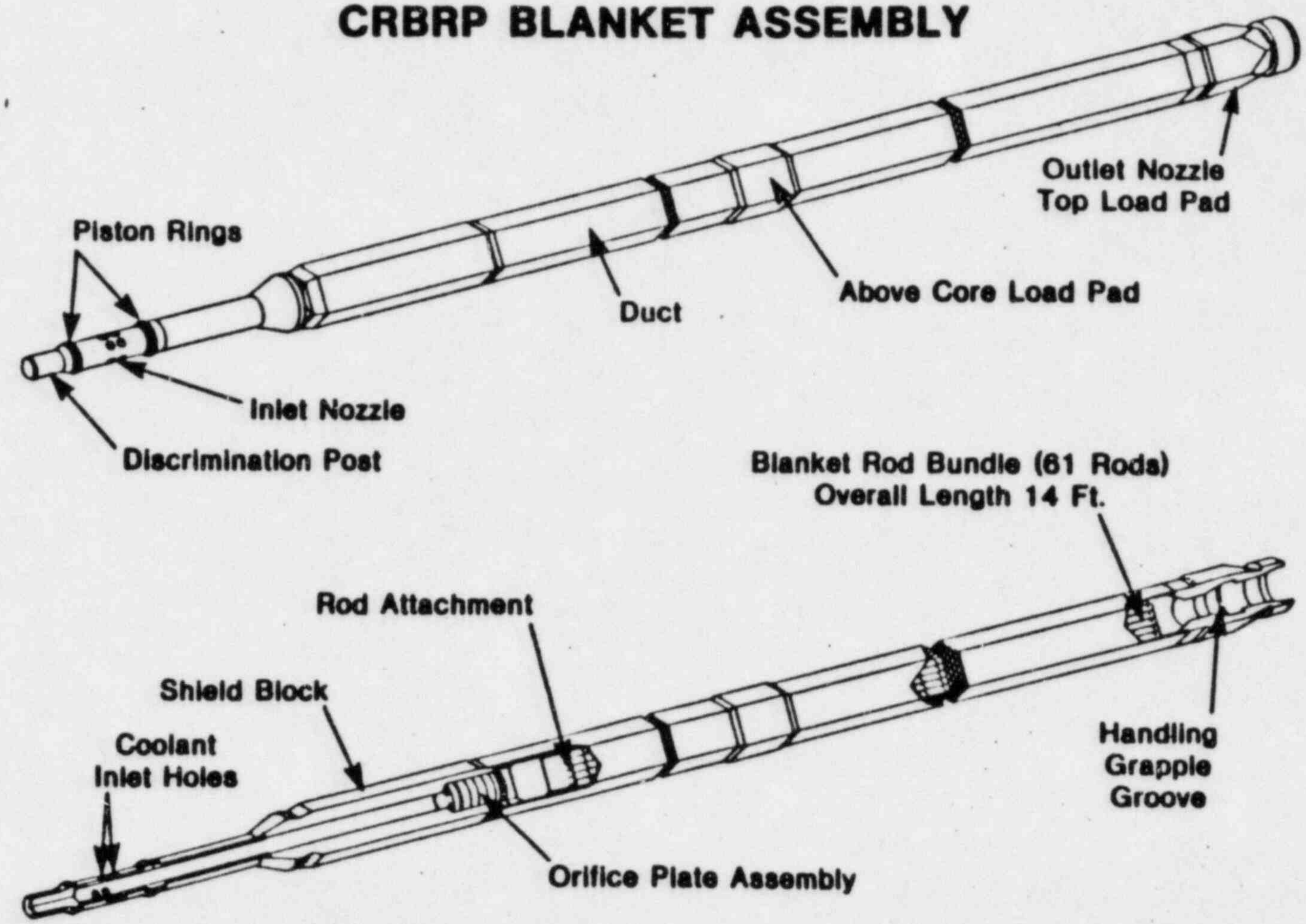
CRBRP BLANKET ASSEMBLY



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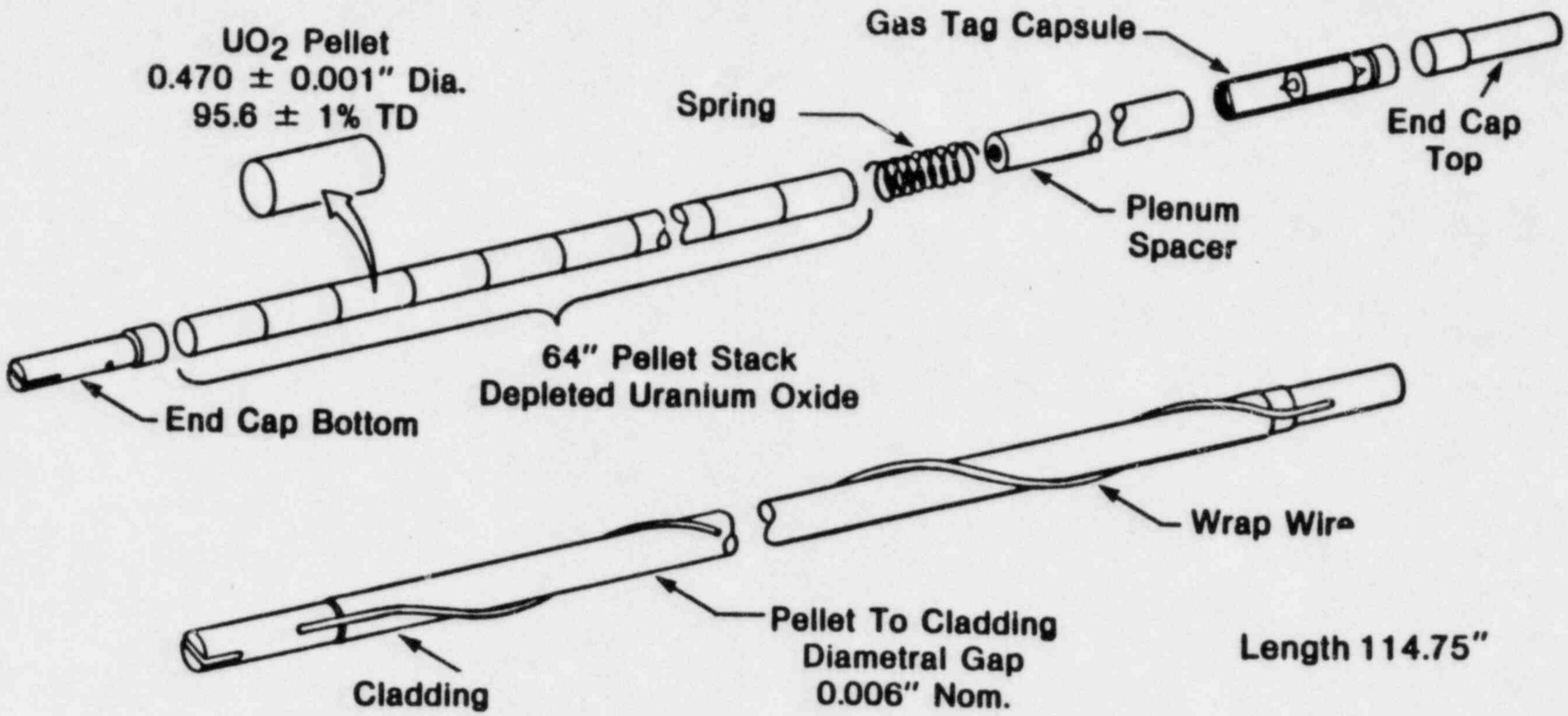
CRBR BLANKET ROD



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DAMAGE SEVERITY LIMITS

Event Category	Damage Severity Level	
Normal Operation	No Significant Loss of Effective Lifetime	Design Limits Preclude Mechanistic Failures for Normal Operation Plus Anticipated Events and the Worst Unlikely Event Including Uncertainties.
Anticipated Events (Upset)	No Reduction of Effective Lifetime Below the Design Values	
Unlikely Events (Emergency)	A General Reduction in the Fuel Burnup Capability and, at Most, a Small Fraction of Fuel Rod Cladding Failures	
Extremely Unlikely Events (Faulted)	Maintain Coolable Configuration	No Sodium Boiling, Limited Fuel Melting, Cladding Solidus*

*PSAR Guideline

FUEL FAILURE MONITORING SYSTEM

FUEL FAILURE MONITORING SYSTEM (FFMS)

COVER GAS MONITORING SYSTEM (CGMS)

INTENT

- PROVIDE DETECTION OF ROD FAILURES IN FUEL AND BLANKET ASSEMBLIES
- PROVIDE COVER GAS ANALYSIS AND ACTIVITY MEASUREMENT
- PROVIDE OPERATOR WITH INFORMATION TO INITIATE OPERATION OF THE FAILED FUEL LOCATION SYSTEM (FFLS)

FAILED FUEL LOCATION SYSTEM (FFLS)

INTENT

- LOCATE FAILED FUEL AND BLANKET ASSEMBLIES
- CHARACTERIZE FAILURES

DELAYED NEUTRON MONITORING SYSTEM (DNMS)

INTENT

- PROVIDE CAPABILITY TO DETECT IN-CORE FUEL/SODIUM CONTACT BREACHES
- PROVIDE CAPABILITY TO IDENTIFY DEGREE OF FUEL/SODIUM INTERACTION AT IN-CORE BREACH

FUEL FAILURE MONITORING SYSTEM (FFMS) RESPONSE TIME

COVER GAS MONITORING SYSTEM (CGMS)

- COVER GAS TO DETECTOR = ~ 15-90 MINUTES (DEPENDS ON RELEASE RATE)

FAILED FUEL LOCATION SYSTEM (FFLS)

• COVER GAS TRANSIT TIME	=	~ 15 MINUTES
• ACCUMULATE SAMPLE	=	~ 15 MINUTES
• TAG GAS CONCENTRATION IN TRAPS	=	~ 7 HOURS
• MASS SPECTROMETER PROCESSING	=	~ 30 MINUTES
TOTAL	=	~ 8 HOURS

DELAYED NEUTRON MONITORING SYSTEM (DNMS)

- FULL FLOW TRANSIT TIME = < 1 MINUTE
- 40% FLOW TRANSIT TIME = 2 MINUTES
- COUNTING TIME ~ 1 TO 3 MINUTES

REQUIREMENTS PLACED ON THE FFMS

GENERAL REQUIREMENT: DETECT, LOCATE AND CHARACTERIZE FUEL AND BLANKET ROD FAILURES.

SPECIFIC REQUIREMENTS:

CGMS - DETECT FUEL AND BLANKET ROD BREACHES VIA COVER GAS ACTIVITY INCREASE. DESIGN SENSITIVITY IS 10^{-11} STANDARD CC FISSION GAS PER CC COVER GAS

FFLS - LOCATE AND CHARACTERIZE FUEL AND BLANKET ROD BREACHES VIA TAG GAS AND FISSION GAS ANALYSES. DESIGN SENSITIVITY IS < 1 PPB OF TAG IN COVER GAS

DNMS - DETECT FUEL-SODIUM CONTACT AT IN-CORE BREACH LOCATIONS FOR FUEL AND BLANKET RODS.

- DESIGN SENSITIVITY IS 1.5 CM^2 EXPOSED FUEL BY RECOIL.
- DESIGN SENSITIVITY PRECLUDES LARGE DIAMETER INCREASES, AND POSTULATED POROUS HEAT GENERATING BLOCKAGES WHICH COULD CAUSE CLADDING TEMPERATURES GREATER THAN 1600°F AND SODIUM BOILING.

RUN BEYOND FAILURE CONCEPT

- Operate with Gas Leakers Until DND Indication of Fuel-Sodium Contact by Continuous DN Signal Above Some TBD Level

- Remove All Known Fuel-Sodium Contact Leakers and Gas Leakers at Each Refueling Interval

RUN BEYOND CLADDING BREACH EXPERIENCE

CONCLUSIONS FROM OPERATING EXPERIENCE WITH BREACHED RODS

Steady State Operation

- Gas Type Leakers Present No Problem in Regard to Rod-to-Rod Propagation
- Finite Lifetime with Fuel-Sodium Contact Leakers - Extended to 22 Days and 0.75 cm^2 with Benign Effects. Limit not yet Found Although an Assembly has Gone ~ 96 Days with DN Signals and Multiple Failures (PIE in Progress)
- Operating Breached Rods have Enhanced DN Emission Over Recoil Making DN Detection More Sensitive - Breached Rods DN Emission is Exponential with Power.

Transient Operation

- The Behavior of Breached Fuel Rods Having Fuel-Sodium Contact (Logging) are not Seriously Degraded. Shutdown/Startup Extends Breach.

Areas Requiring Further Development Effort

- Diameter Increase Versus Fuel Exposure to Sodium
- Additional PPS Terminated TOP and LOF Results from Naturally Occurring Failures
- Detection and Instrumentation Diagnostics

RBCB WITH DEFECTED RODS

Completed Tests

- 2 S.S. EBR-II Pre-Defected Irradiated Fuel Assembly Tests Completed and Results Reported (RBCB-6,7)
- 3 S.S. EBR-II Irradiated Fuel Rod Assembly Tests with Natural Breach Completed (RBCB-1,2,3)
- 1 EBR-II Irradiated Fuel Rod Bundle S.S. Test with Natural Breaches Test Completed (XY-2)
- U.S. Data Indicates Benign Operation - Little Fuel Loss, Operation Possible for at Least ~ 22 Days, Defects Grow Due to Shutdown - Startup Combined Localized Diameter Increase < 15%, Detectable DND Signals, No Rod - Rod Propagation (Although a Low B.U. Rod in RBCB-2 Adjacent to Another Breached Rod Failed. Breach Possibly Due to Reconstitution; Diameter Increase Acceptable)

Planned or On-Going Testing

- 5 Kinetics and Contamination S.S. Bundle Tests (RBCB-K1, K2, K2A,B,C)
- 3 Detection and Instrumentation DND Tests (RBCB-D1, D2, D3 Variable Flow)
- 5 Fuel and Irradiation Variables Tests (RBCB-V2 Plenum Defects, RBCB-V4 Larger Diameter, RBCB-V5 Blanket, RBCB-V6 Unreconst., RBCB-V7 Sodium Storage Pre-Defected)
- ORT Program Includes 1 TOP Test (TOP 1-2)

CRBR Inlet Blockage Considerations

R. A. Markley

ACRS Subcommittee Meeting
March 16, 1983

ASSEMBLY INLET BLOCKAGE AND MODULE BLOCKAGE

- Assembly Inlet and Module Blockages are Highly Improbable Because of the Built-In Flow Blockage Prevention Features Which Provide Flow Path Redundancy:

  -- Multiple Primary Ports
  -- Axial Debris Barriers
  -- Radial Debris Barriers
  -- Multiple Auxiliary Ports
  -- Strainer
  } - Lower Inlet Module

  -- Multiple Inlet Slots - Assembly Nozzle

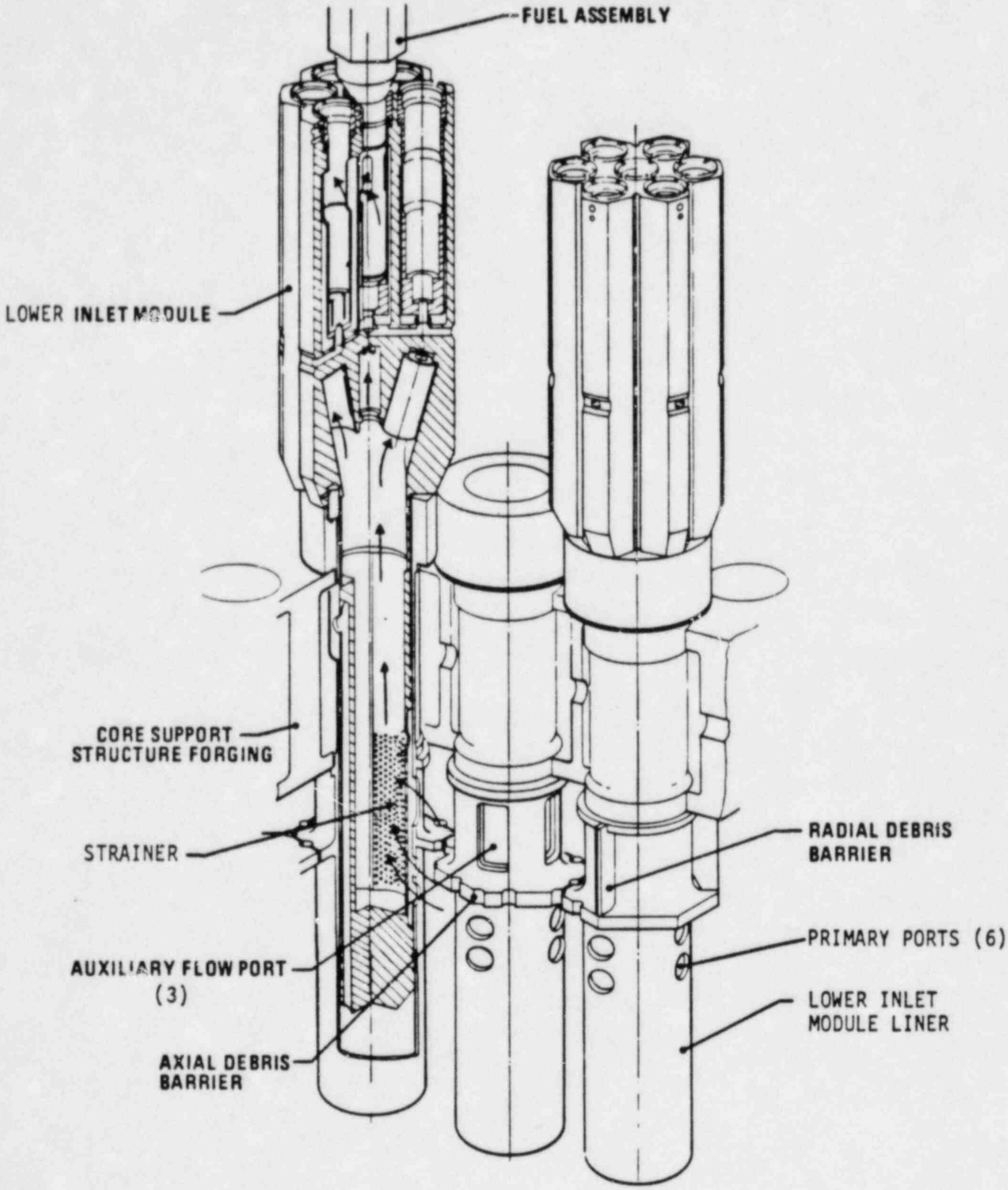
- The Strainer Prevents Debris Larger than <u>0.25</u> Inches from Entering the Modules

- The Orifice Plates Provide Another Level of Screening

- Debris Not Strained by the Fuel Rod Support Keys and Unheated Rod Bundle Entrance Will Pass Through Rod Bundles

- Even a 50% Areal Inlet Blockage Causes Less than 25°F Increase in Outlet Temperature

LOWER INLET MODULES
FLOW BLOCKAGE PREVENTION FEATURES



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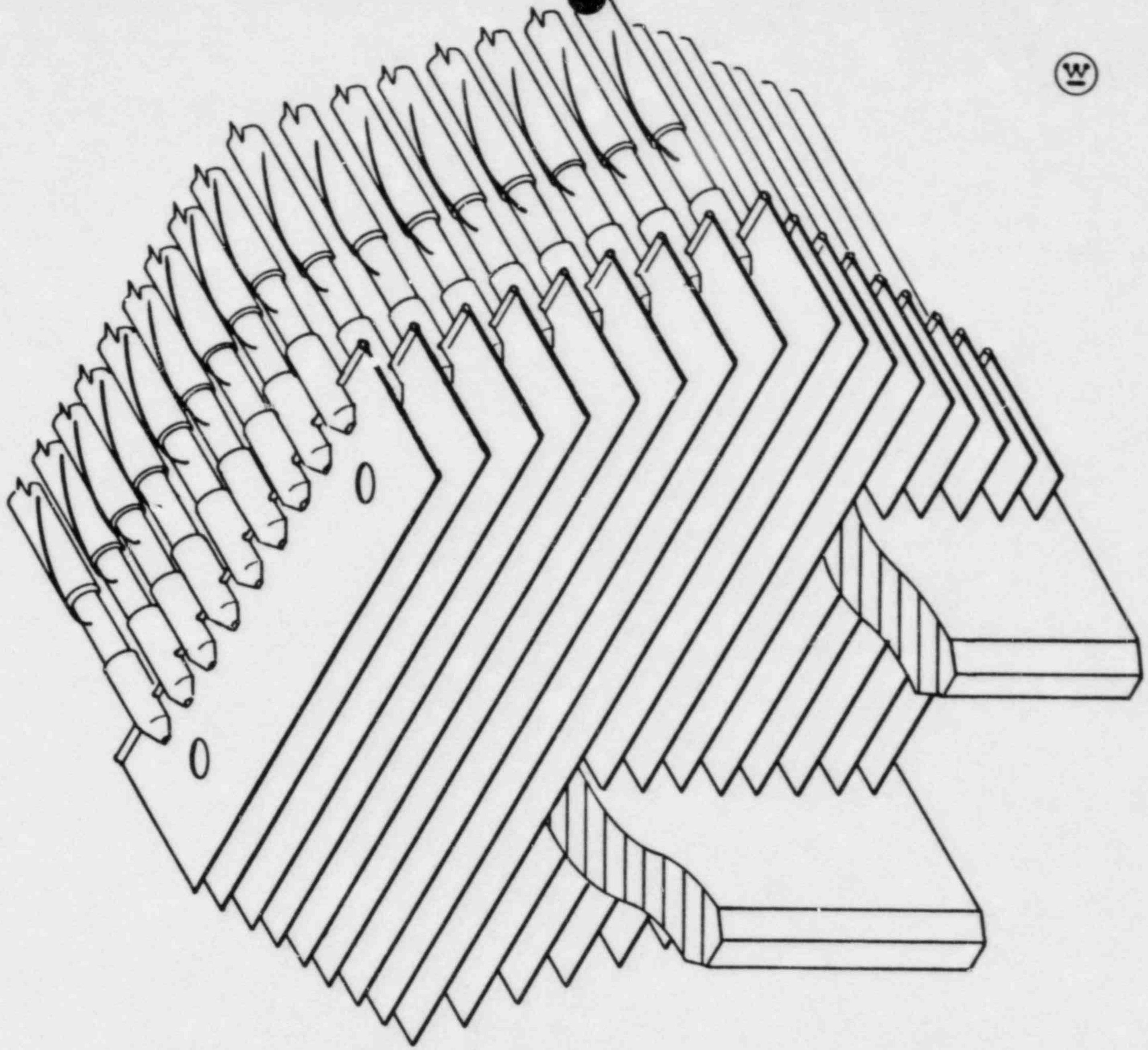
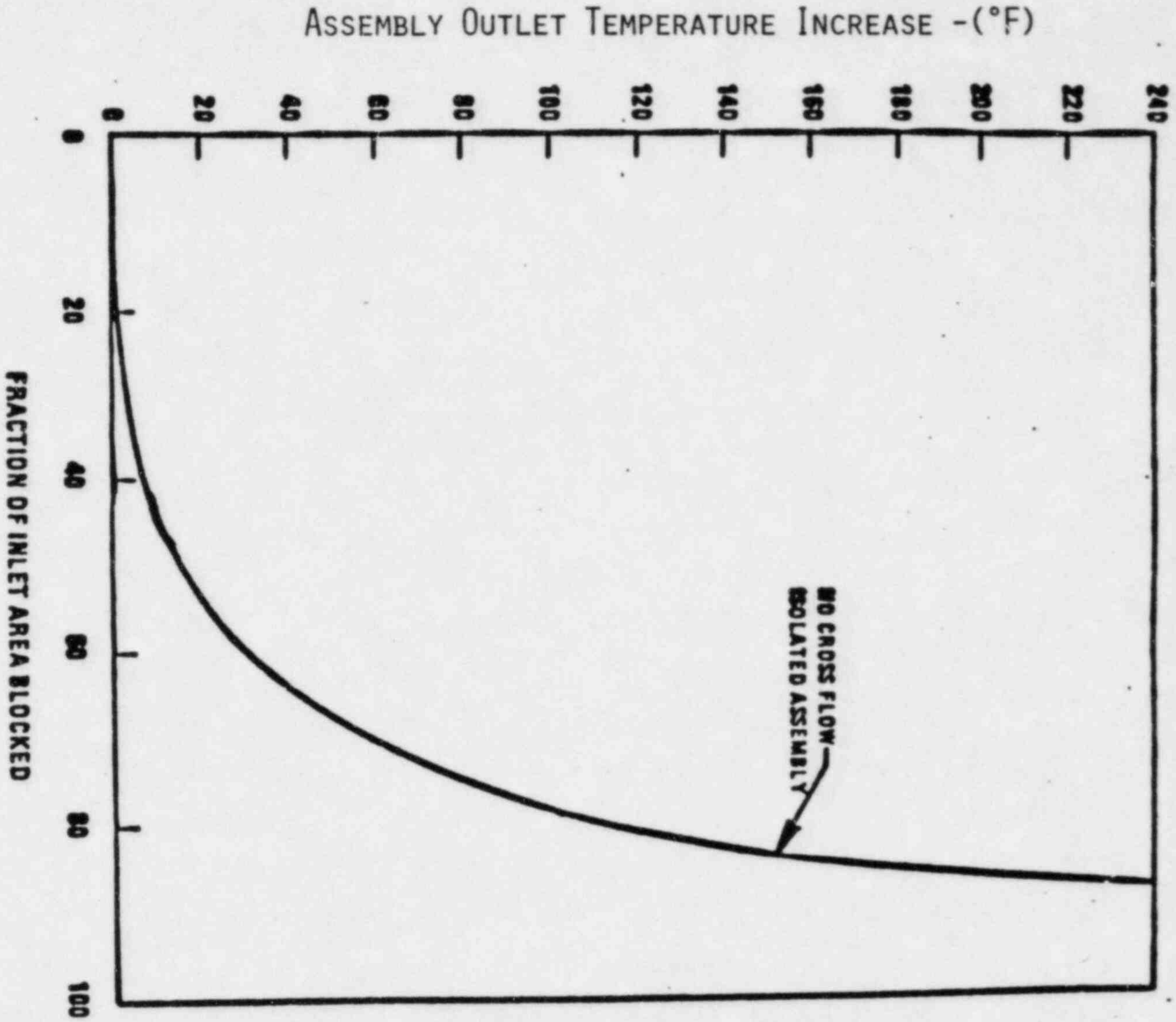
ROD BUNDLE INLET SCREENING

- LOW PROBABILITY
  - -- Q A PROGRAM COMPARABLE TO FFTF
  - -- CORE SPECIAL ASSEMBLIES DURING PREOPERATION TESTING FILTER REACTOR FLOW (>0.004 INCHES)
- ONLY SLOW BUILDUP POSSIBLE
  - -- PARTICULATE BUILDUP NOT PREFERENTIAL
  - -- PARTICLES OF CONCERN MUST BE IN 0.056" TO 0.25" RANGE
  - -- >400 INTERCONNECTED FLOW CHANNELS IN FUEL ASSEMBLY
- MARGINS TO ACCOMMODATE
  - -- REDUNDANT FLOW PATHS
  - -- CASCADING STRAINING HELPS
  - -- EVEN 50% AREAL INLET BLOCKAGE CAUSES LESS THAN 25°F INCREASE IN OUTLET TEMPERATURE

∴ ROD BUNDLE INLET BLOCKAGES VERY LOW PROBABILITY, SLOW, WITH LARGE MARGINS TO ACCOMMODATE



ROD BUNDLE ATTACHMENT RAIL ASSEMBLY


ASSEMBLY OUTLET TEMPERATURE INCREASE -(°F)
0
20
40
60
80
100
120
140
160
180
200
220
240
FRACTION OF INLET AREA BLOCKED
0
20
40
60
80
100
NO CROSS FLOW
ISOLATED ASSEMBLY

FUEL FAILURE PROPAGATION POTENTIAL

R. W. TILBROOK

WESTINGHOUSE ADVANCED REACTORS DIVISION

WALTZ MILL SITE

MARCH 1983

CHAPTER 15.4 LOCAL FAILURE EVENTS

DEFINITION: A FAILURE WHICH IS INITIATED WITHIN A SINGLE FUEL, BLANKET OR CONTROL ASSEMBLY

TYPE OF INITIATOR EVENTS:

1. STOCHASTIC ROD FAILURE
2. LOCAL FLOW BLOCKAGE
3. BUBBLE PASSING-THROUGH CORE

ALSO CONSIDERED:

4. MOLTEN FUEL

1. STOCHASTIC ROD FAILURE

- FISSION GAS RELEASE
- FUEL PARTICLE RELEASE
- OPERATION WITH FAILED FUEL

FISSION GAS RELEASE

- THERMAL EFFECT - GAS JET IMPINGEMENT
  - MAIN LOCATION FOR IMPINGEMENT WOULD BE AT PLENUM
  - LOWER RELEASE RATES IN FUEL REGION WOULD AVOID IMPINGEMENT
  - WIDE RANGE OF HOLE DIAMETER AND PIN PRESSURE TESTED

- THERMAL EFFECT - ROD GAS BLANKETING
  - TESTS SHOW DISPERSION OF GAS ACROSS ASSEMBLY
  - FLOW REDUCTION FROM GAS BLANKETING WOULD NOT CAUSE FAILURE

- TRANSIENT MECHANICAL LOADING
  - DIFFERENTIAL PRESSURE ACROSS ROD CANNOT CAUSE ROD FAILURE
  - DUCT WILL WITHSTAND MAXIMUM PRESSURIZATION (300 PSI VS. 550 PSI CAPABILITY AT 1000°F)

FISSION GAS RELEASE IN REACTOR SAFETY (CSNI REPORT NO. 40)

INTERNATIONAL COOPERATIVE STUDY SPONSORED BY "ORGANIZATION FOR ECONOMIC COOPERATION AND DEVELOPMENT" (1979)

- THE SAFETY TECHNOLOGY ON FISSION GAS RELEASE IS WELL DEVELOPED WITH THE SUPPORT OF A VAST AMOUNT OF EXPERIMENTAL DATA AND MANY YEARS OF EXPERIENCE

- FISSION GAS RELEASE BY ITSELF IS NOT A POSSIBLE CAUSE FOR FUEL ROD FAILURE PROPAGATION

FUEL PARTICLE RELEASE

- OPERATIONAL EXPERIENCE WITH FAILED FUEL INDICATES LITTLE WASHOUT AND NO PARTICULATE RETENTION IN WIRE WRAP ASSEMBLIES

- HEAT GENERATING BLOCKAGES ASSOCIATED WITH ASSUMED PARTICLE RETENTION IN THE BUNDLE ARE DISCUSSED LATER

OPERATION WITH FAILED FUEL

- CONSEQUENCES OF OPERATING WITH FAILED FUEL:
  - FISSION PRODUCT LEACHING: SODIUM CLEANUP SYSTEM REMOVES FPs
  - SODIUM INGRESS: HIGH INTERNAL PRESSURES FROM SODIUM VAPORIZATION HAVE NOT BEEN FOUND IN TESTS
  - FUEL/SODIUM CHEMICAL REACTION: ANY SWELLING WHICH OPENS THE BREACH EXPOSES MORE FUEL AND INCREASES THE D.N. DETECTOR SIGNAL. SWELLING IS SLOW AND HAS NEGLIGIBLE EFFECTS ON NEIGHBOR ROD HEAT TRANSFER

CONCLUSIONS: OPERATION WITH FAILED FUEL IS ACCEPTABLE. OPERATING LIMITS WILL BE ESTABLISHED.

SODIUM-FUEL REACTION

- SODIUM AND URANIUM-PLUTONIUM OXIDE WILL REACT IN THE PRESENCE OF FREE OXYGEN
- REACTION PRODUCT $NA_3\ MO_4$ HAS DENSITY ABOUT 1/2 OF FUEL
- UNDER WORST CASE CONDITIONS AT EOL, THEORETICAL UNIFORM EXPANSION OF CRBRP FUEL WOULD BE 1.7% DIAMETRAL
- KINETIC DATA INDICATE TWO TO SIX DAYS TO REACH EQUILIBRIUM
- EXPERIENCE INDICATES SODIUM-FUEL REACTION DOES NOT LEAD TO FAILURE PROPAGATION; DFR, RAPSODIE, AND BR-5 OPERATED LONGER THAN 100 DAYS WITH FUEL EXPOSED TO SODIUM WITHOUT EVIDENCE OF PROPAGATION

2. LOCAL FLOW BLOCKAGE

(WITHIN ROD BUNDLE)

- IN-CORE PASSIVE BLOCKAGE
- IN-CORE ACTIVE BLOCKAGE (HEAT-GENERATING BLOCKAGES)

CONCLUSIONS ON LOCAL FLOW BLOCKAGE

LOCAL FLOW BLOCKAGE DETAILS ARE IN THE FEBRUARY 3, 1983 PRESENTATION.

- EVEN A SIX-CHANNEL IN-CORE PASSIVE PLANAR BLOCKAGE WILL AT MOST ONLY REDUCE FUEL LIFETIME
- THE FORMATION OF HEAT-GENERATING BLOCKAGES (HGB) WITHIN THE CORE IS HIGHLY IMPROBABLE
- DELAYED NEUTRON DETECTOR (DND) SYSTEM WILL DETECT A HGB SMALLER THAN THAT WHICH COULD PROPAGATE DAMAGE

3. GAS BUBBLES PASSING THROUGH CORE

- THE PLANT IS DESIGNED TO PRECLUDE GAS BUBBLES ENTERING THE CORE, FOR EXAMPLE: VENTS FROM POTENTIAL GAS POCKETS, LOW COVER GAS PRESSURE, IHX CONTINUOUS BLEED, VORTEX SUPPRESSOR PLATE

- THE THERMAL CONSEQUENCES OF EVEN A LARGE BUBBLE (4 INCHES HIGH-OVER 8 ROWS: 3 CU. FT.) ARE SMALL (68°F MAXIMUM CLADDING TEMPERATURE INCREASE)

- THE REACTIVITY CONSEQUENCES OF ISOLATED BUBBLES ARE NEGLIGIBLE

- EVEN A 5 INCH BUBBLE WOULD ONLY INCREASE CLAD TEMPERATURE BY 25°F AT CORE EXIT

CONCLUSION: THERE ARE NO PROPAGATIVE CONSEQUENCES DUE TO BUBBLES

4. SUMMARY OF MOLTEN FUEL ISSUES

- MOLTEN FUEL (M.F.) DOES NOT EXIST IN REACTOR DURING NORMAL, UPSET OR EMERGENCY EVENTS
- M.F. NOT FORMED BEHIND PASSIVE PLANAR BLOCKAGES
- M.F. NOT FORMED IN HEAT GENERATING BLOCKAGES WHICH ARE NON-DETECTABLE
- TESTS D1, D2 DEMONSTRATED TRANSIENT OPERATION WITH M.F.
- SLSF TEST P1 DEMONSTRATED CONTINUED OPERATION WITH SIGNIFICANT AREAL M.F.

WORLD-WIDE EXPERIENCE WITH LIQUID METAL/OXIDE FUELED PLANTS IS EXTENSIVE, AND DEMONSTRATES NON-PROPAGATIVE CONSEQUENCES OF FUEL FAILURE.

WORLD WIDE OXIDE FUEL *

OPERATING EXPERIENCE

(APPROXIMATE NUMBERS)

COUNTRY	REACTOR	OXIDE USE	NO. RODS IRRADIATED	NO. RODS FAILED
FRANCE	PHENIX	DRIVER+	100,000	5
	RAPSODIE	DRIVER TEST	16,000 2,000	36**
UK	PFR	DRIVER+	30,000+	2
	DFR	TEST	2,500	200+ ++
USA	FFTF	DRIVER+	16,000	1
	EBR II	TEST	3,000	55**
JAPAN	JOYO	DRIVER TEST	5,000	0
USSR	BR-5, 10	DRIVER TEST	5,000	100+
	BOR-60		12,000	<60
	BN-350		50,000	
	BN-600			

* (U, Pu) O_2

** (INCLUDES 1/3 TO 1/2 WHICH FAILED AT B.U. $\geq$ 10%)

\+ SOME RODS MAY BE DESIGN PROOF TESTS

++ MOST FAILURES RELATED TO DOWN-FLOW COOLANT IN CORE

CONCLUSIONS FOR LOCAL FAULTS

- OPERATING PLANT FUEL FAILURE RATES LOW
- STOCHASTIC FUEL ROD FAILURE DOES NOT LEAD TO FAILURE PROPAGATION
- FAILED FUEL IS DETECTABLE
- CONTINUED OPERATION WITH FAILED FUEL HAS NOT CAUSED PROPAGATION
- DEGRADATION PROCESSES ARE SLOW AND CAN BE MONITORED WITH REMOVAL AT PREDETERMINED OPERATING LIMITS

THEREFORE, THE CONSEQUENCES OF FUEL FAILURE ARE BENIGN.

CRBR Core Exit Thermocouples

R. A. Markley

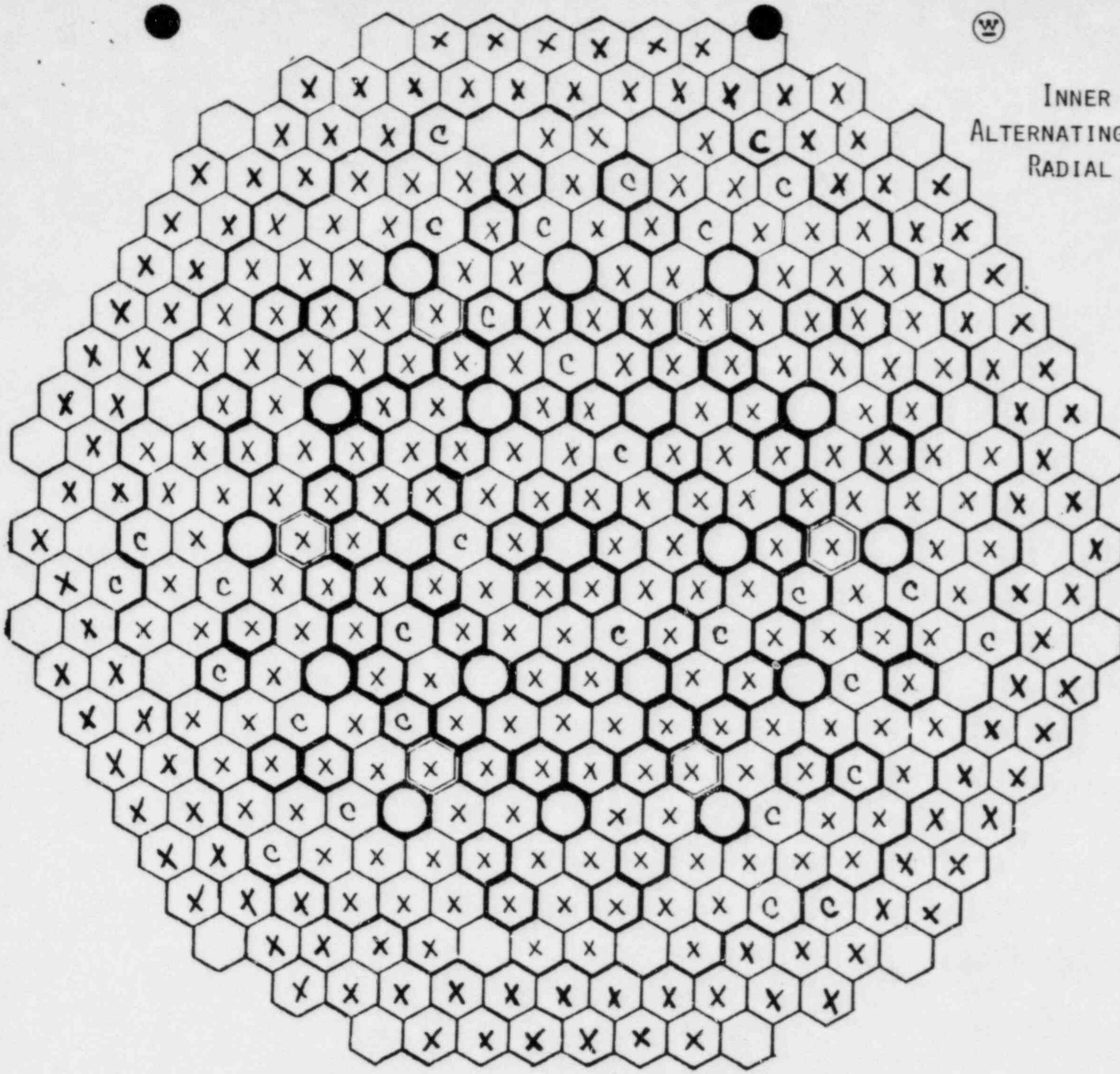
ACRS Subcommittee Meeting

March 16, 1983

CRBRP CORE EXIT THERMOCOUPLES

IDENTIFIED FUNCTIONS

- Reactor Control
- Design Verification
  - -- Design Margins, Power Distribution, Symmetry, Fuel Rod Lifetime
  - -- Operational Disturbances (Local Disturbances in Power Distribution, Shifts in Power/Flow, Etc.)
  - -- Uncertainty Factors Assessment
  - -- Relationship with Post-Irradiation Measurements



COVERAGE

FUEL	148	OUT OF	156
INNER BLANKET	72	"	76
ALTERNATING ASSYS.	6	"	6
RADIAL BLANKET	112	"	126
TOTAL	338		364

C = CONTROL
X = SURVEILLANCE/ DESIGN VERIFICATION

CRBRP CORE EXIT THERMOCOUPLES COVERAGE

REACTOR THERMOCOUPLES FOR AUTOMATIC CONTROL

REQUIREMENTS:

- Maintain Steady State Primary Outlet Temperature Within Fixed Band
- Minimize Temperature Overshoot

Coverage - 30 Positions Over 3 Symmetric 60° Sectors

RATIONALE:

- Closely Approach Core Mixed Mean and Follow Cycle Swing
- Even Coverage
- Provide Redundancy

Type - Dry-Well

Time Constant <10 Seconds

TEMPERATURE MEASUREMENT UNCERTAINTIES

FUEL ASSEMBLY

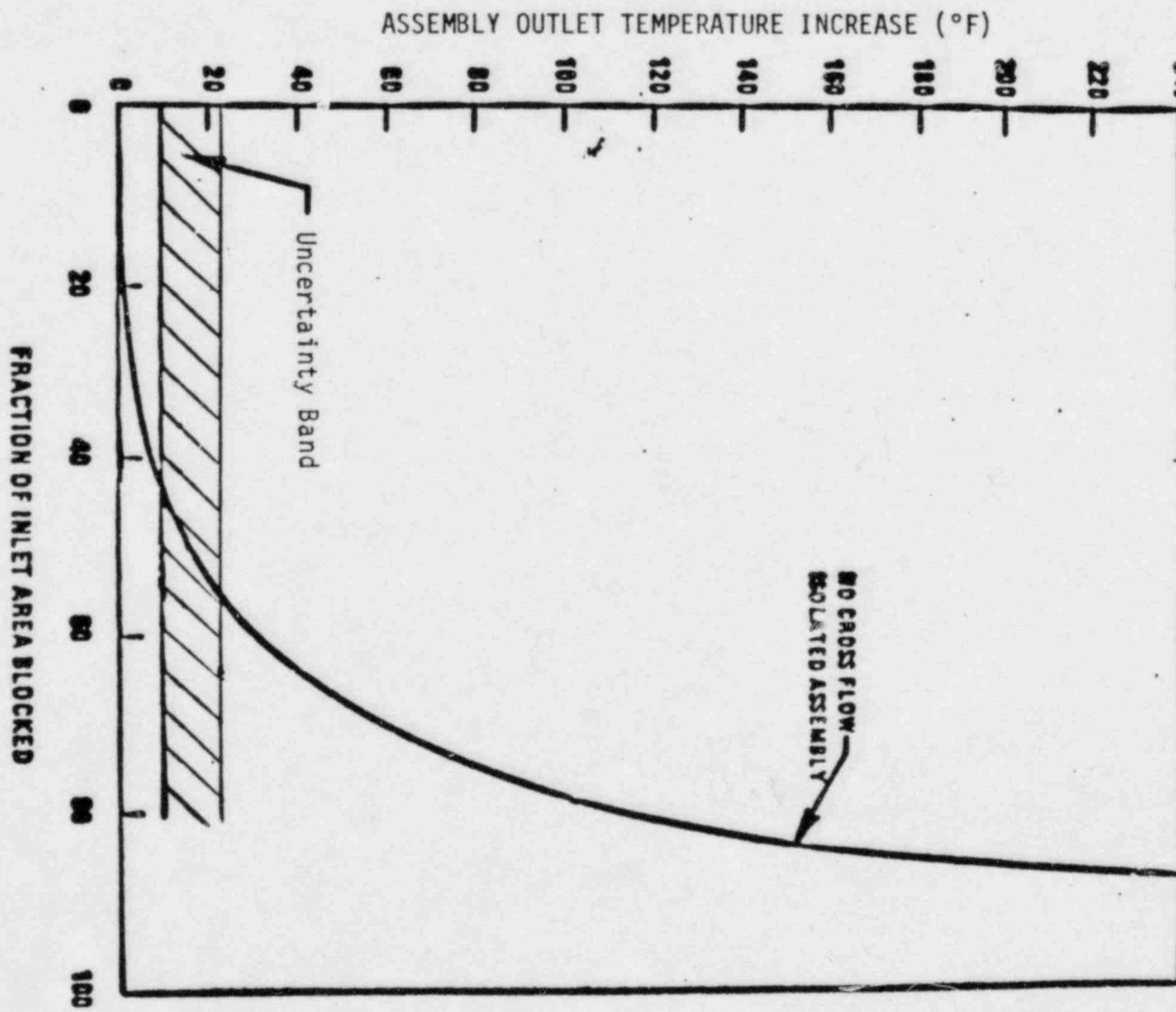
RANGE FROM ~8°F (CORE CENTER, SIX-FOLD COVERAGE BY SYMMETRY) TO ~25°F (PERIPHERY, SINGLE COVERAGE)

INNER BLANKET ASSEMBLY

RANGE FROM ~15°F (HIGH FLOW, SYMMETRY) TO ~40°F (LOW FLOW, SINGLE COVERAGE)

CORE OUTLET THERMOCOUPLES ARE NOT SAFETY RELATED

- For "Core-Wide" Faults:
  - -- Fast Acting PPS Trips (e.g. high flux, flux to flow) Shutdown Before Excessive Temperatures
- For Local Faults (Blockage, High Heat Flux)
  - -- T/C Insensitive to Local Faults (e.g. Blockages <50 to 60%)
  - -- T/C "Blinded" by Adjacent Cross Flow When Blockage ~ >90%
  - -- Low Probability of Occurrence
  - -- If Local Temperature Increase Causes Clad Failure, Detected By Failed Fuel Monitoring System
- Limited Number of Clad Failures Are Expected and Are Not a Safety Concern
  - -- Rapid Propagation Will Not Occur


ASSEMBLY OUTLET TEMPERATURE INCREASE (°F)
0
20
40
60
80
100
120
140
160
180
200
220
240
Uncertainty Band
NO CROSS FLOW
ISOLATED ASSEMBLY
FRACTION OF INLET AREA BLOCKED
0
20
40
60
80
100

13

CORE OUTLET THERMOCOUPLE

ALARM, ALERTS, ETC.

PRESENTATION TO

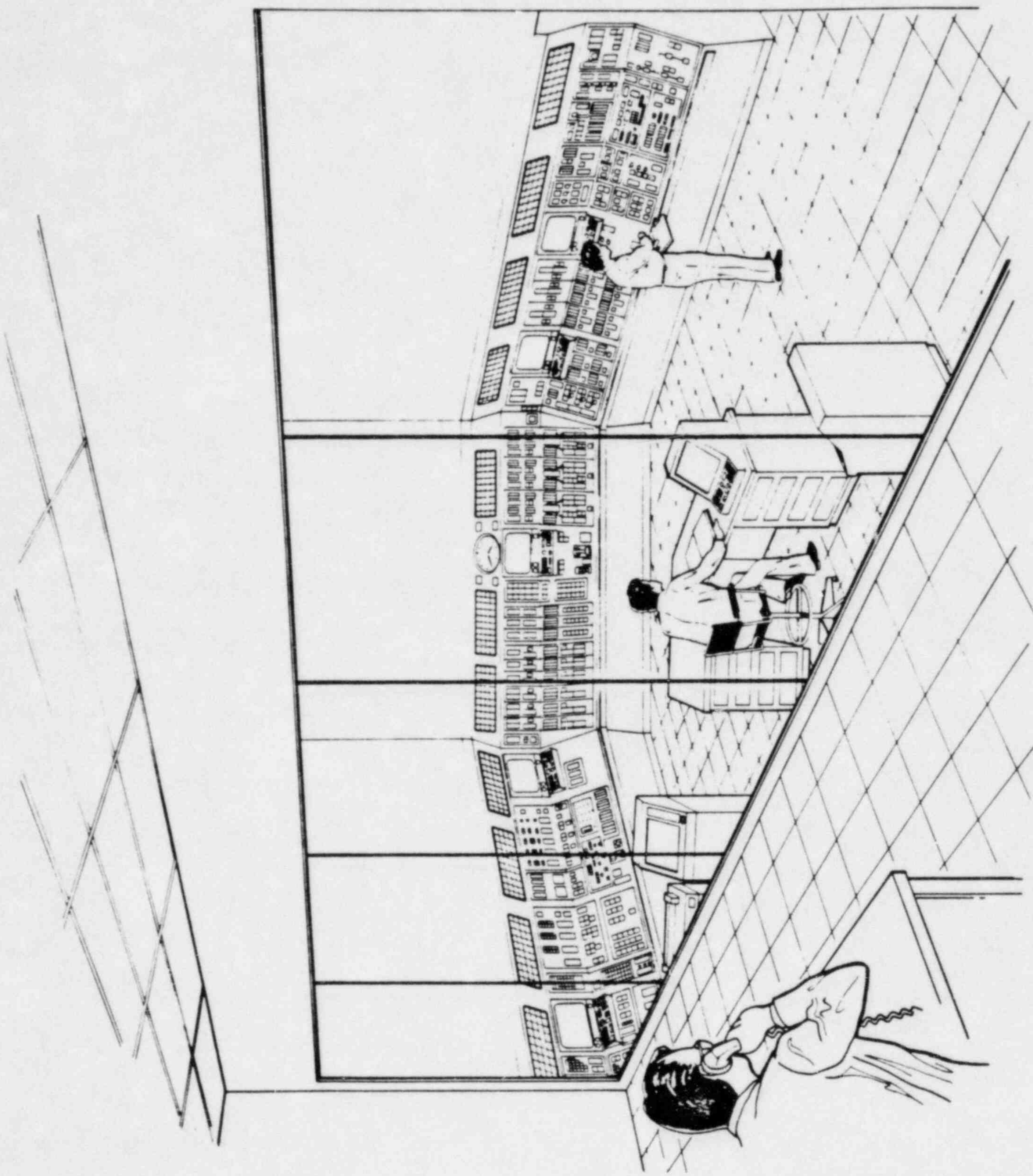
ACRS

MARCH 16, 1983

BY

R. J. TINDER, WESTINGHOUSE

ADVANCED REACTORS DIVISION



WARD

CORE THERMOCOUPLE INDICATIONS AND ALARMS

- 338 FUEL AND BLANKET THERMOCOUPLES
- EACH WIRED TO PLANT DATA HANDLING AND DISPLAY SYSTEM
- COMPARED TO ALGORITHM OF HISTORY-POWER-FLOW-LOCATION
- COMPUTER "ALERT" WHEN OUTSIDE TOLERENCE
- ALERT TYPES ON ALARM TYPEWRITER
- ALERT APPEARS ON ALARM CRT
- LAST THREE ALERTS APPEAR AT BOTTOM OF CRTs
- OPERATOR CAN REQUEST T/C READINGS ANY TIME (CRT OR HARDCOPY)
- SELECTED 30 T/C FOR CONTROL SYSTEM
- LESS THAN 25 CONTROL T/C ACTIVATES MAIN PLANT ANNUNCIATOR (AUDIBLE AND VISUAL)

Review Of World-Wide Application Of Local Fault Instrumentation

L. E. Strawbridge

Westinghouse Advanced Reactors Division

Scope Of Survey

- Available Design Information Collected On 17 Sodium Cooled Fast Reactors.
- Major Plant Design Variations
  - Loop Or Pool
  - Power Or Test
  - Small To Large
- Major Fuel Design Variations
  - Oxide Or Metal Fuel
  - Wire Wrap Or Grid Spacers

INSTRUMENTATION SURVEY

	RAPSODIE	PHENIX	SPX	DFR	PFR	CDFR	SNR-300	JOYO	MONJU
LOOP OR POOL	LOOP	POOL	POOL	LOOP	POOL	POOL	LOOP	LOOP	LOOP
POWER OR TEST	P,T	P	P	P,T	P	P	P	T	P
FUEL/SPACER	OX/WW	OX/WW	OX/WW	M/	OX/G	OX/G	OX/G		OX/
CORE EXIT T/C	YES	YES	YES		YES	YES	YES	YES	YES
T/C IN PPS		YES	YES		YES	YES	A	NO	YES

DEFINITIONS:

WIRE WRAP - WW   OXIDE - OX   ALARM - A

GRID - G   METAL - M

INSTRUMENTATION SURVEY (CONTINUED)

	BR-5/10	BOR 60	BN 350	BN 600	EBR II	FERMI	FFTF	CRBRP
LOOP OR POOL	LOOP	LOOP	LOOP	POOL	POOL	LOOP	LOOP	LOOP
POWER OR TEST	P,T	P,T	P	P	P,T	P	T	P
FUEL/SPACER	OX/	OX/	OX/	OX/	M/WW	M/G	OX/WW	OX/WW
CORE EXIT T/C					YES	YES	YES	YES
T/C IN PPS					NO	NO	NO	NO

DEFINITIONS:

WIRE WRAP - WW　　OXIDE - OX　　ALARM - A

GRID - G　　METAL - M

Survey Of Outlet Thermocouples In PPS For Local Faults

- Strongest Correlation Exists For Loop vs. Pool

	# Plants	T/C In PPS		
		Yes	No	Unknown
Loop	11	1	5	5
Pool	6	4	1	1

Conclusions On World-Wide Application Of Local Fault Instrumentation

- No Universal Agreement On Approach.
- Core Outlet Thermocouples Usually Included In PPS In Pool Reactors; Usually Not Included In Loop Reactors.
- Considering The CRBRP Design, The Application Of Instrumentation Is Consistent With The World-Wide Trends.

Conclusions On Local Faults

- QA Results In Low Probability Of Loading Defective Fuel.
- Design Features And Operational Requirements Prevent Ex-Core Blockages.
- No In-Core Blockage Expected Based On Operation, Tests And Analysis.
- Fuel Failures During Operation Are Anticipated And Will Be Detected.
- Extensive Experience Base Shows No Evidence Of Progagation.
- Thermocouples Do Not Significantly Improve The Margin Of Safety For Local Faults
  - Not Effective For Prevention/Forewarning
  - DNDs More Sensitive For Detecting Failures Of Interest
- Considering The CRBRP Design, The Application Of Instrumentation Is Consistent With The World-Wide Trends.
- Instrumentation For Local Fault Monitoring Is Not Needed In PPS Since No Faults Could Propagate On A Time Scale Requiring PPS Action.

SOURCE INTERDICTION FOR DIRECT RELEASES TO THE HYDROSPHERE AFTER A SEVERE ACCIDENT AT THE CRBRP

S. J. NIEMCZYK

OAK RIDGE NATIONAL LABORATORY

MARCH 16, 1983

WASHINGTON, D.C.

SUMMARY

IT IS NOT EXPECTED THAT THE CRBRP BASEMAT WILL MELT THROUGH.

IF IT DOES AND NO INTERDICTIVE ACTIONS ARE TAKEN, IT IS ESTIMATED THAT THE RESULTING CONSEQUENCES TO THE PUBLIC WILL BE RELATIVELY SMALL.

IF IT IS DECIDED THAT MITIGATING MEASURES ARE INDICATED, THEN IT SHOULD BE POSSIBLE AT THE CRBRP SITE TO ALMOST COMPLETELY PREVENT ANY RADIATION DOSE TO THE PUBLIC BY APPROPRIATE SOURCE INTERDICTIVE PROCEDURES.

PRIMARY BASIS FOR DISCUSSION: NUREG/CR-1596 (SAND80-1669)

"THE CONSEQUENCES FROM LIQUID PATHWAYS AFTER A REACTOR MELTDOWN ACCIDENT"

THAT STUDY

1. CONSIDERED ALL OPERATING AND UNDER-CONSTRUCTION LWRs PLUS SOME PROPOSED ONES

2. ESTIMATED THE CONSEQUENCES OF RELEASES TO THE HYDROSPHERE BOTH WITH AND WITHOUT INTERDICTIVE PROCEDURES

3. CONSIDERED THE FEASIBILITY AND THE EFFECTIVENESS OF BOTH SOURCE AND PATHWAY INTERDICTIVE PROCEDURES

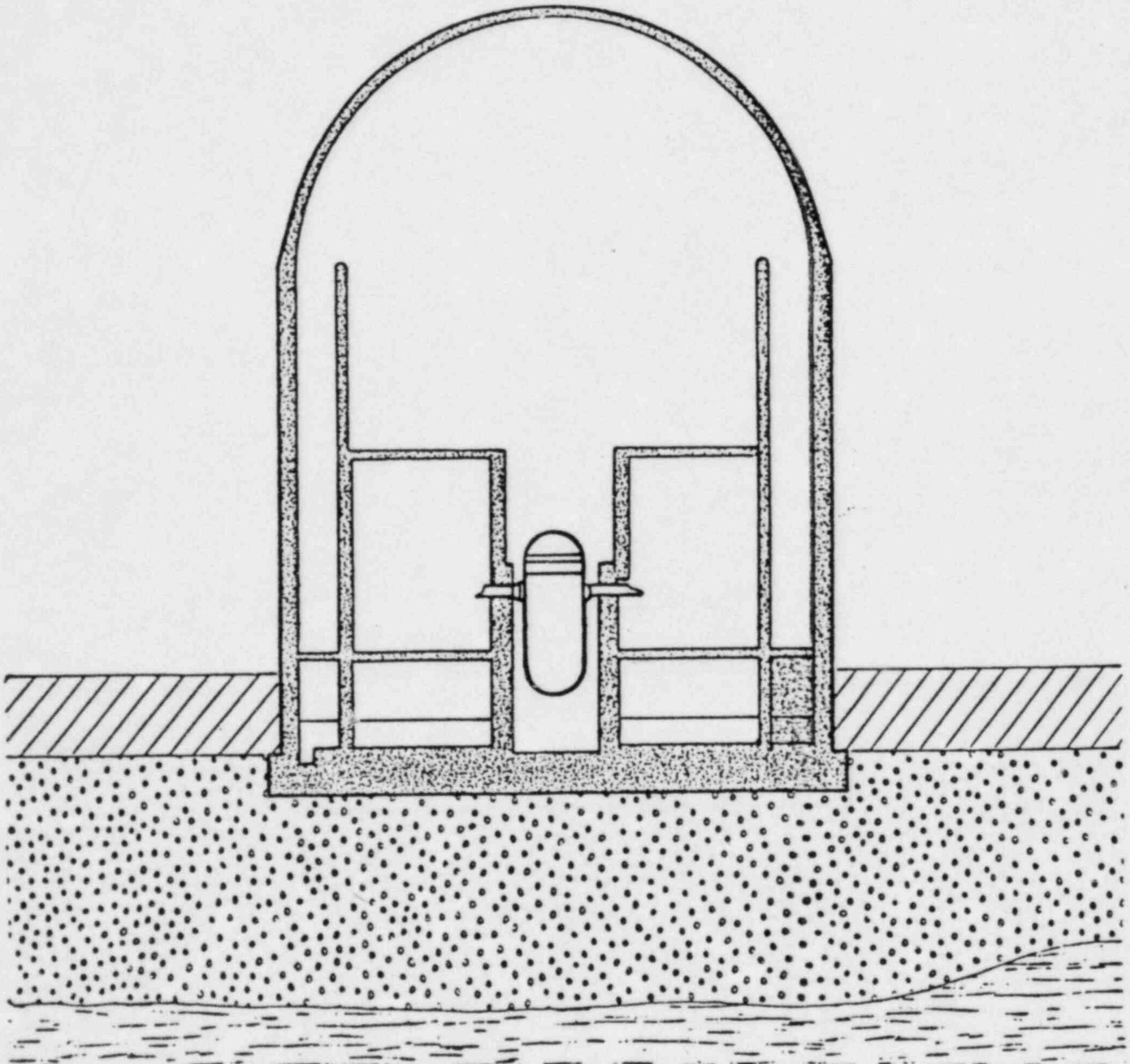
ACCORDING TO THAT STUDY, IF BASEMAT MELT-THROUGH OCCURS,

1. EVEN WITHOUT MITIGATING PROCEDURES, THE CONSEQUENCES OF RELEASES TO THE HYDROSPHERE ARE NOT EXPECTED TO BE SIGNIFICANT FOR MANY LWR SITES

2. HOWEVER, FOR SOME SITES, THE CONSEQUENCES OF SUCH RELEASES MAY BE SIGNIFICANT UNLESS ADEQUATE MITIGATING PROCEDURES ARE TAKEN

3. ALTHOUGH ADEQUATE MITIGATING PROCEDURES CAN PROBABLY BE TAKEN AT MOST LWR SITES, IT IS NOT OBVIOUS THAT THAT THEY CAN BE TAKEN AT ALL SITES

4. ONLY MITIGATING PROCEDURES TAKEN CLOSE TO THE SOURCE CAN TYPICALLY BE VERY EFFECTIVE IN REDUCING THE POTENTIAL RADIATION DOSE TO THE HUMAN POPULATION



MAJOR TYPES OF DIRECT RELEASES TO HYDROSPHERE

LWR

1. LEACHING OF MELT DEBRIS

2. ESCAPE OF SUMPWATER (OR SUPPRESSION POOL WATER)

CRBRP

1. LEACHING OF MELT DEBRIS

CHARACTERISTICS OF RELEASES

MELT DEBRIS

1. COMPOSED MOSTLY OF CONCRETE/SOIL RESIDUA

2. INCLUDES PRIMARILY LESS VOLATILE RADIONUCLIDES

3. RELEASE TO ACCESSIBLE PORTIONS OF THE ENVIRONMENT GENERALLY WILL BE RELATIVELY SLOW

SUMPWATER

1. MAY CONTAIN LARGE FRACTIONS OF MANY OF THE MORE VOLATILE RADIONUCLIDES

2. RELEASE TO THE ACCESSIBLE ENVIRONMENT MAY BE RELATIVELY RAPID

CRBRP SOURCE TERM VERSUS "TYPICAL" LWR SOURCE TERM

1. THICKER BASEMAT IN CRBRP; THEREFORE BASEMAT LESS LIKELY TO MELT THROUGH

2. SMALLER CORE IN CRBRP; THEREFORE, LESS FISSION PRODUCT ACTIVITY IS AVAILABLE FOR RELEASE TO HYDROSPHERE; ALSO, BASEMAT MELT-THROUGH IS LESS LIKELY

3. NO MAJOR RELEASES OF HIGHLY CONTAMINATED WATER FROM CRBRP; THEREFORE EARLY RADIONUCLIDE RELEASES, AS WELL AS RELEASES OF VOLATILE RADIONUCLIDES, ARE MUCH LESS LIKELY

4. MORE PLUTONIUM IN CRBRP; THEREFORE LARGER LONG-TERM SOURCE POSSIBLE

POTENTIAL CONSEQUENCES OF RELEASES TO THE HYDROSPHERE DEPEND ON

1. RATE OF INITIAL RELEASES TO HYDROSPHERE

   A. TIME UNTIL GROUNDWATER CONTACT ESTABLISHED

   B. RATES OF LEACHING

2. RATE OF TRANSPORT TO ACCESSIBLE PORTIONS OF ENVIRONMENT

   A. TRAVEL TIME OF GROUNDWATER FROM SITE TO SURFACE WATER

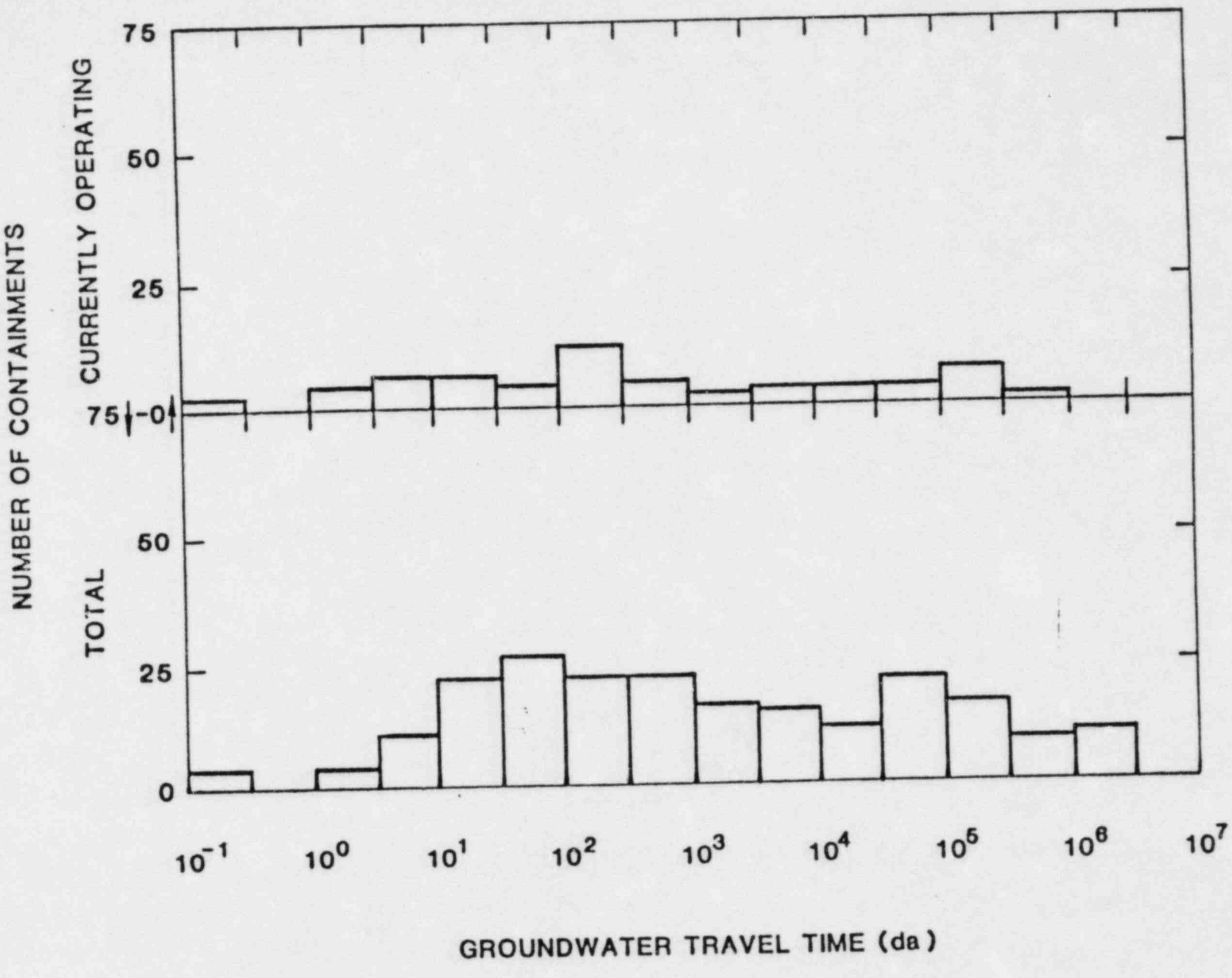
   B. STRENGTHS OF ADSORPTION OF RADIONUCLIDES TO SOIL

3. PRESENCE OF EXPOSURE PATHWAYS

   A. SIZES OF POPULATIONS-AT-RISK

   B. USAGE PATTERNS OF POPULATIONS-AT-RISK

   C. CHARACTERISTICS OF PATHWAYS



NUMBER OF CONTAINMENTS
CURRENTLY OPERATING
TOTAL
75
50
25
75 -0
50
25
0
10^{-1}
10^{0}
10^{1}
10^{2}
10^{3}
10^{4}
10^{5}
10^{6}
10^{7}
GROUNDWATER TRAVEL TIME (da)

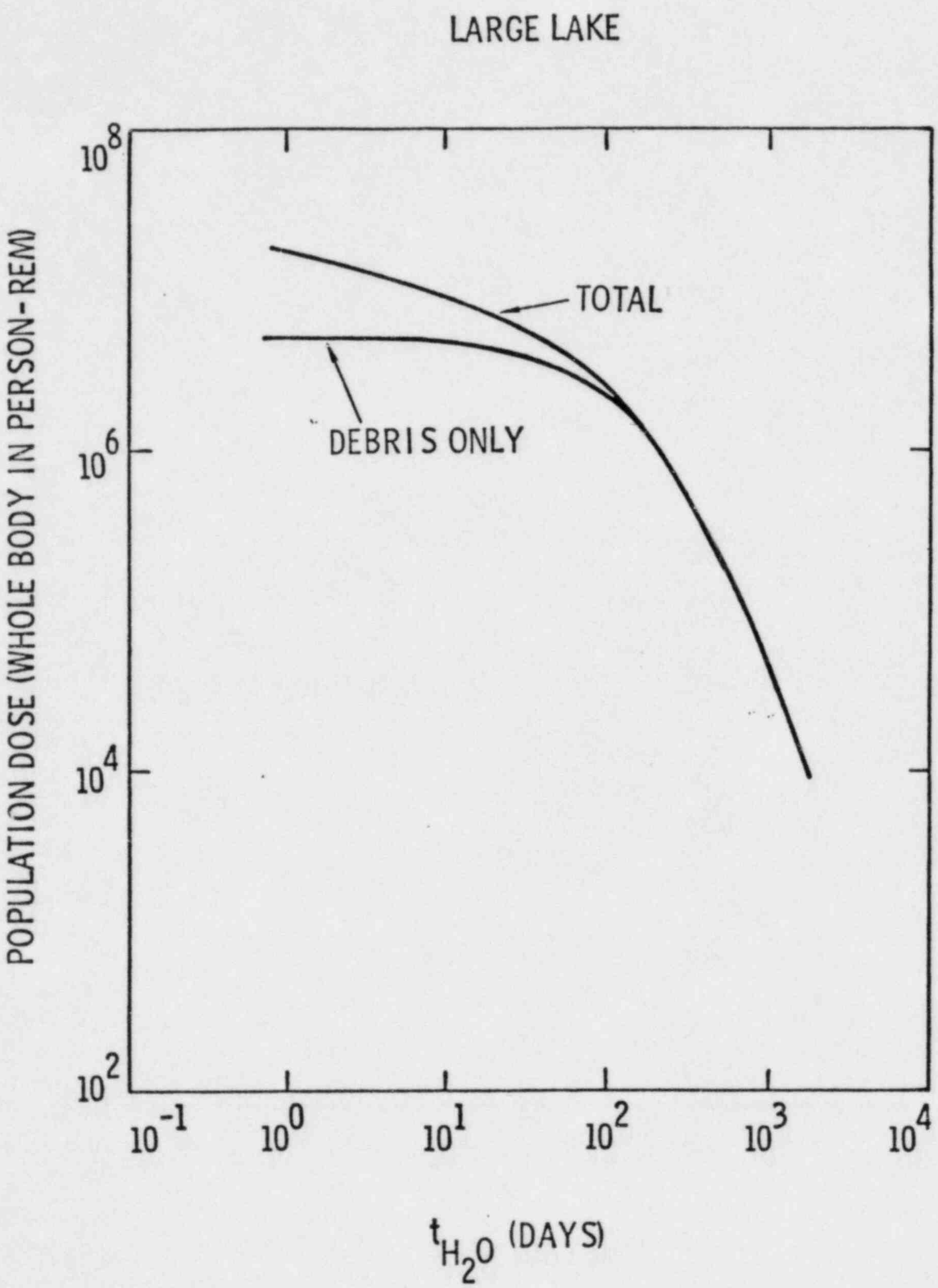

LARGE LAKE
POPULATION DOSE (WHOLE BODY IN PERSON-REM)
10^8
10^6
10^4
10^2
TOTAL
DEBRIS ONLY
10^{-1}
10^0
10^1
10^2
10^3
10^4
t_{H_2O} (DAYS)

Figure 2.5-3A
Topographic Map in the Vicinity of the CRBRP Site

KNOX GROUP
SCALE IN FEET
KNOX GROUP
LOWER SILTSTONE
LIMESTONE
CHICKAMAUGA UNIT A
CHICKAMAUGA UNIT A
UPPER SILTSTONE
COOLING TOWER
CHICKAMAUGA UNIT A
CHICKAMAUGA UNIT B
CHICKAMAUGA UNDIFFERENTIATED
CHICKAMAUGA GROUP
FAULT
D
U
ROME FORMATION
CHICKAMAUGA GROUP
WATTS BAR LAKE
CREEK

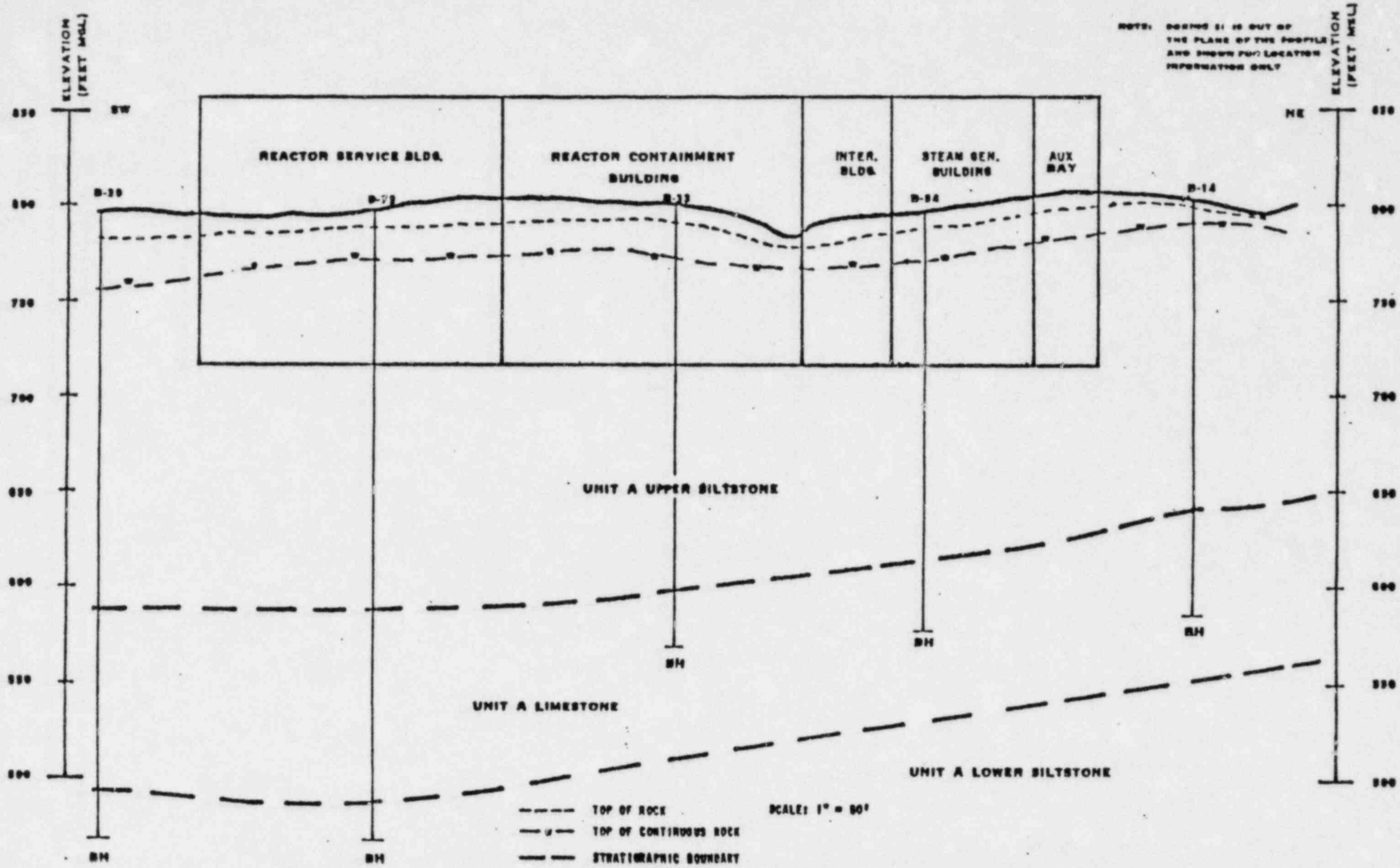


Figure 2.5-25 Nuclear Island Subsurface Profile A-A

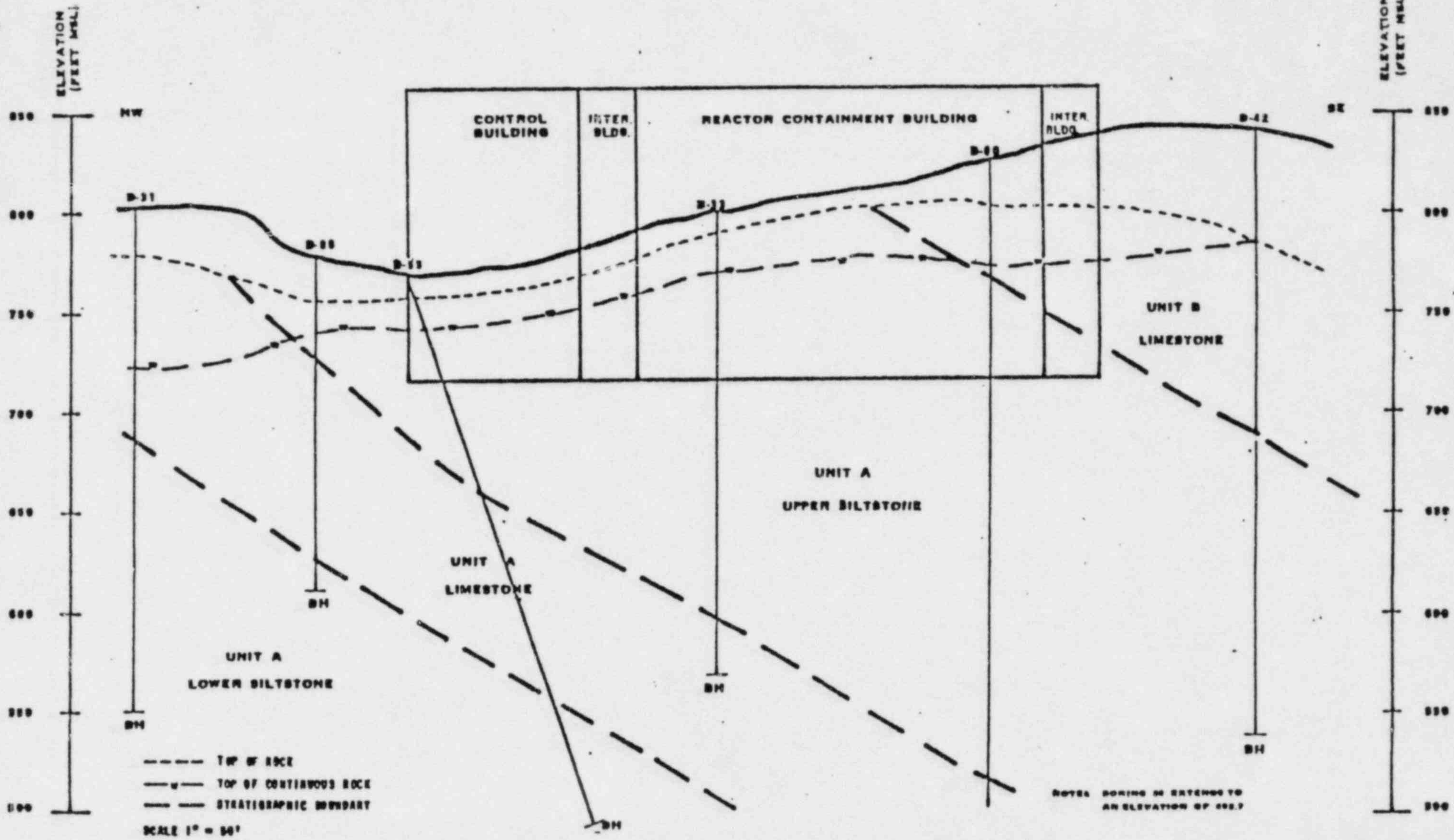


Figure 2.5-26 Nuclear Island Subsurface Profile B-B

POTENTIAL PATHWAYS FROM DEBRIS TO ACCESSIBLE ENVIRONMENT

INITIALLY,

1. ESCAPE VIA TRANSPORT THROUGH SILTSTONE
2. ESCAPE VIA CONSTRUCTION CHANNELS
3. ESCAPE THROUGH FISSURES, ...
4. ESCAPE THROUGH DEGRADED SILTSTONE

LATER,

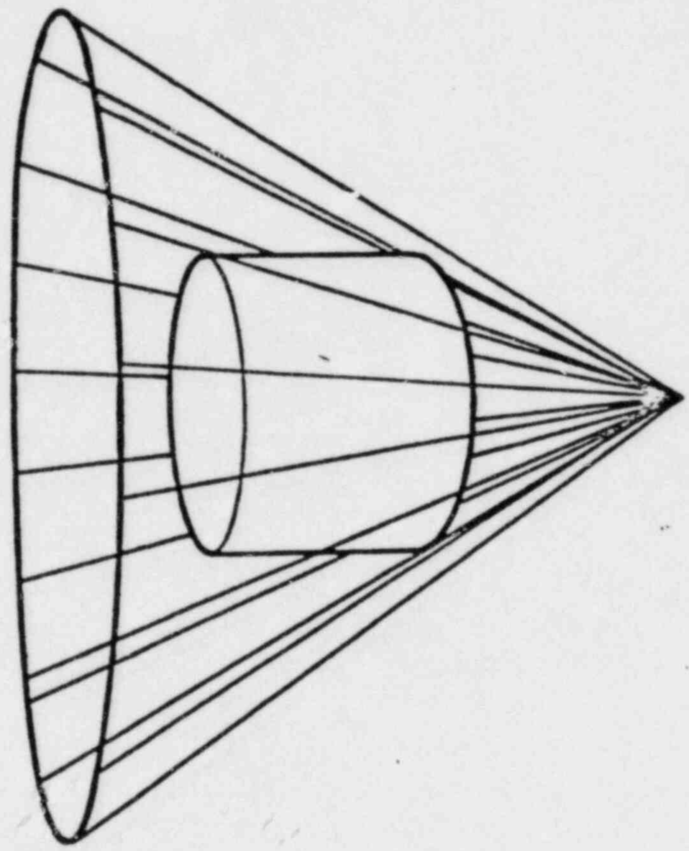
1. TRANSPORT THROUGH UNWEATHERED SILTSTONE
2. TRANSPORT THROUGH UNWEATHERED LIMESTONE
3. TRANSPORT THROUGH WEATHERED SILTSTONE
4. TRANSPORT THROUGH WEATHERED LIMESTONE

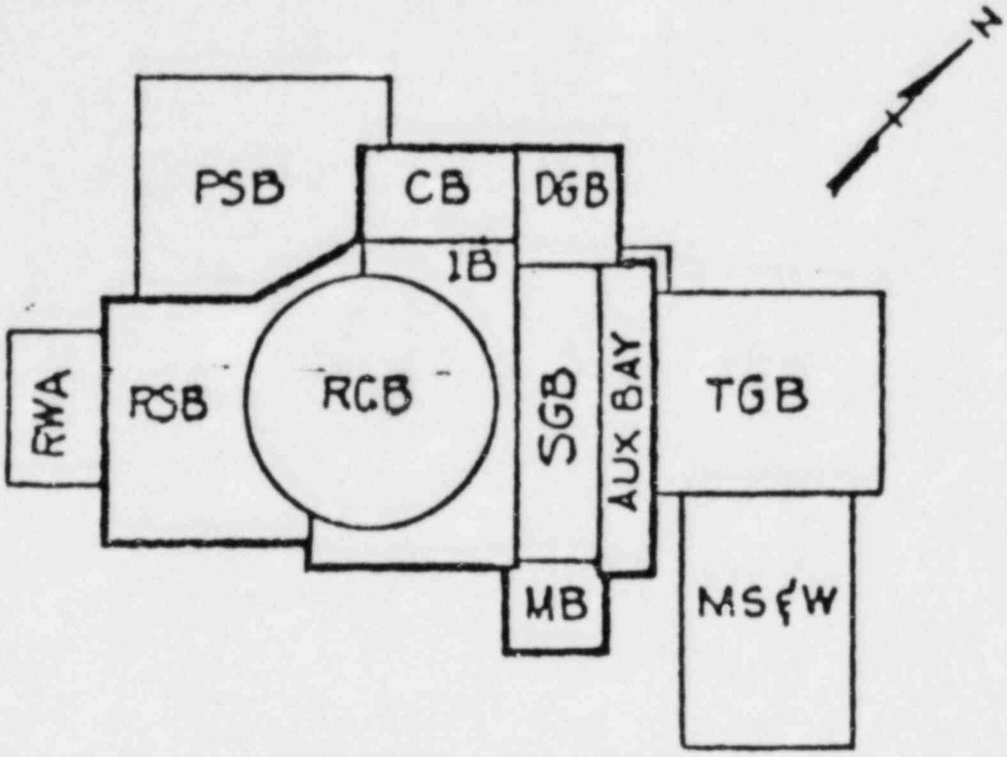
FOR TRANSPORT THROUGH WEATHERED STRATA, ESCAPE TO THE GROUND SURFACE AND TRANSPORT ACROSS IT (PRIOR TO REACHING SURFACE WATER) WOULD ALSO BE A POSSIBILITY.

POSSIBLE "TEMPORARY" SOURCE INTERDICTIVE PROCEDURES

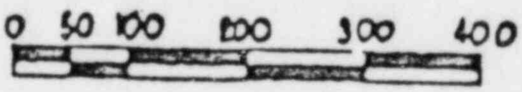
1. DEWATERING
2. WATER INJECTION
3. RECOVERY
4. GROUND FREEZING
5. SLURRY TRENCH CONSTRUCTION
6. GROUTING
7. ...

ANY OF THE ABOVE WOULD NEED TO BE COMBINED WITH AN EXTENSIVE MONITORING PROGRAM.





GRAPHIC SCALE



RCB REACTOR CONTAINMENT BLDG
SGB STEAM GENERATOR BLDG
IB INTERMEDIATE BAY
RSB REACTOR SERVICE BLDG
CB CONTROL BLDG
DGB DIESEL GENERATOR BLDG
TGB TURBINE GENERATOR BLDG
PSB PLANT SERVICE BLDG
MS&W MAINTENANCE SHOP & WAREHOUSE
MB MAINTENANCE BAY
RWA RADWASTE AREA

EFFORT REQUIRED FOR ISOLATION OF DEBRIS BY GROUTING

1. CONVENE EXPERTS AND FORMULATE PLAN
2. CLEAN UP SITE
3. DETERMINE SUBSURFACE CONDITIONS
4. RAZE EXTRANEOUS BUILDINGS, ...
5. ASSEMBLE DRILLS, TEAMS, GROUT, ...
6. FORM INITIAL GROUT BARRIER
7. FORM SECOND BARRIER
8. ADD INTERMEDIATE BARRIER TO ENSURE ISOLATION
9. COVER SURFACE

"PERMANENT" SOLUTION

1. REENFORCE GROUTING

   A. SOLUTIONING IN LIMESTONE LAYERS MAY CAUSE TROUBLE

   B. SETTLEMENT, ... OF GROUND MAY CAUSE TROUBLE

   C. GROUTING AND ITS REENFORCEMENT MAY ULTIMATELY DETERIORATE

2. MOVE DEBRIS TO PERMANENT REPOSITORY

   A. LEGAL PROBLEMS CURRENTLY PRECLUDE THIS

   B. ACCESS TO DEBRIS MAY PERMIT ESCAPE OF RADIOACTIVITY TO ENVIRONMENT

GOOD FEATURES OF CRBR PLANT AND SITE

1. THE CONTAINMENT BUILDING WOULD HAVE A RELATIVELY THICK BASEMAT; THEREFORE, BASEMAT MELT-THROUGH WOULD NOT BE LIKELY.

2. THE CORE WOULD BE COMPARATIVELY SMALL; THEREFORE, ONLY RELATIVELY SMALL AMOUNTS OF MOST RADIONUCLIDES COULD BE RELEASED TO THE HYDROSPHERE; IN ADDITION, BASEMAT MELT-THROUGH WOULD NOT BE LIKELY.

3. NO SIGNIFICANT RELEASES OF HIGHLY CONTAMINATED WATER WOULD OCCUR; THEREFORE FEW IF ANY VOLATILE RADIONUCLIDES WOULD BE RELEASED DIRECTLY TO THE HYDROSPHERE; FURTHERMORE, ANY RELEASES OF RADIOACTIVITY WHICH DID OCCUR WOULD BE RELATIVELY SLOW.

4. IF BASEMAT MELT-THROUGH OCCURRED, THE MELT WOULD RESIDE IN SILTSTONE; THEREFORE ONLY LIMITED WATER MIGHT BE AVAILABLE FOR LEACHING.

5. THE BEDROCK BENEATH THE PLANT WOULD BE RELATIVELY UNDISTURBED BY CONSTRUCTION; THEREFORE, ONLY LIMITED WATER MIGHT BE AVAILABLE FOR LEACHING.

6. THE SITE IS A LONG WAY FROM THE RIVER; THEREFORE, IF INITIAL EFFORTS TO CONTAIN THE RADIOACTIVITY FAILED, THERE WOULD BE TIME TO CARRY OUT ADDITIONAL EFFORTS TO PREVENT THE RADIOACTIVITY FROM REACHING THE RIVER.

SUMMARY

IT IS NOT EXPECTED THAT THE CRBRP BASEMAT WILL MELT THROUGH.

IF IT DOES AND NO INTERDICTIVE ACTIONS ARE TAKEN, IT IS ESTIMATED THAT THE RESULTING CONSEQUENCES TO THE PUBLIC WILL BE RELATIVELY SMALL.

IF IT IS DECIDED THAT MITIGATING MEASURES ARE INDICATED, THEN IT SHOULD BE POSSIBLE AT THE CRBRP SITE TO ALMOST COMPLETELY PREVENT ANY RADIATION DOSE TO THE PUBLIC BY APPROPRIATE SOURCE INTERDICTIVE PROCEDURES.

Planchon

CRBRP ACCIDENT RESPONSE PLANNING

BRIEFING FOR:

ADVISORY COMMITTEEE ON REACTOR SAFEGUARDS (ACRS) CRBRP SUBCOMMITTEE & MATERIALS AND STRUCTURE WORKING GROUP

ON:

DESIGN FOR EMERGENCY MANAGEMENT

PRESENTED BY:

PETE PLANCHON
MANAGER, PLANT SYSTEMS
AND SAFETY RELATED DESIGNS
CRBRP PROJECT
WESTINGHOUSE—AESD
OAK RIDGE, TN.

MARCH 16, 1983

3 83 3275 1

CRBRP DESIGN FOR EMERGENCY MANAGEMENT

- INTRODUCTION
- ON-SITE EMERGENCY CONTRCL CENTERS
  - CONTROL ROOM
  - TECHNICAL SUPPORT CENTER
  - OPERATIONAL SUPPORT CENTER
- ACCIDENT MONITORING INSTRUMENTATION
- EMERGENCY RESPONSE DATA SYSTEM

CRBRP CONTROL ROOM ARTIST CONCEPT

COMMAND AND CONTROL OF OPERATIONS

- PROVIDE THE CONTROL ROOM SUPERVISOR WITH INFORMATION THAT WILL SUPPORT COGNITIVE BEHAVIOR
- PROVIDE CONTROL ROOM OPERATORS WITH INFORMATION AND CONTROLS THAT WILL SUPPORT RULE AND SKILL BASED BEHAVIOR
- TO ACHIEVE THE ABOVE
  - PROVIDE A STATE-OF-THE-ART COMPUTER SYSTEM TO ENHANCE COGNITIVE FUNCTIONS OF THE SUPERVISOR
  - PROVIDE CONTROL ROOM AND CONTROL PANELS LAYOUT WHICH EMPHASIZES MAN-MACHINE INTERFACE CONSIDERATIONS

10-82-3036-4

ON-SITE EMERGENCY CONTROL CENTERS

CONTROL ROOM

- OVERVIEW ARRANGEMENT
- PROCESS CONTROLS INDICATIONS ON PANELS
  - AUTOMATIC AND MANUAL SAFETY CONTROLS
  - KEY ACCIDENT MONITORING INTEGRATED INTO PANEL
- PLANT COMPUTER
  - SAFETY PARAMETER DISPLAY SYSTEM (SPDS)
  - INTEGRATED GRAPHICS
  - SUPPORTS SUPERVISORS DECISIONS

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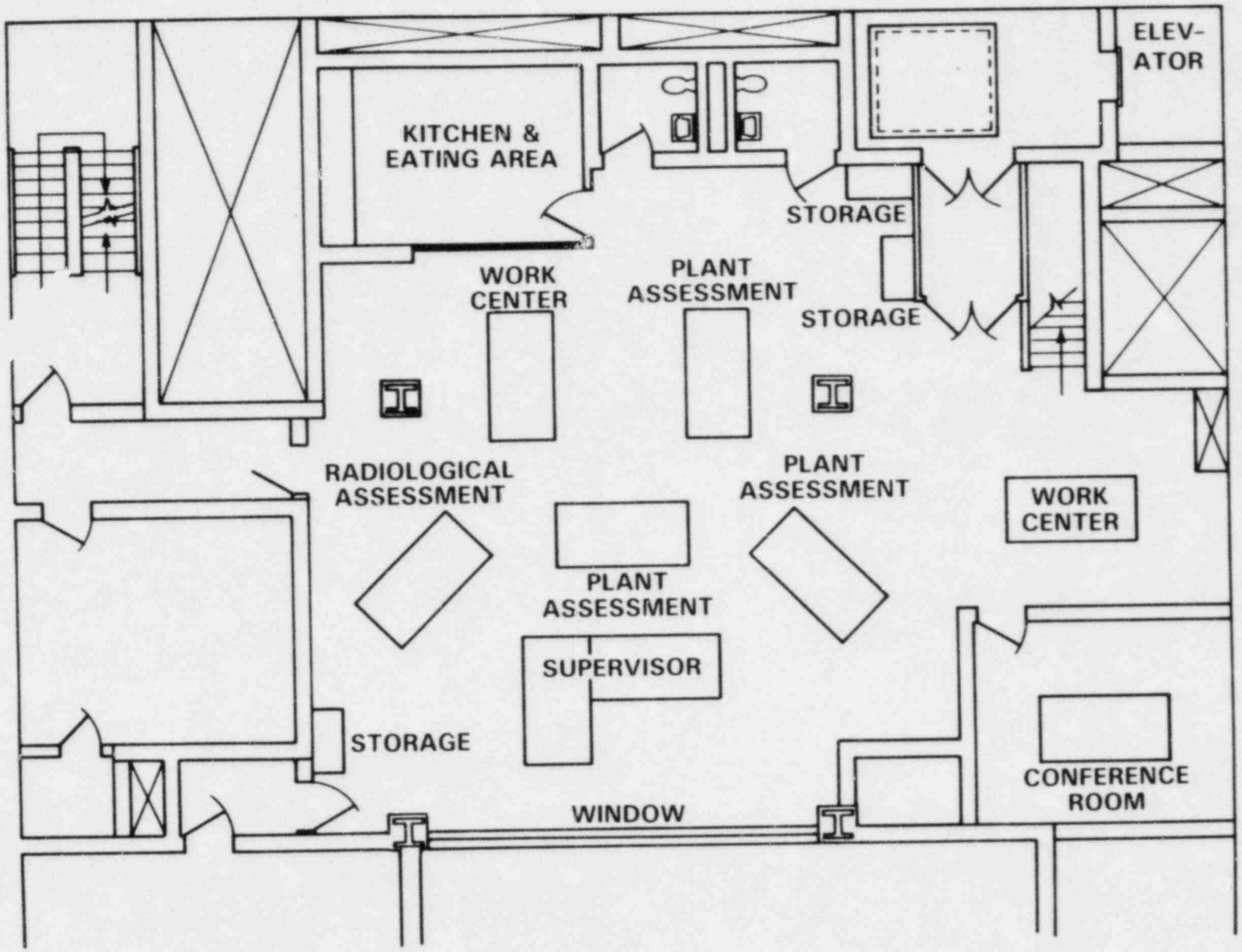
ON-SITE EMERGENCY CONTROL CENTERS

TECHNICAL SUPPORT CENTER

- LOCATED NEAR CONTROL ROOM
- PLACE WHERE PLANT MANAGEMENT AND TECHNICAL EXPERTS MONITOR CONDITIONS AND PROVIDE DIRECTION AND ADVICE
- CONTAINS EXTENSIVE INFORMATION SYSTEMS
  - PLANT COMPUTER
  - RADIATION MONITORING COMPUTER
- COMMUNICATIONS CENTER
  - IN-PLANT COMMUNICATIONS
  - EXTERNAL COMMUNICATIONS

TECHNICAL SUPPORT CENTER



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CLINCH RIVER BREEDER REACTOR PLANT

BRIEFING FOR

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS) CRBRP SUBCOMMITTEE & MATERIALS AND STRUCTURE WORKING GROUP

NSSS LOOSE PARTS MONITORING SYSTEM

PRESENTED BY:
W. J. O'BRYANT
MANAGER, MAINTENANCE AND TEST
CRBRP PROJECT
WESTINGHOUSE—AESD
OAK RIDGE, TN.

MARCH 16, 1983

3 83 3269 1

LOOSE PARTS MONITORING SYSTEM
INTRODUCTION AND BACKGROUND

- CRBRP PROJECT IS COMMITTED TO HAVING A LOOSE PARTS MONITORING SYSTEM CONSISTENT WITH LWR TECHNOLOGY, MODIFIED AS NECESSARY FOR CRBRP ENVIRONMENT.
- GENERAL DESIGN CRITERIA FOR LPMS WERE ESTABLISHED AND MUTUALLY AGREED UPON BY NRC AND THE CRBRP PROJECT IN 11/8/82 MEETING.

LOOSE PARTS MONITORING SYSTEM
GENERAL DESIGN CRITERIA

- SENSOR LOCATIONS WILL INCLUDE AS A MINIMUM, RV, PHTS PUMP, IHX, IHTS PUMP, SG MODULES, RV HEAD AND/OR UPPER PLENUM.
- SENSORS SHALL BE PROVEN STATE OF THE ART, CONSISTENT WITH LWR TECHNOLOGY, MODIFIED AS NECESSARY FOR CRBRP ENVIRONMENT AND SHALL BE REDUNDANT.
- SENSITIVITY (THRESHOLD ENERGY) SHALL BE ADEQUATE TO IDENTIFY ALL LOOSE PARTS THAT COULD POTENTIALLY RESULT IN DEGRADATION OF ABOVE COMPONENTS BY IMPACTING.
- SENSORS SHALL BE LOCATED TO DETECT LOOSE PARTS AT NATURAL COLLECTION POINTS FOR EACH OF THE ABOVE COMPONENTS.
- METHODS OF MOUNTING OF SENSORS SHALL BE EITHER BY DIRECT MOUNTING TO COMPONENTS/PIPING OR BY ATTACHMENT TO SUITABLE STANDOFFS.

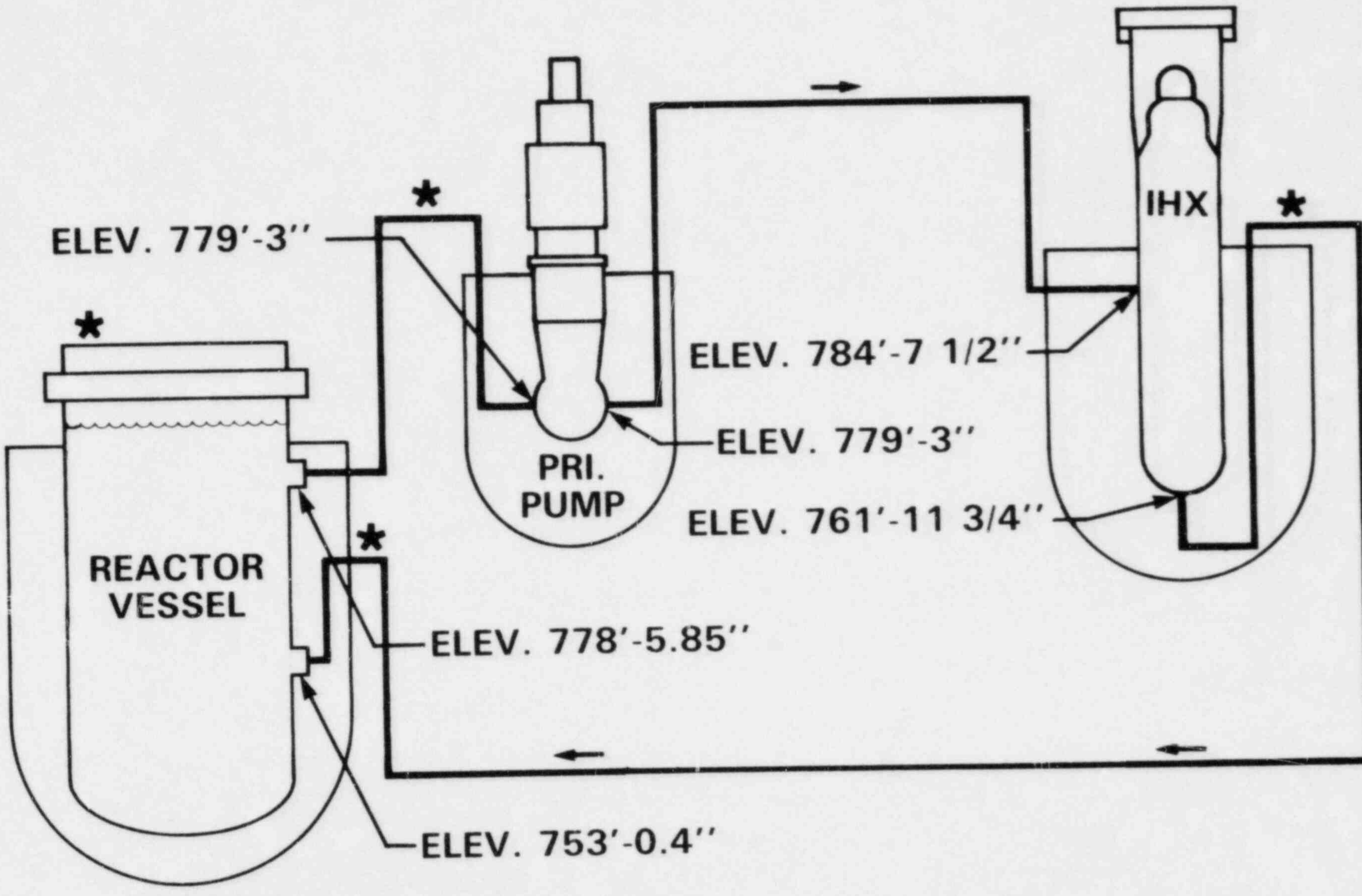
LOOSE PARTS MONITORING SYSTEM
GENERAL DESIGN CRITERIA (CONT.)

- THE CRBRP PROJECT WILL UTILIZE REG. GUIDE 1.133 IN IMPLEMENTING A LOOSE PARTS MONITORING SYSTEM EXCEPT WHERE DIFFERENCES BETWEEN LWR AND LMFBR TECHNOLOGY REQUIRE DIFFERENT METHODS.
- THE LOOSE PARTS MONITORING SYSTEM SHOULD MEET THE REQUIREMENTS OF PARAGRAPH C.1.G, "OPERABILITY FOR SEISMIC AND ENVIRONMENTAL CONDITIONS" OF REG. GUIDE 1.133, REV. 1.
- A BASELINE NOISE SIGNATURE WILL BE ESTABLISHED FOR THE LOOSE PARTS MONITORING SYSTEM AND WILL BE MONITORED EITHER ON A CONTINUING OR PERIODIC BASIS.
- SUITABLE AUDIBLE INDICATIONS/MONITORING OF THE PRESENCE OF LOOSE PARTS SHALL BE PROVIDED IN THE CONTROL ROOM, AND AT OTHER PLANT LOCATIONS AS APPROPRIATE.

SENSOR CHARACTERISTICS

- STATE-OF-THE-ART
- COMMERCIALLY AVAILABLE
- TEMPERATURE RANGE
  - 65°F TO 700°F
- RADIATION LIMITS
  - 6.2×10^{10} R INTEGRATED GAMMA FLUX
  - 3.7×10^{18} n/cm^2 INTEGRATED NEUTRON FLUX

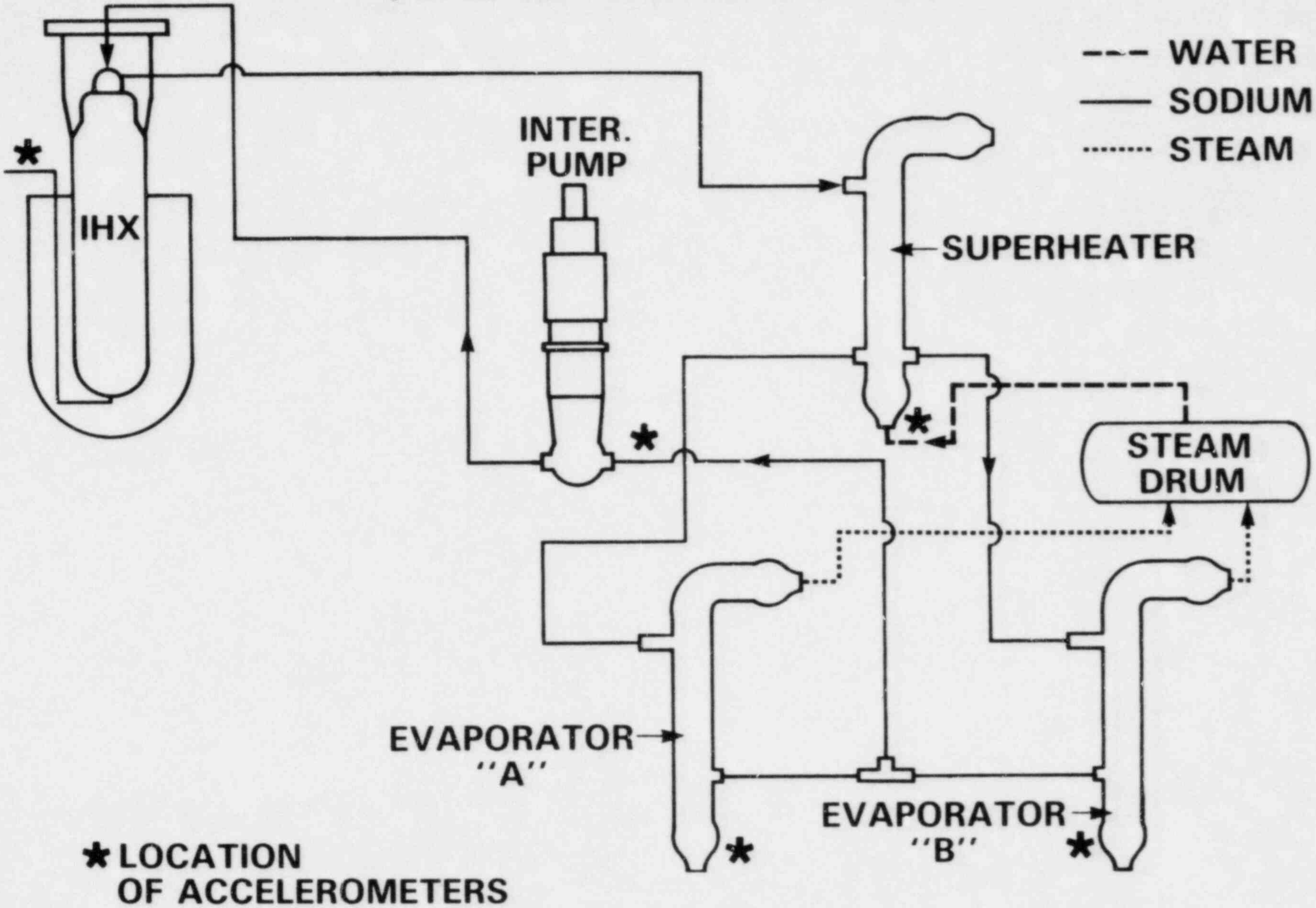
PRIMARY HEAT TRANSPORT SYSTEM
(ONE OF THREE LOOPS)



* LOCATION OF ACCELEROMETERS

INTERMEDIATE HEAT TRANSPORT SYSTEM

(ONE OF THREE LOOPS)



LPMS COMPLIANCE WITH NRC REG GUIDE

- COMPONENTS OF THE LPMS WITHIN CONTAINMENT SHALL BE DESIGNED AND INSTALLED TO PERFORM THEIR FUNCTION FOLLOWING ALL SEISMIC EVENTS THAT DO NOT REQUIRE PLANT SHUTDOWN, I.E., UP TO AND INCLUDING THE OPERATING BASIS EARTHQUAKE (OBE).
- THE LPMS SHALL BE SHOWN TO BE ADEQUATE BY ANALYSIS, TEST, OR COMBINED ANALYSIS AND TEST FOR THE NORMAL OPERATING RADIATION, VIBRATION, TEMPERATURE, AND HUMIDITY ENVIRONMENT.

DEVELOPMENT OF BASELINE NOISE SIGNATURE

- MONITOR TESTS ON THE CRBRP PROTOTYPE STEAM GENERATOR AT ENERGY TECHNOLOGY AND ENGINEERING CENTER (ETEC) FOR BACKGROUND NOISE SIGNATURE.
- BASELINE NOISE SIGNATURE SHALL BE TAKEN DURING STARTUP TESTING.

Ⓦ AESD

CRBRP

CORE SUPPORT CONE WELDS

STRUCTURAL INTEGRITY

PRESENTED TO

ACRS

CRBRP SUBCOMMITTEE AND

MATERIALS AND STRUCTURES WORKING GROUP

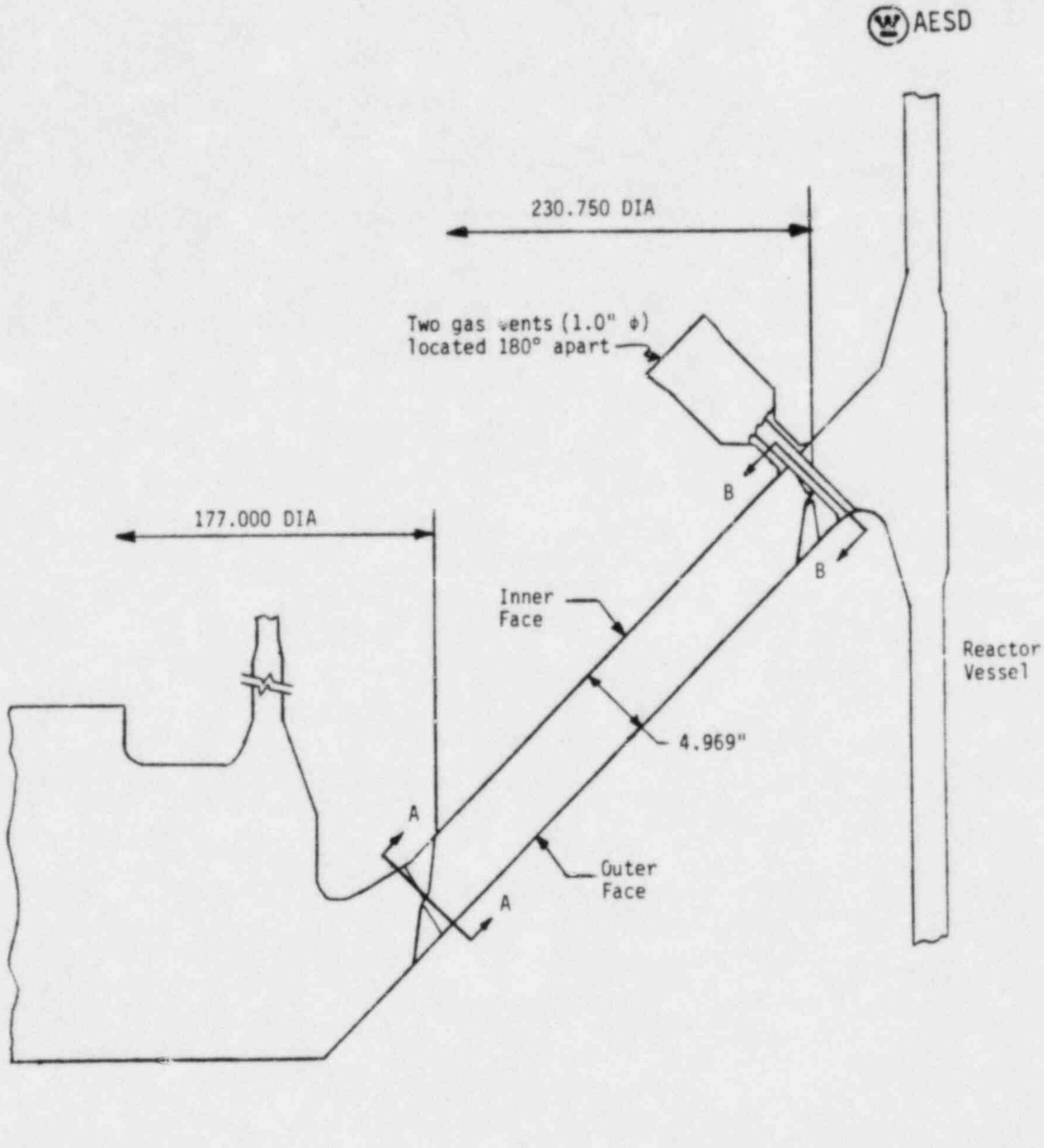
MARCH 16 AND 17, 1983

G. H. NICKODEMUS

W-AESD

THE CORE SUPPORT CONE WELDS HAVE A HIGH LEVEL OF ASSURED STRUCTURAL INTEGRITY FOR THE DESIGN LIFE OF THE PLANT

- DESIGNED, ANALYZED, CONSTRUCTED, AND INSPECTED TO RIGID ASME CODE REQUIREMENTS
- LOCATED IN BENIGN ENVIRONMENT
  - NORMAL OPERATING TEMPERATURE 750°F
  - SODIUM AND AGING EFFECTS NEGLIGIBLE
  - IRRADIATION EFFECTS NEGLIGIBLE
- LOCATED AWAY FROM GEOMETRICAL DISCONTINUITY IN LOW STRESS REGION.
- MEET ALL DESIGN LIMITS WITH SUBSTANTIAL MARGIN
- HIGH DEGREE OF TOLERANCE TO CRACK GROWTH AND INSTABILITY



CORE SUPPORT CONE GEOMETRY

HIGH QUALITY FABRICATION AND INSPECTION PROCEDURES WERE USED FOR THE CORE SUPPORT CONE WELDS

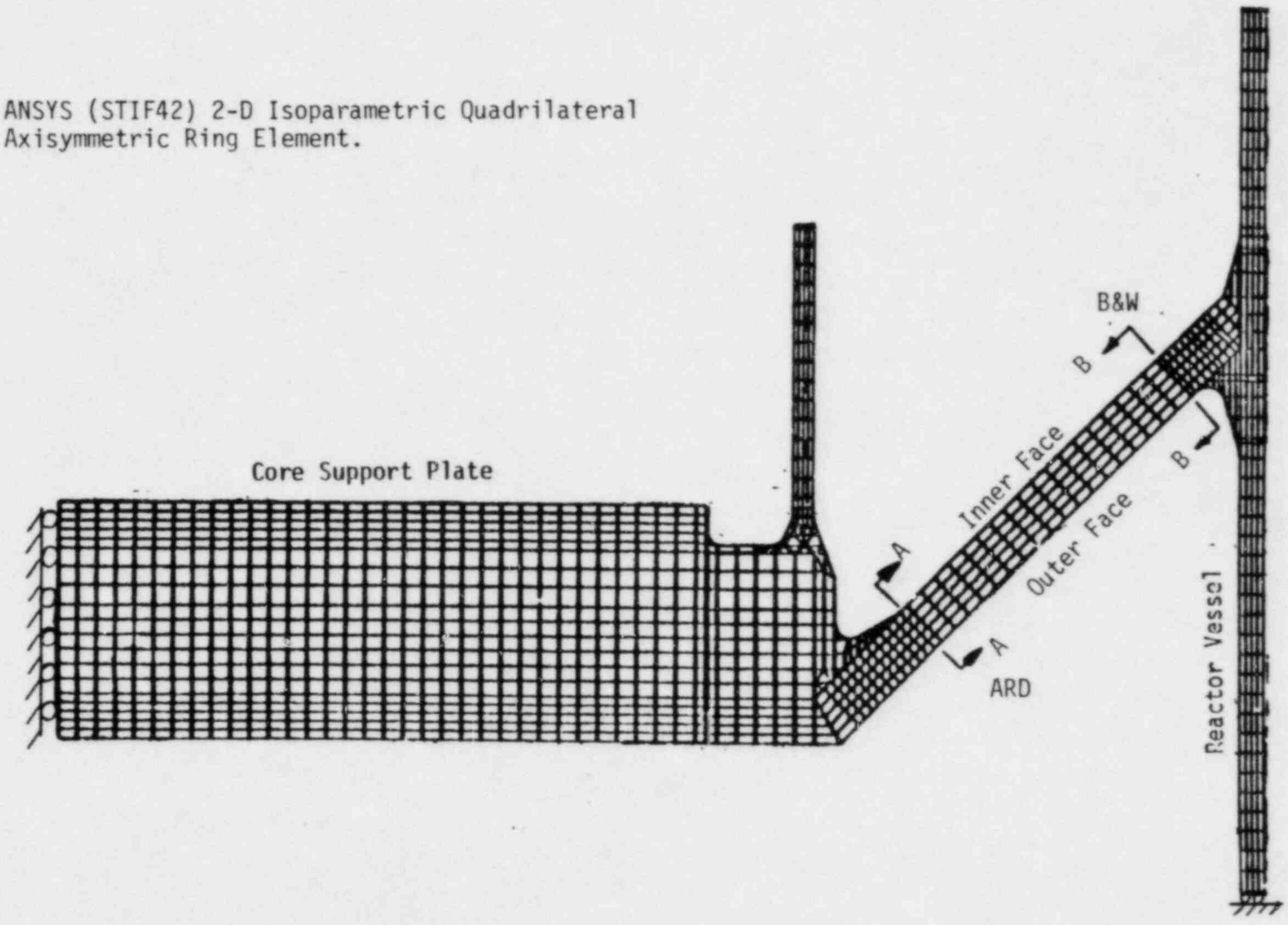
- WELD JOINTS INSPECTED PRIOR TO WELDING.
- QUALIFIED WELDERS AND PROCEDURES.
- EACH BEAD BRUSHED AND ALL STARTS AND STOPS GROUND.
- INNER FACE WELDING COMPLETED, GROUND AND VISUALLY AND PT INSPECTED.
- ROOT AREA BACKGROOVED, CLEANED, AND VISUALLY AND PT INSPECTED.
- PARTIALLY COMPLETED WELD TO CSS RADIOGRAPHED FOR INFORMATION.
- OUTER FACE WELDING COMPLETED, GROUND AND VISUALLY AND PT INSPECTED AND RADIOGRAPHED.
- WELD SURFACES FINISHED.
- FINAL VISUAL, PT, AND RADIOGRAPHIC INSPECTION.
- ALL RADIOGRAPHY BY A WIDE SPECTRUM LINATRON SOURCE.
- NO WELD REPAIRS IN CSS WELD.
- ONLY TWO WELD REPAIRS IN VESSEL WELD (3% OF CIRCUMFERENCE).

ENVIRONMENTAL EFFECTS ON THESE
WELDS ARE NEGLIGIBLE

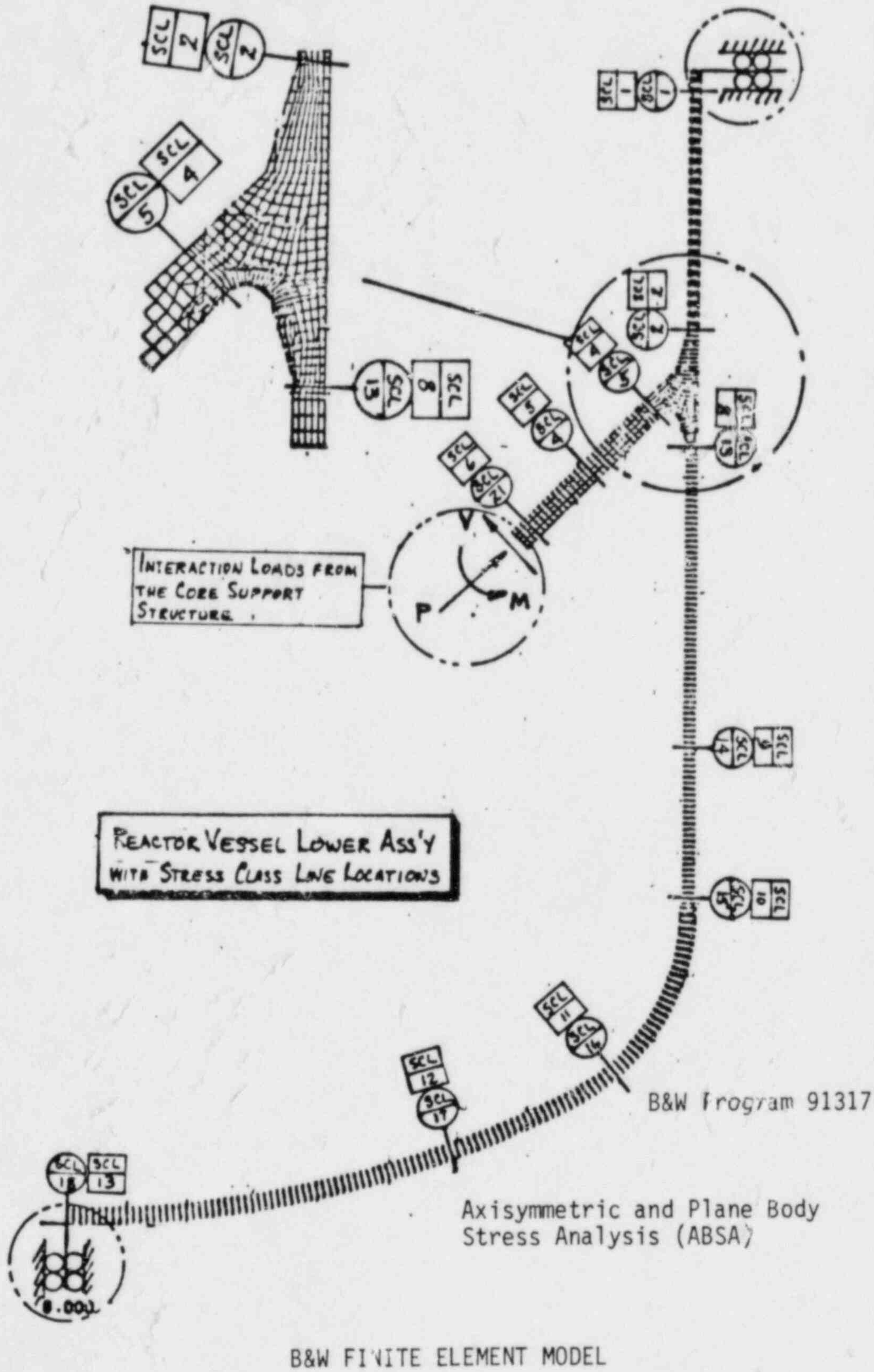
- CORROSION NON-EXISTANT - LOW TEMPERATURE.
- EROSION NEGLIGIBLE - LOW FLOW.
- CARBURIZATION NEGLIBIBLE - LOW TEMPERATURE.
- AGING NEGLIGIBLE - LOW TEMPERATURE.
- IRRADIATION NEGLIGIBLE.
  - HELIUM EMBRITTLEMENT - LOW TEMPERATURE.
  - DISPLACEMENT DAMAGE - LOW EXPOSURE.

CSS AXISYMMETRIC MODEL
WITH CUT IDENTIFICATIONS

ANSYS (STIF42) 2-D Isoparametric Quadrilateral
Axisymmetric Ring Element.



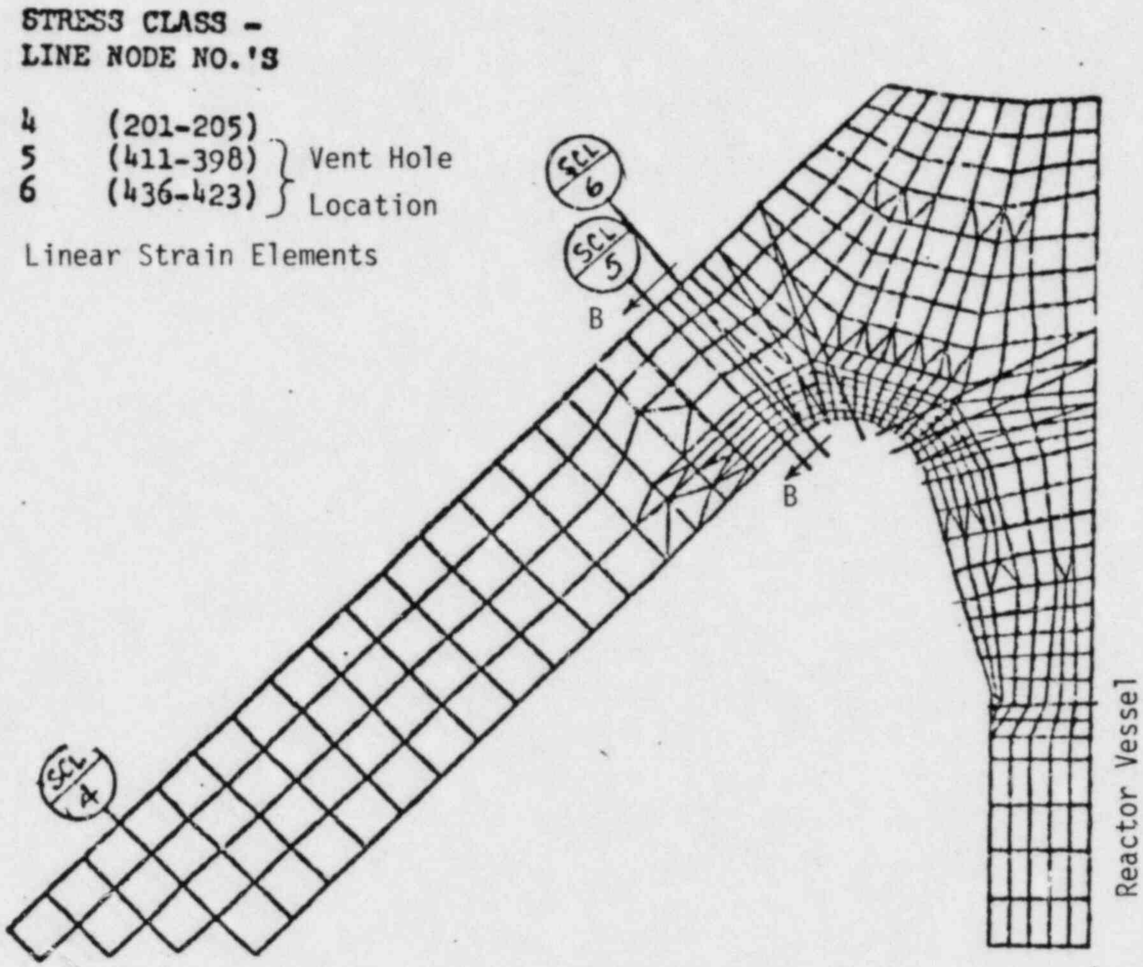
W ARD FINITE ELEMENT MODEL



B&W Program 91317

Axisymmetric and Plane Body Stress Analysis (ABSA)

B&W FINITE ELEMENT MODEL

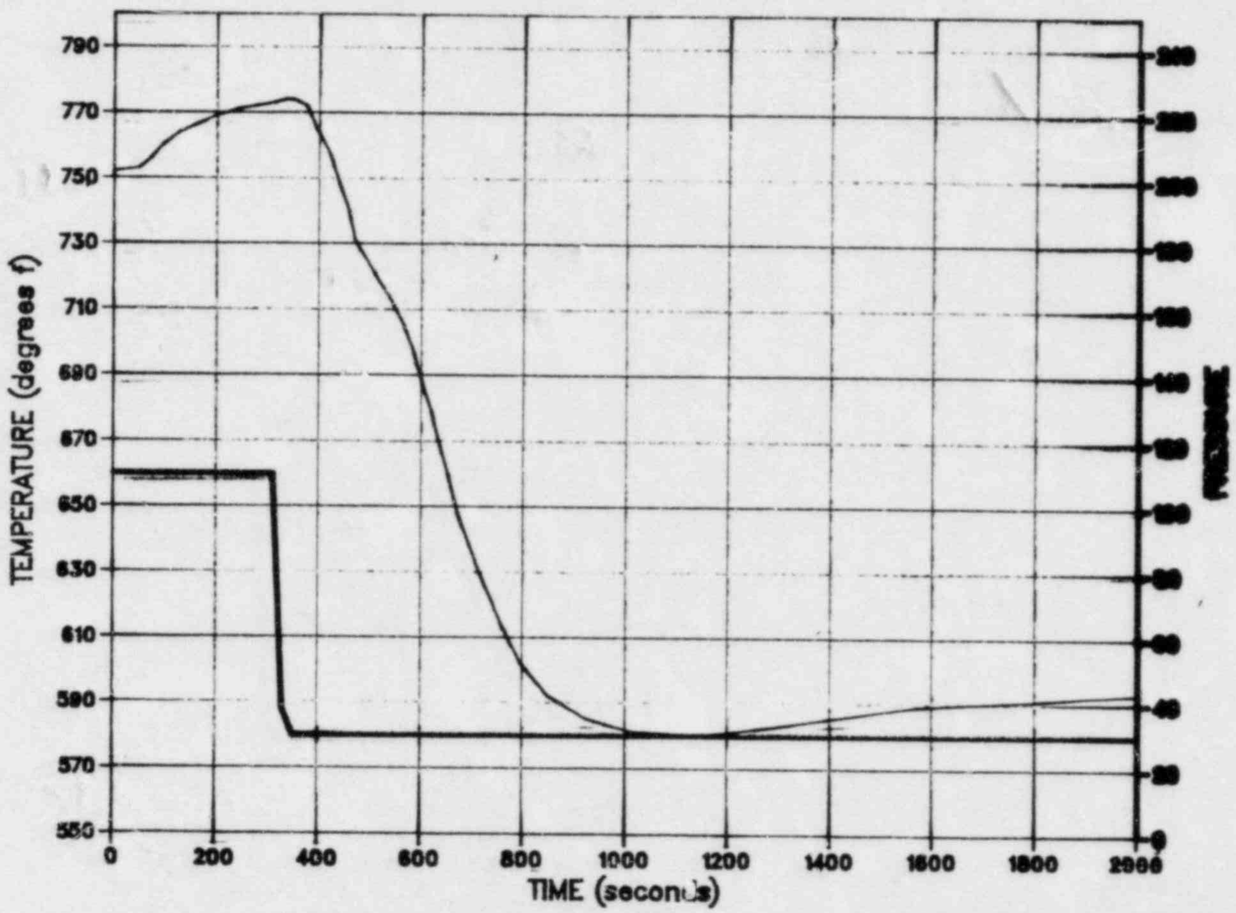


B&W FINITE ELEMENT MODEL AT THE WELD

AESD

(W) AESD

INLET PLENUM TEMPERATURE RESPONSE DURING A U-2B UNCONTROLLED ROD WITHDRAWAL FROM FULL POWER



STRESS SUMMARY AT SECTION A-A

OUTER-FACE

Criteria	Loading Combinations	Stress Categories	Calculated Stress Intensities (psi)	Allowable Stress Intensities (850°F) (psi)	Margin of Safety*
Normal + Upset	OBE + ΔP	P_m	6245	S_{mt} = 14800	1.37
	OBE + ΔP	P_L+P_b	9360	K_tS_t= 21230	1.27
	OBE + Thermal** (Linear)	$(P_L+P_b+Q)_{range}$	41490	$3\bar{S}_m$ = 49185	0.186
	Fatigue** (OBE+ΔP+Thermal)	P_L+P_b+Q+F	$\Sigma\frac{n}{N}$ = 0.06 < 0.9 allowable per T-1435 CC1592		
Faulted	SSE + ΔP	P_m	7560	1.2 S_t = 23160	2.06
	SSE + ΔP	P_L+P_b	10895	$1.2K_tS_t$ = 25500	1.3

* Margin of Safety $= \frac{\text{Allowable Stress}}{\text{Calculated Stress}} - 1$

** Includes Emergency loading condition thermal transients.

STRESS SUMMARY AT SECTION B-B AND GAS VENT

OUTER FACE

Criteria	Loading Combinations	Stress Categories	Calculated Stress Intensities (psi)	Allowable Stress Intensities (850°F) (psi)	Margin of Safety*
Normal + Upset	OBE + ΔP	P_m	6380	S_{mt} = 14800	1.32
	OBE + ΔP	$P_L + P_b$	15270	$K_t S_t$ = 21230	0.39
	OBE + Thermal** (Linear)	$(P_L + P_b + Q)_{range}$	27087	$3\bar{S}_m$ = 49185	0.81
	Fatigue** (OBE+ΔP+Thermal)	$P_L + P_b + Q + F$	$\Sigma \frac{n}{N}$ = 0.02 < 0.9 allowable per T-1435 CC1592 $\sim$ 0.3 < 0.9 allowable per T-1435 CC1592 (at gas vent)		
Faulted	SSE + ΔP	P_m	8950	1.2 S_t = 23160	1.59
	SSE + ΔP	$P_L + P_b$	19475	1.2 $K_t S_t$ = 25500	0.31

* Margin of Safety = $\frac{\text{Allowable Stress}}{\text{Calculated Stress}} - 1$

** Includes Emergency loading condition thermal transients.

IT IS UNLIKELY THAT A CRACK-LIKE DEFECT OF ANY SIGNIFICANT SIZE WILL EXIST IN THE MAIN WELDS OF THE CORE SUPPORT CONE AT THE BEGINNING OF LIFE.

- HIGH QUALITY OF CONSTRUCTION AND INSTALLATION
- THOROUGH INSPECTIONS DURING WELDING

IT IS UNLIKELY THAT CRACKS WILL DEVELOP DURING THE SERVICE LIFE.

- LOW OPERATING TEMPERATURE
- COMPRESSIVE PRIMARY OPERATING STRESSES
- LARGE DESIGN MARGINS ARE EXHIBITED WITH RESPECT TO CONSERVATIVE DESIGN LIMITS

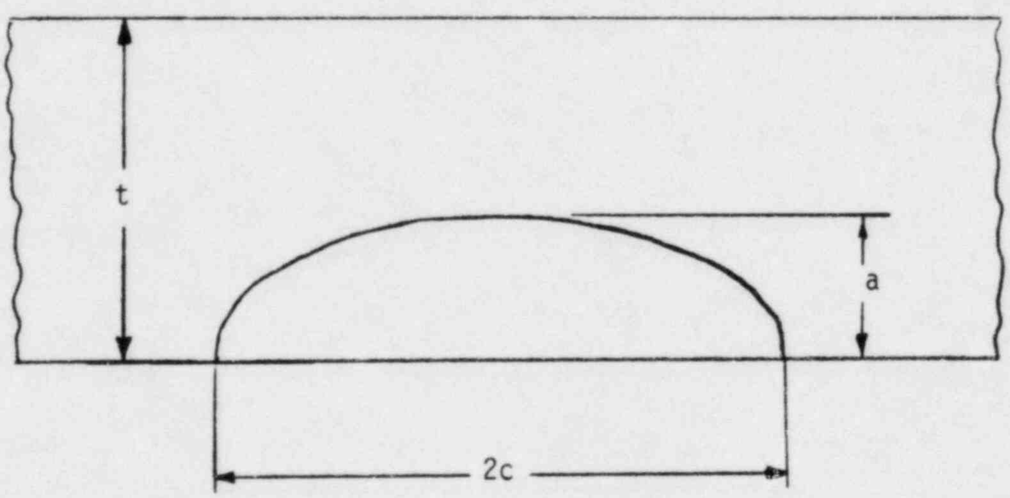
THE EFFECTS OF POSTULATED CRACKS ON THE STRUCTURAL INTEGRITY OF THE MAIN WELDS OF THE CRBRP CORE SUPPORT CONE (FAGIGUE CRACK GROWTH DURING SERVICE AND CRACK INSTABILITY) ARE NEGLIGIBLE.

- THE EVALUATION IS FOCUSED ON THE LOWER MAIN WELD JOINING THE CORE SUPPORT CONE TO THE CORE SUPPORT PLATE.

- THE RESULTS ARE CONSIDERED GENERALLY APPLICABLE TO THE UPPER MAIN WELD JOINING THE CORE SUPPORT CONE TO THE REACTOR VESSEL, ALTHOUGH THE EFFECTS OF THE BY-PASS FLOW GAS VENTS ARE STILL BEING CONSIDERED.

FOR EVALUATING FATIGUE CRACK GROWTH, THE CONE IS MODELED AS A CIRCULAR CYLINDER CONTAINING A CIRCUMFERENTIALLY-ORIENTED SURFACE CRACK.

- CRACK MAY ORIGINATE FROM INNER OR OUTER FACE.
- FATIGUE GROWTH IN THE TWO PRINCIPAL DIRECTIONS OF THE CRACK IS EXPLICITLY CALCULATED.



A CONSERVATIVE APPROACH WAS USED FOR THESE ANALYSES

- STATISTICAL UPPER BOUND IN SODIUM CRACK GROWTH DATA WERE USED.
- CALCULATIONS WERE PERFORMED USING THE CRACK GROWTH DATA AT 800°F AND 1000°F BECAUSE NEITHER DATA BASE IS CONSERVATIVE OVER THE ENTIRE APPROPRIATE K_{eff} RANGE.
- THE CYCLIC STRESS HISTORY WAS CONSERVATIVELY CONSTRUCTED.
- COMPRESSIVE PORTIONS OF THE CYCLIC STRESS RANGE WERE ASSUMED TO CAUSE THE SAME AMOUNT OF CRACK GROWTH AS AN EQUAL MAGNITUDE TENSILE STRESS RANGE.
- THE CRACK GEOMETRY WAS UPDATED FOR EACH OF THE 57930 CYCLES IN THE LOADING HISTORY.
- THE EFFECTIVE STRESS INTENSITY FACTORS WERE RE-CALCULATED USING THE UPDATED CRACK GEOMETRY FOR EACH CYCLE.

FATIGUE CRACK GROWTH RESULTS FOR
OUTER FACE OF CORE SUPPORT CONE

Initial Crack Dimensions		Final Crack Dimensions (800°F)		Final Crack Dimensions (1000°F)	
a (Inches)	c (Inches)	a (Inches)	c (Inches)	a (Inches)	c (Inches)
0.500	1.500	0.500	1.500	0.500	1.500
1.000	3.000	1.002	3.000	1.003	3.000
1.500	4.500	1.506	4.501	1.512	4.502
2.000	6.000	2.017	6.003	2.038	6.006
0.500	2.500	0.501	2.500	0.501	2.500
1.000	5.000	1.003	5.000	1.006	5.000
1.500	7.500	1.511	7.501	1.526	7.501
0.500	0.500	0.500	0.500	0.500	0.500
1.000	1.000	1.000	1.001	1.000	1.001
1.500	1.500	1.501	1.502	1.500	1.502
2.000	2.000	2.001	2.003	2.001	2.006
2.500	2.500	2.503	2.507	2.503	2.516

FATIGUE CRACK GROWTH RESULTS FOR
INNER FACE OF CORE SUPPORT CONE

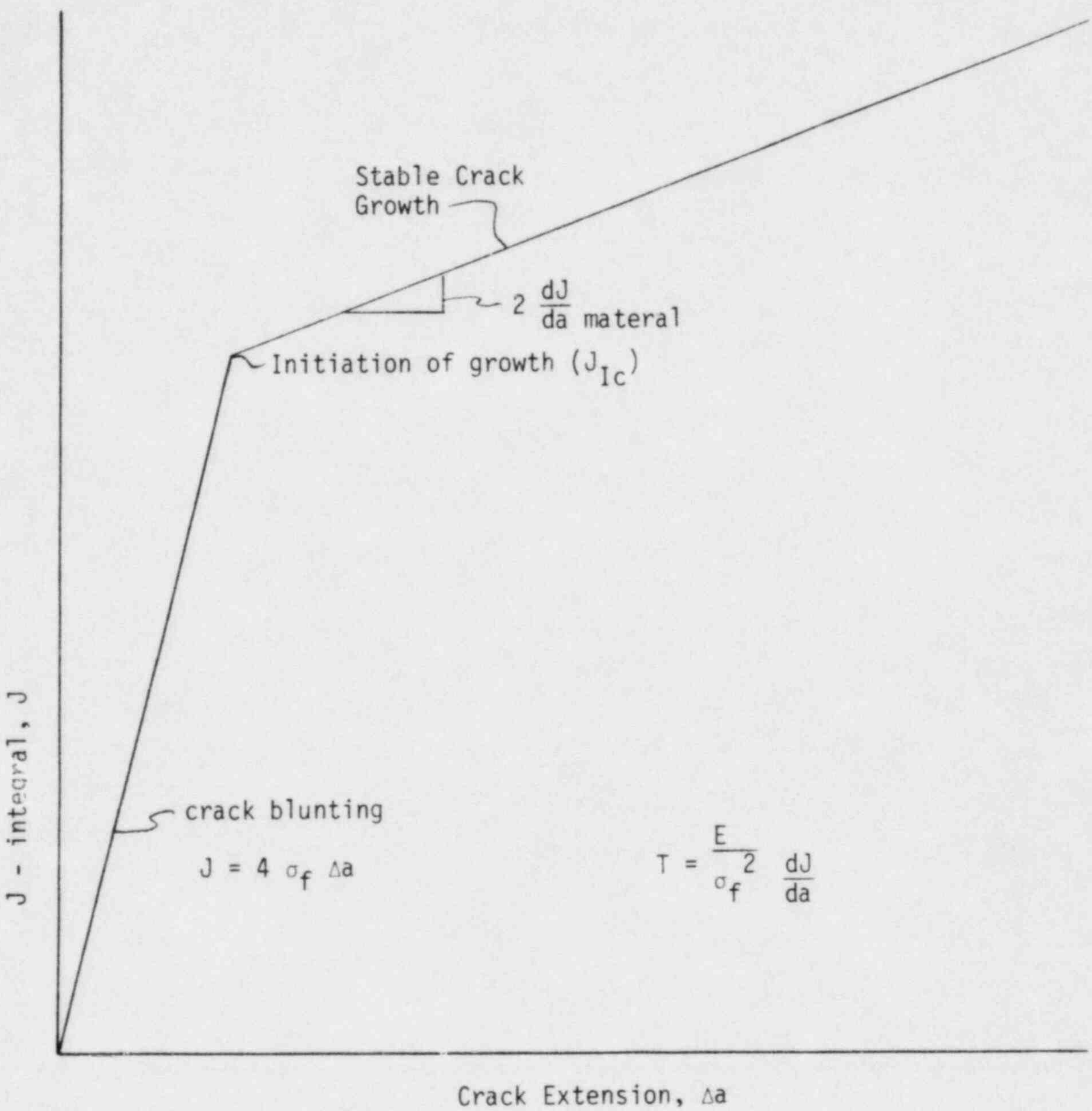
Initial Crack Dimensions		Final Crack Dimensions (800°F)		Final Crack Dimensions (1000°F)	
a (Inches)	c (Inches)	a (Inches)	c (Inches)	a (Inches)	c (Inches)
0.500	1.500	0.518	1.503	0.524	1.502
1.000	3.000	1.082	3.017	1.183	3.031
1.500	4.500	1.691	4.565	2.042	4.771
2.000	6.000	2.342	6.204	3.147	8.248
0.500	2.500	0.527	2.501	0.542	2.501
1.000	5.000	1.138	5.009	1.428	5.024
1.500	7.500	1.896	7.549	3.128	8.862
0.500	0.500	0.502	0.505	0.502	0.504
1.000	1.000	1.009	1.026	1.009	1.039
1.500	1.500	1.516	1.570	1.522	1.651
2.000	2.000	2.022	2.150	2.037	2.433
2.500	2.500	2.525	2.786	2.558	3.624

CRACK STABILITY DEPENDS ON TWO MATERIAL PROPERTIES

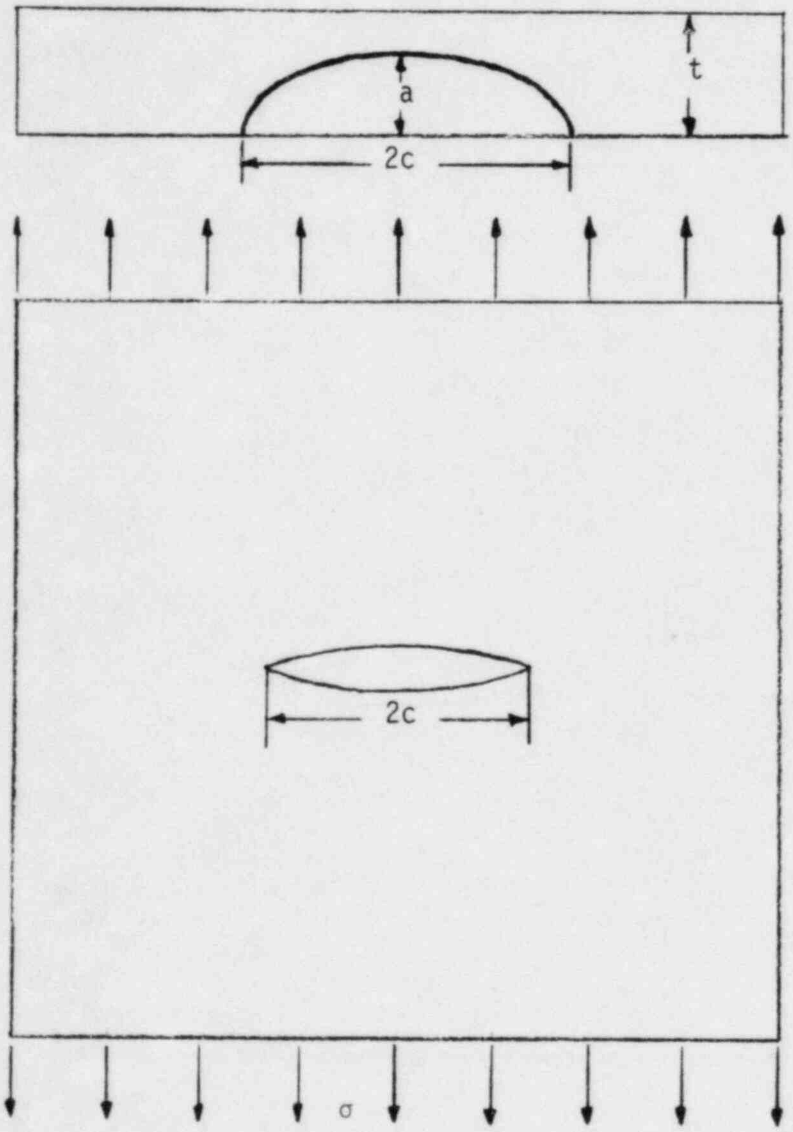
- FRACTURE TOUGHNESS DESCRIBES THE ONSET OF CRACK EXTENSION UNDER MONOTONICALLY INCREASING LOADING CONDITIONS.

- TEARING MODULUS DESCRIBES THE RESISTANCE OF THE MATERIAL TO ADDITIONAL CRACK EXTENSION AS THE LOAD IS INCREASED PAST THE ONSET OF THE CRACK EXTENSION FRACTURE TOUGHNESS LEVEL.

- THE TEARING MODULUS APPROACH WAS USED BECAUSE IT PROVIDES A MORE REALISTIC APPRAISAL OF CRACK STABILITY IN A HIGHLY DUCTILE MATERIAL.


Stable Crack Growth
2 dJ/da materal
Initiation of growth (J_{Ic})
crack blunting
$J = 4\ \sigma_f\ \Delta a$
$T = \frac{E}{\sigma_f^{\ 2}} \frac{dJ}{da}$
J - integral, J
Crack Extension, Δa

THE SURFACE-CRACKED PLATE MODEL INDICATES THAT CRACK INSTABILITY REQUIRES A CRACK LENGTH OF 420 INCHES OR MORE.

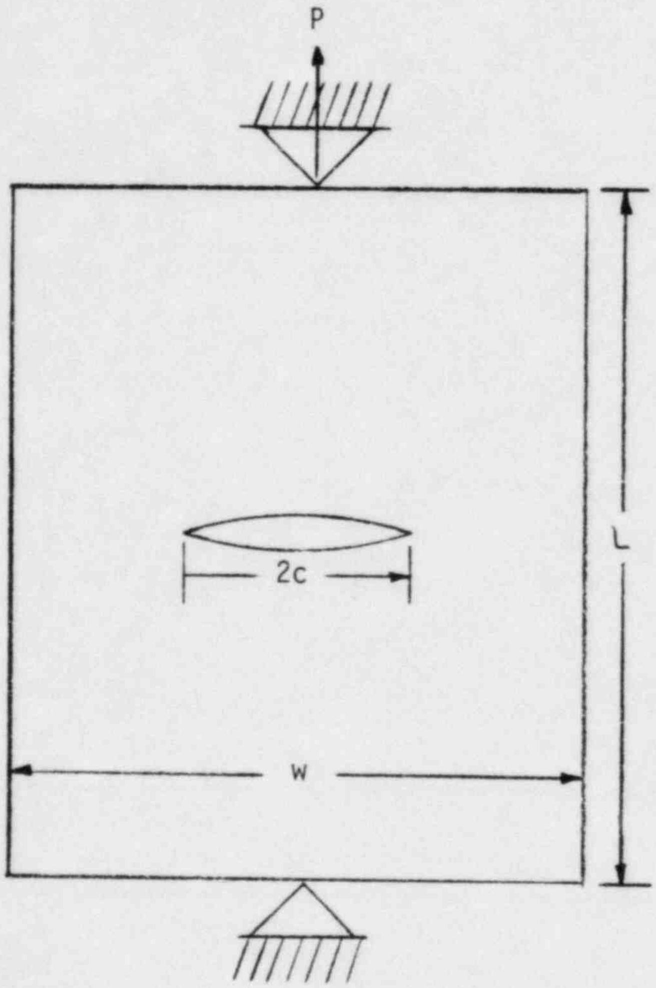


FOR INSTABILIY: $\frac{4c}{t} \geq T_m$

T_m = 168 (weld metal)

$c \geq 210$ Inch

THE THROUGH-CRACKED PLATE MODEL INDICATES THAT CRACK INSTABILITY REQUIRES A CORE SUPPORT CONE LENGTH OF MORE THAN 1900 FEET.



For Instability: $\frac{2L}{W} \geq T_m$

$L = 2L_{cone}$

$W = \pi D_{Lower}$

$L_{cone} \geq 23{,}100$ Inches

Actual cone length = $\sim$ 4 feet

THE FRACTURE MECHANICS ASSESSMENT SHOWS A HIGH DEGREE OF TOLERANCE TO POTENTIAL FLAWS

- A LARGE POSTULATED INITIAL SURFACE FLAW OF 1.5 x 15 INCHES HAS A LIMITED GROWTH TO 3.1 x 17.7 INCHES FOR THE PLANT LIFETIME.
- INSTABILITY SIZE FOR A SURFACE FLAW IS 420 INCHES IN LENGTH.
- THROUGH THICKNESS CRACK REQUIRES A CONE LENGTH OF 1900 FEET FOR INSTABILITY.
- FURTHER ASSESSMENTS AT THE GAS VENT LOCATIONS WILL BE PERFORMED.

THE CORE SUPPORT CONE WELDS HAVE A HIGH LEVEL OF ASSURED STRUCTURES INTEGRITY FOR THE DESIGN LIFE OF THE PLANT

- DESIGNED, ANALYZED, CONSTRUCTED, AND INSPECTED TO RIGID ASME CODE REQUIREMENTS.
- LOCATED IN BENIGN ENVIRONMENT.
  - NORMAL OPERATING TEMPERATURE 750°F
  - SODIUM AND AGING EFFECTS NEGLIGIBLE
  - IRRADIATION EFFECTS NEGLIGIBLE
- LOCATED AWAY FROM GEOMETRICAL DISCONTINUITY IN LOW STRESS REGION.
- MEET ALL DESIGN LIMITS WITH SUBSTANTIAL MARGINS.

  IN PARTICULAR, EXCEPT AT THE GAS VENTS
  SECONDARY MEMBRANE AND BENDING S.I. $< 3\overline{S}_m$
  FATIGUE DAMAGE < 0.1
- HIGH DEGREE OF TOLERANCE TO CRACK GROWTH AND INSTABILITY.

CLINCH RIVER BREEDER REACTOR PLANT

BRIEFING FOR:

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS) CRBRP SUBCOMMITTEE & MATERIALS AND STRUCTURE WORKING GROUP

WHAT IF?

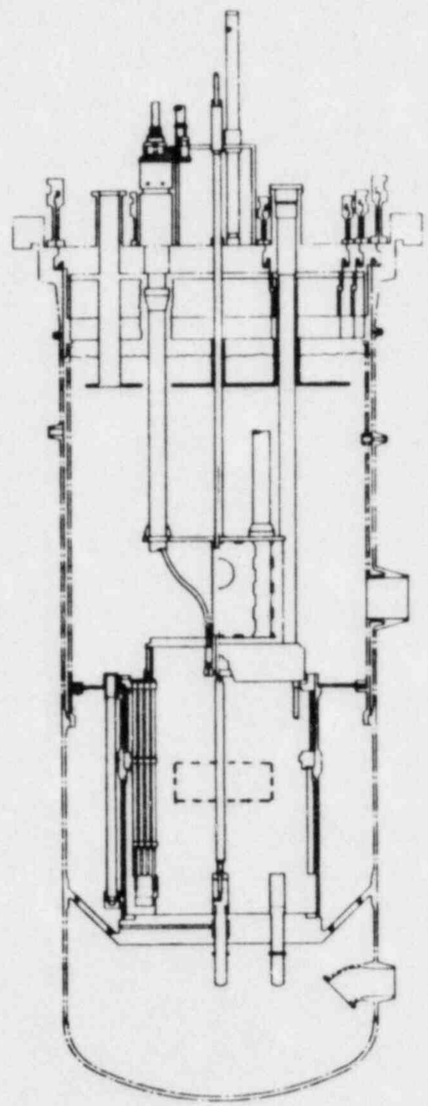
PRESENTED BY:

PAUL W. DICKSON
TECHNICAL DIRECTOR
CRBRP PROJECT
WESTINGHOUSE-AESD
OAK RIDGE, TN

MARCH 16, 1983

3-83-3279-4

REACTOR VESSEL AND INTERNALS CUTAWAY



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CORE SUPPORT CONE FAILURE

- POSTULATED FAILURE OCCURS AFTER SHUTDOWN
- UPON FALLING, THE REACTOR REMAINS SUBCRITICAL WITH SIX SECONDARY RODS INSERTED
- FOR ALMOST ALL OF CRBRP OPERATION, THE REACTOR WOULD REMAIN SUBCRITICAL EVEN AFTER COOLDOWN TO 600°F
- FOR THE FIRST FEW DAYS OF OPERATION OF CYCLES 3 AND 4, THE REACTOR ACHIEVES CRITICALITY AS THE REACTOR DECAY POWER REDUCES TO ~4% FULL POWER
- NUCLEAR POWER INCREASES WILL OCCUR TO MAINTAIN THE REACTOR HOT AS THE INLET PLENUM COOLS AND DECAY POWER DECREASES.
- AS SYSTEM TEMPERATURE TRIES TO ACHIEVE 600°F THE REACTOR POWER INCREASES (TO ~90MW AT BOC 3)

3-83-3279-6

APPLICABLE OPERATING PERIOD

- FOR THE WORST CYCLES (3 AND 4) AFTER LESS THAN 40 FULL POWER DAYS OR FOR CYCLES 1 AND 2 AT BEGINNING OF LIFE, THE REACTOR WOULD REMAIN SUBCRITICAL AFTER COOLDOWN TO 600°F
- FOR CYCLES 3 AND 4 AFTER LESS THAN 75 FULL POWER DAYS (OR CYCLES 1 AND 2 AT BOL), THE REACTOR WOULD REMAIN SUBCRITICAL EVEN IF THE REACTOR OPERATOR TOOK ACTION TO COOL THE PLANT TO 400°F

CORE SUPPORT CONE FAILURE

- WITH NO OPERATOR INTERVENTION, FOR THE WORST TIME IN LIFE, THE SYSTEM SHOULD STABILIZE AT:
  - INLET TEMPERATURE $\sim$ 600° F
  - REACTOR POWER $\sim$90 MW
  - REACTOR OUTLET BULK TEMPERATURE $\sim$1200° F
  - REACTOR PEAK OUTLET TEMPERATURE $\sim$1350° F
  - HOT LEG TEMPERATURE $\sim$ 890° F

3-83 3279-8

CONCLUSION

- ONLY FOR A SMALL FRACTION OF CRBRP OPERATING LIFE WOULD EVEN A RECRITICALITY OCCUR
- NO POWER EXCURSIONS OCCUR (NO ENERGETIC HCDA)
- IF CORE COOLING IS SIGNIFICANTLY LESS THAN PREDICTED, A PARTIAL MELTDOWN OF THE CORE COULD RESULT TO SHUT THE REACTOR DOWN WITH IN-PLACE LONG-TERM DECAY HEAT COOLING
- AT THE WORST, A TOTAL CORE MELTDOWN WOULD BE BOUNDED BY THE BASE CASE TMBDB ANALYSIS.

CRBR - LOCAL FUEL FAILURES

STAFF PRESENTATION TO ACRS - 3/16/83

T. KING

SCOPE OF STAFF REVIEW

REVIEW OF EXPERIMENTAL AND ANALYTICAL DATA ON LOCAL FAILURES AND BLOCKAGES FOR:

- o FUEL ASSEMBLIES
- o BLANKET ASSEMBLIES
- o CONTROL ASSEMBLIES

REVIEW OF DETECTION SYSTEMS:

- FISSION GAS MONITORING
- DELAYED NEUTRON MONITORING
- CORE EXIT INSTRUMENTATION

REVIEW OF FUTURE WORK:

- ANALYTICAL
- EXPERIMENTAL

ASSISTANCE BY LANL

DESIGN FEATURES WHICH MINIMIZE POTENTIAL FOR PROPAGATION

- DUCTED ASSEMBLIES
- FLOW BLOCKAGE PREVENTION DEVICES
- GAS ENTRAINMENT PREVENTION DEVICES
- FEATURES TO PREVENT MISLOADING ERRORS
- QA AND INSPECTION DURING ASSEMBLY FABRICATION
- DND, FISSION GAS DETECTION AND CORE EXIT THERMOCOUPLE SYSTEMS

CAUSES OF FAILURE

- STOCHASTIC PIN FAILURE
- INSUFFICIENT HEAT TRANSFER
- LOW FLOW
- EXCESS POWER

EFFECTS EVALUATED

FISSION GAS RELEASE
- GAS BLANKETING OF PIN *
- FLOW REDUCTION CAUSED BY GAS RELEASE *
- MECHANICAL EFFECTS *
- REACTIVITY EFFECTS *

FUEL RELEASE
- PARTICULATE BLOCKAGE *
- MOLTEN FUEL COOLANT INTERACTION*
- MOLTEN FUEL IMPINGEMENT *

EXCESS POWER EVENTS
- OVERENRICHED PINS *

LOW FLOW EVENTS
- CRACK IN DUCT WALL
- FLOW BLOCKAGE:
  - W/W FAILURE
  - CLAD BOWING OR SWELLING *
  - DEBRIS (HEAT GENERATING & NON-HEAT GENERATING) *

CHANGE IN HEAT TRANSFER
- CORROSION PRODUCT DEPOSITION
- INTRODUCTION OF FOREIGN MATERIAL INTO PHTS
- GAS BUBBLES IN CORE

FUEL PERFORMANCE AFTER A CLADDING BREACH
- STEADY STATE *
- TRANSIENT *

*EXPERIMENTAL DATA EXISTS TO SUPPORT EVALUATION

CONCLUSIONS

FAILURE CHARACTERISTICS

1) NO OBSERVED FUEL FAILURE PROPAGATION IN ANY OPERATING LMFBR.

2) MOST FAILURES ARE SMALL PINHOLE CLADDING BREACHS.

3) ALL ANALYSIS AND EXPERIMENTAL DATA INDICATE THAT FISSION GAS RELEASES AND SMALL LOCAL BLOCKAGES [illegible] LEAD TO ADDITIONAL PIN FAILURES.

4) LARGE FLOW BLOCKAGES OR EXPULSION OF MOLTEN FUEL FROM A PIN ARE REQUIRED TO CAUSE ADDITIONAL PIN FAILURES.

5) TIME REQUIRED FOR ADDITIONAL PIN FAILURES TO OCCUR IS ON THE ORDER OF MINUTES.

6) FUEL WILL BE EXPOSED TO FLOWING SODIUM IN A SITUATION THAT LEADS TO ADDITIONAL PIN FAILURES.

CONCLUSIONS (CONTINUED)

DETECTION NEEDS

NEED A FAST ACTING SYSTEM THAT WILL DETECT PROPAGATION CONDITION.

- CORE EXIT T/C's - MAY NOT DETECT ALL PROPAGATION CONDITIONS.
- CORE EXIT FLOWMETERS - MAY NOT DETECT ALL PROPAGATION CONDITIONS.
- DND - WILL DETECT ANY CONDITION WHERE FUEL IS EXPOSED TO FLOWING SODIUM (CONDITION FOR PROPAGATION).

ADDITIONAL DATA REQUIRED

COMPLETION OF RBCB PROGRAM

CONFIRMATORY FFTF TESTING

EXAMINATION OF P-4 TEST

SUMMARY OF STAFF POSITION

- o APPLICANTS SHOULD INSTALL DND SYSTEM SO AS NOT TO PRECLUDE CONNECTION TO THE RSS AT A LATER DATE.
- o FINAL DECISION ON NEED FOR AUTOMATIC SCRAM ON DND SIGNAL WILL BE MADE AS PART OF THE OL REVIEW.
- o REMOVE FAILED FUEL AT FIRST SHUTDOWN (PLANNED OR UNPLANNED) OR IF DND SIGNAL EXCEEDS PRE-DETERMINED LEVEL.
- o COMPLETE P-4 EXAMINATION AND RBCB, FFTF TESTING PROGRAMS.
- o ACCEPTABLE FOR A CP WITH THE ABOVE CONDITIONS.

Palm

CRBRP BRIEFING FOR

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS)

WORKING GROUP

EVALUATION OF N. I. FOUNDATION MAT FOR THERMAL MARGINS BEYOND THE DESIGN BASIS

PRESENTED BY

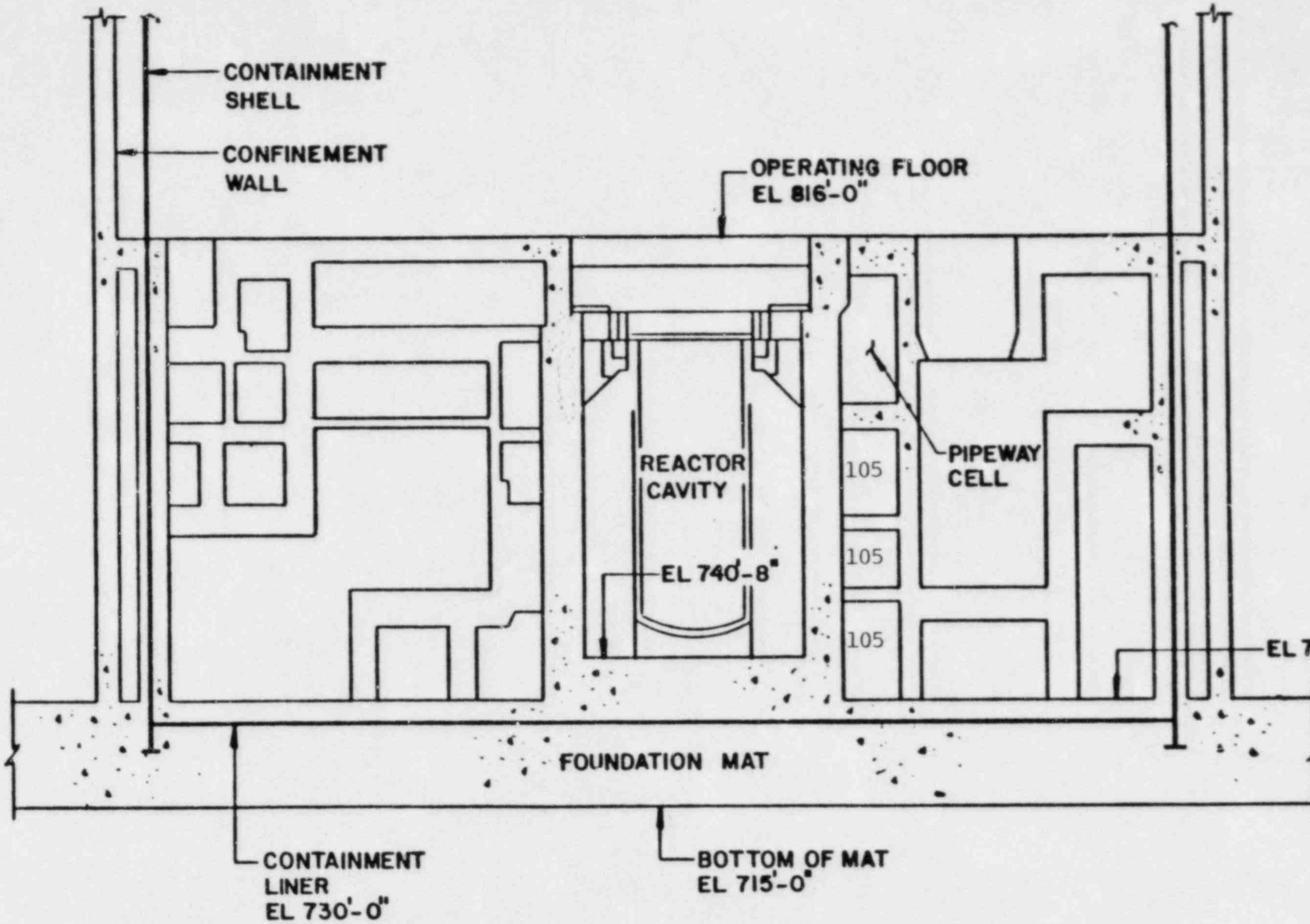
ROBERT E. PALM
CIVIL/STRUCTURAL ENGINEERING MANAGER
BURNS AND ROE, INC.
ORADELL, NEW JERSEY

MARCH 16, 1983

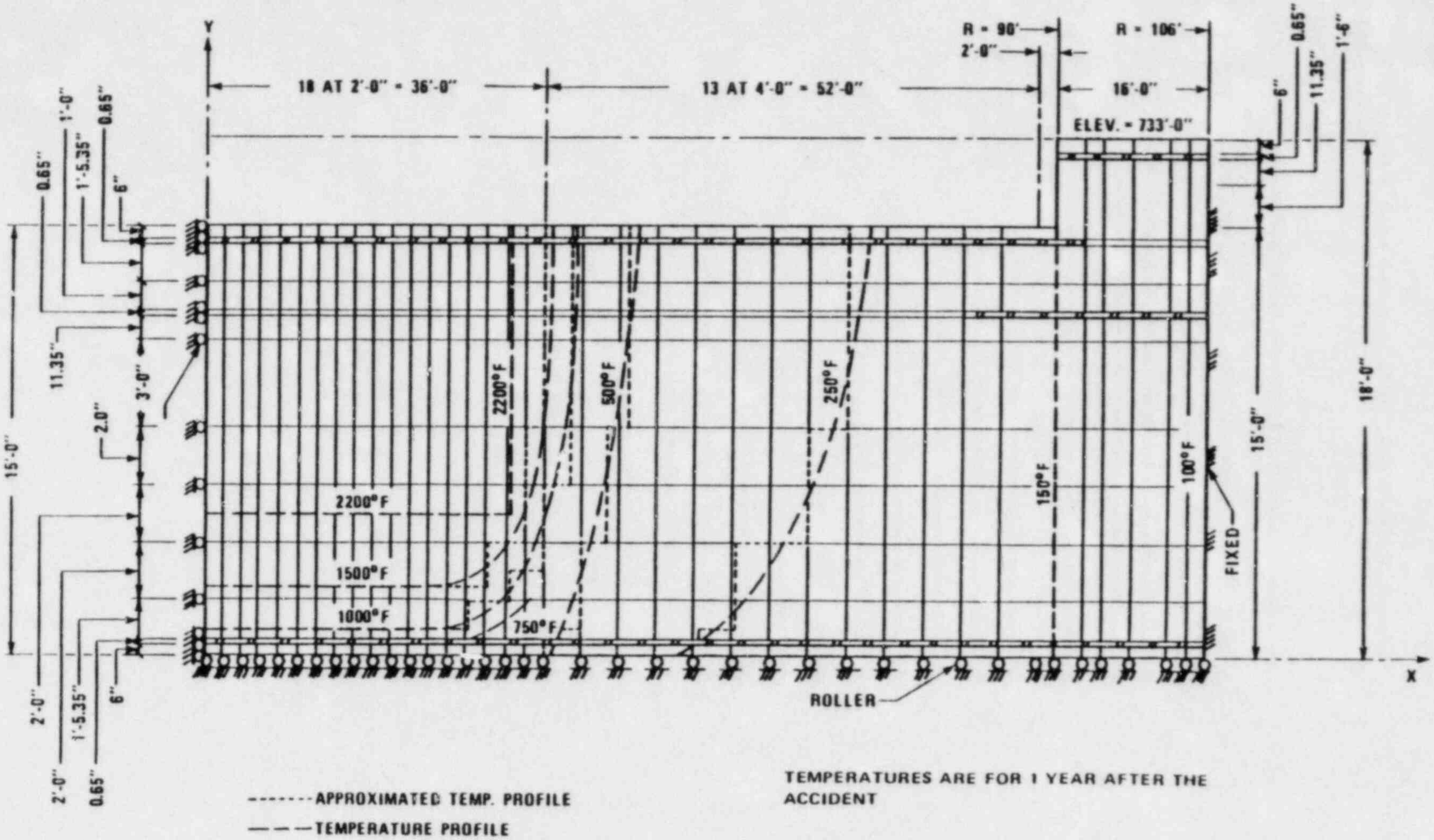
NUCLEAR ISLAND FOUNDATION MAT

STRUCTURAL EVALUATION

- ANALYSIS PERFORMED TO ASSURE DAMAGE TO MAT IS LOCAL AND CONTAINMENT/CONFINEMENT INTEGRITY IS MAINTAINED
- FINITE ELEMENT ELASTIC-PLASTIC ANALYSIS USING COMPUTER PROGRAM ANSYS
- TEMPERATURE PROFILES DEVELOPED TO 8000 HOURS
- MATERIAL PROPERTIES NON-LINEAR AND TEMPERATURE DEPENDENT
- ANALYSIS PERFORMED BY ITERATIVE PROCEDURES TO ACCOUNT FOR CRACKING AND CRUSHING OF CONCRETE



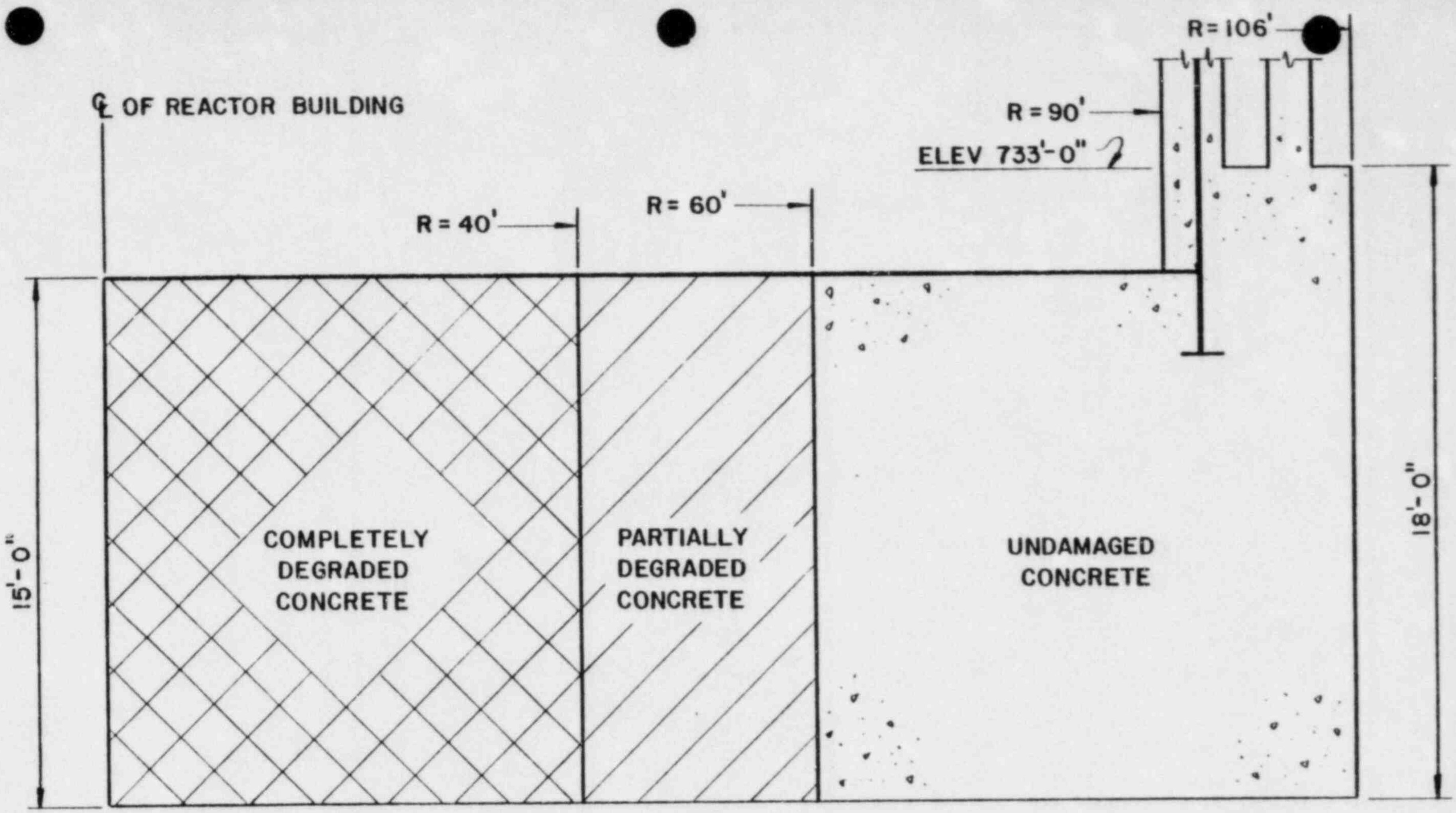
RCB CROSS SECTION



BASEMAT MODEL AFTER BOIL DRY

RESULTS OF STRUCTURAL EVALUATION

- TO TIME OF SODIUM BOIL DRY
  - NO INFLUENCE ON MAT INTEGRITY
- CONDITIONS AT 8000 HOURS FROM CENTERLINE REACTOR CAVITY
  - INNER PORTION FROM RADIUS (R) = 0 FT. TO R = 40 FT.
    CONCRETE DEGRADED AND CRUSHED
  - BETWEEN R = 40 FT AND R = 60 FT.
    PARTIAL DEGRADATION AND CRUSHING
  - FROM R = 60 FT AND BEYOND
    CONCRETE REMAINS STRUCTURALLY SOUND
- ADEQUATE SUPPORT IS PROVIDED FOR THE PERIPHERAL WALL, CONTAINMENT VESSEL AND CONFINEMENT STRUCTURE
- CONTAINMENT/CONFINEMENT INTEGRITY IS MAINTAINED



BASEMAT STRUCTURAL ANALYSIS RESULTS

PRIMARY SYSTEM RESPONSE TO SMBDB

THOMAS A. BUTLER
LOS ALAMOS NATIONAL LABORATORY

MARCH 16, 1983
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

PRIMARY SYSTEM RESPONSE TO SMBDB

- 0 CRITERIA USED IN EVALUATING PRIMARY SYSTEM BOUNDARY
- 0 LOADS EXPERIENCED BY PRIMARY SYSTEM
- 0 CONCLUSIONS FROM SCALE MODEL TESTS AND ANALYSES

SMBDB DESIGN REQUIREMENTS
FOR HEAD AND HEAD MOUNTED COMPONENTS

- NO HEAD MOUNTED COMPONENT SHALL BECOME A MISSLE CAPABLE OF IMPAIRING CONTAINMENT
- HEAD MOUNTED COMPONENT NOZZLES SHALL ACCOMMODATE HEAD LOADINGS.
- HEAD MOUNTED COMPONENTS MUST FUNCTION AS LIMITED LEAKAGE BARRIERS.
- VESSEL HEAD AND HEAD MOUNTED COMPONENTS SHALL ACCOMMODATE LONGER TERM MECHANICAL LOADS FROM SATURATED VAPOR.

SMBDB STRUCTURAL EVALUATION CRITERIA

- MEMBRANE STRAIN LIMIT TO PROTECT AGAINST PLASTIC INSTABILITY
- STRAIN LIMIT IS PROVIDED TO PROTECT AGAINST LOCAL DUCTILE RUPTURE
- APPROPRIATE STRESS LIMITS ARE USED FOR ELASTIC ANALYSES

SMBDB LOAD REQUIREMENTS ARE BASED ON TEST AND ANALYSIS

- Vessel and head loads based on REXCO-HEP calculations.
- Loads verified conservative by scale model tests.
- Vessel head required to accommodate a sodium slug with 75 MJ of kinetic energy.
- Remainder of primary system loads are based on conservative prediction techniques.

SCALE MODEL TEST PROGRAM

TEST	LOADING	CONFIGURATION
SM-1	Hydrostatic	No Shielding Plates
SM-2	Dynamic	No Upper Internal Structure
SM-3	Dynamic	Included Upper Internal Structure
SM-4	Dynamic	Complete Vessel, Internals, Non-Prototypic Head
SM-5	Dynamic	
SM-7	Hydrostatic	Non-Prototypic Shielding Plates
SM-8	Hydrostatic	Prototypic Shielding Plates

CONCLUSIONS OF TESTS AND ANALYSES

- PRESENT HEAD DESIGN CANNOT ACCOMMODATE 75 MJ SODIUM SLUG
- FAILURE WOULD BE KINEMATIC DISENGAGEMENT OF HEAD INTERMEDIATE ROTATING PLUG.
- HEAD CAN BE MODIFIED TO ELIMINATE CURRENT FAILURE MODE
- REMAINDER OF LOAD PATH WILL ACCOMMODATE LOADS RESULTING FROM IMPACT OF A 75 MJ SLUG

OBJECTIVES OF NRC REVIEW OF PSI/ISI PLAN

- FABRICATION EXAMINATIONS AND PSI ARE PERFORMED WITH BEST AVAILABLE TECHNOLOGY
- NDE REQUIRED FOR MAINTENANCE, REPAIR OR MODIFICATION AND IS CONSIDERED IN PLANT DESIGN
- ACCESS PROVIDED FOR PERIODIC VOLUMETRIC ISI
- SPECIALIZED EQUIPMENT DESIGNED FOR PLANNED ISI

CP REVIEW CONSIDERATIONS

FABRICATION/PSI

. DOUBLE ANGLE RT WILL BE PERFORMED ON WELDS IN VESSELS AND PIPING

. UT OF PIPE WELDS ($\geq$ 1/2 INCH WALL) SHOULD BE PERFORMED USING EXISTING TECHNOLOGY

. UT OF VESSEL WELD HAZ AND ADJACENT BASE METAL SHOULD BE PERFORMED WHERE TECHNICALLY FEASIBLE

CP REVIEW CONSIDERATIONS

ISI

- EXAMINATIONS SHOULD BE PERFORMED DURING THE REQUIRED MAINTENANCE, THUS REDUCING OCCUPATIONAL EXPOSURE AND PLANT OUTAGE TIME
- SELECTED VOLUMETRIC EXAMINATIONS SHOULD BE PERFORMED TO DETECT GENERIC, UNANTICIPATED DEGRADATION MECHANISMS
- INTEGRATED LEAKAGE DETECTION SYSTEM WILL BE DEMONSTRATED UNDER PLANT ENVIRONMENT AND TESTED PER TECHNICAL SPECIFICATIONS TO MAINTAIN ITS EFFECTIVENESS

TOPICS STILL BEING EVALUATED BY THE APPLICANT

- SURVEILLANCE PROCEDURES FOR THE REACTOR INTERNALS
- INSPECTION TECHNIQUES FOR THE IHX TUBE BUNDLE IN THE EVENT THAT MAINTENANCE PROVIDES ACCESS TO THE TUBES

OL REVIEW CONSIDERATIONS

- EVALUATION OF EXISTING NDE TECHNOLOGY FOR ISI
- DESIGNATE SPECIFIC LOCATIONS, METHODS AND FREQUENCY OF PERIODIC INSERVICE INSPECTION

SUMMARY

- ADEQUATE EXAMS WILL BE PERFORMED DURING FABRICATION/PSI TO PROVIDE ASSURANCE THAT NO SIGNIFICANT FLAWS ESCAPE DETECTION
- BASELINE DATA WILL BE AVAILABLE IN THE EVENT ISI IS REQUIRED
- NDE TECHNIQUES WILL BE AVAILABLE TO SUPPORT INSERVICE MAINTENANCE, REPAIR AND MODIFICATION
- PERIODIC ISI WILL BE PERFORMED TO DETECT GENERIC DEGRADATION

STRUCTURAL MARGIN BEYOND THE DESIGN BASE

OVERALL TECHNICAL APPROACH

L. E. Strawbridge

Westinghouse Advanced Reactors Division

8064-12

APPROACH

- Hypothetical core disruptive accidents (HCDA's) are not design basis accidents
  - Design features prevent initiation of HCDA's
- Prudent margins beyond the design base incorporated to further reduce public risk
  - Structural margin beyond the design base (SMBDB)
  - Thermal margin beyond the design base (TMBDB)

7120-4

PURPOSE OF SMBDB

PROVIDE REASONABLE MARGINS ON IDENTIFIED COMPONENTS TO ACCOMMODATE DYNAMIC LOADS ASSOCIATED WITH HCDAs

- AVOID LARGE RELEASE OF VAPORIZED FUEL AND FISSION PRODUCTS
- AVOID LARGE RELEASE OF SODIUM

THESE MARGINS PREVENT SHORT-TERM CHALLENGE TO CONTAINMENT AND LARGE RADIOLOGICAL RELEASES

0605-2

FORM OF SMBDB REQUIREMENTS

Dynamic Load Requirements

- Maintain short term integrity of the reactor coolant boundary
- Loads derived from energetic fuel vapor expansion resulting in 101MJ energy release at time of sodium impact with the head

Leakage Requirements

- Limit short term sodium and gas leakage from reactor coolant boundary to avoid challenge to containment integrity

Geometric Requirements

- Provide clearances between components to avoid unacceptable interactions during dynamic loads

6054-117

DERIVATION OF SMBDB LOADS
(Performed in 1975)

Loads required for design prior to completion of extensive energetics analyses

Energetics information available

- Extensive analyses for FFTF
- Preliminary analyses for CRBRP
- Results indicated that even pessimistic assumptions generally resulted in average fuel vapor temperature of < 4300°K

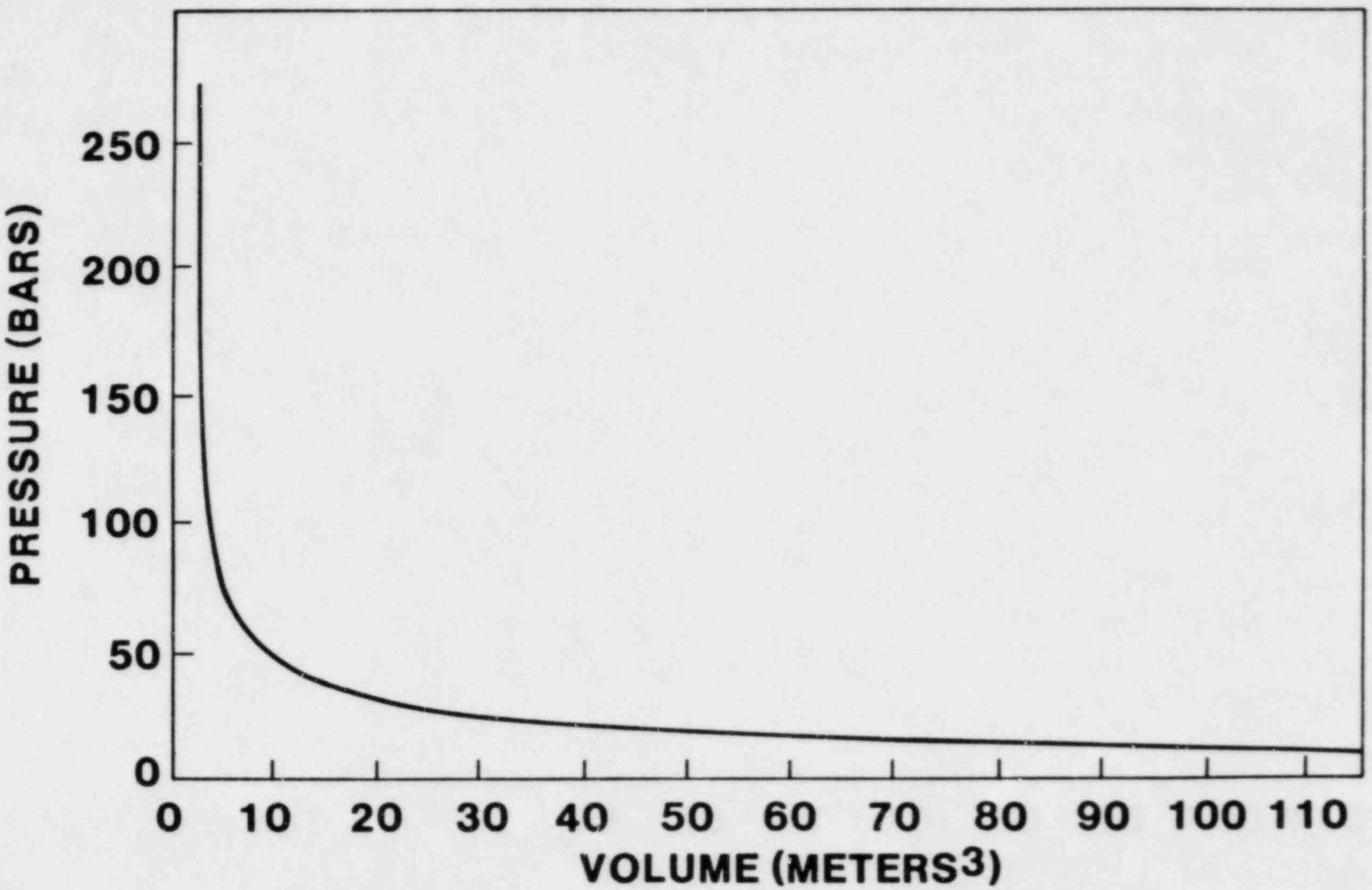
Parametric analyses assessed dynamic loads as a function of fuel vapor temperature

- Whole core involvement (liquid/vapor phases)
- Temperature distribution

Average vapor temperature of 4800°K selected as a basis for SMBDB loads

- Margin for reasonable uncertainties in data and models
- Margin to accommodate design evolution

PRESSURE-VOLUME RELATIONSHIP FOR SMBDB LOADINGS



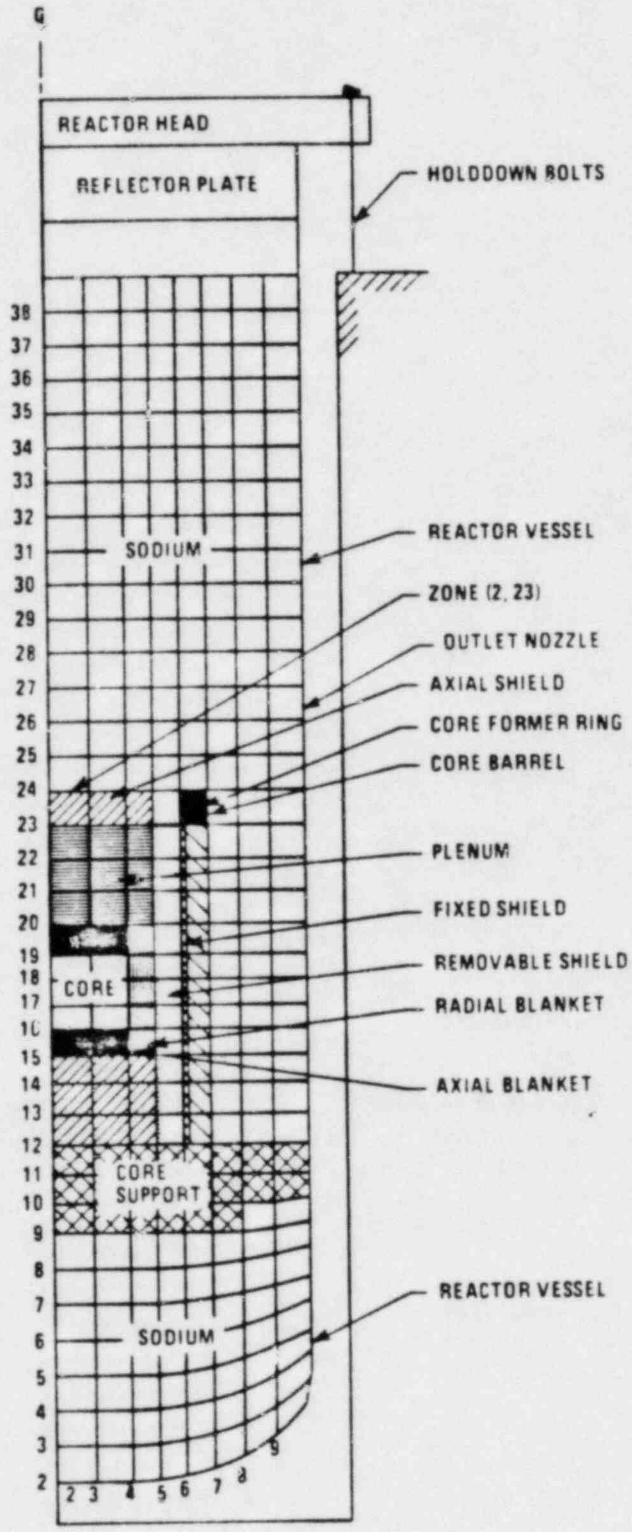
0805-5

THERMODYNAMIC CHARACTERISTICS OF SMBDB

Average fuel vapor temperature	4800°K
Maximum fuel vapor temperature	6030°K
Maximum pressure	273 bars
Initial fuel volume	2.56 m^3
Energy released in expansion to head impact	101 MJ
Energy released in expansion to 1 atmosphere	661 MJ

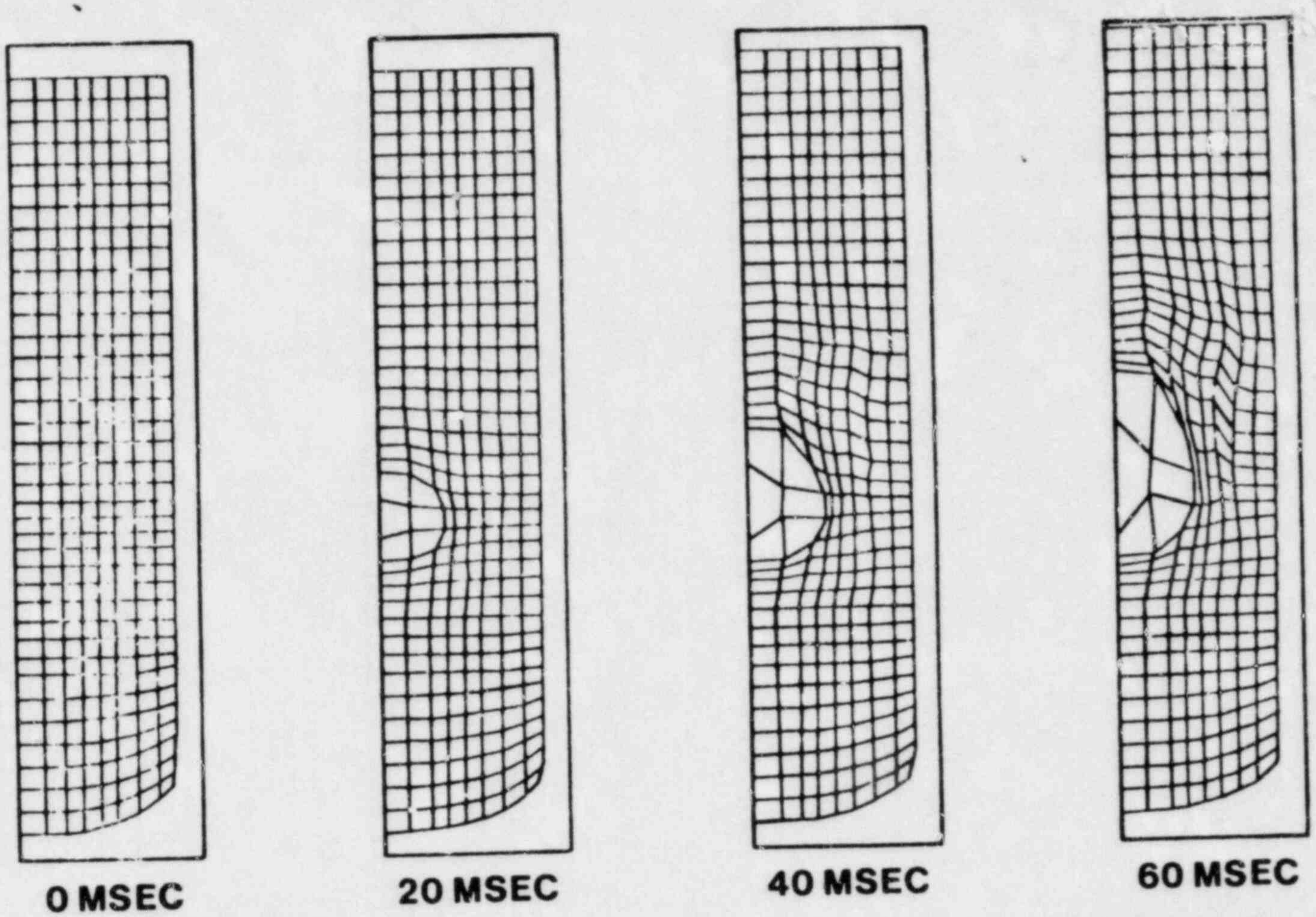
6054-15

REXCO-HEP MATHEMATICAL MODEL OF THE CRBRP REACTOR



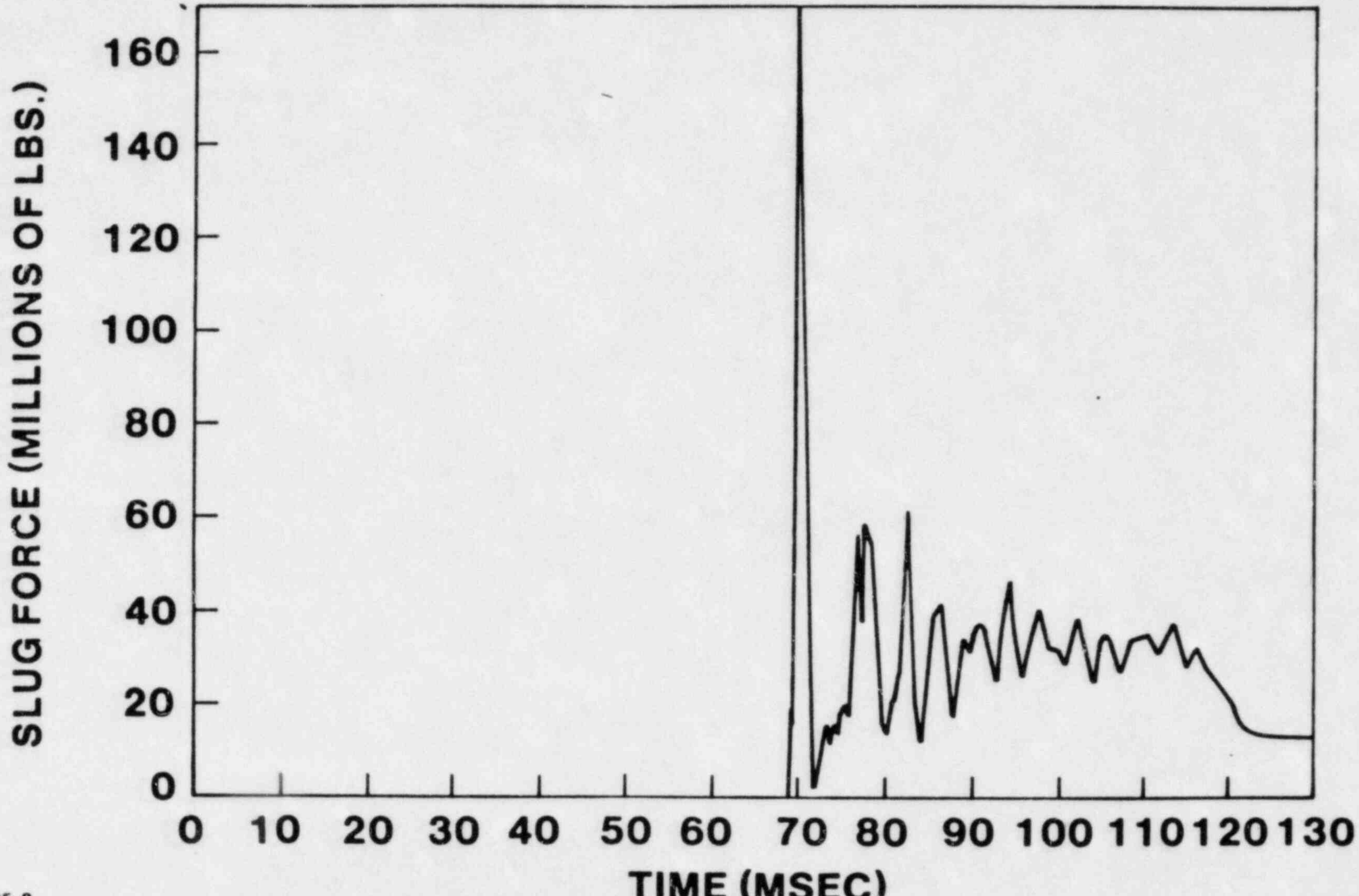
0605-6

CORE EXPANSION FOR SMBDB AS PREDICTED BY REXCO-HEP



0605-7

SLUG FORCE ON UNDER HEAD SHIELDING OF REACTOR VESSEL CLOSURE HEAD



0605-8

AREAS IN WHICH SMBDB LOAD REQUIREMENTS ARE SPECIFIED

Basic loads

- Reactor vessel
- Reactor vessel nozzles
- Reactor vessel closure head
- Core support structure
- Upper internals structure
- Reactor vessel support system
- In-vessel components

Loads derived from basic loads

- Head mounted components
- PHTS piping
- Pump
- Intermediate heat exchanger
- Check valve
- Overflow and makeup system
- Reactor cover gas system
- Impurity monitoring and analysis system

SPECIAL ASPECTS OF MARGIN BEYOND THE DESIGN BASE ANALYSES

- Not part of ASME code requirements
- Not included in stress report showing code compliance
- Stress analyses are provided in separate report
- May use relaxed acceptance criteria compared to ASME code, provided functional requirements are met
- SMBDB loads not combined with seismic loads

6054-110

CONSERVATISMS IN SMBDB LOADS

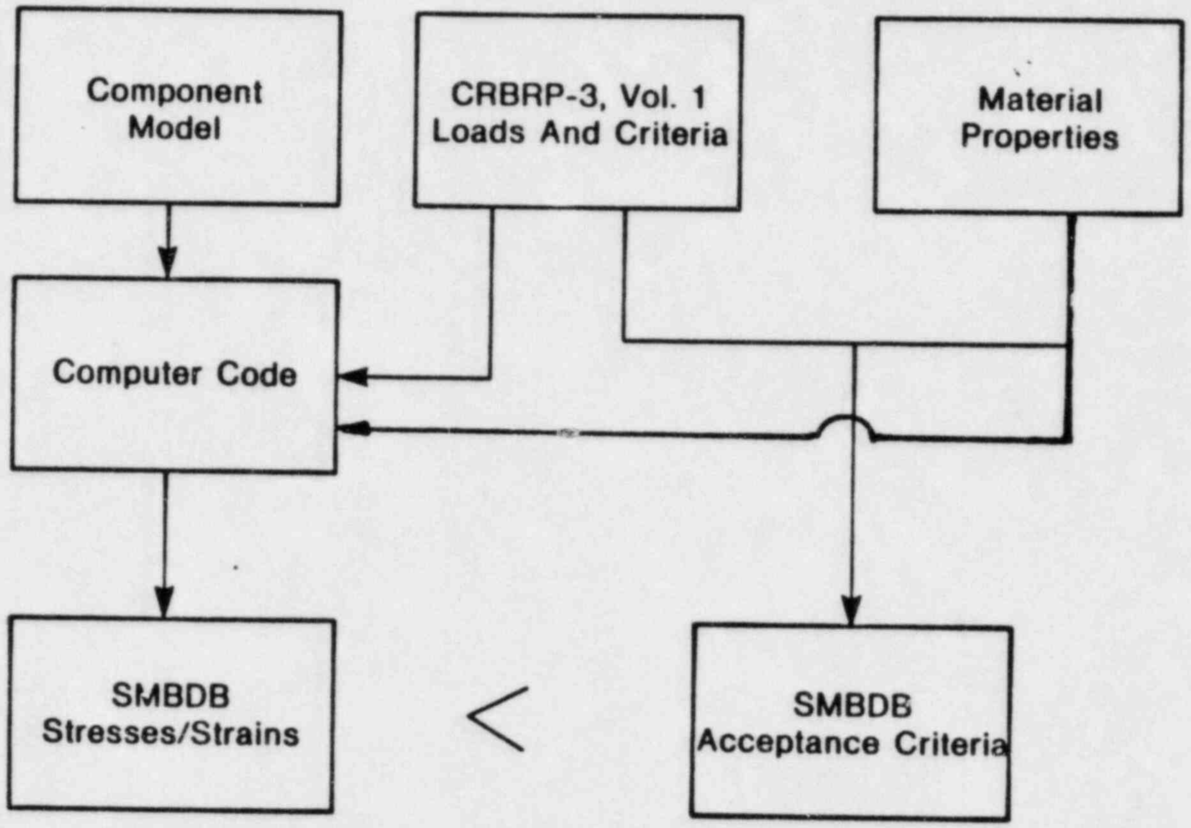
- Assumes low probability initiating conditions that are beyond the design base
- Loads derived from conditions that assume an energetic HCDA even though energetics not expected
- Isentropic fuel vapor expansion
- Attenuation due to upper internals structure ignored
- Heterogeneous core reduces potential for energetics

8054-7

SMBDB ASSESSMENTS

- Finite element elastic or inelastic analyses
- Scale model testing

COMPONENT ANALYSIS METHODOLOGY



Note: Some components are also tested under SMBDB conditions

6054-69

SMBDB ACCEPTANCE CRITERIA
(Ref: CRBRP-3, Vol. 1)

Features are:

- Some relaxation from ASME code criteria permitted
- Stress criterion to be used with elastic analysis
- Strain criterion to be used with inelastic analysis

6054-67

SCALE MODEL TESTING

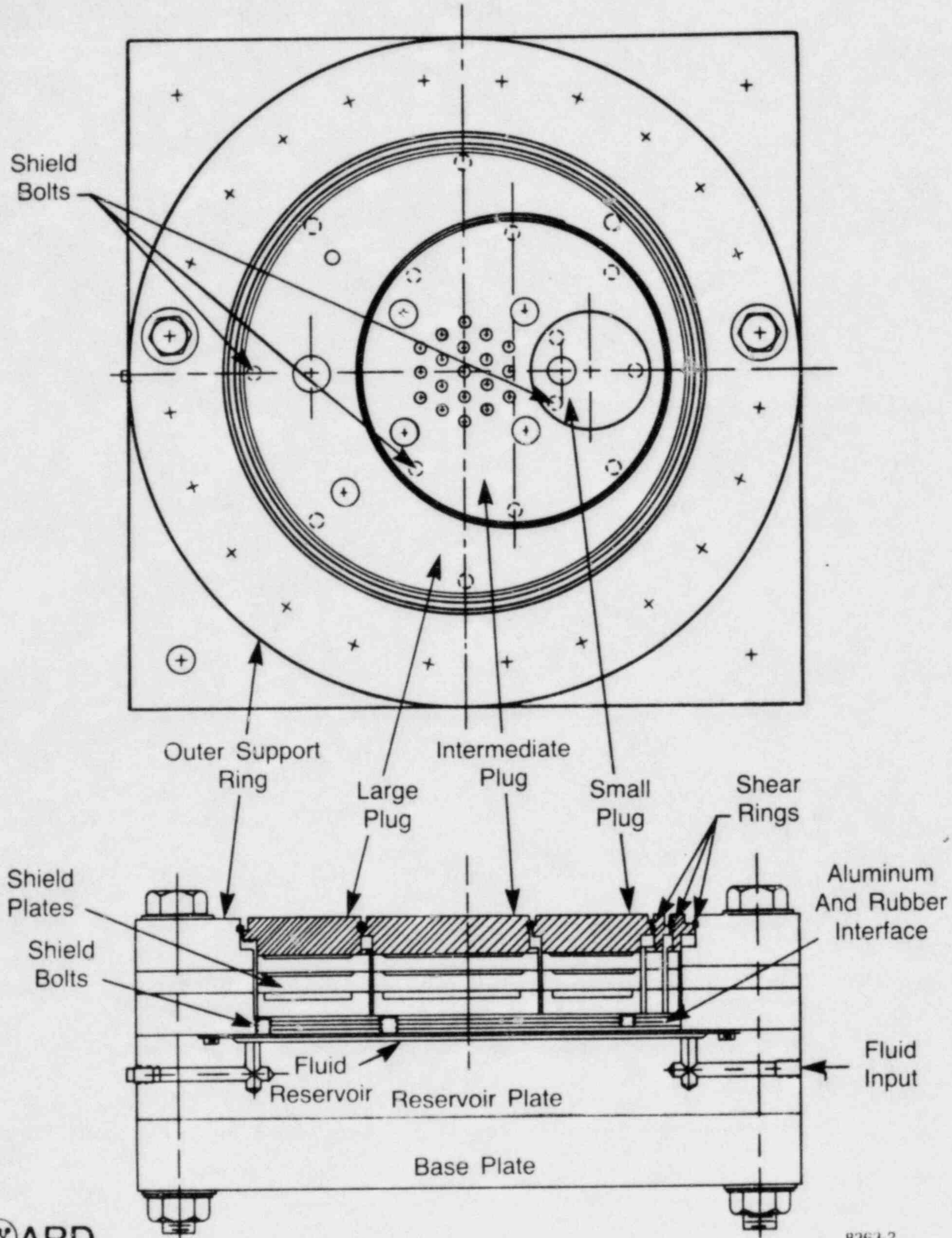
Objectives

- Assess ability of CRBRP scale models to withstand HCDA loads
- Provide understanding of response and interaction of reactor components
- Provide information to support methods verification

Summary of program to date

- Three static head tests performed
- Four dynamic tests performed with scaled SMBDB pressure-volume source

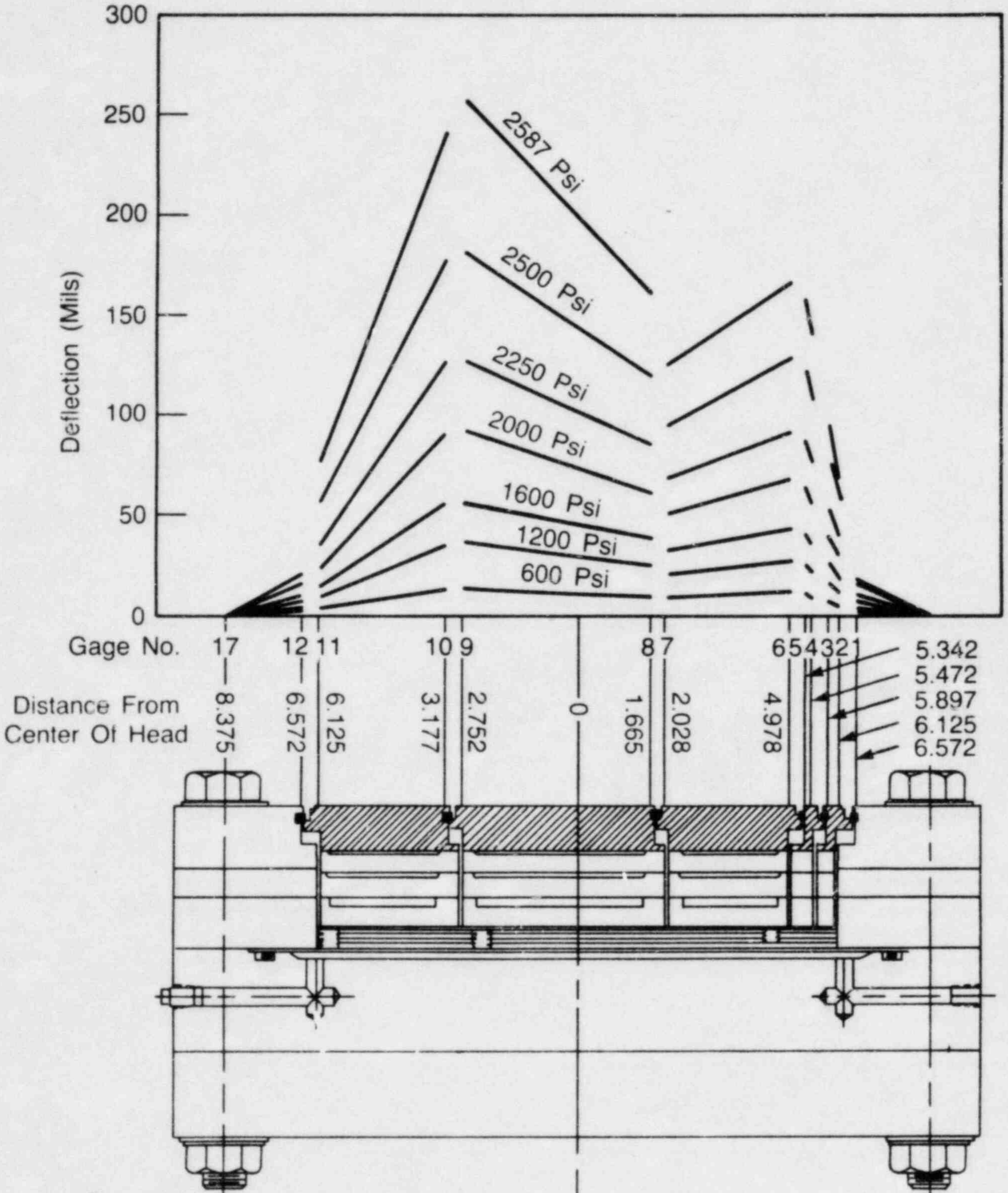
SCHEMATIC OF TEST FIXTURE FOR SM 7



ⓌARD

8363-2

PROFILES OF HEAD AT SEVEN PRESSURES FOR SM 7



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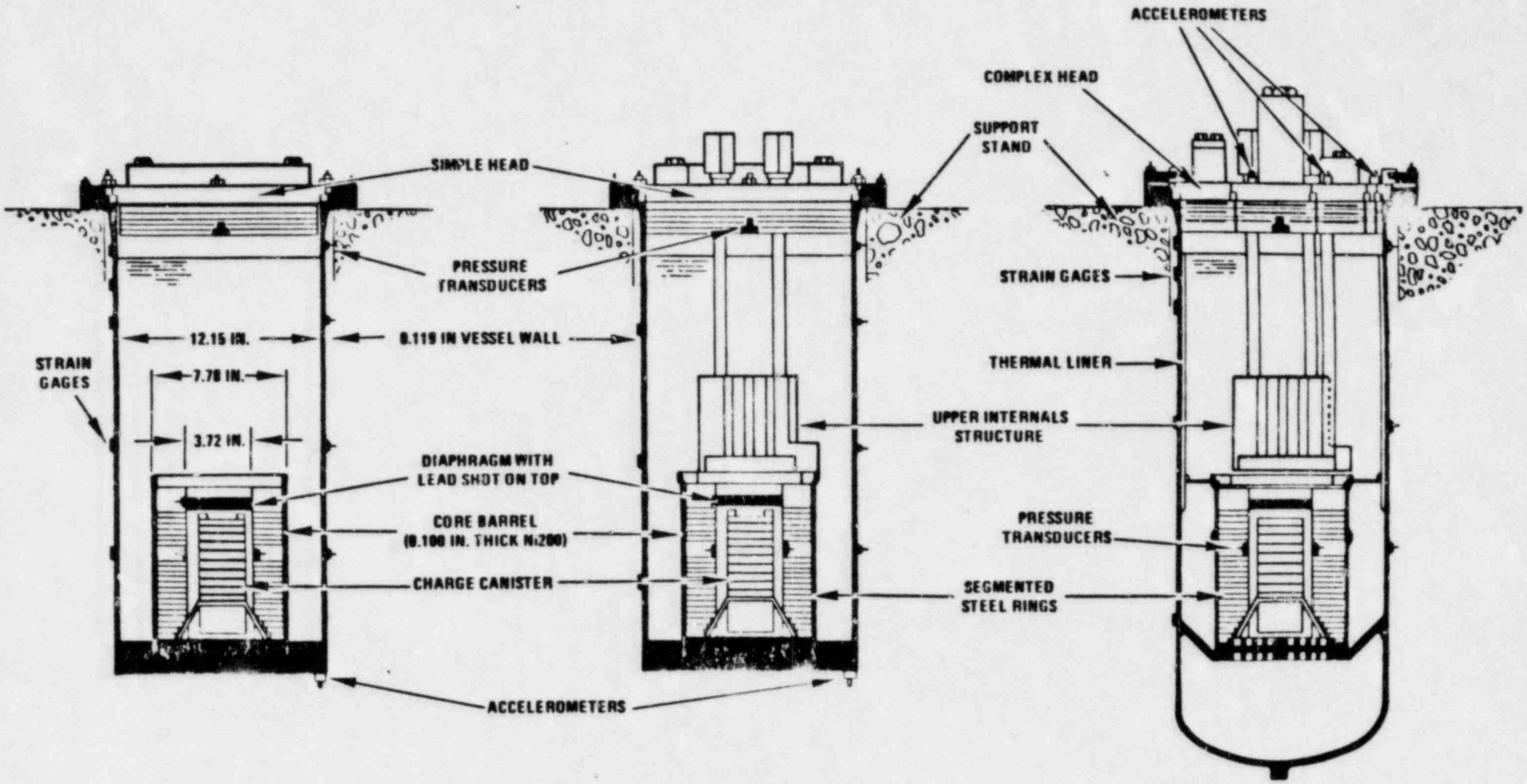
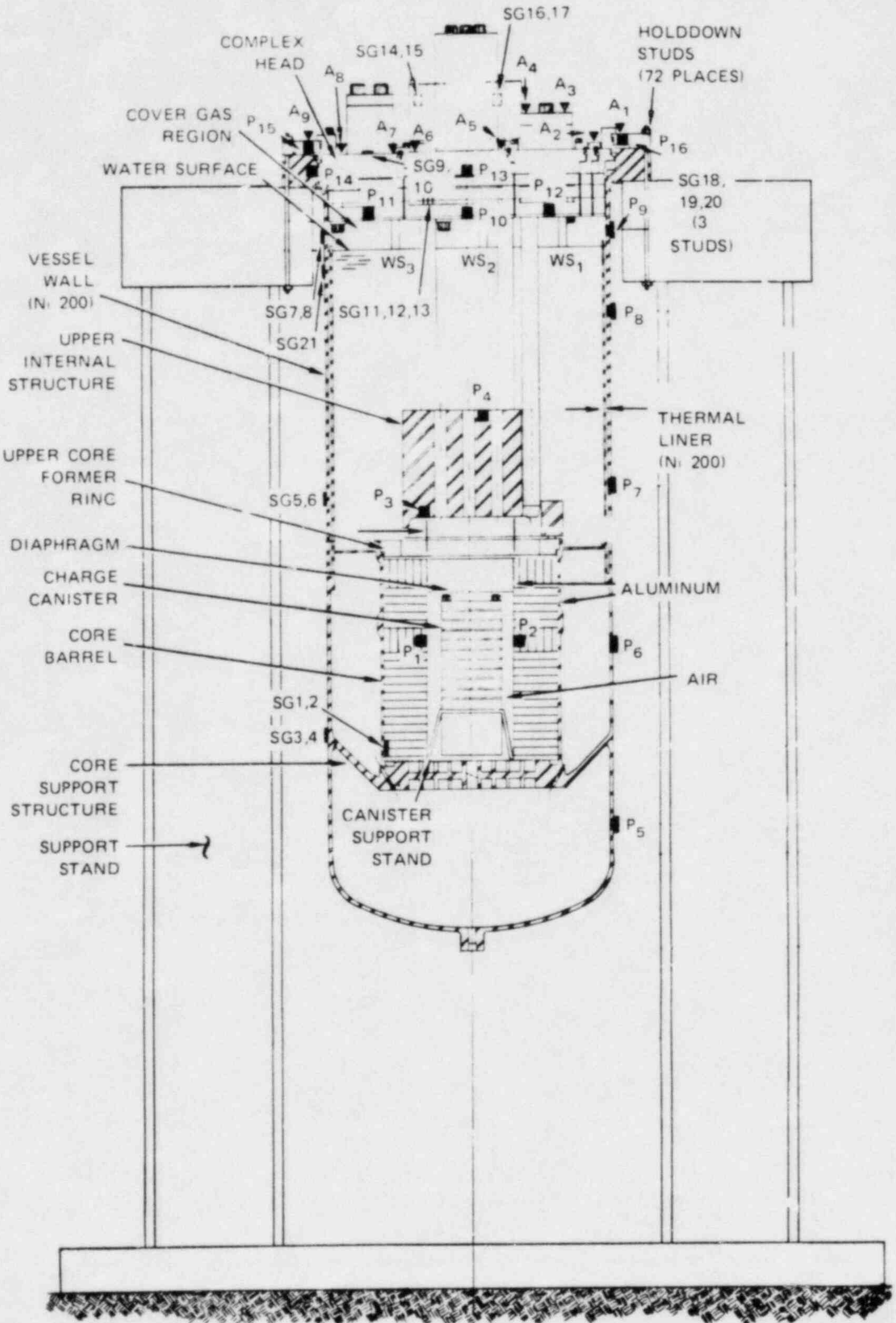
8363-1

SCALE MODEL TESTS



6054-71

SM-5 WITH INSTRUMENTATION



MA-3929-141A

CONCLUSIONS FROM SCALE MODEL TESTS

- Generally confirmed conservatism of methods used to predict dynamic loads
- Vessel and core barrel strains well below failure strains
- Response of upper internals structure (UIS) did not jeopardize boundary integrity
- UIS provides substantial mitigation of head loads
- Head failure mode was by disengagement at interface between large and intermediate plugs
- Head response sensitive to underhead shielding
- Although head plugs and margin rings remained elastic in dynamic tests, capability for SMBDB head load not proven by the tests because:
  - Non-prototypicalities in underhead shielding
  - Head load lower than required load because of mitigation by UIS

SMBDB SUMMARY

- SMBDB loads have been established
- SMBDB loads are used in component design
- Component analyses in progress
- Scale model test program confirmed adequacy of many areas
- Adequacy of head still needs to be confirmed by further design analyses and testing

CRBR LOOSE PARTS MONITORING SYSTEM

STAFF PRESENTATION TO ACRS - 3/16/83

T. KING

SUMMARY OF STAFF REVIEW

- APPLICANTS INITIALLY PROPOSED NO LOOSE PARTS MONITORING SYSTEM.

- STAFF DID NOT FIND ANY COMPELLING REASON TO EXCLUDE SUCH A SYSTEM FROM CRBR AND; THEREFORE, PER SRP SECTION 4.4 SUCH A SYSTEM IS CONSIDERED APPLICABLE TO CRBR.

- APPLICANTS HAVE NOW COMMITTED TO THE DESIGN, INSTALLATION AND OPERATION OF A LPMS PER R.G. 1.133 GUIDELINES.

MAJOR DESIGN CRITERIA

R.G. 1.133 GUIDELINES - LOCATE TWO SENSORS AT EACH NATURAL COLLECTION POINT OF PRIMARY SYSTEM.

- BE SENSITIVE ENOUGH TO DETECT LOOSE PARTS LARGE ENOUGH TO CAUSE DAMAGE
- DESIGN SYSTEM FOR CHANNEL SEPARATION AND OBE
- PROVIDE AUDIBLE ALERT TO OPERATORS
- PROVIDE FOR TESTABILITY AND CALIBRATION

CRBR SENSOR LOCATIONS - REACTOR VESSEL

- PHTS & IHTS PUMPS
- IHXS
- SGS
- NATURAL COLLECTION POINTS OF SYSTEM

COMPONENT NOISE AND VIBRATION MEASUREMENTS TO LOOK FOR DEGRADATION.

EXPERIENCE BASE

EBR-II - SUCCESSFUL TESTING OF HIGH TEMPERATURE IN-SODIUM MICROPHONES.

FFTF - INSTALLATION AND OPERATION OF LPMS FOR REACTOR VESSEL.

- EXPERIENCE WITH HIGH TEMPERATURE ACCELEROMETERS IN-VESSEL.

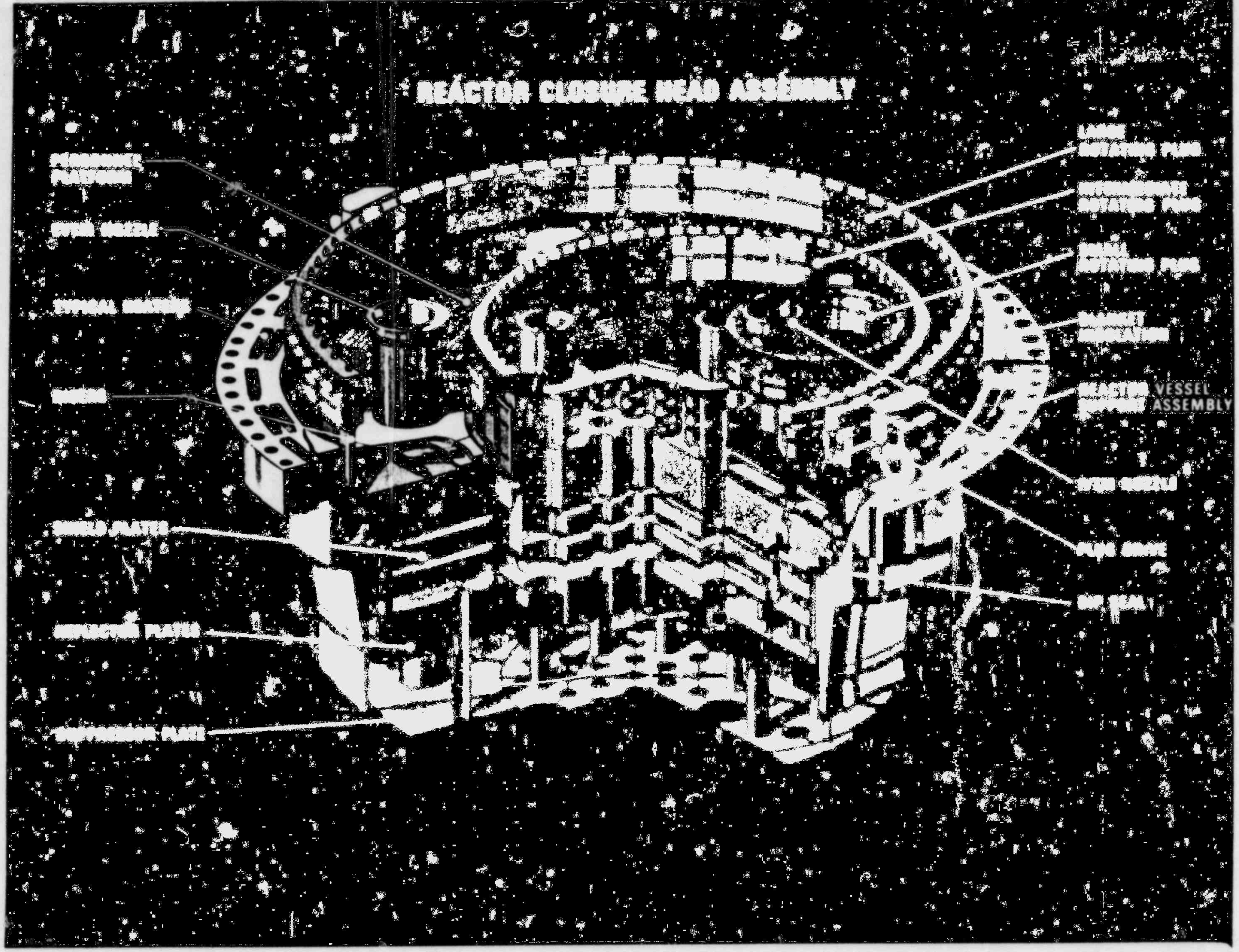
CONCLUSION

COMMITMENT BY APPLICANTS TO DESIGN, INSTALL AND OPERATE A LPMS FOR CRBR IN ACCORDANCE WITH R.G. 1.133 IS ACCEPTABLE FOR CP.

CRBRP

CLOSURE HEAD SMBDB CAPABILITY

- BACKGROUND
- EVALUATION OF TEST RESULTS
- POTENTIAL HEAD MODIFICATIONS
- ENERGY ABSORPTION CAPABILITY OF THE MODIFIED HEAD
- PLANNED TESTS
- SUMMARY


REACTOR CLOSURE HEAD ASSEMBLY
VESSEL ASSEMBLY

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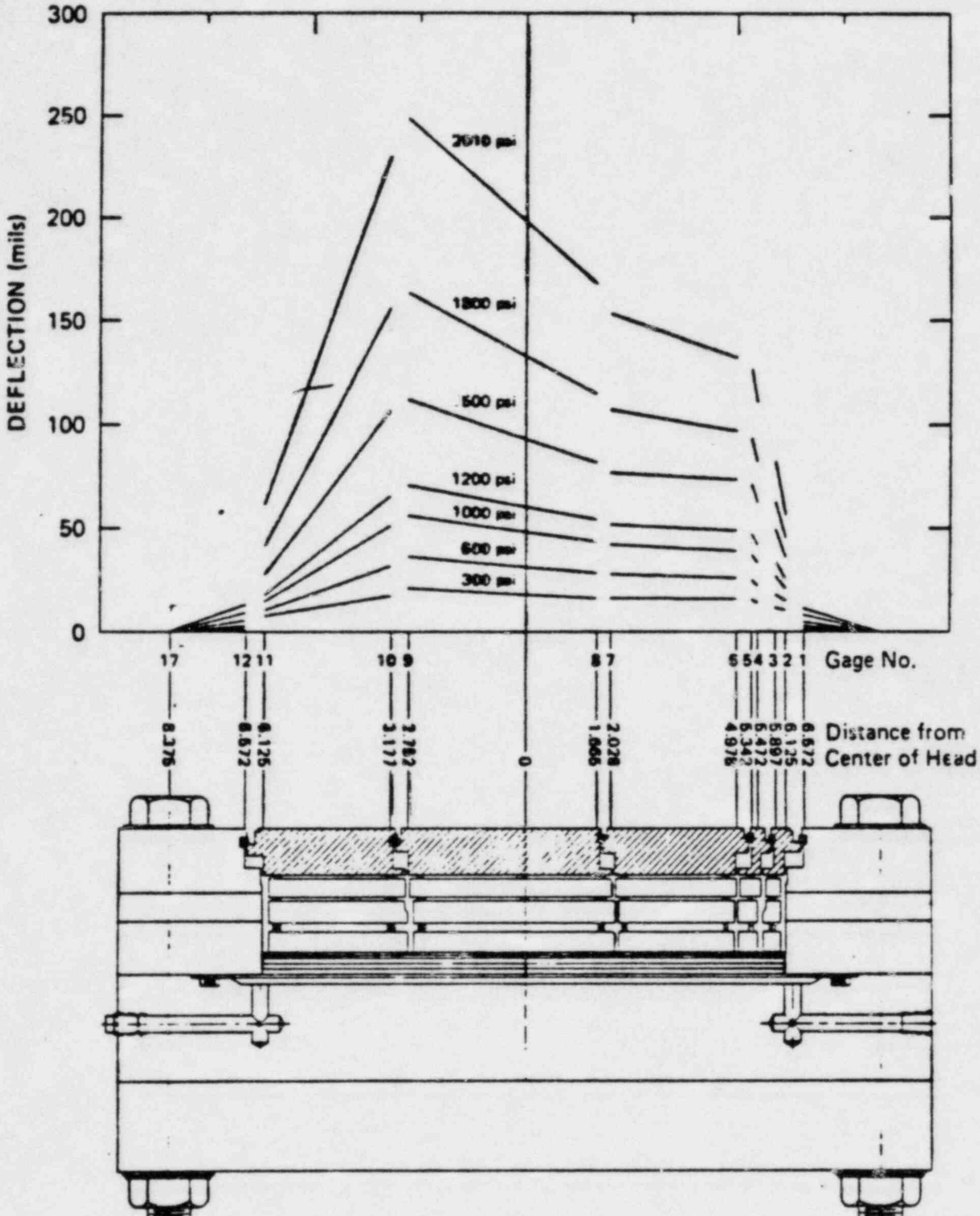
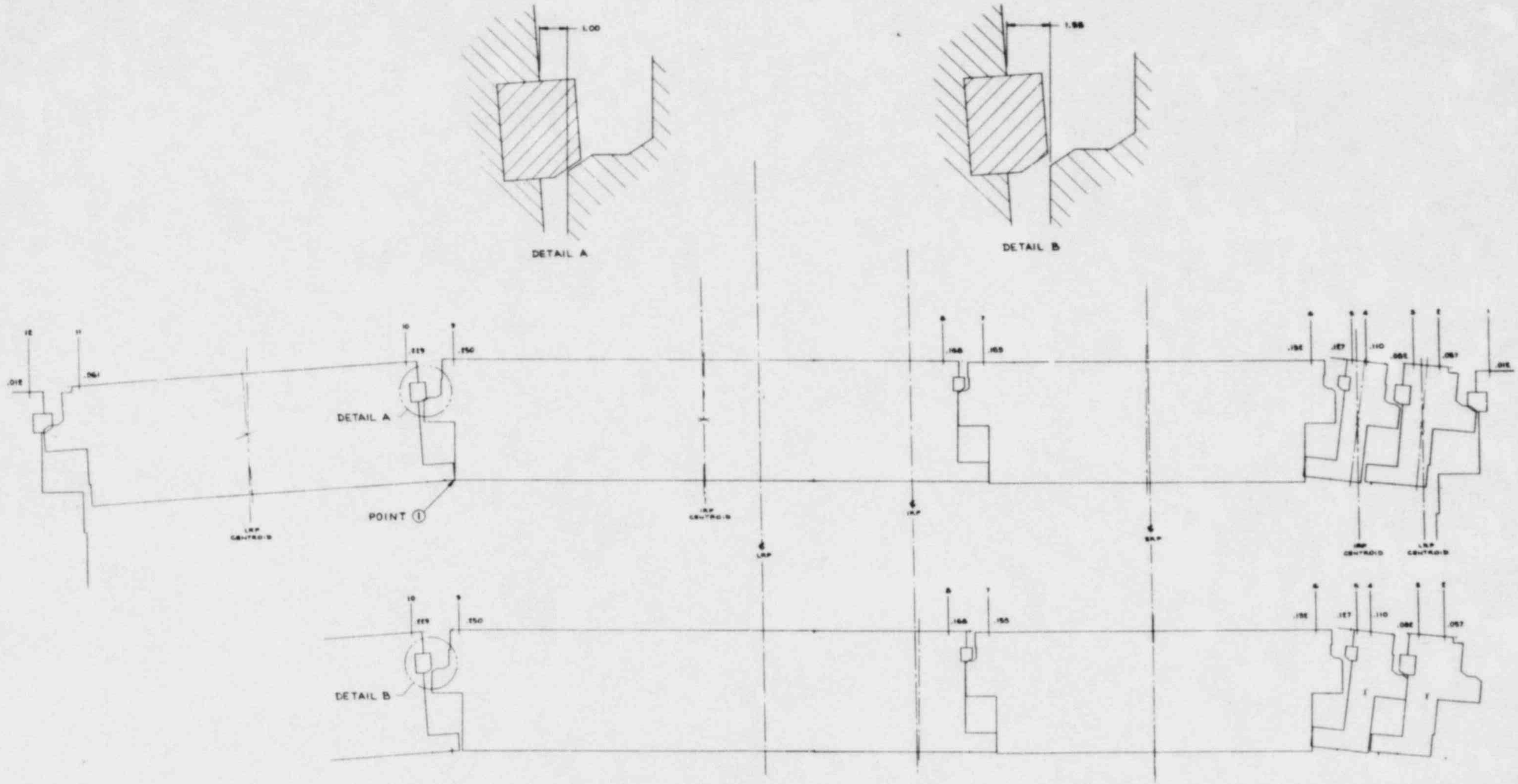


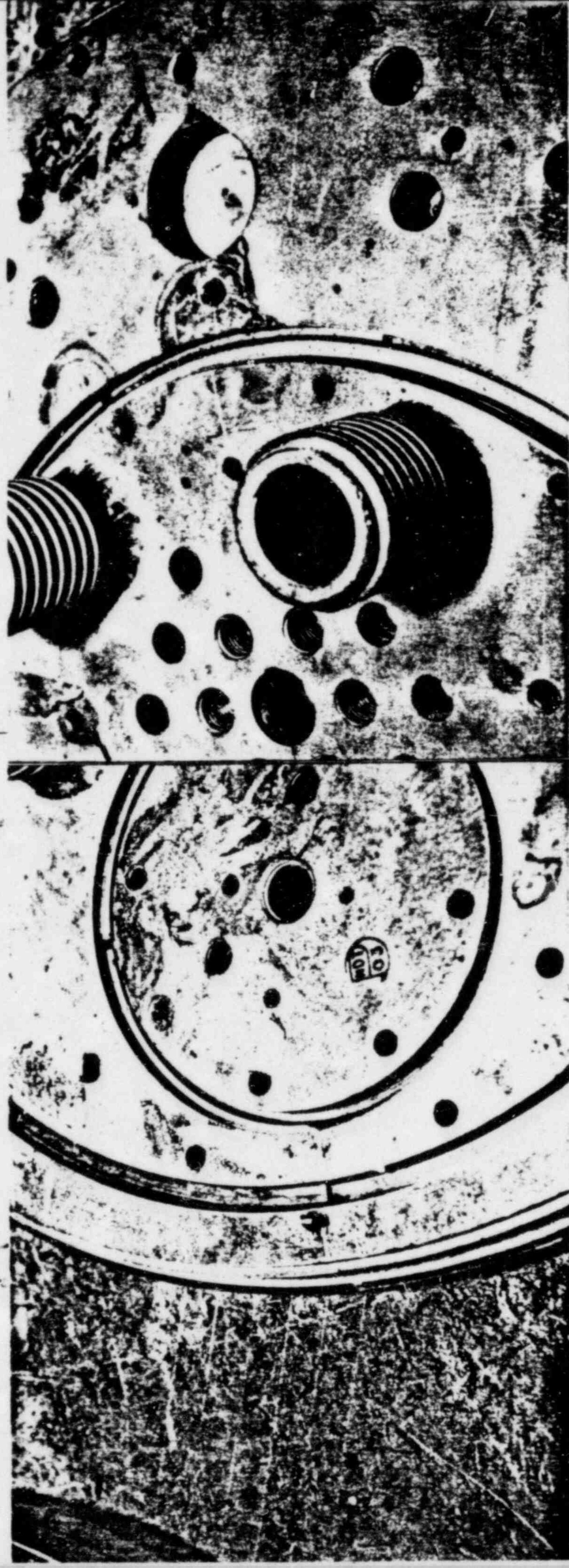
Figure 1
Profiles of Head at Seven Pressures on SM 8

ⓌARD



SM-8 MODEL
PLUG CONFIGURATION
AT FAILURE (2010 PSI)

FIGURE 2



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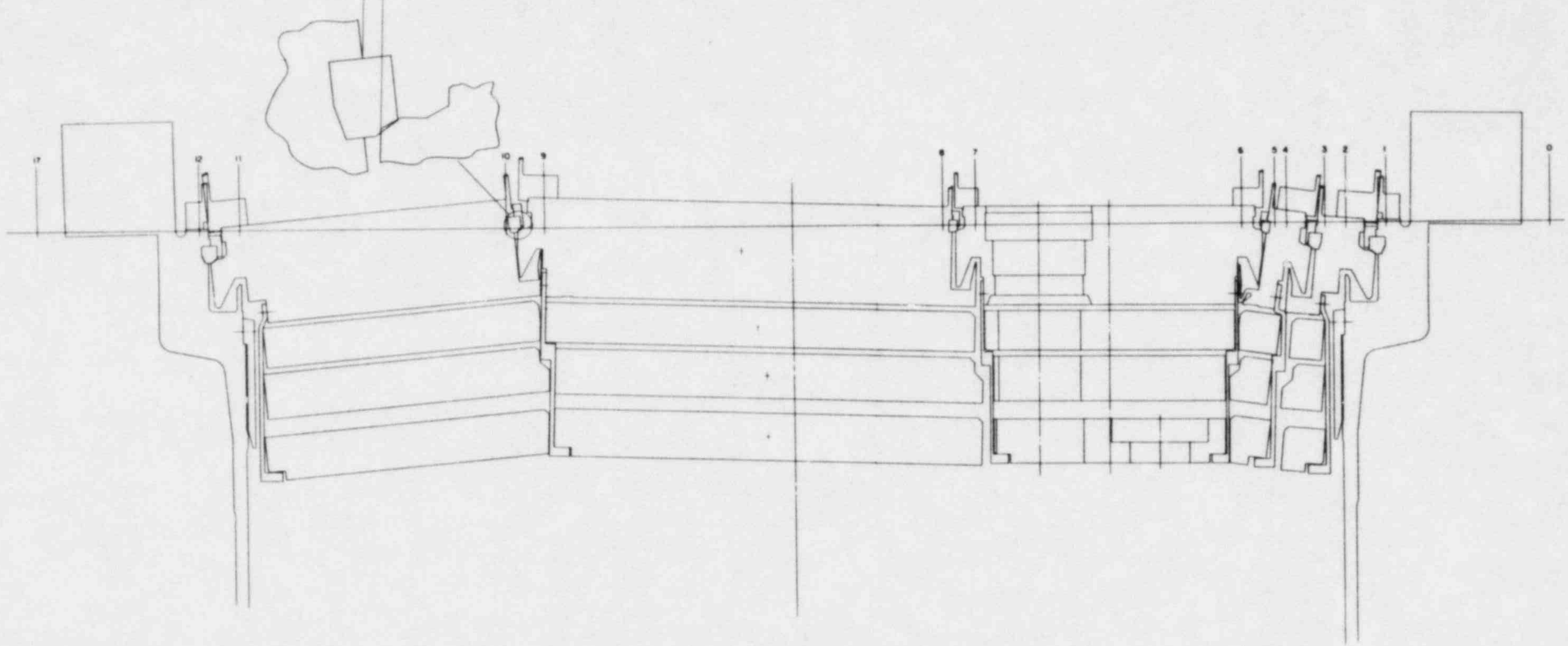
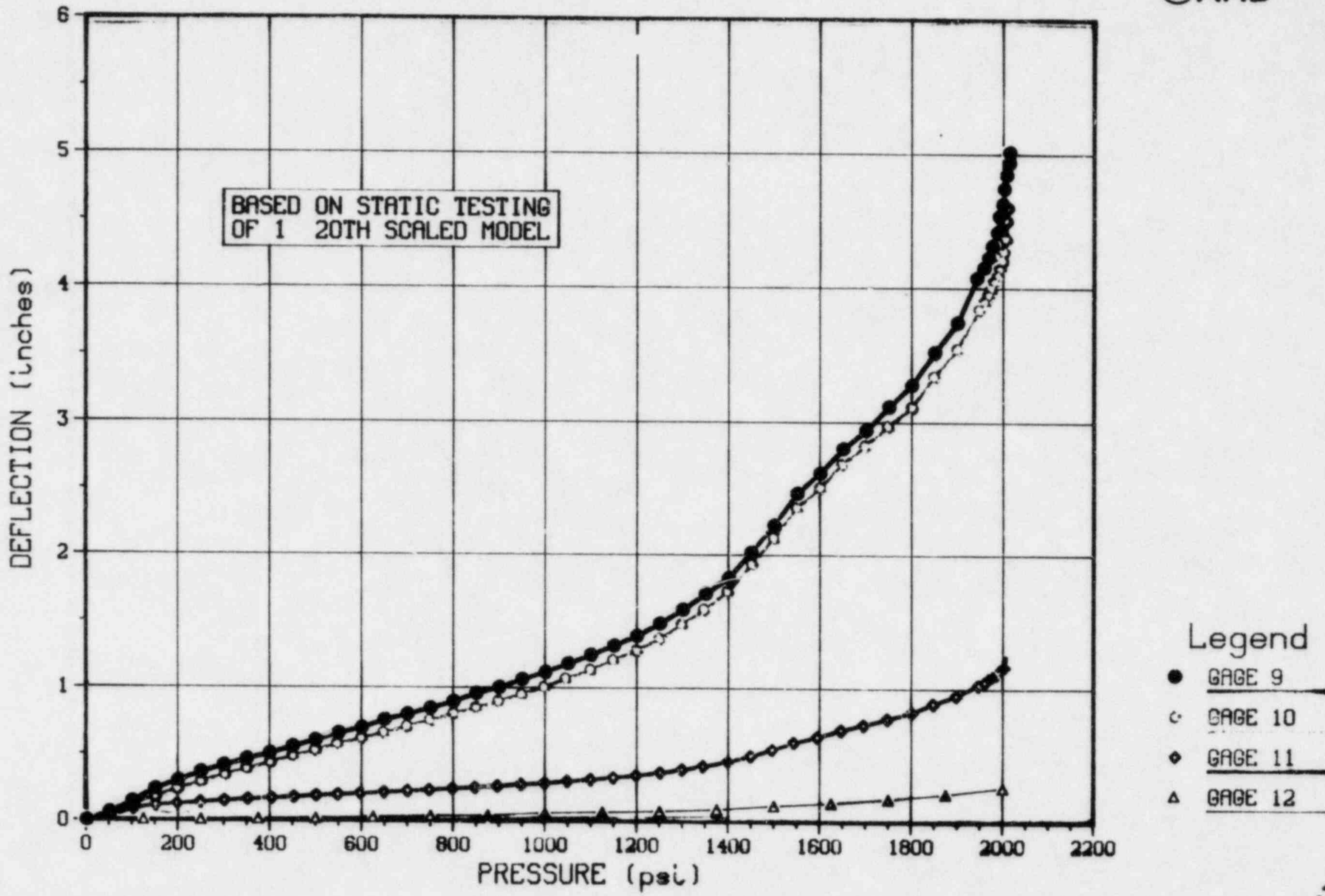


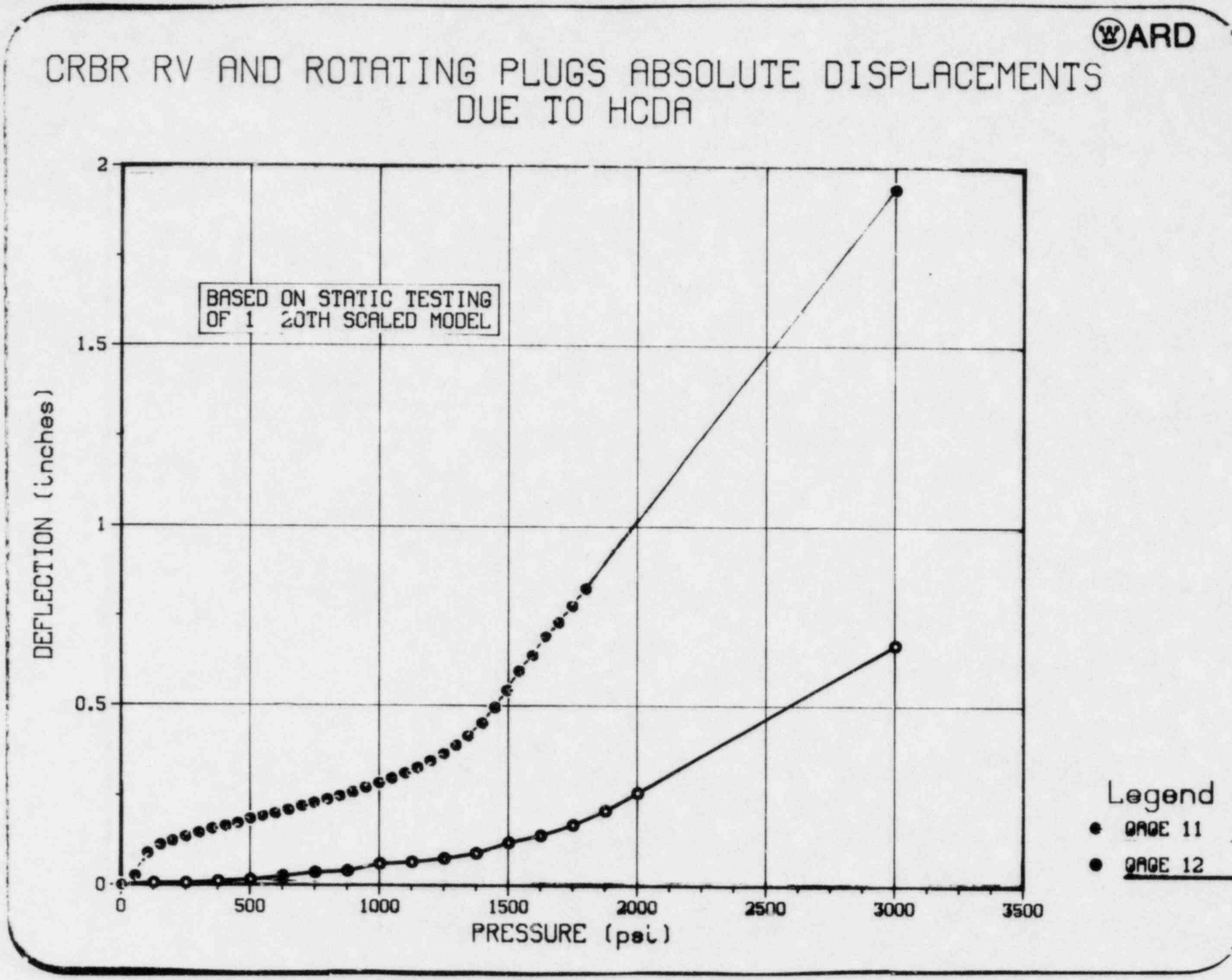
PROJECTED CLOSURE HEAD DEFLECTIONS AT
2700 PSI WITH PROTOTYPE RADIAL RESTRAINT
AT THE LRP-REACTOR VESSEL INTERFACE

FIGURE 11

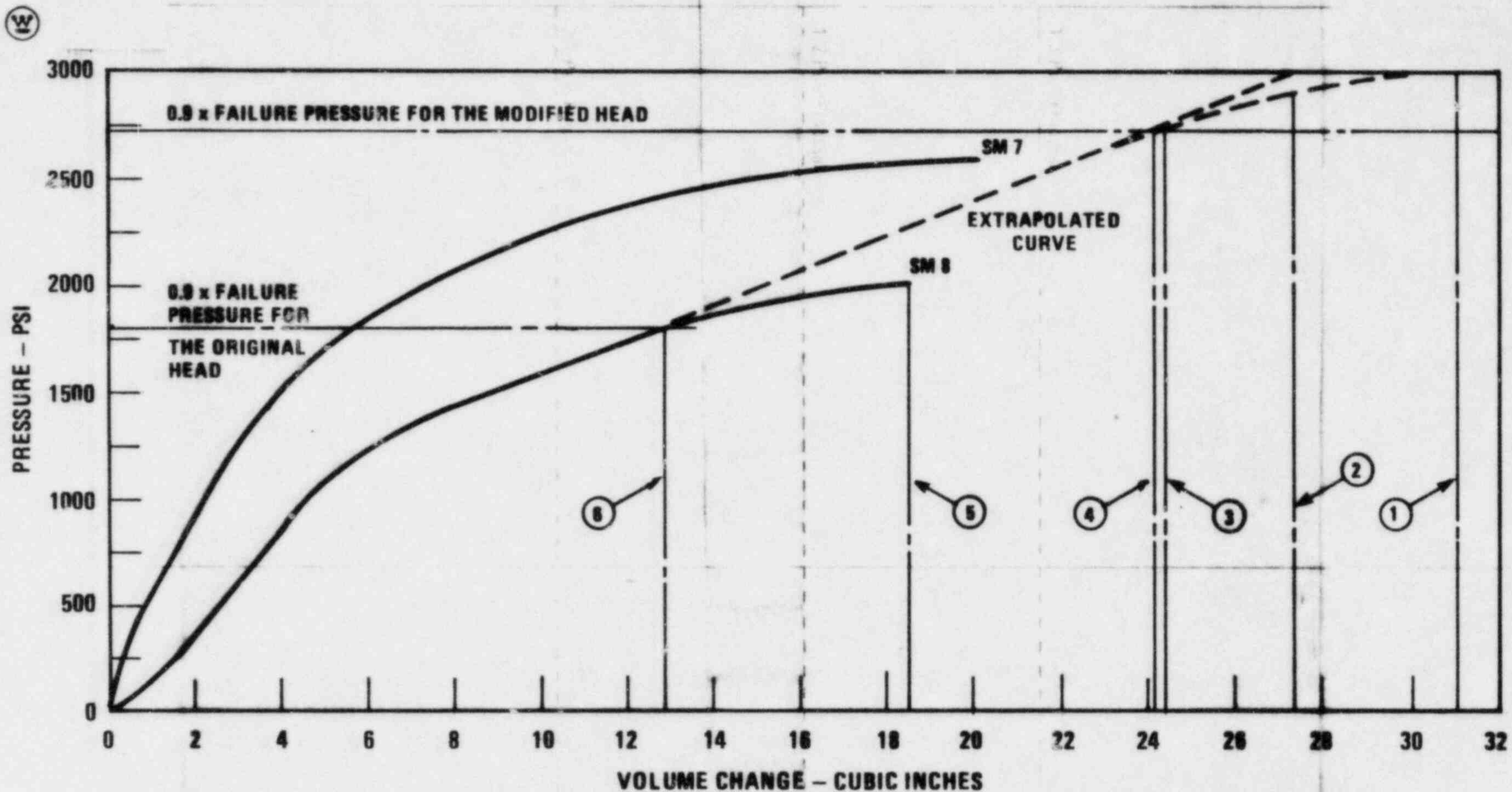
CRBR RV AND ROTATING PLUGS ABSOLUTE DISPLACEMENTS DUE TO HCDA

ⓌARD





ⓌARD
CRBR RV AND ROTATING PLUGS ABSOLUTE DISPLACEMENTS
DUE TO HCDA
BASED ON STATIC TESTING
OF 1 20TH SCALED MODEL
DEFLECTION (inches)
2
1.5
1
0.5
0
0
500
1000
1500
2000
2500
3000
3500
PRESSURE (psi)
Legend
GAGE 11
GAGE 12



① VOLUME CHANGE FOR THE MODIFIED HEAD WITH NON LINEAR DISENGAGEMENT (NLD)

② VOLUME CHANGE FOR THE MODIFIED HEAD WITH NO NON LINEAR DISENGAGEMENT

③ USEABLE VOLUME CHANGE FOR THE MODIFIED HEAD WITH NLD

④ USEABLE VOLUME CHANGE FOR THE MODIFIED HEAD WITH NO NLD

⑤ VOLUME CHANGE FOR THE ORIGINAL HEAD WITH NLD

⑥ USEABLE VOLUME CHANGE FOR THE ORIGINAL HEAD

Figure 24
Pressure-Volume Change for Original & Modified Head Designs

ADDITIONAL SOURCES OF ENERGY ABSORPTION

- MODEL PRESSURE AREA CORRECTION — 36×10^6 LB.IN
- HEAD POTENTIAL ENERGY — 5×10^6 LB.IN
- REACTOR SUPPORT STRUCTURE STRAIN ENERGY — 11×10^6 LB.IN
- TOTAL ADDITIONAL SYSTEM ENERGY ABSORPTION — 52×10^6 LB.IN (5.9 MJ)
- EQUIVALENT SLUG KINETIC ENERGY — 22.5 MJ

M_S = SLUG MASS ($W_S = 0.368 \times 10^6$ LB)

M_H = HEAD MASS ($W_H = 1.046 \times 10^6$ LB)

$M_{H\&S}$ = SLUG MASS + HEAD MASS

V_S = SLUG VELOCITY

$V_{H\&S}$ = VELOCITY OF THE COMBINED SLUG & HEAD MASS

KE_S = SLUG KE AT IMPACT = 75 MJ

$KE_{(H\&S)}$ = ESTIMATE OF THE KE DELIVERED TO THE HEAD

FOR CONSERVATION OF MOMENTUM: -

$$V_{M\&S} = V_S\left(\frac{M_S}{M_S + M_H}\right) \qquad (1)$$

$$KE_S = \left(\frac{M_S\, V_S^{\,2}}{2}\right) \qquad (2)$$

$$KE_{H\&S} = \left(\frac{M_S + M_H}{2}\right) \times V_S^{\,2}\left(\frac{M_S}{M_S + M_H}\right)^2$$

$$KE_{H\&S} = \frac{V_S^{\,2}\ M_S^{\,2}}{2(M_S + M_H)} = KE_S\left(\frac{M_S}{M_S + M_H}\right) \qquad (3)$$

$$KE_{H\&S} = \frac{75 \times 0.368}{(1.046 + 0.368)} = 19.5 \text{ MJ}$$

$$KE_{H\&S} = 19.5 \times 8.85 \times 10^6 = 173 \times 10^6 \text{ LB.IN.}$$

Figure 21
Head Energy Input

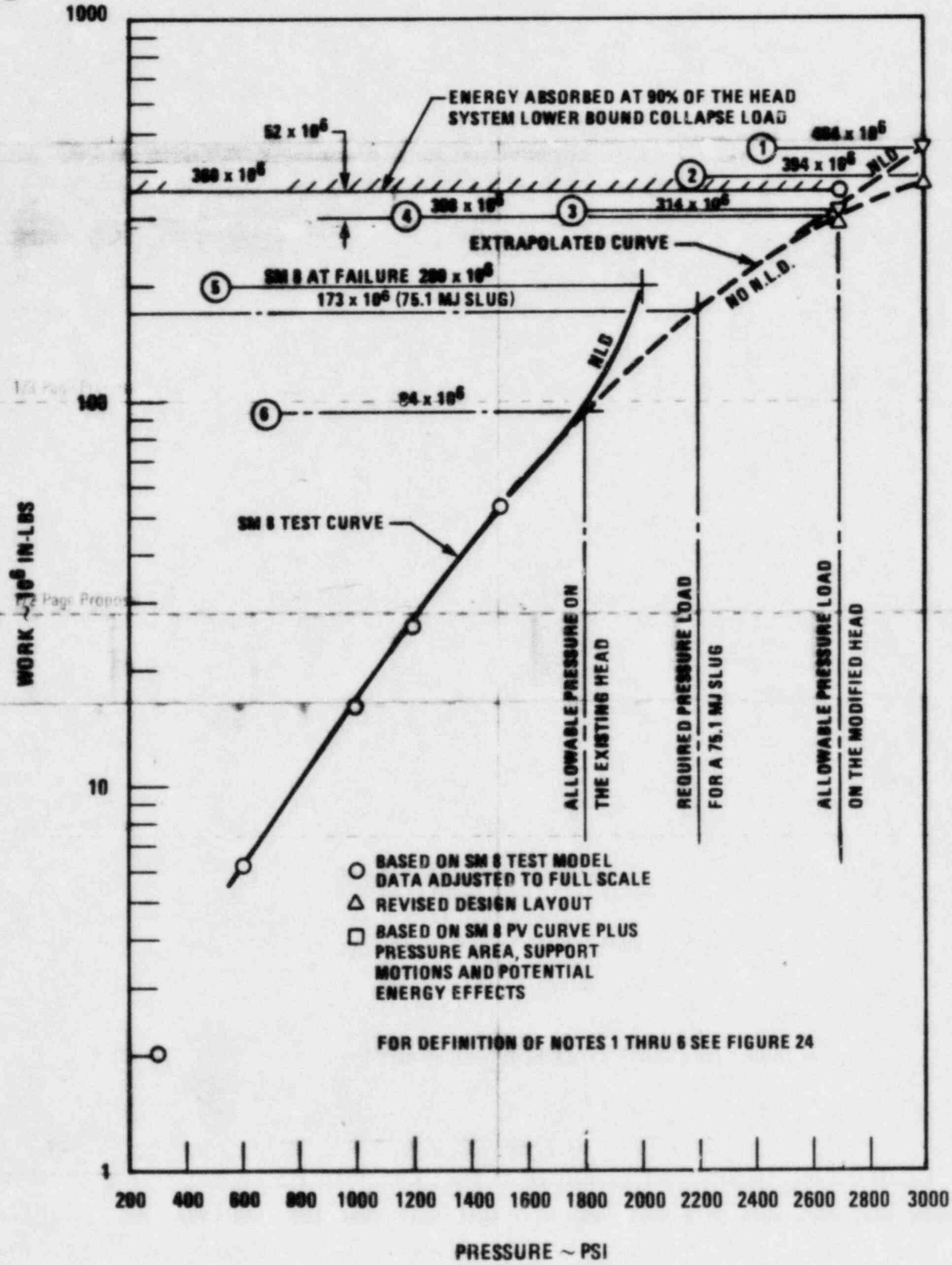


Figure 25
Head System Energy Absorbtion Capability

PLANNED TESTS

TEST NO.	LOADING	MODEL CONFIGURATION
SM-9	STATIC PRESSURE	EXISTING HEAD GEOMETRY WITH PROTOTYPIC REPRESENTATION OF RISERS, KEEPER RINGS AND SHIELD PLATE SUPPORT CYLINDERS
SM-10	STATIC PRESSURE	MODEL FEATURES AS IN SM-9 PLUS MODIFICATIONS TO ELIMINATE PREMATURE KINEMATIC DISENGAGEMENT OF THE SHEAR RINGS
SM-11	75 MJ SLUG IMPACT	MODEL CONFIGURATION TO BE DETERMINED BY RESULTS FROM TESTS SM-9 AND SM-10

TEST SM-11 WILL BE COMPLETED IN CY 1984

HEAD CONFIGURATION	LOADING CONDITION	HEAD PRESSURE PSI	ENERGY ABSORBTION FROM THE EXTRAPOLATED TEST P-V CURVE LB-IN	ADDITIONAL HEAD SYSTEM ENERGY CAPACITY (1) LB-IN	TOTAL HEAD SYSTEM ENERGY CAPACITY LB-IN
EXISTING HEAD	COLLAPSE LOAD	2010	200×10^6	~	~
EXISTING HEAD	90% COLLAPSE LOAD	1800	94×10^6	~	$\geqslant 94 \times 10^6$ (40.8 MJ)
MODIFIED HEAD (2)	COLLAPSE LOAD	3000	394×10^6	~	~
MODIFIED HEAD (2)	90% COLLAPSE LOAD	2700	308×10^6	$\geqslant 52 \times 10^6$ (2)	360×10^6 (156.3 MJ)

THE MODIFIED HEAD ENERGY ABSORBTION CAPABILITY
CORRESPONDS TO A SLUG ENERGY OF 156 MJ

NOTES:

1. THE ADDITIONAL HEAD SYSTEM ENERGY ABSORBTION CAPACITY DERIVES FROM (a) HEAD SYSTEM POTENTIAL ENERGY (b) HEAD SUPPORT SYSTEM STRAIN ENERGY AND (c) ADJUSTMENTS TO ACCOUNT A SMALL NON PROTOTYPIC REPRESENTATION OF THE HEAD PRESSURE AREA IN THE SM 8 MODEL
2. THE ADDITIONAL HEAD SYSTEM ENERGY ABSORBTION CAPACITY WAS ESTIMATED FOR THE MODIFIED HEAD ONLY. SMALLER VALUES WOULD APPLY TO THE EXISTING HEAD DESIGN

Figure 30
Closure Head System Energy Absorbtion Summary

SMBDB BACKGROUND

- PURPOSE OF SMBDB (STRUCTURAL MARGINS BEYOND DESIGN BASIS)

  ASSURE THAT DURING A CDA, AND IMMEDIATELY FOLLOWING, THE REACTOR CLOSURE HEAD AND HEAD MOUNTED COMPONENTS WILL NOT CHALLENGE CONTAINMENT ABOVE THE OPERATING FLOOR. THE CHALLENGES TO CONTAINMENT FROM SMBDB ARE MISSILE GENERATION ABOVE THE OPERATING FLOOR OR DIRECT SODIUM SPRAY CONTAINING CORE FISSION PRODUCTS OUTSIDE OF THE HEAD ACCESS AREA.

SMBDB BACKGROUND (CONTINUED)

- EVALUATION

  THE APPLICANTS AND THE STAFF HAVE HAD NUMEROUS WORKING MEETINGS AND SEVERAL HUNDRED QUESTIONS RELATING TO THIS SUBJECT INCLUDING THREE APPLICANT FURNISHED DOCUMENTS LRA-1109, WARD-2108 CRBRP-3 VOLUME 1 REVISION 4 AND SRI SCALE MODEL TESTS SM-1 THROUGH SM-8.

  THE STAFF'S TECHNICAL ASSESSMENT OF THE INFORMATION PROVIDED BY THE APPLICANTS FURNISHING THE ABOVE REPORTS, LETTERS TO NRC AND MEETINGS WITH THE STAFF DECEMBER 9, 1982 AND FEBRUARY 14, 1983 WILL BE GIVEN IN A CONCLUSIONARY VIEW GRAPH.

STAFF CONCLUSION

SUBJECT TO THE FOLLOWING COMMITMENTS BEING MET BY THE APPLICANTS:

- UPON SATISFACTORY COMPLETION OF CONFIRMATORY TESTS SM-9 AND SM-10 WE BELIEVE THAT THE REACTOR CLOSURE SYSTEM WILL COMPLY WITH 10 CFR 50.34 PART iii RELATING TO "PROVIDING REASONABLE ASSURANCE THAT THE FINAL DESIGN WILL CONFORM TO THE DESIGN BASIS WITH ADEQUATE MARGIN FOR SAFETY" HAS BEEN MET FOR STRUCTURAL MARGINS BEYOND THE DESIGN BASIS.
- 75 MJ* KE IN THE SODIUM SLUG AT TIME OF IMPACT AT THE BOTTOM OF PLUG
- PLUG MODIFICATION AS NECESSARY TO ACCOMMODATE 75 MJ*
- SCALE MODEL TEST PROGRAM SM-9 AND SM-10 WITH A) AND B) OPTION
- SODIUM RELEASES INTO HEAD ACCESS AREA
- REVISE CRBRP-3 AND INCLUDE LOAD PATH FROM VESSEL FLANGE THROUGH REACTOR CAVITY WALL

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* REFERENCE LETTER DATED FEBRUARY 14, 1983.

ACRS SUBCOMMITTEE MEETING
ACCOMMODATION OF LOCAL FUEL FAULTS
AND ROLE OF INSTRUMENTATION

March 16, 1983

Westinghouse Electric Co.
Advanced Energy Systems Division
Madison, PA 15663

R. A. Markley
A. L. Schwallie
L. E. Strawbridge
R. W. Tilbrook
R. J. Tinder

INTRODUCTION

INTRODUCTION

- Summarize Local Faults Technical Information
  - Provide Description of Fuel Design Characteristics
  - Review Pertinent Data Base and Conclusions on Fuel Failure Propagation
  - Provide Description of Fuel Failure Monitoring System Response
  - Intended Use and Utility of Core Outlet Thermocouples
  - Foreign and Domestic Experience on PPS Instrumentation Relative to Local Faults

TECHNICAL SUMMARY ON LOCAL FAULTS

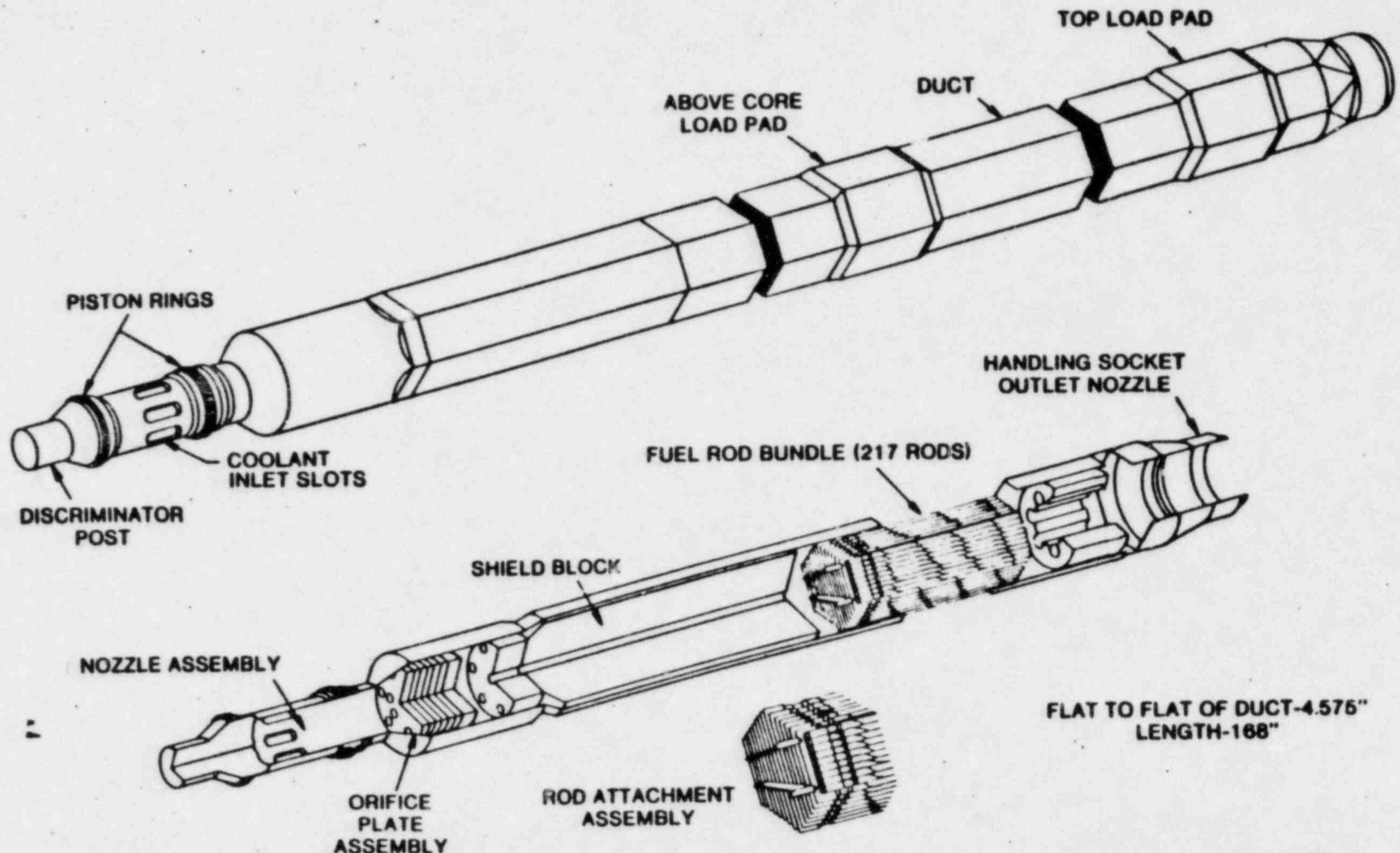
- Rapid Rod-to-Rod Failure Propagation Due to Local Faults Leading to a Condition of Loss of Coolable Geometry is Incredible. It has not Occurred in Run Beyond Cladding Breach (RBCB) Testing nor in Operating Reactors with Failed Fuel.

- Instrumentation to Alert the Operator to Local Fault Conditions is Available in CRBRP, and Considering the Long Time Intervals Necessary for Rod-to-Rod Propagation, Provides Sufficient Warning for Operator Corrective Action.

- Since Rapid Rod-to-Rod Propagation is Incredible, Requirements for Local Fault Detection and Protection Against Fuel Failure Propagation in the PPS are not Considered Necessary.

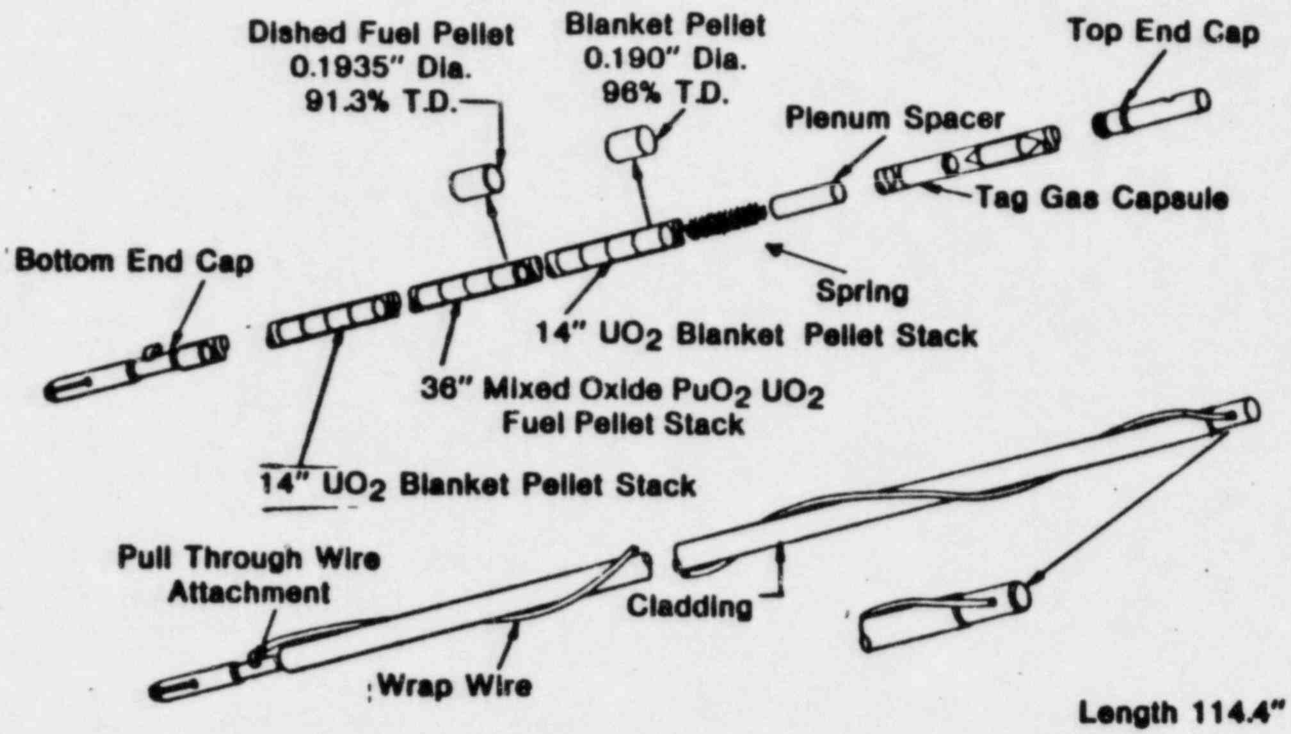
FUEL DESIGN OVERVIEW

CRBRP FUEL ASSEMBLY



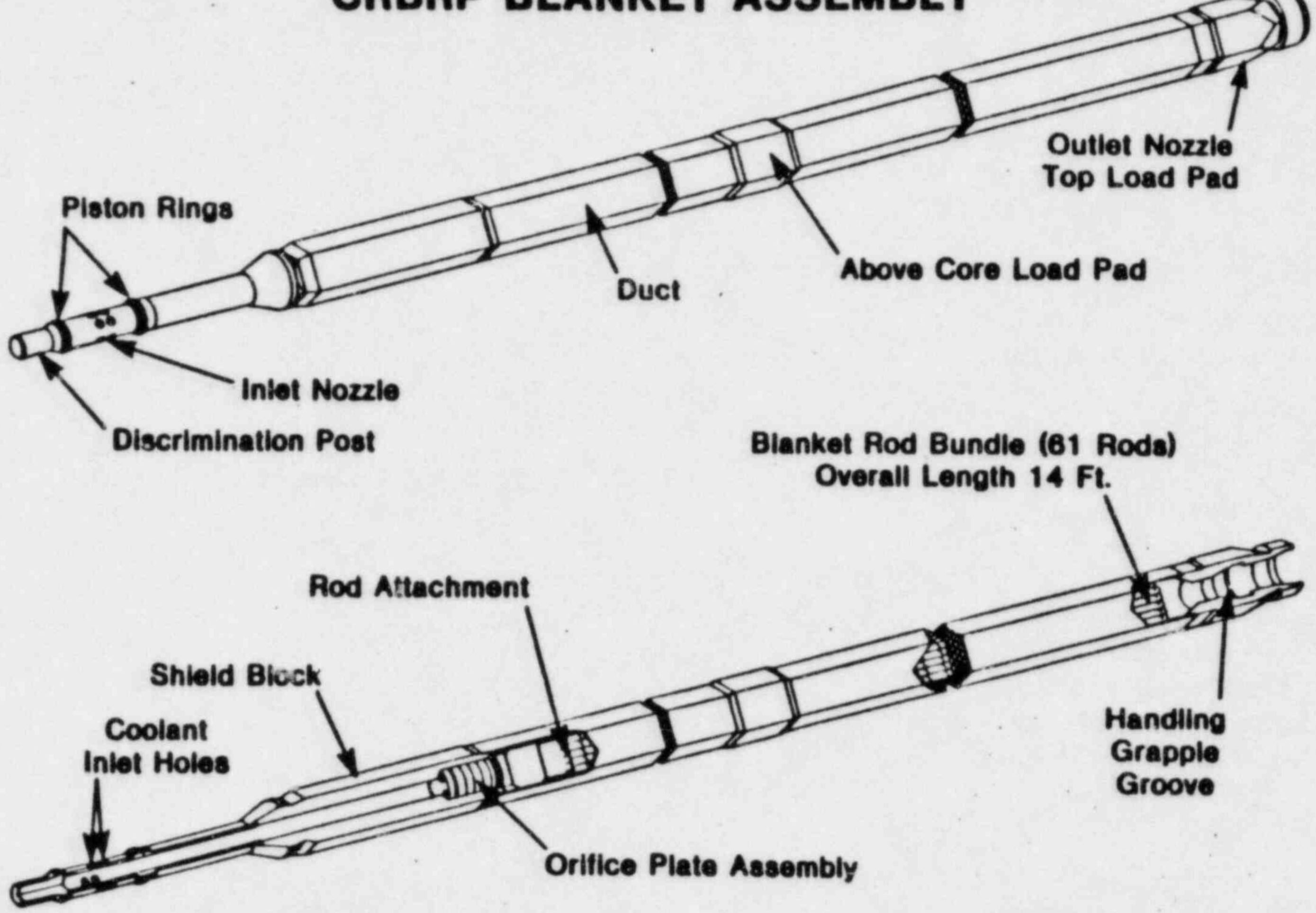
FLAT TO FLAT OF DUCT-4.575"
LENGTH-168"

CRBRP FUEL ROD



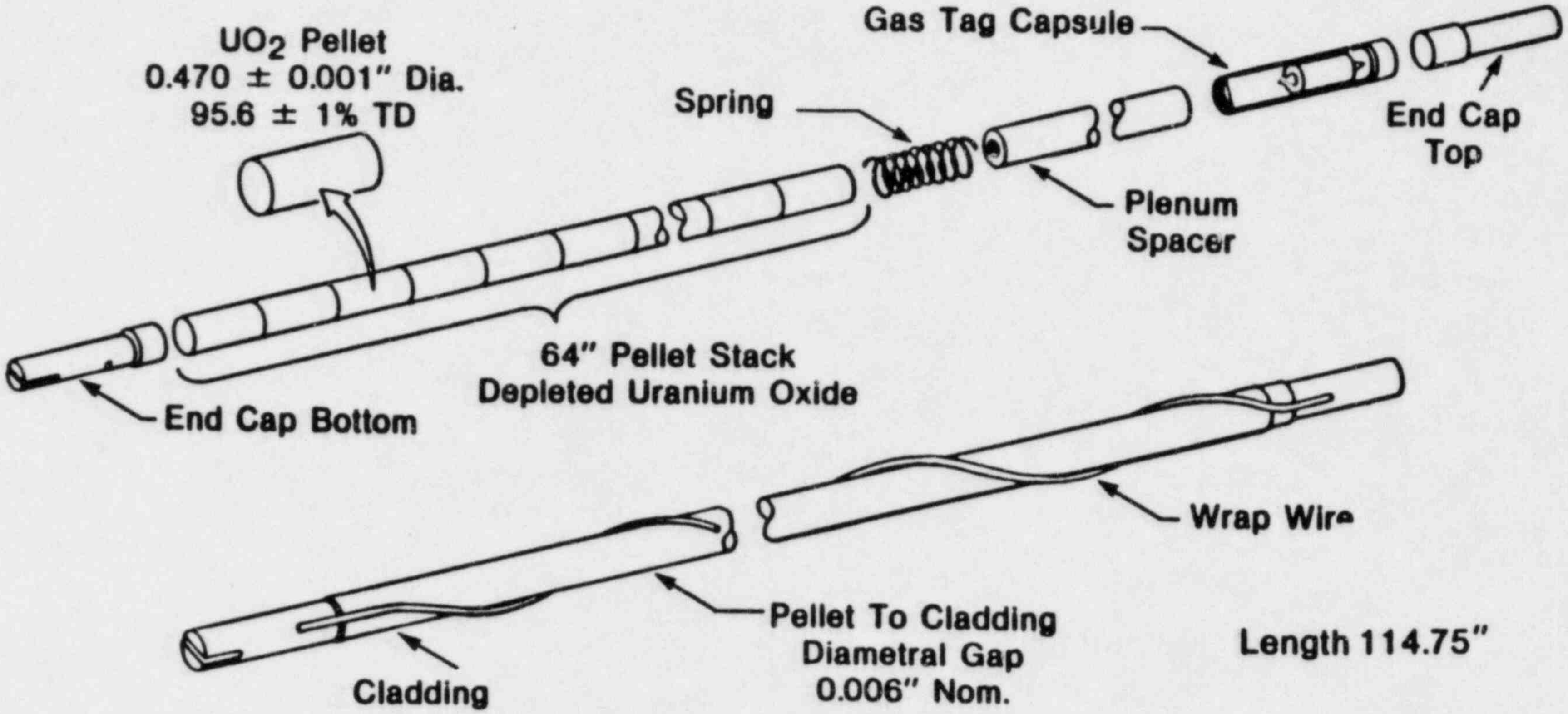
8115-1

CRBRP BLANKET ASSEMBLY



6115-5

CRBR BLANKET ROD



6112-28

DAMAGE SEVERITY LIMITS

Event Category	Damage Severity Level	
Normal Operation	No Significant Loss of Effective Lifetime	
Anticipated Events (Upset)	No Reduction of Effective Lifetime Below the Design Values	Design Limits Preclude Mechanistic Failures for Normal Operation Plus Anticipated Events and the Worst Unlikely Event Including Uncertainties.
Unlikely Events (Emergency)	A General Reduction in the Fuel Burnup Capability and, at Most, a Small Fraction of Fuel Rod Cladding Failures	
Extremely Unlikely Events (Faulted)	Maintain Coolable Configuration	No Sodium Boiling, Limited Fuel Melting, Cladding Solidus*

*PSAR Guideline

FUEL FAILURE MONITORING SYSTEM

FUEL FAILURE MONITORING SYSTEM (FFMS)

Cover Gas Monitoring System (CGMS)

Intent

- Provide Detection of Rod Failures in Fuel and Blanket Assemblies
- Provide Cover Gas Analysis and Activity Measurement
- Provide Operator with Information to Initiate Operation of the Failed Fuel Location System (FFLS)

Failed Fuel Location System (FFLS)

Intent

- Locate Failed Fuel and Blanket Assemblies
- Characterize Failures

Delayed Neutron Monitoring System (DNMS)

Intent

- Provide Capability to Detect In-Core Fuel/Sodium Contact Breaches
- Provide Capability to Identify Degree of Fuel/Sodium Interaction at In-Core Breach

FUEL FAILURE MONITORING SYSTEM (FFMS) RESPONSE TIME

COVER GAS MONITORING SYSTEM (CGMS)

- COVER GAS TO DETECTOR = ~ 15-90 MINUTES (DEPENDS ON RELEASE RATE)

FAILED FUEL LOCATION SYSTEM (FFLS)

• COVER GAS TRANSIT TIME	=	~ 15 MINUTES
• ACCUMULATE SAMPLE	=	~ 15 MINUTES
• TAG GAS CONCENTRATION IN TRAPS	=	~ 7 HOURS
• MASS SPECTROMETER PROCESSING	=	~ 30 MINUTES
TOTAL	=	~ 8 HOURS

DELAYED NEUTRON MONITORING SYSTEM (DNMS)

- FULL FLOW TRANSIT TIME = < 1 MINUTE
- 40% FLOW TRANSIT TIME = 2 MINUTES
- COUNTING TIME ~ 1 TO 3 MINUTES

REQUIREMENTS PLACED ON THE FFMS

GENERAL REQUIREMENT: DETECT, LOCATE AND CHARACTERIZE FUEL AND BLANKET ROD FAILURES.

SPECIFIC REQUIREMENTS:

CGMS - DETECT FUEL AND BLANKET ROD BREACHES VIA COVER GAS ACTIVITY INCREASE. DESIGN SENSITIVITY IS 10^{-11} STANDARD CC FISSION GAS PER CC COVER GAS

FFLS - LOCATE AND CHARACTERIZE FUEL AND BLANKET ROD BREACHES VIA TAG GAS AND FISSION GAS ANALYSES. DESIGN SENSITIVITY IS $<$ 1 PPB OF TAG IN COVER GAS

DNMS - DETECT FUEL-SODIUM CONTACT AT IN-CORE BREACH LOCATIONS FOR FUEL AND BLANKET RODS.

- DESIGN SENSITIVITY IS 1.5 CM^2 EXPOSED FUEL BY RECOIL.

- DESIGN SENSITIVITY PRECLUDES LARGE DIAMETER INCREASES, AND POSTULATED POROUS HEAT GENERATING BLOCKAGES WHICH COULD CAUSE CLADDING TEMPERATURES GREATER THAN 1600°F AND SODIUM BOILING.

RUN BEYOND FAILURE CONCEPT

- Operate with Gas Leakers Until DND Indication of Fuel-Sodium Contact by Continuous DN Signal Above Some TBD Level

- Remove All Known Fuel-Sodium Contact Leakers and Gas Leakers at Each Refueling Interval

RUN BEYOND CLADDING BREACH EXPERIENCE

CONCLUSIONS FROM OPERATING EXPERIENCE WITH BREACHED RODS

Steady State Operation

- Gas Type Leakers Present No Problem in Regard to Rod-to-Rod Propagation
- Finite Lifetime with Fuel-Sodium Contact Leakers - Extended to 22 Days and 0.75 cm^2 with Benign Effects. Limit not yet Found Although an Assembly has Gone ~ 96 Days with DN Signals and Multiple Failures (PIE in Progress)
- Operating Breached Rods have Enhanced DN Emission Over Recoil Making DN Detection More Sensitive - Breached Rods DN Emission is Exponential with Power.

Transient Operation

- The Behavior of Breached Fuel Rods Having Fuel-Sodium Contact (Logging) are not Seriously Degraded. Shutdown/Startup Extends Breach.

Areas Requiring Further Development Effort

- Diameter Increase Versus Fuel Exposure to Sodium
- Additional PPS Terminated TOP and LOF Results from Naturally Occurring Failures
- Detection and Instrumentation Diagnostics

CRBR Inlet Blockage Considerations

R. A. Markley

ACRS Subcommittee Meeting

March 16, 1983

RBCB WITH DEFECTED RODS

Completed Tests

- 2 S.S. EBR-II Pre-Defected Irradiated Fuel Assembly Tests Completed and Results Reported (RBCB-6,7)
- 3 S.S. EBR-II Irradiated Fuel Rod Assembly Tests with Natural Breach Completed (RBCB-1,2,3)
- 1 EBR-II Irradiated Fuel Rod Bundle S.S. Test with Natural Breaches Test Completed (XY-2)
- U.S. Data Indicates Benign Operation - Little Fuel Loss, Operation Possible for at Least ~ 22 Days, Defects Grow Due to Shutdown - Startup Combined Localized Diameter Increase < 15%, Detectable DND Signals, No Rod - Rod Propagation (Although a Low B.U. Rod in RBCB-2 Adjacent to Another Breached Rod Failed. Breach Possibly Due to Reconstitution; Diameter Increase Acceptable)

Planned or On-Going Testing

- 5 Kinetics and Contamination S.S. Bundle Tests (RBCB-K1, K2, K2A,B,C)
- 3 Detection and Instrumentation DND Tests (RBCB-D1, D2, D3 Variable Flow)
- 5 Fuel and Irradiation Variables Tests (RBCB-V2 Plenum Defects, RBCB-V4 Larger Diameter, RBCB-V5 Blanket, RBCB-V6 Unreconst., RBCB-V7 Sodium Storage Pre-Defected)
- ORT Program Includes 1 TOP Test (TOP 1-2)

ASSEMBLY INLET BLOCKAGE AND MODULE BLOCKAGE

- Assembly Inlet and Module Blockages are Highly Improbable Because of the Built-In Flow Blockage Prevention Features Which Provide Flow Path Redundancy:

  -- Multiple Primary Ports
  -- Axial Debris Barriers
  -- Radial Debris Barriers
  -- Multiple Auxiliary Ports
  -- Strainer
  } - Lower Inlet Module

  -- Multiple Inlet Slots - Assembly Nozzle

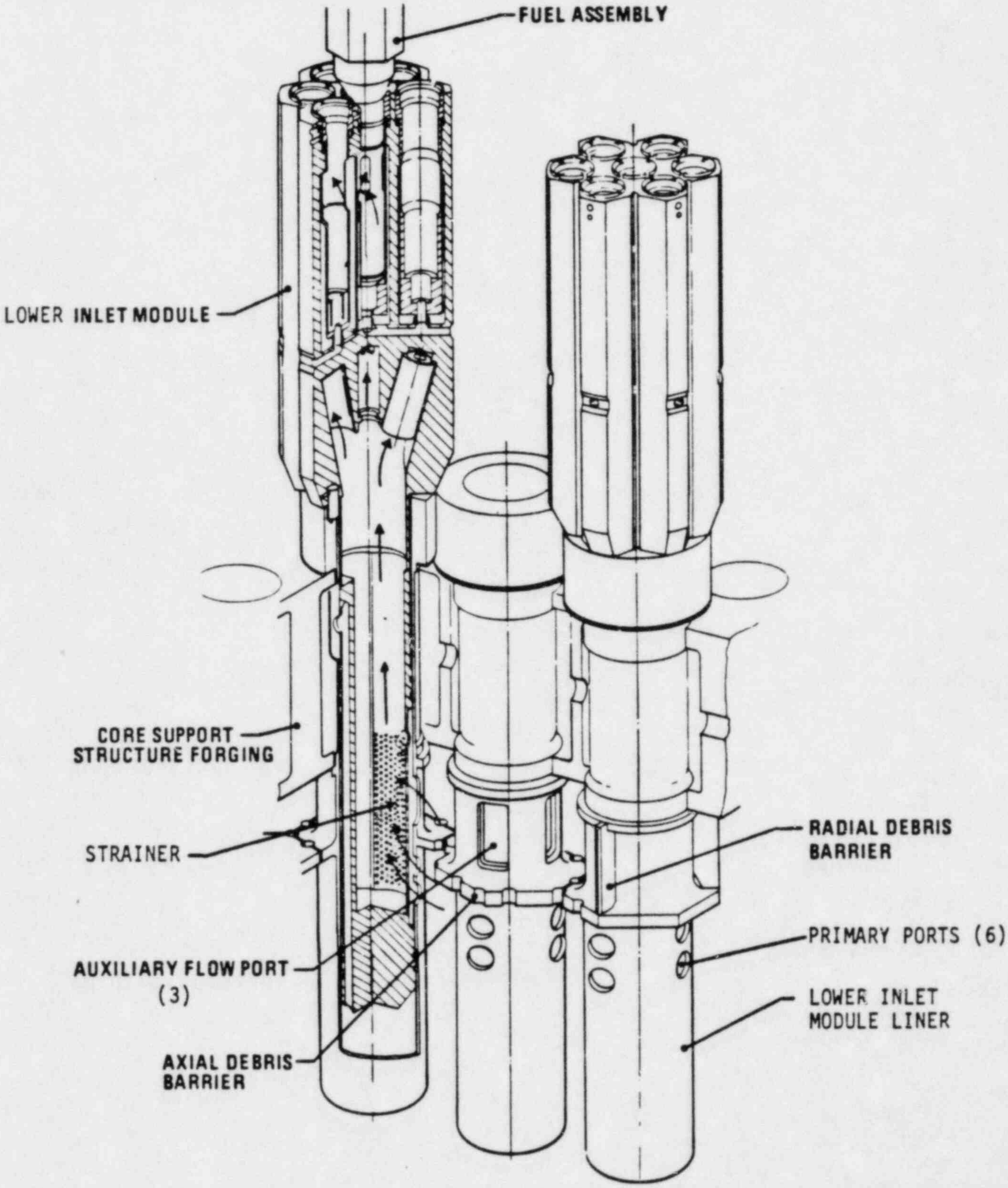
- The Strainer Prevents Debris Larger than 0.25 Inches from Entering the Modules

- The Orifice Plates Provide Another Level of Screening

- Debris Not Strained by the Fuel Rod Support Keys and Unheated Rod Bundle Entrance Will Pass Through Rod Bundles

- Even a 50% Areal Inlet Blockage Causes Less than 25°F Increase in Outlet Temperature

LOWER INLET MODULES
FLOW BLOCKAGE PREVENTION FEATURES

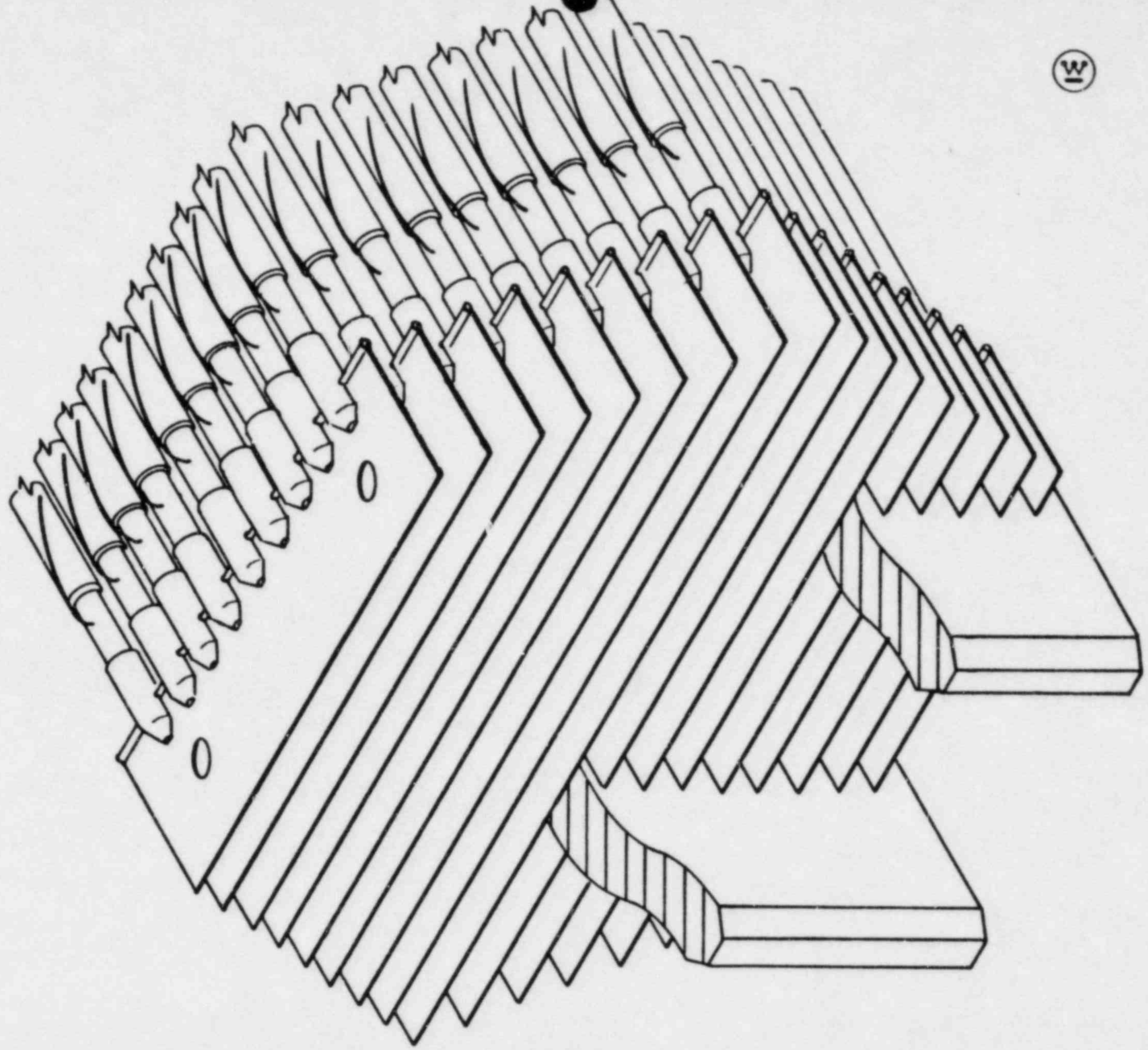


9657-2

ROD BUNDLE INLET SCREENING

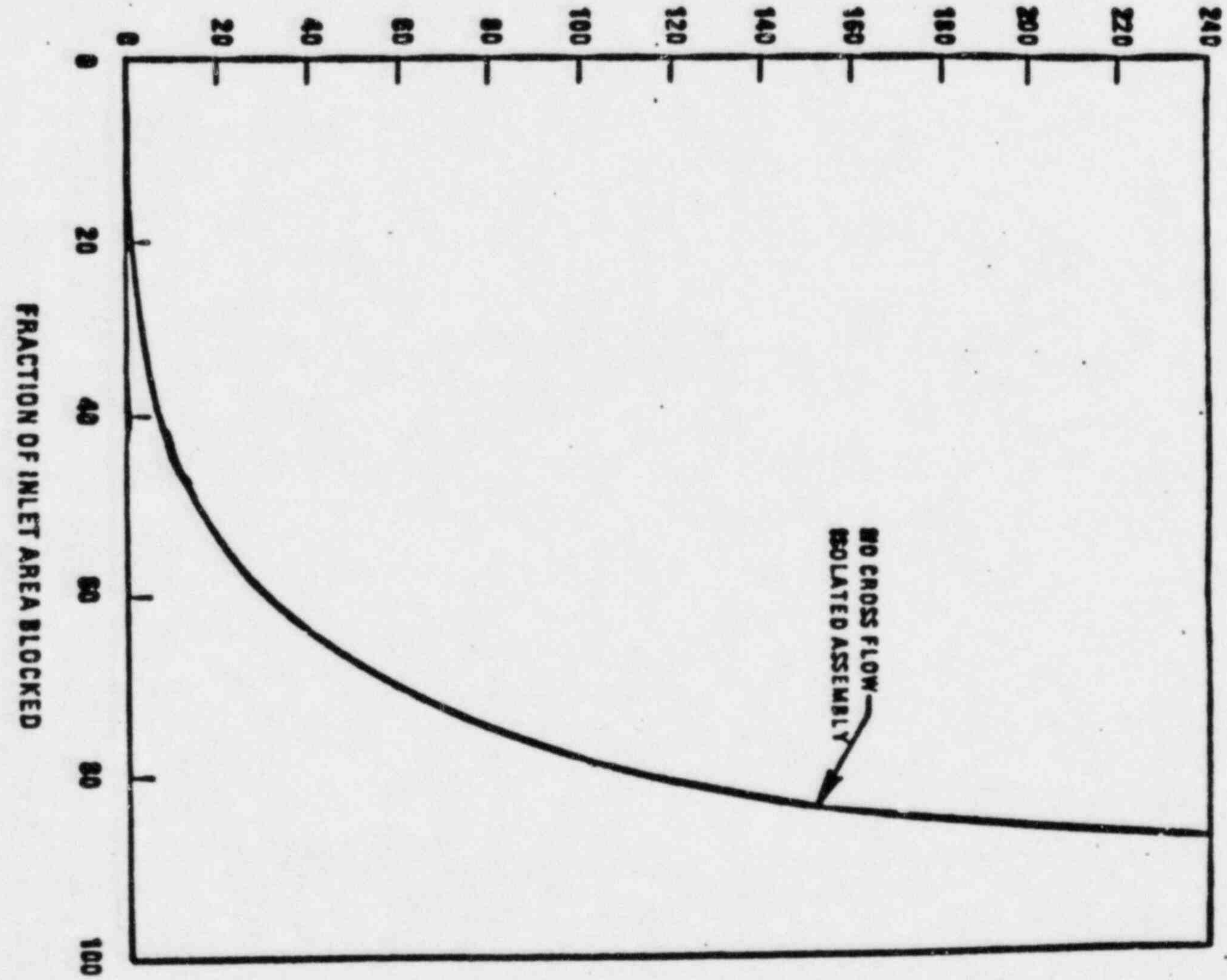
- Low Probability
  - -- Q A Program Comparable to FFTF
  - -- Core Special Assemblies During Preoperation Testing Filter Reactor Flow (>0.004 Inches)
- Only Slow Buildup Possible
  - -- Particulate Buildup Not Preferential
  - -- Particles of Concern Must Be in 0.056" to 0.25" Range
  - -- >400 Interconnected Flow Channels in Fuel Assembly
- Margins to Accommodate
  - -- Redundant Flow Paths
  - -- Cascading Straining Helps
  - -- Even 50% Areal Inlet Blockage Causes Less Than 25°F Increase in Outlet Temperature

∴ Rod Bundle Inlet Blockages Very Low Probability, Slow, With Large Margins to Accommodate



ROD BUNDLE ATTACHMENT RAIL ASSEMBLY

ASSEMBLY OUTLET TEMPERATURE INCREASE -(°F)



FUEL FAILURE PROPAGATION POTENTIAL

R. W. TILBROOK

WESTINGHOUSE ADVANCED REACTORS DIVISION

WALTZ MILL SITE

MARCH 1983

CHAPTER 15.4 LOCAL FAILURE EVENTS

DEFINITION: A FAILURE WHICH IS INITIATED WITHIN A SINGLE FUEL, BLANKET OR CONTROL ASSEMBLY

TYPE OF INITIATOR EVENTS:

1. STOCHASTIC ROD FAILURE
2. LOCAL FLOW BLOCKAGE
3. BUBBLE PASSING-THROUGH CORE

ALSO CONSIDERED:

4. MOLTEN FUEL

1. STOCHASTIC ROD FAILURE

- FISSION GAS RELEASE
- FUEL PARTICLE RELEASE
- OPERATION WITH FAILED FUEL

FISSION GAS RELEASE

- THERMAL EFFECT - GAS JET IMPINGEMENT
  - MAIN LOCATION FOR IMPINGEMENT WOULD BE AT PLENUM
  - LOWER RELEASE RATES IN FUEL REGION WOULD AVOID IMPINGEMENT
  - WIDE RANGE OF HOLE DIAMETER AND PIN PRESSURE TESTED

- THERMAL EFFECT - ROD GAS BLANKETING
  - TESTS SHOW DISPERSION OF GAS ACROSS ASSEMBLY
  - FLOW REDUCTION FROM GAS BLANKETING WOULD NOT CAUSE FAILURE

- TRANSIENT MECHANICAL LOADING
  - DIFFERENTIAL PRESSURE ACROSS ROD CANNOT CAUSE ROD FAILURE
  - DUCT WILL WITHSTAND MAXIMUM PRESSURIZATION (300 PSI VS. 550 PSI CAPABILITY AT 1000°F)

FISSION GAS RELEASE IN REACTOR SAFETY (CSNI REPORT NO. 40)

INTERNATIONAL COOPERATIVE STUDY SPONSORED BY "ORGANIZATION FOR ECONOMIC COOPERATION AND DEVELOPMENT" (1979)

- THE SAFETY TECHNOLOGY ON FISSION GAS RELEASE IS WELL DEVELOPED WITH THE SUPPORT OF A VAST AMOUNT OF EXPERIMENTAL DATA AND MANY YEARS OF EXPERIENCE

- FISSION GAS RELEASE BY ITSELF IS NOT A POSSIBLE CAUSE FOR FUEL ROD FAILURE PROPAGATION

FUEL PARTICLE RELEASE

- OPERATIONAL EXPERIENCE WITH FAILED FUEL INDICATES LITTLE WASHOUT AND NO PARTICULATE RETENTION IN WIRE WRAP ASSEMBLIES

- HEAT GENERATING BLOCKAGES ASSOCIATED WITH ASSUMED PARTICLE RETENTION IN THE BUNDLE ARE DISCUSSED LATER

OPERATION WITH FAILED FUEL

- CONSEQUENCES OF OPERATING WITH FAILED FUEL:
  - FISSION PRODUCT LEACHING: SODIUM CLEANUP SYSTEM REMOVES FPs
  - SODIUM INGRESS: HIGH INTERNAL PRESSURES FROM SODIUM VAPORIZATION HAVE NOT BEEN FOUND IN TESTS
  - FUEL/SODIUM CHEMICAL REACTION: ANY SWELLING WHICH OPENS THE BREACH EXPOSES MORE FUEL AND INCREASES THE D.N. DETECTOR SIGNAL. SWELLING IS SLOW AND HAS NEGLIGIBLE EFFECTS ON NEIGHBOR ROD HEAT TRANSFER

CONCLUSIONS: OPERATION WITH FAILED FUEL IS ACCEPTABLE. OPERATING LIMITS WILL BE ESTABLISHED.

SODIUM-FUEL REACTION

- SODIUM AND URANIUM-PLUTONIUM OXIDE WILL REACT IN THE PRESENCE OF FREE OXYGEN

- REACTION PRODUCT $NA_3\ MO_4$ HAS DENSITY ABOUT 1/2 OF FUEL

- UNDER WORST CASE CONDITIONS AT EOL, THEORETICAL UNIFORM EXPANSION OF CRBRP FUEL WOULD BE 1.7% DIAMETRAL

- KINETIC DATA INDICATE TWO TO SIX DAYS TO REACH EQUILIBRIUM

- EXPERIENCE INDICATES SODIUM-FUEL REACTION DOES NOT LEAD TO FAILURE PROPAGATION; DFR, RAPSODIE, AND BR-5 OPERATED LONGER THAN 100 DAYS WITH FUEL EXPOSED TO SODIUM WITHOUT EVIDENCE OF PROPAGATION

2. LOCAL FLOW BLOCKAGE

(WITHIN ROD BUNDLE)

- IN-CORE PASSIVE BLOCKAGE
- IN-CORE ACTIVE BLOCKAGE (HEAT-GENERATING BLOCKAGES)

CONCLUSIONS ON LOCAL FLOW BLOCKAGE

LOCAL FLOW BLOCKAGE DETAILS ARE IN THE FEBRUARY 3, 1983 PRESENTATION.

- EVEN A SIX-CHANNEL IN-CORE PASSIVE PLANAR BLOCKAGE WILL AT MOST ONLY REDUCE FUEL LIFETIME

- THE FORMATION OF HEAT-GENERATING BLOCKAGES (HGB) WITHIN THE CORE IS HIGHLY IMPROBABLE

- DELAYED NEUTRON DETECTOR (DND) SYSTEM WILL DETECT A HGB SMALLER THAN THAT WHICH COULD PROPAGATE DAMAGE

3. GAS BUBBLES PASSING THROUGH CORE

- THE PLANT IS DESIGNED TO PRECLUDE GAS BUBBLES ENTERING THE CORE, FOR EXAMPLE: VENTS FROM POTENTIAL GAS POCKETS, LOW COVER GAS PRESSURE, IHX CONTINUOUS BLEED, VORTEX SUPPRESSOR PLATE

- THE THERMAL CONSEQUENCES OF EVEN A LARGE BUBBLE (4 INCHES HIGH-OVER 8 ROWS: 3 CU. FT.) ARE SMALL (68°F MAXIMUM CLADDING TEMPERATURE INCREASE)

- THE REACTIVITY CONSEQUENCES OF ISOLATED BUBBLES ARE NEGLIGIBLE

- EVEN A 5 INCH BUBBLE WOULD ONLY INCREASE CLAD TEMPERATURE BY 25°F AT CORE EXIT

CONCLUSION: THERE ARE NO PROPAGATIVE CONSEQUENCES DUE TO BUBBLES

4. SUMMARY OF MOLTEN FUEL ISSUES

- MOLTEN FUEL (M.F.) DOES NOT EXIST IN REACTOR DURING NORMAL, UPSET OR EMERGENCY EVENTS
- M.F. NOT FORMED BEHIND PASSIVE PLANAR BLOCKAGES
- M.F. NOT FORMED IN HEAT GENERATING BLOCKAGES WHICH ARE NON-DETECTABLE
- TESTS D1, D2 DEMONSTRATED TRANSIENT OPERATION WITH M.F.
- SLSF TEST P1 DEMONSTRATED CONTINUED OPERATION WITH SIGNIFICANT AREAL M.F.

WORLD-WIDE EXPERIENCE WITH LIQUID METAL/OXIDE

FUELED PLANTS IS EXTENSIVE, AND DEMONSTRATES

NON-PROPAGATIVE CONSEQUENCES OF FUEL FAILURE.

WORLD WIDE OXIDE FUEL*

OPERATING EXPERIENCE

(APPROXIMATE NUMBERS)

COUNTRY	REACTOR	OXIDE USE	NO. RODS IRRADIATED	NO. RODS FAILED
FRANCE	PHENIX	DRIVER+	100,000	5
	RAPSODIE	DRIVER TEST	16,000 2,000	36**
UK	PFR	DRIVER+	30,000+	2
	DFR	TEST	2,500	200+++
USA	FFTF	DRIVER+	16,000	1
	EBR II	TEST	3,000	55**
JAPAN	JOYO	DRIVER TEST	5,000	0
USSR	BR-5, 10	DRIVER TEST	5,000	100+
	BOR-60		12,000	<60
	BN-350		50,000	
	BN-600			

* (U, Pu) O_2

** (INCLUDES 1/3 TO 1/2 WHICH FAILED AT B.U. $\geq$ 10%)

\+ SOME RODS MAY BE DESIGN PROOF TESTS

++ MOST FAILURES RELATED TO DOWN-FLOW COOLANT IN CORE

CONCLUSIONS FOR LOCAL FAULTS

- OPERATING PLANT FUEL FAILURE RATES LOW
- STOCHASTIC FUEL ROD FAILURE DOES NOT LEAD TO FAILURE PROPAGATION
- FAILED FUEL IS DETECTABLE
- CONTINUED OPERATION WITH FAILED FUEL HAS NOT CAUSED PROPAGATION
- DEGRADATION PROCESSES ARE SLOW AND CAN BE MONITORED WITH REMOVAL AT PREDETERMINED OPERATING LIMITS

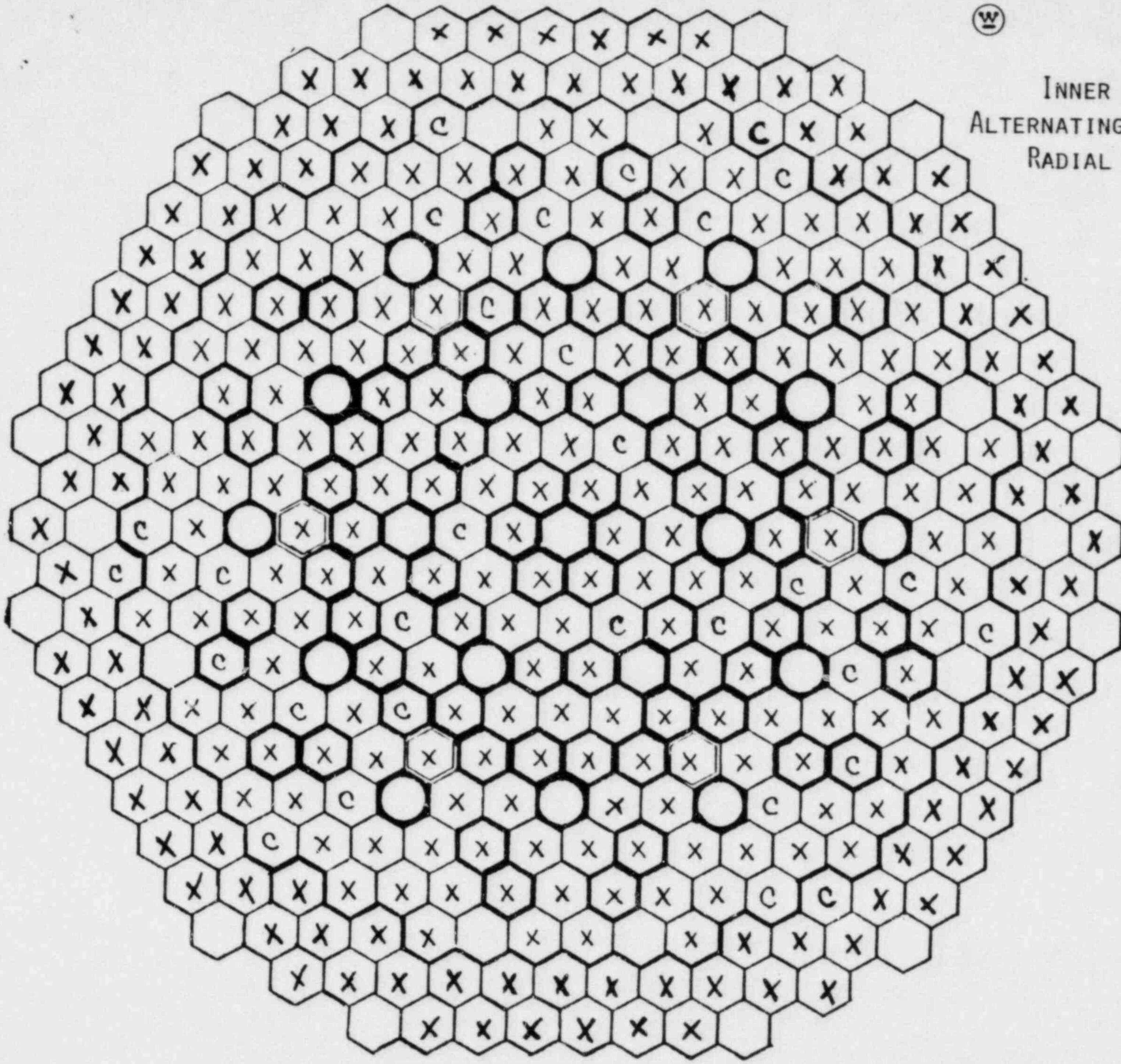
THEREFORE, THE CONSEQUENCES OF FUEL FAILURE ARE BENIGN.

CRBR Core Exit Thermocouples

R. A. Markley

ACRS Subcommittee Meeting

March 16, 1983



COVERAGE

FUEL	148	OUT OF	156
INNER BLANKET	72	"	76
ALTERNATING ASSYS.	6	"	6
RADIAL BLANKET	112	"	126
TOTAL	338		364

C = CONTROL

X = SURVEILLANCE/ DESIGN VERIFICATION

CRBRP CORE EXIT THERMOCOUPLES COVERAGE

CRBRP CORE EXIT THERMOCOUPLES

IDENTIFIED FUNCTIONS

- Reactor Control
- Design Verification
  - -- Design Margins, Power Distribution, Symmetry, Fuel Rod Lifetime
  - -- Operational Disturbances (Local Disturbances in Power Distribution, Shifts in Power/Flow, Etc.)
  - -- Uncertainty Factors Assessment
  - -- Relationship with Post-Irradiation Measurements

REACTOR THERMOCOUPLES FOR AUTOMATIC CONTROL

REQUIREMENTS:

- Maintain Steady State Primary Outlet Temperature Within Fixed Band
- Minimize Temperature Overshoot

Coverage - 30 Positions Over 3 Symmetric 60° Sectors

RATIONALE:

- Closely Approach Core Mixed Mean and Follow Cycle Swing
- Even Coverage
- Provide Redundancy

Type - Dry-Well

Time Constant <10 Seconds

TEMPERATURE MEASUREMENT UNCERTAINTIES

FUEL ASSEMBLY

RANGE FROM ~8°F (CORE CENTER, SIX-FOLD COVERAGE BY SYMMETRY) TO ~25°F (PERIPHERY, SINGLE COVERAGE)

INNER BLANKET ASSEMBLY

RANGE FROM ~15°F (HIGH FLOW, SYMMETRY) TO ~40°F (LOW FLOW, SINGLE COVERAGE)

CORE OUTLET THERMOCOUPLES ARE NOT SAFETY RELATED

- For "Core-Wide" Faults:
  - -- Fast Acting PPS Trips (e.g. high flux, flux to flow) Shutdown Before Excessive Temperatures
- For Local Faults (Blockage, High Heat Flux)
  - -- T/C Insensitive to Local Faults (e.g. Blockages <50 to 60%)
  - -- T/C "Blinded" by Adjacent Cross Flow When Blockage ~ >90%
  - -- Low Probability of Occurrence
  - -- If Local Temperature Increase Causes Clad Failure, Detected By Failed Fuel Monitoring System
- Limited Number of Clad Failures Are Expected and Are Not a Safety Concern
  - -- Rapid Propagation Will Not Occur

CORE OUTLET THERMOCOUPLE

ALARM, ALERTS, ETC.

PRESENTATION TO

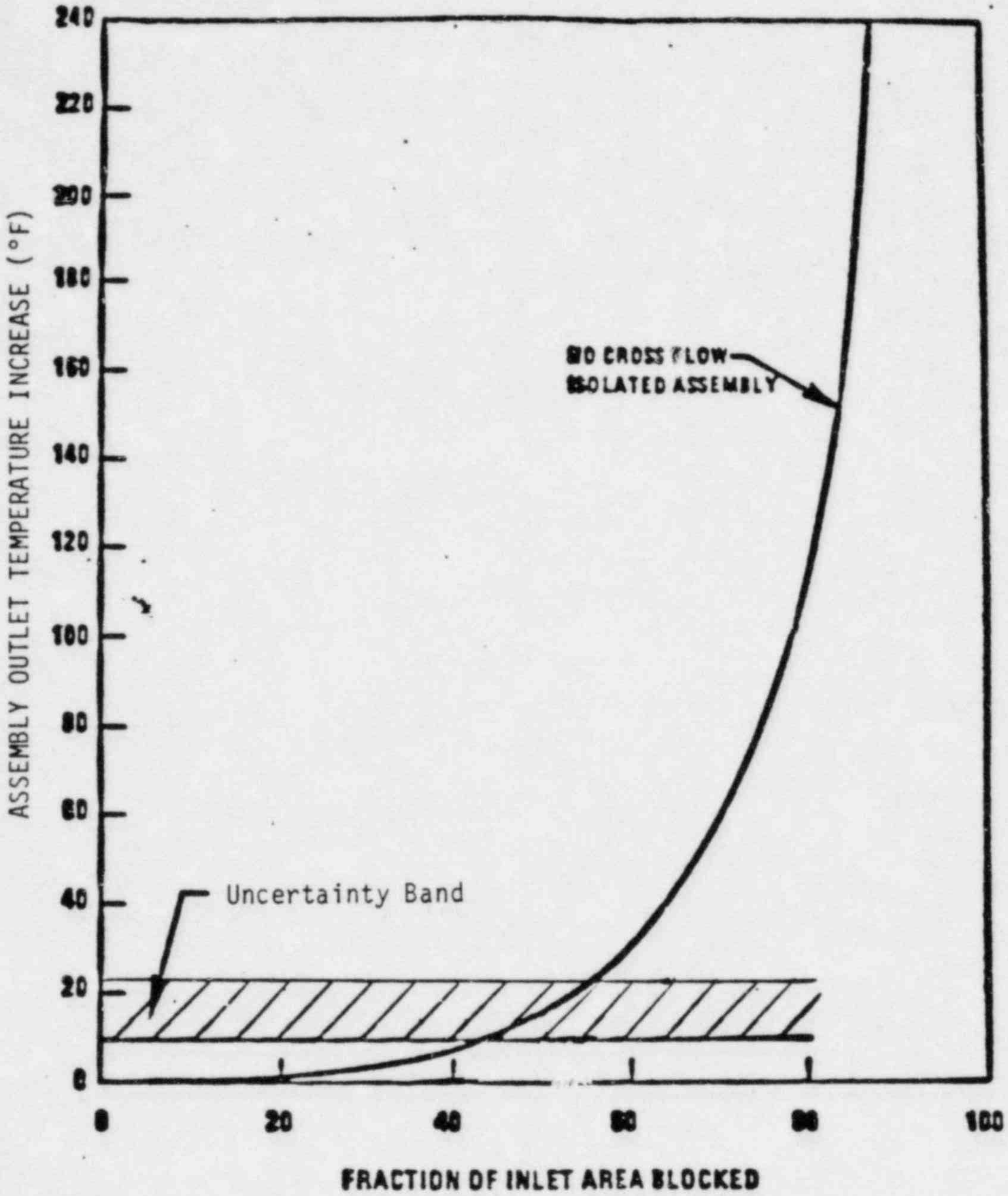
ACRS

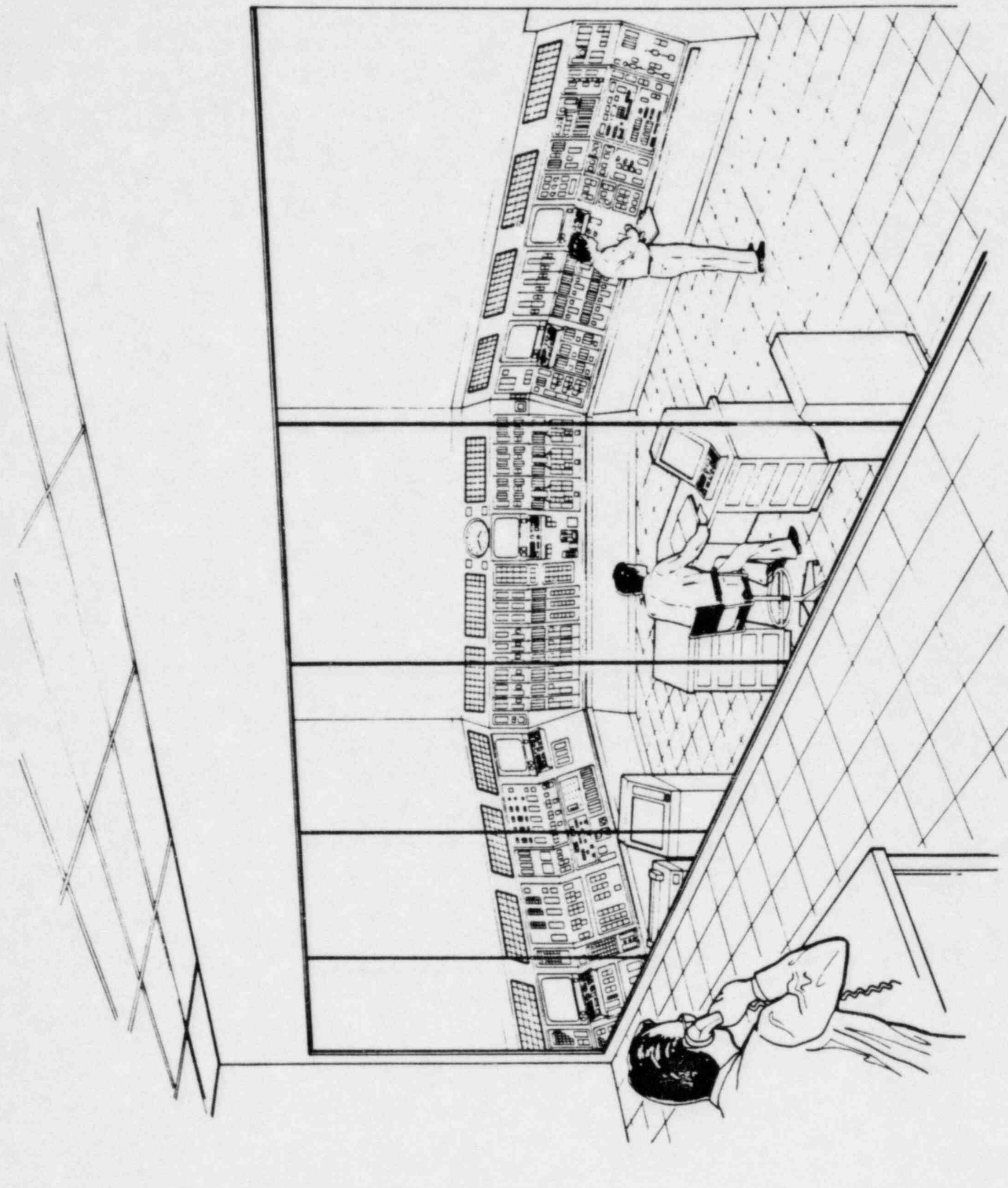
MARCH 16, 1983

BY

R. J. TINDER, WESTINGHOUSE

ADVANCED REACTORS DIVISION


ASSEMBLY OUTLET TEMPERATURE INCREASE (°F)
0
20
40
60
80
100
120
140
160
180
200
220
240
NO CROSS FLOW
ISOLATED ASSEMBLY
Uncertainty Band
20
40
60
80
100
FRACTION OF INLET AREA BLOCKED



ⓌARD

CORE THERMOCOUPLE INDICATIONS AND ALARMS

- 338 FUEL AND BLANKET THERMOCOUPLES
- EACH WIRED TO PLANT DATA HANDLING AND DISPLAY SYSTEM
- COMPARED TO ALGORITHM OF HISTORY-POWER-FLOW-LOCATION
- COMPUTER "ALERT" WHEN OUTSIDE TOLERENCE
- ALERT TYPES ON ALARM TYPEWRITER
- ALERT APPEARS ON ALARM CRT
- LAST THREE ALERTS APPEAR AT BOTTOM OF CRTs
- OPERATOR CAN REQUEST T/C READINGS ANY TIME (CRT OR HARDCOPY)
- SELECTED 30 T/C FOR CONTROL SYSTEM
- LESS THAN 25 CONTROL T/C ACTIVATES MAIN PLANT ANNUNCIATOR (AUDIBLE AND VISUAL)

Review Of World-Wide Application Of Local Fault Instrumentation

L. E. Strawbridge

Westinghouse Advanced Reactors Division

Scope Of Survey

- Available Design Information Collected On 17 Sodium Cooled Fast Reactors.
- Major Plant Design Variations
  - Loop Or Pool
  - Power Or Test
  - Small To Large
- Major Fuel Design Variations
  - Oxide Or Metal Fuel
  - Wire Wrap Or Grid Spacers

INSTRUMENTATION SURVEY

	RAPSODIE	PHENIX	SPX	DFR	PFR	CDFR	SNR-300	JOYO	MONJU
LOOP OR POOL	LOOP	POOL	POOL	LOOP	POOL	POOL	LOOP	LOOP	LOOP
POWER OR TEST	P,T	P	P	P,T	P	P	P	T	P
FUEL/SPACER	OX/WW	OX/WW	OX/WW	M/	OX/G	OX/G	OX/G		OX/
CORE EXIT T/C	YES	YES	YES		YES	YES	YES	YES	YES
T/C IN PPS		YES	YES		YES	YES	A	NO	YES

DEFINITIONS:

WIRE WRAP - WW   OXIDE - OX   ALARM - A

GRID - G   METAL - M

INSTRUMENTATION SURVEY (CONTINUED)

	BR-5/10	BOR 60	BN 350	BN 600	EBR II	FERMI	FFTF	CRBRP
LOOP OR POOL	LOOP	LOOP	LOOP	POOL	POOL	LOOP	LOOP	LOOP
POWER OR TEST	P,T	P,T	P	P	P,T	P	T	P
FUEL/SPACER	OX/	OX/	OX/	OX/	M/WW	M/G	OX/WW	OX/WW
CORE EXIT T/C					YES	YES	YES	YES
T/C IN PPS					NO	NO	NO	NO

DEFINITIONS:

WIRE WRAP - WW　　OXIDE - OX　　ALARM - A

GRID - G　　METAL - M

Survey Of Outlet Thermocouples In PPS For Local Faults

- Strongest Correlation Exists For Loop vs. Pool

	# Plants	T/C In PPS Yes	No	Unknown
Loop	11	1	5	5
Pool	6	4	1	1

Conclusions On World-Wide Application Of Local Fault Instrumentation

- No Universal Agreement On Approach.
- Core Outlet Thermocouples Usually Included In PPS In Pool Reactors; Usually Not Included In Loop Reactors.
- Considering The CRBRP Design, The Application Of Instrumentation Is Consistent With The World-Wide Trends.

Conclusions On Local Faults

- QA Results In Low Probability Of Loading Defective Fuel.
- Design Features And Operational Requirements Prevent Ex-Core Blockages.
- No In-Core Blockage Expected Based On Operation, Tests And Analysis.
- Fuel Failures During Operation Are Anticipated And Will Be Detected.
- Extensive Experience Base Shows No Evidence Of Progagation.
- Thermocouples Do Not Significantly Improve The Margin Of Safety For Local Faults
  - Not Effective For Prevention/Forewarning
  - DNDs More Sensitive For Detecting Failures Of Interest
- Considering The CRBRP Design, The Application Of Instrumentation Is Consistent With The World-Wide Trends.
- Instrumentation For Local Fault Monitoring Is Not Needed In PPS Since No Faults Could Propagate On A Time Scale Requiring PPS Action.