APR 11 1986

MEMORANDUM FOR: Chairman Palladino

FROM: Victor Stello, Jr.
Executive Director for Operations

SUBJECT: MEETING WITH ELECTRIC POWER RESEARCH INSTITUTE NUCLEAR POWER PROGRAMS

In order to further improve our technical interchange with the industry, NRR met with representatives of the Electric Power Research Institute (EPRI) on February 27, 1986. The EPRI programs address many substantive issues in the areas of nuclear plant design, construction, operation and life extension. EPRI's budget for its nuclear power program is approximately $60 million in 1986. However, many of the associated programs are jointly funded and managed, resulting in a very economical and effective use of these resources. Since the scope of the information discussed in this meeting bears on the activities of the NRC, I am providing a copy of the slides for your information.

Original signed by
Victor Stello

Victor Stello, Jr.
Executive Director for Operations

Enclosures:
1. Meeting Summary
2. Copy of slides

cc: Commissioner Roberts
Commissioner Asselstine
Commissioner Bernthal
Commissioner Zech
SECY
OPE
OGC

Contact: T. Speis, NRR
49-27517

Distribution:
Central Files
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EDO Rdg. File
TAMB Rdg. File
PPAS
V. Stello
H. Denton
J. Funches
T. Speis
M. Williams
J. Roe
T. Rehm
J. Sniezek

* See previous concurrence

TAMB:PPAS*	TAMB:PPAS*	DIR:PPAS*	DIR:NRR	DIR:EDO
HPolk:js	M. Williams	J. Funches	H. Denton	V. Stello
4/3/86	4/3/86	4/7/86	4/9/86	4/11/86

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EPRI

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HPolk:js	M. Williams	J. Funches for	H. Denton	V. Stello
4/3 /86	4/3 /86	4/7 /86	4/ /86	4/ /86

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Enclosure 1

ELECTRIC POWER RESEARCH INSTITUTE MEETING SUMMARY

On February 27, 1986, NRR met with EPRI, represented by the Department Directors of Nuclear Power Programs (NPD) to discuss the current and future research. EPRI provided an overview of their nuclear power program followed by a fairly detailed discussion of several major projects. Some of the more significant products from their work are the development of guidelines for steam generator chemistry control, nondestructive examination, BWR pipe crack guidelines on inspection and chemistry control and remote examination and sampling by robots. This summary report of the meeting is provided to you for your information. For further information consult the detailed EPRI slides contained in Enclosure 2.

EPRI gave a presentation that was divided into four major divisions which paralleled the organization of EPRI NPD. These divisions are:

1. Nuclear Safety Analysis Center (NSAC)
2. Engineering and Operations Center (E&O)
3. Safety Technology (ST)
4. Systems and Material Department (S&M)

The NSAC was organized at EPRI by the utilities as a result of the TMI event. NSAC International Program has 13 Associate Members. Programs are focused on Generic Safety Issues. Some are conducted using the matrix management system that involves the other departments in the following areas:

1. Pressurized Thermal Shock
2. Steam Generator Integrity
3. Decay Heat Removal
4. Piping Integrity
5. Technical Specification Improvement
6. Diesel Generator Reliability

The results of the NSAC work is integrated toward resolution of safety issues. Approximately 32 individual issues are currently being tracked in the Generic Safety Analyses Program. The issues have been prioritized in four levels and only a few can be worked on in light of budget limitations. The top priority items are:

1. Steam Generator Integrity
2. Environmental Qualification of Equipment
3. ATWS
4. Seismic Design Criteria
5. Station Blackout
6. Hydrogen Control Measures

The Engineering and Operations Department is concentrating its research in the areas of:

1. Nuclear Plant Life Extension
2. Nuclear Plant Constructibility
3. Plant Availability
4. Low Level Waste and Coolant Technology

The Nuclear Plant Life extension work is using the Surry 1 and Monticello plants as models. The work is concentrated in a detailed assessment of the major components, systems and structures. The study is identifying the utility actions that are recommended to establish a basis for plant life extension. These actions are identifying maintenance requirements along with necessary plant equipment records and implementation of material sampling plans which would show the effects of aging on material properties. Aging specimens have been installed in 7 plants at this time.

In the area of nuclear plant constructibility EPRI is researching ways to reduce plant construction costs for new plants and modifications to operating plants. The work is centering on using computer-aided engineering techniques, providing program management for the Nuclear Construction Issues Group (NCIG) and conducting workshops on plant lay-up and equipment preservation. As a tool for developing better plant layout EPRI has developed a three dimensional computer model for the Midland plant.

The 1985 key accomplishments for the E&O Department in these areas were:

1. Surry 1 and Monticello Pilot Project on plant life extension under way.
2. Developed interim guidelines for application of computer-aided engineering.
3. Completed Midland three dimensional construction model.
4. Provided program management for Nuclear Construction Issues Group (NCIG).
5. Conduct workshop on plant lay-up and equipment preservation.
6. Published guidelines/databook on qualification on mechanical equipment.
7. Conducted seminars on maintaining equipment qualification.
8. Aging specimens (cable, devices, lubricants) installed in 7 plants.

The E&O is studying the minimization of waste by various techniques and evaluation of the impact of these techniques. EPRI has developed a computer program to perform radwaste volume reduction economic analysis. In the area of Coolant Technology the work is centered in Cobalt 60 control through zinc injection, surface modification, prefilming of surfaces and cobalt alloy replacement.

The Maintenance and Equipment Application Center (MEAC) has developed two robots to assist the utilities in inspection and limited sampling of areas within a plant that would be too dangerous for humans to enter or work continuously. These robots are the Surveyor Mobil Robot and the Surbot Mobile Robot. Additional MEAC has developed and demonstrated an underwater inspection vehicle. The MEAC provides training courses on pressure boundry bolting and diesel generator maintenance. Current R&D emphasis is in the areas of Plant Life Extension, Decontamination, Robotic Devices, Equipment Qualification, Computer Generated Displays and Computer Aided Construction.

The Safety Technology Department is active in the areas of:

1. Source Term
2. Seismicity Resolution
3. Codes and Standards on Structural Response
4. Integration of PRA on major issues
5. Digital Systems Implementation

The Safety Technology Department has recently focused heavily on source term and seismic technology. There is growing emphasis on safety control development. There is continuing work on a variety of computer codes for safety analysis, fuel management and near-real-time plant monitoring.

All of the work requiring experimental input is geared toward large scale demonstration experiments and tests that are frequently co-sponsored by government, industry or foreign organizations.

The seismic work involves the EUS seismicity study and large installation in Taiwan that consists of large scale concrete containment type structure to measure the effects of earthquakes. This information will be used to validate the current methods that are in use for predicting structural responses. Considerable work has been done in the areas of internal pressure effects on concrete containment walls that have liners attached. Work has been done and is still ongoing that will be used in producing computer codes including one that can be used to predict the transport and deposition of sediment in the coolant piping. The source term experimental work is nearing completion and the next task will focus on integration of information to support implementation of the technology in a meaningful way.

The Systems and Materials Department is active in the following areas:

1. LWR Fuel and Spent Fuel Storage
2. Component Reliability
3. Corrosion Control
4. Advanced LWR
5. BWR Owners

The Systems and Material Department LWR fuel group is developing a technical basis for designing LWR fuel for higher burnup. Limited work is being done on zircaloy corrosion and fission gas release. Additionally, the group is nearly finished with a computer code that will provide the utilities with the capability to evaluate fuel designs and support licensing of reloads. Final version is scheduled for release on March 31, 1986. The group is also developing and transferring to field practice qualified NDE equipment and procedures. The group is conducting a test program structured to increase the allowable pipe stresses and determining the feasibility of using simplified design methods. In the area of pressurized thermal shock, the Department is working to develop correct elastic models for crack arrest in the ductile regions of the pressure vessel. A significant result of this work will be additional life for neutron damaged vessel steel. Corrosion fatigue crack growth has been studied extensively during the past five years. The Department is studying the relationships between hydrogen water chemistry and intergranular stress corrosion and how this corrosion can be mitigated. The Department has demonstrated the feasibility of an on-line, in-plant corrosion cracking sensor. Work that has been done by EPRI shows that the current seismic piping design is very conservative.

AGENDA

EPRI R&D BRIEFING FOR NRC MANAGEMENT

EPRI Washington DC Office
1800 Massachusetts Ave. NW
Suite 700
Washington, DC, 20036

FEBRUARY 27, 1986

8:30 a.m.	CONTINENTAL BREAKFAST
9:00	WELCOME AND INTRODUCTIONS

Bill Rasin, Duke Power Company
Chairman, Safety Technology Task Force
(for Ed Kintner, GPU
Chairman, Nuclear Power Divisional Committee)

John Taylor, Vice President - Nuclear
EPRI

Harold Denton, Director of Regulation
NRC

9:15 NUCLEAR POWER DIVISION PROGRAM

John Taylor
- Overview

9:30 NUCLEAR SAFETY ANALYSIS CENTER (NSAC)

Jeff Jeffries, Carolina Power & Light Company
Chairman, NSAC Task Force

A. David Rossin, Director, NSAC
- Overview
- Issue Prioritization
- Decay Heat Removal
- PTS Code Acceptance
- Station Blackout

10:00 BREAK

10:15 a.m.	ENGINEERING & OPERATIONS DEPARTMENT (E&O)	
		Bud Fay, Wisconsin Electric & Power Company Chairman, E&O Task Force
		Don Rubio, Director, E&O Department - Overview - Nuclear Plant Life Extension & Constructability - Plant Availability - Low-Level Waste and Coolant Technology
11:00	SAFETY TECHNOLOGY DEPARTMENT (ST)	
		Bill Rasin, Duke Power Company Chairman, ST Task Force
		Walt Loewenstein, Director, ST Department - Overview - Source Term Research - Seismic Center - Safety Control - Testing to Support Analyses - Disciplined Software
12:00 Noon	LUNCH	
12:45 p.m.	SYSTEMS & MATERIALS DEPARTMENT (S&M)	
		Lou DelGeorge, Commonwealth Edison Company Chairman, S&M Task Force
		Karl Stahlkopf, Director, S&M Department - Overview - Hydrogen Water Chemistry - BWR Water Chemistry and Impurities - NDE Center/Ultrasonic Inspection - Seismic Response of Piping and Supports
2:00	DISCUSSION	
		John Taylor
		Harold Denton
2:45	ADJOURNMENT	

EPRI

EPRI R&D BRIEFING FOR

NRC MANAGEMENT

FEBRUARY 27, 1986

NUCLEAR POWER DIVISION

ELECTRIC POWER RESEARCH INSTITUTE
WASHINGTON, DC

EPRI

OVERVIEW OF EPRI'S NUCLEAR POWER PROGRAM

ELECTRIC POWER RESEARCH INSTITUTE

NUCLEAR POWER DIVISION

JOHN J. TAYLOR

EPRI R&D BRIEFING FOR

U.S. NUCLEAR REGULATORY COMMISSION MANAGEMENT

FEBRUARY 27, 1986

EPRI

OVERVIEW OF EPRI'S NUCLEAR POWER PROGRAM

UTLINE

- INTRODUCTION
- EXAMPLES OF PRODUCTS FROM 1985
- STATUS OF NPD R&D
- NEW MAJOR PROJECTS
- GUIDELINES
- SUMMARY

EPRI

OVERVIEW OF EPRI'S NUCLEAR POWER PROGRAM

- SOURCE TERM
  - + LARGE TEST PROGRAMS
  - + WORLDWIDE TECHNOLOGY INTEGRATION
  - + IDCOR INTERFACE
- SEISMIC
  - + SEISMIC MARGINS
  - + LOTUNG TEST
- PIPING INTEGRITY
  - + SEISMIC
  - + BLOWDOWN
- CONTROL TECHNOLOGY
  - + SIMULATORS
  - + AI
  - + DIGITAL CONTROL

EPRI

OVERVIEW OF EPRI'S NUCLEAR POWER PROGRAM

- TESTING TO SUPPORT ANALYSIS
  - + MIST
- DECAY HEAT REMOVAL
- TECHNICAL SPECIFICATION IMPROVEMENT
- STATION BLACKOUT
- SOFTWARE
- LWR FUEL & SPENT FUEL STORAGE
  - + HIGH BURNUP FUEL

EPRI

OVERVIEW OF EPRI'S NUCLEAR POWER PROGRAM

- MAINTAINABILITY
  - IMPROVE U.S. PLANT AVAILABILITY
  - PROVIDE TECHNICAL INITIATIVES TO CONTAIN O & M COST ESCALATION
  - REDUCE MAINTENANCE RELATED ABNORMAL EVENTS
  - 5 YEAR, $6.9 MILLION EFFORT
- ROBOTICS
  - ESTABLISH UTILITY ADVISORY PANEL
  - DEVELOP GUIDELINES FOR POWER PLANT ROBOTIC AIDS
  - DEVELOP ROBOTS OPTIMIZED TO ASSIST IN PLANT OPERATION, SURVEILLANCE AND MAINTENANCE
  - 6^+ YEAR, $12.9 MILLION EFFORT

EPRI

OVERVIEW OF EPRI'S NUCLEAR POWER PROGRAM

- PLANT LIFE EXTENSION
  - PILOT PLANT EVALUATION
  - GENERIC BWR & PWR EVALUATIONS
  - CRITICAL STRUCTURES, COMPONENTS & SYSTEMS
  - RECORDS REQUIREMENTS
  - Product → LIFE EXTENSION GUIDELINES
  - 5 YEAR, $7 MILLION EFFORT

EPRI

OVERVIEW OF EPRI'S NUCLEAR POWER PROGRAM

- COMPONENT RELIABILITY
  - NDE R&D - INSPECTION & DETECTION
  - DYNAMIC PLUS STATIC PIPING LOADS
  - CRACK GROWTH
- CORROSION CONTROL
  - HYDROGEN WATER CHEMISTRY
- STEAM GENERATOR
  - LONG TERM EFFECTIVENESS OF CORRECTIVE MEASURES
  - NEWER PROBLEMS NOT COMPLETELY ADDRESSED
  - INCENTIVE: COSTS OF LOST CAPACITY HIGH
  - 5 YEAR, $30 MILLION EFFORT
  - INTERACTIONS WITH NRC, INPO, DOE

OVERVIEW OF EPRI'S NUCLEAR POWER PROGRAM

- LOW LEVEL WASTE
  + WASTE & VOLUME REDUCTION
  + RADWASTE DIRECT ASSAY
  + LLW DISPOSAL TECHNOLOGIES
- COOLANT TECHNOLOGY
  + CHEMICAL CONTROL
  + RADIATION CONTROL
- NUCLEAR PLANT CONSTRUCTABILITY
  + EQUIPMENT QUALIFICATION DATA BASE
  + CAE GUIDELINES

OVERVIEW OF EPRI'S NUCLEAR POWER PROGRAM

STEAM GENERATOR GUIDELINES

- STEAM GENERATOR REFERENCE BOOK
  - COMPENDIUM OF OPERATING EXPERIENCE, LABORATORY DATA, AND GUIDELINES
  - RECOMMENDS CORRECTIVE ACTION FOR S/G CORROSION
  - RECOMMENDS REPLACEMENT S/G AND B-O-P DESIGN FEATURES
- SECONDARY WATER CHEMISTRY GUIDELINES
  - WITHOUT REGULATION, ADOPTED BY U.S. & FOREIGN UTILITIES
  - ESSENTIALLY ELIMINATES DENTING AS A CONCERN
- NDE GUIDELINES
  - EPRI NDE CENTER CLASSES TO TRAIN EDDY CURRENT ANALYSTS
  - EPRI NDE CENTER CLASSES ON PLANNING UTILITY ISI PROGRAMS AND MONITORING VENDORS (STARTS APRIL 1986)

EPRI

OVERVIEW OF EPRI'S NUCLEAR POWER PROGRAM

BWR PIPE CRACK GUIDELINES

- NDE INSPECTION GUIDELINES
(COORDINATION PLAN - EPRI, BWROG & NRC - LEADS TO MUTUALLY SATISFACTORY CREDIBLE INSPECTION)
- BWR WATER CHEMISTRY GUIDELINES
(WITHOUT REGULATIONS, THESE GUIDELINES ARE WIDELY ADOPTED)
- HYDROGEN WATER CHEMISTRY IMPLEMENTATION GUIDELINES
(COMMON BASE FOR DESIGNING FACILITIES FOR STORING AND INJECTING WATER INTO A BWR)

NUCLEAR SAFETY ANALYSIS CENTER

NRC BRIEFING

A. DAVID ROSSIN

FEBRUARY 27, 1986

NSAC REPORTS
-1985-

REPORT NUMBER	TITLE
75	FRACTURE EVALUATION OF A W REACTOR DURING A PTS TRANSIENT
83	BRUNSWICK DECAY HEAT REMOVAL PROBABILISTIC SAFETY STUDY
84	ZION RESIDUAL HEAT REMOVAL SYSTEM PROBABILISTIC RISK ASSESSMENT
85	LOSSES OF OFF-SITE POWER AT U.S. NUCLEAR POWER PLANTS-ALL YEARS THROUGH 1984-
86	REALISTIC ECCS EVALUATION METHODOLOGY FOR ADVANCED LWRs
87	PLANT SPECIFIC COMPARED TO GENERIC ASSESSMENT OF STATION BLACKOUT
88	RHR EXPERIENCE REVIEW AND SAFETY ANALYSIS
89	FRACTURE EVALUATION OF A C-E REACTOR DURING A PTS TRANSIENT
90	DEVELOPING A LIVING SCHEDULE FUNDAMENTAL CONCEPT
91	A PARAMETRIC STUDY OF AN ANTICIPATED TRANSIENT WITHOUT SCRAM IN A W FOUR-LOOP PLANT
92	SETPOINT RELAXATION ANALYSIS FOR SCRAM REDUCTION USING RETRAN-02 ON C-E ANALOG PWRs
93	SETPOINT RELAXATION ANALYSIS FOR SCRAM REDUCTION USING RETRAN-02 ON C-E DIGITAL PWRs

NSAC REPORTS
-1985-
(CONTINUED)

REPORT NUMBER	TITLE
94	REDUCING SCRAM BY MODIFYING REACTOR SETPOINTS FOR A W 4-LOOP PLANT
NP-3975	ANALYSIS OF THE HYDROGEN BURN IN THE TMI-2 CONTAINMENT
NP-3938	A FULL RANGE DRIFT FLUX CORRELATION FOR VERTICAL FLOW
NP-4146	EPRI'S R&D CONTRIBUTIONS TO THE TECHNICAL BASIS FOR REVISION OF ECCS RULES

NSAC REPORTS
-1986-

95	GENERIC SAFETY ISSUE TRACKING AND EVALUATION SUMMARY DESCRIPTIONS-1985 (IN PRESS)
96	EFFECT OF DIESEL START TIME ON BWR/6 PEAK CLAD TEMPERATURE-LICENSING BASIS CALCULATIONS

NSAC INTERNATIONAL ASSOCIATES

13 ASSOCIATES

NSAC ASSOCIATES (DOMESTIC)

AMERICAN ELECTRIC POWER
FLORIDA POWER CORPORATION
NEW YORK POWER AUTHORITY
DAIRYLAND
TOLEDO EDISON
NEBRASKA PUBLIC POWER DISTRICT
OMAHA PUBLIC POWER DISTRICT (JOINED EPRI)

ALL PENNSYLVANIA AND NEW JERSEY NUCLEAR
UTILITIES TREATED LIKE MEMBERS OR ASSOCIATES

DETROIT EDISON (1986)
CONSUMERS POWER (1986)
GULF STATES (1986)
MIDDLE SOUTH (1986)
LONG ISLAND LIGHTING (1985/86)

COMBINED ALL NSAC ACTIVITIES INTO ONE RESEARCH PROJECT

RP-2420

GENERIC SAFETY ANALYSIS

- ALWR SUPPORT
- NUMARC SUPPORT
- MATRIX MANAGEMENT

(IN-HOUSE RESEARCH COSTS)

REGULATORY SAFETY ASSESSMENT

HYDROGEN CONTROL OWNERS GROUP

(NO EPRI BASE PROGRAM FUNDING IN 1986)

NUCLEAR POWER DIVISION
PROGRAMS UNDER NSAC MATRIX MANAGEMENT

MATRIX	MANAGER
PRESSURIZED THERMAL SHOCK (COMPLETED)	CHEXAL
STEAM GENERATOR INTEGRITY	LANG
DECAY HEAT REMOVAL	VINE
PIPING INTEGRITY	CHEXAL
TECHNICAL SPEC. IMPROVEMENT	POWER
DIESEL GENERATOR PERFORMANCE	WYCKOFF

SCRAM ON LOW SG LEVEL

- SOME MILITARY PLANTS HAVE NOT USED S/G LEVEL IN SCRAM SYSTEMS

- HISTORICALLY S/G LEVEL SCRAM WAS A COST REDUCTION TRADE-OFF MADE RATHER THAN TO INCORPORATE MORE PRESSURIZER SAFETY VALVE CAPACITY INDICATED BY TRANSIENT ANALYSIS

- BETTER ANALYSIS TOOLS AVAILABLE NOW

SCRAM ON TURBINE TRIP

- CAN WE RAISE THE SCRAM SETPOINT ABOVE 10% POWER LEVEL?

CONCLUSIONS

SG LEVEL SCRAM

1. IN ALL CASES WHERE REACTOR PROTECTION IS REQUIRED BECAUSE OF LOSS OF THE S/G HEAT SINK, ADEQUATE PROTECTION IS PROVIDED BY SCRAMS OTHER THAN S/G LEVEL.

2. W PLANT STEAM FLOW/FEED MISMATCH SCRAM IS NOT NECESSARY.

3. W PLANT LOW-LOW SG LEVEL CAN BE REMOVED FOR REACTOR POWER BELOW 50%.

TURBINE TRIP

4. W PLANT WITH 40% STEAM DUMP CAN INCREASE SCRAM SETPOINT FROM 10% TO 50% POWER ON TURBINE TRIP.

5. CE PLANT WITH 40% STEAM DUMP CAN INCREASE SCRAM SETPOINT ON TURBINE TRIP FROM 15% TO AT LEAST 35%. A HIGHER POWER LEVEL PROBABLY CAN BE DEMONSTRATED THROUGH FURTHER ANALYSIS.

CONCLUSION

ALL NSAC WORK INTEGRATED TOWARD RESOLUTION OF SAFETY ISSUES

EXAMPLE: STATION BLACKOUT

- LOSS OF OFF-SITE POWER
- DIESEL GENERATOR RELIABILITY
- DIESEL GENERATOR START TIME
- DC POWER RELIABILITY WORKSHOP
- DECAY HEAT REMOVAL
- ENGINEERING REVIEW OF DESIGN CHANGES
- PRA ON DHR (BASED ON OCONEE PRA EXPERIENCE)
- COST/BENEFIT ANALYSIS; ON-SITE AVERTED COSTS

GENERIC SAFETY ANALYSIS PROGRAM

W. LAYMAN

J.CHAO
B. CHEXAL
J. LANG
J. POWER
W. REULAND
G. VINE

- ABOUT 30 ISSUES ARE BEING TRACKED

- ANALYSIS UNDERWAY ON THE FOLLOWING ISSUES:
  - SHUTDOWN DECAY HEAT REMOVAL
  - SCRAM REDUCTION
  - STATION BLACKOUT
  - REDUCTION OF PIPING RESTRAINTS
  - TECHNICAL SPECIFICATION IMPROVEMENT

- ANALYSES COMPLETED FOR THE FOLLOWING ISSUES:
  - PRESSURE VESSEL THERMAL SHOCK
  - BWR ATWS
  - STEAM GENERATOR SAFETY ISSUES
  - LOSS OF OFFSIGHT POWER DATA BASE
  - DHR RISK EVALUATION USING PRA
    (WITH REGULATORY TECHNICAL ASSESSMENT GROUP)

	OVERALL INDUSTRY PRIORITY
REACTOR VESSEL THERMAL SHOCK	A
STEAM GENERATOR INTEGRITY	AAA
SHUTDOWN HEAT REMOVAL SYSTEMS	AAA
ENVIRONMENTAL QUALIFICATION OF EQUIPMENT	AA
EMERGENCY PLANNING	AA
SINGLE FAILURE CRITERIA	A
ATWS	AAA
RELIABILITY OF VITAL EQUIPMENT	AA
NEAR TERM REGULATORY REQUIREMENT PRIORITIZATION	A
SITING CRITERIA	A
WATER HAMMER	A
CONTAINMENT INTEGRITY	AA
FRACTURE TOUGHNESS OF S/G AND RCP SUPPORTS	INACTIVE
SYSTEMS INTERACTIONS	AA
SEISMIC DESIGN CRITERIA	AAA

	OVERALL INDUSTRY PRIORITY
CONTAINMENT EMERGENCY SUMP PERFORMANCE	A
STATION BLACKOUT	AAA
SEISMIC QUALIFICATION OF EQUIPMENT	AA
SAFETY IMPLICATIONS OF CONTROL SYSTEMS	AA
HYDROGEN CONTROL MEASURES AND BURN EFFECTS	AAA
DC POWER RELIABILITY	AA
I&C SYSTEM RELIABILITY	AA
PLANT AIR SYSTEMS ANOMALIES	AA
LIGHTNING PROTECTION	A
IN-CORE INSTRUMENTATION TUBE RUPTURE	INACTIVE
TWO-PHASE NATURAL CIRCULATION	A
HP INJECTION SYSTEMS	AA
INADVERTENT CRITICALITY	INACTIVE
HIGH STRENGTH BOLTING DEGRADATION	INACTIVE
PIPE CRACKS IN BOILING WATER REACTORS	AA
PIPING INTEGRITY	AA
TECHNICAL SPECIFICATIONS	AA

RESIDUAL HEAT REMOVAL EXPERIENCE REVIEW AND SAFETY ANALYSIS

PWRs: NSAC-52
BWRs: NSAC-88

- ACTUAL LOSSES OR DEGRADATIONS OF OPERATING RHR SYSTEMS IN SHUTDOWN COOLING MODE:

  PWRs: 96 EVENTS OVER FIVE YEARS

  BWRs: 90 EVENTS OVER SEVEN YEARS

- MAJOR SAFETY BENEFIT IN COLD SHUTDOWN COMES FROM MAINTAINING ADEQUATE COOLANT INVENTORY IN REACTOR (RECOGNITION, PREVENTION, AND RECOVERY ARE ALL IMPORTANT).

- SHUTDOWN REACTOR SAFETY IS ENSURED BEST BY TRAINING, PROCEDURES, MANAGEMENT AND ADMINISTRATIVE CONTROLS WHICH ADDRESS POTENTIAL CHALLENGES AND MAKE EFFECTIVE USE OF ALL AVAILABLE EQUIPMENT.

- MAJOR DESIGN CHANGES ARE NOT NECESSARY TO COPE WITH CONCERNS RAISED BY OPERATING EXPERIENCE ANALYSIS.

- MAJOR CONCLUSIONS AND RECOMMENDATIONS OF NSAC-52 FOR PWRs ARE CONSISTENT WITH RECENT INPO SOER AND AEOD CASE STUDY. INPO IS DEVELOPING SOER BASED ON NSAC-88 FOR BWRs.

- MANY AREAS OF POTENTIAL SAFETY CONCERN HAVE ALREADY BEEN ANALYZED. OVER 100 NRC AND INPO EVENT REPORTS HAVE BEEN WRITTEN ON EVENTS IN NSAC-52 AND NSAC-88.

- PLANT SPECIFIC PRAs OF ZION (NSAC-84) AND BRUNSWICK (NSAC-83) VALIDATED CONCLUSIONS OF OPERATING EXPERIENCE STUDIES. PRINCIPLE CONCLUSIONS CONCERNING RISK OF CORE OR PLANT DAMAGE DURING SHUTDOWN:

  - RHR WEAKNESSES OBSERVED IN OPERATING EXPERIENCE EVALUATED BY PRA TO RESULT IN ACCEPTABLY SMALL RISK OF CORE MELT.

  - RISK OF PLANT DAMAGE FURTHER REDUCED BY ENSURING ADEQUATE AVAILABILITY OF DHR SYSTEMS, INVENTORY ADDITION SYSTEMS, AND KEY SUPPORT SYSTEMS DURING SHUTDOWN. THIS INCLUDES MAINTAINING MAIN CONDENSER (AND STEAM GENERATORS ON PWRs) AVAILABLE DURING HOT SHUTDOWN.

  - PRA SENSITIVITY STUDIES SHOWED PROCEDURES, TRAINING, ADMINISTRATIVE AND MAINTENANCE IMPROVEMENTS ARE VERY IMPORTANT.

PLANT INCIDENT ASSISTANCE

PLANT: RANCHO SECO

DATE: DECEMBER 26, 1985

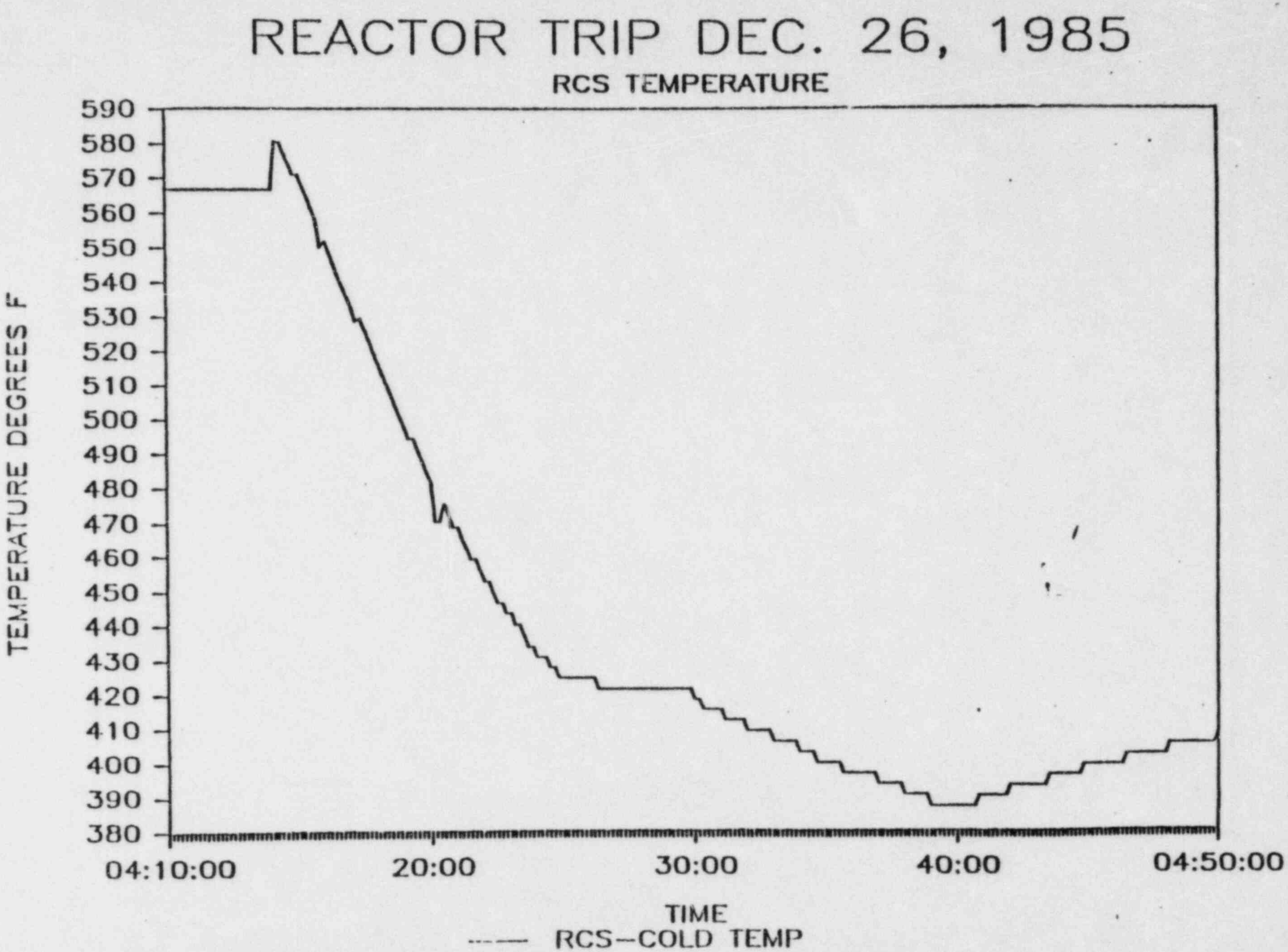
CONTACT: RON COLOMBO PHONE CALL

ACTION: LAYMAN, ROSSIN
TO PLANT 12/26

FOLLOW-UP: PTS CALCULATION REPORT TO SMUD 1-7-86

CONCLUSION: COOLDOWN RATE SIMILAR TO "LIGHT BULB" EVENT, BUT SHORTER DURATION AND LESS TEMPERATURE CHANGE

CONTACT ARRANGED FOR ICS CONSULTATION


REACTOR TRIP DEC. 26, 1985
RCS TEMPERATURE
TEMPERATURE DEGREES F
590
580
570
560
550
540
530
520
510
500
490
480
470
460
450
440
430
420
410
400
390
380
04:10:00
20:00
30:00
40:00
04:50:00
TIME
RCS-COLD TEMP

EPRI ASSESSMENT OF RANCHO SECO
REACTOR VESSEL TRANSIENT 12/26/85

12/31 RECEIVED CONFIRMATION OF TRANSIENT INFORMATION AND REACTOR VESSEL FLUENCE FROM SMUD

1/6 COMPLETED ANALYSIS REPORT AND MAILED FINAL REPORT TO SMUD

CONCLUSION

RANCHO SECO REACTOR VESSEL BELTLINE REGION HAS ADEQUATE STRUCTURAL INTEGRITY FOR RETURN TO SERVICE WITHOUT FURTHER EVALUATION.

DRAFT ASME SECTION XI APPENDIX FOR EVALUATION OF UNANTICIPATED OPERATIONAL TRANSIENTS

ALLOWS DEMONSTRATION OF ADEQUATE STRUCTURAL INTEGRITY OF THE REACTOR VESSEL BELTLINE WITHOUT DOING FURTHER INTEGRITY ANALYSES AS LONG AS

1. REACTOR COOLANT PRESSURE HAS NOT EXCEEDED DESIGN PRESSURE.

2. T_C-RT_{NDTS} HAS NOT BEEN LESS THAN 55°F DURING THE TRANSIENT.

$RT_{NDTS} = RT_{NDTo} + \Delta RT_{NDT} + 2\sigma$

$\Delta RT_{NDT} = [CF]\, F^{(.28-0.1 \text{ LOG } F)}$ (REG GUIDE 1.99)

CF FROM SECY-82-465

ASME SECTION XI APPENDIX FOR EVALUATION OF UNANTICIPATED TRANSIENTS

1/83 GRIESBACH AND MARSTON AS ASME SECTION XI MEMBERS GOT COMMITTEE TO ESTABLISH A WORKING GROUP ON OPERATIONAL TRANSIENTS.

GRIESBACH HAS BEEN THE CHAIRMAN OF THE WORKING GROUP SINCE 1/83.

83-85 VIRTUALLY ALL OF THE FUNDING FOR THE WORKING GROUP TECHNICAL WORK WAS FROM EPRI.

2/86 THE DRAFT ASME SECTION XI APPENDIX FOR EVALUATION OF UNANTICIPATED OPERATIONAL TRANSIENTS WILL GO TO THE SECTION XI SUBCOMMITTEE FOR ITS APPROVAL.

NRC TECHNICAL PERSONNEL HAVE BEEN INVOLVED IN THE PREPARATION AND REVIEW OF THE DRAFT AND HAVE ENDORSED IT.

WHAT DETERMINES THE RISK FROM STATION BLACKOUT

-FOUR IMPORTANT FACTORS-

- OFF-SITE POWER UNRELIABILITY
  - IS OF DECLINING IMPORTANCE BECAUSE
  - OVERALL UNRELIABILITY IS EXTREMELY LOW
  - PLANT SPECIFIC UNRELIABILITY IS DIFFICULT TO DETERMINE

- EMERGENCY ON-SITE POWER REDUNDANCY AND UNRELIABILITY
  - THESE ARE GENERALLY RECOGNIZED AS THE BACKBONE OF A NUCLEAR PLANTS SHUTDOWN POWER RELIABILITY

- OTHER RECOGNIZED BUT NOT FULLY QUALIFIED AC POWER SOURCES
  - THESE PLAY A MAJOR ROLE OUTSIDE THE U.S.
  - THEY MUST BE COVERED BY OPERATING AND SURVEILLANCE TESTING PROCEDURES
  - DO NOT HAVE TO BE SEISMICALLY QUALIFIED

WHAT DETERMINES THE RISK FROM STATION BLACKOUT

-Four Important Factors-

- COPING DURATION
  - IS THE PERIOD DURING WHICH THE PLANT CAN PREVENT CORE UNCOVERY FROM THE TIME WHEN OFF-SITE AND EMERGENCY ON-SITE AC POWER IS LOST
  - OVERSEAS UTILITIES AND REGULATORS GENERALLY ACCEPT THAT <u>RECOGNIZED BUT NOT QUALIFIED</u> AC POWER SOURCES PROVIDE COPING TIME
  - HOW TO INCLUDE SUCH SOURCES IN COPING DURATION MUST YET BE WORKED OUT BY U.S. NUCLEAR INDUSTRY AND NRC.

EMERGENCY DIESEL GENERATOR NOTES FROM OVERSEAS

- GERMANS HAVE 4 EDGs PER NUCLEAR UNIT
  - ONE IS SUFFICIENT IF NO LOCA
  - HAVE 2 ADDITIONAL BACKUP EDGs FOR AFW
  - BUT WITHOUT CHARGING, BATTERIES WILL LAST ONLY 30 MINUTES

- IN ADDITION TO EDGs, SWEDISH PLANTS HAVE PERMANENTLY INSTALLED BACKUP GAS TURBINE GENERATORS THAT CAN LOAD IN 3 MINUTES

- FRENCH HAVE ONE PORTABLE GAS TURBINE GENERATOR FOR EACH 4 NUCLEAR UNITS AND FACILITIES FOR CONNECTING IN 2 HOURS

- FRENCH HAVE INSTALLED ONE 140 KW STEAM TURBINE GENERATOR PER NUCLEAR UNIT TO:
  - PROVIDE SEAL INJECTION WATER IN PRIMARY SYSTEM OF PWR
  - CHARGE BATTERIES
  - BUT WITHOUT CHARGING, BATTERIES WILL LAST ONLY ONE HOUR

- FINNISH AND SWEDISH EDG UNRELIABILITY IS 1.2%

EMERGENCY DIESEL GENERATOR NOTES FROM OVERSEAS

- GERMANS REPORT THEIR UNRELIABILITY TO BE 0.8%

- FOR PLANNED STARTS, SWISS BRING EDGs TO SPEED OVER A 10 MINUTE PERIOD AND THEN LOAD OVER 4 MINUTES

- BELGIUM HAS DROPPED BACK TO 2 FAST STARTS PER YEAR

- SWEDEN HAS BACKFIT A SOFT START DEVICE ONTO ITS NUCLEAR PLANT EDGs

  -FOR PLANNED STARTS THEY ARE BROUGHT TO SPEED IN 30 SECONDS (VS. U.S. 10 SECONDS)

U.S. DIESEL GENERATOR RELIABILITY SURVEY

PRELIMINARY DATA
1983 AND 1984

- 67% UTILITIES

UNRELIABILITY ~1.3%

EPRI ACTIVITIES ON REQUIRED
EMERGENCY DIESEL GENERATOR START TIME

BINDI CHEXAL
NUCLEAR POWER DIVISION

FEBRUARY 1986

CURRENT DIESEL START TIME REQUIREMENT AND ITS BASIS

REQUIREMENT

- PER NRC REG. GUIDE 1.108, EMERGENCY DIESEL GENERATORS MUST REACH RATED SPEED AND VOLTAGE WITHIN 10 SECONDS.

BASIS

- HYPOTHETICAL LARGE BREAK LOCA
- COINCIDENT LOSS OF OFF-SITE POWER
- FAILURE OF ONE OF THE EDGs TO SUPPLY POWER
- APPENDIX K CONSERVATISM

<u>PROBLEM</u>

- FAST STARTS, FAST LOADING, AND A LARGE NUMBER OF TESTS CONTRIBUTE TO STESS AND WEAR ON EMERGENCY DIESEL GENERATORS. FAST STARTING IS ACCEPTED WORLDWIDE AS A MAJOR CAUSE OF EDG DEGRADATION.

  - NRC IS ENCOURAGING A REDUCTION IN THE NUMBER OF FAST STARTS.

  - OVERSEAS MOST COUNTRIES ARE SLOWING DOWN THE RATE AT WHICH THEY START AND LOAD THEIR DIESELS.

  - ONE U.S. DIESEL MANUFACTURER HAS STATED THAT IN NUCLEAR POWER PLANT SERVICE, A DIESEL'S WEAR IS DEPENDENT PRIMARILY ON THE NUMBER OF FAST STARTS.

  - EDF HAS PERFORMED ANALYSES THAT SHOW THAT FAST START MECHANICAL STRESSES ARE 21% HIGHER THAN THE STRESSES AT RATED SPEED

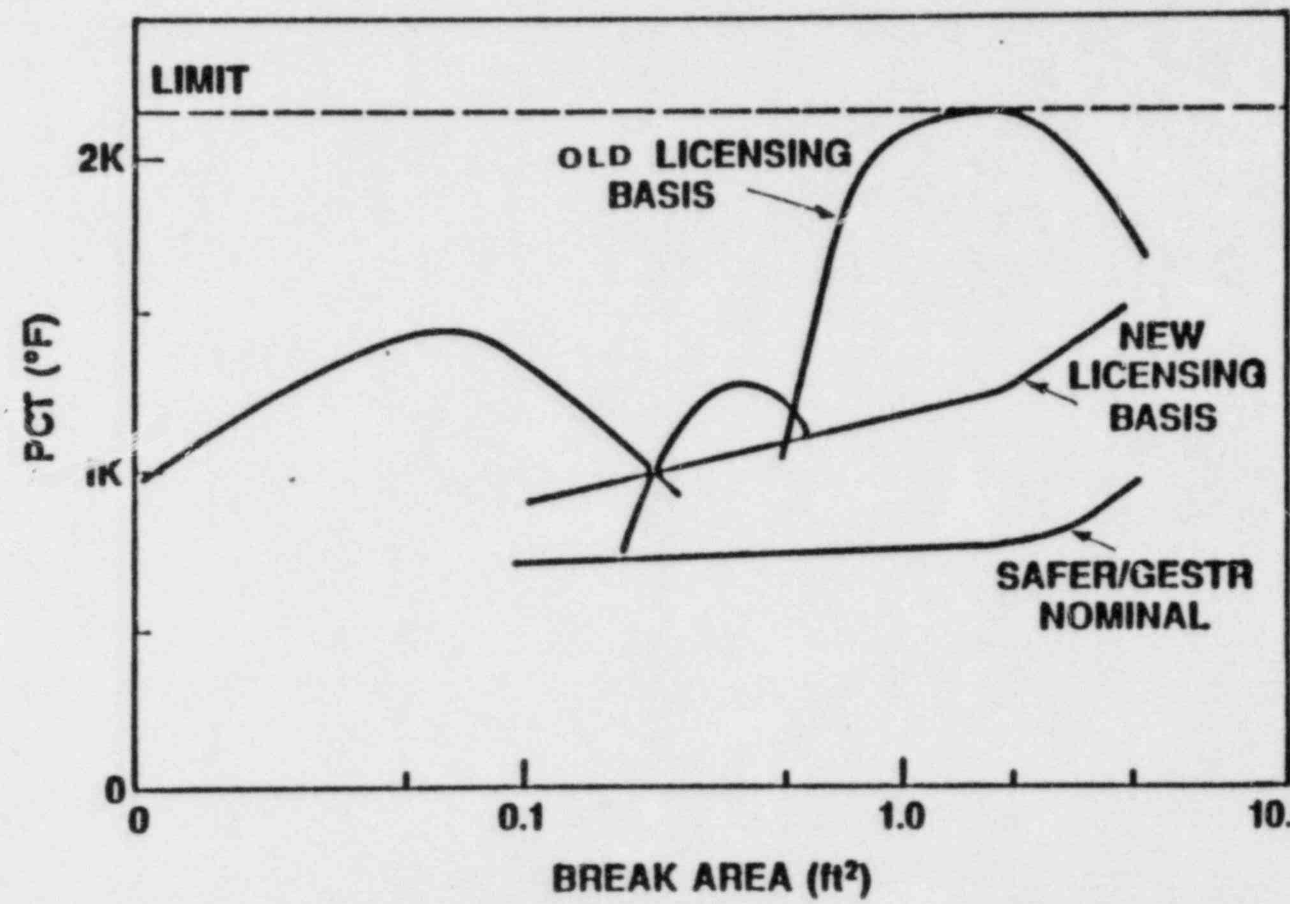
RESOLUTION

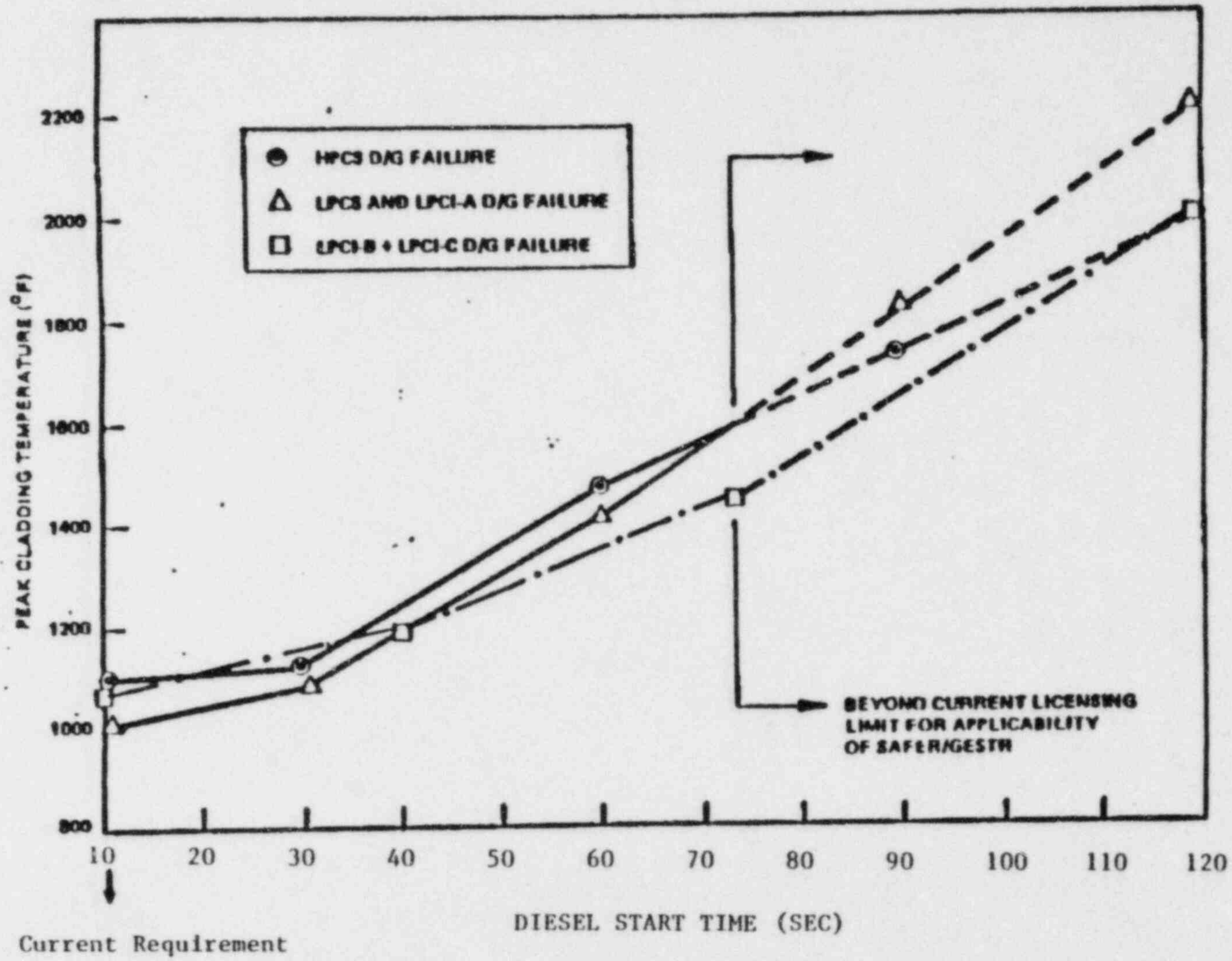
- USE OF REALISTIC APPROACH TO ECCS LICENSING BASED ON NRC PROPOSED DOCUMENT SECY 83-472.

- AVAILABLE IN GE SAFER/GESTR LOCA PACKAGE LICENSED FOR USE ON JET PUMP BWRS.

- CURRENTLY UNDER DEVELOPMENT FOR W PWRS BY W.

JET PUMP BWRs





PCT vs Diesel Start Time

CONCLUSIONS

THIS STUDY HIGHLIGHTS THE USE OF A REALISTIC APPROACH TO ECCS LICENSING (ACCEPTED BY NRC) TO REDUCE DIESEL DEGRADATION AND HELP RESOLVE THE STATION BLACKOUT ISSUE. OTHER AREAS WHERE OPERATIONAL FLEXIBILITY CAN BE IMPROVED USING THE SAME APPROACH ARE:

- IMPROVED FUEL UTILIZATION
- RELAXATION OF PERFORMANCE REQUIREMENTS
- IMPROVEMENTS IN TECH SPECS
- ALLOW PLANT POWER INCREASE

LOSSES OF ALL OFF-SITE POWER
AT U.S. NUCLEAR PLANTS

-1985-

	NUMBER OF EVENTS IN 63.9 TOTAL SITE CALENDAR YEARS	LOSSES PER SITE YEAR
Ia. Longer than 30 minutes	2	0.031
Ib. Less than 30 minutes	4	0.063
Total	6	0.094
II.	0	-
III.	1	0.016
IV.	1	0.016

LOSSES OF ALL OFF-SITE POWER
AT U.S. NUCLEAR PLANTS

Three Most Recent Years (1983, 1984, 1985)

	NUMBER OF EVENTS IN 172.9 TOTAL SITE CALENDAR YEARS	LOSSES PER SITE YEAR
Ia. Longer than 30 minutes	4	0.023
Ib. Less than 30 minutes	9	0.052
Total	13	0.075
II.	3	0.017
III.	8	0.046
IV.	8	0.046

LOSSES OF ALL OFF-SITE POWER
AT U.S. NUCLEAR PLANTS

-ALL YEARS THROUGH 1985-

		NUMBER OF EVENTS IN 664.9 TOTAL SITE CALENDAR YEARS	LOSSES PER SITE YEAR
Ia. Longer than 30 minutes		25	0.038
Ib. Less than 30 minutes		30	0.045
	Total	55	0.083
II.		7	0.011
III.		34	0.051
IV.		12	0.018

LOSS OF ALL OFF-SITE POWER COMPARISONS

-LOSSES PER SITE YEAR-

LATEST YEAR

			1984	1985
1a	Longer than 30 minutes		0.000	0.031
1b	Less than 30 minutes		0.089	0.063
		Total	0.089	0.094

3 MOST RECENT YEARS

		Thru 1984	Thru 1985
1a		0.013	0.023
1b		0.031	0.052
	Total	0.044	0.075

ALL YEARS

		Thru 1984	Thru 1985
1a		0.038	0.038
1b		0.043	0.045
	Total	0.081	0.083

LOSS OF OFF-SITE POWER
AT U.S. NUCLEAR PLANTS

- ALL YEARS THROUGH 1985 -

THE FIFTEEN OUTAGES LONGER THAN ONE HOUR

Plant	Duration hours:minutes	Year	Where Caused by Severe Weather
Arkansas Nuclear One	1.29	1978	
Dresden	4:00	1965	Tornado[1]
Farley	2:45	1983	
Ft. St. Vrain	1:45	1983	Snow & Ice
Indian Point	6:28	1977	Lightning[2]
	1:45	1980	Lightning
Millstone	5:00	1976	Hurricane
	5:30	1985	Hurricane[3]
Pilgrim	2:40	1977	Snow & Ice[4]
	8:54	1978	Snow & Ice & Salt Spray
Prairie Island	1:02	1980	
St. Lucie	2:50	1977	
Turkey Point	1:02	1977	
	2:00	1977	
	2:05	1985	

[1]At the time of this event Dresden had 5 transmission lines all on one right-of-way. Today it has 12 transmission lines located on two right-of-ways.

[2]Line arrangement entering plant has been improved and it is now a technical specification that at least one of three gas turbines is operable.

[3]As a precautionary step, both units were shutdown in advance of peak storm conditions-units were shutdown when power was lost.

[4]Have equipment to wash insulators while energized.

LOSS OF OFF-SITE POWER
AT U.S. NUCLEAR PLANTS

- ALL YEARS THROUGH 1985 -

SOME INTERESTING OBSERVATIONS .

- Losses of off-site power longer than:

0 minutes	27 of 65 sites
30 minutes	14 of 65 sites
1 hour	10 of 65 sites
2 hours	7 of 65 sites

- 38 of 65 sites have never lost all off-site power.

- 51 of 65 sites have not had a loss longer than 30 minutes.

- The median duration of all losses is 1/2 hour.

- There were only 4 losses of all off-site power longer than 30 minutes in the last 3 years.

WEATHER-RELATED LOSSES OF OFF-SITE POWER

- ALL YEARS THROUGH 1985 -

- Total weather-related losses of off-site power and unit trip
    - 15 events in 664.9 site years
    - 0.023 events/site year
- Seven weather events have been substantial and 8 minor.
- One of the weather-related losses occurred in 1985 and one in 1983. The others occurred a number of years ago: two in 1980, one in 1965, and the remainder in the 1970's.
- The small number of weather losses is the result of correcting identified weaknesses.
- Median duration for weather-related losses 1 to 2 hours
- Longest duration 8:54 in 1978
    - Corrective actions are expected to prevent recurrence.

ENGINEERING & OPERATIONS DEPARTMENT

NRC REVIEW

DEPARTMENT DIRECTOR: A. RUBIO

FEBRUARY 27, 1986

ENGINEERING & OPERATIONS DEPT.

- NUCLEAR PLANT LIFE EXTENSION & CONSTRUCTIBILITY

- PLANT AVAILABILITY

- LOW LEVEL WASTE & COOLANT TECHNOLOGY

NUCLEAR PLANT LIFE EXTENSION AND CONSTRUCTIBILITY

OBJECTIVES:

- EXTEND OPERATING LIFETIME BEYOND 40 YEAR LICENSED TERM
- REDUCE CONSTRUCTION COSTS FOR OPERATING PLANT MODIFICATIONS
- ESTABLISH AND MAINTAIN EQUIPMENT QUALIFICATION

NUCLEAR PLANT LIFE EXTENSION

1985 KEY ACCOMPLISHMENTS

SURRY–1 AND MONTICELLO PILOT PROJECTS:

- DETAILED ASSESSMENT OF MAJOR COMPONENTS, SYSTEMS & STRUCTURES
- IDENTIFICATION OF UTILITY ACTIONS REQUIRED; MAINTENANCE, RECORDS, MATERIAL SAMPLING PLANS
- LIFE EXTENSION COSTS; MAINTENANCE, REPLACEMENTS, REPAIRS, INSPECTIONS
- LIFE EXTENSION IMPLEMENTATION & MANAGEMENT PLAN

STATUS: ~ 50% COMPLETE

NUCLEAR PLANT CONSTRUCTIBILITY

ISSUE:

- EXCESSIVE PLANT CONSTRUCTION COSTS; NEW CONSTRUCTION—OPERATING PLANT MODIFICATIONS

SOLUTIONS (PARTIAL):

- RESOLUTION OF KEY QUALITY ASSURANCE ISSUES
- INCREASED APPLICATION OF COMPUTER—AIDED ENGINEERING; IMPROVE PLANT INFORMATION MANAGEMENT
- IMPROVED DESIGN/CONSTRUCTION PRACTICES

NUCLEAR PLANT CONSTRUCTIBILITY

1985 KEY ACCOMPLISHMENTS

- DEVELOPED INTERIM GUIDELINES FOR APPLICATION OF COMPUTER—AIDED ENGINEERING
- COMPLETED MIDLAND 3—D CONSTRUCTION COMPUTER MODEL
- EPRI TO PROVIDE PROGRAM MANAGEMENT FOR NUCLEAR CONSTRUCTION ISSUES GROUP (NCIG)
- CONDUCTED WORKSHOP ON PLANT LAYUP & EQUIPMENT PRESERVATION

INTERIM GUIDELINES FOR CAE APPLICATION
DESIGN/CONSTRUCTION/OPERATIONS

NEED : REDUCE COSTS
- AUTOMATE PLANT WORK ACTIVITIES
- IMPROVED PLANT INFORMATION ACCESS AND CONTROL

SOLUTION: COMPUTER—AIDED ENGINEERING
- INTERACTIVE COMPUTER APPLICATIONS
- COMMON SUPPORTING PLANT DATA BASE

FURTHER NEEDS:
- SYSTEMATIC DEFINITION OF PLANT DATA STRUCTURE
- GUIDELINES FOR USEAGE
- INDUSTRY ACCEPTANCE

EQUIPMENT QUALIFICATION—ENVIRONMENTAL

KEY 1985 ACCOMPLISHMENTS

- PUBLISHED GUIDELINES/DATA BANK ON QUALIFICATION OF MECHANICAL EQUIPMENT

- CONDUCTED SEMINARS ON MAINTAINING EQUIPMENT QUALIFICATION

- AGING SPECIMENS(CABLE, DEVICES, LUBRICANTS) INSTALLED IN 7 PLANTS

EQUIPMENT QUALIFICATION

DEGRADED CORE

- PUBLISHED EQUIPMENT DATA FROM FULL SCALE HYDROGEN BURN TESTS AT NTS

- REPORT REVIEWED BY NRC SENIOR ADVISORY REVIEW PANEL PRIOR TO PUBLICATION

- NRC TO ISSUE POSITION ON RULEMAKING FOR LARGE DRY CONTAINMENTS BY MID–1986

LOW LEVEL WASTE

WASTE MINIMIZATION:

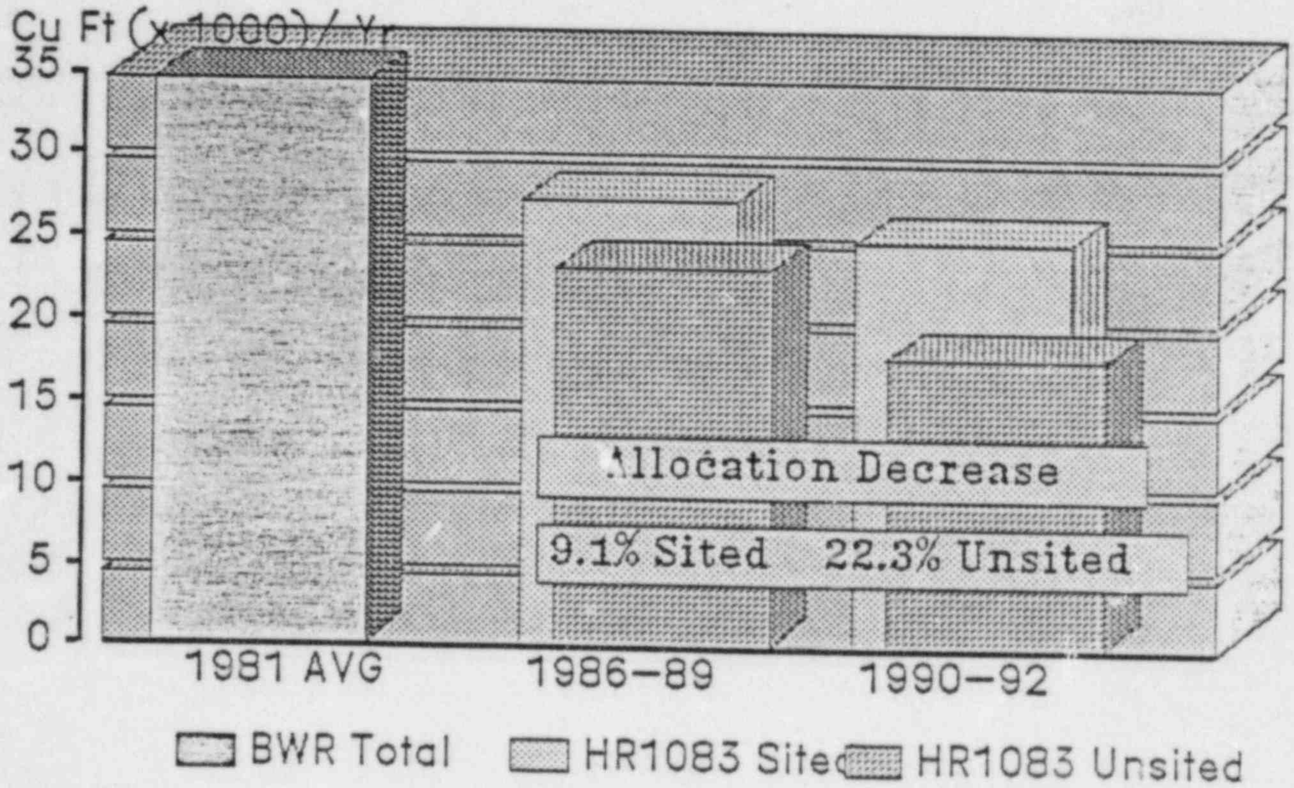
- DRY WASTE REDUCTION
- WET WASTE REDUCTION
- ADVANCED VOLUME REDUCTION TECHN.

REGULATORY CONCERNS/IMPACT:

- MONITORING
- DISPOSAL TECHNOLOGY
- STABILIZATION & STORAGE

BWR RADWASTE GENERATION

1981 ACTUAL VS. HR 1083 ALLOCATION



LOW LEVEL WASTE

TOPIC: Economics of Radwaste Volume Reduction

UTILITY NEED: Ability to perform side by side cost comparisons of available in-plant and mobile volume reduction technology.

APPROACH: Develop user friendly IBM compatible program to perform radwaste volume reduction economic analysis.

LOW LEVEL WASTE

TOPIC: Radwaste Package Direct Assay Tech.

UTILITY NEED: Capability to accurately identify and quantify radionuclides in the waste burial container.

APPROACH: Develop and demonstrate methods which would perform direct gamma & TRU radionuclide assay of waste containers.

LOW LEVEL WASTE

TOPIC: LLW Disposal Technologies

UTILITY NEED: Ability to influence & respond to state disposal agencies' alternatives to shallow land burial.

APPROACH: Develop generic design. Assess cost and performance features for disposal technology alternatives.

COOLANT TECHNOLOGY

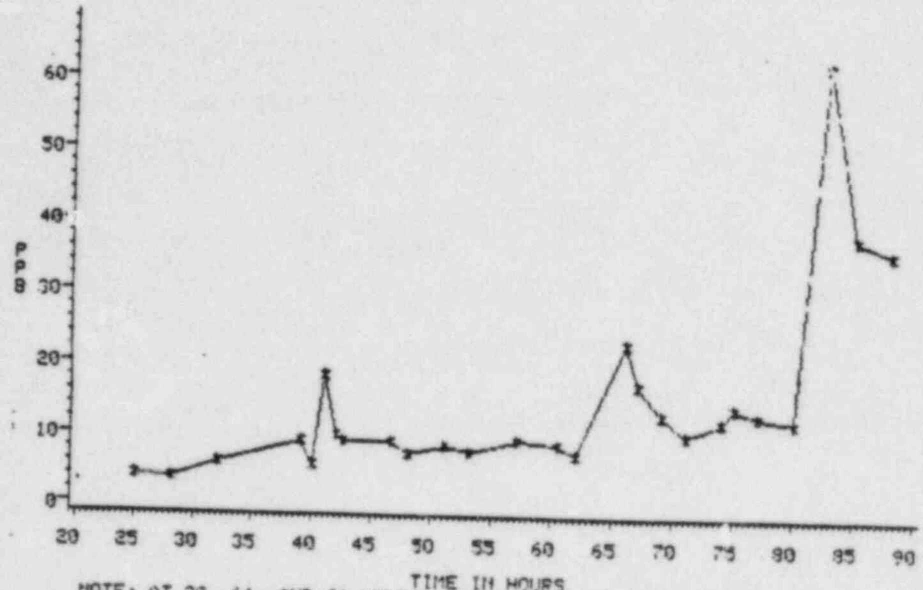
CHEMICAL CONTROL:

- ADVANCED MONITORING TECHNIQUES
- WATER QUALITY IMPROVEMENTS
- PRIMARY CHEMISTRY GUIDELINES

RADIATION CONTROL:

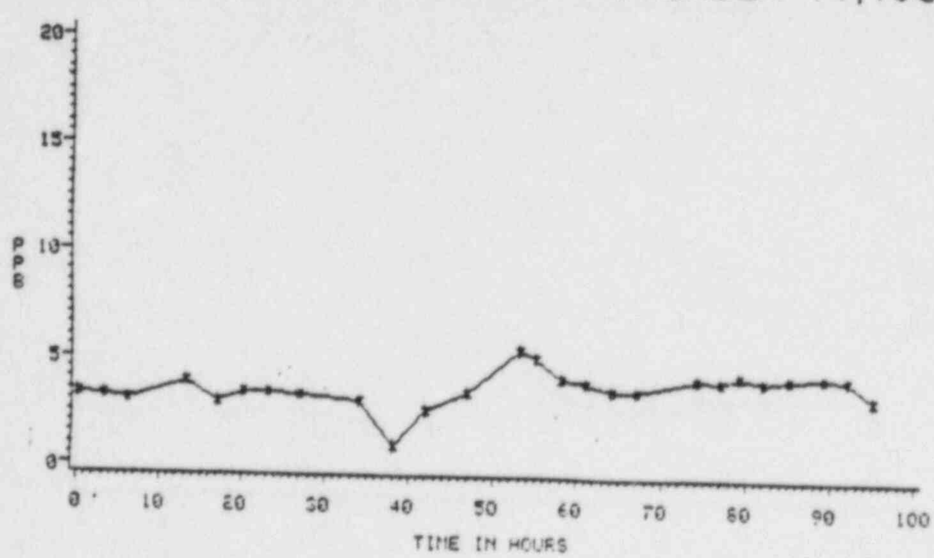
- ZINC INJECTION PASSIVATION
- SURFACE MODIFICATION
- PREFILMING
- COBALT REPLACEMENT
- LOMI DECONTAMINATION

SULFATE VALUES FOR S/G A
JUNE 19,1984 THROUGH JUNE 21,1984



NOTE: AT 32. 64. AND 81 HOURS. NEW PRECOATS WERE PLACED IN SERVICE

SULFATE VALUES FOR S/G A
NOVEMBER 8,1984 THROUGH NOVEMBER 11,1984



NOTE: AT 53 HOURS AND 88 HOURS. NEW PRECOATS WERE PLACED IN SERVICE

COOLANT TECHNOLOGY

TOPIC: Water Quality Improvements

UTILITY NEED: Water Chemistry controlled particularly for impurities.

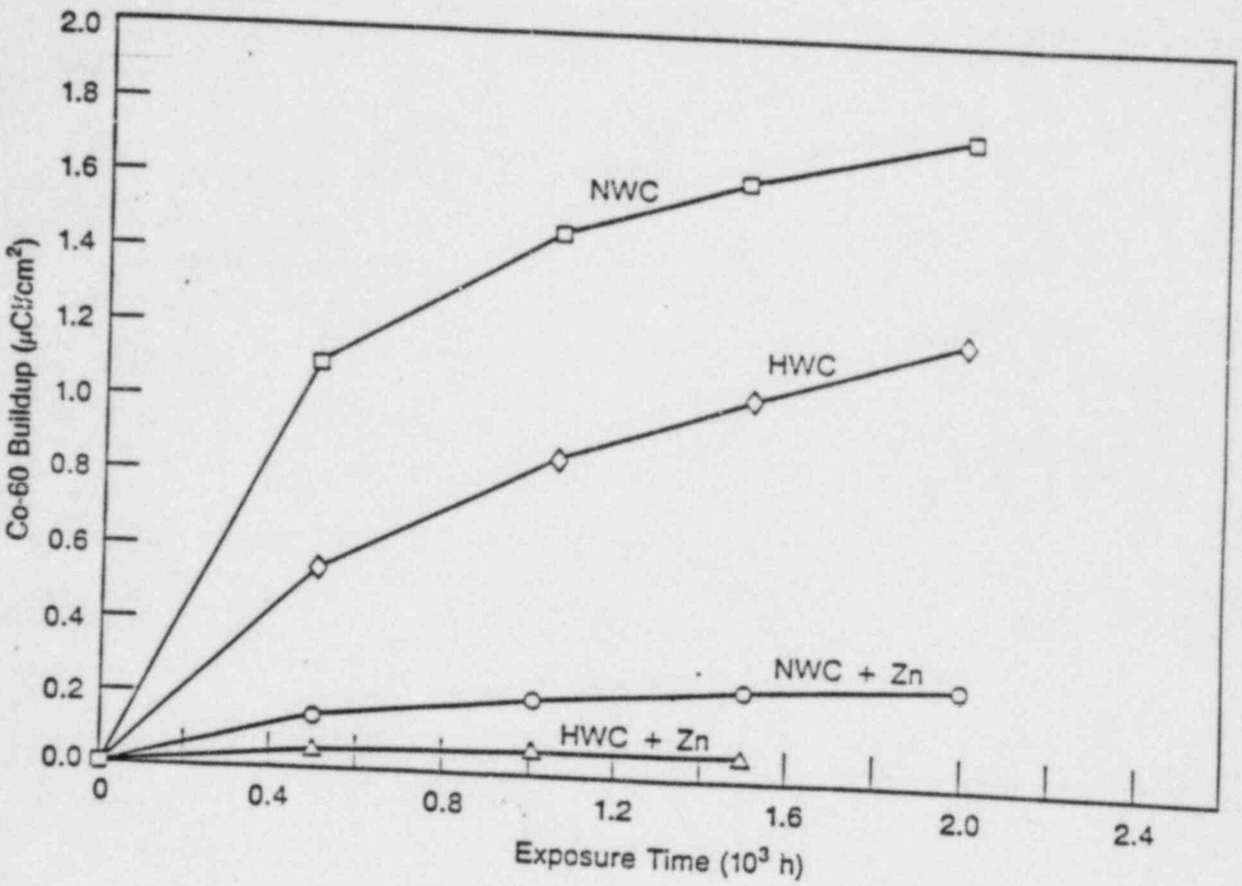
APPROACH: Develop robust configurations for Condensate Polishers. Provide generic cost/benefit evaluations for retrofit options.

COOLANT TECHNOLOGY

TOPIC: PWR Primary Water Chemistry Guidelines

UTILITY NEED: Guidelines have proved a powerful aid in improving plant water chemistry.

APPROACH: Committee of 12 Utility & 3 NSSS vendor Representatives prepared guidelines using format of PWR Secondary and BWR water chemistry guidelines.



The effect of zinc on cobalt deposition on 316 SS in normal water chemistry and hydrogen water chemistry.

EPRI7238

COOLANT TECHNOLOGY

TOPIC:	Surface Modification/Co-60 Reduction
UTILITY NEED:	Reduce Radiation Field buildup on BWR RCS piping and SG Channel Heads.
APPROACH:	Field test electropolishing techniques and demonstrate no adverse effects.

COOLANT TECHNOLOGY

TOPIC:	Prefilming of BWR Piping
UTILITY NEED:	Reduce rapid contamination of new plants & replaced piping in old plants.
APPROACH:	Demonstrate benefit of prefilming to give adherent oxide film on surfaces.

COOLANT TECHNOLOGY

TOPIC:	Cobalt Replacement
UTILITY NEED:	Minimize cobalt release by wear and corrosion to reduce inventory available for activation to Co-60.
APPROACH:	Qualify cobalt-free wear-resistant alloys for use in components and as a substitute for cobalt-base hardfacing alloys.

COOLANT TECHNOLOGY

TOPIC: LOMI Chemical Decontamination Process

UTILITY NEED: Low cost, effective, corrosion-free decontamination method.

APPROACH: Demonstrated no corrosion problems. LOMI modified to reduce radwaste generated.

PWR RADIATION FIELD CONTROL TECHNIQUES

BEFORE POWER ASCENSION:	Electropolish channel heads. Extended hot functional tests with good water quality.
DURING OPERATION:	pH Control. Hydrogen Peroxide addition at shutdown.
REFUELING:	Use Zircaloy grids in replacment fuel.
MAINTENANCE:	Replace wearing valves with cobalt-free alternatives (especially CVCS flow controllers). Valve Maintenance procedures to remove debris.
SPEC. REPAIRS & MAINTENANCE:	Chemical decontamination. Electropolish channel heads. Use low-cobalt for replacement tubing.
PLANT LIFE EXTENSION:	Complete primary system decontamination.

BWR RADIATION FIELD CONTROL TECHNIQUES

BEFORE POWER ASCENSION:	Replace control blade cobalt alloys. Control oxygen at 200–400 ppb during hot functional tests.
DURING OPERATION:	Minimize feedwater iron input. Maintain low reactor water conductivity Zinc Injection Process.
REFUELING:	Low cobalt materials in replacement fuel. Cobalt–free control blades.
MAINTENANCE:	Replace wearing valves with cobalt–free alternatives (especially feedwater flow controllers). Valve Maintenance procedures to remove debris.
SPEC. REPAIRS & MAINTENANCE:	Chemical decontamination. Electropolish replacement piping. Air oxidation or water prefilming of pipe.
PLANT LIFE EXTENSION:	Complete Reactor Coolant System decontamination.

M. E. A. C.

WORKSHOPS/SEMINARS/MEETINGS

* Instrument Improvement – March
* Computer Modeling, Data Management, and Video Applications – June
* Dose Reduction and Personnel Factors – September
* Bolting and Bolted Connections – November

M. E. A. C.

TECHNOLOGY TRANSFER

* Improved BWR CRD Handling System
  Design Review/Bid Evaluation
* Underwater Inspection Vehicle
  Demonstrated at Catawba
  Four Utilities Interested
* Surveyor Mobile Robot
  Lab tests/Field Hardening
  Demonstrated at ICUEF Show

M. E. A. C.

TECHNOLOGY TRANSFER(Cont.)

* Surbot Mobile Robot
  Simulated–Plant Evaluation
* Training Courseware
  3 Video Tapes – Pressure
  Boundary Bolting
  Available from MEAC

HUMAN ENGINEERING: MAINTAINABILITY GUIDELINES

UTILITY NEED

Incorporate improved maintenance as a plant design goal

SOLUTION

Develop human engineering guidelines for designing new or backfitting existing plants

STATUS

Guidelines available (NP-4350)

HEAT STRESS MANAGEMENT PROGRAM
GUIDELINES

UTILITY NEED

Worker productivity, health & safety challenged by conditions leading to heat stress

SOLUTION

Develop comprehensive guidelines for managing heat stress

STATUS

Guideline available (NP-4453)

DIAGNOSTIC/TROUBLESHOOTING SKILLS
SOFTWARE

UTILITY NEED

Improve the diagnostic and troubleshooting skills of plant personnel

SOLUTION

Develop and evaluate prototype program for training these skills

STATUS

Diesel generator courseware tested

PERFORMANCE DATA ACQUISITION GUIDELINES

UTILITY NEED

Insufficient data to locate chronic "lost megawatts"

SOLUTION

Develop guidelines for a structured approach to heat rate increase diagnosis

STATUS

Guidelines available (NP-3915)

MOBILE SURVEILLANCE ROBOT

COMMERCIAL HARDWARE

UTILITY NEED

Reduce human exposure to hostile environments for radiation, visual, and environmental surveying

SOLUTION

Develop untethered, remotely operated robotic system

STATUS

SURVEYOR commercially available

V. M. O. IMPROVEMENTS

Commercial Hardware

UTILITY NEED

Failure of motor operated valves is a major cause of abnormal events

SOLUTION

Develop microprocessor control and diagnostic system

STATUS

Prototype tested. Field test ongoing

VALVE PACKING IMPROVEMENTS

Commercial Hardware

UTILITY NEED

Packing leaks increase maintenance costs and radiation levels

SOLUTION

Develop improved graphite packing and guidelines for "live loading"

STATUS

Prototype tested. Field test ongoing

FIELD HARDENED INSTRUMENTS

Guidelines & Commercial Hardware

UTILITY NEED

Instrument & electrical events cause plant trips & 30% of LERs

SOLUTION

Develop Guidelines

Demonstrate with hardware

STATUS

Guidelines developed. Module tested

MACHINERY VIBRATION MONITORING GUIDELINES

UTILITY NEED

Assistance in evaluating plant monitoring program alternatives

SOLUTION

Develop Guidelines giving technical and skills requirements and corporate goals

STATUS

Interim Guidelines (NP-4346)

ENGINEERING & OPERATIONS DEPT.

CURRENT R&D EMPHASIS

- PLANT LIFE EXTENSION
- SURFACE ELECTROPOLISHING & PASSIVATION
- DECONTAMINATION
- ROBOTICS DEVICES
- RADIATION CONTROL
- MAINTAINABILITY GUIDELINES
- EQUIPMENT QUALIFICATION
- COMPUTER GENERATED DISPLAYS
- COMPUTER AIDED CONSTRUCTION

SAFETY TECHNOLOGY DEPARTMENT

W. B. LOEWENSTEIN

RISK ASSESSMENT	I. WALL
SOURCE TERM	R. VOGEL
ANALYTICAL METHODS & VERIFICATION	R. BREEN
SAFETY CONTROL & TESTING	R. DUFFEY

WBL:AV/3932ST6B

PROGRAM 1310

RISK ASSESSMENT

I. WALL

PROBABILISTIC ANALYSIS & APPLICATION	D. WORLEDGE
PLANT & GEOTECHNICAL ENGINEERING	R. KASSAWARA
CIVIL ENGINEERING	H. TANG
SEISMOLOGY	C. STEPP

WBL:AV/3932ST6B

PROGRAM 1315

SOURCE TERM

R. VOGEL

AEROSOL TECHNOLOGY LARGE SCALE EXPERIMENTS	F. RAHN
FISSION PRODUCT BEHAVIOR	R. RITZMAN
TMI-2 DATA USE	G. THOMAS
CODE DEVELOPMENT, BWR POWER LEVEL	R. SEHGAL
IDCOR COORDINATION	M. LEVERETT

WBL:AV/3932ST6B

PROGRAM 1320

ANALYTICAL METHODS & VERIFICATION

R. BREEN

REACTOR PERFORMANCE	O. OZER
THERMAL PERFORMANCE	L. AGEE
METHODS SUPPORT & ENHANCEMENT	R. BREEN

WBL:AV/3932ST6B

PROGRAM 1330

SAFETY CONTROL & TESTING

R. DUFFEY

SAFETY CONTROL SYSTEMS	B. SUN
SAFETY MARGINS & TESTING	J.-P. SURSOCK

WBL:AV/3932ST6B

SAFETY TECHNOLOGY OVERVIEW

SAFETY RESEARCH IN TRANSITION

FROM: ACCIDENT DESCRIPTION

TO: ACCIDENT PREVENTION &
ACCIDENT ACCOMODATION

VIA: CONSOLIDATION & CONVERGENCE
OF RESULTS

WBL:AV/3999ST6D

SAFETY TECHNOLOGY PRIORITIES

BASE PROGRAM

SOURCE TERM

SEISMIC

SAFETY CONTROL

TESTING TO SUPPORT ANALYSIS

DISCIPLINED SOFTWARE

WBL:AV/3999ST6D

SAFETY TECHNOLOGY OVERVIEW
HIGHLIGHTS FOR 1985 (I)

SOURCE TERM

- MAJOR LARGE TEST PROGRAMS WINDING DOWN
- EXPLOITATION & CONSOLIDATION OF RESULTS
- IDCOR DIALOGUE ESTABLISHED
- IDCOR METHODOLOGY VERIFICATION
- OVERALL INTEGRATIOON IS NEW FOCUS

SEISMIC

- SEISMICITY OWNERS GROUP PROCEEDING WELL
- SEISMIC MARGINS IN HAND
- LOTUNG TEST READY
- PIPING RESPONSE WORK DEPLOYED

SAFETY TECHNOLOGY OVERVIEW

HIGHLIGHTS FOR 1985 (II)

SAFETY CONTROL & TESTING

- MIST PROJECT PROCEEDING WELL
- SIMULATOR QUALIFICATION GROWTH
- AI SPECULATIONS PROMISING
- OPERATION AIDS BEING TESTED & DEPLOYED
- DIGITAL FEEDWATER CONTROL ON SITE
- SGTR CLOSURE PROMISED

PRA

- APPLICATIONS FOR RELIABILITY WITH GROWING INTEREST
- COMMON MODE STUDY PROMISING
- HUMAN RELIABILITY MODELING PROGRESS

SOFTWARE

- MMS COMMERCIALIZATION SUCCESSFUL
- VIPRE-SER BEING DEVELOPED
- RETRAN CONSORTIUM INITIATED
- ATHOS/PORTHOS VERY SUCCESSFUL
- RASP EVOLUTION CONTINUING
- TECHNOLOGY TRANSFER IN CORE MONITORING

SAFETY TECHNOLOGY OVERVIEW

EMPHASIS FOR 1986 IS ON

DEPLOYMENT & CLOSURE

SOURCE TERM

- CONSOLIDATION & CLOSURE
- PROBABILITICS FOCUS FOR CLOSURE

SEISMIC

- ESSENTIAL COMPLETION OF OWNERS' GROUP SCOPE
- METHODOLOGY FOR MARGINS DEVELOPED
- CODES & STANDARDS WORK ESSENTIAL

PRA

- DEPLOY APPLICATIONS FOR RELIABILITY EVALUATIONS

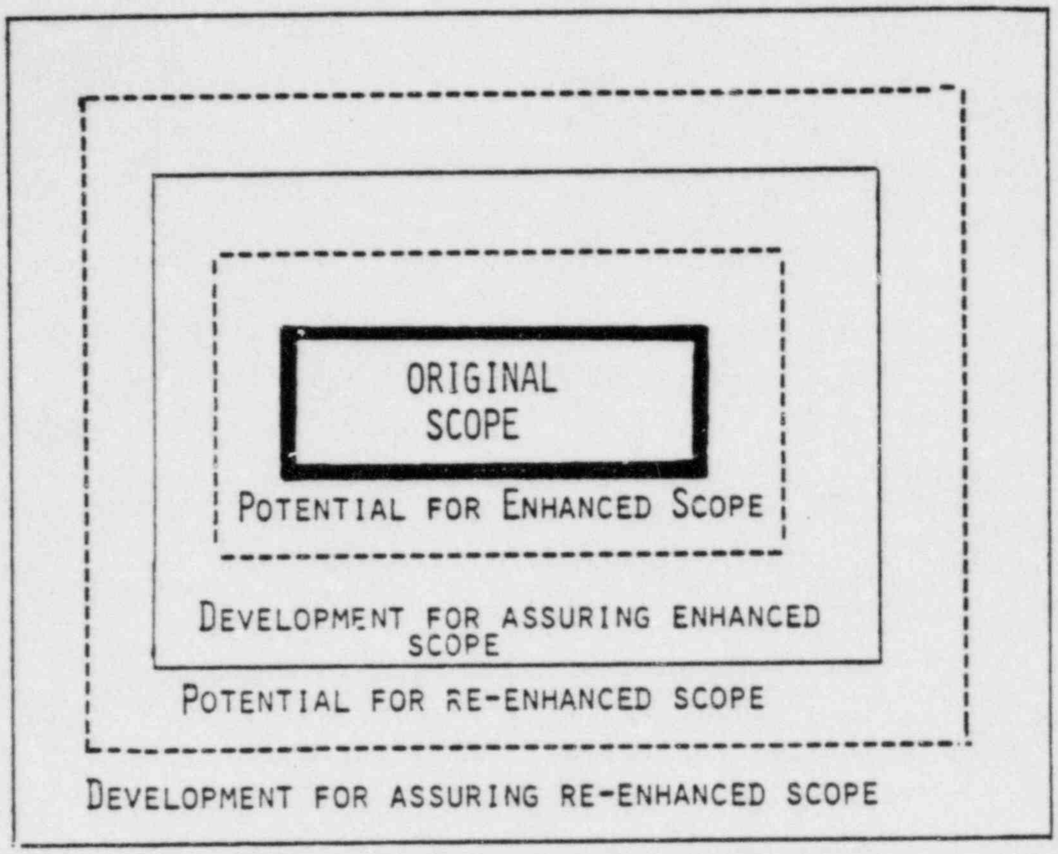
SOFTWARE

- DE-EMPHASIZE ENHANCEMENT & MAINTENANCE
- EMPHASIZE a) COMMERCIALIZATION AND/OR
  b) USER SUPPORTED ACTIVITIES

SAFETY CONTROL & TESTING

- EMPHASIZE DEPLOYMENT & USE OF PRODUCTS
- EXPLORE DIGITAL TECHNOLOGY FOR INTEGRATED SYSTEMS
- MIST SAFETY DEFINITION

COMPLETION/CLOSURE



SAFETY TECHNOLOGY OVERVIEW
1986 THRUSTS

DEPLOYMENT OF RESEARCH RESULTS

- SOURCE TERM ACTION

- SEISMICITY RESOLUTION

- CODES & STANDARDS ON STRUCTURAL RESPONSE

- SOFTWARE COMMERCIALIZATION OR USER SUPPORTED SOFTWARE ACTIVITIES

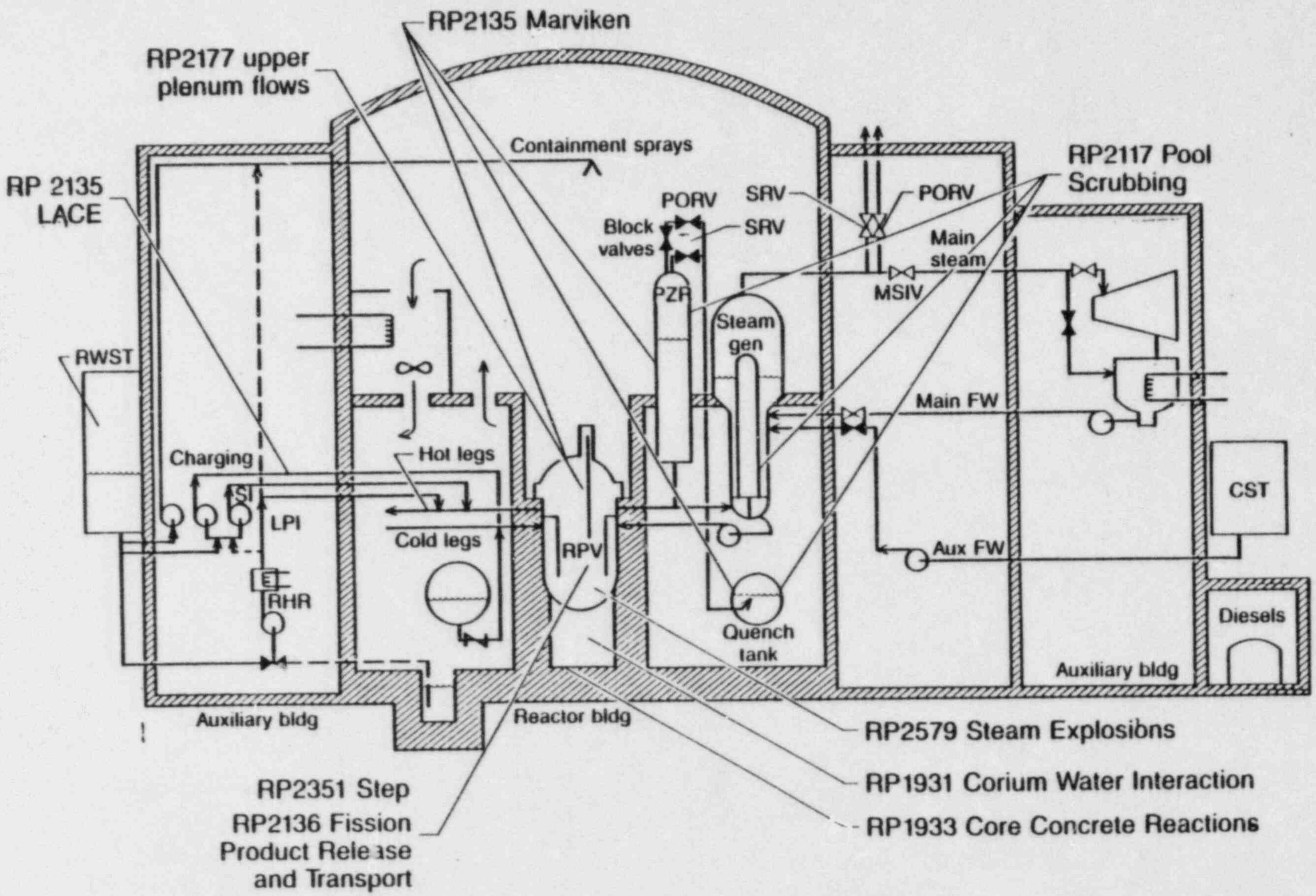
- INTEGRATION OF PRA ON MAJOR ISSUES (E.G. SOURCE TERM)

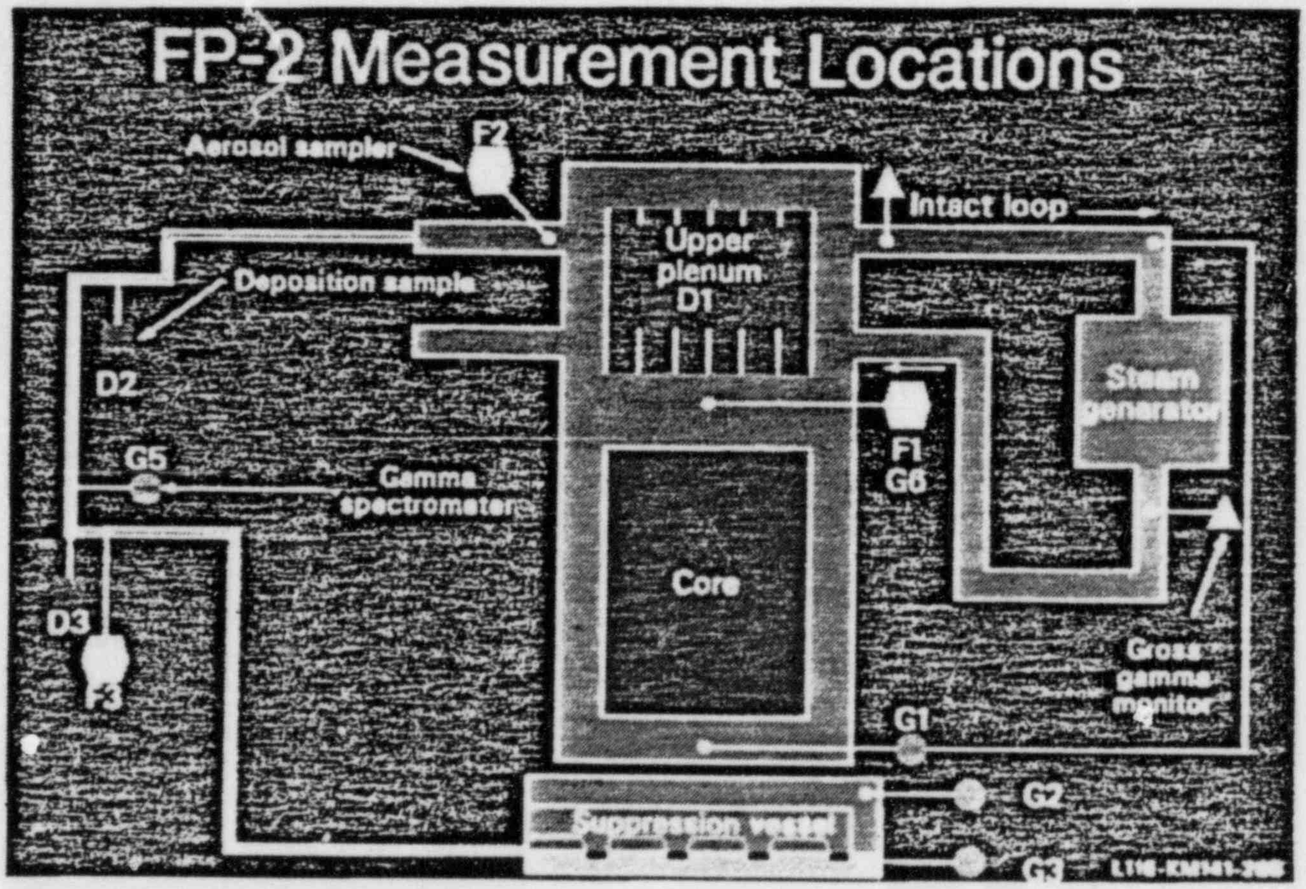
- DIGITAL SYSTEMS IMPLEMENTATION

- CLOSURE ON B&W SAFETY TESTING

WBL:AV/3999ST6D

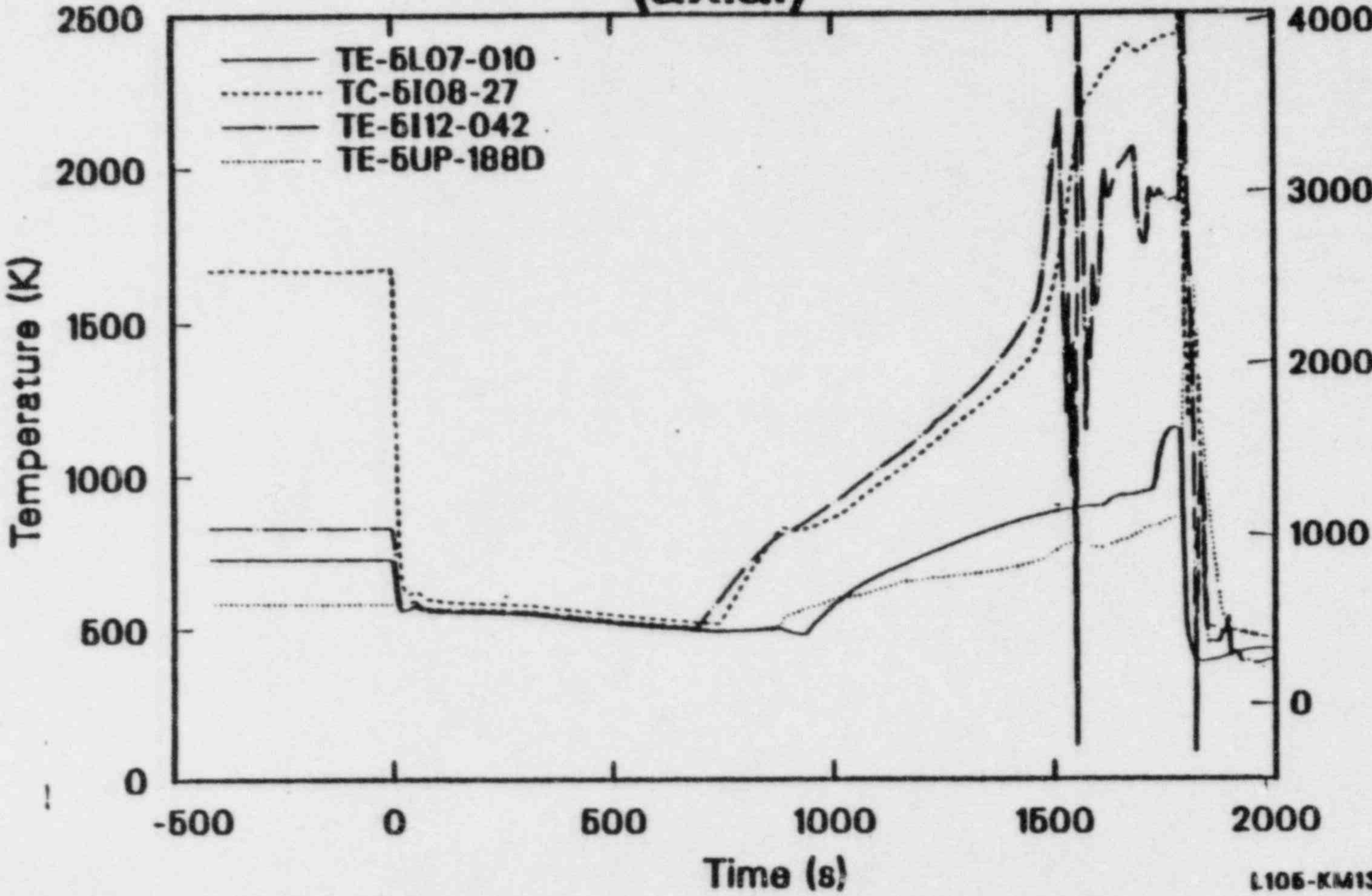
SOURCE TERM RESEARCH FOR A TYPICAL PWR



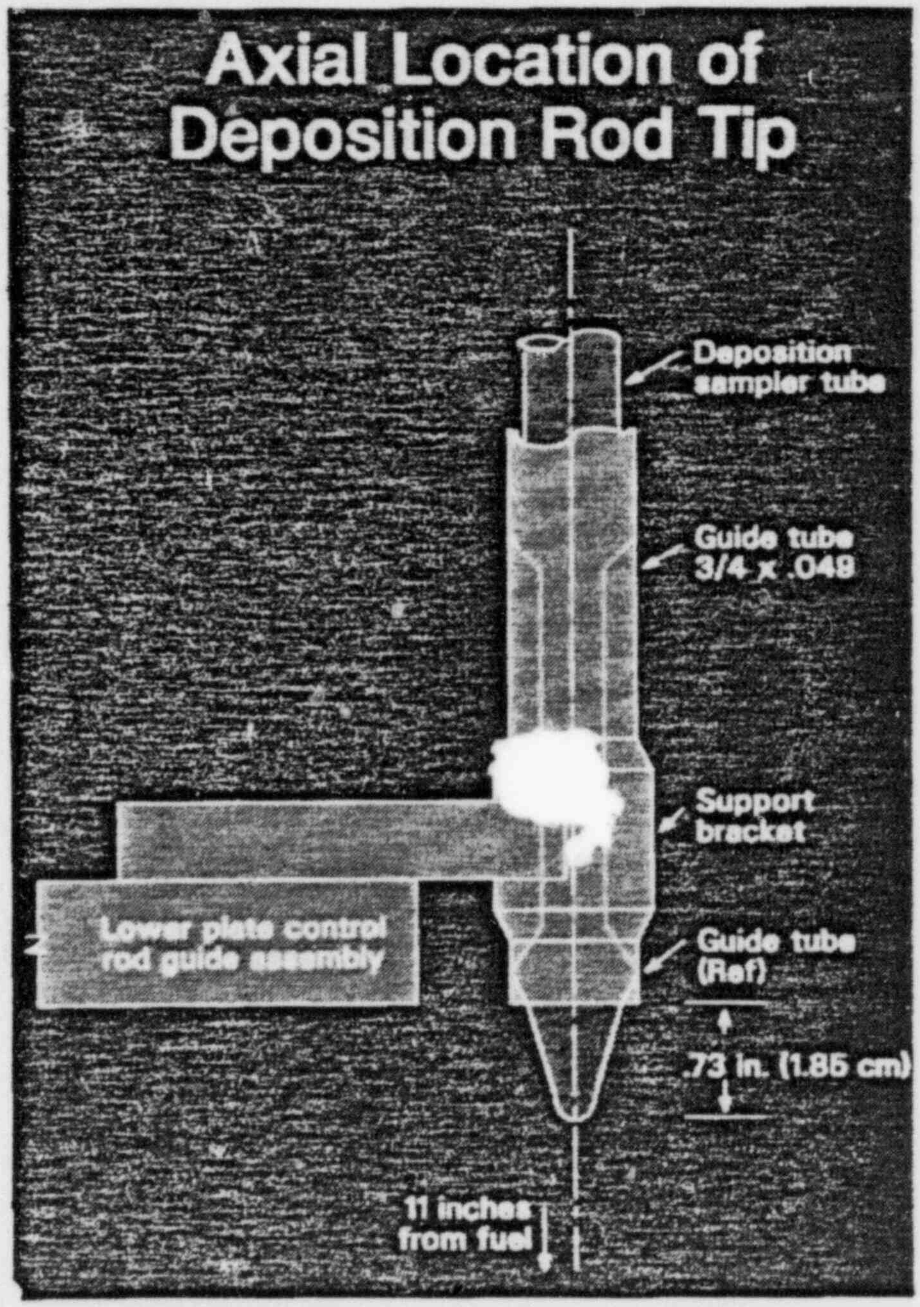


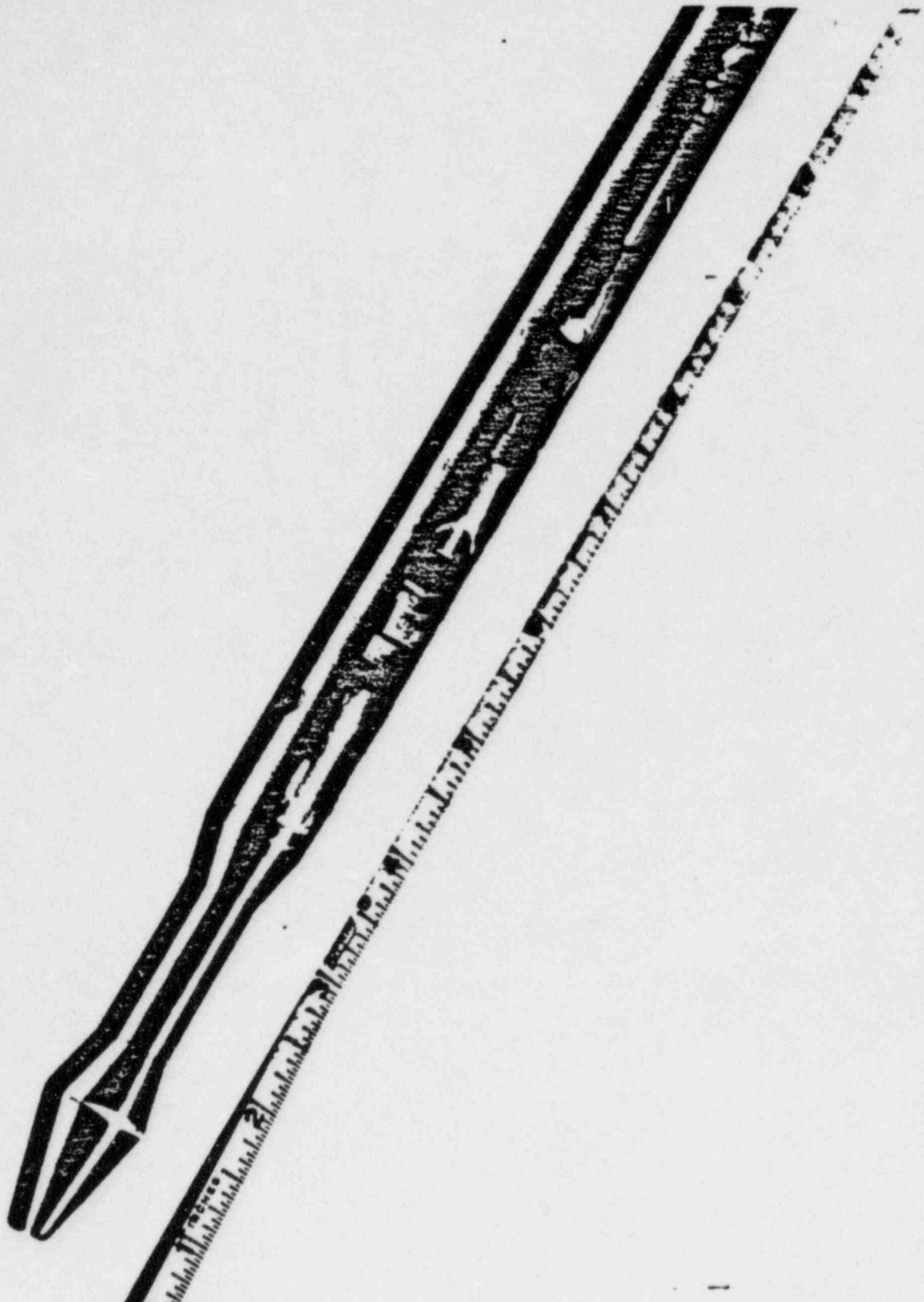
FP-2 Measurement Locations
Aerosol sampler
F2
Intact loop
Upper plenum D1
Deposition sample
D2
Steam generator
F1
G6
G5
Gamma spectrometer
Core
D3
F3
Gross gamma monitor
G1
G2
Suppression vessel
G3

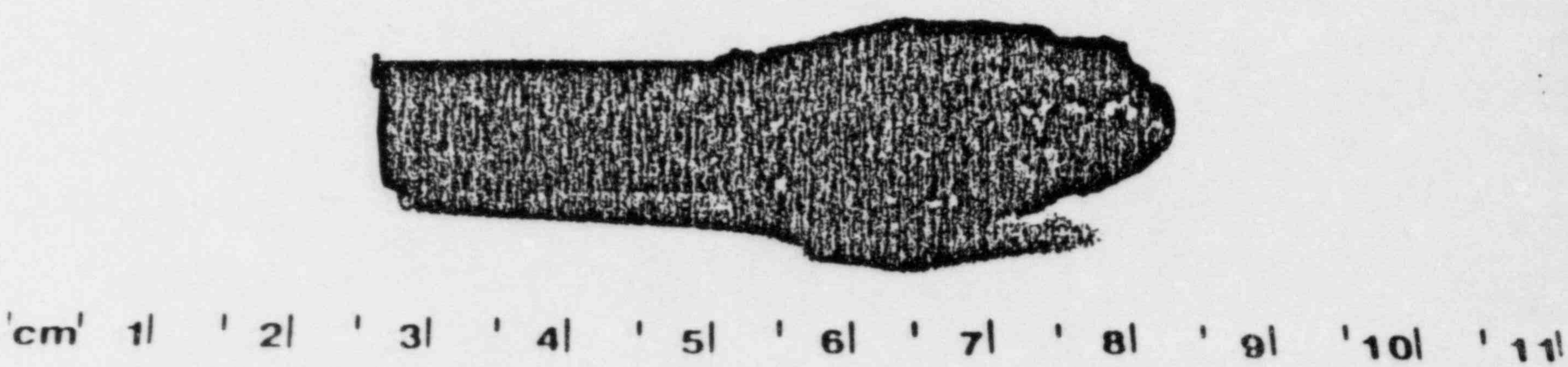
LP-FP-2 Center Module Temperatures (axial)

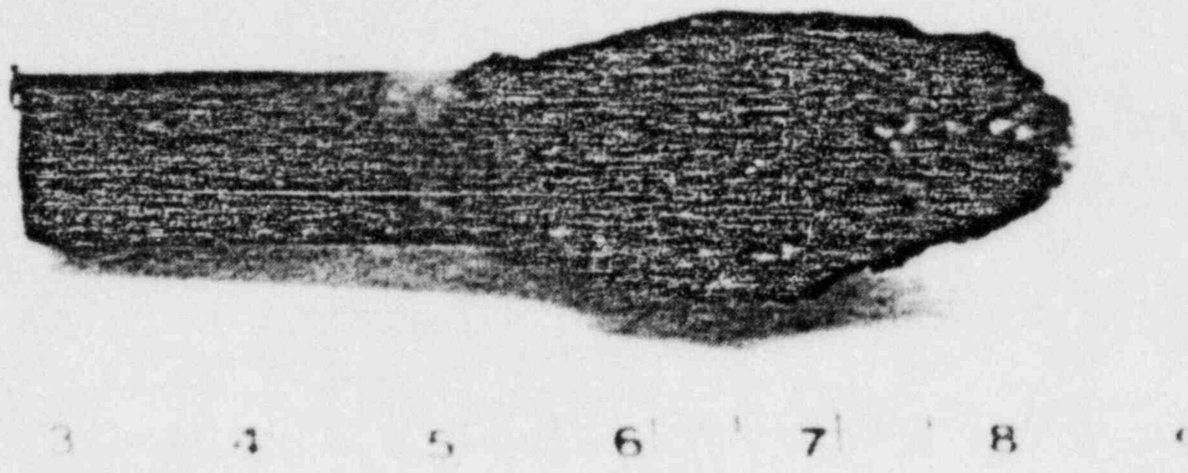


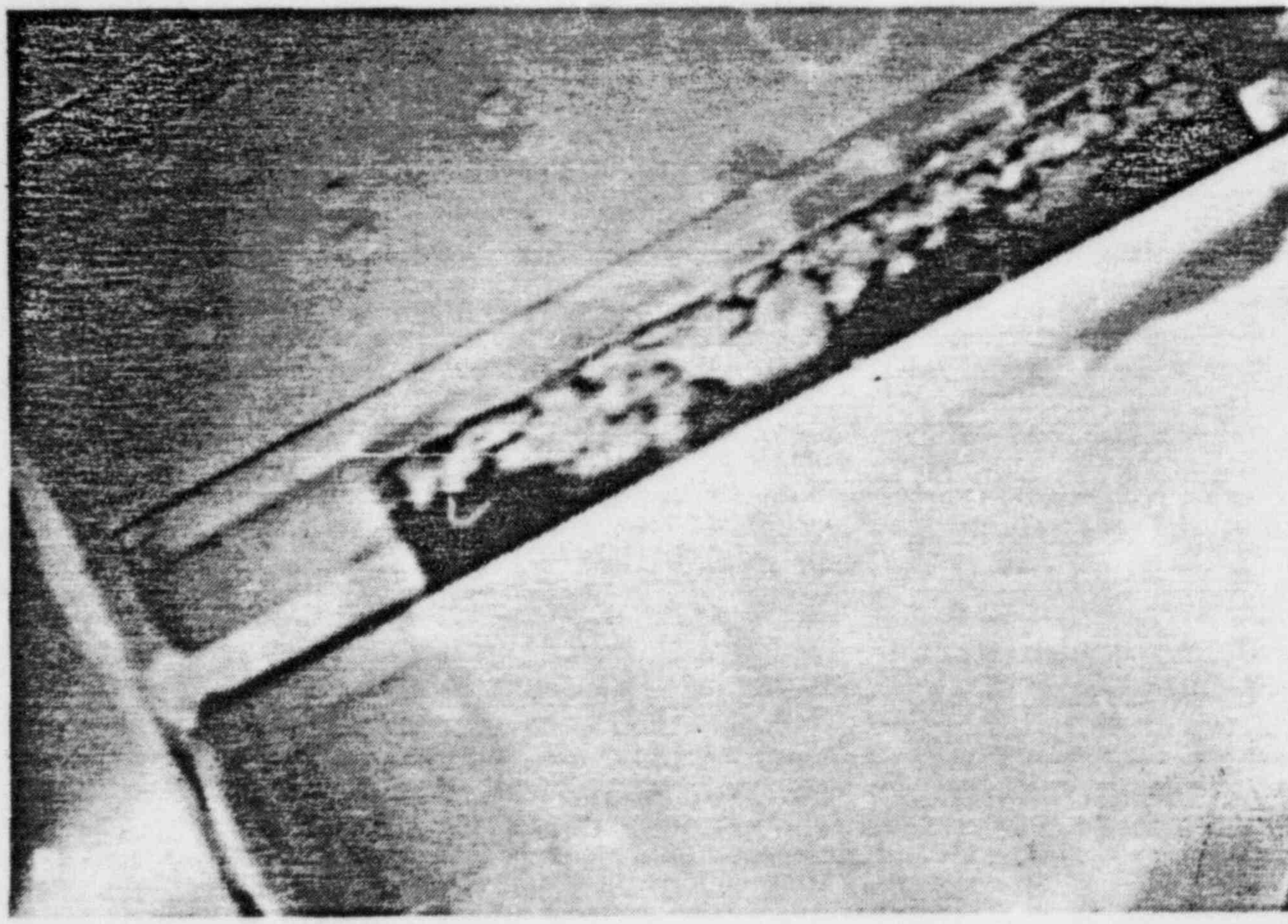
L106-KM130-08M

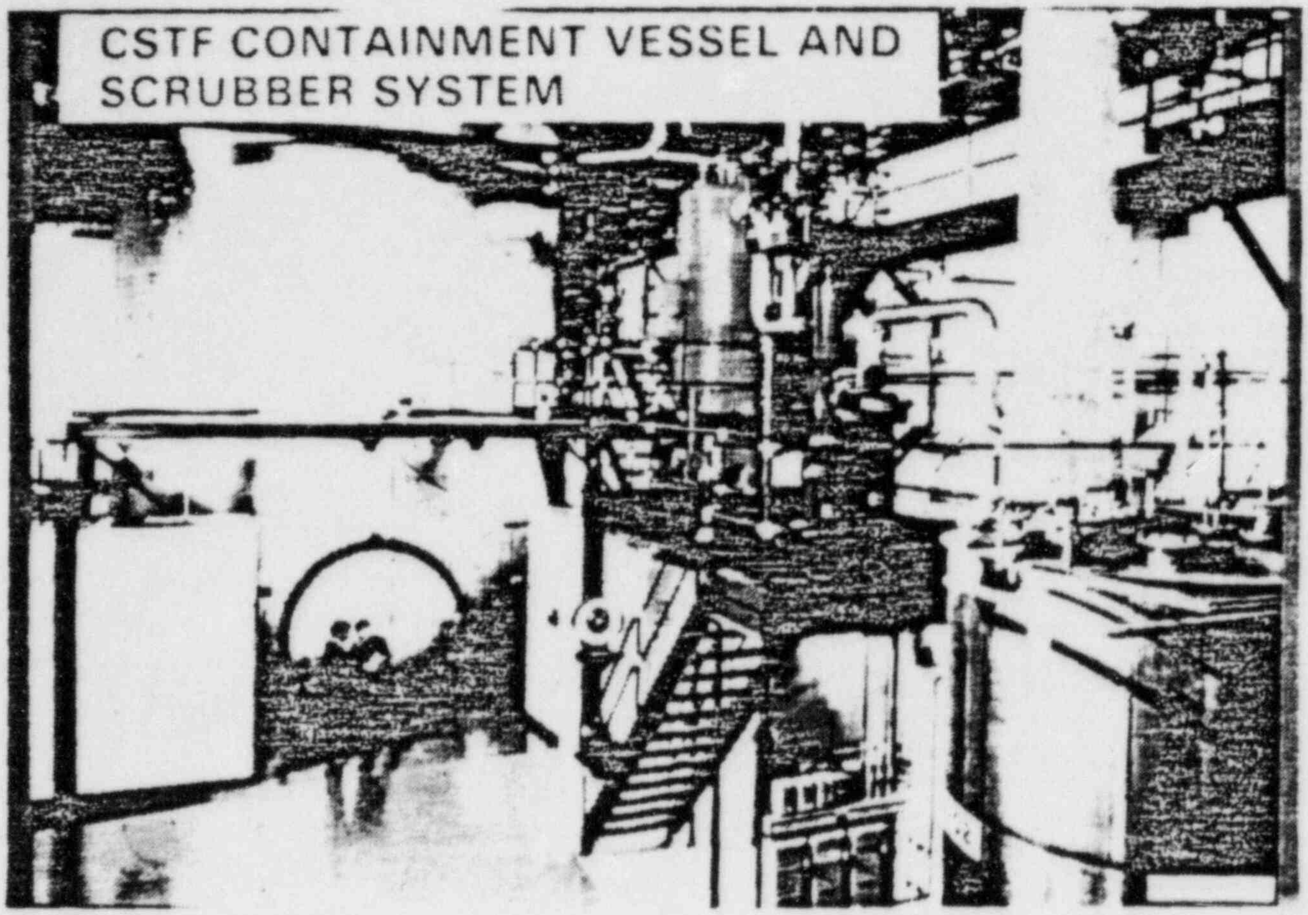

Axial Location of Deposition Rod Tip
Deposition sampler tube
Guide tube 3/4 x .049
Support bracket
Lower plate control rod guide assembly
Guide tube (Ref)
.73 in. (1.85 cm)
11 inches from fuel

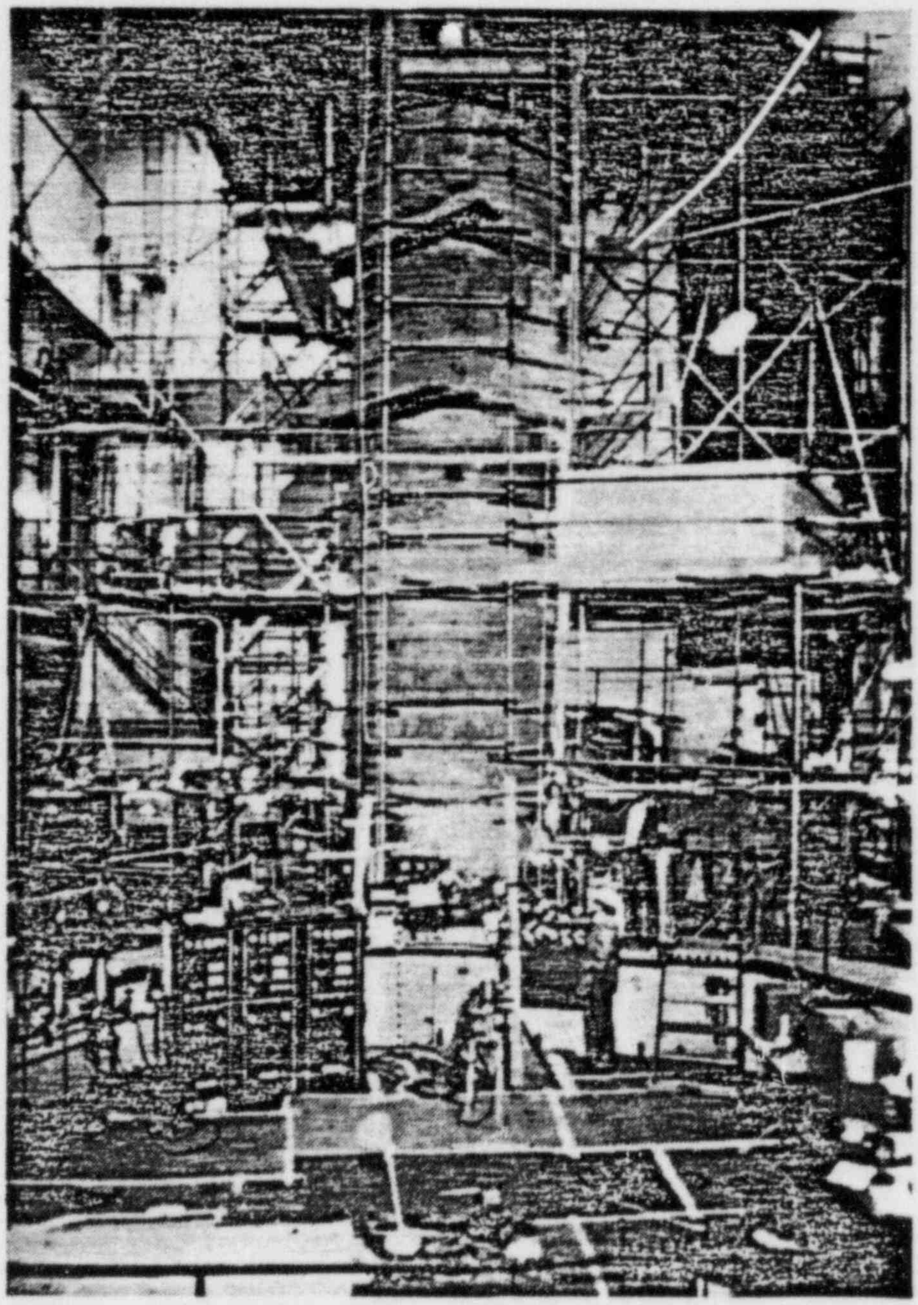


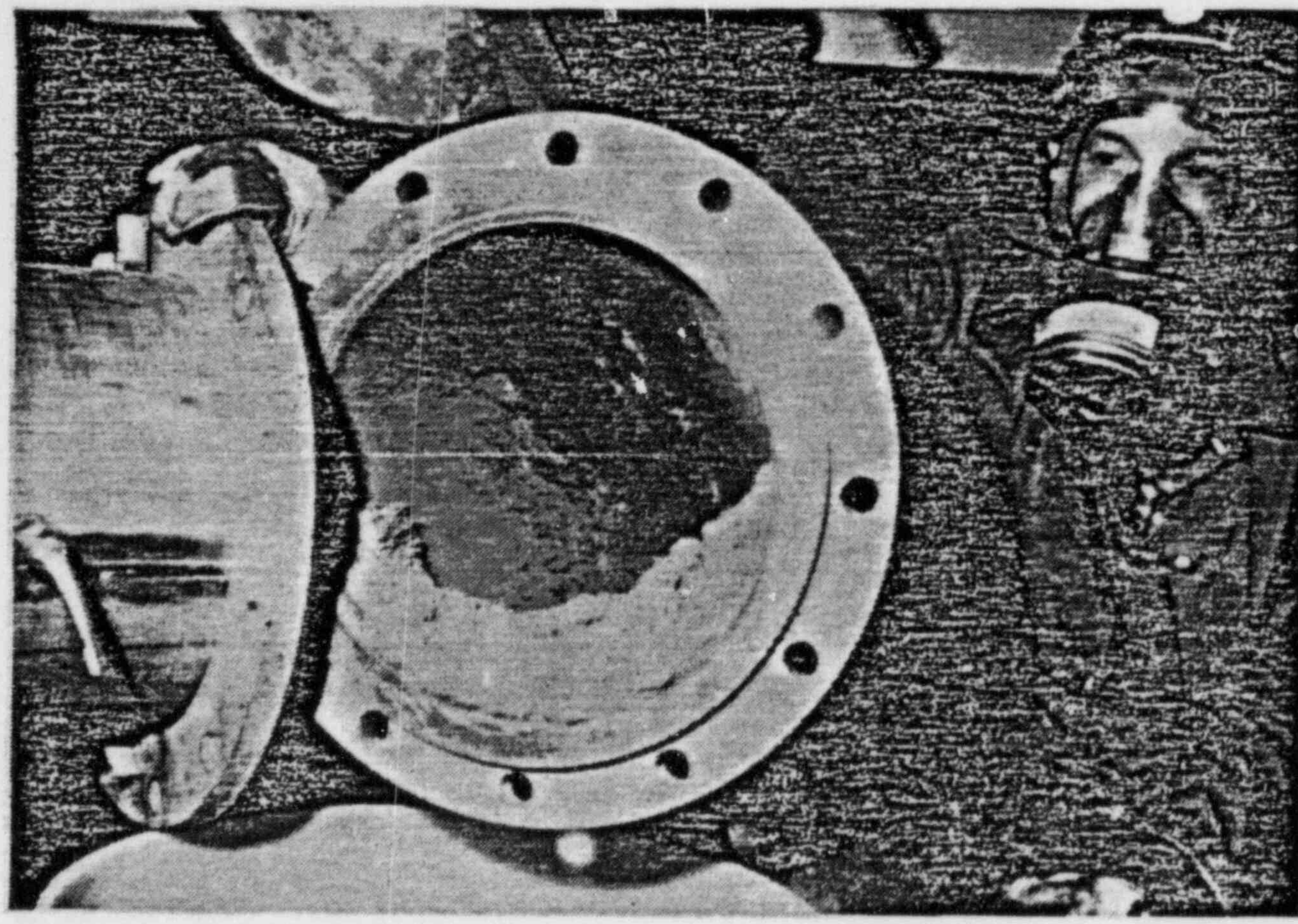

cm 1 2 3 4 5 6 7 8 9 10 11

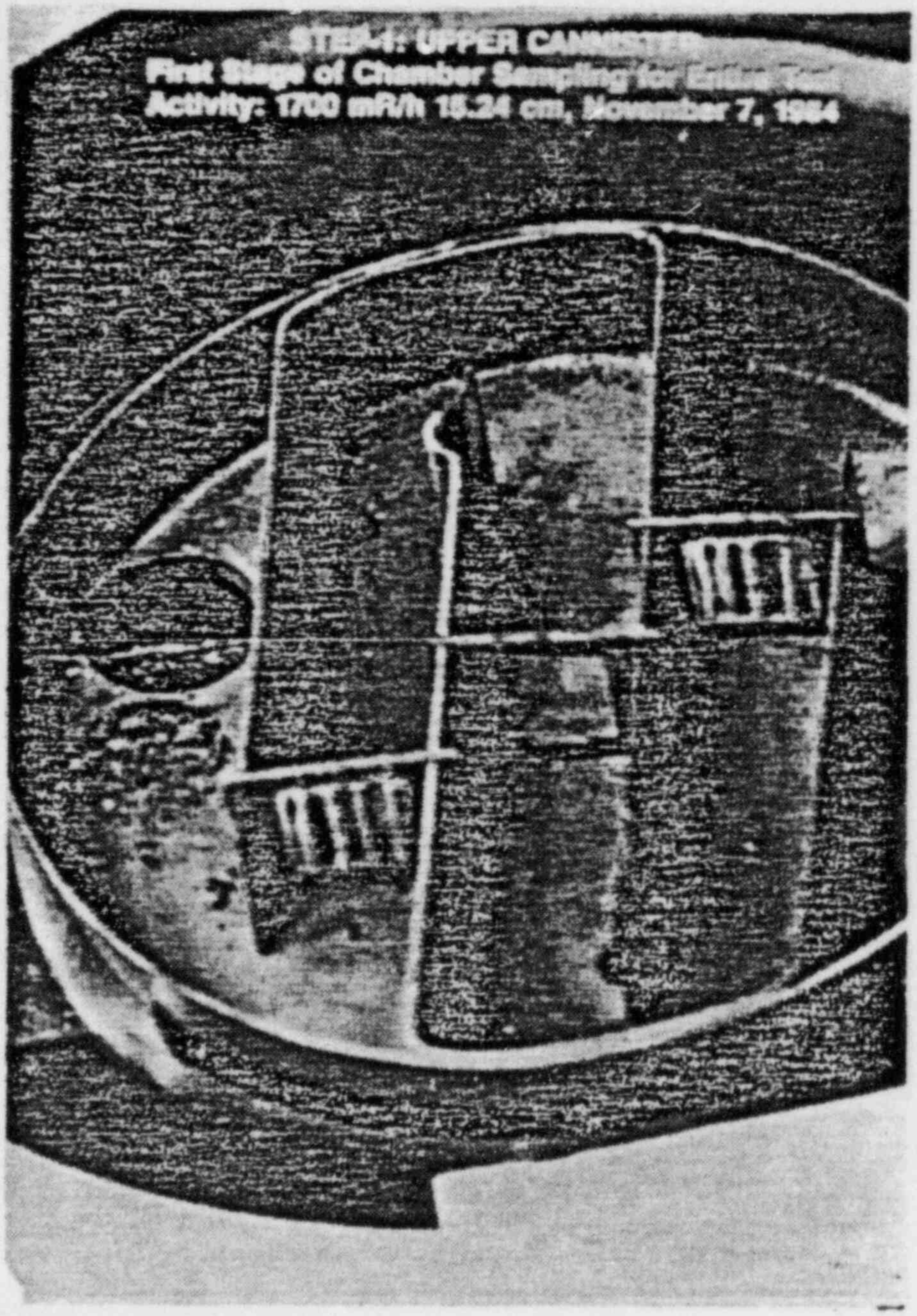

3 4 5 6 7 8




CSTF CONTAINMENT VESSEL AND
SCRUBBER SYSTEM



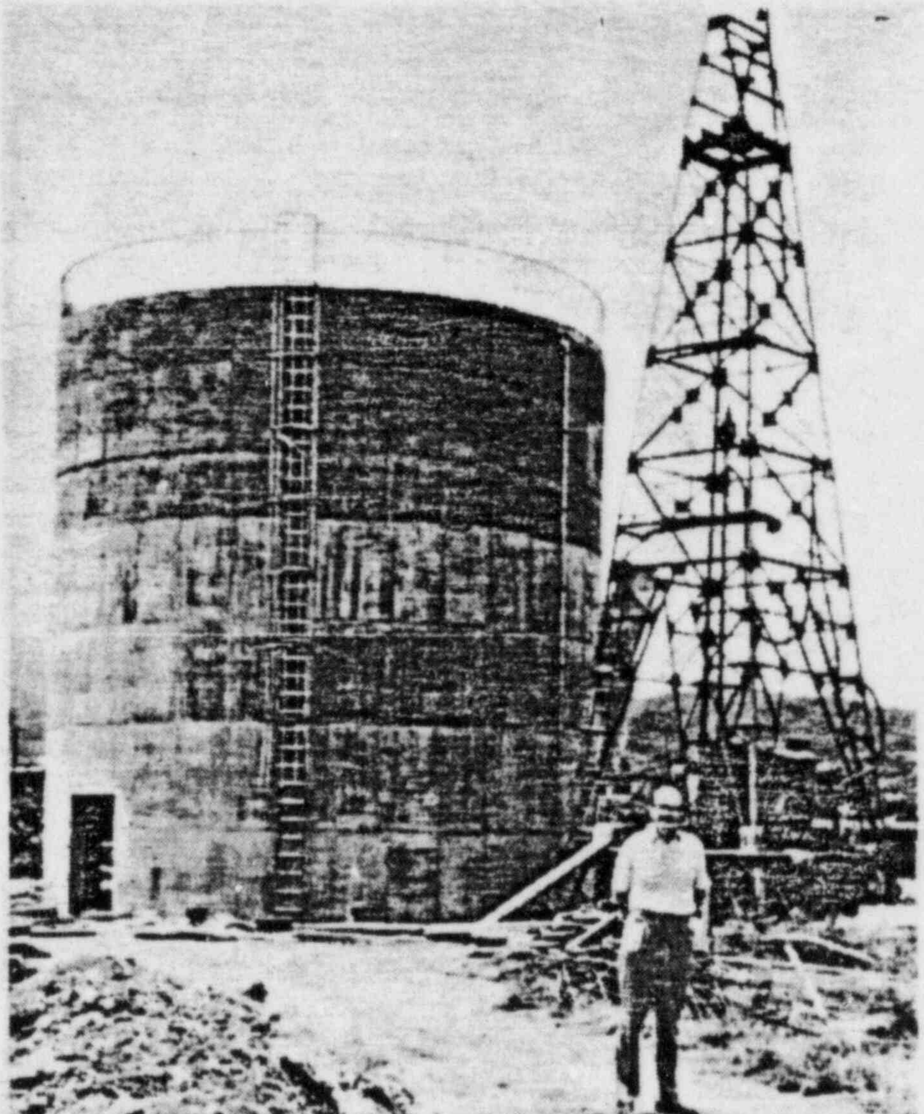


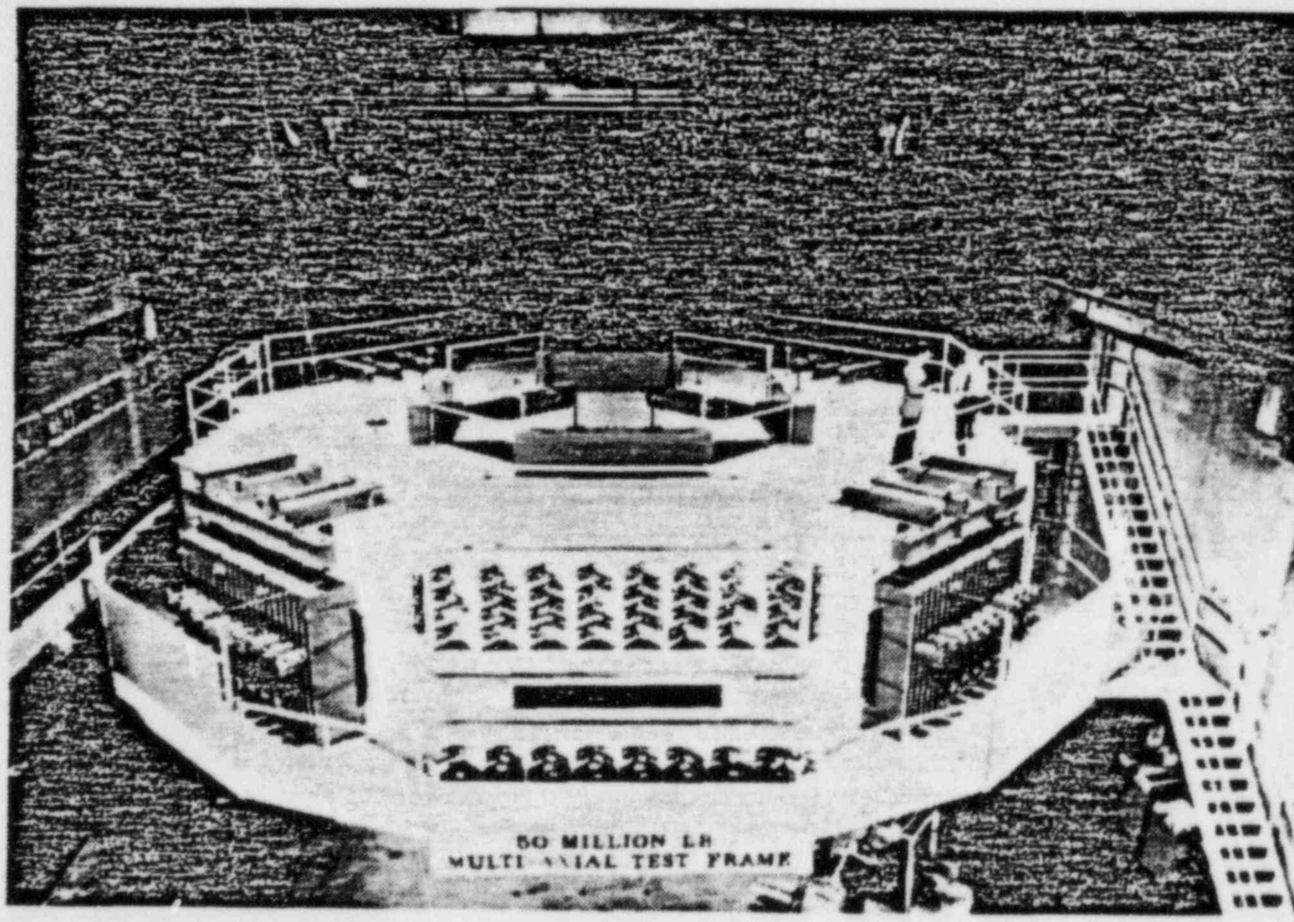

STEP-1: UPPER CANNISTER
First Stage of Chamber Sampling for
Activity: 1700 mR/h 15.24 cm, November 7, 1984

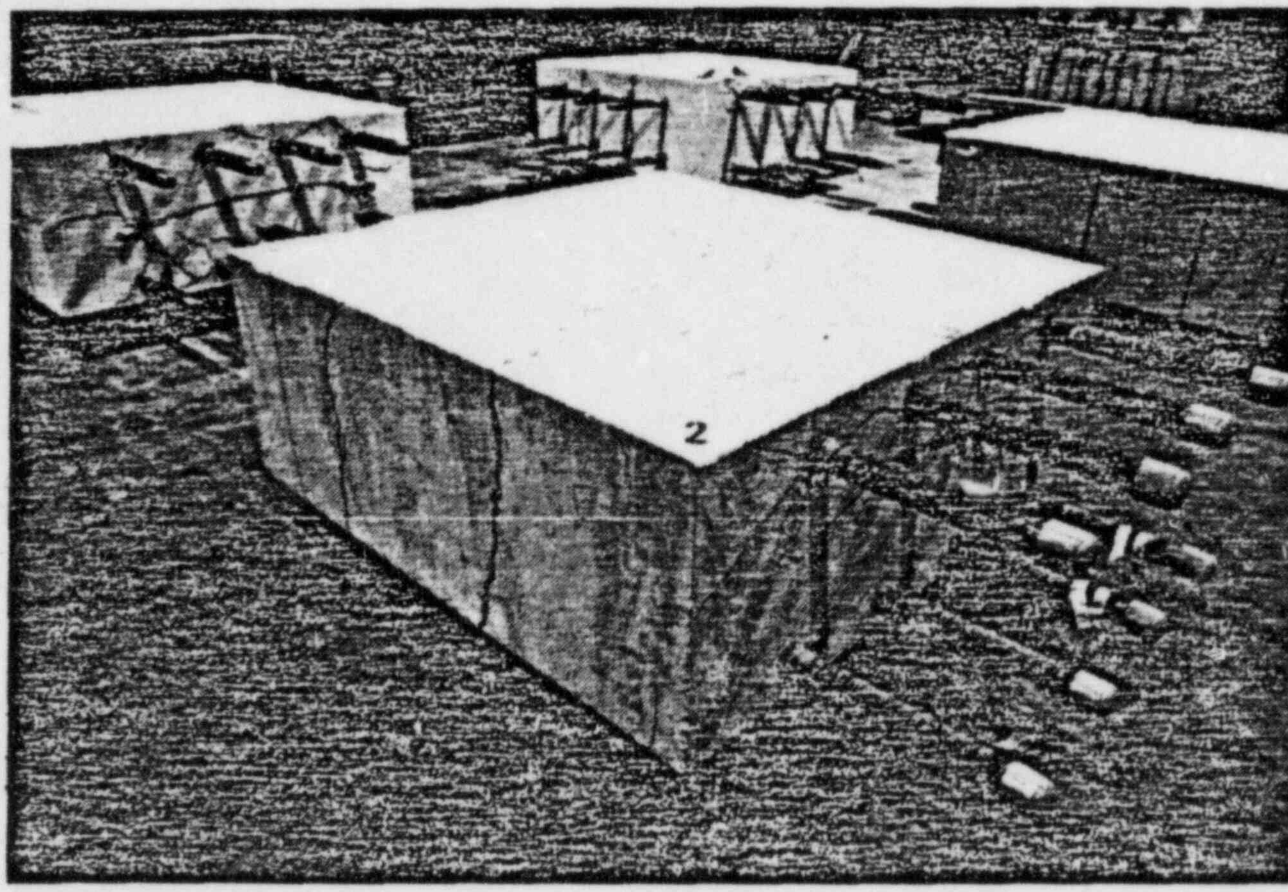
RAFT CODE VERSUS EXPERIMENTS

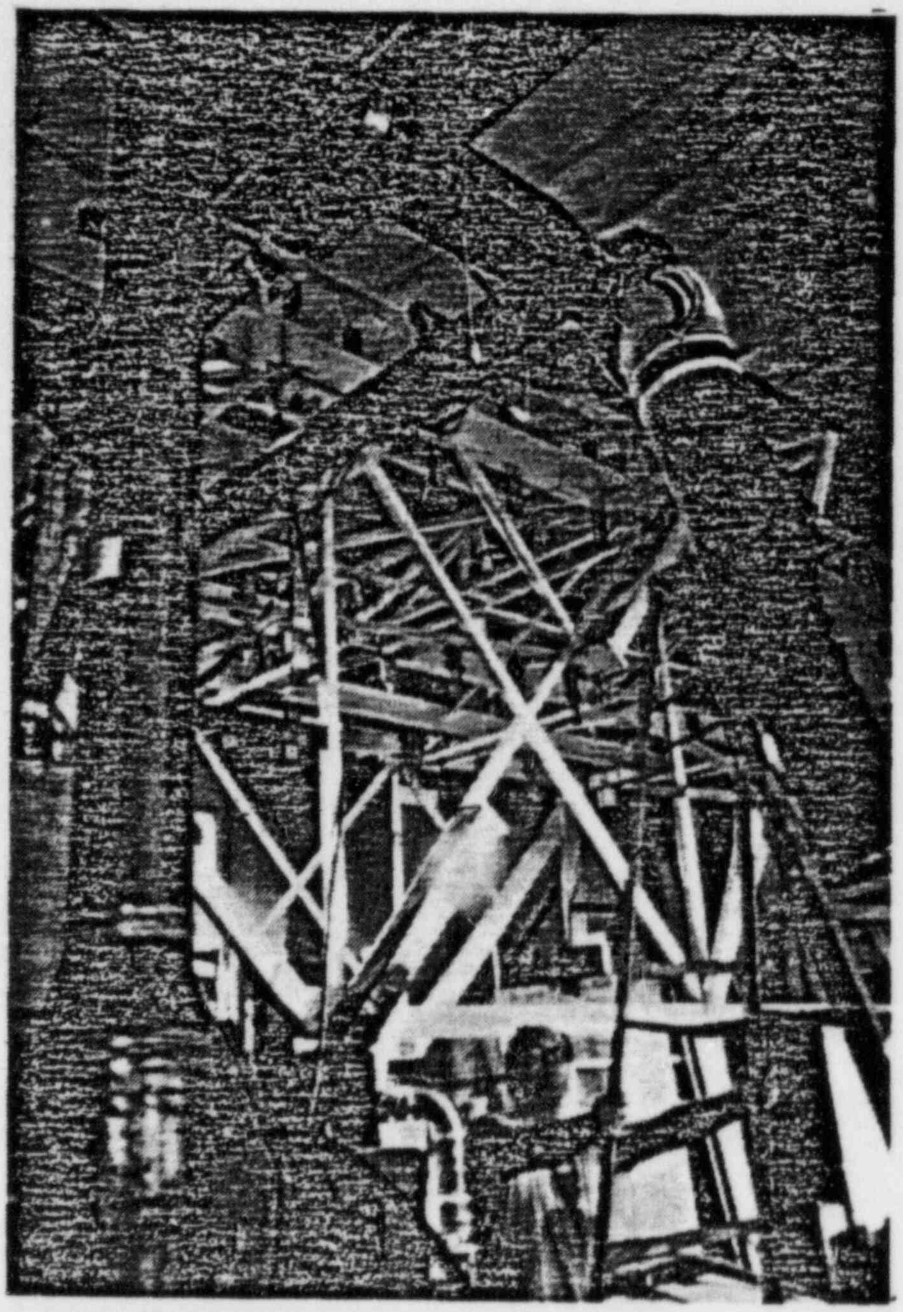
AEROSOL TEST	TEST NUMBER	AEROSOL PARTICLE SIZE (µM)		TOTAL MASS DEPOSITED (KG)	
		CODE	MEASUREMENT	CALCULATION	MEASUREMENT
MARVIKEN	2A	- 5	- 6	8.4	3.7
	1	- 7	- 6	14.5	18.0
	2B	- 9	-	18.1	37.1
	7	-11	-	31.3	22.5
				% DEPOSITION	
				CALCULATION	MEASUREMENT
ANL HOT TUBE	2	-0.2	0.2-1	20	18
	4	-0.8	0.1-1	11	13
				AEROSOL CONCENTRATION (PART/CM^3)	
				CALCULATION	MEASUREMENT
STEP	1	-0.5	-0.2	10^7	10^6-10^7

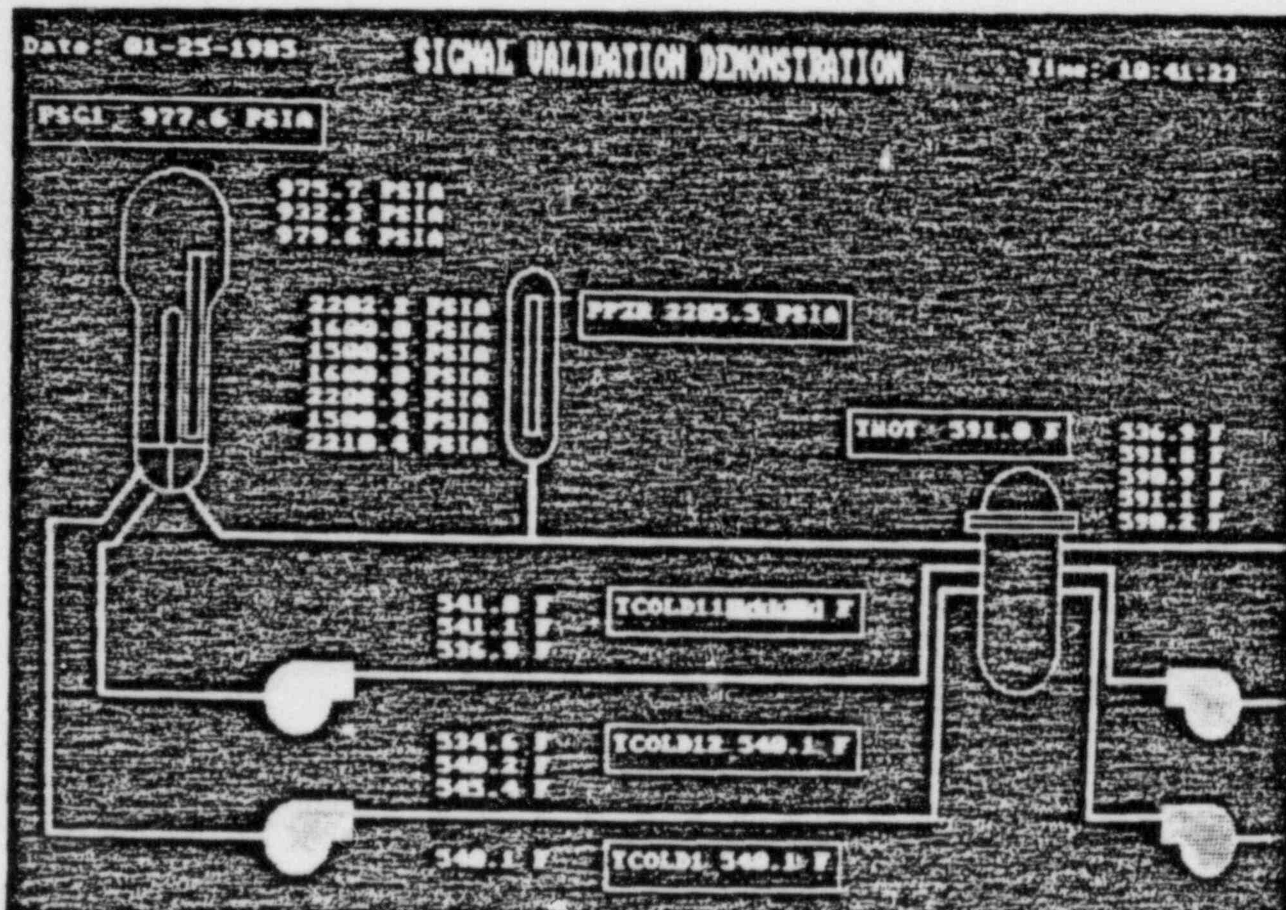
RLR3997ST5A

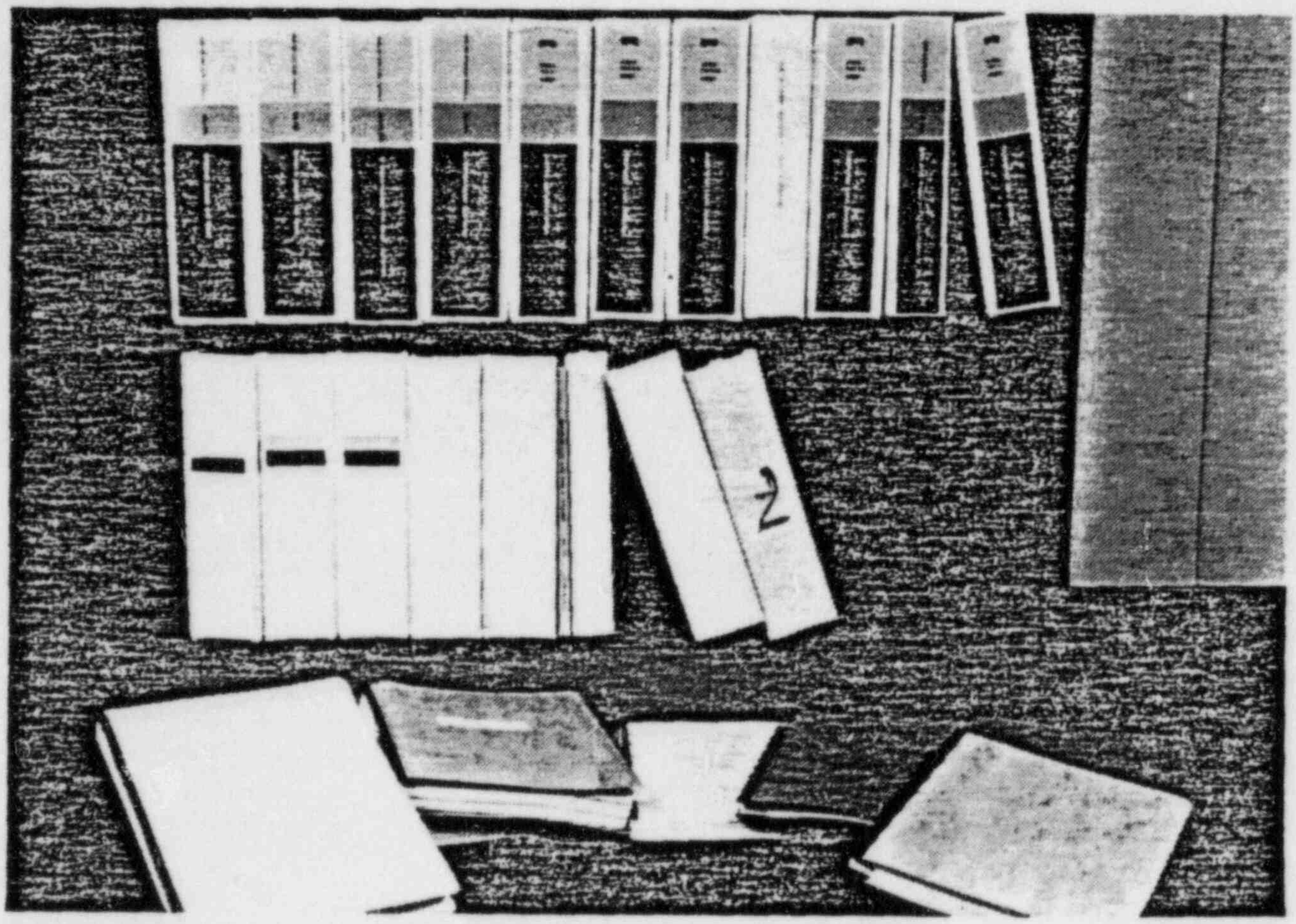


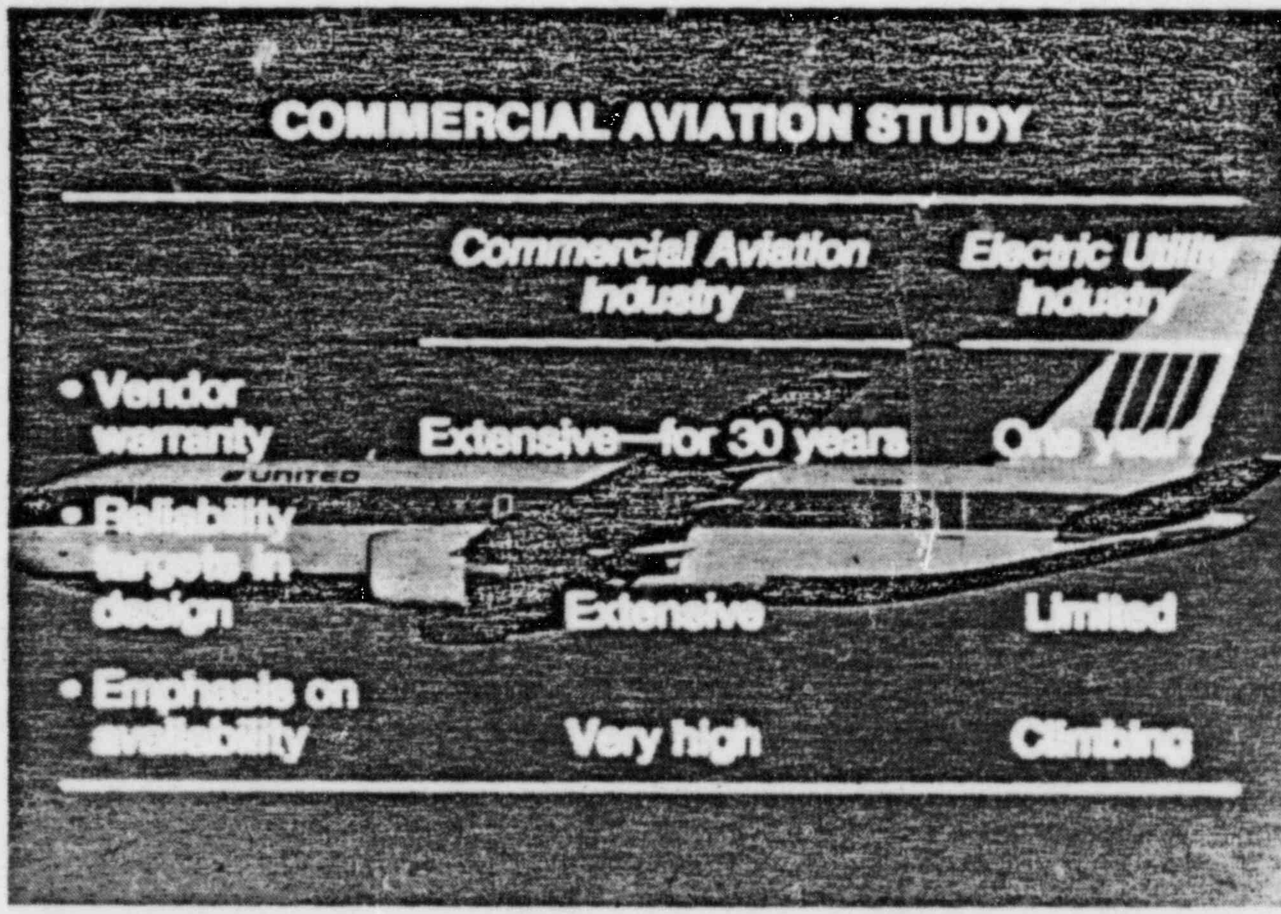

50 MILLION LB
MULTI AXIAL TEST FRAME


2



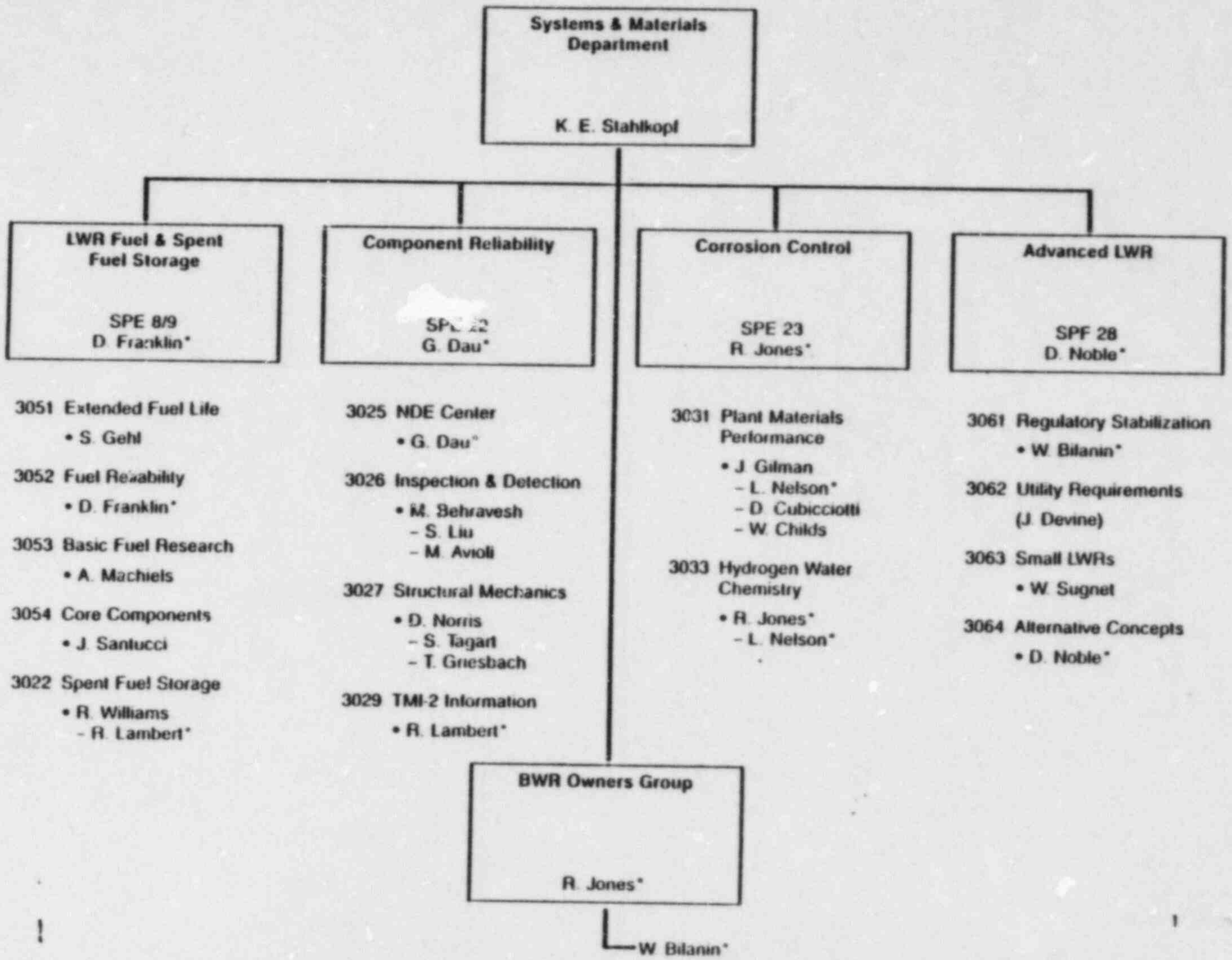

Date: 01-25-1985
SIGNAL VALIDATION DEMONSTRATION
Time: 10:41:23
PSG1 977.6 PSIA
975.7 PSIA
932.3 PSIA
979.6 PSIA
2202.2 PSIA
1600.0 PSIA
1500.5 PSIA
1600.0 PSIA
2200.9 PSIA
1500.4 PSIA
2210.4 PSIA
PPZR 2205.5 PSIA
THOT 591.0 F
536.9 F
591.0 F
590.9 F
591.1 F
590.2 F
541.0 F
541.1 F
536.9 F
TCOLD11
534.6 F
540.2 F
545.4 F
TCOLD12 540.1 F
540.1 F
TCOLD1 540.1 F


2


COMMERCIAL AVIATION STUDY
Commercial Aviation Industry
Electric Utility Industry
• Vendor warranty
Extensive—for 30 years
One year
• Reliability targets in design
Extensive
Limited
• Emphasis on availability
Very high
Climbing
UNITED

EPRI
SYSTEMS & MATERIALS DEPARTMENT

KARL E. STAHLKOPF



Systems and Materials Department Structure

*Dual Capacity

HIGH BURNUP FOR LWR FUEL

OBJECTIVE

- DEVELOP A TECHNICAL BASIS FOR USING AND LICENSING STANDARD—DESIGN LWR FUEL TO HIGH BURNUPS

HIGH BURNUP FOR LWR FUEL
(cont.)

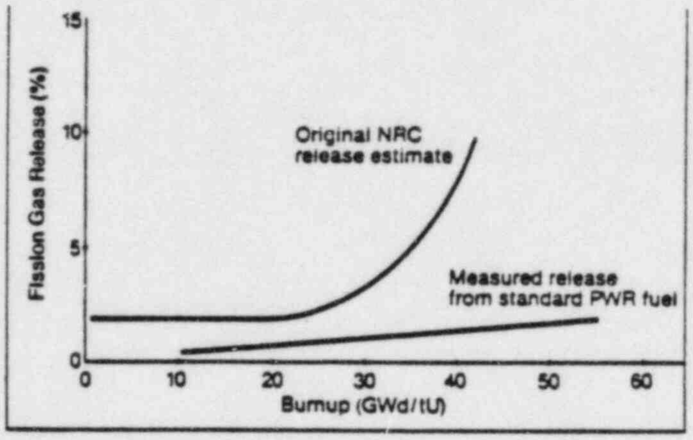
BENEFIT

- REDUCED FUEL—CYCLE COSTS
- ALLOWS ECONOMICAL IMPLEMENTATION OF LONG CYCLES, WHICH IMPROVE CAPACITY FACTORS
- EPRI CONTRACTORS (C—E AND W) OBTAINED NRC APPROVAL ON HIGH—BURNUP TOPICAL REPORTS BASED ON EPRI DATA

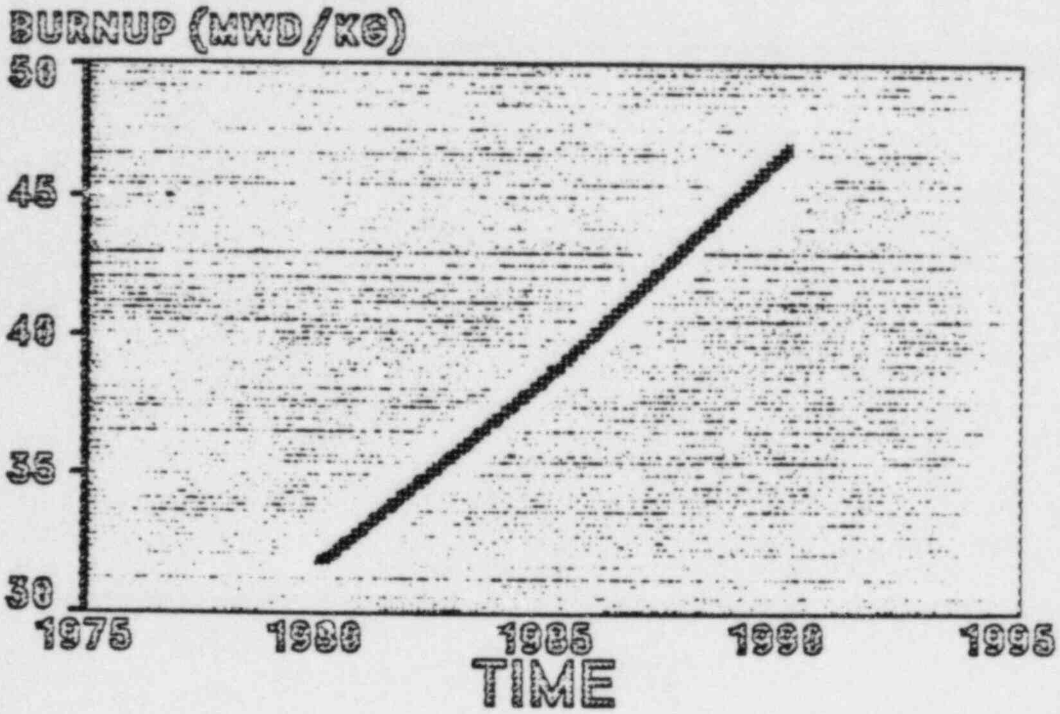
HIGH BURNUP FOR LWR FUEL (cont.)

STATUS

- EPRI COMPLETED LAST OF SURVEILLANCE PROGRAMS BY COMPLETING HOT CELL EXAMINATIONS OF PWR FUEL AFTER 55 GWd/t
- LIMITED WORK ON SELECTED PHENOMENA ONGOING:
  - — ZIRCALOY CORROSION
  - — FISSION—GAS RELEASE


Fission Gas Release (%)
15
10
5
0
Original NRC
release estimate
Measured release
from standard PWR fuel
0
10
20
30
40
50
60
Burnup (GWd/tU)

LEAD PWR UTILITY BURNUP SCHEDULE



HIGH BURNUP FOR LWR FUEL
(cont.)

APPLICATION

- ESSENTIALLY ALL U.S. PWRS EXTENDING BURNUP
- ABOUT HALF OF U.S. BWRS EXTENDING BURNUP
- BALTIMORE GAS & ELECTRIC DOCUMENTS "FIRST—USE" TOTAL BENEFITS OF $58 MILLION UP TO 1990 FOR EACH OF THE TWO CALVERT CLIFFS UNITS, BASED ON THE RESULTS OF RP586—1

ESCORE
FUEL—PERFORMANCE CODE

OBJECTIVE

- PROVIDE UTILITIES WITH CAPABILITY TO
  - EVALUATE FUEL DESIGNS
  - SUPPORT LICENSING OF RELOADS

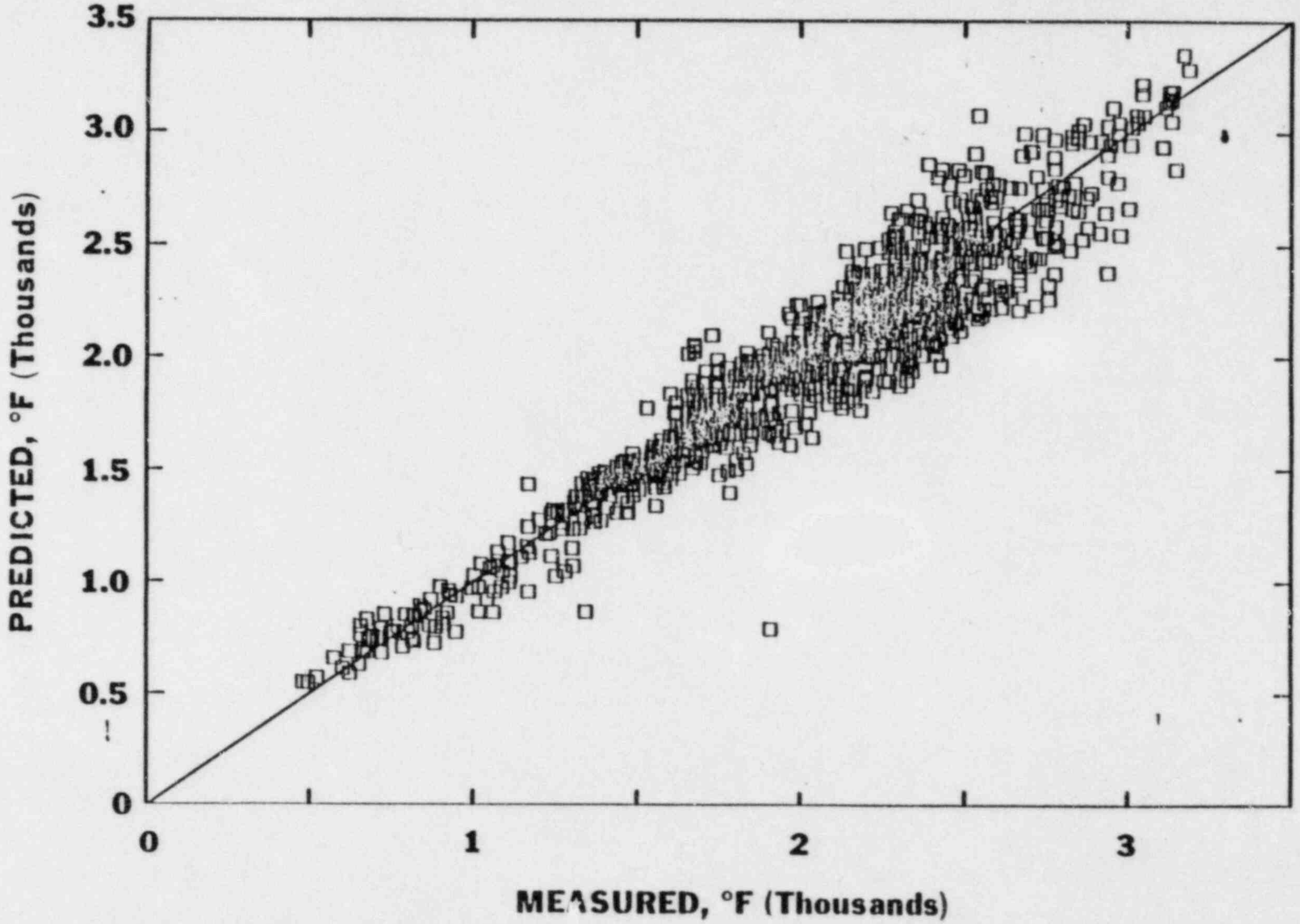
ESCORE
FUEL—PERFORMANCE CODE (cont.)

STATUS

- PRERELEASE VERSION RELEASED IN 1985
- FINAL VERSION COMPLETED: TO BE RELEASED BY SOFTWARE CENTER MARCH 31, 1986
- UTILITY GROUP FOR REGULATORY ACTION (UGRA) BEING FORMED TO SUPPORT SUBMITTAL TO NRC

ESCORE VERSION 1

PREDICTED VERSUS MEASURED TEMPERATURES



ESCORE

APPLICATION

- 15 UTILITIES OBTAINED PRERELEASE VERSION
- 20 UTILITIES INTERESTED IN FINAL VERSION AND GENERAL USERS GROUP
- 11 UTILITIES INTEND TO JOIN UGRA
- 1 UTILITY SCHEDULED TO USE ESCORE IN 1986 IN NRC LICENSING APPLICATION

INSPECTION AND DETECTION (NDE R&D)

OBJECTIVE

DEVELOP AND TRANSFER TO FIELD PRACTICE QUALIFIED, IMPROVED NDE EQUIPMENT AND PROCEDURES

SCOPE

- DETECTION AND SIZING OF IGSCC IN BWR PIPING
- PWR PIPING
- VESSEL INSPECTION
- STEAM GENERATOR INSPECTION
- BOLTING
- TURBINE

PIPING AND FITTING DYNAMIC RELIABILITY

OBJECTIVE

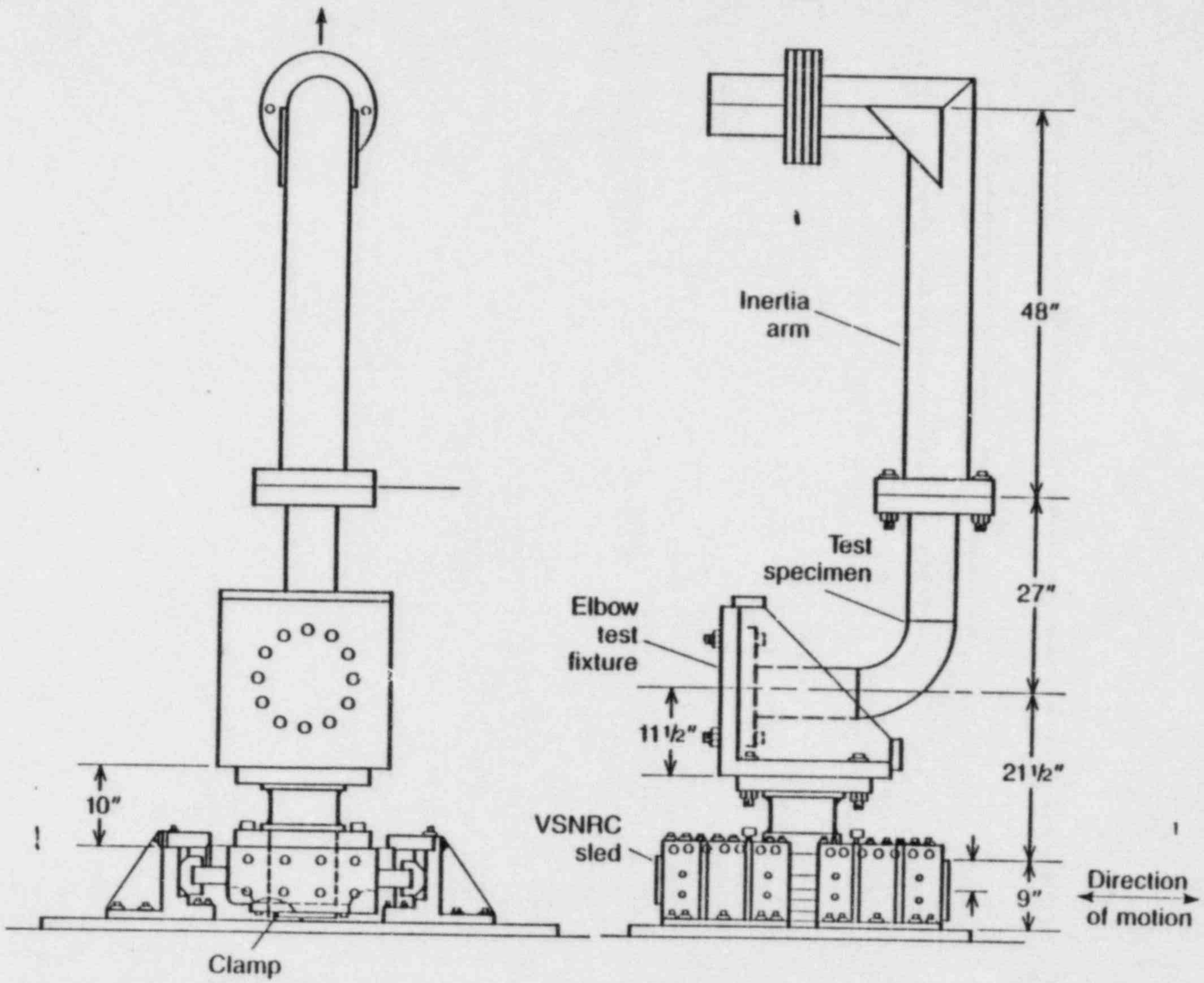
- TO DEVELOP AN IMPROVED, REALISTIC, AND DEFENSIBLE SET OF DESIGN RULES WHICH CAN BE ADOPTED BY THE ASME CODE FOR TREATMENT OF DYNAMIC PLUS STATIC PIPING LOADS

PIPING AND FITTING DYNAMIC RELIABILITY

STATUS

- PROJECT STARTED MARCH 15, 1985 AND EXPECTED TO LAST THREE YEARS
- THREE TYPES OF TESTS PLANNED; MATERIALS, PIPE COMPONENTS, PIPING SYSTEMS
- AS OF JANUARY 1, 1988, 8 OF 40 COMPONENT TESTS ARE COMPLETE

IN-PLANE SETUP FOR ELBOW COMPONENT TEST



PIPING AND FITTING DYNAMIC RELIABILITY

SIGNIFICANCE

- EARLY COMPONENT TEST RESULTS SUGGEST EXCESSIVE MARGIN OF 15 TO 20 TIMES SSE REQUIRED FOR PIPE FAILURE (MARGIN OF 1.5 IS CODE BASIS)
- ALLOWABLE STRESS CAN BE SIGNIFICANTLY INCREASED TO REDUCE USE OF SNUBBERS
- SIMPLIFIED DESIGN METHODS MAY BE FEASIBLE

PRESSURIZED THERMAL SHOCK

OBJECTIVE

- TO CORRECT ELASTIC MODELS FOR CRACK ARREST IN THE DUCTILE REGIONS OF NUCLEAR PRESSURE VESSELS

PRESSURIZED THERMAL SHOCK

STATUS

- THREE YEAR PROGRAM COMPLETED DECEMBER 31, 1985 — $1.4M
- OUR EXPERIMENTS HAVE SHOWN 50% HIGHER CRACK ARREST TOUGHNESS THAN THAT USED FOR NRC CRITERIA DEVELOPMENT

PRESSURIZED THERMAL SHOCK

SIGNIFICANCE

- ADDITIONAL LIFE FOR NEUTRON DAMAGED VESSEL STEEL

PREDICTION OF CRACK GROWTH IN REACTOR PRESSURE VESSEL (RPV) STEELS

BACKGROUND

- RELIABLE PREDICTION OF RPV FLAW BEHAVIOR IS AN IMPORTANT ELEMENT OF PLANT SAFETY AND LIFE EXTENSION EVALUATIONS

- LWR SERVICE ENVIRONMENTS ACCELERATE CRACK GROWTH RATES IN RPV MATERIALS

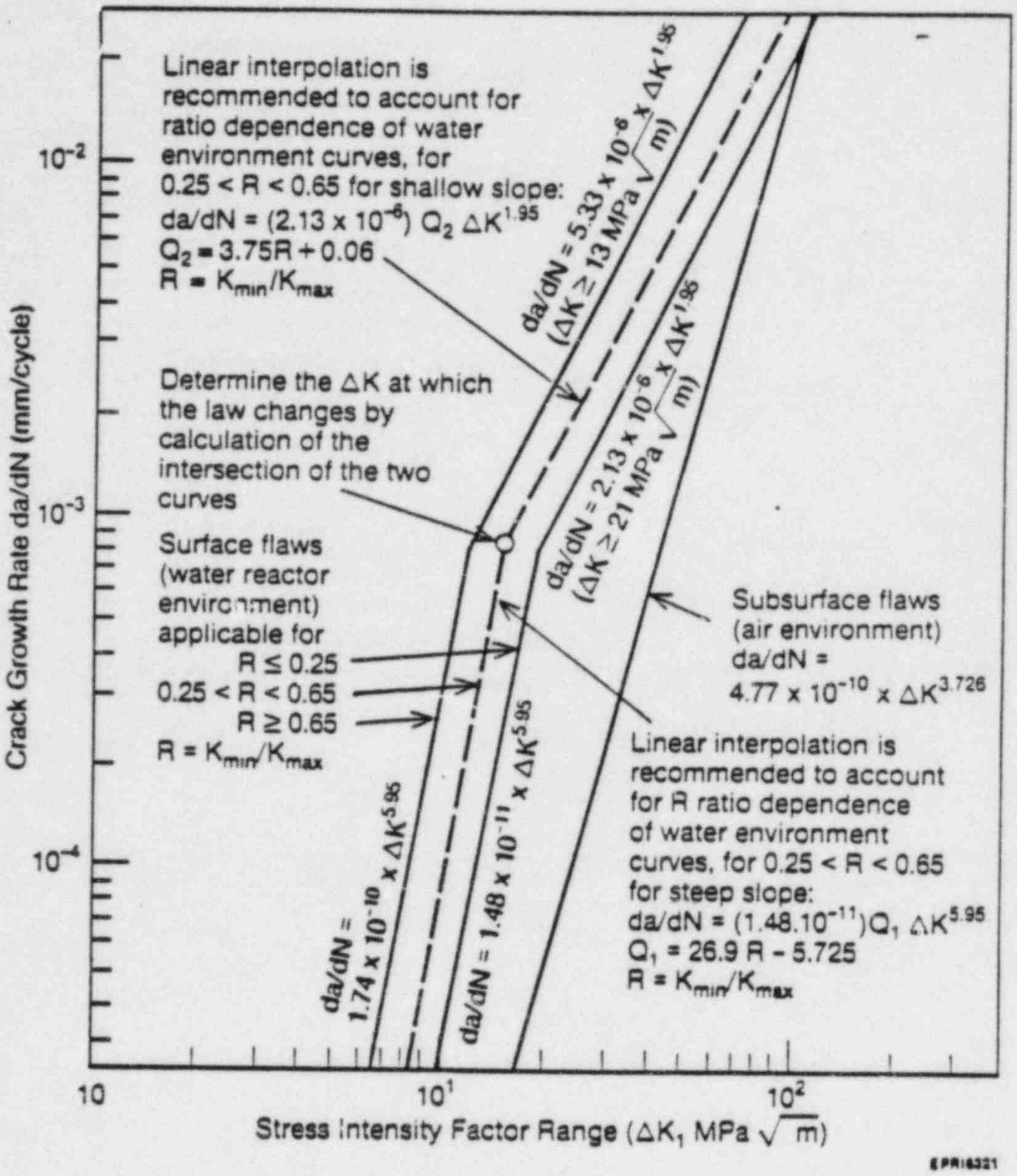


Figure 4. Revised Appendix A crack growth reference curves from the 1980 addenda. (SI edition)

PREDICTION OF CRACK GROWTH IN REACTOR PRESSURE VESSEL (RPV) STEELS

BACKGROUND (cont'd)

- CORROSION FATIGUE CRACK GROWTH HAS BEEN STUDIED EXTENSIVELY DURING THE PAST FIVE YEARS

- RELIABLE PREDICTIVE METHODS ARE NOW BEING DEVELOPED AND PROPOSED FOR ADOPTION BY INDUSTRY CODES

PREDICTION OF CRACK GROWTH IN REACTOR PRESSURE VESSEL (RPV) STEELS

STATUS

- SYSTEMATIC MEASUREMENTS OF CRACK GROWTH RATES HAVE BEEN MADE FORE A WIDE RANGE OF RELEVANT CONDITIONS. METALLURGICAL VARIABLES, ENVIRONMENTAL CONDITIONS, STRESSES, AND LOADING RATES ALL HAVE IMPORTANT EFFECTS ON CRACK GROWTH RATES

- EPRI DATA AND OTHER INDUSTRY DATA HAVE BEEN COLLECTED AND MADE ACCESSIBLE THROUGH EDEAC (EPRI DATABASE FOR ENVIRONMENTALLY ASSISTED CRACKING)

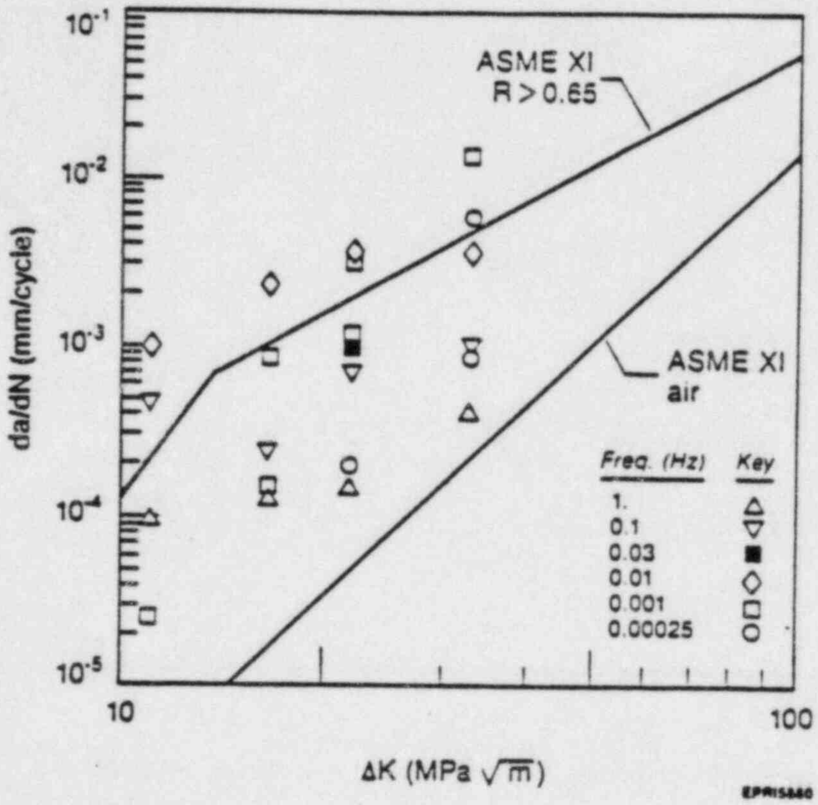
PREDICTION OF CRACK GROWTH IN REACTOR PRESSURE VESSEL (RPV) STEELS

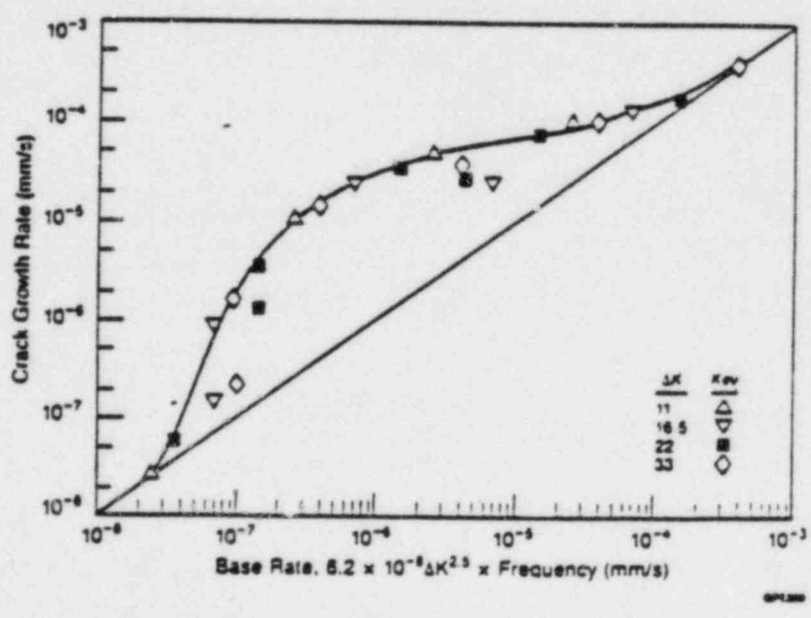
STATUS (cont'd)

- PARALLEL RESERACH HAS DEFINED CORROSION CRACKING MECHANISMS AND DEVELOPED PHYSICALLY— BASED MODELS OF THE PROCESSES

- ALL RAW DATA ARE BEING REANALYZED IN LIGHT OF THE PHYSICALLY—BASED MODELS, USING COMPUTER TOOLS DEVELOPED FOR THIS PURPOSE

SA533B-1, 0.025%S, 288°C BWR WATER, R = 0.7, 0.8





PREDICTION OF CRACK GROWTH IN REACTOR PRESSURE VESSEL (RPV) STEELS

PLANS/UTILIZATION

- DATA ANALYSIS HAS RESULTED IN A PROPOSED REVISION TO THE ASME CODE CRITERIA FOR EVALUATION OF FLAWS EXPOSED TO AIR

- A FUNDAMENTALLY NEW EVALUATION METHOD, TERMED "TIME DOMAIN ANALYSIS" HAS RESULTED FROM THE PHYSICALLY—BASED MODEL OF THE CORROSION CRACKING PROCESS IN WATER. REANALYSIS OF ALL DATA IN THE "TIME DOMAIN" IS IN PROGRESS. TRIAL APPLICATIONS HAVE BEEN CONDUCTED

PREDICTION OF CRACK GROWTH IN REACTOR PRESSURE VESSEL (RPV) STEELS

PLANS/UTILIZATION (cont'd)

- RESEARCH CONTINUES ON THE EFFECTS OF METAL-LURGICAL SULFUR ON CRACK GROWTH RATES. HIGHER RATES ARE ASSOCIATED WITH HIGHER SULFER LEVELS, TYPICAL OF DOMESTIC REACTOR PRESSURE VESSELS OF OLDER MANUFACTURE. DISCRIMINATING CRITERIA ARE NEEDED TO AVOID UNDUE PENALTIES IN FLAW EVALUATIONS.

HYDROGEN WATER CHEMISTRY

BACKGROUND

- INTERGRANULAR STRESS CORROSION CRACKING (IGSCC) OF BWR PIPING HAS HAD SIGNIFICANT IMPACT ON PLANT AVAILABILITY
- NUMEROUS REMEDIES HAVE BEEN DEVELOPED---ALTERNATE ALLOYS AND STRESS IMPROVEMENT TREATMENTS
- LABORATORY STUDIES AND LIMITED IN-REACTOR TESTS HAVE INDICATED THAT OXYGEN SUPPRESSION VIA HYDROGEN INJECTION CAN PREVENT IGSCC AS LONG AS WATER PURITY IS HIGH

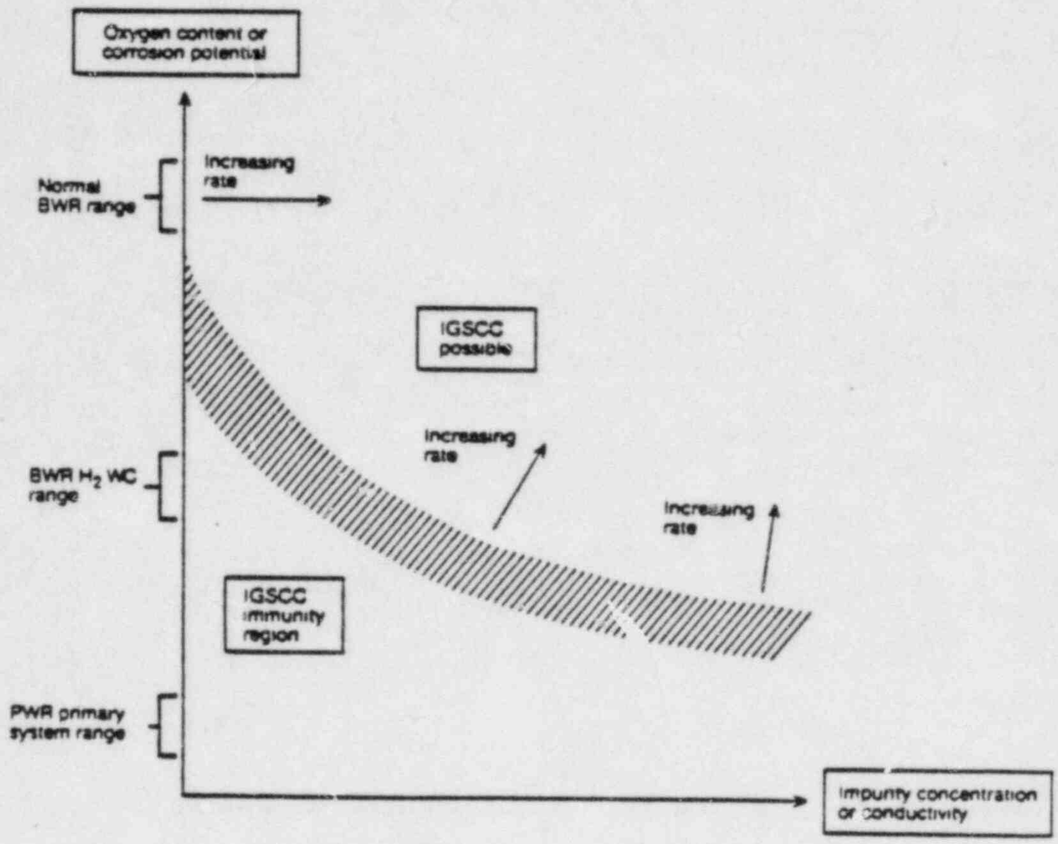
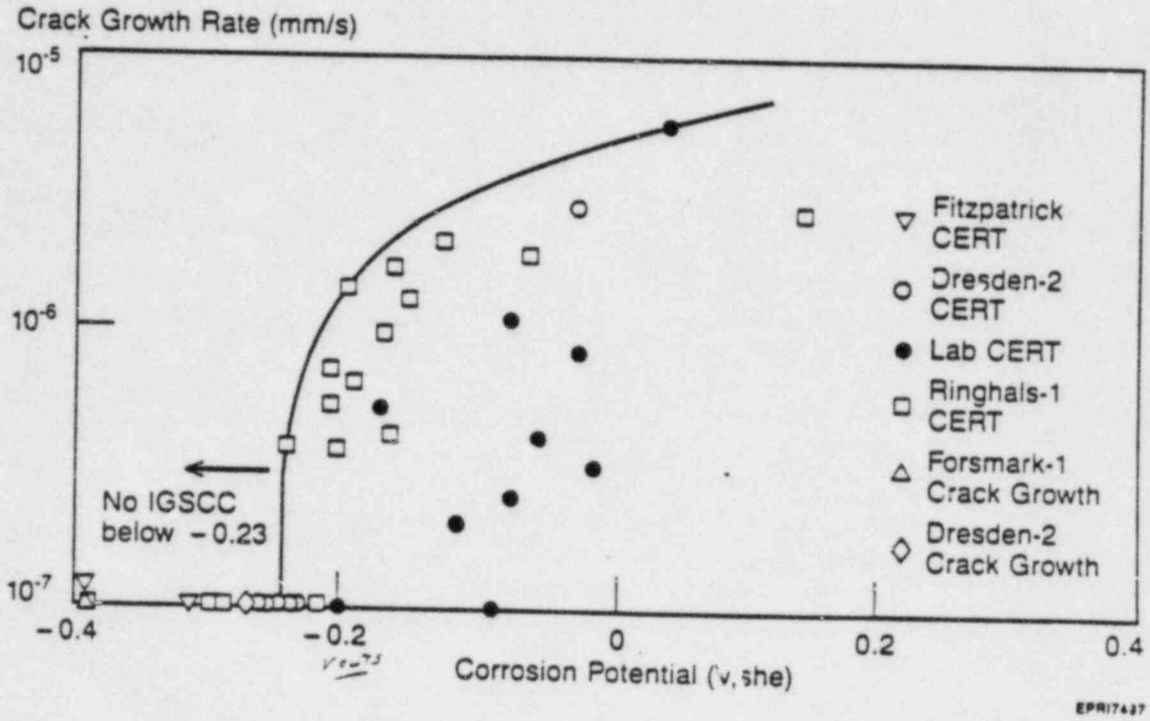


Figure 2-1 Schematic Summary of the Results of Laboratory Studies of Effect of Impurities on SCC of Sensitized Austenitic Stainless Steels

DEPENDENCE OF IGSCC IN SENSITIZED TYPE 304 STAINLESS STEEL ON ECP



HYDROGEN WATER CHEMISTRY

BACKGROUND (cont'd)

EPRI PROJECT WITH COMMONWEALTH EDISON CO., GENERAL ELECTRIC AND APPLIED PROCESS TECHNOLOGY TO

- DEMONSTRATE ABILITY TO OPERATE A BWR CONTIN-UOUSLY WITHIN HWC SPECIFICATION
- VERIFY THAT IGSCC WILL NOT INITIATE AND EXISTING FLAWS DO NOT PROPAGATE IN HWC
- DETERMINE THE EFFECT OF HWC ON OTHER STRUCTURAL MATERIALS AND ON THE FUEL
- DEFINE THE RADIOLOGICAL IMPACT OF HWC

HYDROGEN WATER CHEMISTRY (HWC)

1985 ACCOMPLISHMENTS (1)

- DOCUMENTED EFFECTS OF OPERATING DRESDEN-2 ON HWC DURING CYCLE 9
  - WATER CHEMISTRY
  - IN PLANT MATERIALS TESTING
  - RADIOLOGICAL IMPACT
  - FUEL SURVEILLANCE

HYDROGEN WATER CHEMISTRY (HWC)

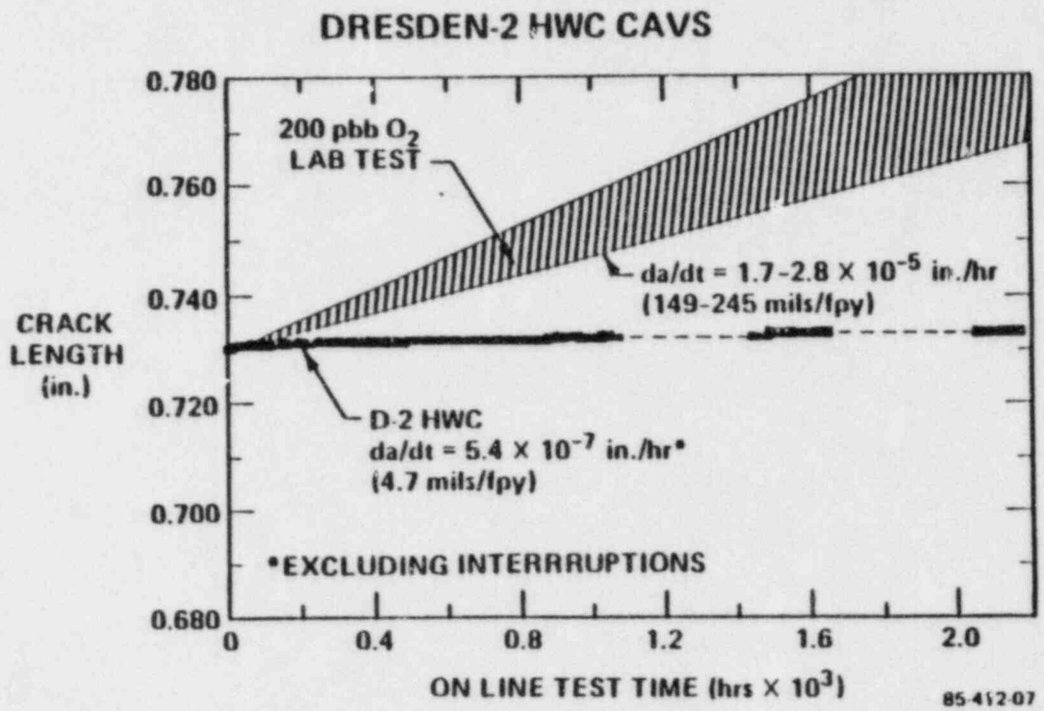
CYCLE 9 FUEL SURVEILLANCE RESULTS

- POOLSIDE AND HOT CELL EXAMINATIONS
- OXIDATION AND HYDRIDING BEHAVIOR
  - NO MEASURABLE EFFECTS OF HWC
- CRUD CHARACTERISTICS
  - SOME DIFFERENCES COMPARED TO NWC
  - NO IMPACT ON FUEL PERFORMANCE AT DRESDEN-2
  - WILL NEED TO BE ASSESSED AT OTHER PLANTS

HYDROGEN WATER CHEMISTRY (HWC)

1985 ACCOMPLISHMENTS (2)

- DEMONSTRATED THE EFFECTIVENESS OF HWC IN MITIGATING IGSCC AT DRESDEN–2 DURING THE FIRST 6 MONTHS OF CYCLE 10
  - WATER CHEMISTRY AND ECP MEASUREMENTS
  - ON–LINE CRACK GROWTH MONITOR (CAVS)


DRESDEN-2 HWC CAVS
0.780
0.760
0.740
0.720
0.700
0.680
CRACK LENGTH (in.)
200 pbb O_2 LAB TEST
da/dt = 1.7-2.8 × 10^{-5} in./hr
(149-245 mils/fpy)
D-2 HWC
da/dt = 5.4 × 10^{-7} in./hr*
(4.7 mils/fpy)
*EXCLUDING INTERRRUPTIONS
0
0.4
0.8
1.2
1.6
2.0
ON LINE TEST TIME (hrs × 10^3)
85-452-07

HYDROGEN WATER CHEMISTRY (HWC)

1985 ACCOMPLISHMENTS (3)

- DEVELOPED GENERIC GUIDELINES FOR HWC IMPLEMENTATION

  — DEFINES ASPECTS OF IMPLEMENTATION THAT CAN PROCEED UNDER 10CFR 50.59 AND CITES PERTINENT CODES AND STANDARDS

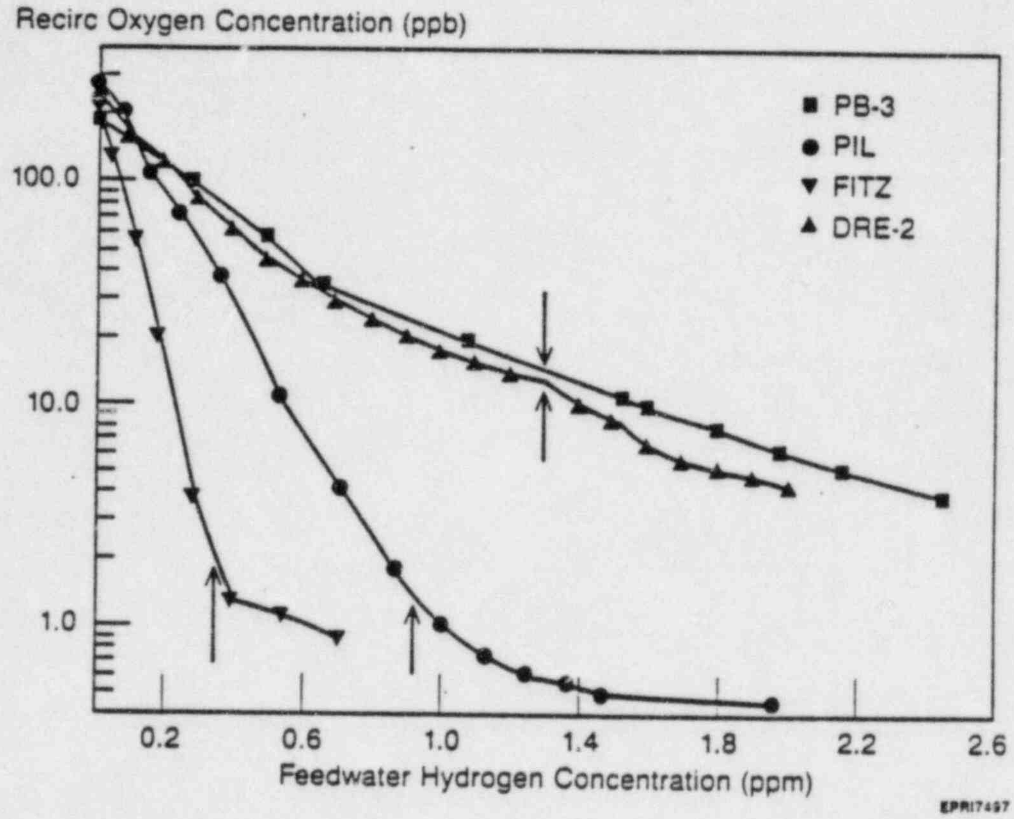
  — PROVIDES CONSERVATIVE ANALYSIS PROCEDURES FOR ASPECTS OF IMPLEMENTATION THAT NEED ANALYSIS (E.G., SITING OF CRYOGENIC STORAGE TANK)

  — NRC REVIEW IS IN PROCESS

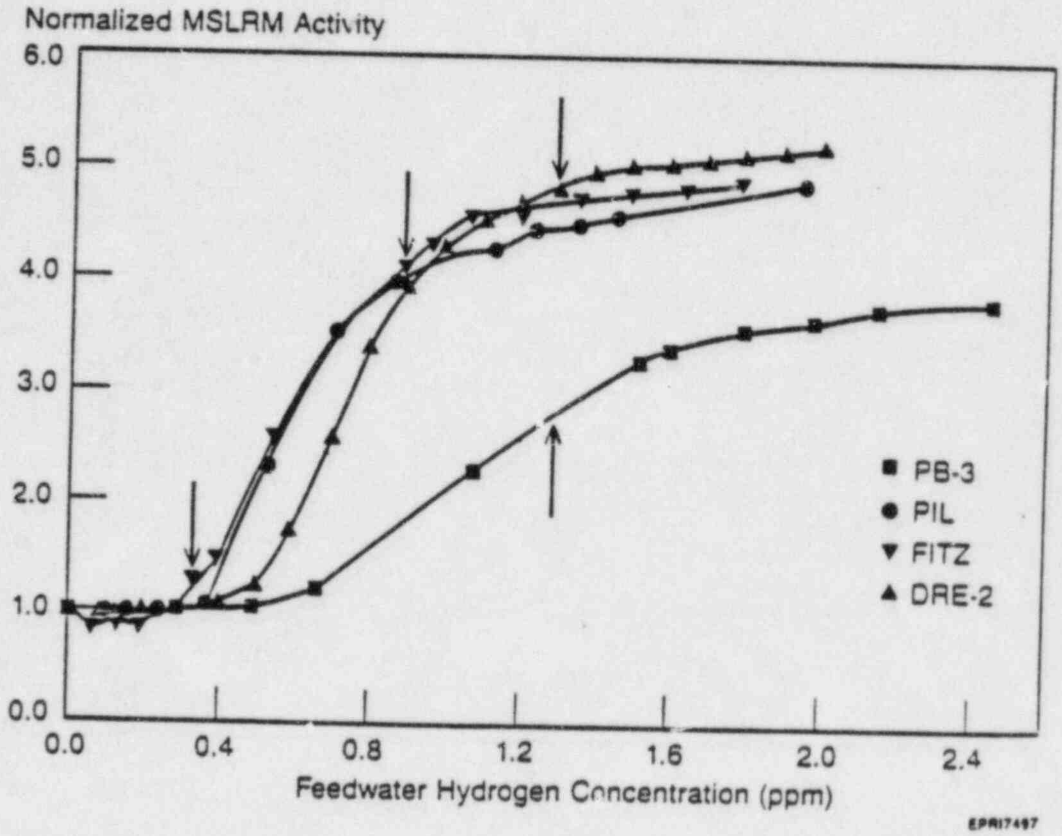
HYDROGEN WATER CHEMISTRY (HWC)

1985 ACCOMPLISHMENTS (4)

- ANALYZED THE RESULTS OF SHORT–TERM HYDROGEN INJECTION TESTS AT THREE U.S. PLANTS
  - CONTROL OF IGSCC BY HWC IS POSSIBLE AT ALL THREE PLANTS
  - EACH PLANT RESPONDS DIFFERENTLY TO HYDROGEN INJECTION
  - VARIABILITY OF RESPONSES CANNOT BE EXPLAINED BY DESIGN DIFFERENCES
  - THE KINETICS OF THE RADIOLYTIC DECOMPOSITION AND RECOMBINTAION REACTIONS INVOLVED IN HWC NEED TO BE UNDERSTOOD



Recirc Oxygen Concentration (ppb)
100.0
10.0
1.0
0.2
0.6
1.0
1.4
1.8
2.2
2.6
Feedwater Hydrogen Concentration (ppm)
PB-3
PIL
FITZ
DRE-2
EPRI7497



Normalized MSLRM Activity
6.0
5.0
4.0
3.0
2.0
1.0
0.0
0.0
0.4
0.8
1.2
1.6
2.0
2.4
Feedwater Hydrogen Concentration (ppm)
PB-3
PIL
FITZ
DRE-2
EPRI7497

HYDROGEN WATER CHEMISTRY

SUMMARY OF STATUS AND PLANS

- FIRST CYCLE OF HWC OPERATION AT DRESDEN—2 WAS COMPLETED 10/84 (CYCLE 9). OUTCOME WAS FAVORABLE
- SECOND CYCLE (CYCLE 10) IS IN PROGRESS—BEGAN 4/85. RESULTS TO DATE CONTINUE POSITIVE
- NEED FOR A THIRD CYCLE TO CONFIRM SATISFACTORY FUEL PERFORMANCE WILL BE ASSESSED DURING 1986
- SCOPE OF WATER CHEMISTRY WORK WILL BE EXPANDED TO ADDRESS PLANT—TO—PLANT VARIABILITY

PLANT MATERIALS PERFORMANCE

1985 ACCOMPLISHMENTS

- DEMONSTRATED THE FEASIBILITY OF AN ON—LINE IN—PLANT CORROSION CRACKING SENSOR WHICH CAN MONITOR THE AGGRESSIVENESS OF PLANT COOLANTS TOWARD STRUCTURAL MATERIALS

- DOCUMENTED THE EFFECTS OF A RANGE OF BWR WATER CHEMISTRY IMPURITIES ON IGSCC OF STAINLESS STEELS AND IDENTIFIED AN APPROACH TOWARD INHIBITOR DEVELOPMENT

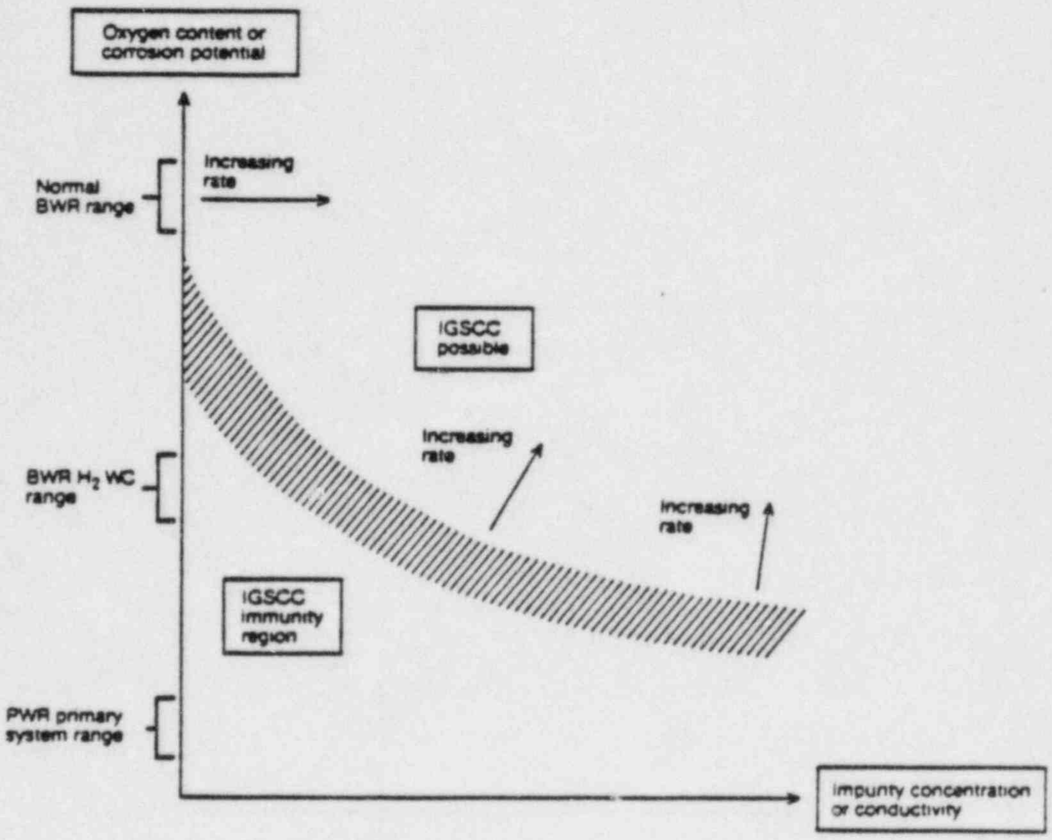


Figure 2-1 Schematic Summary of the Results of Laboratory Studies of Effect of Impurities on SCC of Sensitized Austenitic Stainless Steels

Crack Growth Rates of Types 304 and 316NG
Stainless Steels in 200 ppb Oxygen Water
at 550°F Containing Sodium Sulfate