Description of document:	Nuclear Regulatory Commission (NRC) Materials on the following Communities of Practice Site on the NRC Knowledge Center (on Sharepoint/Intranet): Mitigation Strategies for Beyond-Design-Basis External Events and New Reactor Technical Reviews, 2007- 2013
Requested date:	24-February-2017
Release 1 date:	01-May-2017
Release 2 date:	01-May-2017
Release 3 date:	01-May-2017
Posted date:	26-November-2018
Source of document:	FOIA Request U.S. Nuclear Regulatory Commission FOIA Officer Mailstop: T-2 F43 Washington, DC 20555-0001 Fax: 301-415-5130 Email: FOIA.resource@nrc.gov

From: Chidichimo, Gabriele <Gabriele.Chidichimo@nrc.gov>
Sent: Thu, Aug 17, 2017 1:36 pm
Subject: your FOIA request - 2017-0617 list of NRO records

Attached please find the list of NRO records you requested in order to narrow your scope.

Thank you again for all your help with this!

Gaby

Turbine Missiles - Explained
Lessons Learned from Flow-Induced Vibration to New Reactors
Digital I&C Operating Experience
Corrosion in Nuclear Power Plants
Graphite - Advanced Training
BWR Plant Startup and Shutdown
High Temperature Reactor Materials - Licensing and Regulatory Issue
Seismic Design of Small Modular Reactors

Regulating I&C Diversity for Advanced Reactors
SRP Section 3.9.4 - Control Rod Drive Systems
ASME Code - Explained
Confidence Interval on Estimate of Mean Value
Earthquake Effects on North Anna
Overview of ASME Section XI
PWR Startup and Shutdown
Insights on Performing SRP 6.2.1 Reviews
Containment ISI review and Action Plan
Containment Thermal-Hydraulic and Source Term Phenomena
Station Blackout and Emergency Diesel Generator
Pumps and Valves Training Slides
Seismic and Dynamic Qualification of Mechanical and Electrical Equipment
SRP Section 3.9.2 - Dynamic Testing and Analysis of Systems, Structures, and Components
SRP Section 3.2.1 - Seismic Classification

High Temperature Metallic Materials in HTGR & VHTR Systems
High Temp Reactors - Construction Code Issues
Alternatives to ASME Code Section III and IEEE 603 Requirements
Leak before Break – History, Updates, and Future Plans
Surry EMD Diesel Failure, Notice of Enforcement Discretion, and Generic Implications
Fundamentals of Pressurized Thermal Shock (PTS)
Reactor Vessel and Internals: History, Issues & Resolution
Commercial Grade Dedication of I&C Equipment
Sodium-Cooled Fast Reactors and LWRs
Environmentally Assisted Fatigue
ABCs of Welding
Key principles of I&C System Architecture

NRC FORM 464 Part I
(03-2017)

U.S. NUCLEAR REGULATORY COMMISSION

RESPONSE TO FREEDOM OF INFORMATION ACT (FOIA) REQUEST

FOIA	RESPONSE NUMBER
2017-0617	1

RESPONSE TYPE: ☑ INTERIM ☐ FINAL

REQUESTER:

DATE: AUG 2 8 2017

DESCRIPTION OF REQUESTED RECORDS:

A copy of the materials on the following Communities of Practice Site on the NRC Knowledge Center (on Sharepoint/ Intranet):
Mitigation Strategies for Beyond-Design-Basis External Events and New Reactor Technical Reviews

PART I. -- INFORMATION RELEASED

You have the right to seek assistance from the NRC's FOIA Public Liaison. Contact information for the NRC's FOIA Public Liaison is available at https://www.nrc.gov/reading-rm/foia/contact-foia.html

☑ Agency records subject to the request are already available on the Public NRC Website, in Public ADAMS or on microfiche in the NRC Public Document Room.

☐ Agency records subject to the request are enclosed.

☐ Records subject to the request that contain information originated by or of interest to another Federal agency have been referred to that agency (see comments section) for a disclosure determination and direct response to you.

☑ We are continuing to process your request.

☑ See Comments.

PART I.A -- FEES

AMOUNT*
$0.00
*See Comments for details

☐ You will be billed by NRC for the amount listed.

☐ You will receive a refund for the amount listed.

☐ Fees waived.

NO FEES

☑ Minimum fee threshold not met.

☐ Due to our delayed response, you will not be charged fees.

PART I.B -- INFORMATION NOT LOCATED OR WITHHELD FROM DISCLOSURE

☐ We did not locate any agency records responsive to your request. *Note*: Agencies may treat three discrete categories of law enforcement and national security records as not subject to the FOIA ("exclusions"). 5 U.S.C. 552(c). This is a standard notification given to all requesters; it should not be taken to mean that any excluded records do, or do not, exist.

☐ We have withheld certain information pursuant to the FOIA exemptions described, and for the reasons stated, in Part II.

☐ Because this is an interim response to your request, you may not appeal at this time. We will notify you of your right to appeal any of the responses we have issued in response to your request when we issue our final determination.

☐ You may appeal this final determination within 90 calendar days of the date of this response by sending a letter or e-mail to the FOIA Officer, at U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001, or FOIA.Resource@nrc.gov. Please be sure to include on your letter or email that it is a "FOIA Appeal." You have the right to seek dispute resolution services from the NRC's Public Liaison, or the Office of Government Information Services (OGIS). Contact information for OGIS is available at https://ogis.archives.gov/about-ogis/contact-information.htm

PART I.C COMMENTS (Use attached Comments continuation page if required)

Please note:
As discussed, you narrowed the scope of your request as follows:
KM materials ONLY, Indian Point records ONLY (NRR) and a list of 28 presentations as specified (NRO)
(continued on next page)

Signature - Freedom of Information Act Officer or Designee

NRC FORM 464 Part I (03-2017) U.S. NUCLEAR REGULATORY COMMISSION	FOIA	RESPONSE NUMBER
RESPONSE TO FREEDOM OF INFORMATION ACT (FOIA) REQUEST Continued	2017-0617	1
	RESPONSE TYPE ☑ INTERIM ☐ FINAL	

REQUESTER:	DATE:
	AUG 2 8 2017

PART I.C COMMENTS (Continued)

Please note:

The following responsive records have been made publicly available in their entirety (NRR, related to Indian Point):

ML14251A227
ML12319A008
ML15149A140
ML13337A597
ML15069A028
ML13247A032
ML14251A227
ML13079A348
ML14251A227
ML14070A365
ML130720080
ML12107A014
ML15246A119

Records with a ML Accession Number are publicly available in the NRC's Public Electronic Reading Room at http:www.nrc.gov/reading-rm.html. If you need assistance in obtaining these records, please contact the NRC's Public Documents Room (PDR) at 301-415-4737 or 1-800-397-4209, or by Email to PDR.Resource@nrc.gov.

NRC FORM 464 Part I
(03-2017)

U.S. NUCLEAR REGULATORY COMMISSION

RESPONSE TO FREEDOM OF INFORMATION ACT (FOIA) REQUEST

FOIA	RESPONSE NUMBER
2017-0617	2

RESPONSE TYPE: ☐ INTERIM ☑ FINAL

REQUESTER:

DATE: OCT 2 5 2017

DESCRIPTION OF REQUESTED RECORDS:

A copy of the materials on the following Communities of Practice Site on the NRC Knowledge Center (on Sharepoint/ Intranet):
Mitigation Strategies for Beyond-Design-Basis External Events and New Reactor Technical Reviews

PART I. -- INFORMATION RELEASED

You have the right to seek assistance from the NRC's FOIA Public Liaison. Contact information for the NRC's FOIA Public Liaison is available at https://www.nrc.gov/reading-rm/foia/contact-foia.html

- ☐ Agency records subject to the request are already available on the Public NRC Website, in Public ADAMS or on microfiche in the NRC Public Document Room.
- ☑ Agency records subject to the request are enclosed.
- ☐ Records subject to the request that contain information originated by or of interest to another Federal agency have been referred to that agency (see comments section) for a disclosure determination and direct response to you.
- ☐ We are continuing to process your request.
- ☑ See Comments.

PART I.A -- FEES

AMOUNT*

$0.00

*See Comments for details

- ☐ You will be billed by NRC for the amount listed.
- ☐ You will receive a refund for the amount listed.
- ☐ Fees waived.

NO FEES

- ☑ Minimum fee threshold not met.
- ☐ Due to our delayed response, you will not be charged fees.

PART I.B -- INFORMATION NOT LOCATED OR WITHHELD FROM DISCLOSURE

- ☐ We did not locate any agency records responsive to your request. *Note*: Agencies may treat three discrete categories of law enforcement and national security records as not subject to the FOIA ("exclusions"). 5 U.S.C. 552(c). This is a standard notification given to all requesters; it should not be taken to mean that any excluded records do, or do not, exist.
- ☑ We have withheld certain information pursuant to the FOIA exemptions described, and for the reasons stated, in Part II.
- ☐ Because this is an interim response to your request, you may not appeal at this time. We will notify you of your right to appeal any of the responses we have issued in response to your request when we issue our final determination.
- ☑ You may appeal this final determination within 90 calendar days of the date of this response by sending a letter or e-mail to the FOIA Officer, at U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001, or FOIA.Resource@nrc.gov. Please be sure to include on your letter or email that it is a "FOIA Appeal." You have the right to seek dispute resolution services from the NRC's Public Liaison, or the Office of Government Information Services (OGIS). Contact information for OGIS is available at https://ogis.archives.gov/about-ogis/contact-information.htm

PART I.C COMMENTS (Use attached Comments continuation page if required)

Please note:
As discussed, you narrowed the scope of your request as follows:
KM materials ONLY, Indian Point records ONLY (NRR) and a list of presentations as specified (NRO)
(continued on next page)

Signature - Freedom of Information Act Officer or Designee

Stephanie [signature]

NRC FORM 464 Part I (03-2017) U.S. NUCLEAR REGULATORY COMMISSION

RESPONSE TO FREEDOM OF INFORMATION ACT (FOIA) REQUEST Continued

FOIA	RESPONSE NUMBER
2017-0617	2

RESPONSE TYPE: ☐ INTERIM ☑ FINAL

REQUESTER:

DATE: OCT 25 2017

PART I.C COMMENTS (Continued)

Please note:

The NRC regrets to inform you that we are unable to locate the following records:

BWR Plant Startup and Shutdown
Unable to find presentation
NRO has no record

Insights on Performing SRP 6.2.1 Reviews
Unable to find presentation
No record found

Containment ISI review and Action Plan
Unable to find presentation
No record found

NRC FORM 464 Part II (08-2013)

U.S. NUCLEAR REGULATORY COMMISSION

RESPONSE TO FREEDOM OF INFORMATION ACT (FOIA) / PRIVACY ACT (PA) REQUEST

FOIA/PA 2017-0617

DATE OCT 25 2017

PART II.A -- APPLICABLE EXEMPTIONS

GROUP X

Records subject to the request that are contained in the specified group are being withheld in their entirety or in part under the Exemption No.(s) of the PA and/or the FOIA as indicated below (5 U.S.C. 552a and/or 5 U.S.C. 552(b)).

- [ ] Exemption 1: The withheld information is properly classified pursuant to Executive Order 12958.
- [ ] Exemption 2: The withheld information relates solely to the internal personnel rules and practices of NRC.
- [ ] Exemption 3: The withheld information is specifically exempted from public disclosure by statute indicated.
  - [ ] Sections 141-145 of the Atomic Energy Act, which prohibits the disclosure of Restricted Data or Formerly Restricted Data (42 U.S.C. 2161-2165).
  - [ ] Section 147 of the Atomic Energy Act, which prohibits the disclosure of Unclassified Safeguards Information (42 U.S.C. 2167).
  - [ ] 41 U.S.C., Section 4702(b), prohibits the disclosure of contractor proposals in the possession and control of an executive agency to any person under section 552 of Title 5, U.S.C. (the FOIA), except when incorporated into the contract between the agency and the submitter of the proposal.
- [x] Exemption 4: The withheld information is a trade secret or commercial or financial information that is being withheld for the reason(s) indicated.
  - [x] The information is considered to be confidential business (proprietary) information.
  - [ ] The information is considered to be proprietary because it concerns a licensee's or applicant's physical protection or material control and accounting program for special nuclear material pursuant to 10 CFR 2.390(d)(1).
  - [ ] The information was submitted by a foreign source and received in confidence pursuant to 10 CFR 2.390(d)(2).
  - [ ] Disclosure will harm an identifiable private or governmental interest.
- [ ] Exemption 5: The withheld information consists of interagency or intraagency records that are not available through discovery during litigation. Applicable privileges:
  - [ ] Deliberative process: Disclosure of predecisional information would tend to inhibit the open and frank exchange of ideas essential to the deliberative process. Where records are withheld in their entirety, the facts are inextricably intertwined with the predecisional information. There also are no reasonably segregable factual portions because the release of the facts would permit an indirect inquiry into the predecisional process of the agency.
  - [ ] Attorney work-product privilege. (Documents prepared by an attorney in contemplation of litigation)
  - [ ] Attorney-client privilege. (Confidential communications between an attorney and his/her client)
- [ ] Exemption 6: The withheld information is exempted from public disclosure because its disclosure would result in a clearly unwarranted invasion of personal privacy.
- [ ] Exemption 7: The withheld information consists of records compiled for law enforcement purposes and is being withheld for the reason(s) indicated.
  - [ ] (A) Disclosure could reasonably be expected to interfere with an enforcement proceeding (e.g., it would reveal the scope, direction, and focus of enforcement efforts, and thus could possibly allow recipients to take action to shield potential wrong doing or a violation of NRC requirements from investigators).
  - [ ] (C) Disclosure could constitute an unwarranted invasion of personal privacy.
  - [ ] (D) The information consists of names of individuals and other information the disclosure of which could reasonably be expected to reveal identities of confidential sources.
  - [ ] (E) Disclosure would reveal techniques and procedures for law enforcement investigations or prosecutions, or guidelines that could reasonably be expected to risk circumvention of the law.
  - [ ] (F) Disclosure could reasonably be expected to endanger the life or physical safety of an individual.
- [ ] OTHER (Specify)

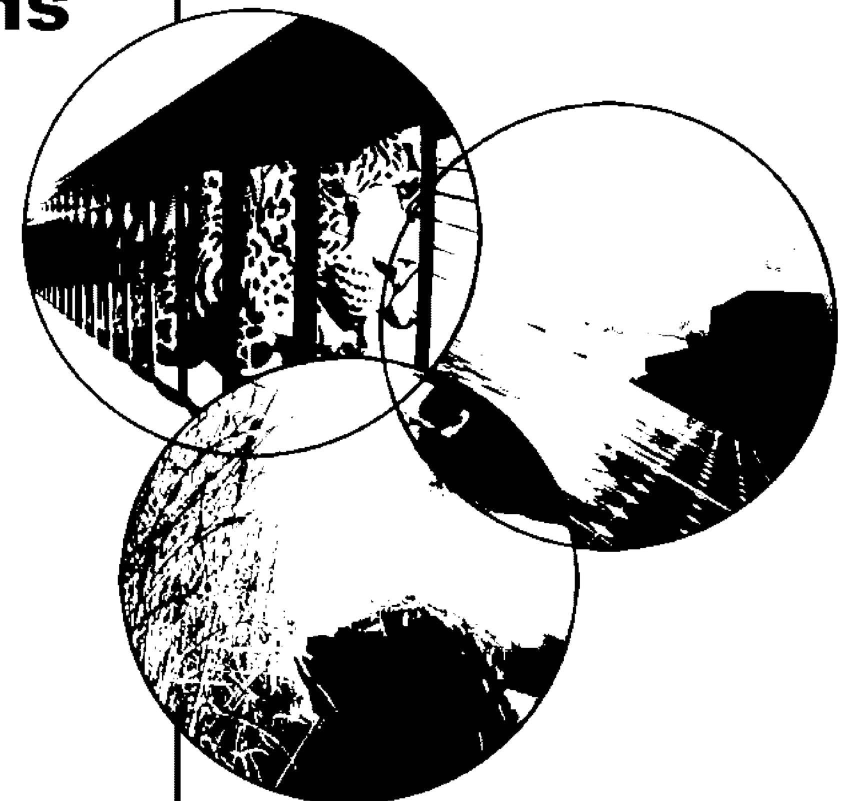
PART II.B -- DENYING OFFICIALS

Pursuant to 10 CFR 9.25(g), 9.25(h), and/or 9.65(b) of the U.S. Nuclear Regulatory Commission regulations, it has been determined that the information withheld is exempt from production or disclosure, and that its production or disclosure is contrary to the public interest. The person responsible for the denial are those officials identified below as denying officials and the FOIA/PA Officer for any denials that may be appealed to the Executive Director for Operations (EDO).

DENYING OFFICIAL	TITLE/OFFICE	RECORDS DENIED	APPELLATE OFFICIAL EDO	SECY	IG
Stephanie Blaney	FOIA/PA Officer	X	✓		

Appeal must be made in writing within 30 days of receipt of this response. Appeals should be mailed to the FOIA/Privacy Act Officer, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, for action by the appropriate appellate official(s). You should clearly state on the envelope and letter that it is a "FOIA/PA Appeal."

High Temperature Metallic Materials in HTGR & VHTR Systems



Presented at the
NRC Tutorial on High Temperature Metallic Materials for HTGR and SFR Reactor Systems

William Corwin,
Oak Ridge National Laboratory

Rockville, Maryland
February 17, 2011

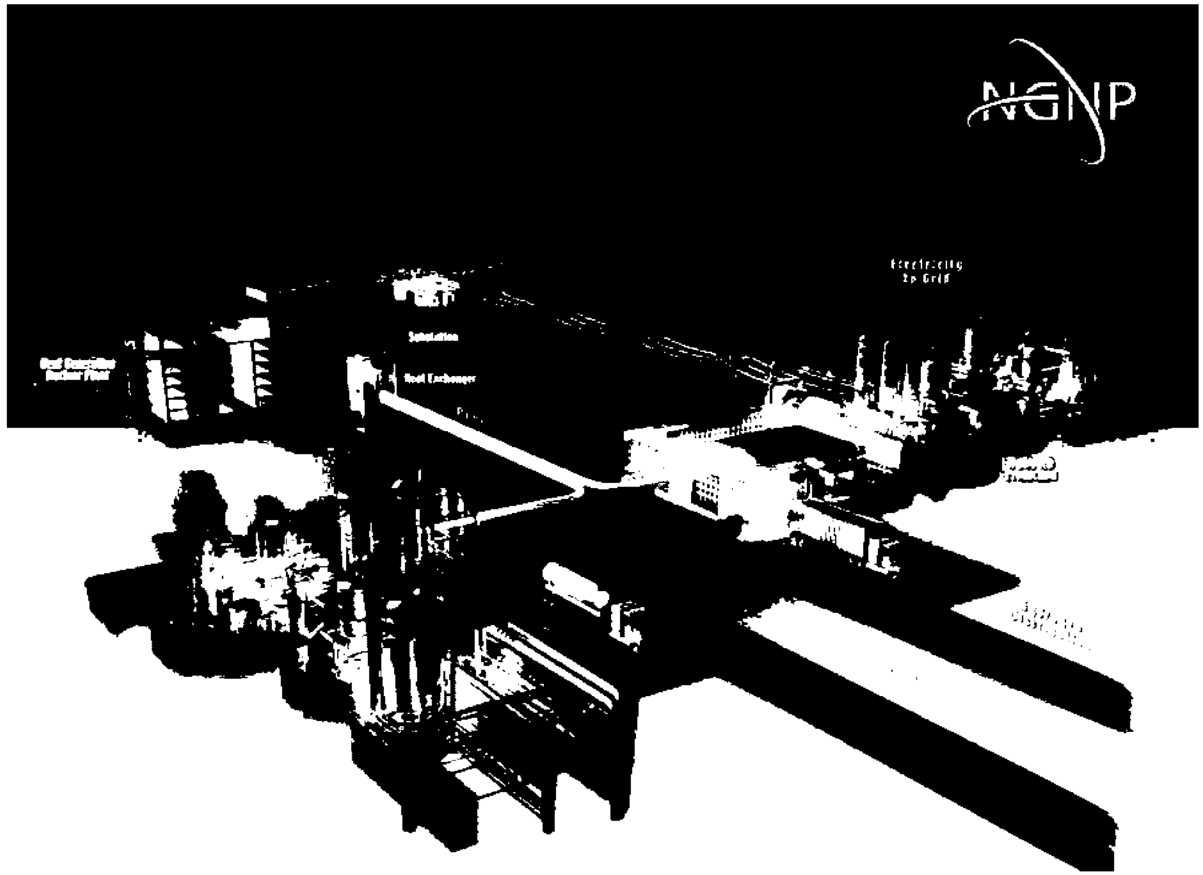
U.S. DEPARTMENT OF
ENERGY

OAK RIDGE NATIONAL LABORATORY
MANAGED BY UT-BATTELLE FOR THE DEPARTMENT OF ENERGY

(V)HTRs Can Provide High Efficiency Electricity and High Quality Process Heat

Characteristics

- He coolant
- Up to 1000°C outlet temperature (long term, <850°C near term)
- <600 MW_{th}
- Solid graphite block or pebble bed core



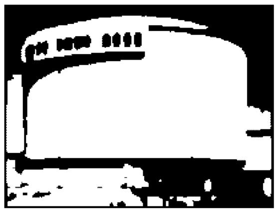
Benefits

- High thermal efficiency
- Process heat applications
- High degree of passive safety

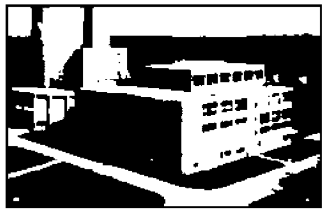
High Temperature, Gas-Cooled Reactor Experience Is Widespread

HTGR PROTOTYPE PLANTS

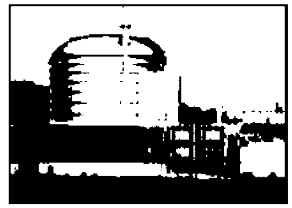
DEMONSTRATION PLANTS



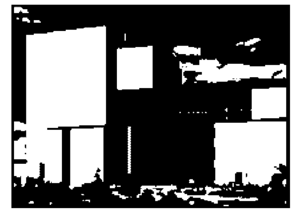
DRAGON
(U.K.)
1963 - 76



AVR
(FRG)
1967 - 1988



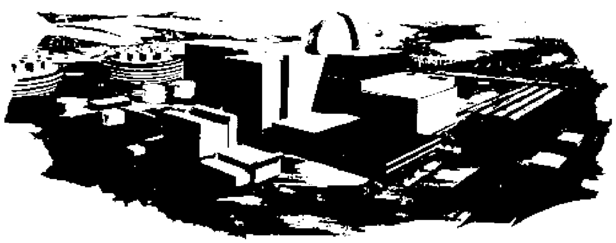
PEACH BOTTOM 1
(U.S.A.)
1967 - 1974



FORT ST. VRAIN
(U.S.A.)
1976 - 1989

THTR
(FRG)
1986 - 1989

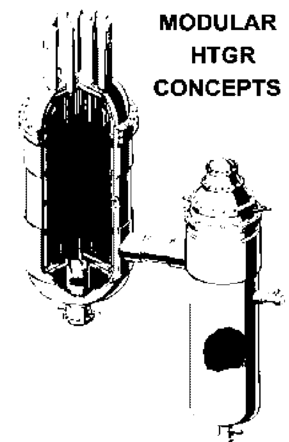
LARGE HTGR PLANTS



HTGR TECHNOLOGY PROGRAM

MATERIALS
COMPONENTS
FUEL
CORE
PLANT TECHNOLOGY

MODULAR HTGR CONCEPTS



Pebble Bed Reactor Experience

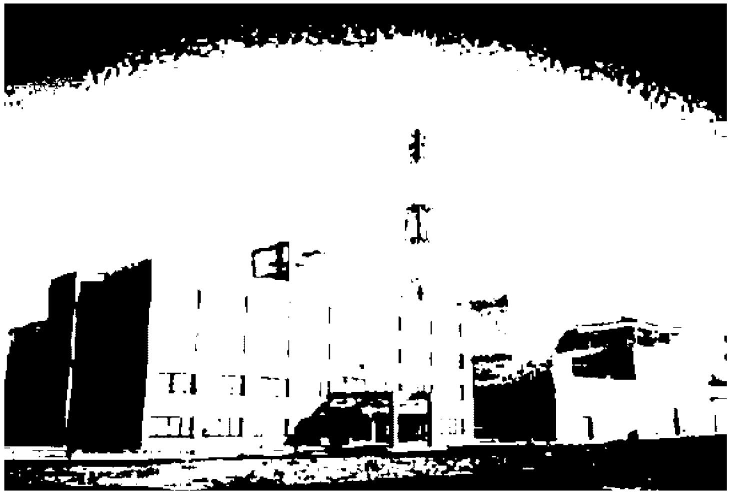
Major Projects	Power (MWt)	Status
AVR (Germany)	50	Being Decommissioned
THTR 300 (Germany)	750	Decommissioned
HTR 500 (Germany)	1390	Prel Design/Lic Review Archived
HTR 100 (Germany)	250	Prel Design/Lic Review Archived
HTR Modul (US, Germany)	200	Prel Design/Lic Review Archived – Safety Concept License Approved
DPP 400 (South Africa)	400	Prel Design/Lic Review Archived – Major Components Canceled
HTR-10 (China)	10	Operating
PM-250 (China)	250	Construction Underway
PBMR-CG (NGNP)	250	Conceptual Design Underway

PBR conceived in US in ‘44; 1st patent filed in US in ‘59; 1st pebbles mfg by Union Carbide

from Sten Caspersson, Westinghouse Electric Co., PBMR Conceptual Design, VHTR R&D FY10 Technical Review Meeting, Denver, Apr 2010

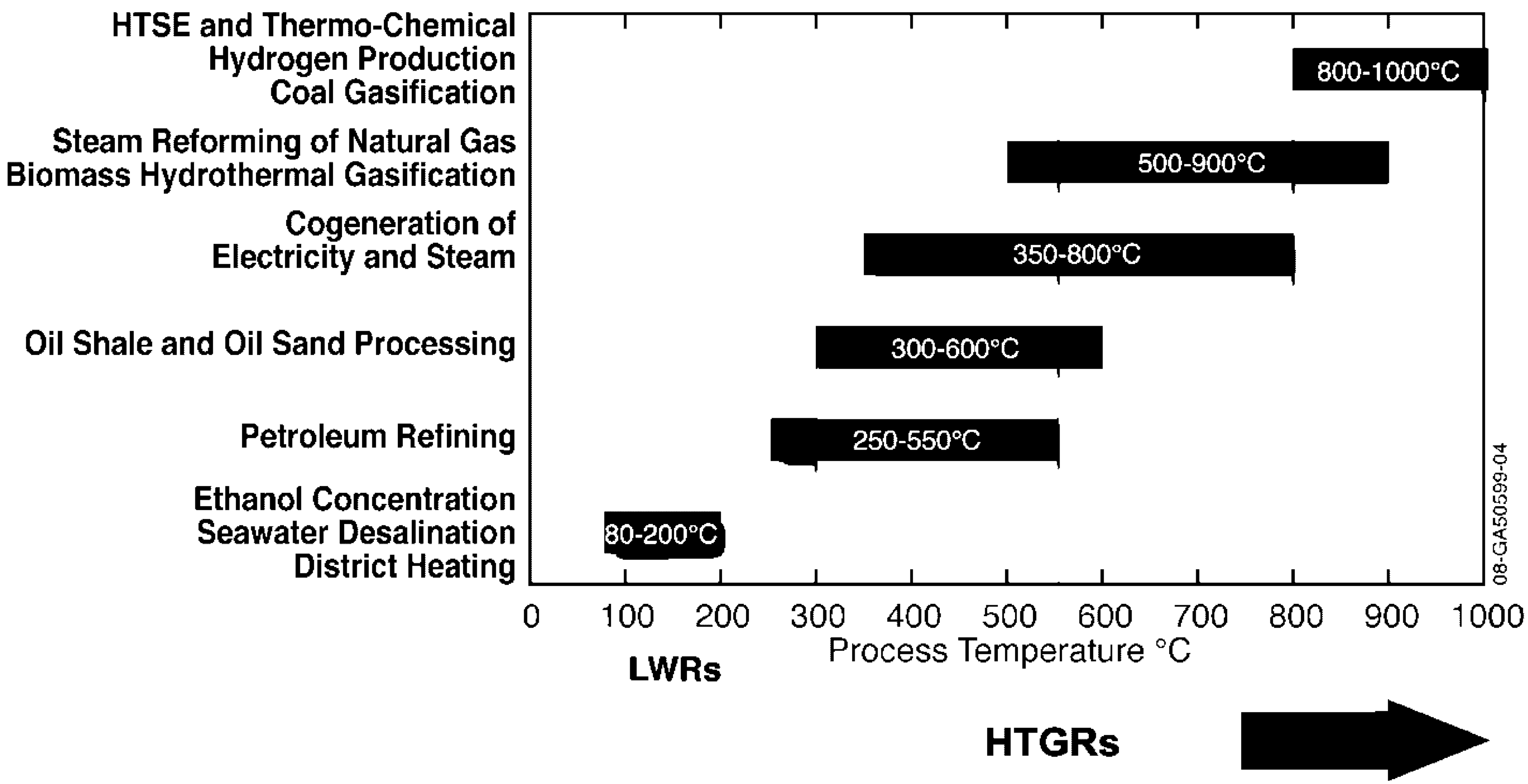
Two HTRs Are Currently Operating

- HTTR (prismatic core)
  - JAEA, Oarai, Japan
  - Up to 950°C ROT
  - 50MWth
  - Targeted IS hydrogen production
  - GTHTR300C (600MWth) to follow



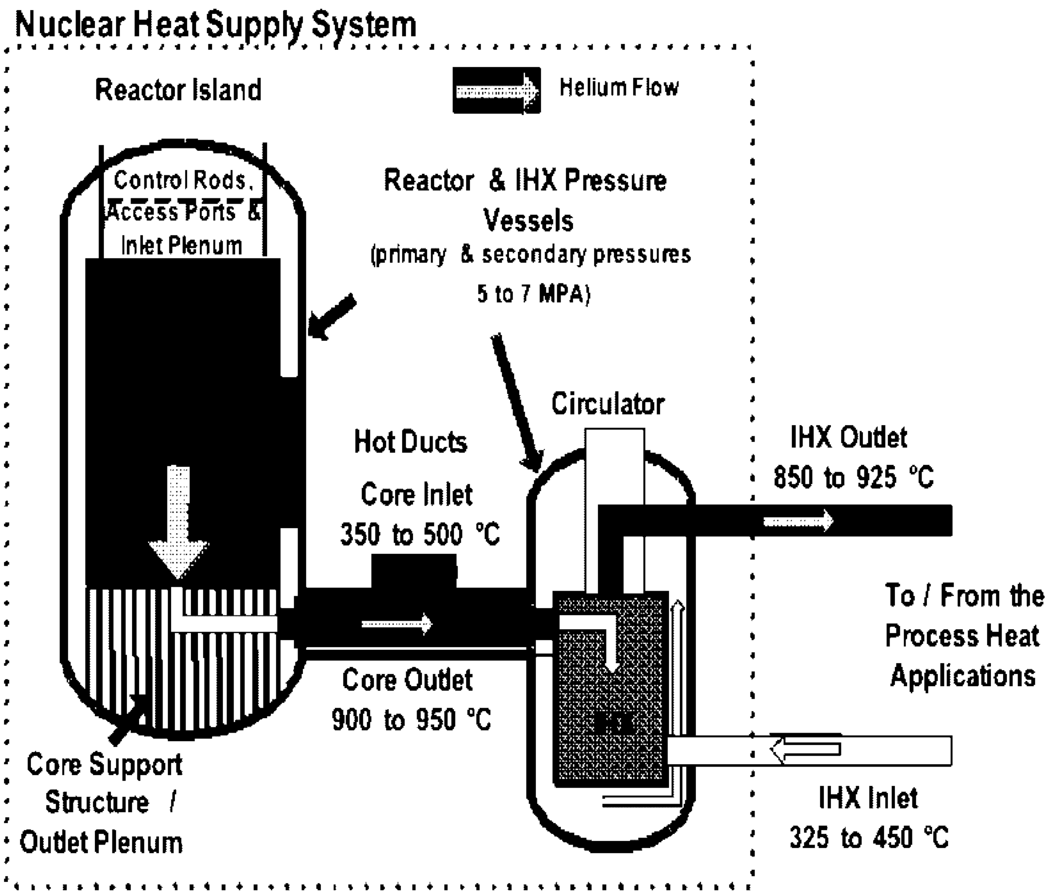
- HTR-10 (pebble bed core)
  - Tsinghua University (INET), China
  - 750°C ROT
  - 10MWth
  - Steam generation
  - PM250 (2x250MWth) to follow

VHTRs & HTGRs Can Provide Energy for Many Applications beyond Electricity



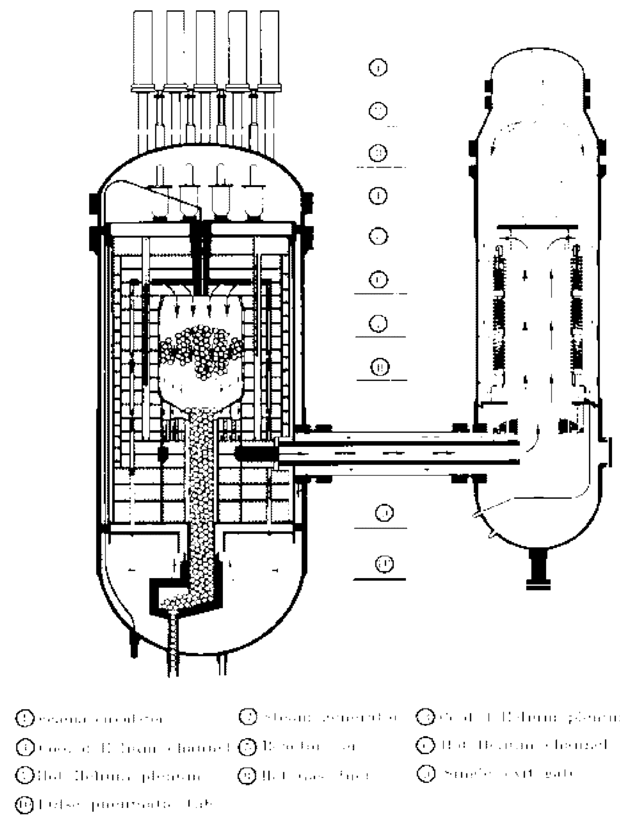
(V)HTR Process Heat Generation Is Simple in Concept

Generic He-He VHTR



* Core includes fuel, graphite, core structural and other ceramic components and the metallic core barrel

He-Steam HTR-10



from Qin Zhenya, *Tsinghua Univ. (INET), HTGR Reactor Devl. in Mainland China, Intl. Symp. for Gen IV Nucl. Reactors, Taipei, Apr. 2009*

Possible NGNP Configurations Described by INL

- Courtesy of Lee Nelson, INL, Leader NGNP Design Activities for pebble bed variants of NGNP
  - 208-526-3093
  - Lee.Nelson@INL.gov
- Presented at INL, October 28, 2009, to NRC-RES staff and updated January 18, 2011
- Based primarily on completed preconceptual design studies by INL and multiple vendors through 2007, as modified by subsequent design studies

Preconceptual Designs (May, 2007) Targeted High Outlet Temperatures

Item	Recommended Operating Conditions and Plant Configuration		
	Westinghouse	AREVA	General Atomics
Power Level, MWth	500 MWth	565 MWth	550-600 MWth
Reactor Outlet Temperature, °C	950°C	900°C	Up to 950°C
Reactor Inlet Temperature, °C	350°C	500°C	490°C
Cycle Configuration	Indirect – Series hydrogen process and power conversion	Indirect – Parallel hydrogen process (He) and power conversion (Helical Shell and Tube IHX)	Direct PCS (Brayton) Parallel indirect hydrogen process (IHX with He)
Secondary Fluid	He	He – Nitrogen mixture to PCS He to H_2 Process	He
Power Conversion Power	100% of reactor power	100% of reactor power	100% of reactor power
Hydrogen Plant Power	10% of reactor power	10% of reactor power	5 MWth – THE 60 MWth – S-I
Reactor Core Design	Pebble Bed	Prismatic	Prismatic

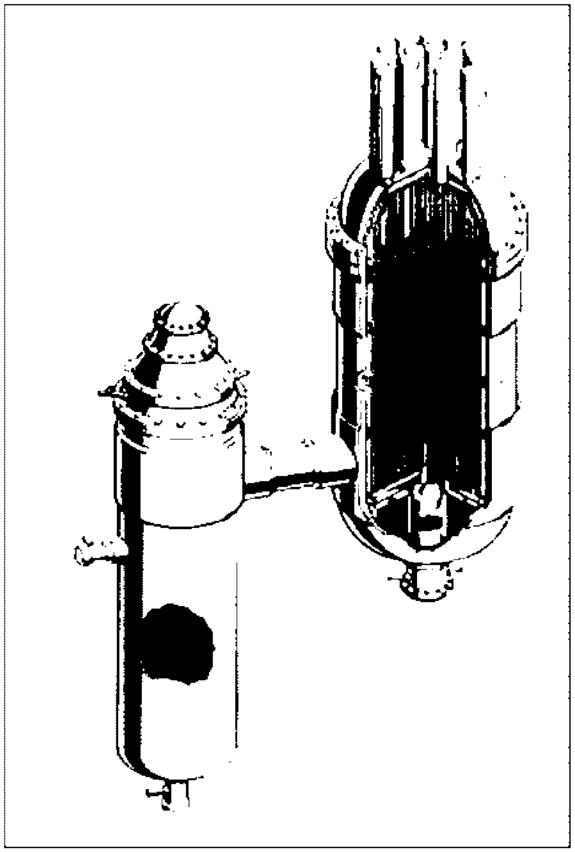
from Lee Nelson, INL

Preconceptual Designs (May, 2007) Targeted High Outlet Temps (cont)

Item	Recommended Operating Conditions and Plant Configuration		
	Westinghouse	AREVA	General Atomics
Fuel	TRISO UO_2 1st and subsequent cores	TRISO UCO – 1st and subsequent cores	TRISO UO_2 1st core Variable subsequent cores
Graphite	PCEA & NBG-18	NGG-17 and NBG-18	IG-110 & NBG-18
RPV Design	Exposed to the gas inlet temperature	Exposed to the gas inlet temperature; insulation and vessel cooling options may be pursued	Exposed to the gas inlet temperature
RPV Material	SA508/533	9Cr – 1 Mo	2-1/4 Cr – 1 Mo 9 Cr – 1 Mo
IHX	2-Stage Printed Circuit Heat Exchanger (PCHE), In 617 material	PCS – 3-Helical Coil Shell & Tube, In 617 Process – PCHE or Fin-Plate, In 617	Process – single stage PCHE, In 617
Hydrogen Plant	Hybrid thermo-chemical plus electrolysis	Initial-High Temperature Electrolysis Longer Term – Sulfur-Iodine	Initial-High Temperature Electrolysis Longer Term – Sulfur-Iodine
Power Conversion	Rankine; standard fossil power turbine generator set	Rankine; standard fossil power turbine generator set	Direct gas turbine Option – Direct Combined Cycle

from Lee Nelson, INL

Prismatic Reactors Based on MHTGR (GA) with Cross Vessel and Steam Generator



MHTGR Typical Plant Parameters

Thermal Power, MW(t)	600
Fuel Columns	102
Fuel Cycle	LEU/Natural U
Average Power Density, W/cm^3	6.6
Primary Side Pressure, MPa (psia)	7.07 (1025)
Induced Helium Flowrate	281 kg/s
Core Inlet Temperature, °C (°F)	288(550)
Core Outlet Temperature, °C (°F)	704(1300)

from Lee Nelson, INL

Effect of Power Level on Reactor Vessel – GA
(based on existing design information)

Reactor Parameter	350 MWth	450 MWth	550 MWth	600 MWth
Reactor Vessel ID, m*	6.55	7.22	7.22	7.22
RPV Thickness, m	0.133	---	---	0.216
RPV Height, m*	22.0	---	---	24.0
RPV Weight, t	728	---	---	1328
Reactor Vessel Material	SA 508/533	SA508/533	2 ¼ Cr-1Mo or 9 Cr-1 Mo (with active vessel cooling would be SA508/533)	2 ¼ Cr-1Mo or 9 Cr-1 Mo (with active vessel cooling would be SA508/533)

* Pebble Bed RPV for 500 MWth plant is 6.8m OD and height of 30m

from Lee Nelson, INL

Metallic Materials (GA) Function of ROT

Component	Temperature Conditions	750C	Mat'l Selection	850C	Mat'l Selection	950C	Mat'l Selection
Inner Control Rod	Normal Ops PCCD max DCCD Max	**808** **1164** **1418**	**C-C or SiC-SiC**	**850** **1174** **1428**	**C-C or SiC-SiC**	**871** **1179** **1433**	**C-C or SiC-SiC**
Outer Control Rod	Normal Ops PCCD max DCCD Max	**440** **929** **980**	**C-C or SiC-SiC**	**482** **939** **990**	**C-C or SiC-SiC**	**526** **1129** **1000**	**C-C or SiC-SiC**
CR and RSM Guide Tubes	Normal Ops PCCD max DCCD Max	346 933 418	Hastelloy X		**C-C or SiC-SiC**		**C-C or SiC-SiC**
UCR	Normal Ops PCCD max DCCD Max	**346** **1028** **604**	**C-C or SiC-SiC**	**346** **1038** **614**	**C-C or SiC-SiC**	**346** **1048** **624**	**C-C or SiC-SiC**
UPS T/B	Normal Ops PCCD max DCCD Max	318 877 455	Hastelloy X	318 887 465	Hastelloy X	318 897 475	Hastelloy X
MCS Load pads	Normal Ops PCCD max DCCD Max	653 653 653	Macor Glass Ceramic	730 730 730	Macor Glass Ceramic	807 807 807	Macor Glass Ceramic

PCCD = Pressurized Conduction Cool Down
DCCD = Depressurized Conduction Cool Down
RSM = Reserve Shutdown Material
UCR = Upper Control Rod
UPS T/B= Upper Plenum Shroud Thermal Barrier
MCS = Metallic Core Supports

from Lee Nelson, INL

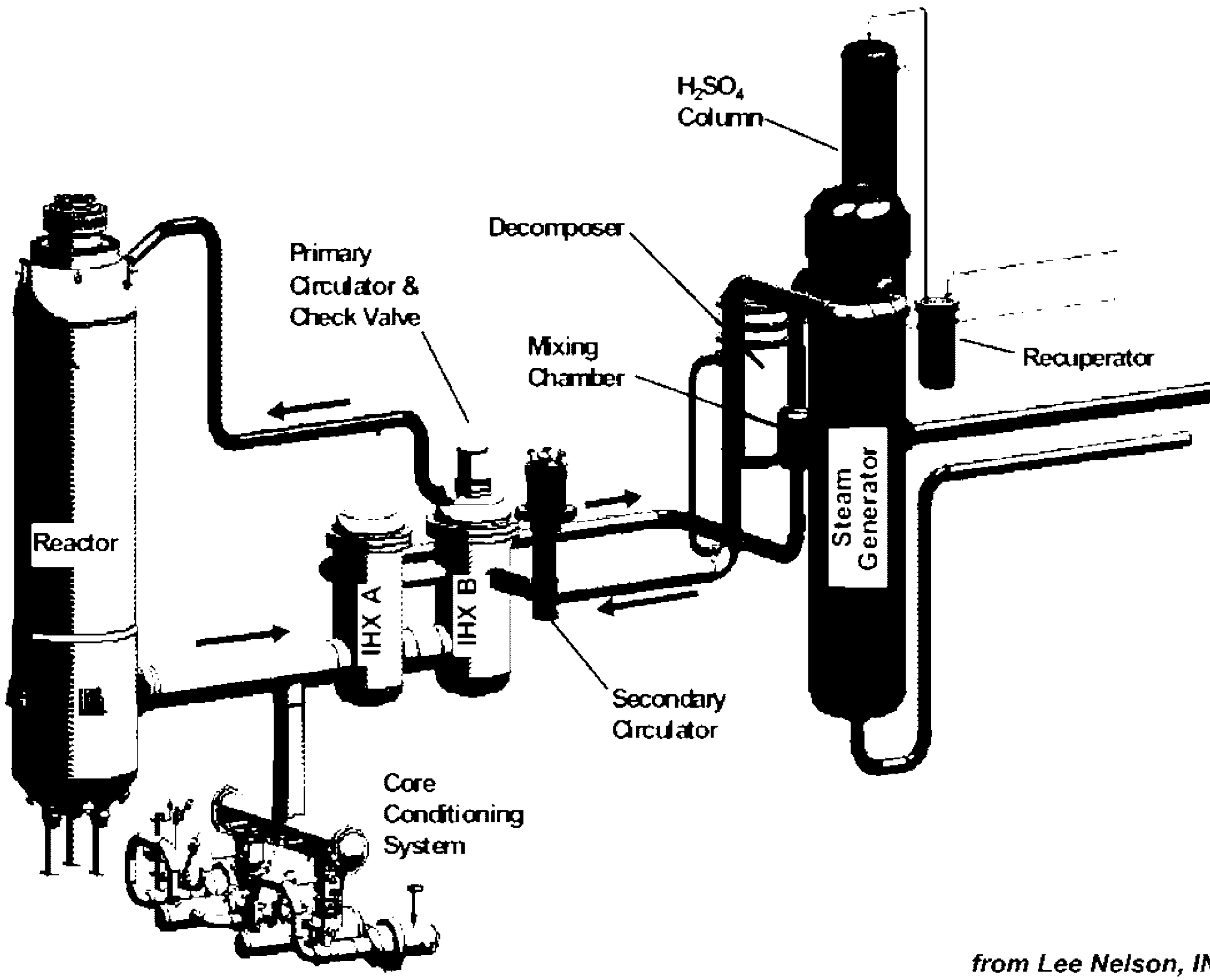
Metallic Materials (GA) Function of ROT (cont)

Component	Temperature Conditions	750C	Mat'l Selection	850C	Mat'l Selection	950C	Mat'l Selection
Hot Duct T/B	Normal Ops PCCD Max DCCD Max	749 786 820 749 749	Hastelloy X	848 837 923 848 848	Hastelloy X	**948** **986** **1022** **948** **948**	**C-C or SiC-SiC**
LPS T/B	Normal Operation PCCD Max DCCD Max	670 707 742 670 670	800H	752 791 826 752 752	Hastelloy X	833 871 907 833 833	Hastelloy X
SCS Entrance Tubes	Normal Operation PCCD Max DCCD Max	653 690 724 653 653	800H	729 768 804 729 729	Hastelloy X	806 844 880 806 806	Hastelloy X
SCS T/B	Normal Operation PCCD max DCCD Max	350 350 350	800H	350 350 350	800H	350 350 350	800H

LPS = Lower Plenum Shroud
SCS = Shutdown Cooling System

from Lee Nelson, INL

Layout of the Pebble Bed Reactor Unit Included SG and IHX(s) for Electricity and H_2 Generation



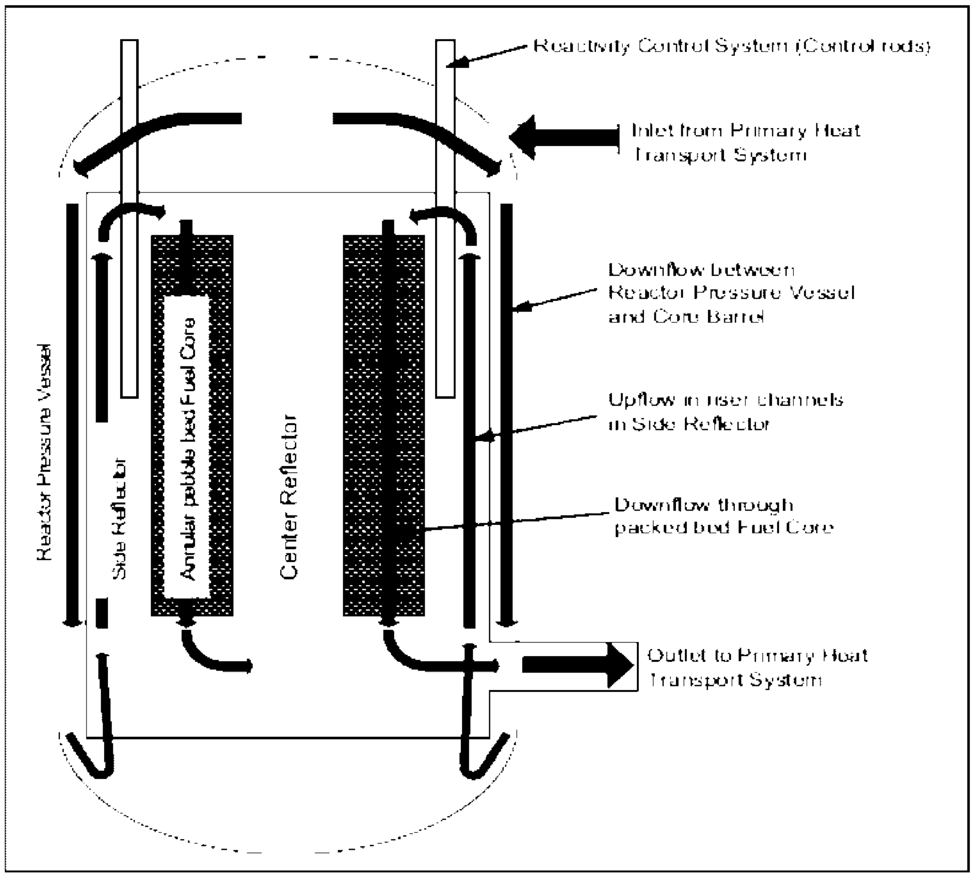
from Lee Nelson, INL

Helium Temperatures in Pebble Bed NGNP Piping Sections (950°C)

Piping Location	Temperature (°C)*
Primary Heat Transport System	
RPV to IHX A	950
IHX A to IHX B	760
IHX B to Circulator	337
Circulator to RPV	350
Secondary Heat Transport System	
IHX A to PCHX	900
IHX A to Mixing Chamber	900
Mixing Chamber to SG	840
PCHX to Mixing Chamber	659
SG to Circulator	273
Circulator to IHX B	287
IHX B to IHX A	700
*Initial studies indicate that transient temperatures are only very slightly higher than these.	

from Lee Nelson, INL

Helium Flow Path Configuration through the Pebble Bed Reactor



from Lee Nelson, INL

Comparison of PBMR DPP and NGNP Key Operational Parameters

Parameter	Normal Operation		DLOFC		PLOFC[b,c]	
	NGNP	DPP	NGNP[d]	DPP[b]	NGNP	DPP
RIT (°C)	350	500	-	-	-	-
ROT (°C)	950	900	-	-	-	-
Tmax, CB (°C)	350[d]	414[b]	466-634	579 (48h)	565	482
Tmax, RPV (°C)	308[d]	324[b]	328-452	419 (56h)	401 (56h)	373 (48h)
He Mass Flow ($kg.s^{-1}$)	160	192	-	-	-	-
Thermal Power (MW)	500	400	-	-	-	-

[a] 25% increase in power level, hence 25% higher flux level assumed for NGNP compared to DPP

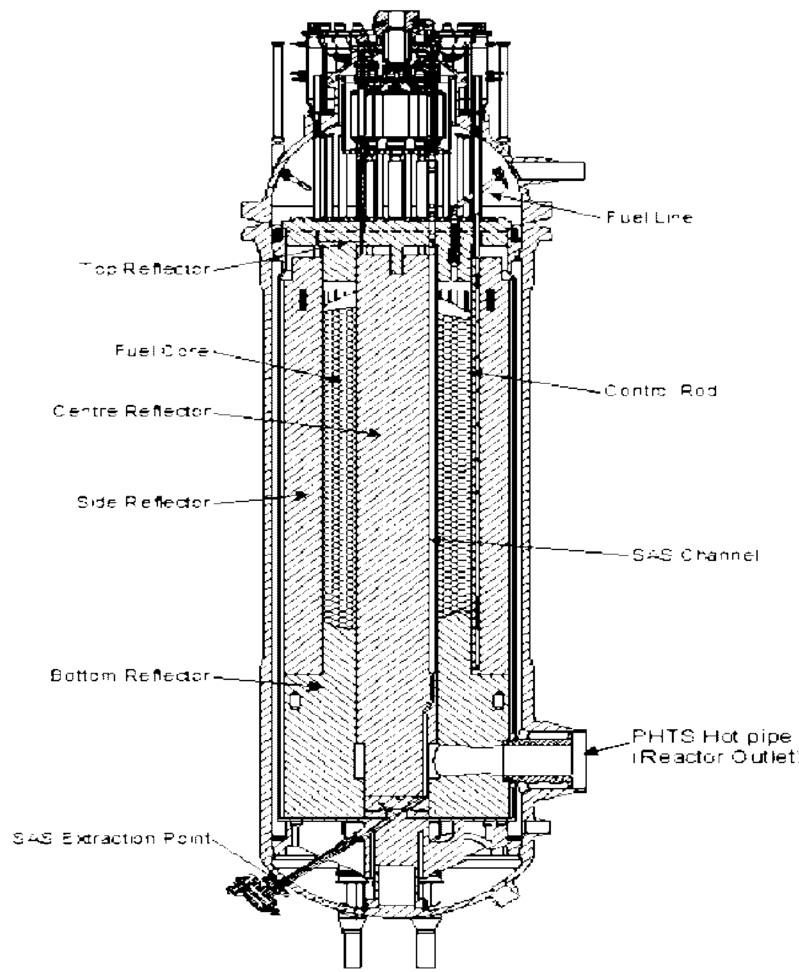
[b] Based on Case 5, NGNP Special Study 20 2: Prototype Power Level Study, NGNP-20-RPT-002, 26-01-07 [1]

[c] Indirect cycle NGNP design, hence operating pressure in system assumed to remain constant at 9 MPa

[d] Reactor Parametric NGNP Special Study, NGNP-NHS 90 PAR, August 2008

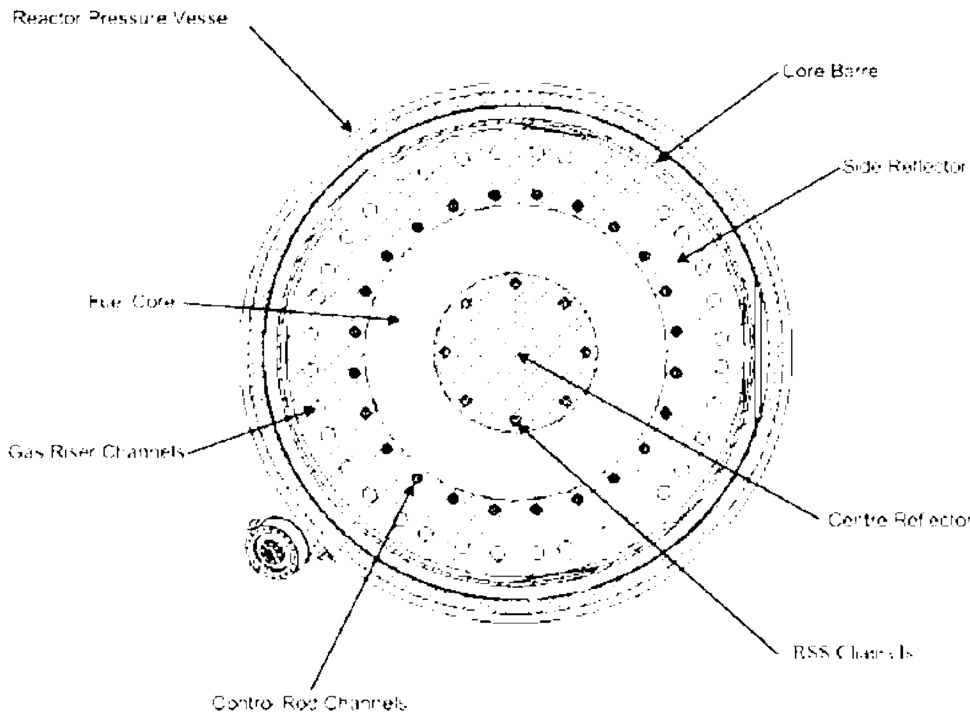
from Lee Nelson, INL

Vertical Schematic Section through the Reactor Unit - PBR



from Lee Nelson, INL

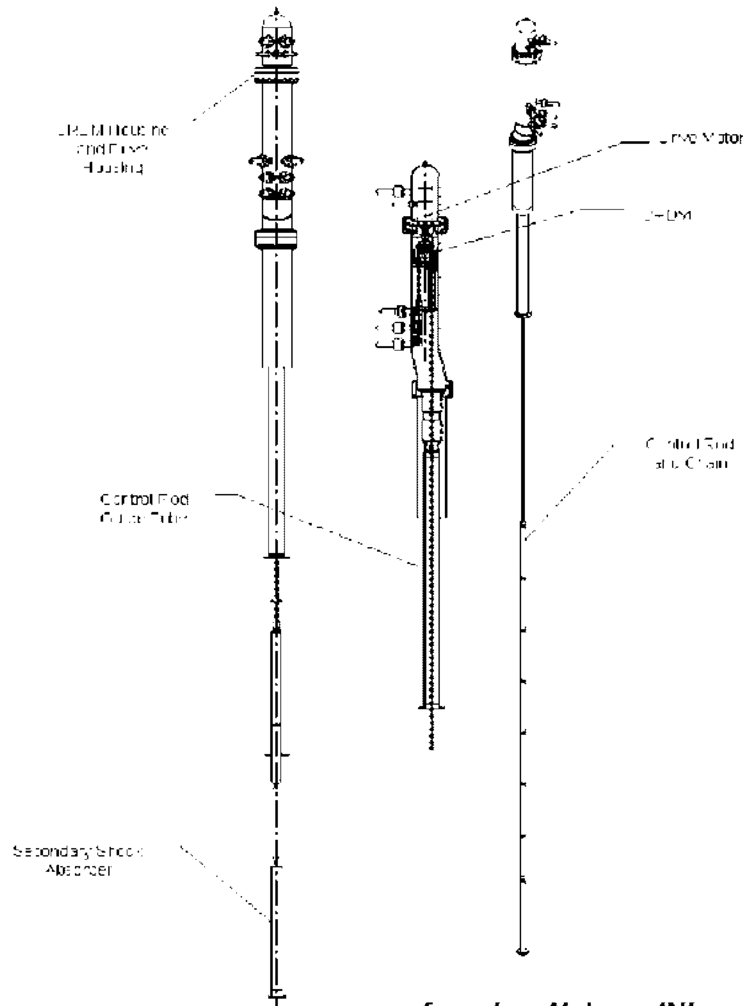
Horizontal Section through the RU without the Core Inlet and Outlet Pipes



from Lee Nelson, INL

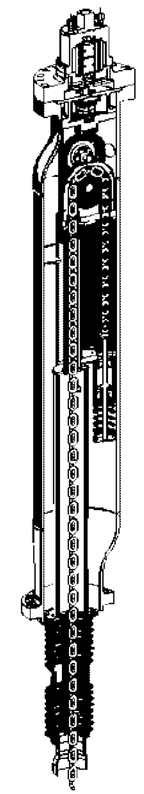
Control Rods for the PBMR NGNP

- Reactivity Control System (RCS) used to control reactivity in the core, to quickly shut the reactor down and to keep it in a shutdown mode
- RCS consists of 24 identical control rods.
  - Control rods are one group of 12
  - Shutdown rods are one group of 12
- Each rod has six segments containing absorber material (sintered B_4C rings between two coaxial cladding tubes separated by joints)
- DPP design uses Alloy 800H for cladding and joints
- Chain lowers and raises the control rods through segmented graphite liners in the Inner Side Reflector (ISR)
- During SCRAM event, control rods drop by gravity but are limited by the characteristics of the drive and the shock is absorbed by secondary shock absorber



from Lee Nelson, INL

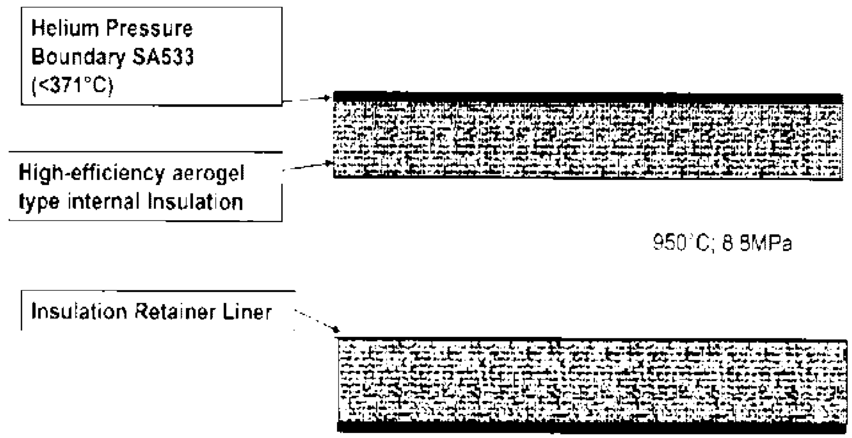
Tentative PCDR Reactivity Control System Metallic Materials for the PBMR NGNP



Component	Materials	Applicable ASME Design Code	Qualification Approach
RCS Chain	SB-446 Alloy 625	Not applicable	Design by analysis, supported by appropriate test data
RCS Absorber Cladding Tubes	SB-407 Alloy 800H	Not applicable	Design by analysis, based on KTA 3221, supported by appropriate test data
RCS Shock Absorber	SB-408 Alloy 800H	Not applicable	Design by analysis, based on KTA 3221, supported by appropriate test data
RCS Guide Tubes	SA-182 F316H SA-312 Gr 316H	Section III, Subsection NG (Tubes)	Use NRC-accepted ASME Specification + EJEMA8 (Bellows)

from Lee Nelson, INL

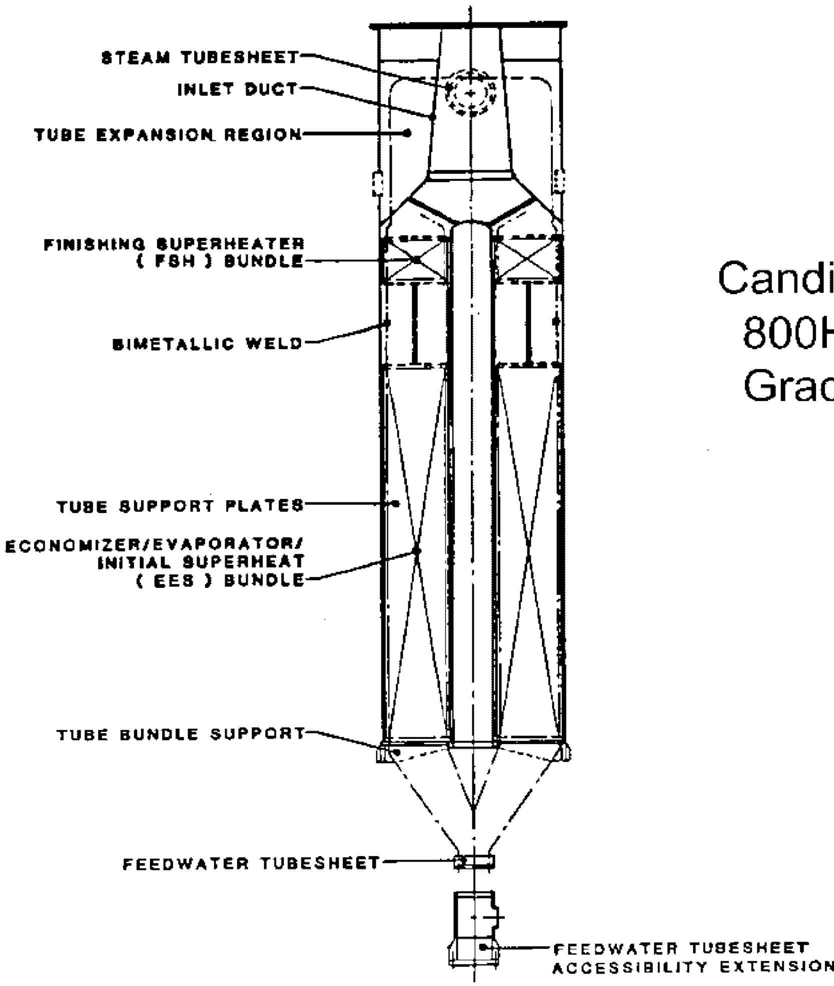
Hot Gas Duct System Materials for the PBMR DPP



Component	Material	Applicable ASME Design Code	Qualification Approach
Core Outlet/CCS Inlet Pressure Pipe	SA-672 Grade J90 (Made from SA-533 Type B, Cl 2 plate)	Section III, Subsection NC	Use NRC-accepted ASME Specification
Core Outlet/CCS Inlet Liner (950 C)	SB-409 Alloy 800H	Not applicable	Design by analysis, based on KTA 3221, supported by appropriate test data
Insulation	Al_2O_3 and SiO_2 (Saffil)	Not applicable	TBD

from Lee Nelson, INL

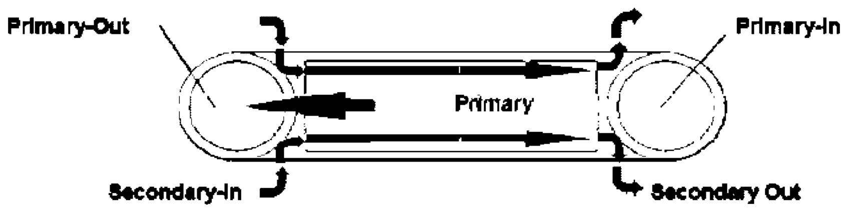
Steam Generator



Candidate Materials Include:
- 800H and 2 ¼ Cr/1 Mo (most likely)
- Grade 91, 617, Hastelloy 617

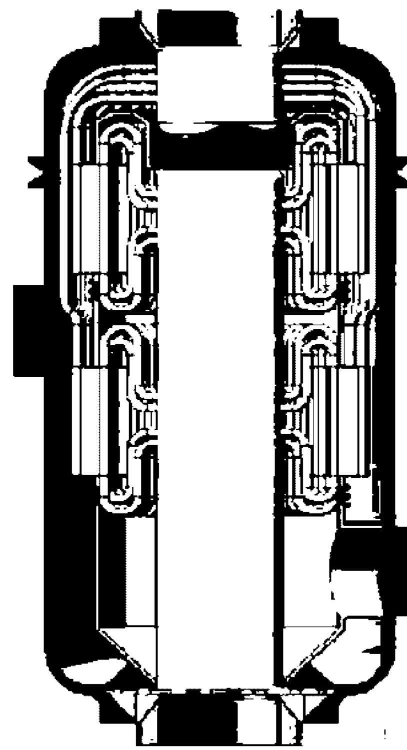
from Lee Nelson, INL

Compact Heat Exchanger



Unit Cell – Plate-Fin Compact IHX

Candidate Materials:
- 230
- 800H
- 617
- Hastelloy X



Concept of Compact IHX

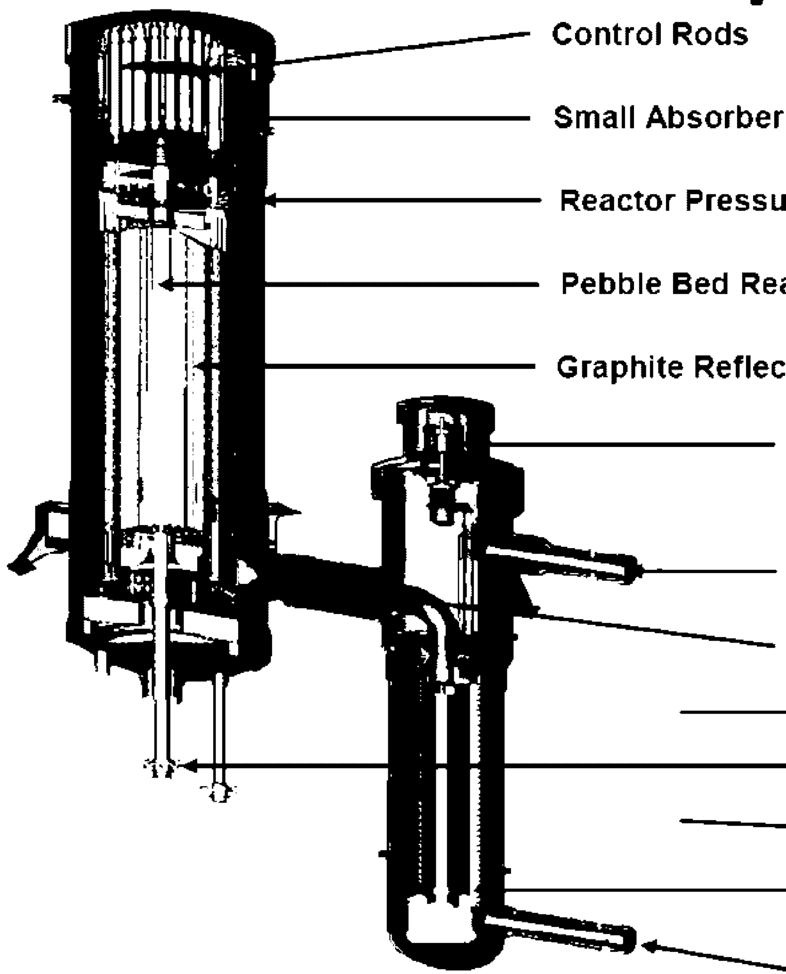
from Lee Nelson, INL

By Sept. 2009, Reference Configurations Featured Much Lower ROTs and Included SGs, not Primary-Secondary IHXs

Item	Tentative Operating Conditions and Plant Configuration [to be finalized during Conceptual Design]		
	Westinghouse	AREVA	General Atomics
Power Level, MWth	200	625	350-600
Reactor Outlet Temperature, °C	750°C	750°C	750°C
Reactor Inlet Temperature, °C	280°C	325°C	322°C
Cycle Configuration	Steam Generator in primary loop	Steam Generator in primary loop	Steam Generator in primary loop
Secondary Fluid	He	NA	NA
Reactor Core Design	Pebble Bed (cylindrical)	Prismatic	Prismatic

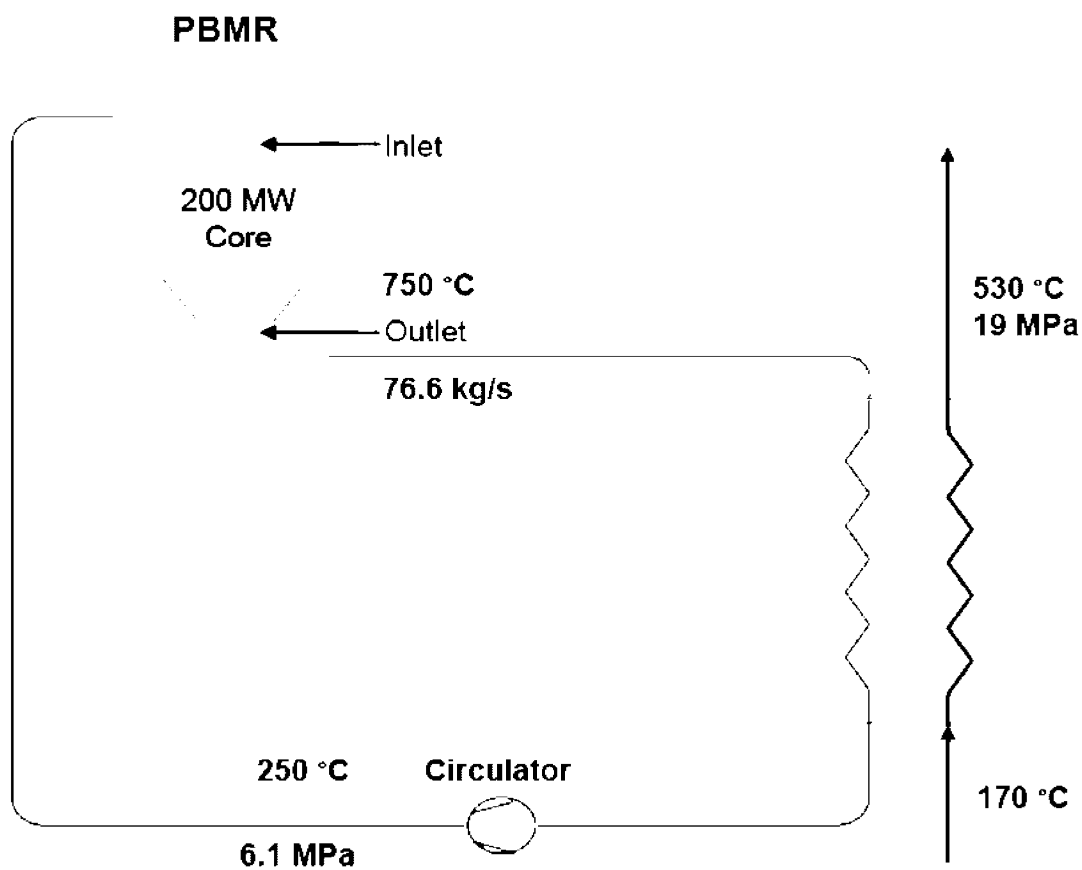
from Lee Nelson, INL

NHSS Layout – Builds on German HTR Modul Experience

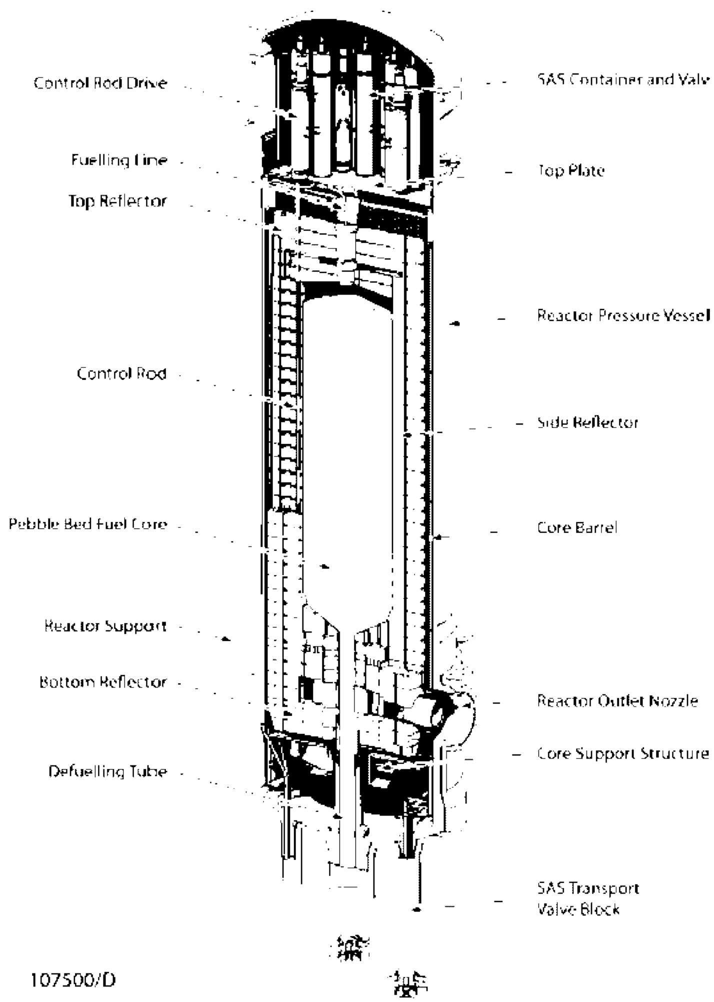


from Sten Caspersson, Westinghouse Electric Co., PBMR Conceptual Design, VHTR R&D FY10 Technical Review Meeting, Denver, Apr 2010

NHSS Primary Loop Heat and Mass Balance



from Sten Caspersson, Westinghouse Electric Co., PBMR Conceptual Design, VHTR R&D FY10 Technical Review Meeting, Denver, Apr 2010

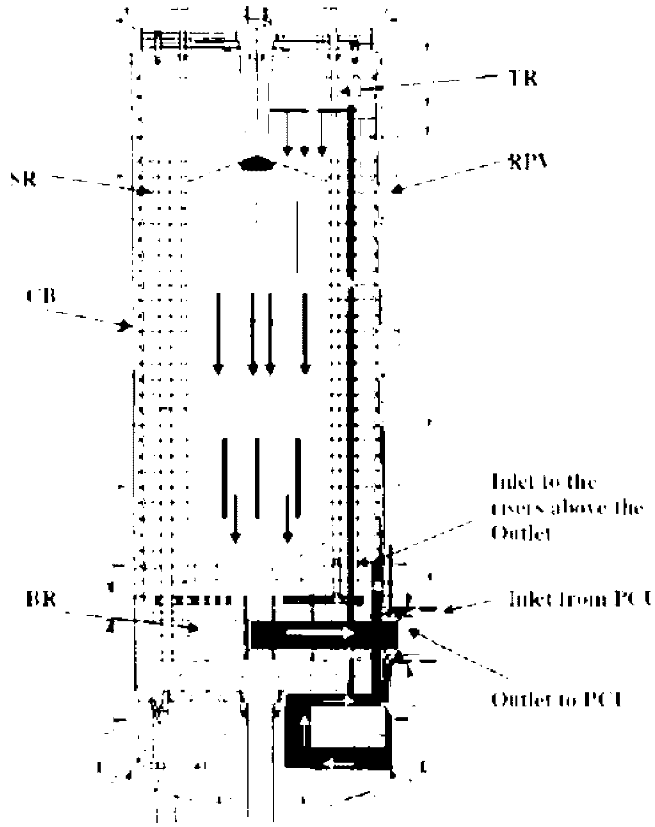


Pebble Bed Reactor Features

- Passive Safety Features
  - **Ceramic coated-particle fuel**
    - **Maintains integrity during loss-of-coolant accident**
  - **Ceramic core with high heat capacity**
    - **High temperature structural integrity**
    - **Slow thermal response times**
  - **Passive heat transfer path**
    - **Limits fuel temperature during loss-of-coolant accident**
  - **Low power density**
  - **Inert Helium Coolant**
  - **Negative Temperature Coefficient**
  - **Two diverse shutdown systems**
    - **Inserts under gravity when power is lost**
- Operating Features
  - **On-line refueling**
    - **No refueling outages**
  - **High gas temperature**
    - **Efficient power conversion cycles**
    - **Process heat applications**

from Sten Caspersson, Westinghouse Electric Co., PBMR Conceptual Design, VHTR R&D FY10 Technical Review Meeting, Denver, Apr 2010

Coolant Flow Design

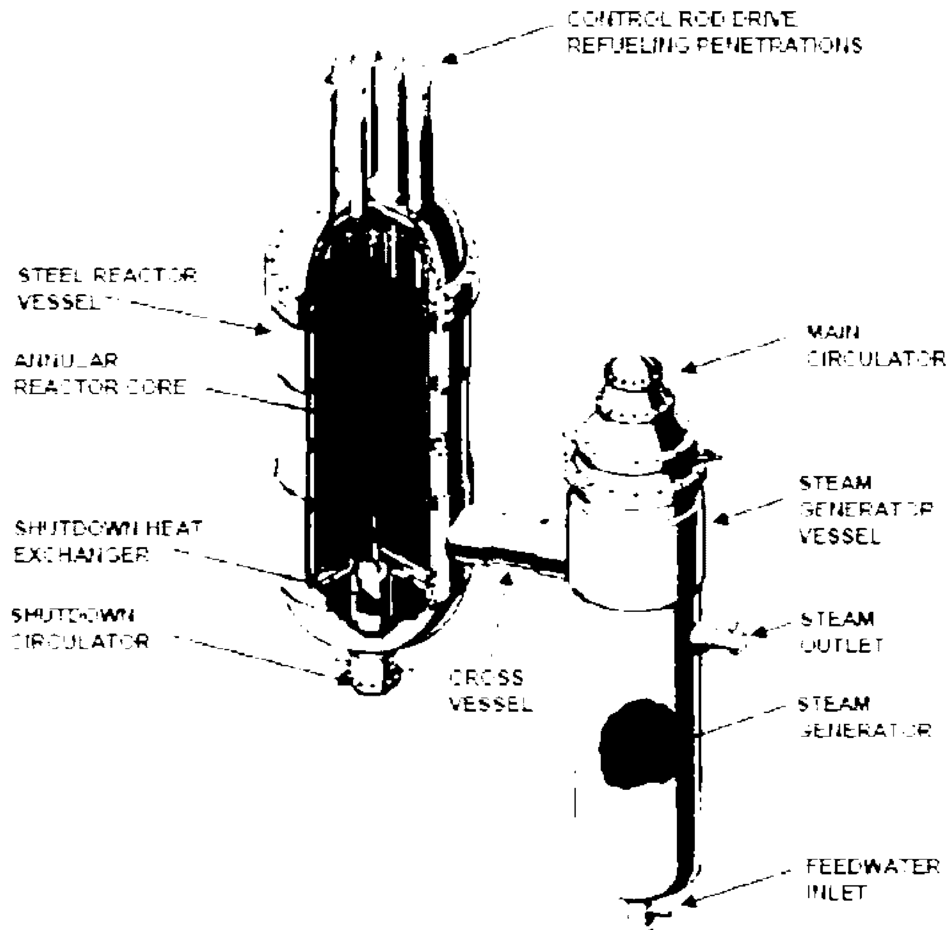


- The coolant flow path design needs to consider the following aspects:
  - cool the metallic structures where necessary
  - reduce bypass flows
  - provide a uniform temperature distribution
  - mix the bypass flows to lower the thermal stratification in the outlet gas

from Sten Caspersson, Westinghouse Electric Co., PBMR Conceptual Design, VHTR R&D FY10 Technical Review Meeting, Denver, Apr 2010

Prismatic NGNP Primary System

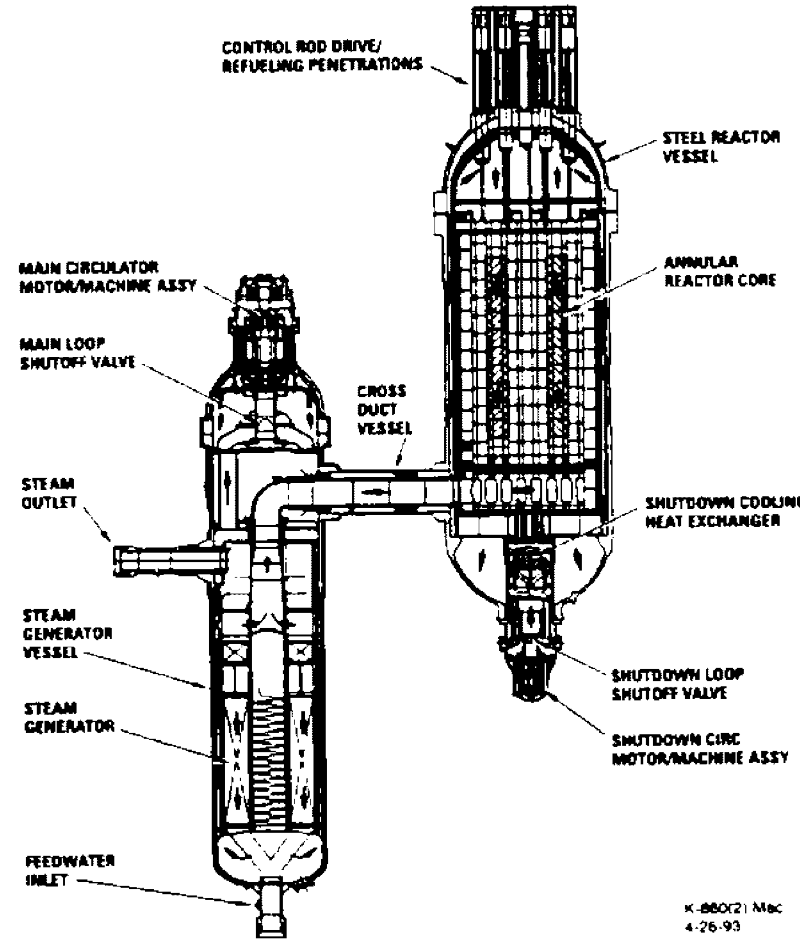
- Modular Nuclear Heat Supply System (NHS)
  - 350 MWt annular core
  - Triso coated particle fuel
  - Helium cooled
  - Graphite moderated
  - 750°C core outlet temp
  - 540° C/17.3 MPa steam heat
- NHS Contained in 2 steel vessels, RV and SGV, interconnected by a cross vessel



from Arkal Shenoy, General Atomics, Prismatic Conceptual Design, VHTR R&D FY10 Technical Review Meeting, Denver, Apr 2010

Main NHS Systems, Subsystems & Components

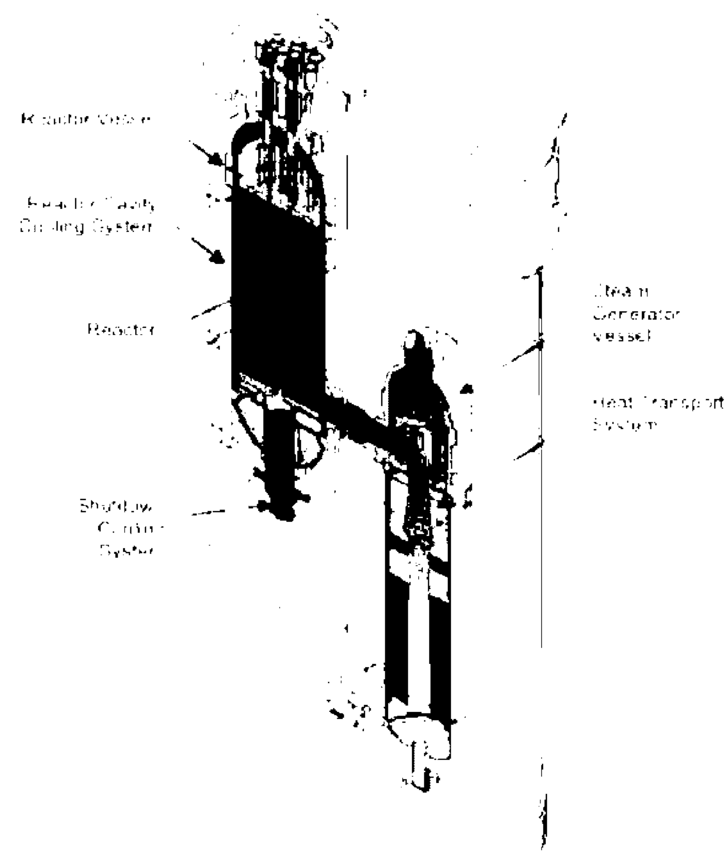
- Reactor System (RS) - Fuel, graphite, CRs, CRD mechanisms, metallic internals, insulation
- Vessel System (VS) – RV, SGV, XV, supports, pressure relief
- Heat Transport System (HTS) – SG, main circulator, hot duct
- Shutdown Cooing System (SCS) – SCS circulator, SCS heat exchanger
- Helium Service System (HSS) - He Purification, transport & storage
- Fuel Handling and Storage System (FHSS) - Refueling machine, transport cask, cask transporter
- Reactor Cavity Cooling System (RCCS)



from Arkal Shenoy, General Atomics, Prismatic Conceptual Design, VHTR R&D FY10 Technical Review Meeting, Denver, Apr 2010

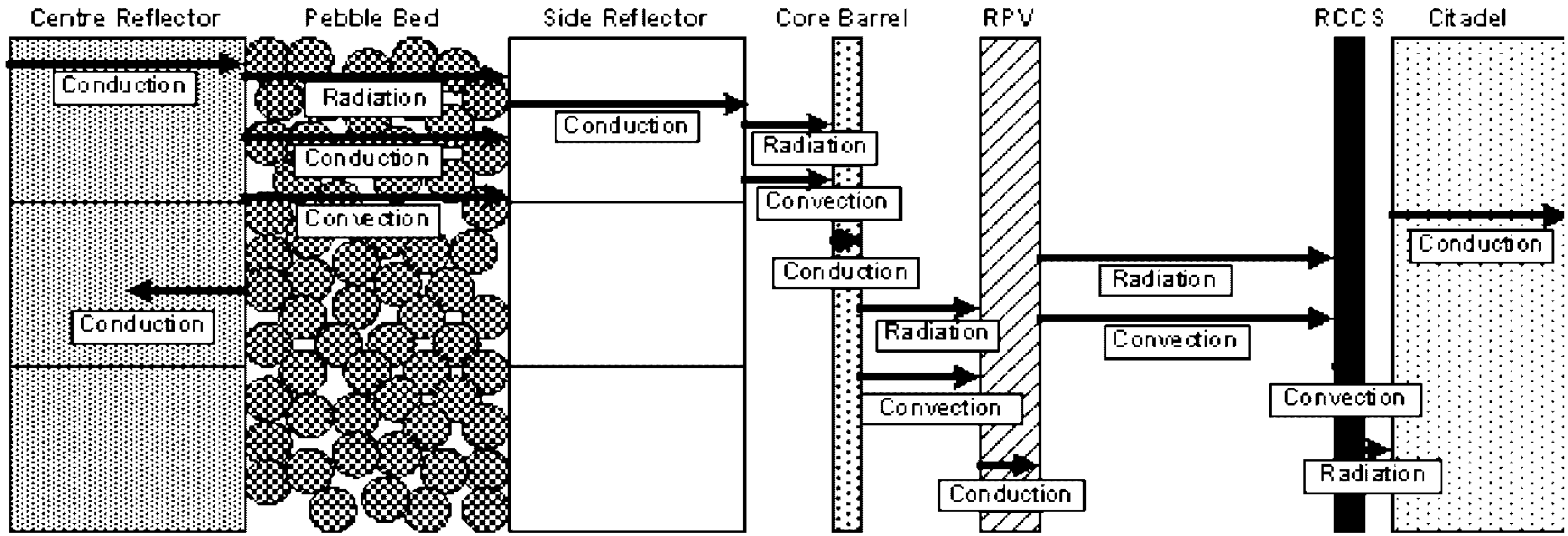
Prismatic NHS Contained in Below Grade Silo

- Cylindrical silo with 2 main cavities:
  - Reactor cavity
  - Steam generator cavity
- Silo depth based on placing SG thermal center well below core
- Main advantages of below grade silo:
  - Sabotage/damage resistant
  - Increased safety approach
    - Decay heat can be dissipated to earth in the event of RCCS failure
    - More resistant to damage from seismic events
  - Improved economics relative to above-grade installations



from Arkal Shenoy, General Atomics, Prismatic Conceptual Design, VHTR R&D FY10 Technical Review Meeting, Denver, Apr 2010

Passive Heat Transfer Path Description



from Sten Caspersson, Westinghouse Electric Co., PBMR Conceptual Design, VHTR R&D FY10 Technical Review Meeting, Denver, Apr 2010

Prismatic NGNP Key Design Selections for CD

NGNP mission	Co-generation of process steam and electricity
Reactor type	Modular Helium Reactor (1 module prototype)
Exclusion Area Boundary	425 m (commercial site requirement)
Off-site accident dose limits	PAGs (1 rem whole body; 5 rem thyroid)
Occupational exposure limits	10% of 10CFR20
Reactor power	350 MW(t)
Reactor pressure vessel material	SA 508/SA 533 steel (LWR vessel material)
Core	Prismatic core
Fuel particles	LEU UCO (Single or Multiple enrichment)
Fuel compact matrix	Thermosetting Resin matrix
Fuel block	10-row block (same as used in FSV)
Graphite grade – fuel elements	Near-isotropic, nuclear grade (having properties similar to H-451
Graphite grade – replaceable reflectors	TBD

from Arkal Shenoy, General Atomics, Prismatic Conceptual Design, VHTR R&D FY10 Technical Review Meeting, Denver, Apr 2010

Key Design Selections for Tech Dev Identification (cont)

Graphite grade – permanent coré structures	TBD
Number of primary coolant loops	Single primary loop, single secondary
Primary coolant	Helium
Hot duct material	Alloy 800H
Core inlet helium temperature	322° C
Core outlet helium temperature	750 ° C
Energy transfer system	Primary to secondary transfer via steam generator in primary
Secondary working fluid	Water/steam
Steam generator inlet/outlet conditions	200° C, 19 MPa / 540° C, 17.3 MPa
Electricity generation	In secondary system via Rankine Cycle
Process steam generation	In tertiary system via steam-to-steam heat exchanger
Reactor Cavity Cooling System	Air-cooled RCCS
Containment	Vented Low-Pressure Containment

from Arkal Shenoy, General Atomics, Prismatic Conceptual Design, VHTR R&D FY10 Technical Review Meeting, Denver, Apr 2010

(V)HTRs Require Additional Qualification and/or Development of High-Temperature Materials

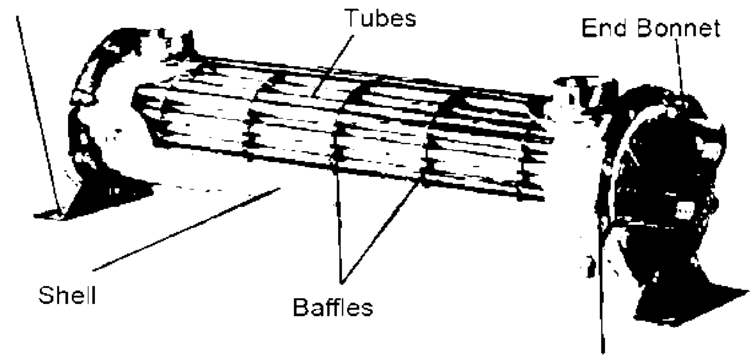
- Designs for near-term deployment include He-cooled reactors with outlet temperature of 750 to 800°C in service for 60 years
- Outlet temps for advanced VHTRs may exceed 950°C
- Primary challenges for VHTR structural materials are irradiation-induced and/or time-dependent failure and microstructural instability in the operating environments
- Additional materials issues related to fabrication, codes and standards, modeling, and design methods must be addressed
- Useful to consider structural materials in four categories
  - Graphite (e.g., core support structures, fuel matrix, etc.)
  - Very high temperature metals (e.g., IHX, SG, turbomachinery, etc.)
  - Medium high temperature metals (e.g., RPV, piping, IHX shell, etc.)
  - Ceramic & composites (e.g., core restraints, control rods, duct liners, etc.)

Metals in Very High Temperature Service Have Major Challenges

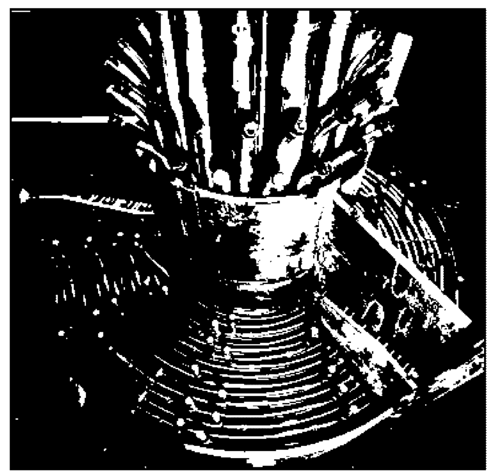
- High-temperature mechanical properties (e.g., tensile, creep, creep fatigue, stress rupture, high and low-cycle fatigue, creep-fatigue crack growth, fracture toughness) in air and impure helium environments
- Environmental degradation processes from exposure to high-temperature helium with contaminants such as CO, CO_2, H_2, H_2O, and CH_4
- High-temperature metallurgical stability (thermal aging effects)
- Long-term irradiation-induced effects on core internals
- Extension of ASME and similar design Code approval for metallic materials for higher VHTR operating temperatures, longer service times, and complex loading conditions
- Validated methodologies for inelastic design analysis

Alloys 617, 230, Incoloy 800H & Hastelloy X(R) Are High Temperature Alloys for IHXs and SGs

- Temperatures up to 950°C
- Expected principal damage mechanism: creep-fatigue
- Only one alloy, 800H, is ASME Code qualified and only to 762°C
- Alloys 617, 230 & X(R) suitable but not in nuclear section of the ASME Code
- 2 1/4Cr-1Mo code qualified for lower SG temperatures
- Dissimilar metal welds remains an issue

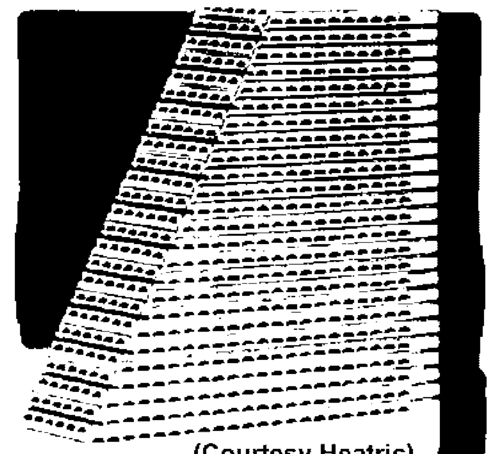


Straight-Tube HX



THTR-300 Steam Generator

Printed Circuit Board Heat Exchanger



(Courtesy Heatric)

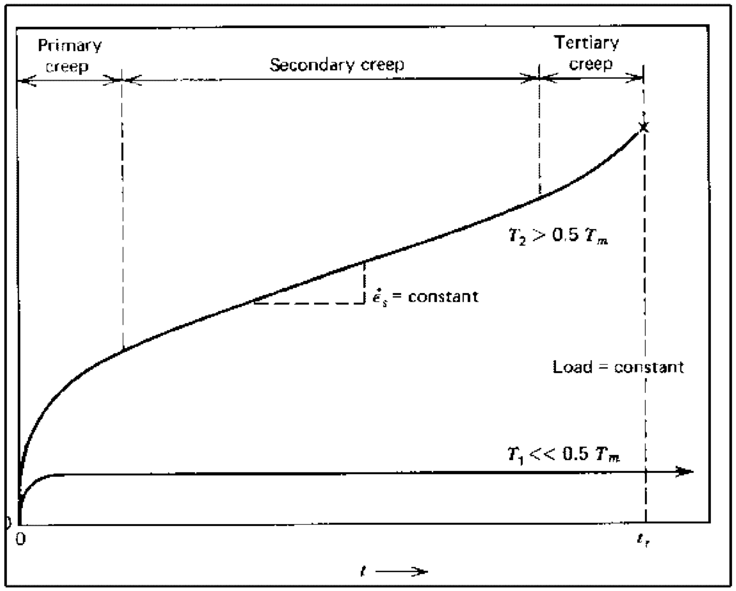
Candidate Materials for IHX & SG Applications Must Have High-Temperature Strength & Corrosion Resistance

Wrought high-Ni creep resistant alloys (20-22 wt% Cr) are creep resistant and offer protection against oxidation up to about 900°C by formation of chromia scale

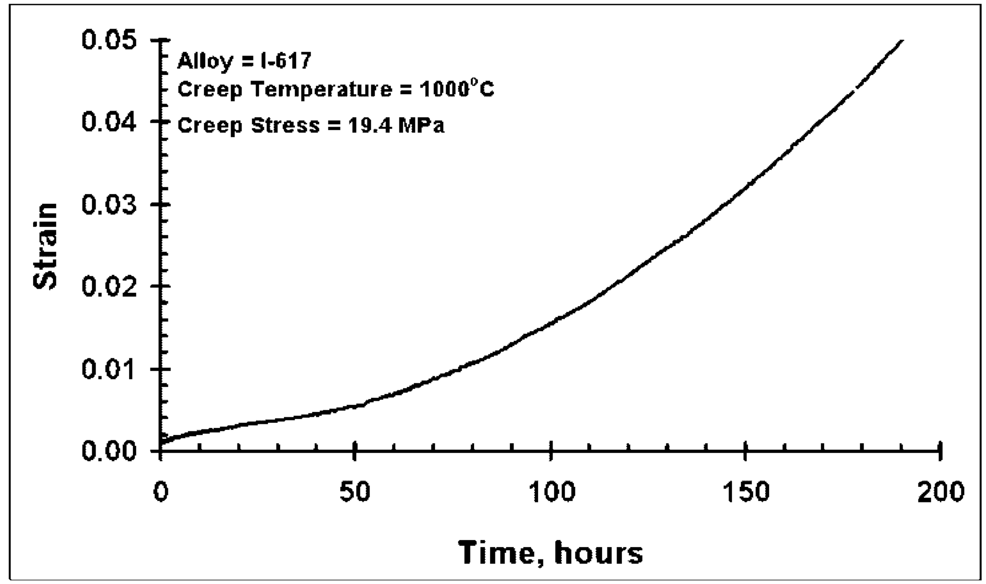
	Ni	Cr	Mn	Co	C	Fe	Ti	Al	W	Si	Mo
Inconel 617	base	22.0	0.40	12.0	0.10	2.0	0.40	1.2	-	0.40	9.0
Haynes 230	base	22.0	0.65	5.0	0.10	3.0	-	0.30	14.0	0.50	2.0
Alloy 800H	32.0	21.0	1.00	-	0.06	bal	0.40	0.40	-	0.60	-
Hastelloy X	base	22.0	1.00	1.50	0.10	18.5	0.15	0.50	0.60	1.00	9.0

None are fully qualified in ASME or similar codes
for VHTR nuclear applications

Need to Model and Codify High-Nickel Alloy Creep Behavior



Typical Metal



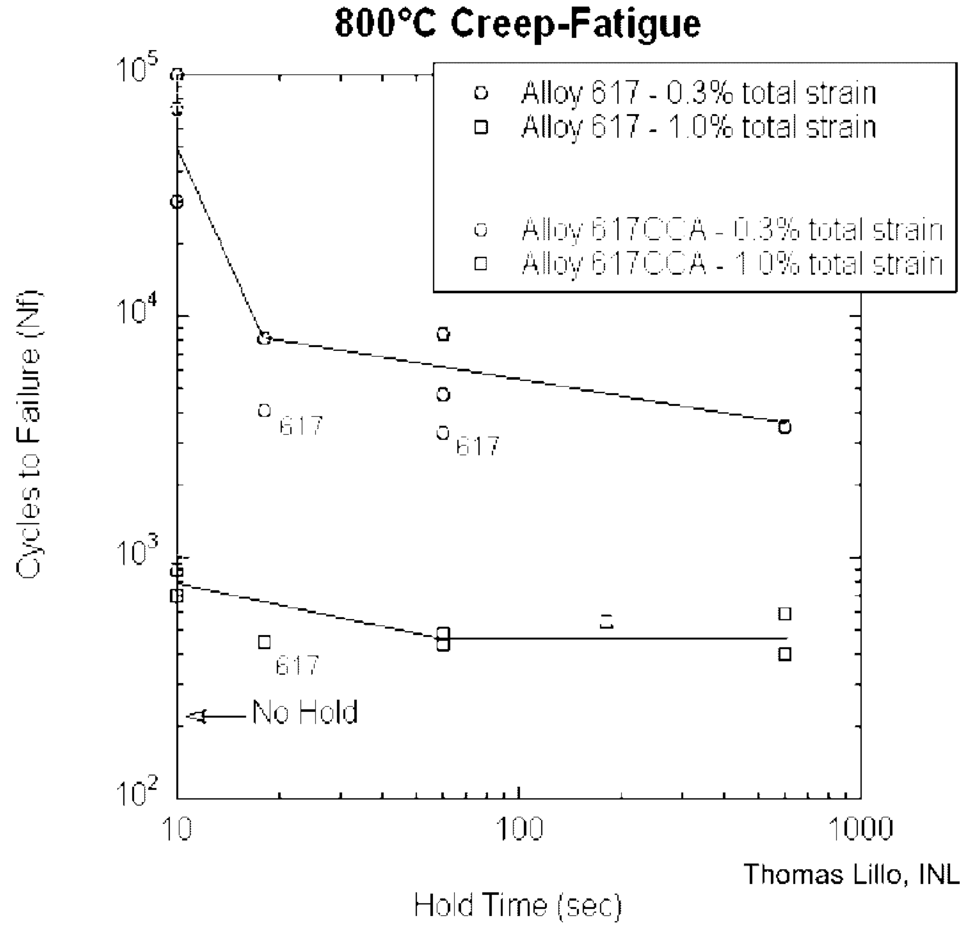
Alloy 617

Thomas Lillo, INL

- **Alloy 617 creep behavior**
  - **Majority of life spent in tertiary creep regime, not in primary and secondary creep.**
  - **Need to determine amount of creep that can be tolerated before load carrying capability is significantly compromised**

Need to Improve and Validate Creep-Fatigue Interaction Models for High-Nickel Alloys

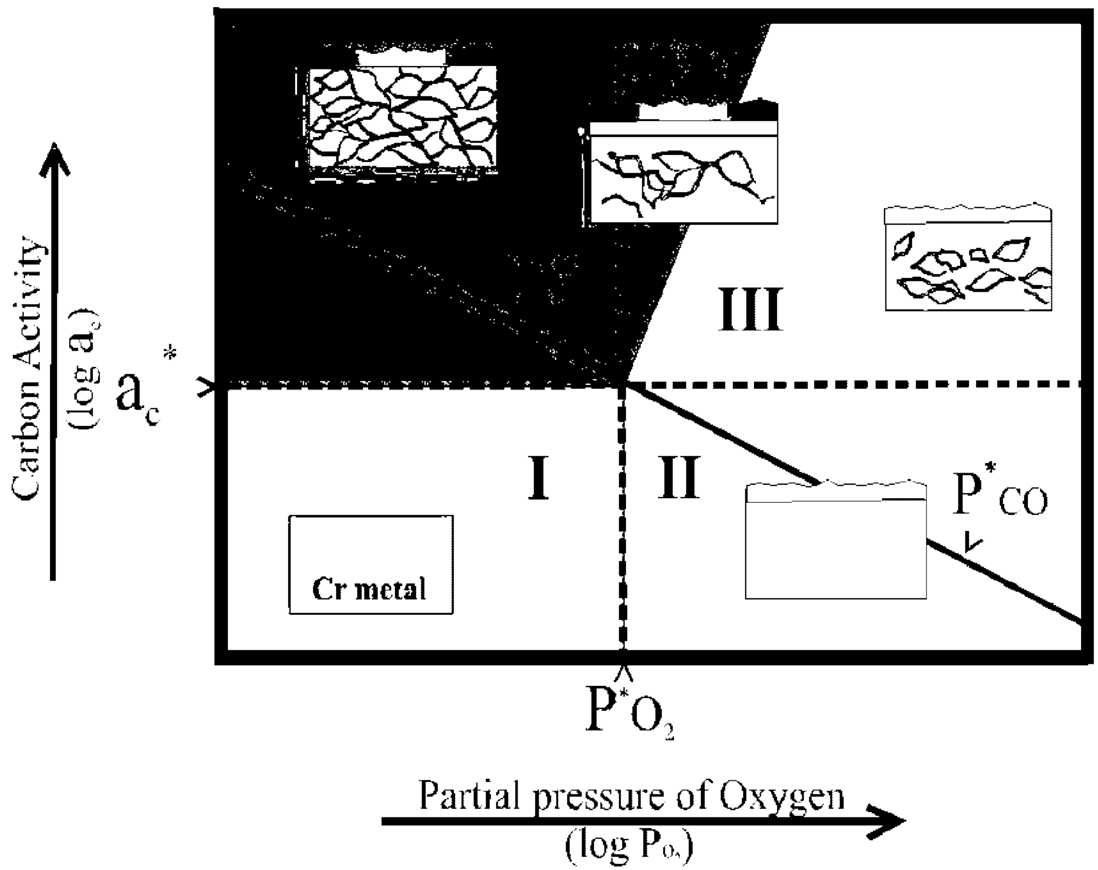
- **Better understanding of operative mechanisms**
  - **Fatigue-dominated regime**
  - **Creep-dominated regime**
- **Effect of hold time: saturation or continuous degradation**
- **Development of improved constitutive models**
- **Verification of creep-fatigue interaction diagram**
- **Incorporation into ASME Code**



Thomas Lillo, INL

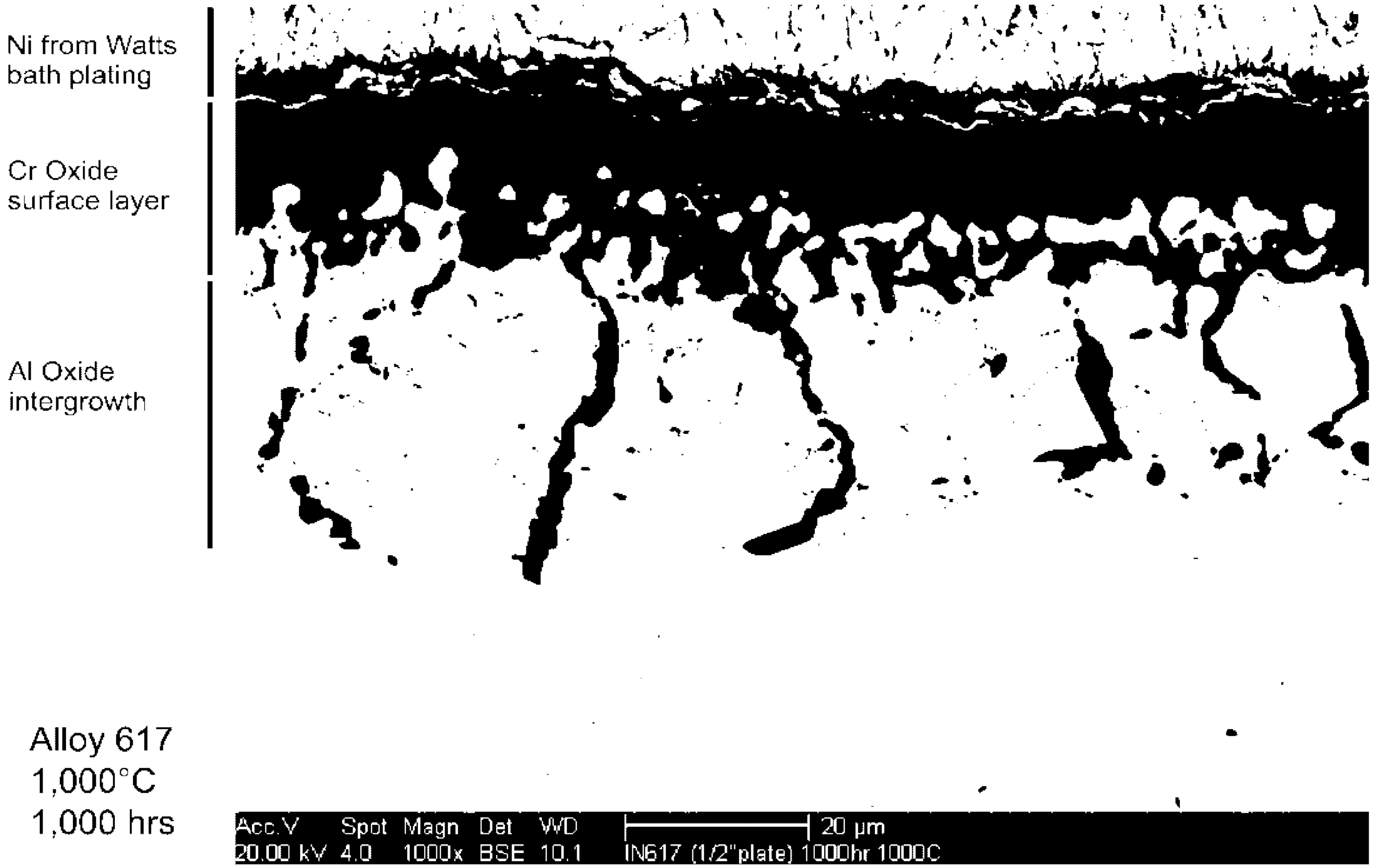
*indicates premature failure

Aging and Environmental Effects Must Be Assessed for VHTRs



- There is no VHTR environment that is inert with respect to alloys
- Environmental-effects maps will help in specification of He impurity content of primary coolant for long-term stability of heat exchangers and steam generators (region II desirable)
- Even without environmental effects, long-term aging results in formation of new phases that can affect mechanical properties

Typical Surface of Alloy 617 Exposed in Air Develops Protective Oxide Layer Containing Islands of Co & Ni

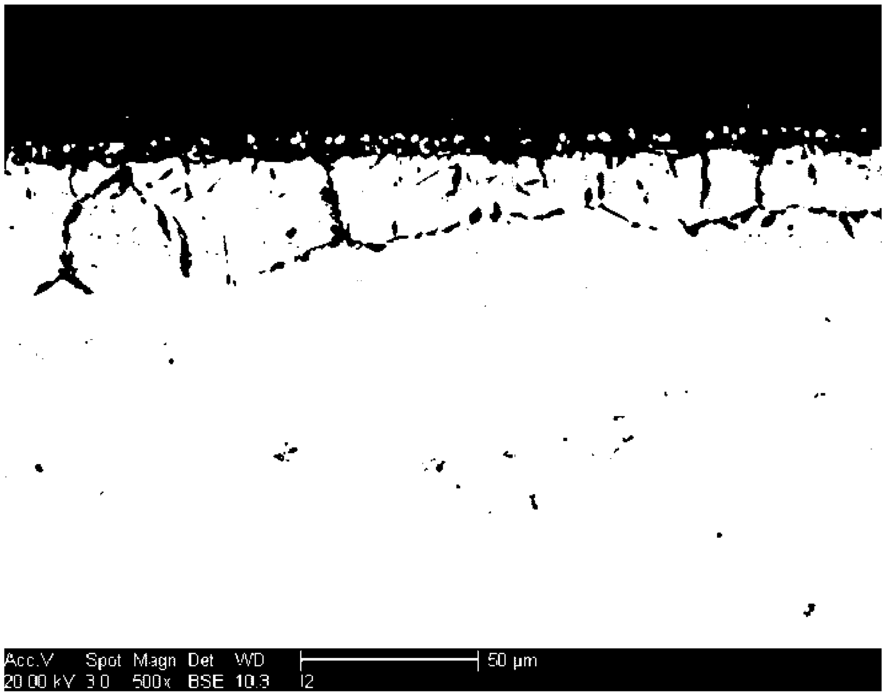


Alloy 617
1,000°C
1,000 hrs

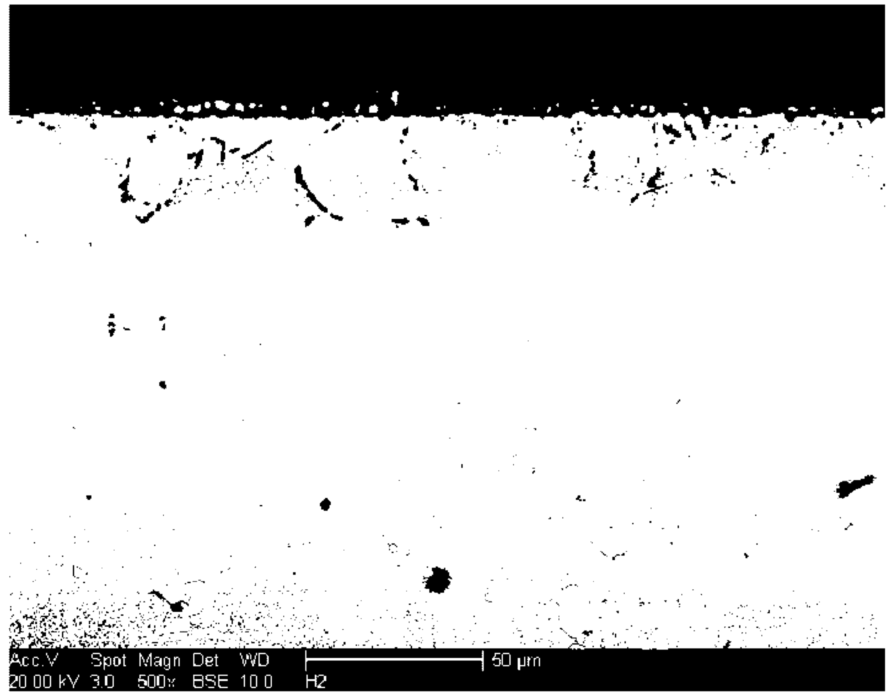
Richard Wright, INL

Alloys Exposed to Oxidizing VHTR He at 1000°C Produce Slow Growing Protective Scales

Alloy 617 Exhibits Generally Similar Behavior as Air Exposure



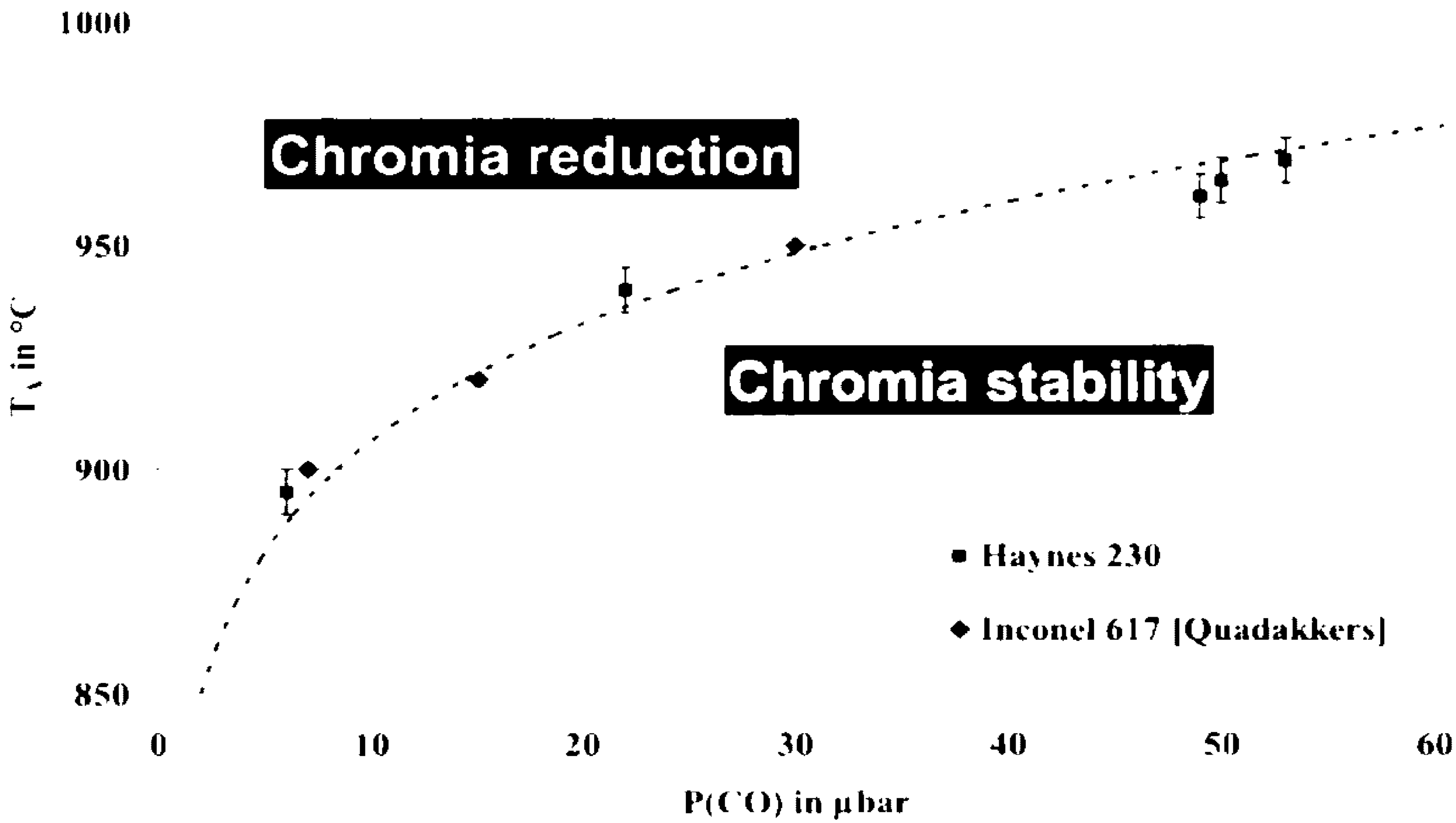
Alloy 230 Forms Thinner Oxide Layer and Less GB Oxidation. Internal WC precipitates visible



Richard Wright, INL

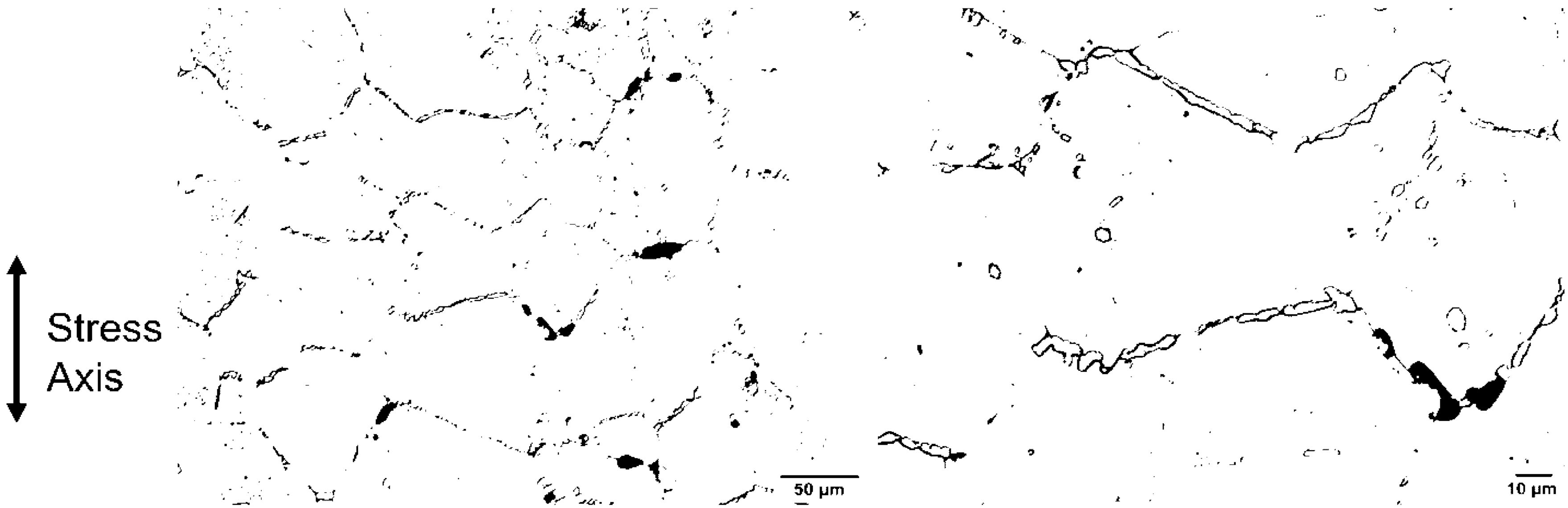
Neither Alloy Is Clearly Superior

Stability of Chromia in VHTR Coolants Is Primarily a Function of P(CO) at Low H_2O



from Rouillard F., Cabet C. et al. Oxid Met 68 (2007) 133
data on Inconel 617 after W. J. Quadakkers, Werkstoffe und Korrosion 36 (1985) 335

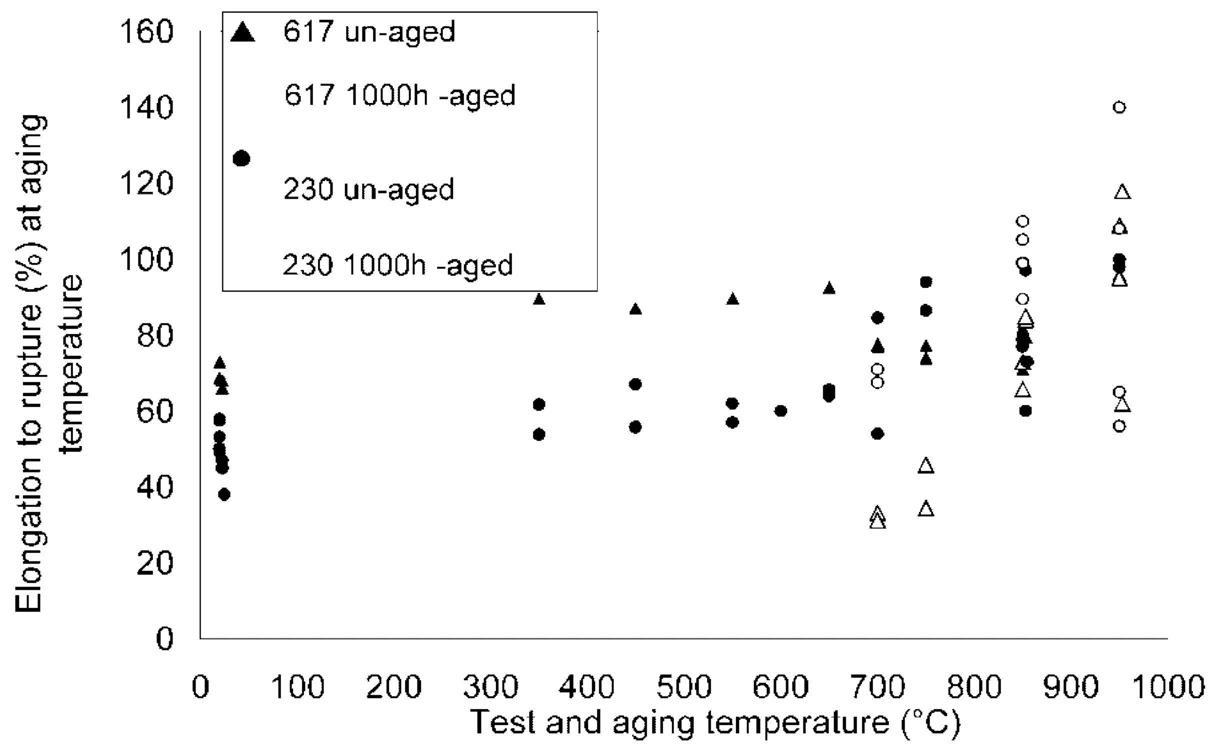
Aging of 617 Results in Microstructural Instability and Loss of Ductility



Richard Wright, INL

- Aging under load results in carbide redistribution and cavitation
- Thermal aging can result in precipitation of additional phases that differ from those under load

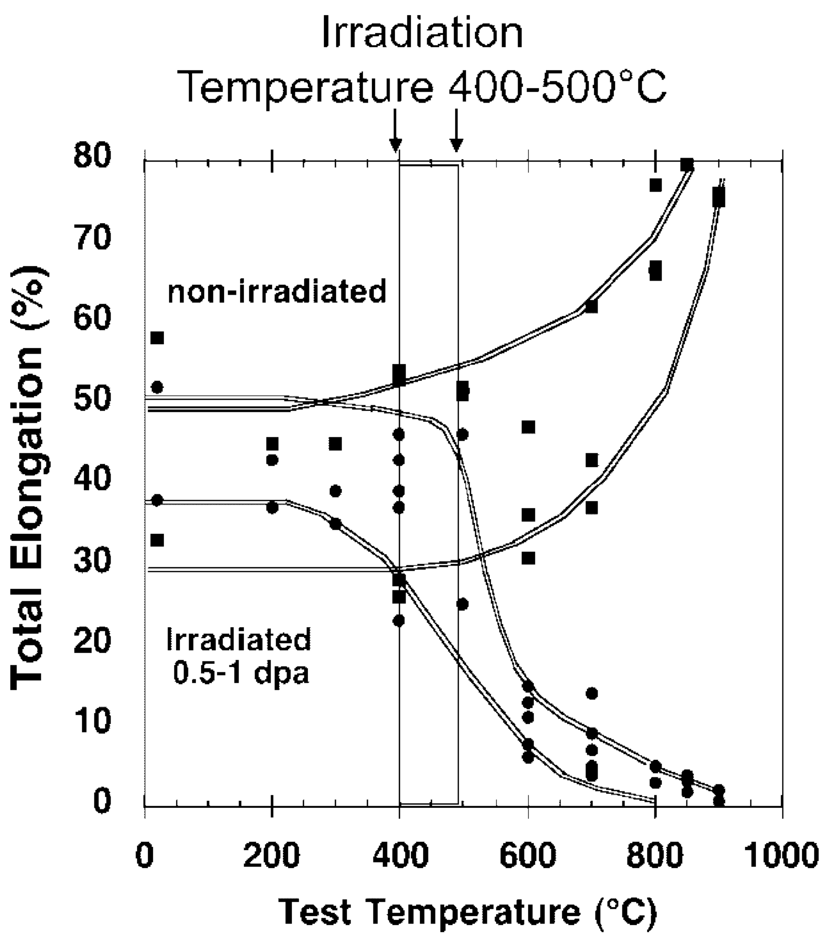
Even with Reductions due to Aging Effects, Tensile Ductility of Aged Alloys Is Still Okay



- **Un-aged material : 617 > 230 up to 700°C and 230 ~ 617 at T > 750°C**
- **1000 hrs-aged 617 : Loss of ductility in the range [700-750°C]**
- **1000 hrs-aged 230 : scattered data at T > 850°C**

Richard Wright, INL, Alloys supplied by CEA

Control Rods Must Also Deal with Irradiation Effects

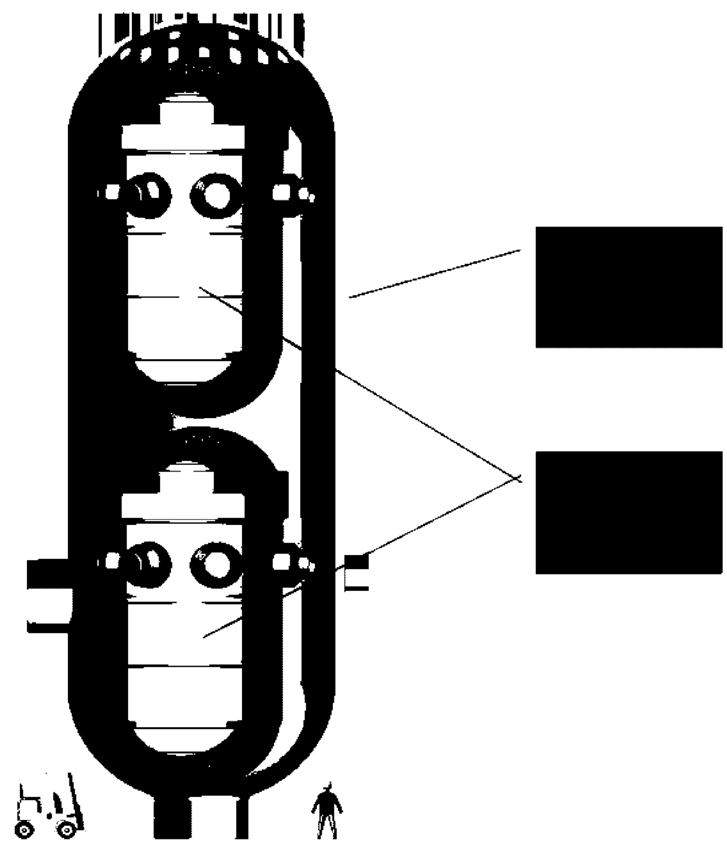


Lance Snead, ORNL

- Irradiation-induced embrittlement is a common feature for high-alloyed heat-resistant materials
- Ni-based alloys are highly susceptible to embrittlement due to helium generation and phase instability during irradiation
- Alloy 800 exhibited the best irradiated ductility in material screening experiments for Japanese HTTR
- However, even Alloy 800 undergoes a major loss of ductility after ≈ 0.5 dpa irradiation at 400-600°C and an order of magnitude loss in creep life

VHTR Pressure Vessels Are the Components of Greatest Concern for Medium-Temperature Metallic Service

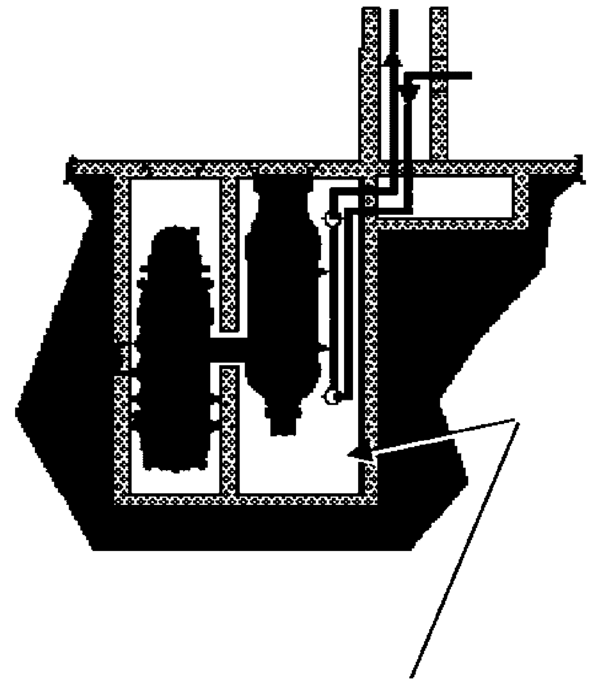
- Normal/off-normal service temperatures and vessel size dominate materials requirements
- With engineered cooling of the vessel, the use of LWR A508/533 steels may be acceptable
  - Limited to <<371°C operation and small, short excursions
  - Assessment still needed for short-time exposures at temperatures approaching or beyond 371°C
- Higher temperature operation of VHTR RPVs requires higher temperature alloys, e.g., 2.25Cr-1Mo(V) or 9Cr-1MoV
  - Very large vessel sizes (up to 30m high x 8-9m diameter) will require scale-up of ring forging & on-site joining technologies, plus Code modifications



Expected Irradiation Dose Low Enough to Avoid Embrittlement but Very Long Term Service May Produce Excessive Creep at Low Temperatures

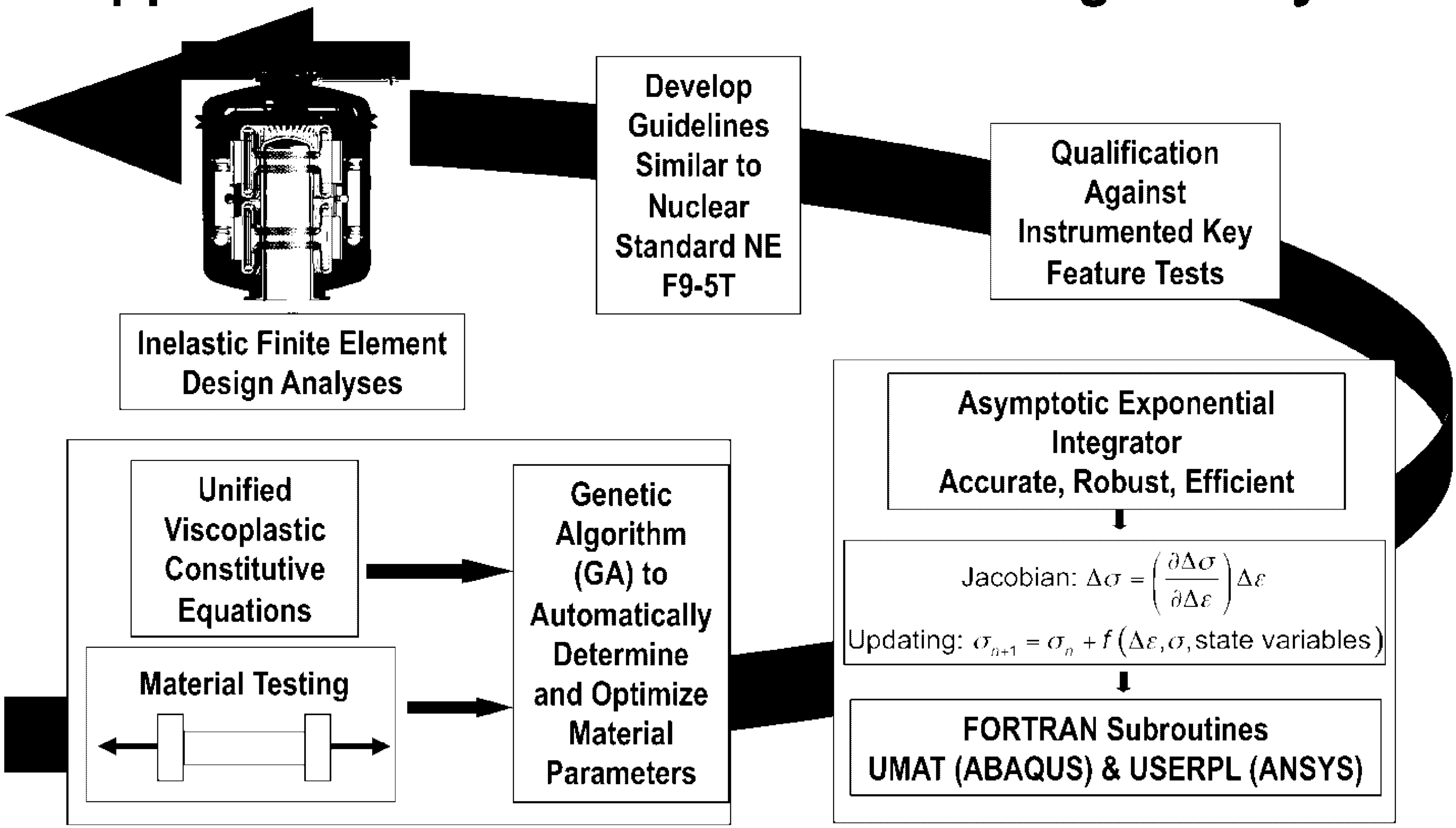
Methods to Ensure High Emissivity of RPV Surfaces Must Be Qualified

- Accident conditions (PLOFC & DLOFC) require high RPV emissivity for passive rejection of decay heat
- Limited studies have shown emissivities from 0.3 for cleaned to 0.9 for oxidized surfaces
- Composition and long-term stability of corrosion products must be evaluated
- Evaluation of core barrel emissivity is also needed
- Additional surface treatments or coating may be needed



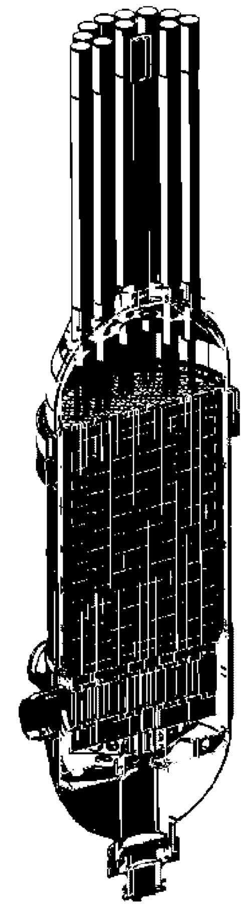
Passive cooling by radiation to water or air panels or ground

Improved Multi-Scale Modeling Is Needed to Support Inelastic Finite Element Design Analyses



GIF Activities Are Underway to Address VHTR Materials Issues

- The Generation IV International Forum is coordinating materials research on graphite, metals, and ceramics & composites performed in seven countries and the EU in direct support of VHTR system developments
- Active programs in China, Japan, Korea, France, Russia, and the U.S. are developing designs and materials requirements for gas cooled reactor systems
- DOE-NE is U.S. participant in VHTR Materials Project Arrangement that has been established to develop and share data among GIF signatories



PIRT Techniques Were Used to Identify Safety-Relevant Phenomena for NGNP*

- Materials degradation phenomena for major components and the materials comprising them were identified
- Phenomena were evaluated for their potential contribution to and pathway for off-site release
- Importance and state of knowledge used to prioritize phenomena
- 58 phenomena identified
  - 17 deemed to have high importance & low/medium state of knowledge on RPV, piping, control rods, internals and valves

*Reference: Next Generation Nuclear Plant Phenomena Identification and Ranking Tables (PIRTs), Volume 4: High-Temperature Materials PIRTs, NUREG/CR-6944, Vol. 4, ORNL/TM-2007/147, Vol. 4, March 2008

Recommended Update on NGNP High Temp Matls PIRT Completed in 2010*

- Considered new, lower temperature versions of NGNP
- Several high-priority phenomena on RPV, piping & HX lowered due to lower ROTs or elimination of HX
- 6 phenomena on SG added and 1 on control rods elevated to high priority; total of 10 high-priority items
- Document produced during IPA & presented to RES staff, but not peer reviewed externally
- PIRT update available for internal NRC use only and contains recommended R&D for phenomena

*Reference: "Recommendation for a High Temp Metals PIRT Update 02-27-10," sent to Shah Malik, NRC-RES, from Bill Corwin, ORNL, via email on March 29, 2010.

High Priority Phenomena in Update on NGNP High Temp Matls PIRT Include:

- Compromised RPV integrity and excessive fuel temperatures caused by inadequate heat transfer from RPV surface due to potential loss of desired surface layer properties and associated reduction of emissivity (#11)
- Breach to secondary system via SG tube failure from initiation & propagation of flaws due to creep crack growth, creep, creep-fatigue, aging (with or without load) & subcritical crack growth (#40a)
- Breach to secondary system via SG tube failure arising from primary boundary design methodology limitations for high-temperature structures (#40b)

High Priority Phenomena in Update on NGNP High Temp Matls PIRT Include: (cont)

- Breach to secondary system via SG tube failure from materials degradation from long-term aging (#40c)
- Breach to secondary system via tube failure from undetected initiation & propagation of flaws due to inadequacy of ISI for high-temperature SGs (#40d)
- Breach of primary to secondary system boundary resulting in water ingress & attack on graphite core structures due to higher secondary system pressures in SG (#40e)

High Priority Phenomena in Update on NGNP High Temp Matls PIRT Include: (cont)

- Inadequate heat transfer from through core barrel due to potential compromise of emissivity from loss of desired surface layer properties (#46)
- Inability to maintain core structure geometry due to potential excessive deformation from radiation-creep in metallic core barrel (#47)
- Two high-priority phenomena on insulation & core restraint failure for non-metallic materials (#s 52,53)

Contributions to this presentation are gratefully acknowledged

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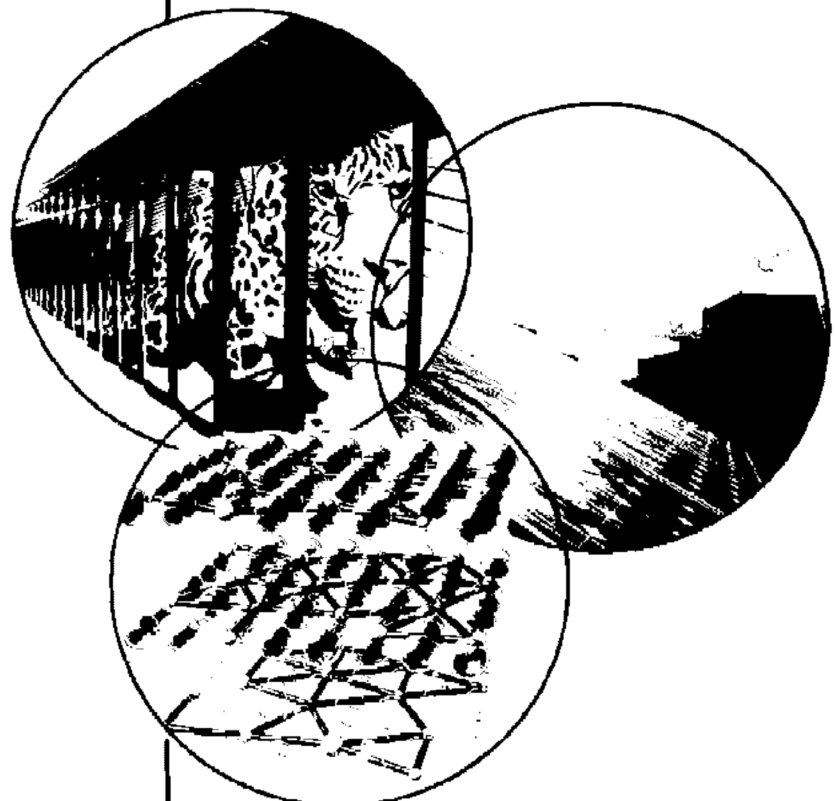
High Temperature Reactors - Construction Code Issues



Sam Sham
Oak Ridge National Laboratory

For Presentation at the Nuclear Regulatory Commission Meeting

February 17, 2011

U.S. DEPARTMENT OF ENERGY

OAK RIDGE NATIONAL LABORATORY
MANAGED BY UT-BATTELLE FOR THE DEPARTMENT OF ENERGY

SUBSECTION NH OVERVIEW AND KEY FEATURES

- Development initiated in late 60's in response to needs of LMR and HTGR
  - Continuous review and improvement since
    - Accelerated development in last few years after 10 – 15 year hiatus
- Implemented on FFTF and CRBR
  - Experience summarized in 4 volume set of WRC Bulletins
  - ACRS review identified concerns for CRBR licensing
    - Program plan for resolution identified and initiated – CRBR canceled prior to completion
- Failure modes addressed
  - NH addresses the rules for Class 1 nuclear components above the temperature limits of NB – 700F for ferritic and 800F for austenitic materials
    - Ductile rupture from short-term loadings
    - Creep rupture from long-term loadings
    - Creep-fatigue failure
    - Gross distortion due to incremental collapse and ratcheting
    - Buckling due to short-term loadings
    - Creep buckling due to long-term loadings
    - Loss of function due to excessive deformation
- Scope of rules
  - Materials
  - Design
  - Fabrication and installation
  - Examination
  - Testing
  - Overpressure protection

ELEVATED TEMPERATURE CODE CASES

Section III, Div 1	Coverage
N-201-5	Class CS (Core Support) Components in Elevated Temperature Service.
N-290-1	Expansion Joints in Class 1, Liquid Metal Piping.
N-253-14	Construction of Class 2 or Class 3 Components for Elevated Temperature Service.
N-254	Fabrication and Installation of Elevated Temperature Components, Class 2 and 3.
N-257	Protection Against Overpressure of Elevated Temperature Components, Classes 2 and 3.
N-467	Testing of Elevated Temperature Components, Classes 2 and 3.
N-499-2	Use of SA-533 Grade B, Class 1 Plate and SA-508 Class 3 Forgings and their Weldments for Limited Elevated Temperature Service.

SOME RELEVANT NH KEY FEATURES

- Allowable primary stress (those determining required wall thickness) based on time-dependent creep properties
- Degradation factors provided for effects of time at temperature (aging)
- Weld strength reduction factors (SRFs) provided as a function of time, temperature, process and weld metal
- Cyclic life assessed through strain limits and creep-fatigue
  - Strain limits reduced by a factor of two for welds to help insure that they are located in low stress areas
- Negligible creep criteria provided to permit application of NB rules for primary plus secondary stress limits
- Restricted material specifications to improve performance
  - Optional for 304SS and 316SS
- Cold work limitations and reheat-treatment requirements
- Severely restricted use of partial penetration welds
- Requires double volumetric examination of welds
  - Applies to category A, B, C and D welds in components greater than 4 in. diameter

MATERIALS

- Only 304 & 316 SS, 800H, $2^1/_4$Cr-1Mo and, recently, 9Cr-1Mo-V are approved pressure boundary materials; 718 for bolting
  - All are in annealed condition for long term stability except 9Cr-1Mo-V & 718
  - N&T for 9Cr-1Mo-V for time independent allowables & minimize ratcheting
  - SA508 & 533 in Code Case 499 for limited cycles, time & temperature
- Reduction in yield and ultimate due to elevated temperature service addressed in NH-2160*(d)*
  - Off normal operation could reduce allowables for subsequent events, i.e. seismic
- Fatigue acceptance test for 304 & 316 SS
  - Creep-fatigue test at 1% strain and 1hr hold time

MATERIALS TIME & TEMPERATURE

NH code materials (other than bolting)	Maximum temperature	
	For stress allowables S_0, S_{mt}, S_t, S_r up to 300,000 hours[a]	For fatigue curves
304 stainless steels (UNS S30400, S30409)	816°C	704°C
316 stainless steel (UNS S31600, S31609)	816°C	704°C
Alloy 800H (UNS N08810)	760°C	760°C
2¼Cr 1Mo steel, annealed condition (UNS K21590)	593°C[b]	593°C
Grade 91 steel (UNS K90901)[c]	649°C	538°C

a. The primary stress limits are very low at 300,000 hours and the maximum temperature limit

b. Temperatures up to 649°C (1200°F) are allowed up to 1,000 hours

c. The specifications for Grade 91 steel covered by Subsection NH are SA-182 (forgings), SA-213 (small tube), SA-335 (small pipe), and SA-387 (plate). The forging size for SA-182 is not to exceed 4540 kg.

DESIGN – Load Controlled (Primary) Stresses

- NH applies above temperature limit for NB allowable stresses
  - 700F for ferritic materials and 800F for austenitic materials
- NB applies above temperature limits if creep effects demonstrated not significant
  - Documented in Stress Report and included in Design Spec. (NH-3211*(c)*)
    - See also T-1324
  - Source for allowable stresses not defined
    - Presumably S_m from NH
  - Disconnect between temperature definition in NB and NH
    - Section average in NH vs. local maximum in NB for some cases
- NH Design Condition evaluation same as Section VIII, Div 1 with same allowables
  - Don't include short term loads in Design Conditions
- NH Service Condition allowable stress criteria same as NB for time independent *Sm* but different and more conservative than VIII, Div 1 for time dependent allowable S_t

VIII, Div 1	NH
80% of min. or 67% of avg. creep rupture stress	67% of min. creep rupture stress
Avg. strain rate = 0.01% per 1000 hr	100% avg. stress to 1% strain
None	80% of min. stress to tertiary creep

  - Wall thickness probably governed by Service Conditions

- Evaluation of Design Conditions and all Service Conditions except Level D are based on a linear elastic material model.
  - Requires stress classification
  - Reference stress methodology under consideration
- Time fraction summations used to evaluated different stress, time and temperature conditions
  - Time fractions summed over all Service Conditions
    - Time fraction is time in a specific condition divided by allowable time at that condition
- Weld strength reduction factors are provided for permitted weld metal and properties
  - Analysis based on parent metal properties

DESIGN - Deformation Controlled Limits

- Acceptable Deformation Controlled Limits in Appendix T
  - Alternative criteria may be used subject to Owner's approval and incorporation in Design Specification
  - Covers strain limits/ratcheting (analogous to P + Q), creep-fatigue damage (analogous to P + Q + F), buckling and welds
- Strain Limits and Creep-Fatigue Damage rules can be satisfied using either elastic or inelastic analysis methods
  - Elastic analysis rules originally envisioned as simpler, more conservative and less costly screening method to satisfy strain limits and creep-fatigue
    - Actually considerably more complex than analogous 'low" temperature rules in NB
  - Inelastic rules envisioned as a more costly and time consuming "gold standard"
    - Conceptually simple but require sophisticated modeling of material behavior in creep regime
    - Requirements for material modeling only addressed in general terms/performance criteria
      - Didn't want to stifle development of improved methods

• Creep-Fatigue

- Major source of conservatism in NH
  - Criteria: $\sum(n/N_d) + \sum(\Delta t/T_d) \leq D$
  - n, number of cycles of a given strain range
  - N_d, allowable number of cycles at that strain range
  - Δt, time at a stress level calculated from average properties
  - T_d, allowable time at the calculated stress level divided by a factor, $K' = 0.67$ for inelastic analysis and 0.9 for elastic analysis, as determined from plot of min. stress to rupture vs. time to rupture
  - D, damage factor to account for combined effects of creep and fatigue
- Rationale for conservatism
  - $K' = 0.67$ is based on Eddystone piping failure and component test results
  - Recent reassessment based on elastic analysis led to $K' = 0.9$
  - D for 9Cr-Mo-V due in part to environmental effects and in part to evaluation methodology

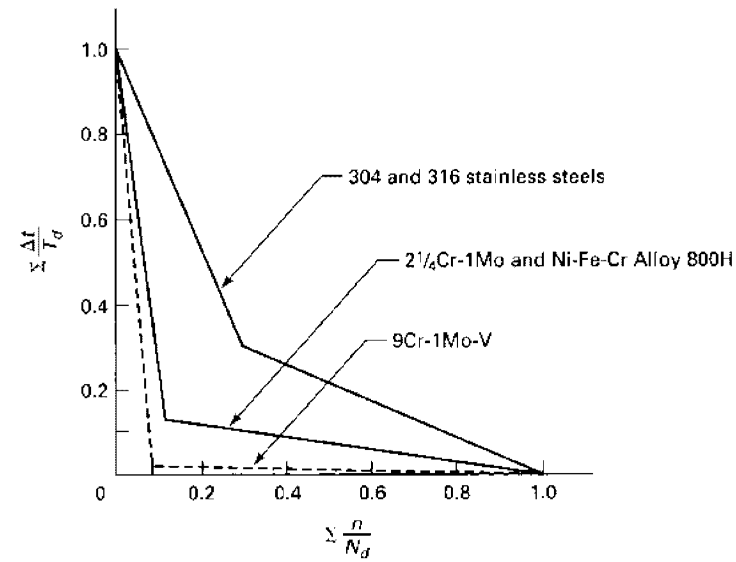


FIG. T-1420-2 CREEP-FATIGUE DAMAGE ENVELOPE

- Currently under review in SG-ETD
- Result
  - Wall thickness may be limited by creep-fatigue rather than load controlled stress criteria

- **Buckling**
  - Buckling charts for Section VIII, Div. 1 & Section 1 do not consider creep
    - Figures provided in NH to define time & temperature limits for applicability of charts
  - A matrix of load factors is supplied in NH to address buckling and instability
    - Factors are a function of:
      - Source, load controlled or deformation controlled
      - Duration, time independent or time dependent
      - Service Level

- **Welds**
  - Weld strength reduction factors
  - Strain limits half that of parent material
  - Creep-fatigue limits:
    - N_d, allowable number of cycles reduced by factor of two
    - Minimum parent metal creep rupture strength reduced by weld strength reduction factor
  - Weld geometry
    - Worst case geometry used in analysis
      - » No upper limit on stress concentration factor implies ground welds
    - Confirmed by inspection

EXAMINATION

- **Dual weld exam required**
  - Radiography plus ultrasonic
  - Radiography plus eddy current
  - Radiography at two different angles
- **NH-5000 refers to NB-5000**
  - NB-5000 invokes Section V “Nondestructive Examination“
  - Article 14 "Examination System Qualification“

Recommended Reading

- Chapter 12 of "Companion Guide to the ASME Boiler & Pressure Vessel Code" K. R. Rao, Editor, ASME Press
  - Background and discussion of Subsection NH rules and their implementation including relevant Code Cases

DOE/ASME GEN IV MATERIALS PROJECT

- Collaboration between DOE and ASME established in 2007 to address technical topics that were identified by DOE, ORNL, INL, and ASME to have particular value with respect to the ASME Code
- In support of an industrial stakeholder's application for licensing of a Gen IV nuclear reactor

- Phase I
  - Tasks 1– 5 completed
    - 2007/2008
- Phase II
  - Tasks 6-11 completed
    - 2009/2010
- Phase III
  - Proposed Tasks 13 – 14
    - Started in 2010, ongoing
    - Task 12 on NDE was funded by NRC

Task 1: Verification of Allowable Stresses in Section III, NH with Emphasis on Alloy 800H and Grade 91

- Rationale
  - Longer design lives at higher temperatures to support HTGR
  - Discrepancies in 800H allowable stresses and differences in allowables for Grade 91 between RCC-MR and NH/II-Part D in thick sections
- Results
  - Alloy 800H base metal values for Sy & SU established to 900 C, SRmin values to 600,000 hr and 900 C
    - 1 % strain controlled long time allowable stress at 850 and above, testing required
  - Alloy 800H weldments require testing above 750 C for long times
  - Grade 91 base metal data support allowable stresses to 600,000 hr and 650 C
  - Grade 91 weldments should adopt Section II report on SRFs
- Next step
  - Use data in Task 13 for Code approved allowable stress values

Task 2: Regulatory Safety Issues in Subsection NH and for Very High Temperatures for VHTR & GEN IV

- Rationale
  - Avoid licensing delays due to unresolved concerns
    - NRC has not endorsed NH
- Results
  - Creep crack growth in weldments and notches, inelastic analysis, and environmental effects are primary issues identified in prior reviews
    - Time and temperature extension and additional concerns identified
  - How NH and Codes Cases address NRC issues was summarized
    - Materials behavior, creep-fatigue and environmental effects
    - Structural integrity of welds
    - Development and verification of simplified design analysis methods
    - Verification testing
- Next step
  - Used to identify follow-on tasks

Task 3: Improvement of Subsection NH Rules for Grade 91 Steel – (Negligible Creep and Creep Fatigue)

- Rationale
  - Current NH criteria for negligible creep and creep-fatigue damage in Grade 91 overly conservative and too restrictive for realistic design application
- Results
  - Negligible creep
    - Criteria for cyclically hardening materials, e.g. austenitic stainless steel, inappropriate for cyclically softening material, e.g. Grade 91
    - Detail modifications proposed & further testing
  - Creep-Fatigue
    - ASME design procedure is very conservative
    - Proposed modifications
      - Reduce safety factor for creep damage calculation with elastic analysis (k' = 0.9) Incorporated in 2008 Addenda
      - Additional modifications and testing
- Next step
  - Proposed modifications evaluated in Task Force on Creep-Fatigue &Task Force on Negligible Creep
  - Data & recommendations used in Task 10

Task 4: Updating of ASME Nuclear Code Case N-201 to Accommodate the Needs of the HTGR

- Rationale
  - CC N-201 (Elevated temperature core support structures) last updated prior to NH
  - Limited materials selection
  - HTGR core support structures expected to see very high temperatures
- Results
  - Questionnaire on metallic core support structure design and requirements
    - Normal and transient temperatures benign and current design methods applicable
    - Additional material needs
      - 316FR/316LN, 321 & 347, Grade 91 and Inconel 718
      - life extension to 60yr
  - Comprehensive line by line review performed against NG and NH
    - CC N-201 revised to correct errors and omissions
- Next step
  - CC revisions approved in SG-ETD – final edit in WG-CSS

Task 5: Collect and study Available Creep-Fatigue Data and Procedures for Grade 91 Steel and Hastelloy XR

- Rationale
  - Significant data on Grade 91 exists in Japan
  - Hastelloy XR used successfully in the Japanese HTGR
- Results
  - Grade 91
    - Numerous data collected
    - NH procedure significantly conservative compared to test data and RCC-MR and Japanese FBR procedures
    - Potential improvements to NH identified and evaluated
    - R&D needs identified
  - Hastelloy XR
    - Available data collected
    - Elevated temperature response characteristics similar to austenitic SS
    - Material models for inelastic analysis were developed for HGTR IHX
- Next step
  - Data and assessments used in Tasks 3 and 11

Task 6: Operating Condition Allowable Stress Levels

- Rationale
  - Minimum Stress to rupture values in NH not consistent with Section II, Part D
- Results
  - Inconsistencies confirmed for current NH materials except Alloy 800H
  - Comprehensive data collection and evaluation for NH materials
    - Current data support 304H and 316H allowable stress values to 1200°F
    - Concern for low creep ductility
    - 800H data support extended stress values to 800 – 850°C
    - Grade 91 data support 500,000 hr below 600°C and 650°C up to 100,000 hr
    - Data for annealed 2.25Cr-1Mo support values to 1200°F
  - Prioritized table of suggested action to revise allowable stress to accommodate HTGR needs
- Next step
  - Implement recommended actions in follow-on task to develop allowable stresses for code committee action

Task 7: ASME Code Considerations for the Intermediate Heat Exchanger (IHX)

- Rationale
  - IHX exposed to full gas outlet temperature at primary to secondary interface
  - Potential use of compact micro channel heat exchangers with unique design features raises concerns
- Results
  - Conventional and compact experience reviewed
    - Tubular Helical Coil Heat Exchanger most mature
    - Compact HX less mature but potential cost and volume savings
    - 617 most promising material followed by 230, XR and 800H
  - Recommended Code approaches defined
    - IHX should be considered non-safety related component
    - Current C & S basically OK for shell and tube
    - Difficult ISI suggests periodic replacement of compact IHX
  - Extensive review of ASTM Standards development
- Next step
  - Implement recommendations in a draft code case

Task 8: Creep and Creep-Fatigue Crack Growth at Structural Discontinuities and Welds

- Rationale
  - Top NRC concern
  - NH has design factors and procedures for weldments and structural discontinuities but not a quantitative assessment of crack growth
- Results
  - Current crack growth methodologies assessed for applicability to design and IS
  - UK R5 approach selected based on development status and current use
  - Theoretical limitations identified
  - Design procedure described
- Next step
  - Extensive discussion in SG-ETD
  - Recommended for use with ISI for inspection intervals and flaw evaluation
  - Applicability for HTGR materials and conditions needs to be established
  - Potential joint BPV SC-III & XI Task Force

Task 9: Update and Improve NH – Simplified Elastic and Inelastic Design Analysis Methods

- Rationale
  - Current NH rules based on 70's – 80's technology
  - Advances in computing technology
  - Advances in understanding of creep behavior and failure mechanisms
- Results
  - Comprehensive review and comparison of elevated temp. design codes and fitness for service manuals
  - Recommended improvements to NH
  - Elastic analysis
  - Reference stress methods
  - Limit load, shakedown, and ratcheting analysis
  - Recommended available benchmark problems and key feature tests
- Next Step
  - Recommended approaches currently under review in SG-ETD

Task 10: Update and Improve NH – Alternative Simplified Creep-Fatigue Design Methods

- Rationale
  - Phase I Tasks 3 & 5 identified a number of deficiencies in current methodologies
  - New methods have been developed
  - Identified as an NRC concern
- Results
  - Creep-fatigue methodologies including damage based, strain based, modified strain range partitioning and methods not separating creep and fatigue damage evaluated with Grade 91 data from Tasks 3 & 5
  - All methods correlate reasonably with short term data, differences in long term extrapolation
  - Near term: incorporate key features in current time fraction approach
  - Generally currently deployable
  - Long term: incorporate SMT methodology which requires signification test & validation
- Next step
  - Review and implementation in Task Force on Creep-Fatigue Criteria
  - Apply methodology to assess other materials of interest

Task 11: New Materials for NH

- Rationale
  - Additional material options in NH & CC N-201 needed to address unique VHTR requirements, e. g. very high temperatures and environmental effects
- Results
  - Comprehensive review of prior design studies and operating conditions
  - Requirements for codification reviewed in detail
  - Candidate alloy characteristics discussed
  - I00,000 hr creep rupture strength for 7 candidate alloys plotted vs. temperature
  - In descending order at 800°C: 230, 617, 625, 556, NF 709, 120, X, 811, 810
  - Hastelloy XR covered in separate report
  - Though 617 is stronger in air than XR they are comparable in HTGR He which doesn't affect XR
  - Testing requirements and cost estimates based on review of NGNP IHX Materials R&D Plan for Alloy 800H and Alloy 617
- Next step
  - Primarily intended for project use

Task 13: Recommend allowable stress values

- Benefit
  - Extends the time and temperatures for which allowable stresses are provided to be compatible with NGNP/GEN IV needs.
  - Specific stakeholder interest in 800H
- Key Points
  - Develops draft code rules for extending 800H limits
  - Provides allowables at time (60yr) and temperature (850°C) compatible with NGNP/GEN IV needs for normal operation
  - Provides higher temperature, shorter time allowables for off normal events
  - Results in code formatted submittals to applicable committees
- Status
  - Ongoing

Task 14: Corrections to stainless steel allowable stresses

- Benefit
  - Corrects recently identified problems with allowable stress values that impact on-going design studies.
- Key Points
  - Task 6 identified errors and potential limitations on current SS stress values in NH
  - Some heats of 304 & 316SS had creep rupture lives below current NH values, particularly above 1200F (650°C)
    - NIMS (post NH) data
  - Identify restrictions to preclude bad heats or
  - Delete impacted allowables
  - Results in code formatted submittals to applicable committees
- Status
  - Ongoing

DIVISION 5, CONSTRUCTION RULES FOR HIGH TEMPERATURE REACTORS

- Need
  - Renewed interest and acceptance of nuclear fission as a source of energy on a global level
  - High temperature reactors are being considered by various countries and companies for future reactor applications
  - Many efforts to collect and develop data for use in high temperature reactor applications and in the development of appropriate Codes and Standards
  - Some current rules have not been properly maintained and are out of date
  - Need a new Division to cover construction rules for components in high temperature reactors

DIVISION 5 SCOPE

The rules of Division 5 constitute the construction requirements associated with components and structures used in high temperature gas-cooled reactors and liquid metal reactors

DIVISION 5

Subsection HA — General Requirements
- Subpart A – Metallic Materials (NCA)*
- Subpart B – Graphite Materials (New)
- Subpart C – Composite Materials (New)

Subsection HB — Class A Metallic Pressure Boundary Components
- Subpart A – Low Temperature Service (NB)
- Subpart B – Elevated Temperature Service (NH)
  - Mandatory Appendix HBB-I (CC N-499)

Subsection HC — Class B Metallic Pressure Boundary Components
- Subpart A – Low Temperature Service (NC)
- Subpart B – Elevated Temperature Service (CC N-253)

Subsection HF — Class A and B Metallic Supports
- Subpart A – Low Temperature Service (NF)

Subsection HG — Class A Metallic Core Support Structures
- Subpart A – Low Temperature Service (NG)
- Subpart B – Elevated Temperature Service (CC N-201)

Subsection HH — Class A Non-Metallic Core Support Structures
- Subpart A – Graphite Materials (New)
- Subpart B – Composite Materials (New)

Two Safety Classes – Class A and Class B
( )* Code Basis

SAFETY CLASSIFICATIONS

Class A - “safety-related”
Class B - “non-safety related with special treatment”

- Reflect the risk-based approach derived from safety criteria established for high temperature reactor plants
- Remaining items not in these two classifications shall satisfy requirements of other appropriate non-nuclear codes and standards

FUTURE IMPROVEMENT OF DIVISION 5

- An Ad-Hoc project team within ASME BPV III, Working Group on Liquid Metal Reactors (LMRs) was formed to establish strategic goals, structure and scope, and execution plan for LMRs in Div 5
- Two white papers were drafted
- The purpose was to develop a consensus on the path forward to provide ASME Code rules for the construction of the next generation LMRs which also includes LMR-based advanced small modular reactors (SMRs)
- Near Term LMR White Paper focused on the 2011, 2013 and 2015 Code editions
- Long Term LMR White Paper focused on the 2017, 2019 and beyond editions

NEAR TERM LMR WHITE PAPER OVERVIEW

- '11, '13 & '15 Code Editions
  - Start approval cycle in 1 – 3 years
- Conventional scope
- Based on current, '11, Div 5 format
  - References other sections
  - Includes Code Cases
- Two classes of construction, A & B

Highest priority items	Next priority
Correct and extend allowable stresses	Update and revise CC N-253 etc.
Resolve 2 vs 3 component classification issue	Add 316LN/FR
Inelastic analysis procedures and models	Incorporate creep-fatigue crack growth in Section XI, Div 3
Improvements to creep-fatigue and negligible creep • Emphasis on Mod9Cr	Add exemption rules for creep-fatigue evaluation
Add reference stress for wall sizing	Add exemption rules for creep-fatigue evaluation

LONG TERM LMR WHITE PAPER OVERVIEW

- ’17 & beyond Code Editions
  - Start approval cycle in 5 years
- Expanded scope
  - Add “leak-before-break”, fitness-for-service & environmental effects
  - All temperature
  - Stress classification replaced by new methods
- New, self-contained format
  - Still reference other sections i.e. Section II
- Two classes of construction, A & B (?)

Highest priority items	Next priority
Feasibility assessment of new methodology	Incorporate improvements to creep-fatigue rules
Develop proposed template for long term Div 5	Develop specific recommendations for new, all temperature, stress classification free methodology
	Add advanced materials e.g. NF616 & HT-UPS
	Add creep-fatigue crack growth to Div 5 design
	Add environmental effects, coolant and irradiation
	Guidelines for thermal striping
	Guidelines for leak-before-break
	Guidelines for fitness for service/remaining life

TASK GROUP ON INELASTIC ANALYSIS METHODS

- Develop non-mandatory NH appendix on inelastic analysis methods for current NH materials
- Envision having draft code rules in place by end of 2013

Approach

- Use NE F9-5T, "Nuclear Standard, Guidelines and Procedures for Design of Class 1 Elevated Temperature Nuclear System Components" (developed by DOE for CRBR vendors) as guideline
- Use currently available state-of-the-art models
  - Models might not be perfect
  - New development will be kept to a minimum
- Use currently available specimen test data, to the extent possible
  - Experiments and testing by Task Group are out-of-scope
- Data source for material models
  - Literature
  - Contributions from vendors, international agencies, US national labs - all on the basis of supporting ASME code committee work
- Perform verification analyses, to the extent practical

Contents of non-mandatory appendix

- Part A
  - 3D unified viscoplastic constitutive equations for current NH materials
    - 304, 316 stainless steels, 9Cr-1Mo-V, 2¼ Cr-1Mo, Alloy 800H, Alloy 718 (bolting)
    - Material parameters for each material, covering NH temperature range, at every 50C
    - Will consider the inclusion of 1D equations, if a need is identified
  - Potential Issues
    - Could lead to a need for updating isochronous stress-strain curves for consistency
- Part B
  - Provide guidance on inelastic finite element analyses
  - Example problems on how to use results from inelastic finite element analyses to satisfy NH deformation limits

High Temperature Reactor Materials

Licensing and Regulatory Issues

February 17 , 2011

Saurin Majumdar, Ken Natesan
Argonne National Laboratory

Argonne NATIONAL LABORATORY
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U.S. Department of Energy

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Office of Science

Summary of Licensing Issues for Clinch River Breeder Reactor

- NRC/ACRS conducted a licensing review (NUREG-0968) of a construction permit for CRBR in the 1980's
- Construction permit was supported with the stipulation that numerous technical issues be resolved prior to requesting an operating license
- The R&D program that was agreed to was not completed
  - Materials
  - Design analysis
  - Weldment integrity
  - Creep ratcheting
  - Creep cracking
  - Creep/creep-fatigue damage evaluation procedure

Major Concerns for CRBR Licensing: Background

- Up until now, the maximum temperature experienced by the LWR industry is ~350°C
  - Primary components designed by ASME Code Section III, Subsection NB
  - Section III, Subsection NB and Section XI have been approved by NRC
  - Time/temperature-dependent deformation and damage not major concerns
  - Significant industry experience in material/design/fabrication of LWR components
  - Significant industry experience in in-service inspection techniques
- Reactor outlet temperature for CRBR was 995°F (535°C)
  - Austenitic stainless steels and low-alloy ferritic steels
  - Time/temperature-dependent deformation cannot be ignored
  - Time/temperature-dependent damage cannot be ignored
- Based on the review of the material submitted by the applicant, NRC listed nine areas of concern

Major NRC Concerns for CRBR Licensing

- Weldment Cracking
- Notch weakening
- Material property representation for inelastic analysis
- Steam generator tubesheet evaluation
- Elevated temperature seismic effects
- Elastic follow-up in piping
- Creep-fatigue evaluation
- Plastic strain concentration factors
- Intermediate piping transition welds

NRC Concern - Weldment Cracking

- Identified as the most significant concern
- Early crack initiation in HAZ
- Variation of material properties within the weld - creep damage
- Effect of cycle rate, hold time on propagation of long shallow cracks in HAZ
- Effect of enhanced creep in uncracked ligaments of cracks due to residual stress and thermal cycling on crack stability and creep crack growth
- These effects must be evaluated to determine the safety margins of weldments in elevated temperature service.
- Effect of long-term aging on creep-fatigue damage
- Effect of loading sequence on creep-fatigue behavior

NRC Concern - Notch Weakening

- Cracking at notches and other discontinuities due to stress-strain concentration effect and exhaustion of local ductility
  - Subsection NH (Code Case N47) ignores notches for load controlled stresses
  - They are considered for strain-controlled loading and creep-fatigue
- Main concern was that the creep-fatigue limits are set on the basis of tests on smooth specimens and, therefore, do not consider stress gradient near notches
- Also, concerned about loss of ductility due to long-term cyclic and monotonic loading.

NRC Concern - Material Property Representation for Inelastic Analysis

- NRC was concerned about lack of verification of computer programs used for conducting inelastic analyses
- Concerned about impact of new technology developments on safety
  - Safety margins that worked well with conservative simplified analyses may be eroded by the use of more accurate analysis techniques
  - Concerned with using average properties rather than minimum properties for inelastic analysis
  - Does the safety factors adequately cover departure from average behavior
  - Is creep rupture damage calculated conservatively in the presence of hardening due to cyclic loading or fabrication processes?

NRC Concern - SG Tubesheet Evaluation

- The major concern was the assurance of adequate tubesheet design life
  - Are the calculations adequate to account for the highly localized inelastic stresses in the outer row of ligaments due to thermal gradients between the perforated and unperforated regions?
  - The use of equivalent solid plate may not be applicable to tubesheets with large thermal gradients
  - Detailed inelastic analysis of mechanically and thermally interacting tubes, tubeholes and ligaments for evaluating ratcheting and creep-fatigue damage is highly complex

NRC Concern - Elevated Temperature Seismic Effects

- Can creep strain accumulation or creep-fatigue damage be enhanced by seismic events?
  - Seismic events may change the residual stress field by short-term primary loading
  - Relaxation of high residual stresses following a seismic event may enhance accumulated creep strain (ratcheting) and creep-fatigue damage
  - Sequence effects may be important at elevated temperatures

NRC Concern - Elastic Follow-up in Piping

- The concern was related to categorizing thermal expansion stresses as secondary for evaluating hot leg piping
- Under creeping condition, relaxation of stresses in highly stressed areas may cause additional cyclic strain and strain accumulation due to elastic follow up.
  - Subsection NH recommends that thermal stresses with large elastic follow up be considered as primary, but does not define when elastic follow up is large

NRC Concern - Creep-Fatigue Evaluation

- CRBR project changed the Code damage calculation procedure for austenitic stainless steel non-safety components by considering compressive holds as less damaging than tensile holds
- Second concern was related to the extrapolation of high cycle fatigue curves in Code Case N47 beyond 10^6 cycles using a slope of -0.12 for load controlled situations

NRC Concern - Plastic Strain Concentration Factors

- The concern was related to using the plastic strain concentration factor as unity for ranges of primary plus secondary stress intensity less than $3S_m$, whereas plastic strain will occur when the locally concentrated stress range exceeds $2S_y$
  - Also, the multiplier for strain concentration on the weaker side of a joint or interface was not included in the formulas for K_e

NRC Concern - Intermediate Piping Transition Welds

- The IHX transition weld reference design was a tri-metalic joint of Type 316H stainless steel, Alloy 800H and 2-1/4Cr-1Mo steel
  - The concern was related to the variability of material properties between the different materials
  - Possible increase of creep rupture damage resulting from the higher yield properties produced by hardening in a multipass welding process.

Summary CRBR Licensing Review

- Major NRC concerns were related to treatment of discontinuities
  - Weldments
  - Notches
  - Tubesheets
- Areas where ASME Code treatment was lacking

Current Safety Issues for Structural Design of VHTR and Gen-IV Reactors

- Evaluate Materials and design bases in ASME Code Case N-47 (NUREG/CR-5955, 1993)
  - Identify issues that must be resolved in order to avoid creep rupture, creep-fatigue, creep ratcheting and creep buckling
  - Advanced LMRs, gas-cooled reactors and CANDU reactors
  - 23 issues were identified. Most important issues are
    - *Lack of materials design allowable data for 60yr life*
    - *Degradation of properties - long-term radiation, corrosion at high temperatures*
    - *Lack of validated thermal striping design methodology*
    - *Reliable creep-fatigue and ratcheting design rules*
    - *Lack of validated weldment design methodology*
    - *Lack of flaw assessment procedure*
    - *Lack of inelastic deign procedure for piping*
    - *Lack of validation of notch weakening effect*

Current Safety Issues for Structural Design of VHTR and Gen-IV Reactors (Cont'd)

- Pre-application safety evaluation of power reactor innovative small module (PRISM) LMR (NUREG-1368,1994)
- NRC concerned primarily with
  - inelastic and limit analyses
  - Extrapolation of Code Case N47 design allowables from 34 to 60 yrs.
  - Environmental issues related to stress corrosion, flowing sodium effects and irradiation embrittlement
  - Weld between core support structure and the RV
  - Neutron Embrittlement for RV with 60 yr design life

Review and Assessment of Existing Design Codes for HTGRs (NUREG-CR/6816, 2003)

- Most of the materials needed for HTGR (Alloy 617, 9Cr-1Mo-V*, Hastelloy X) are not included Subsection NH
- The Code materials that are acceptable for HTGR need to have their upper temperature limit extended to 850°C
- Subsection NH rules are written for materials with classical creep curve (primary, secondary and tertiary). HTGR materials do not show such a behavior
- Advanced unified constitutive equations (no distinction between creep and plastic strain) are needed for the HTGR materials
- The effects of impure helium on fatigue and creep-fatigue properties are needed
- Draft Code Case for Alloy 617 needs to be completed
- Need a more suitable damage theory for creep-fatigue than linear damage rule

Materials Behavior in HTGR Environment (NUREG/CR-6824, 2003)

- Among the three materials for high temperature application in HTGR - Alloys 800H and 617 and Hastelloy X - only Alloy 800H is code certified up to 760°C and 34 yrs.
  - Substantial database exists for Alloys 800H and 617
  - Limited database for Hastelloy X
- Need data on effects of contaminated helium (at pressure) on properties
- Structural alloys can be corroded by gaseous impurities in helium

Design Features and Technology Uncertainties for NGNP (INEEL/EXT-04-01816, 2004)

- Few choices exist for metals for use in VHTR design conditions
- New materials such as, ODS alloys, refractory metals or ceramics need to be developed for long range application at ~1000°C
- For near-term applications, a maximum metal temperature of 900°C was recommended

Framework for Development of Risk-Informed Alternative to 10 CFR Part 50 (NUREG-1860, 2006)

- Report documents a technical basis to support the development of a risk-informed and performance-based process for licensing of future reactors
  - Provides broad guidance for safety review
  - Does not provide guidance for codes and standards
  - The evaluation approach relies heavily on PRA
  - Does not imply that structural design codes be based on PRA
    - *Code assessment results should be in a form that allows PRA of individual components.*

How Regulatory Issues are Addressed in ASME Code

- A new Division 5 of the ASME code has been established to handle HTGRs and LMRs (GEN IV Systems)
  - Materials creep behavior, creep-fatigue, environment effect
    - *Improve structural analysis methods for cyclic loading at high temperatures*
    - *Negligible creep curves*
  - Structural Integrity of Welds
    - *Allowable life and ductility limits are reduced at welds*
    - *Need to account for material variability within the weld and HAZ*
  - Development and verification of simplified design analysis methods
    - *Need new validated methods for HTGR applications*
  - Verification testing
    - *Will need component testing to validate VHTR designs*

ASME Code - In-Service Inspection Issues

- A special Working Group has been set up to look at ISI issues (T. Lupold is a member)
  - Developing requirements for HTGRs
  - Reliability Integrity Management program (RIM)
    - *risk based program combined with some deterministic inspection requirements*
    - *the designer/owner of the plant has to establish reliability requirements for the systems/components*
    - *ISI established to meet those requirements*
    - *The idea is to change the design during the design phase if the reliability requirements cannot be met by the available inspection techniques*
  - RIM is still in a developmental stage
  - Preliminary version is available in ADAMS

Research Needed to Address Regulatory Issues for VHTR

- Material creep behavior, creep-fatigue and environmental effects
- Structural Integrity of Welds
- Development/verification of simplified analysis methods
- Verification testing of components
- Development of ISI techniques

Leak-Before-Break
History, Updates, and Future Plans

David Rudland

U.S. NRC/RES/DE/CIB

Thursday October 11, 2012

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United States Nuclear Regulatory Commission
Protecting People and the Environment

Outline

History of LBB

Early application of LBB in regulatory environment

Updates in technology since original SRP 3.6.3

Effects of active degradation (PWSCC) on LBB – Technical and regulatory

xLPR(extremely low probability of rupture) summary and Regulatory plan for LBB

What is LBB?

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Generally, LBB is the demonstration that a postulated flaw will leak and be detected, before catastrophic failure

or....

Specifically, LBB is the application of fracture mechanics technology to demonstrate that high energy fluid piping is very unlikely to experience double-ended ruptures or their equivalent as longitudinal or diagonal splits

History of LBB

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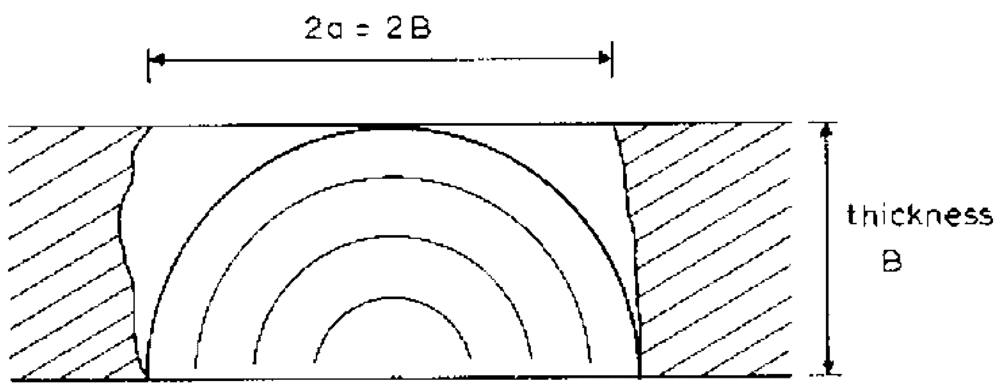
- Earliest approach by Irwin in 1961 for an axial flaw in pipe or pressure vessel
- Linear elastic
- Crack driving force in radial direction is greater than axial direction for 2a>2B

$$K_{1c} \geqslant \sigma_{ys}\sqrt{\pi(B + r_p^*)}\,.$$

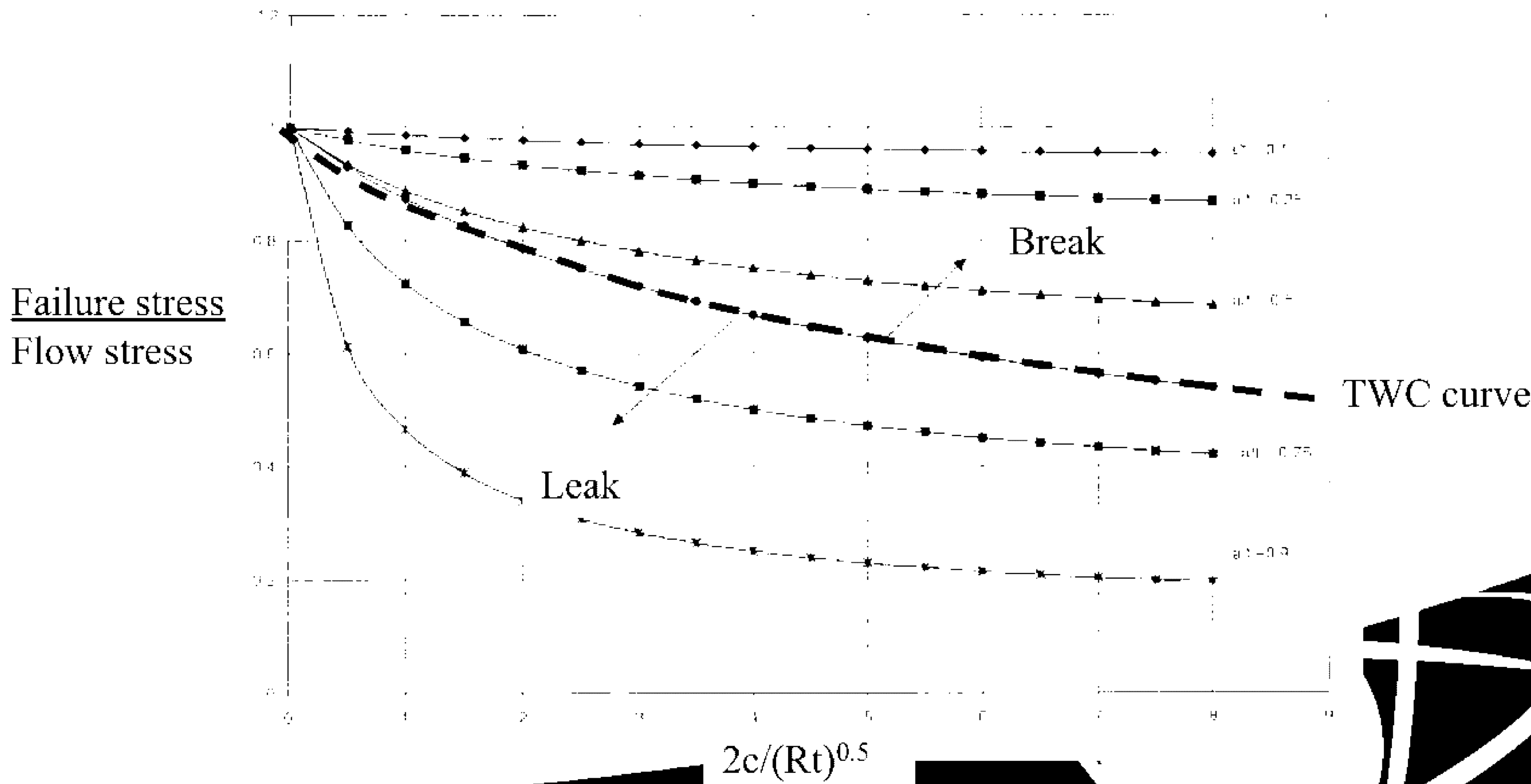
By taking $r_p^* = \sigma^2 a / 2\pi\sigma_{ys}^2$ with $a = B$, and $\sigma = \sigma_{ys}$, it follows that:

$$K_{1c}^2 \geqslant (\pi + \tfrac{1}{2}) B\sigma_{ys}^2 \quad \text{or} \quad \frac{K_{1c}^2}{B\sigma_{ys}^2} \geqslant \pi + \tfrac{1}{2}\,.$$



History of LBB

Battelle work by Duffy, Eiber, Kiefner and Maxey assumed ductile fracture behavior of axial thin-wall gas pipelines (1960's), then nuclear pipe for USAEC (late 1960's and early 1970's)



Early Application

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The U.S. Code of Federal Regulations (10CFR50) states that systems and structures shall be designed to accommodate accidents and postulated ruptures

- Pipe-whip restraints, jet impingement shield barriers are needed

General Design Criterion 4 (GDC-4) allows the use of analyses, approved by the NRC, to demonstrate extremely low probability of pipe rupture for removal of protective hardware

In 1984, leak-before-break (LBB) was accepted as an analytical procedure for demonstrating extremely low probability of rupture events

NUREG-1061

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In 1984, a five volume report was published on a review of the NRC requirements in the area of nuclear piping

Volume 3 of this document reviews the evaluation of potential to pipe breaks

- Gives recommendation for application of LBB in the NRC licensing process

The conclusions and recommendations from this volume were implemented into Standard Review Plan on LBB (3.6.3) in 1987

SRP 3.6.3

The SRP is applicable to Class 1 piping with the following caveats:

Screening Criteria

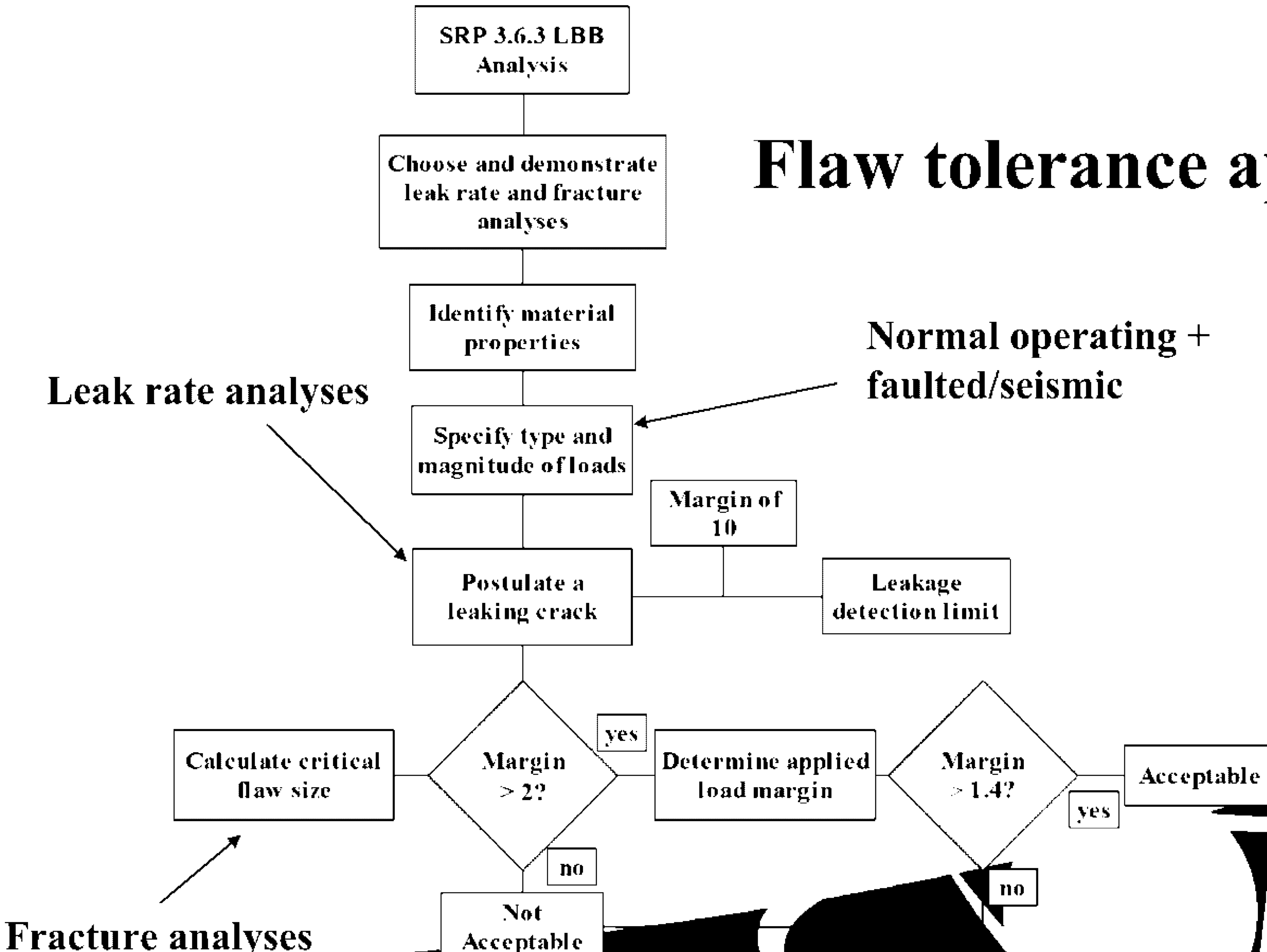
- Must be applied to entire system
- Cannot be used for piping susceptible to SCC, erosion-corrosion, creep, etc. (i.e., no degradation mechanisms that can cause long surface cracks)
- Pipes with weld overlays cannot be considered (removed in later version)
- Systems with a history of fatigue cracking cannot be considered
- Pipe with likely water hammer are not considered
- Piping systems with possible brittle fracture are not considered
- Indirect failure must be shown not to cause rupture

SRP 3.6.3 was revised in 2007 to include Alloy 690/52/152 and overlays

- "Alloy 690/52/152 material is not currently considered susceptible to PWSCC for the purposes of LBB application"

Steps in SRP 3.6.3

Flaw tolerance approach



Defense in Depth

ECCS is designed to handle break in the largest piping in RCS

ASME Section III and screening criteria in SRP3.6.3 provide assurance of extremely low probability events

LBB analyses provide defense in depth against rupture or large break opening to ensure confidence that the probability of pipe rupture is extremely low

US Accepted LBB

Accepted LBB applications in U.S. (All PWRs)

SYSTEM	***NUMBER OF APPROVALS***
Primary Coolant Loop (Hot & Cold Legs)	*76*
Pressurizer Surge Lines	*14*
Safety Injection Accumulator Lines	*11*
Residual Heat Removal Lines	*9*
Safety Injection Charging Lines	*4*

BWR Approvals – NONE

- *Susceptibility to IGSCC Not Adequately Addressed*
- *Few requests for LBB*

Recent Research Since Original SRP 3.6.3

Factors that affect leakage size cracks

- Crack morphology
- Restraint of pressure-induced bending
- Welding residual stress

Material Issues

- Cyclic Effects
- Dynamic Strain aging
- PWSCC testing – Alloy 82/182 and Alloy 52/152

Fracture/Stability Issues

- J-estimation Schemes

Recent Research Since Original SRP 3.6.3

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Factors that affect leakage size cracks

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- Dynamic Strain aging
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Fracture/Stability Issues

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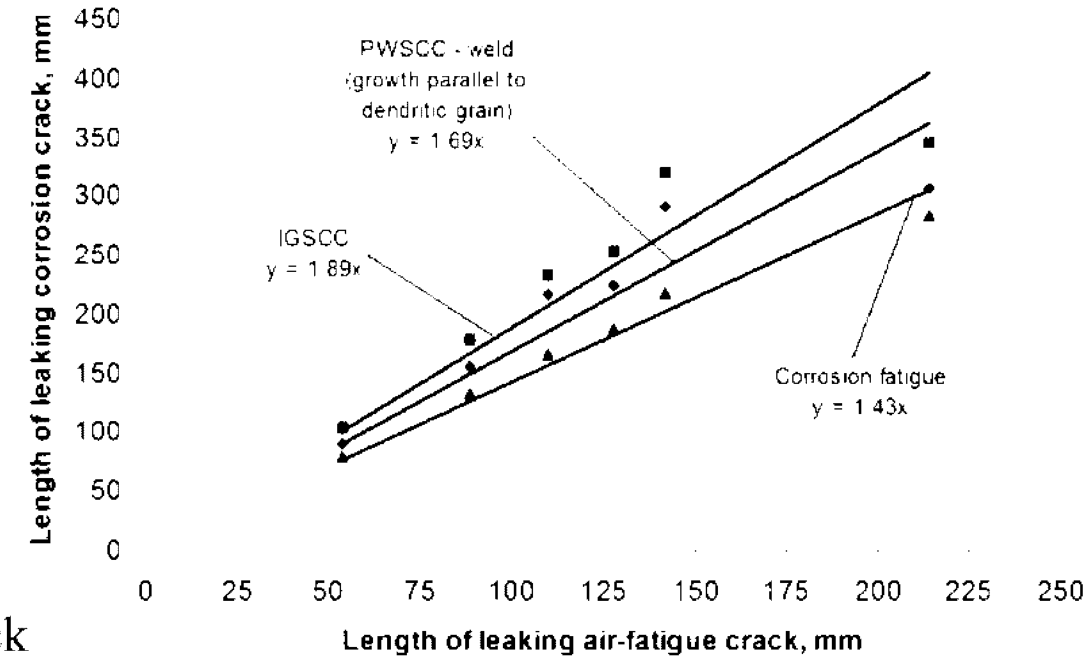
Crack Morphology Parameters

Using ISCCC or PWSCC morphology, leaking crack length at same leak rate can by 89% longer than when air fatigue is assumed

Air fatigue crack

SCC Alloy 82/182 crack



Current leak rate analyses have limited validation with SCC and thus have large uncertainties

Welding Residual Stress

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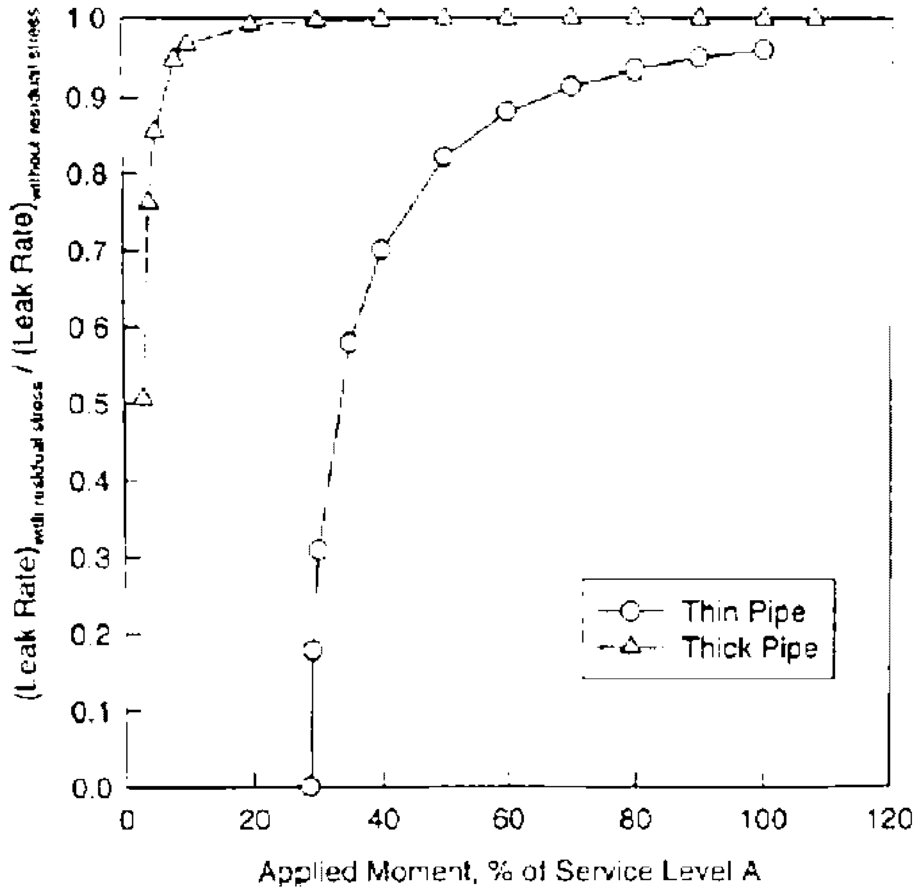
Through thickness welding residual stresses can affect the crack opening area

- Crack-face closure due to the through-wall bending residual stresses (thin-wall pipes),
- Non-elliptical opening (assumed in many leak-rate calculations), and
- Through-wall residual stress distribution being a function of weld preparation geometry, total number of passes, start-stop locations, and the bulk heat input.



Dissimilar Metal Welds

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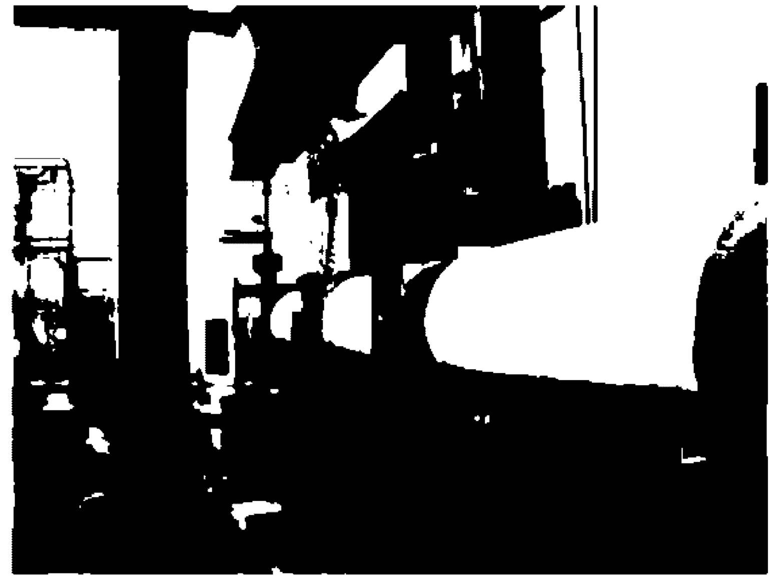
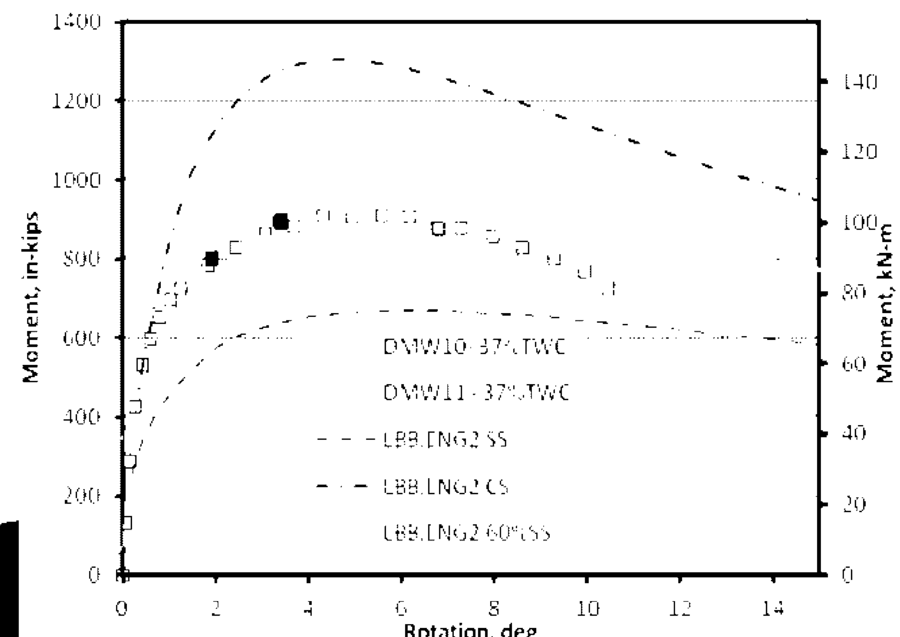
- Most stability calculation for cracks in weld were developed for base metal
  - DM weld connects stainless and carbon steel with nickel-based welds



- Modification of analyses needed
  - Used FE analyses for development
  - Used experiments for validation



LBB Regulation Guide

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Technical basis for LBB Regulatory Guide (NUREG/CR-6765) was published in May 2002

- Suggested tiered approach to LBB
- Draft Regulation guide followed

With the occurrence of PWSCC in previously approved LBB lines, LBB Regulation Guide was put on hold

PWSCC and LBB

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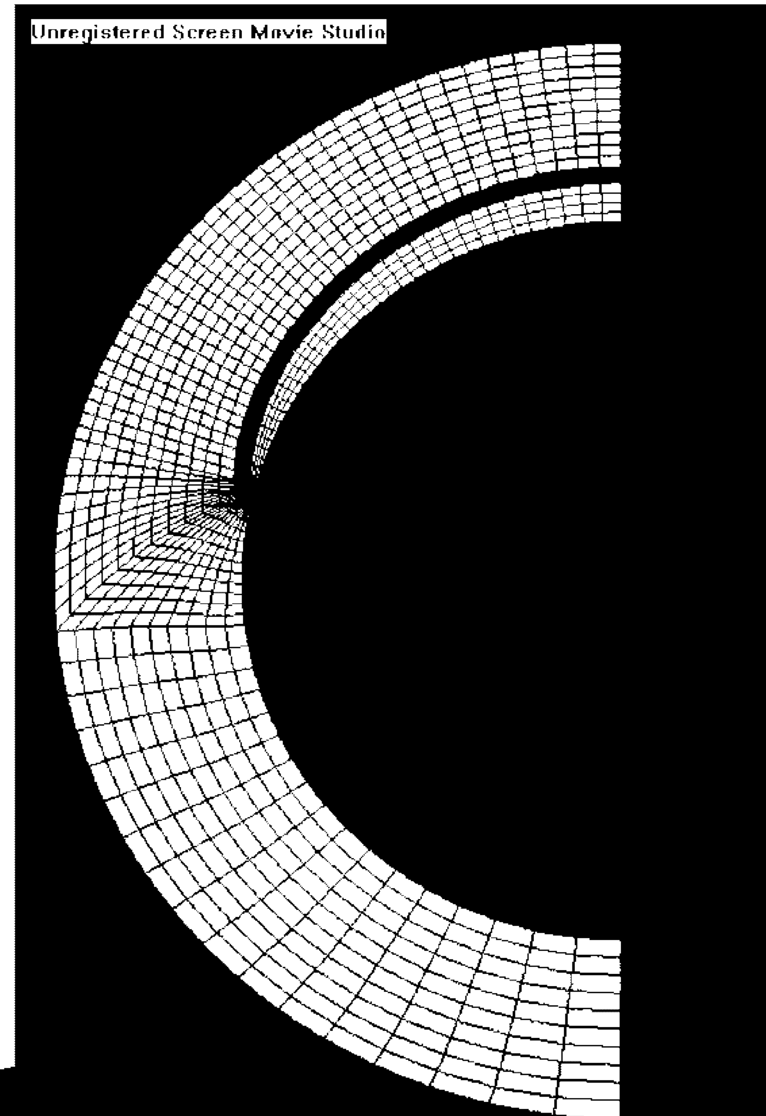
SRP 3.6.3 stipulates that no active degradation is allowed

LBB analyses assumes idealized through-wall crack for leakage and stability calculations – Flaws could grow non-idealized

Current LBB analyses may be non-conservative for this type of behavior



PWSCC Experiments

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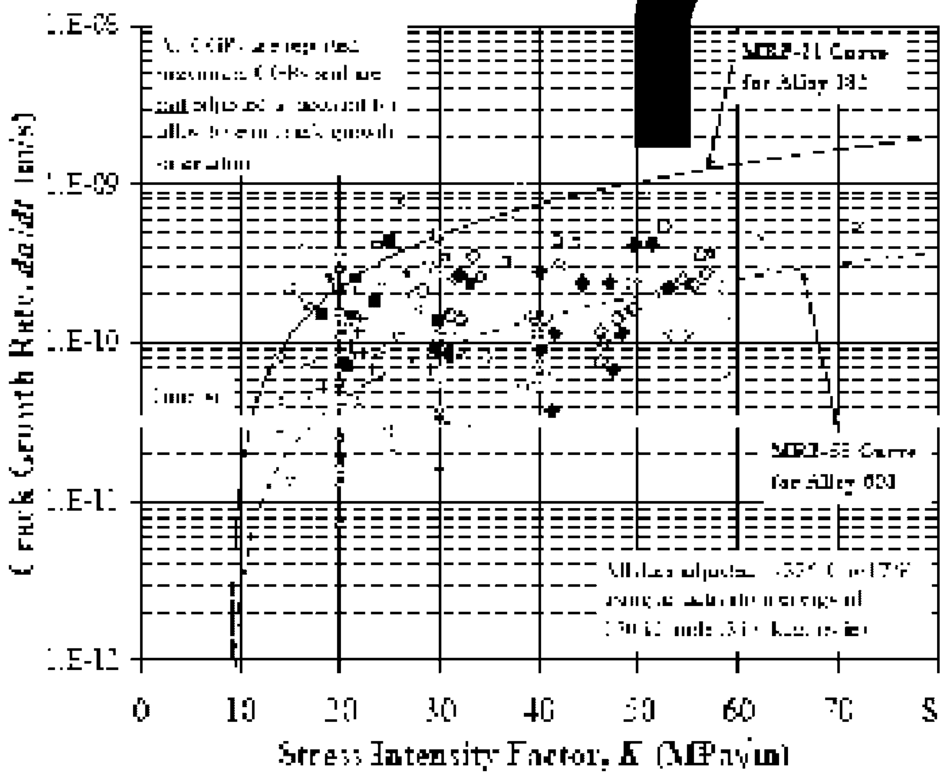
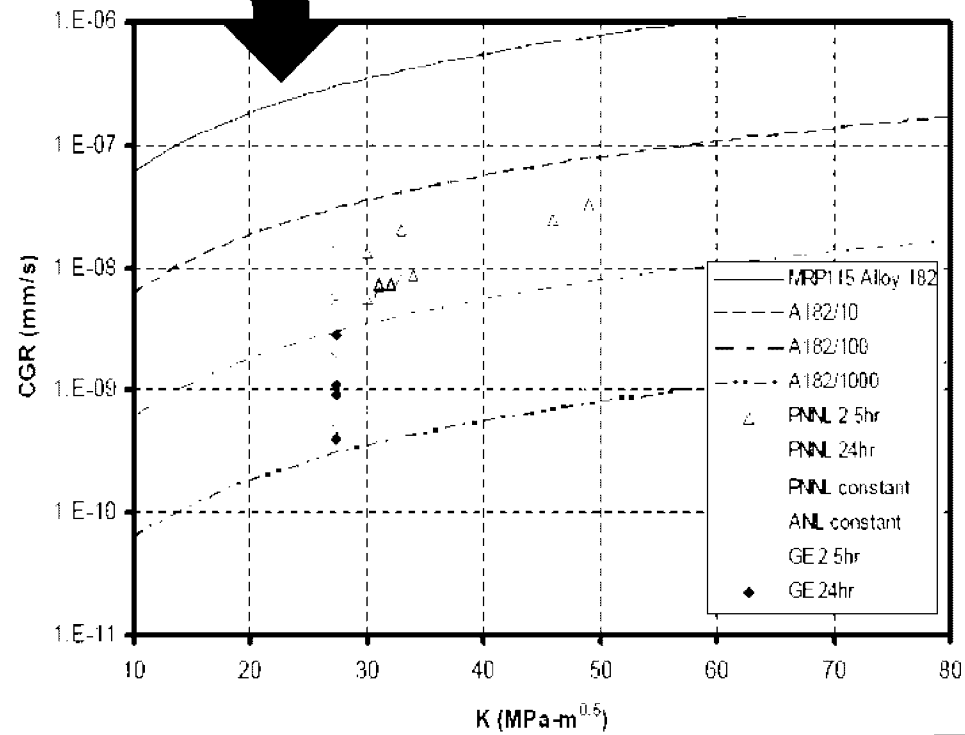


Alloy 82/182



Alloy 52/152

On average Alloy 52/152 crack growth is ~ 100 times slower than Alloy 82/182
Testing of Alloy 52/152 still underway!

Regulatory Impact

Due to PWSCC in susceptible butt welds, the industry developed and released MRP-139, which described the mitigation and inspection efforts to mitigate PWSCC

NRR released RIS2008-25, which stated that MRP-139 provided adequate protection of public health and safety for addressing PWSCC in butt welds for the ***near term***

NRR released RIS2010-07, which reminded licensees that a weld overlay in a piping system approved for LBB may affect the design basis of the plant and may ... ent request ...

Long Term

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For the near term, PWSCC in LBB lines with mitigation and augmented inspection are acceptable

For the long term, quantitatively assess compliance with 10CFR50App-A GDC-4. Include the effects of active degradation, mitigation, inspection, leak detection, uncertainty, etc.

RES is developing the xLPR modular probabilistic fracture mechanics code in cooperation with EPRI through an addendum to the Memorandum of Understanding

vg*

• [redacted]

xLPR Timeline

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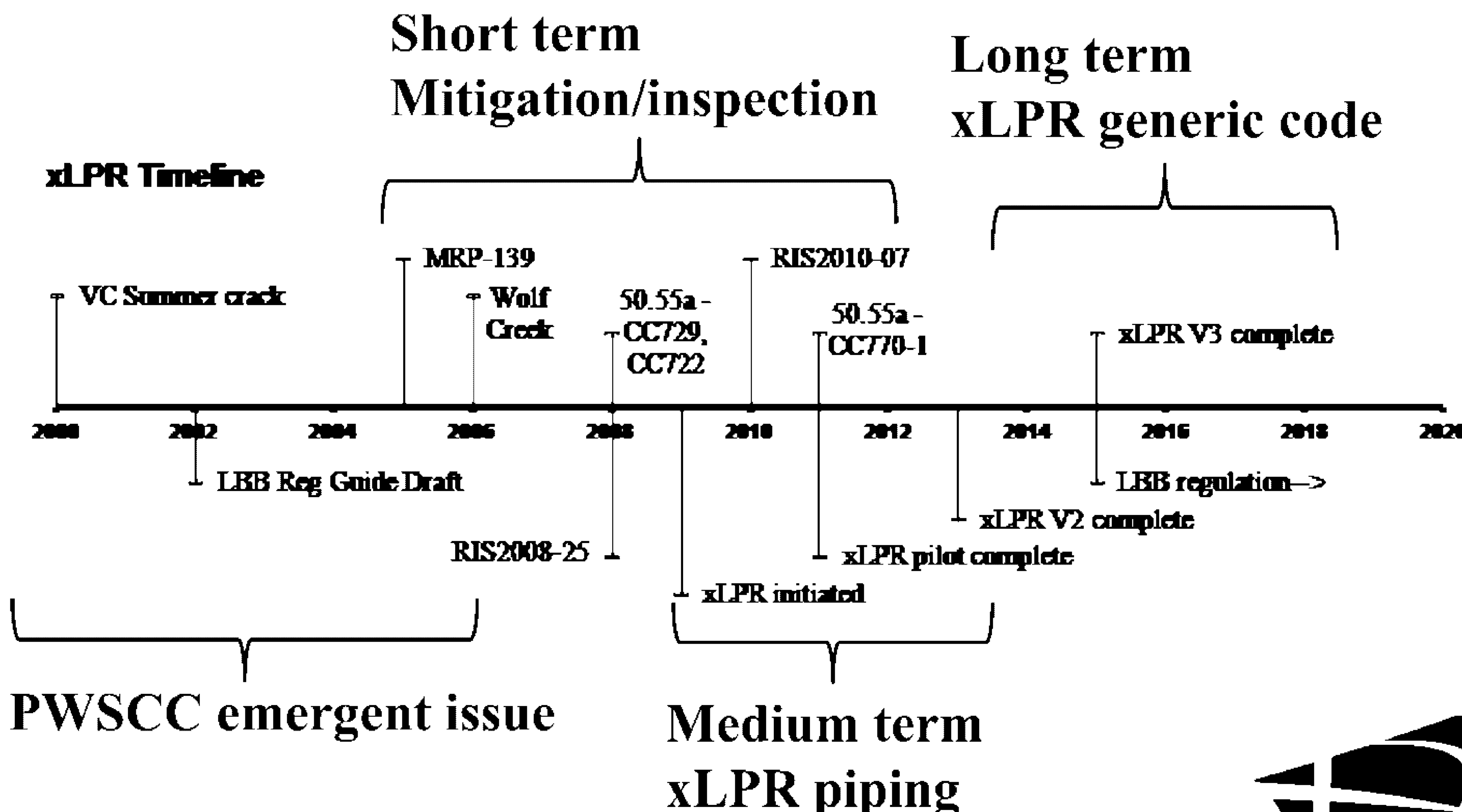
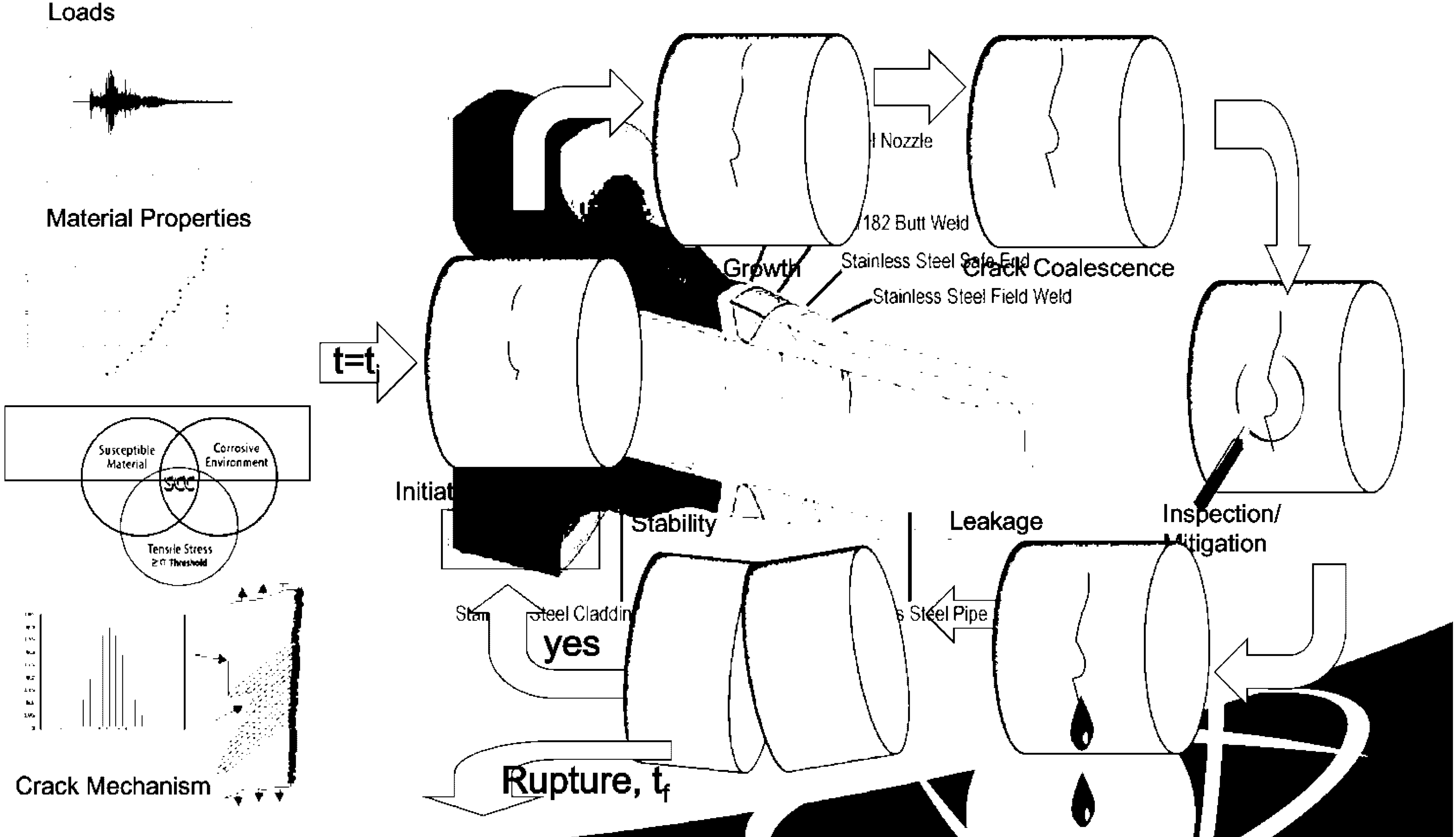


xLPR Technical Flow



Benefits of xLPR

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- Quantified solution to LBB issue
  - Regulation guide
  - Update to SRP3.6.3
- Fully QA'ed modular probabilistic fracture mechanics code for reactor pressure boundary integrity
  - LBB including evaluation of mitigation for DM welds
  - Research tool for prioritization
  - TBS – 50.46a
  - Risk informed ISI
  - GSI 191
  - Easily adaptable to other applications
    - CRDM ejection probabilities
    - RPV

Path Forward

Version 2.0 Development underway

Ongoing meetings

- ACRS meeting - March 2012 (yearly updates to subcommittee)
- NRC and EPRI Management (as needed)
- External reviews (annually)
- Internal reviews (bi-annually)

Version 2.0 release – End 2013

Technical basis and Regulatory Guide for LBB - 201[illegible]

Questions?

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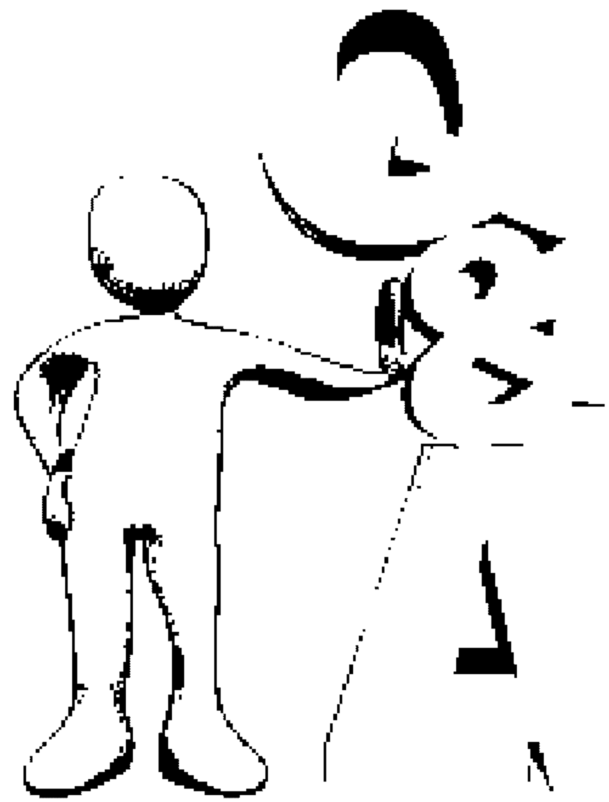


N O In-Depth Series (IDS Presentation

E vironmentally Assi ted Fatigue

Gary L. S vens

nior Mate s Engineer

CIB

(currently on tion to N

Tu day, July 16, 2013

09:00 – 10:00

U.S.NRC

United States Nuclear Regulatory Commission

Protecting People and the Environment

Objectives

- In this presentation, you will learn:
  - What is fatigue?
  - How is fatigue measured?
  - What is environmentally assisted fatigue (EAF)?
  - Why is the NRC interested in EAF?
    - Background - where did EAF requirements come from and why?
  - Methodology for EAF assessment
  - Current NRC requirements
    - License Renewal
    - New Reactors
  - Future NRC Requirements
  - What is the ASME Code doing on EAF?
  - Other questions

What is Fatigue?

- ASTM Specification No. E 1823-09a definition*:
  - *"The process of progressive localized permanent structural change occurring in a material subjected to conditions that produce fluctuating stresses and strains at some point or points and that may culminate in cracks or complete fracture after a sufficient number of fluctuations."*
- There is controversy associated with the definition of fatigue with respect to nuclear power plant design
  - Does "fatigue failure" mean crack initiation? If so, how deep is the "initiated crack"?
  - Or, does "fatigue failure" mean through-wall a crack that leaks?
  - Etc.
- NRC position:
  - *"Based on the results of the majority of the test data evaluated, fatigue life is defined as the number of cycles of a specified character that a given specimen sustains before the formation of a specified size crack (i.e., an "engineering crack"). A fatigue cumulative usage factor (CUF) less than unity provides reasonable assurance that no crack has been formed, and that the probability of forming a crack is low."*
  - NRC defines an "engineering crack" as an initiated fatigue crack with a depth of ~3 mm.

*** A simple and commonly under ood example of a ved by repeatedly bending a paper clip.**

How is Fatigue Measured?

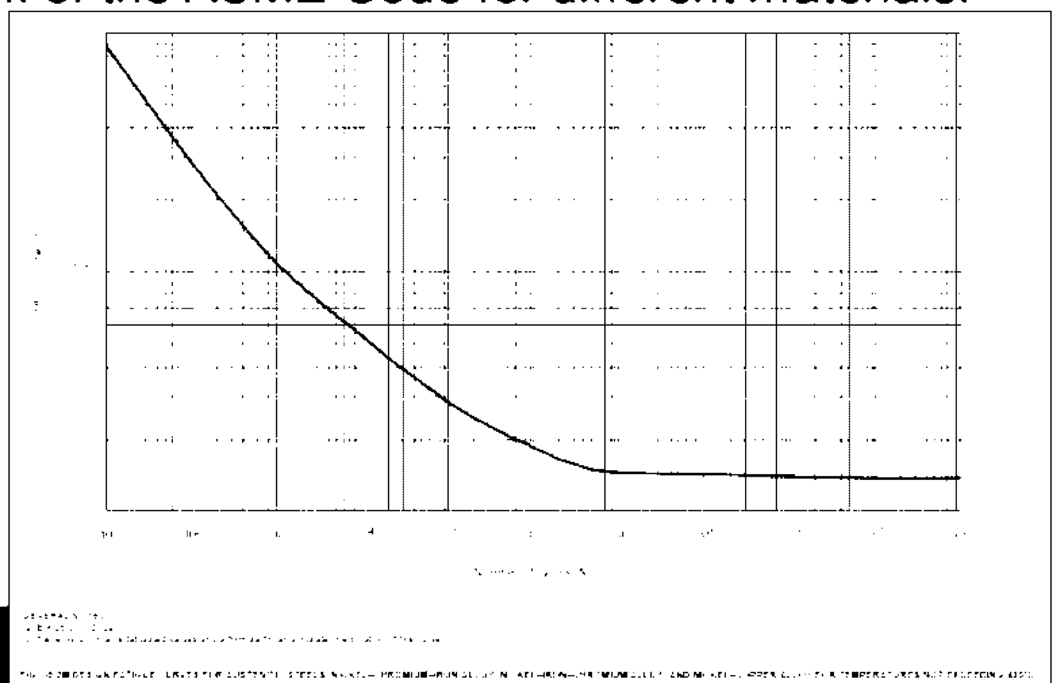
U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- For nuclear plant design, fatigue is "measured" (calculated) using a variable called "cumulative usage factor," or CUF:

$$CUF = \sum_i \frac{n}{N}$$

where: n is the applied number of cycles for load i
N is the allowable number of cycles for the stress associated with load i

- N is a function of the alternating stress, S_a, applied to a component, and is material dependent (i.e., it is a material property)
- S-N curves ("fatigue curves") are given in Mandatory Appendix I to Section III of the ASME Code for different materials:



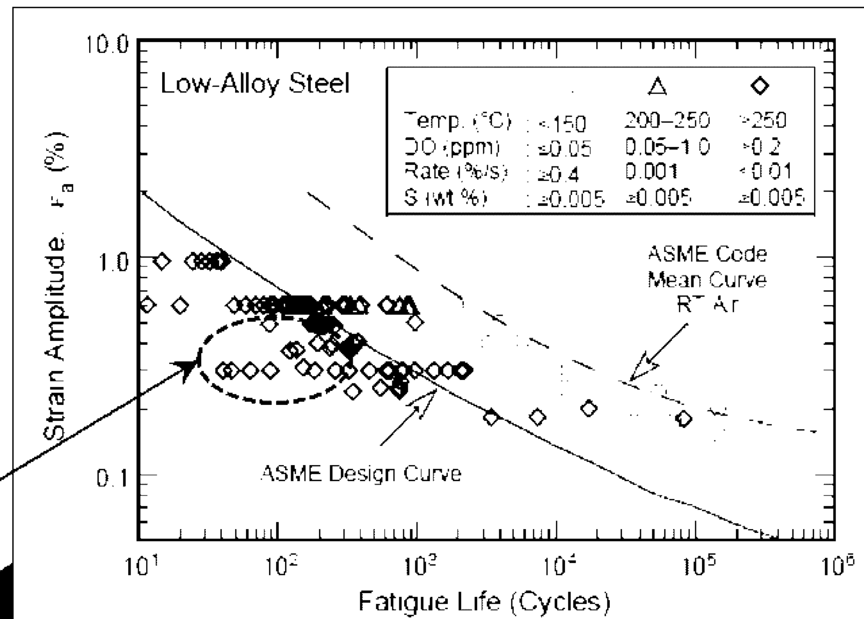
S-N curves are always defined in log -log form

What is Environmentally Assisted Fatigue (EAF)?

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- The fatigue curves in Section III of the ASME Code were developed from laboratory test data from specimens tested in **AIR**
- The **AIR** test data were used to develop design fatigue curves suitable for design:
  - Develop best fit log-log curves for the **AIR** data for each material type
  - Adjust the best fit curves to account for worst-case mean stress effects using the Modified Goodman relationship
  - Apply factors* of 2 on stress (S_a) or 20 on cycles (N), whichever is more conservative, to develop **AIR** design curves for each material
- More recent laboratory testing of specimens tested in **WATER** indicated that the **AIR** design curves may not adequately define fatigue life for materials exposed to **WATER** environments:

*Note how some of the points for tests in **WATER** fall below the **AIR** design curve.*



* Factors to account for data sca er, size effects (i. cimens vs. large power plant components), su ce finish, a osp

Why is the NRC interested in EAF?

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- Around 1990, the NRC initiated a Fatigue Action Plan:
  - Focus on operating plant fatigue to address several outstanding technical concerns
  - Arose from early license renewal activities in the 1980s that uncovered potential technical issues for all operating plants
  - Four basic issues identified:
    - Older vintage plants
    - Environmental effects on fatigue curves
    - Generic Issue 78, "Monitoring of Design Basis Transient Fatigue Limits for Reactor Coolant Systems"
      - To determine whether transient cycle monitoring is necessary at operating plants
    - Actions when CUF exceeds 1.0
      - No current regulatory position
      - Flaw tolerance analysis

Why is the NRC interested in EAF? (cont'd)

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- Completion of Fatigue Action Plan (SECY-95-245):
  - Simplistic cycle counting not a good measure of CUF
  - Required no immediate action by nuclear plant operators to address environmental effects
  - Concurred that essentially all locations could be qualified by monitoring or alternate analysis
  - Alternate risk studies were performed by Nuclear Regulatory Research (NUREG/CR-6674, "Fatigue Analysis of Components for 60-Year Plant Life," June 2000)
    - Showed that a fatigue failure of piping systems is not a significant contributor to core damage frequency
    - Leakage probability increased significantly after 40 years of operation
  - Based on these conclusions, the U.S. NRC could not justify requiring backfit of environmental data to operating plants

Why is the NRC interested in EAF? (cont'd)

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- Completion of Fatigue Action Plan (SECY-95-245): (cont'd)
  - However, effects would have to be considered for some components for license renewal to address leakage concerns
  - Documented as Generic Safety Issue (GSI) 166, "Adequacy of Fatigue Life of Metal Components"
  - "*The staff will consider, as part of the resolution of GSI-166, ...the need to evaluate a sample of components with high fatigue usage, using the latest available environmental fatigue data..*"
  - Renumbered to GSI-190, "Fatigue Evaluation of Metal Components for 60-Year Plant Life" for license renewal
    - Renumbered to eliminate 40-year issue and only focus on 60-year issue

Methodology for EAF Assessment

- Initially, the NRC reviewed two methods for incorporating LWR effects; the second method was adopted:
  1. Develop new environmental fatigue curves
  2. Use of an environmental correction factor, F_{en}
- F_{en} is defined as the ratio of fatigue life in air at room temperature to the fatigue life in water <u>at the service temperature</u>:

$$F_{en} = N_{air}/N_{water}$$

- F_{en} is multiplicative to the calculated fatigue usage in air:

$$U_{en} = U_1 F_{en,1} + U_2 F_{en,2} \ldots\ldots U_n F_{en,n}$$

Methodology for EAF Assessment (cont'd)

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- Carbon steels: $F_{en} = \exp(0.632 - 0.101\ S^*T^*O^*R^*)$
- Low-alloy steels: $F_{en} = \exp(0.702 - 0.101\ S^*T^*O^*R^*)$
  - $S^* = 0.001$ $(S \leq 0.001\ \text{wt.\%})$
    $S^* = S$ $(S \leq 0.015\ \text{wt.\%})$
    $S^* = 0.015$ $(S > 0.015\ \text{wt.\%})$
  - $T^* = 0$ $(T < 150°C)$
    $T^* = (T - 150)$ $(150 < T \leq 350°C)$
  - $O^* = 0$ $(DO \leq 0.04\ \text{ppm})$
    $O^* = \ln(DO/0.04)$ $(0.04 < DO \leqq 0.5\ \text{ppm})$
    $O^* = \ln(12.5)$ $(DO > 0.5\ \text{ppm})$
  - $R^* = 0$ $(R > 1\%/s)$
    $R^* = \ln(R)$ $(0.001 \leq R \leq 1\%/s)$
    $R^* = \ln(0.001)$ $(R < 0.001\%/s)$
- Note that there is an F_{en} of ≈2 even at temperatures below 150°C a very high strain rates; this seems inconsistent with any mecha proposed for environmental fatigue

Methodology for EAF Assessment (cont'd)

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- Stainless steels: $F_{en} = \exp(0.734 - T'\,O'\,R')$
  - $T' = 0$ ($T < 150°C$)
    $T' = (T - 150)/175$ ($150 < T \leq 325°C$)
    $T' = 1$ ($T \geq 325°C$)
  - $O' = 0.281$ (all DO levels)
  - $R' = 0$ ($R > 0.4\%/s$)
    $R' = \ln(R/0.4)$ ($0.001 \leq R \leq 0.4\%/s$)
    $R' = \ln(0.001)$ ($R < 0.001\%/s$)
- Again, an F_{en} of ≈ 2 even at temperatures below 150°C and very high strain rates seems inconsistent with any mechanism proposed for environmental fatigue

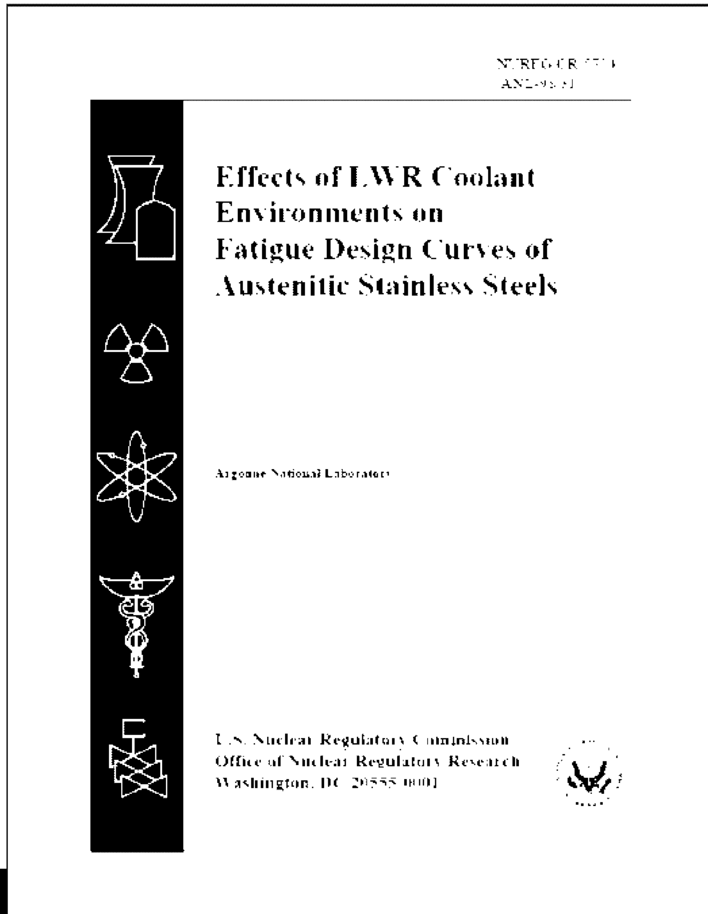
Methodology for EAF Assessment (cont'd)

- Ni-Cr-Fe steels: $F_{en} = \exp(-T' O' R')$
  - $T' = T/325$ $(T < 325°C)$
    $T' = 1$ $(T \geq 325°C)$
  - $O' = 0.09$ (NWC BWR water)
    $O' = 0.16$ (PWR or HWC BWR water)
  - $R' = 0$ $(R > 5.0\%/s)$
    $R' = \ln(R/5.0)$ $(0.0004 \leq R \leq 5.0\%/s)$
    $R' = \ln(0.0004/5.0)$ $(R < 0.0004\%/s)$

Current NRC Requirements – License Renewal

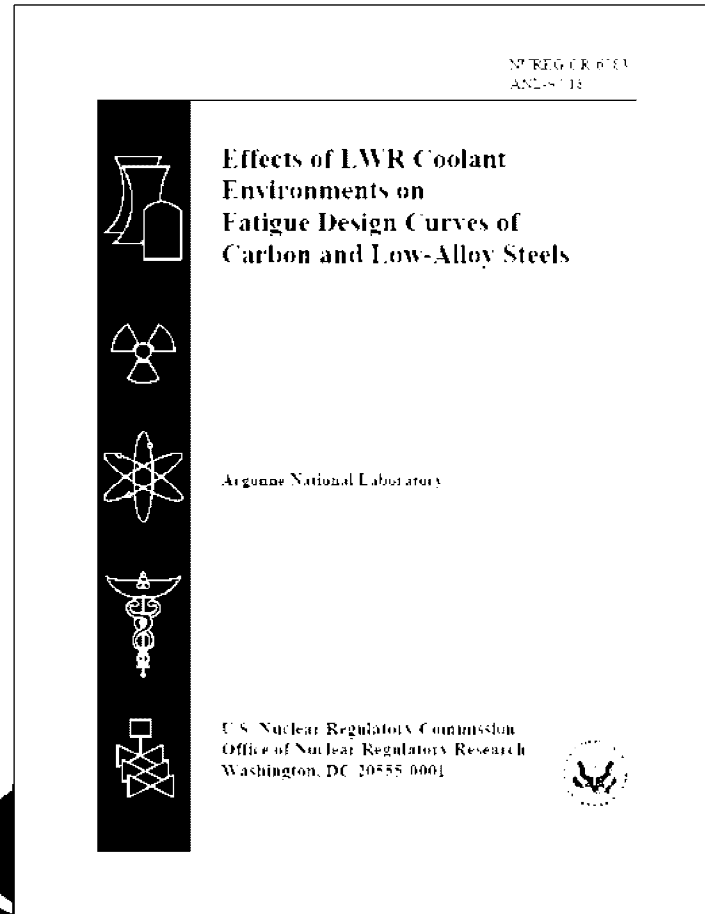
Protecting People and the Environment

- GALL Report (NUREG-1801, Chapter X.M1)
- NUREG/CR-5704 for stainless steels and NUREG/CR-6583 for ferritic steels
  (may also use new reactor requirements – next slide)


Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels

Argonne National Laboratory

U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Washington, DC 20555-0001


Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels

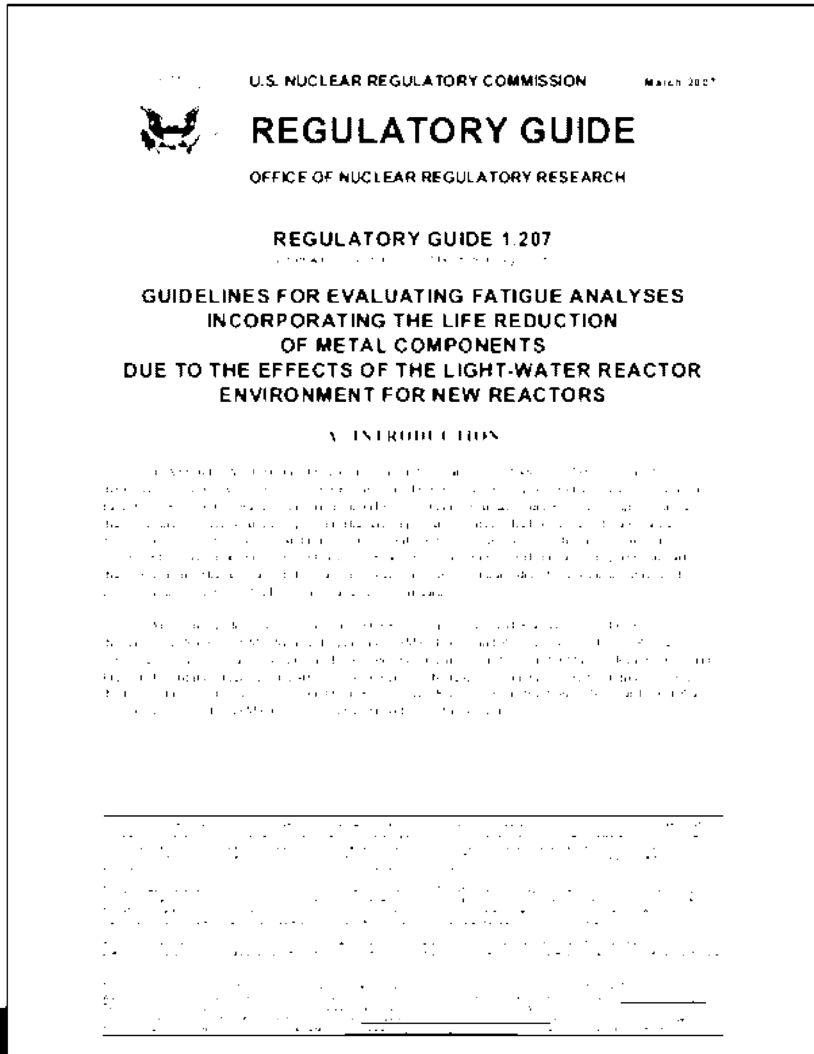
Argonne National Laboratory

U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Washington, DC 20555-0001

Current NRC Requirements – New Reactors

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- Regulatory Guide 1.207
- Supporting technical basis documented in NUREG/CR-6909



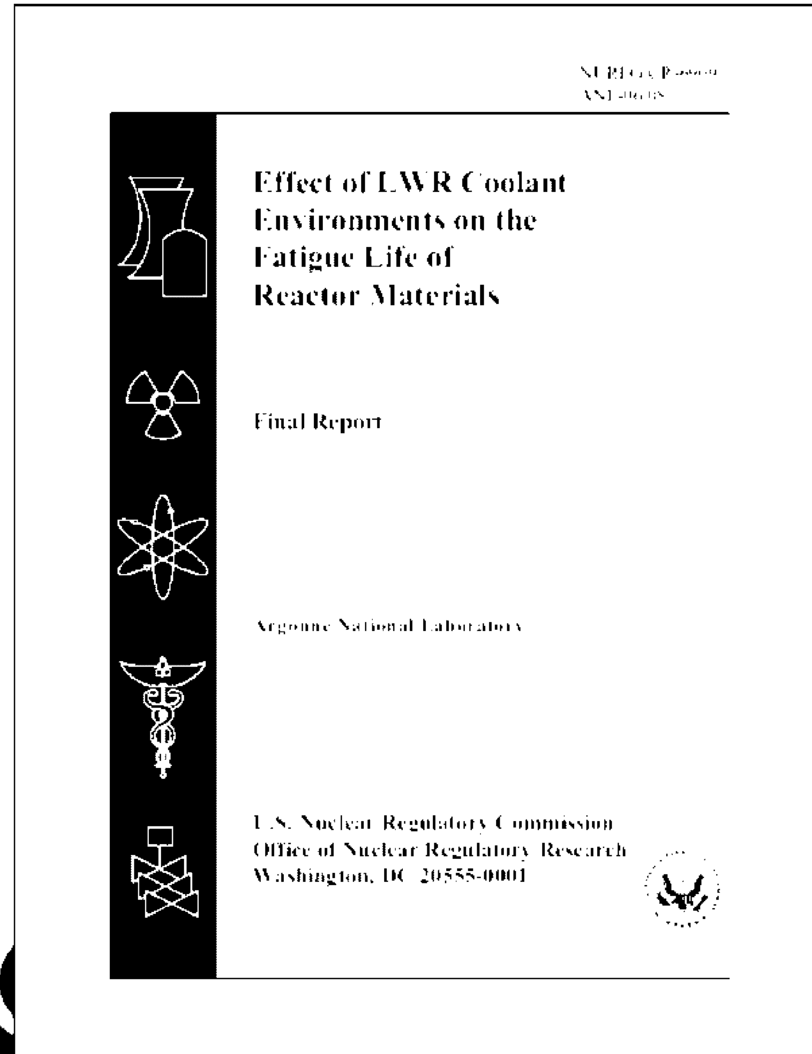
U.S. NUCLEAR REGULATORY COMMISSION

REGULATORY GUIDE

OFFICE OF NUCLEAR REGULATORY RESEARCH

REGULATORY GUIDE 1.207

GUIDELINES FOR EVALUATING FATIGUE ANALYSES
INCORPORATING THE LIFE REDUCTION
OF METAL COMPONENTS
DUE TO THE EFFECTS OF THE LIGHT-WATER REACTOR
ENVIRONMENT FOR NEW REACTORS



Effect of LWR Coolant
Environments on the
Fatigue Life of
Reactor Materials

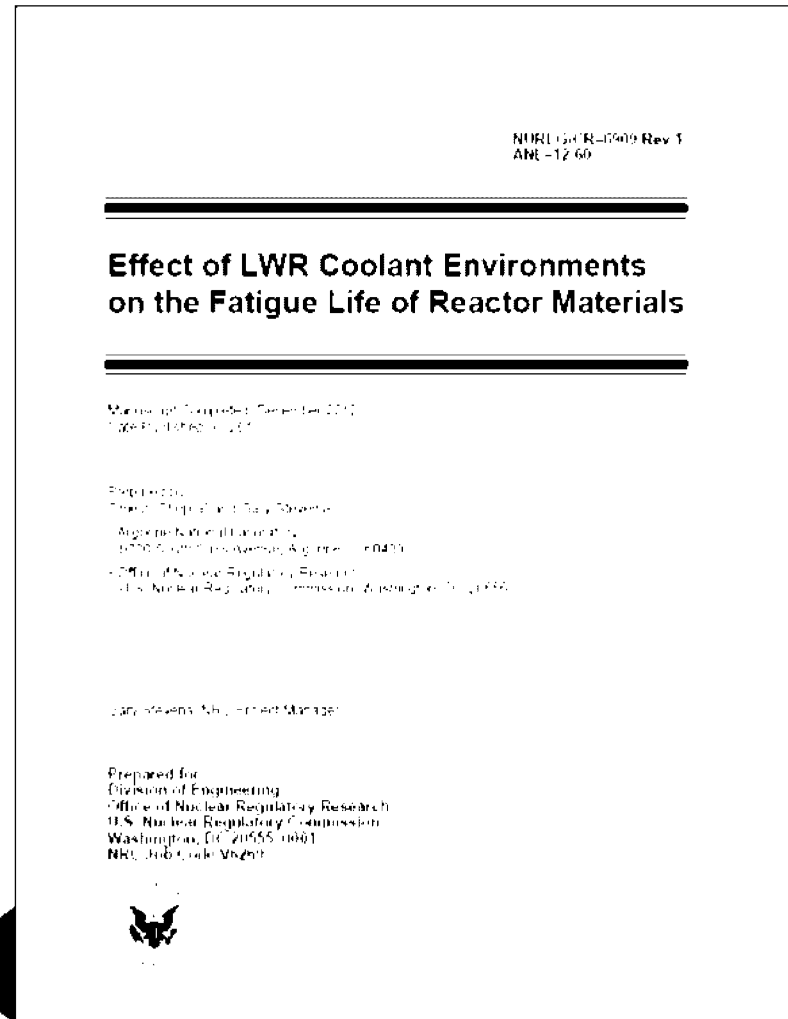
Final Report

Argonne National Laboratory

U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Washington, DC 20555-0001

Future NRC Requirements

- Revision to RG 1.207 and NUREG/CR-6909 currently underway:
  - Will apply to both operating and new reactors
  - Based on 2010-2013 NRC research activities
    - Dual User Need NRR-2010-019/NRO-2010-006
  - Both documents expected to go out for public comment in December 2013



Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials

Prepared for
Division of Engineering
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

What is the ASME Code doing on EAF?

- ASME has been struggling with this issue for more than 10 years
- From Section III, NB-3121, "Corrosion":
  - *Material subject to thinning by corrosion, erosion, mechanical abrasion, or other environmental effects shall have provision made for these effects during the design or specified life of the component by a suitable increase in or addition to the thickness of the base metal over that determined by the design formulas. Material added or included for these purposes need not be of the same thickness for all areas of the component if different rates of attack are expected for the various areas. It should be noted that the tests on which the design fatigue curves (Figs. I-9.0) are based did not include tests in the presence of corrosive environments which might accelerate fatigue failure.*

What is the ASME Code doing on EAF? (cont'd)

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- ASME Section III has published recommended methods for addressing environmental effects in two Code Cases:
  - Code Case N-761:
    - New fatigue design curves for LWR environments
  - Code Case N-792
    - Environmental fatigue correction factor F_{en} method
- NRC has not formally endorsed either Code Case
- Two other Section III Code Cases are under development:
  - Flaw Damage Code Case
    - Similar to ASME Code, Section XI, Nonmandatory Appendix L
  - Strain Rate Code Case
    - Provides methods for determining strain rate for the F_{en} Code Case

Other questions

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- 30+ years of industry experience and no thermal fatigue issues*. What's the big deal? Why are we imposing this stringent criteria on the industry?
- Is it true that the NRC is not imposing this environmental impact on fatigue uniformly to new reactors?
- Does EAF apply to the RCPB only?
- Is there a difference between BWR and PWR environments? Is it a bounding calculation?

Questions?

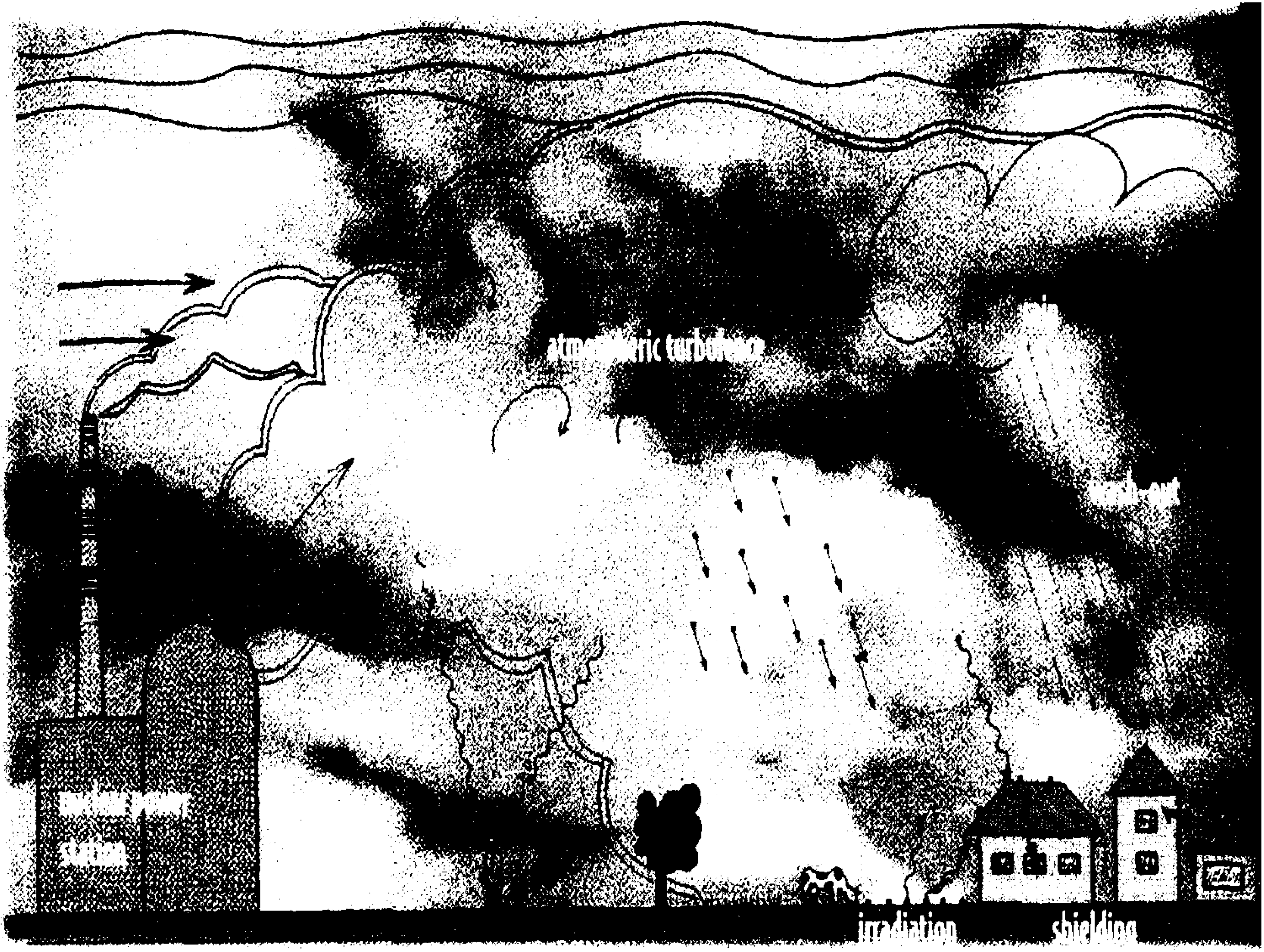
?

Containment Review

Thermal-Hydraulic and Source Term Issues

[redacted]

November 10, 2011

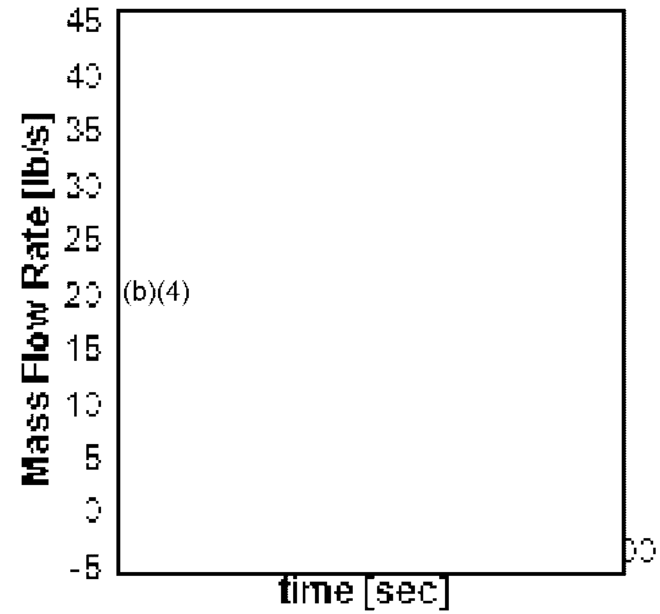

irradiation
shielding

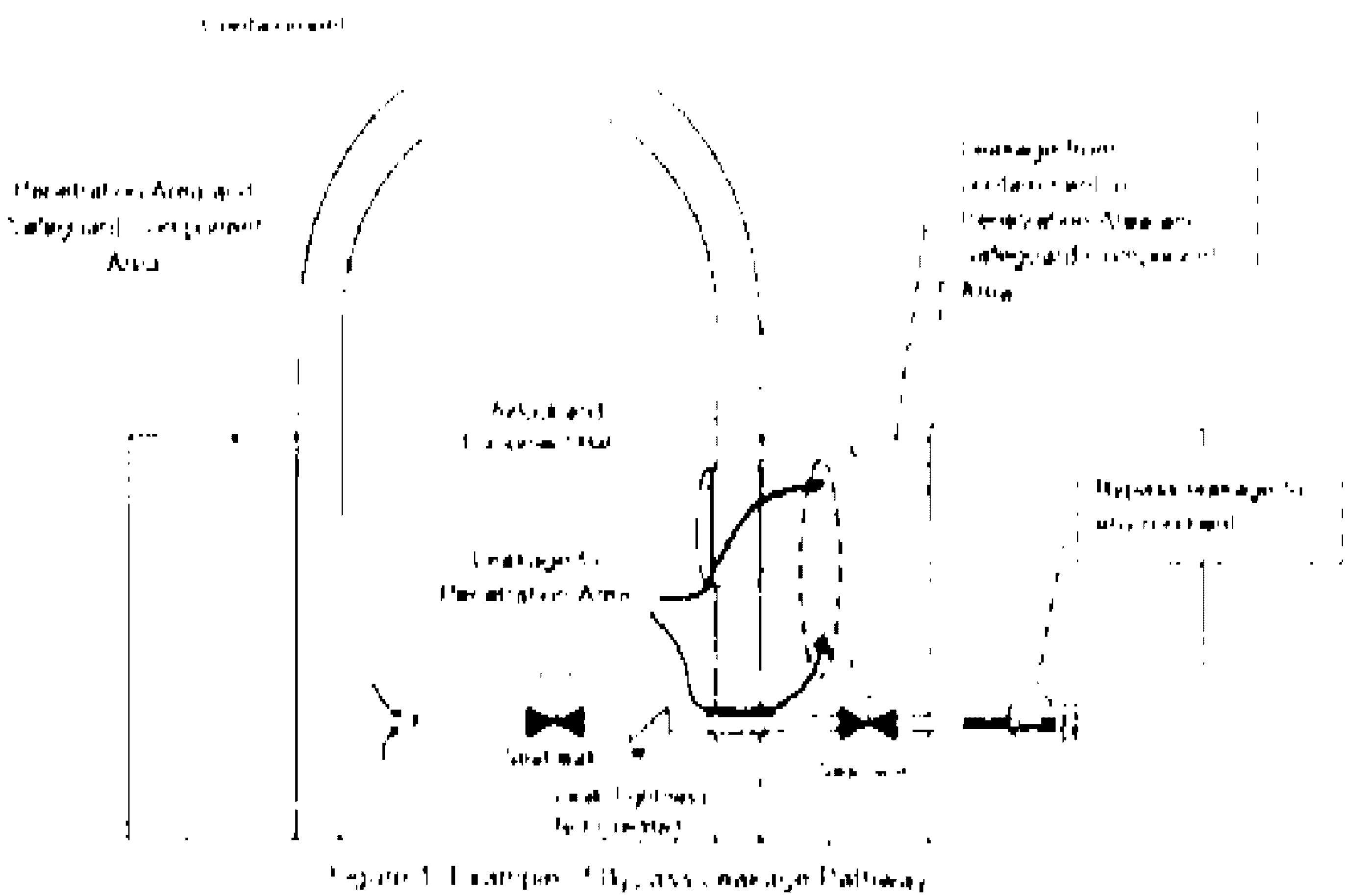
[redacted]

Main issue: containment integrity

- **Leakage (normal operation)**
- **Breach (abnormal / accident**

[redacted]





[redacted]

8” containment breach

[redacted]

NRC regulations for containment - examples:

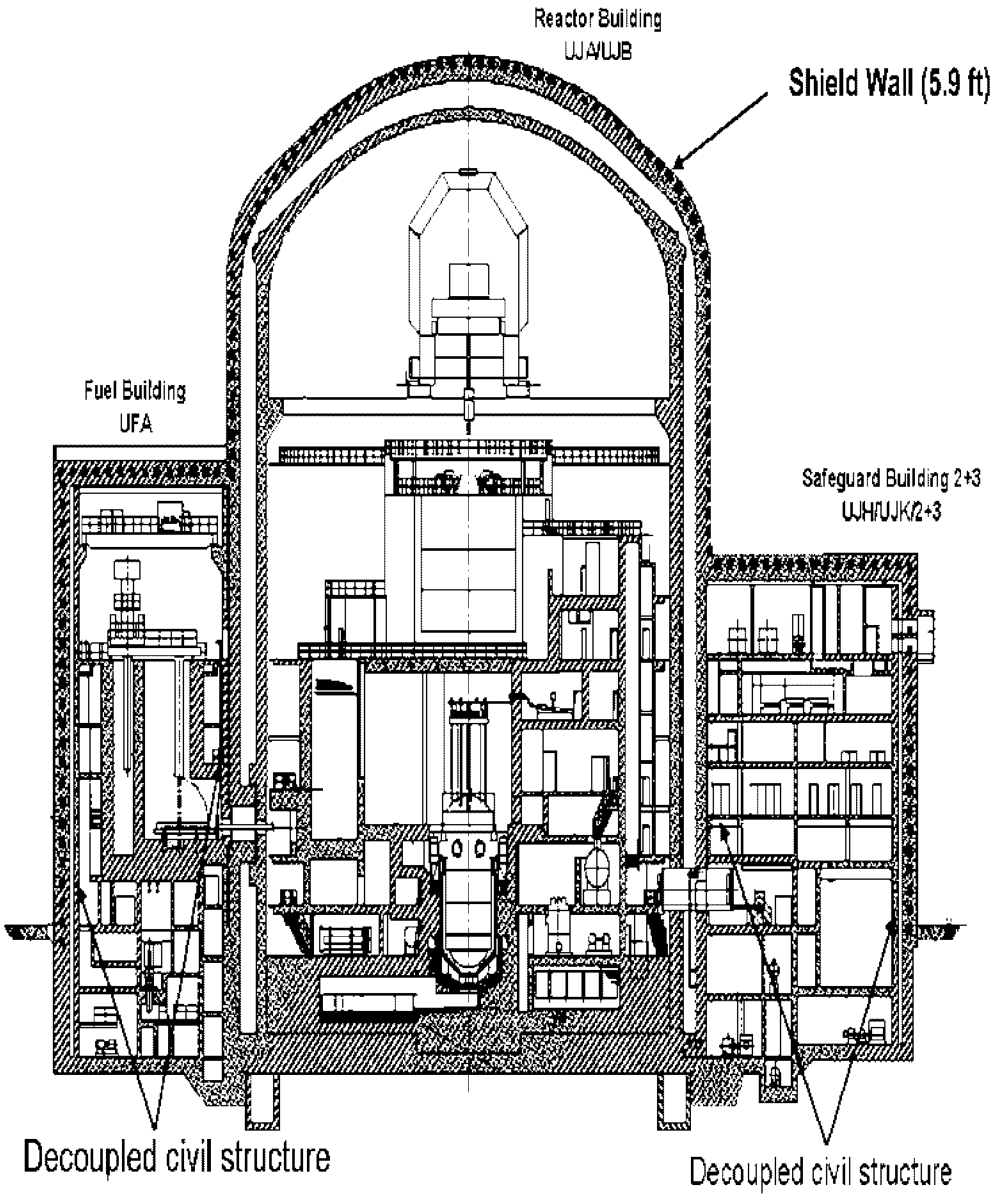
1. General Design Criterion (GDC) 16 - preserving containment integrity under conditions imposed by postulated LOCAs.

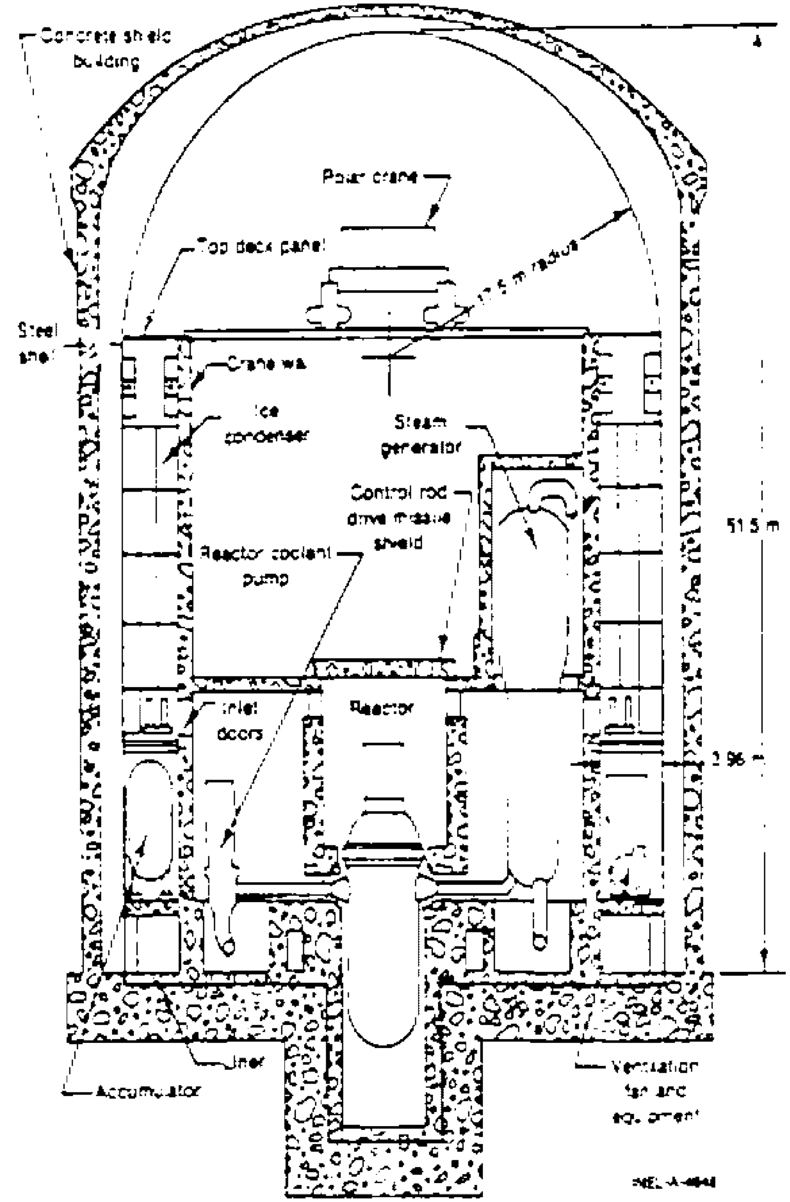
2. GDC 50 - accommodate the calculated P/T without exceeding the design leakage rate

3. GDC 38 – rapid reduction of containment P/T following any LOCA.

4. 10 CFR 50.34(f)(3)(v)(A)(1) - maintain containment integrity assuming H2 burning, as generated from a 100-percent fuel clad metal-water reaction.

5. 10 CFR 52.47(b)(1) – requires a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC)



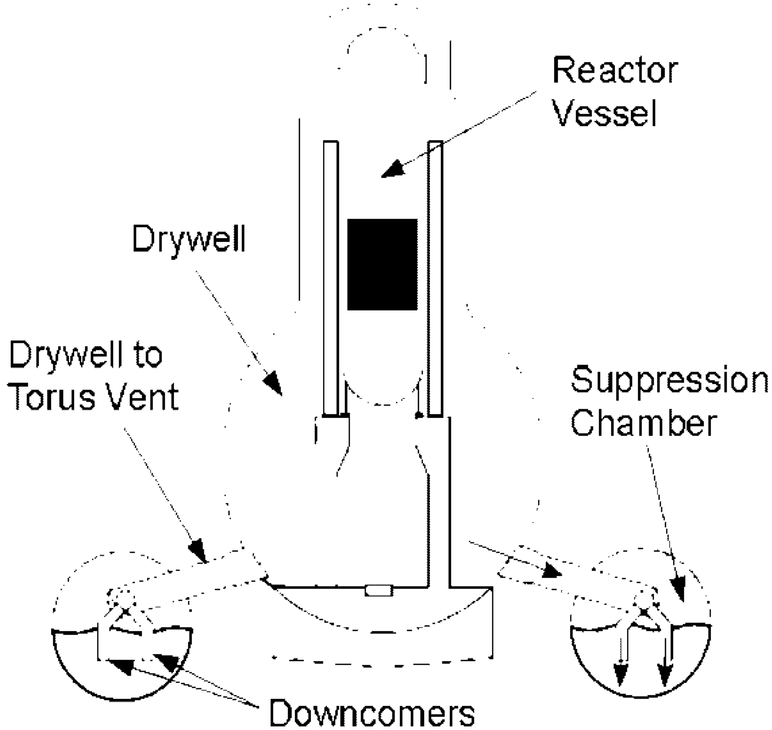


Ice condenser

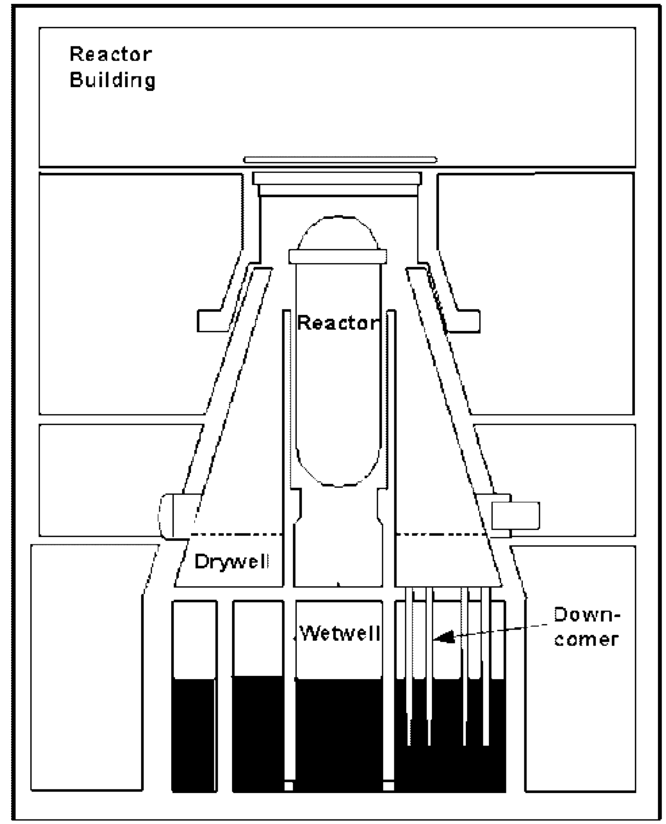
Large dry:
-single or double shell
-subatmopheric

Typical PWR designs

Typical BWRs

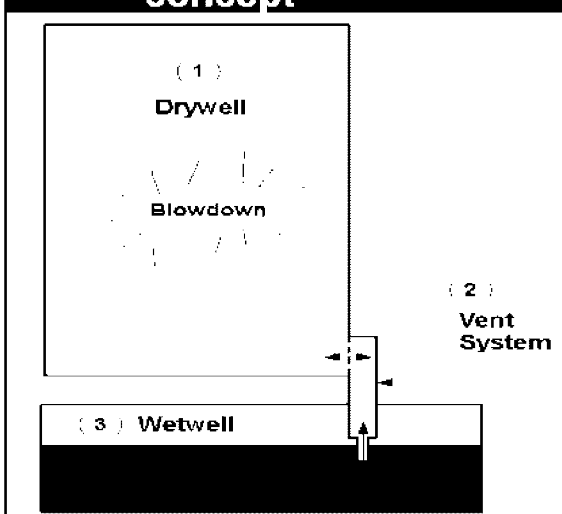


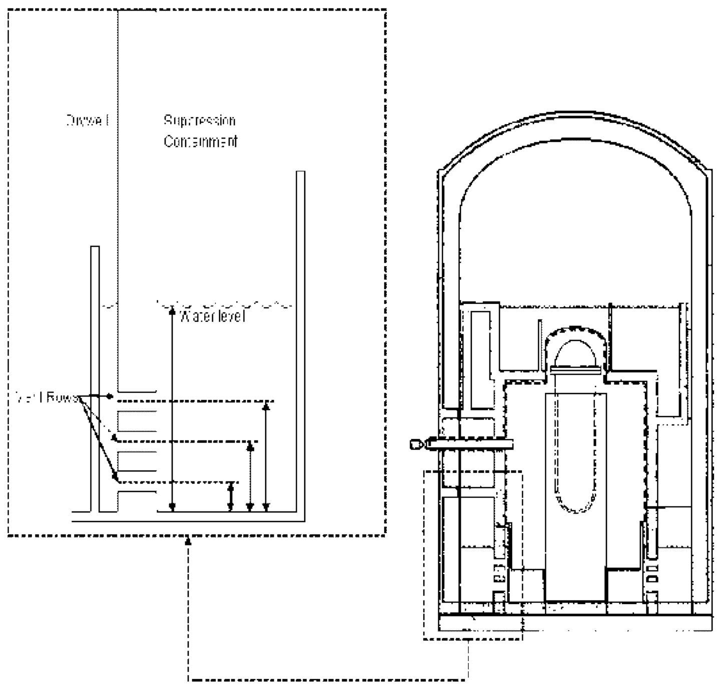
MK I



MK II

Pressure suppression concept





MK III

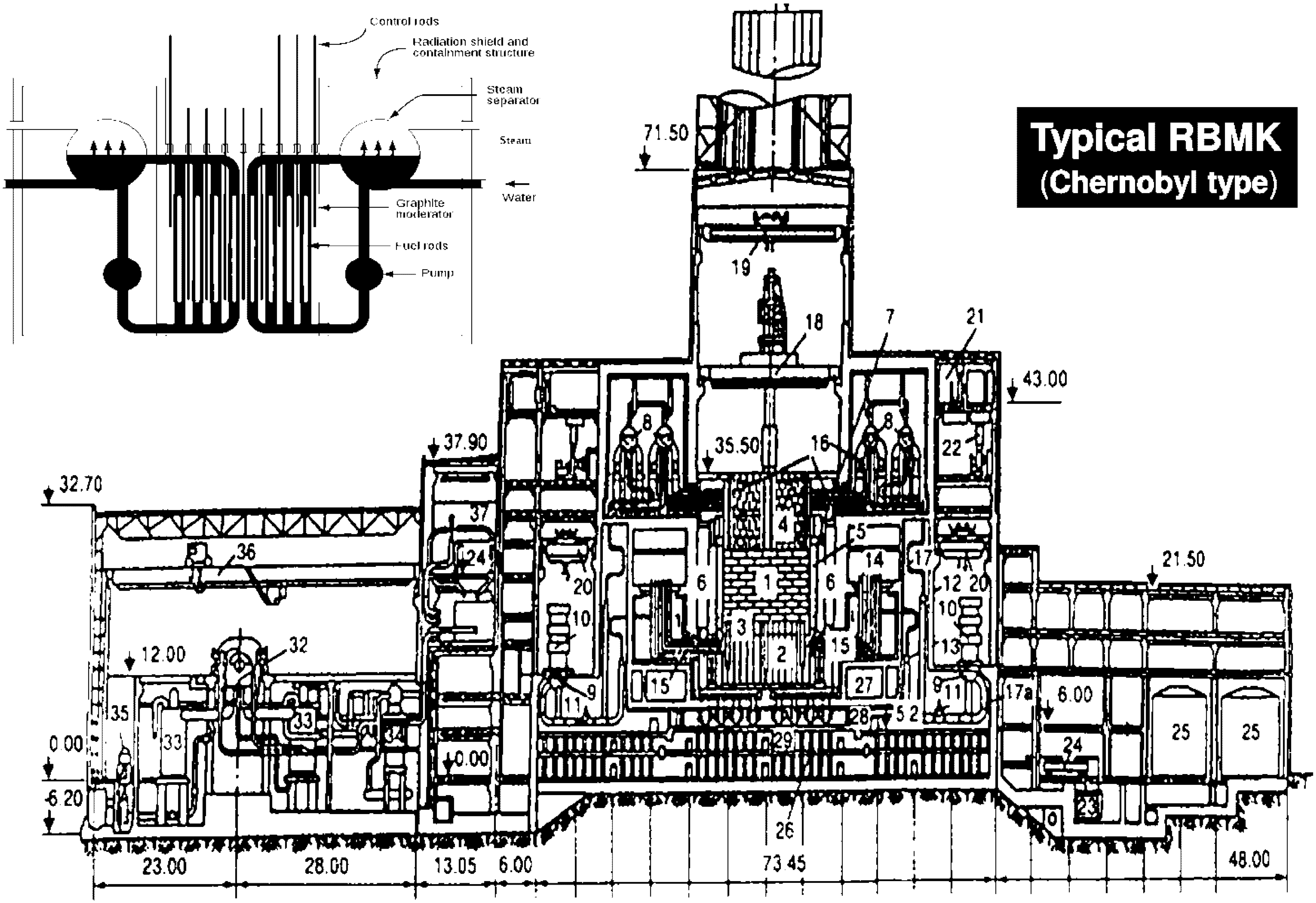
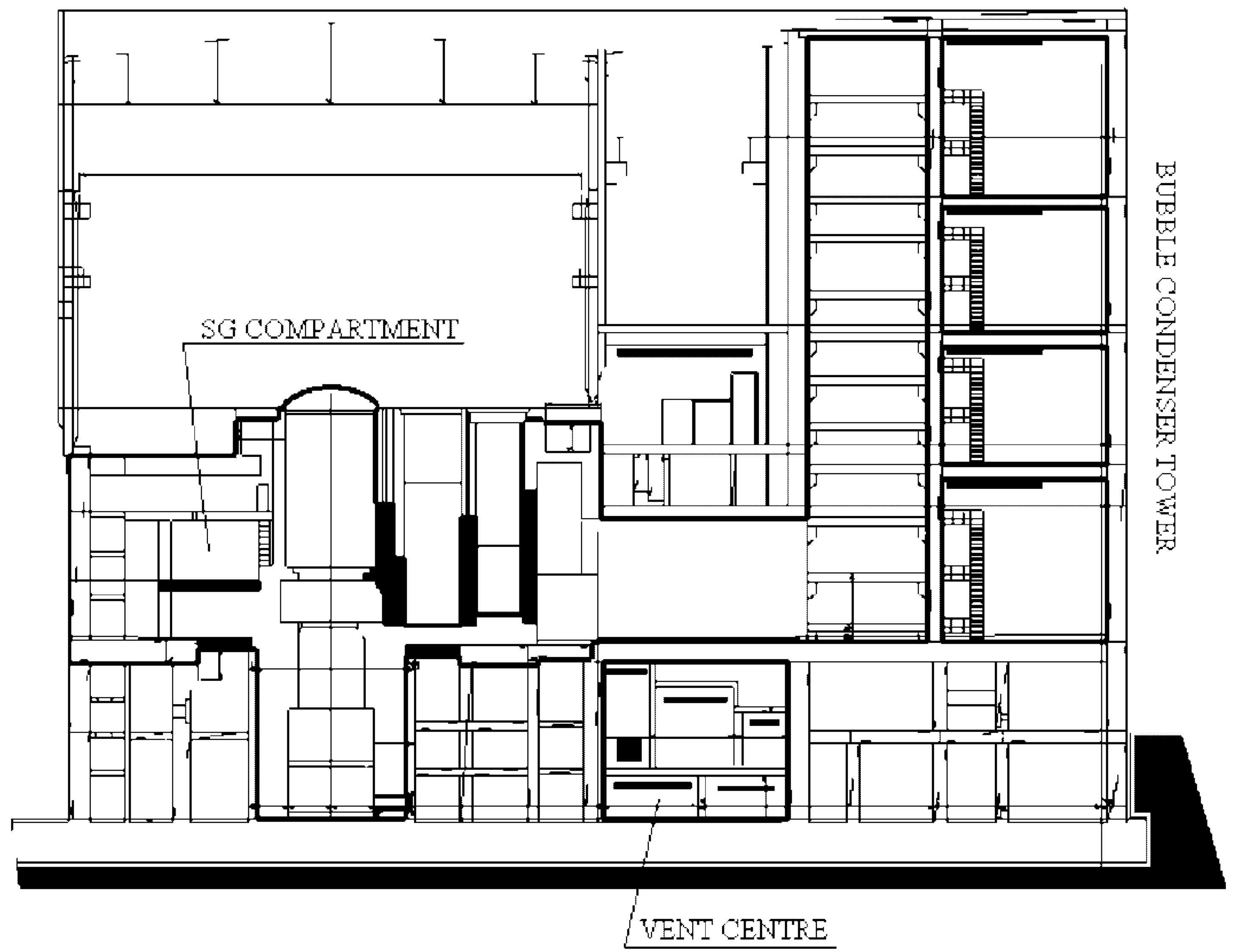
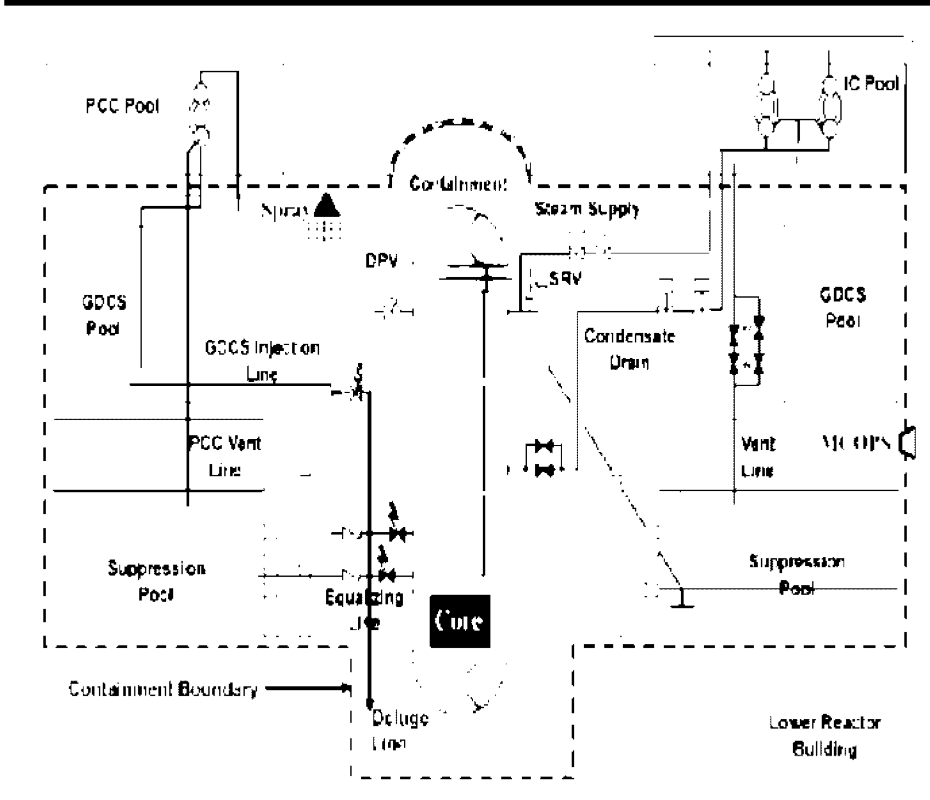


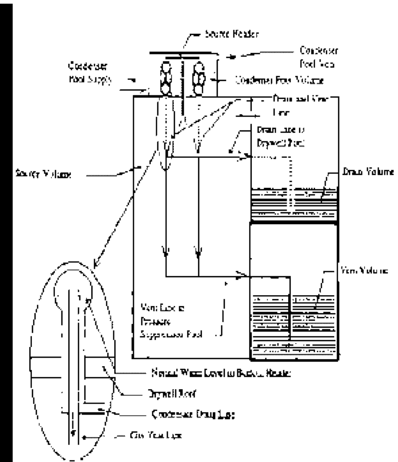
FIG. II-2. Vertical section through the main building of an RBMK unit, including the localization zone. [Numbers refer to itemization of equipment and components in Table II-II. Dimensions are given in metres.]

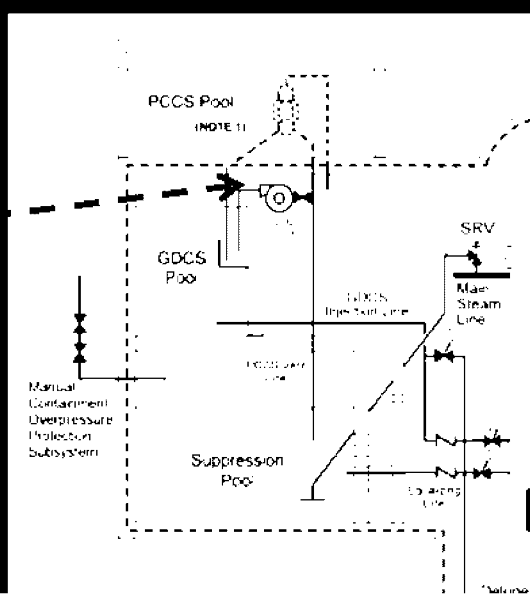


Typical VVER-440

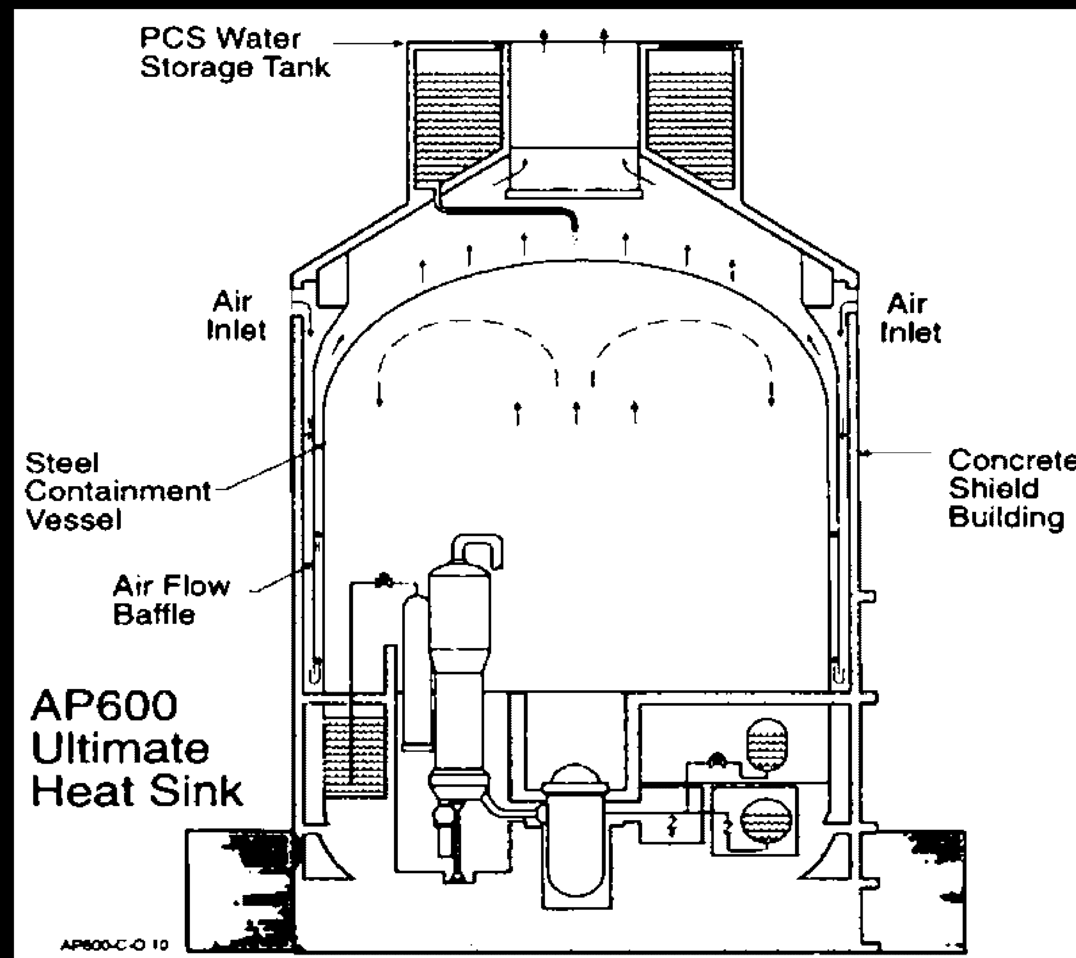
Passive systems



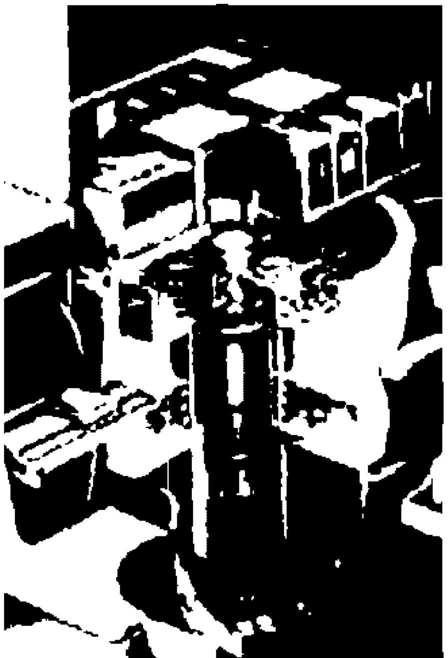




modified design

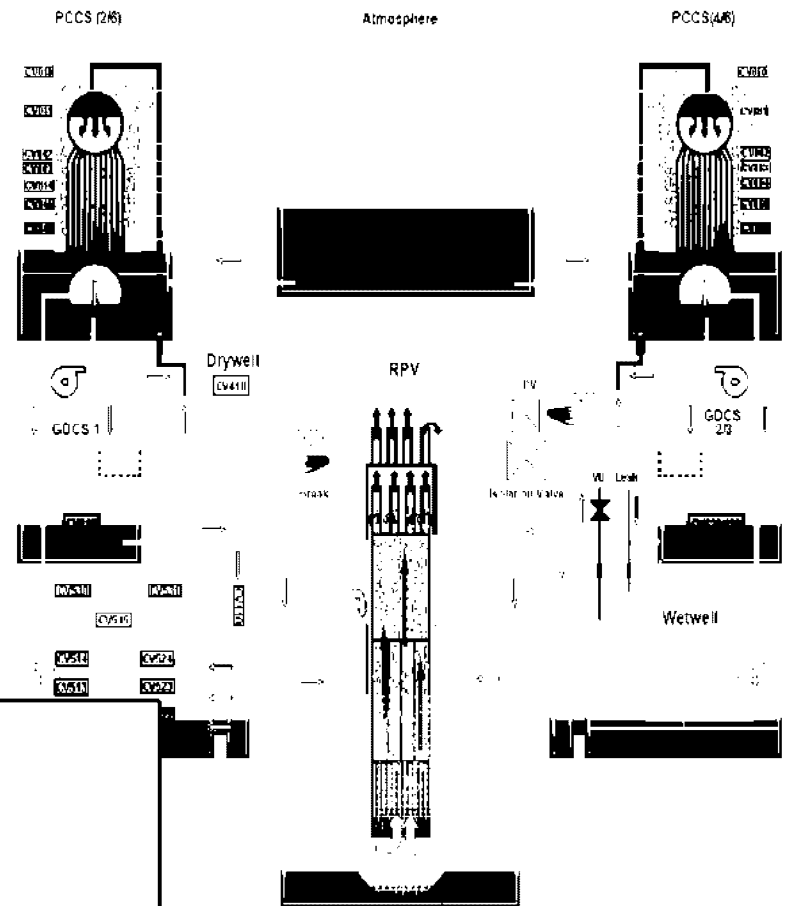


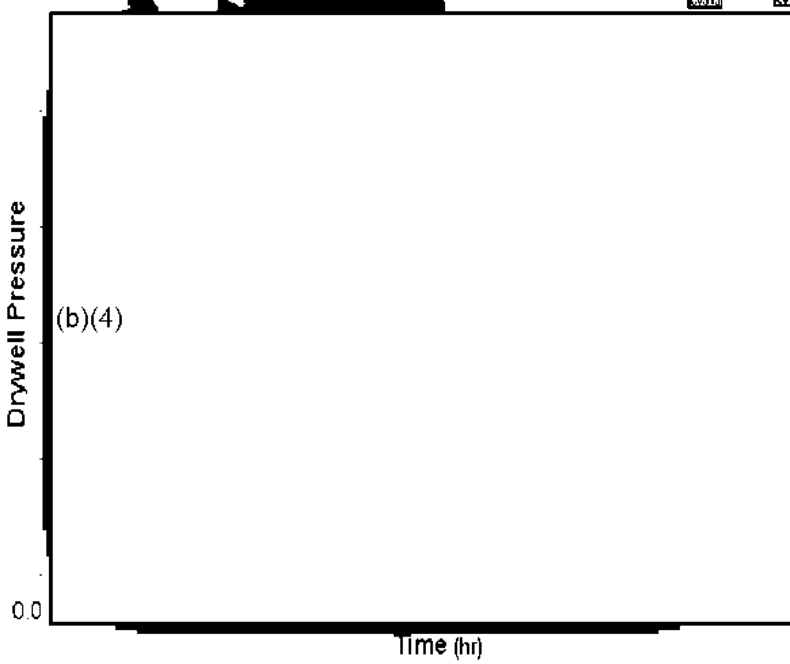
ESBWR Long Term Containment Cooling



MELCOR

MODEL





(b)(4)

SWR 1000 Passive Safety Concept

Containment cooling condensers for containment heat removal (4)

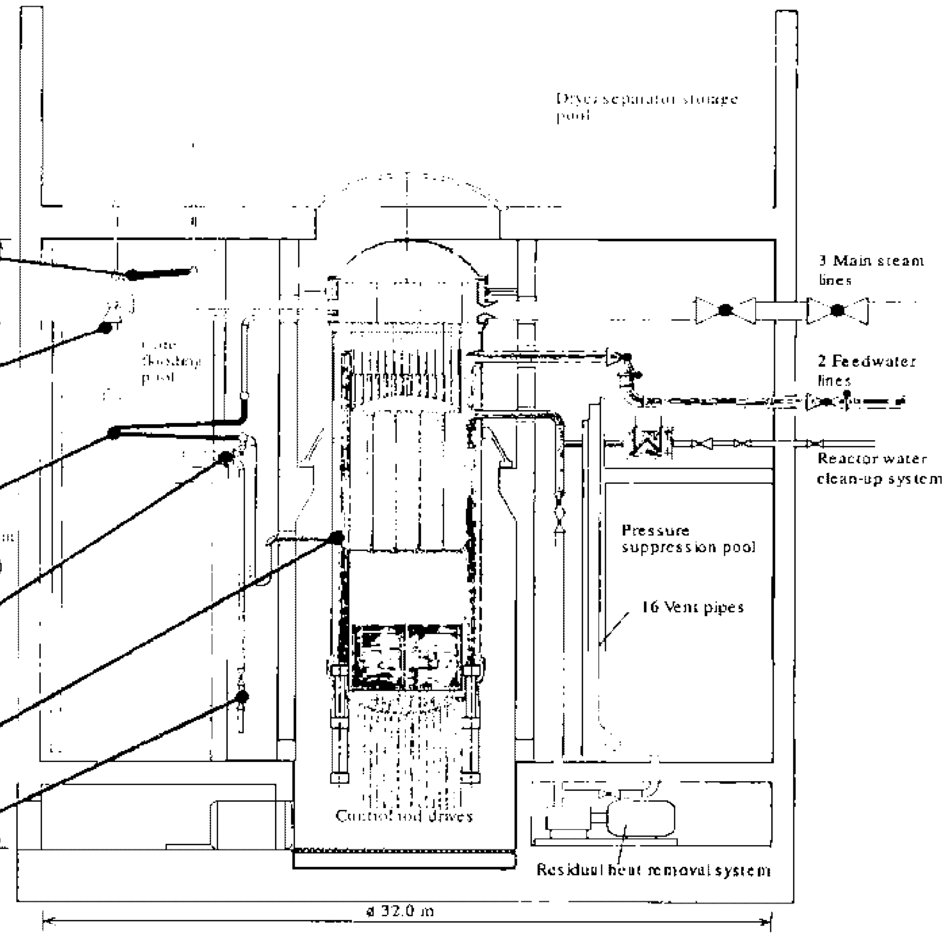
SRV for reactor over-pressure protection and reactor depressurization (8)

Emergency condensers for heat removal from the RPV (4)

Flooding lines for passive core flooding in the event of LOCA (4)

Passive Outflow Reducer (4)

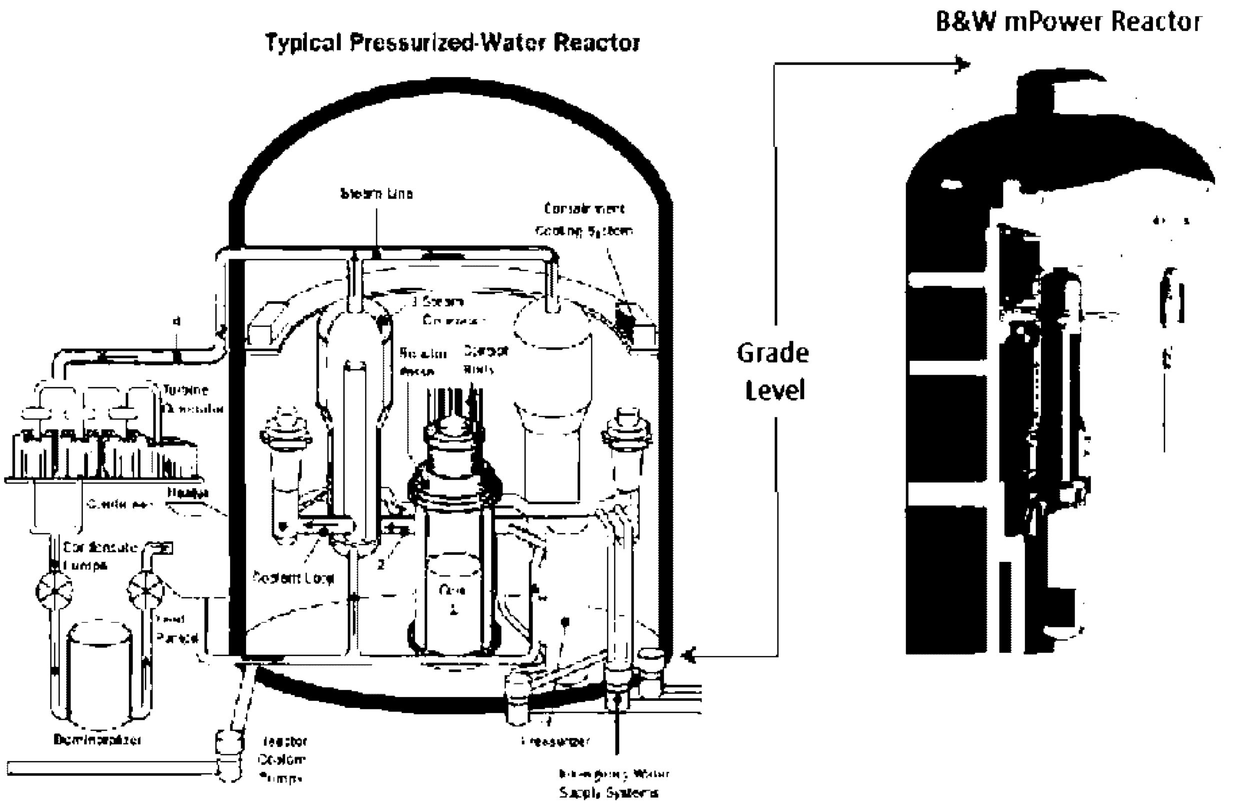
Drywell flooding line (event of a core melt accident)



FRAMATOME ANP

mPower

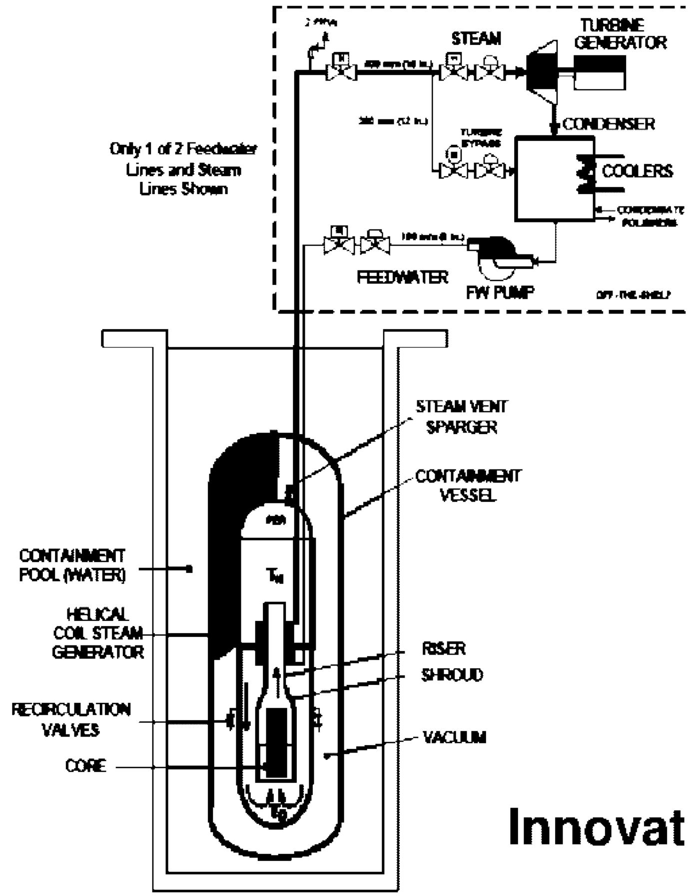
Traditional PWR versus B&W mPower Reactor

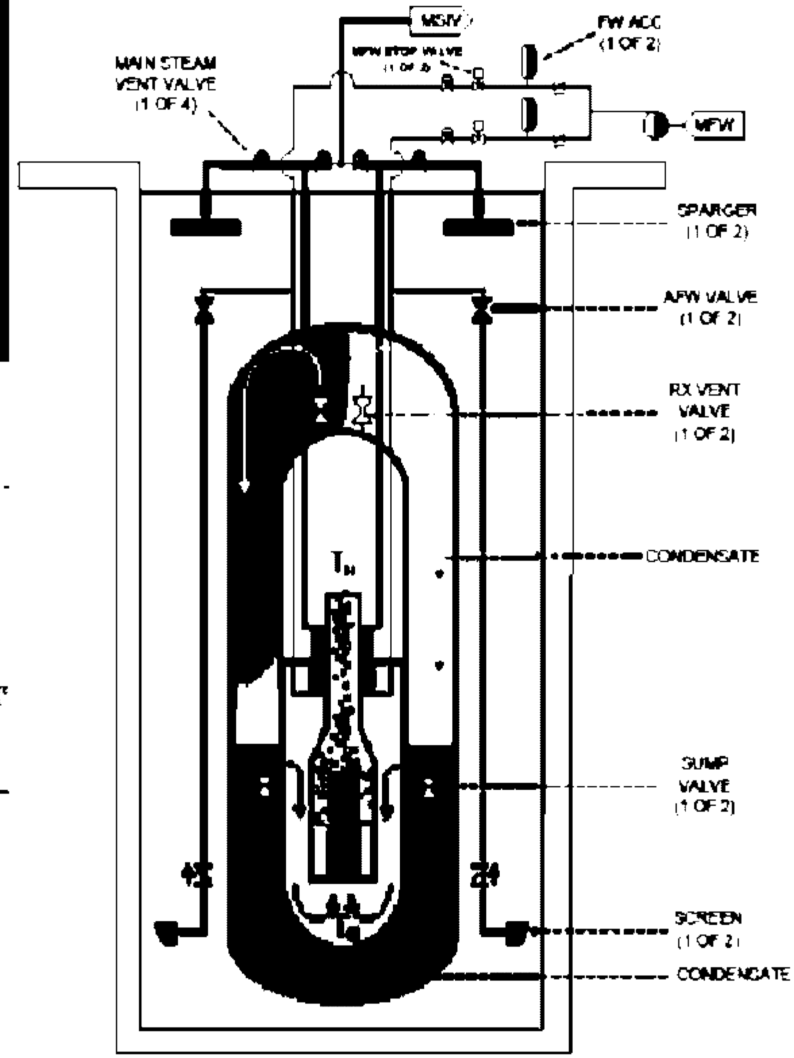


NuScale concept

Power Module

- Integrated Reactor Vessel enclosed in an air evacuated Containment Vessel immersed in a pool of water located below grade.

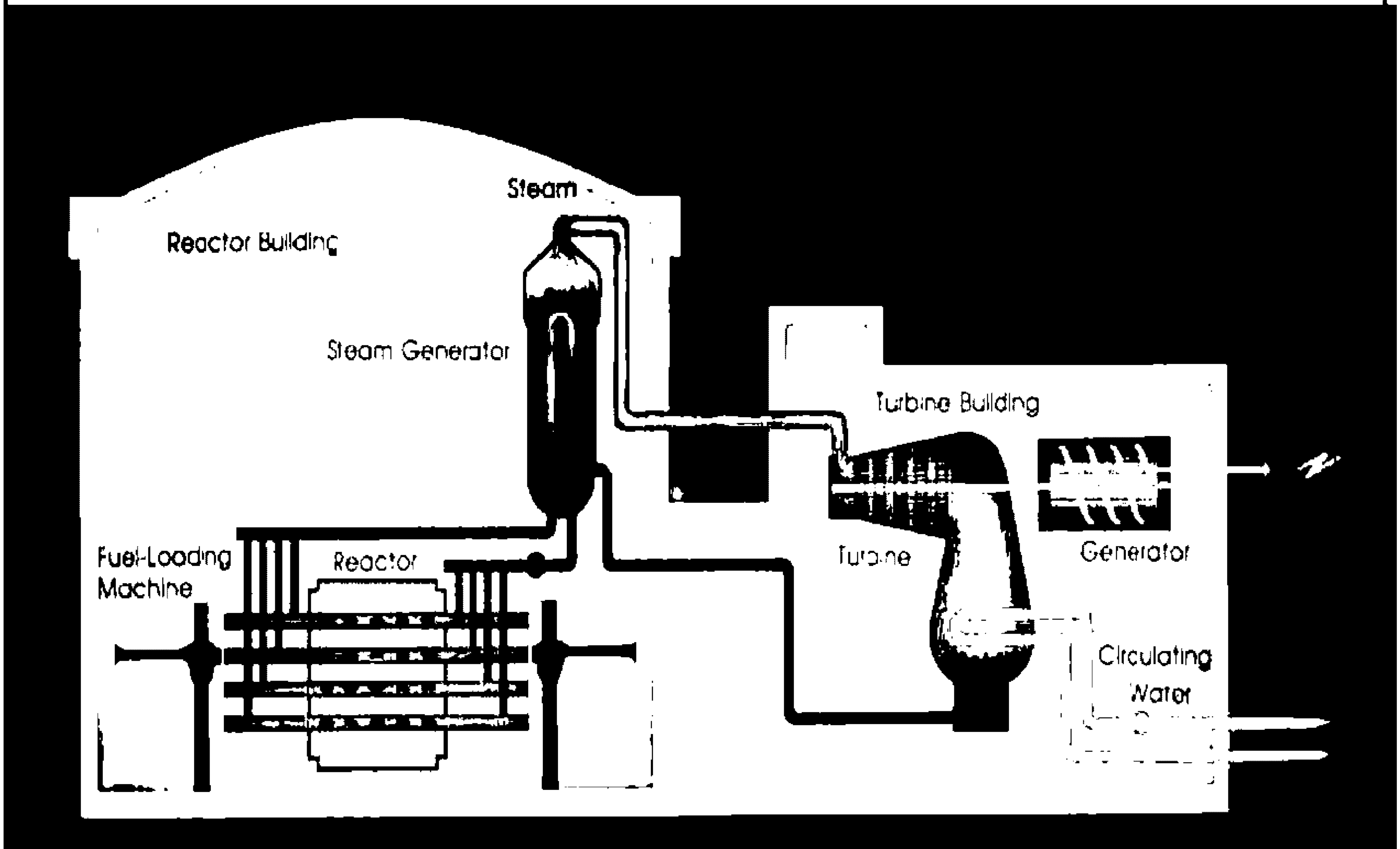




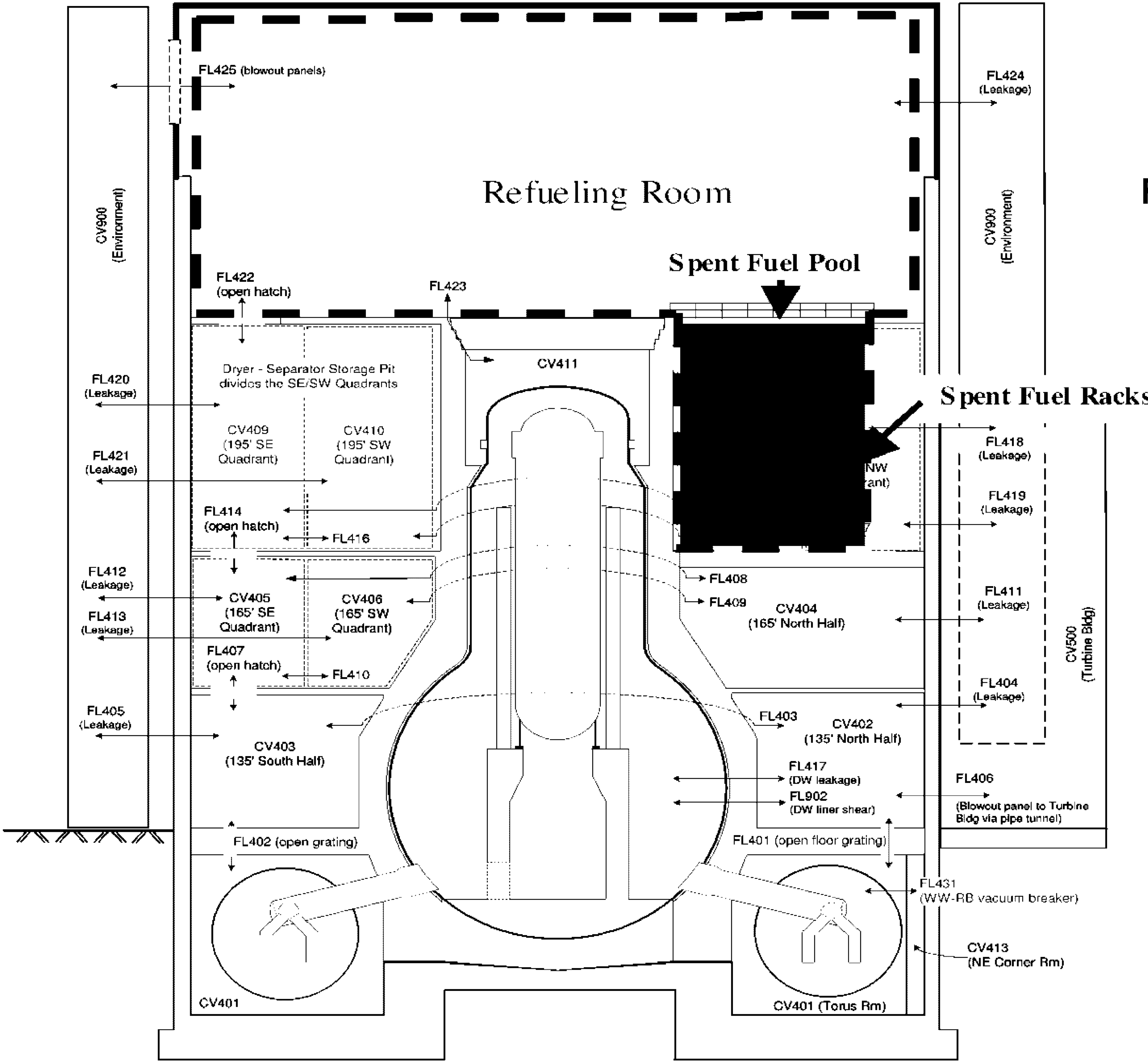
Innovative feature:

vacuum between Rx and containment

CANDU Reactor



MK I
Fukushima-like



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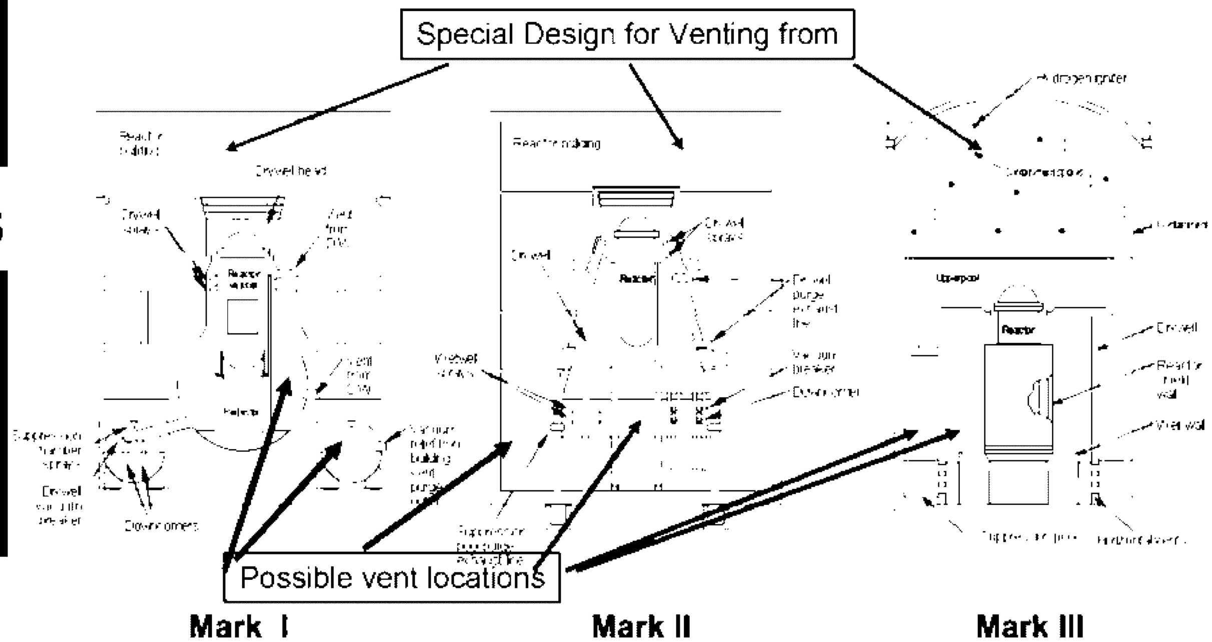
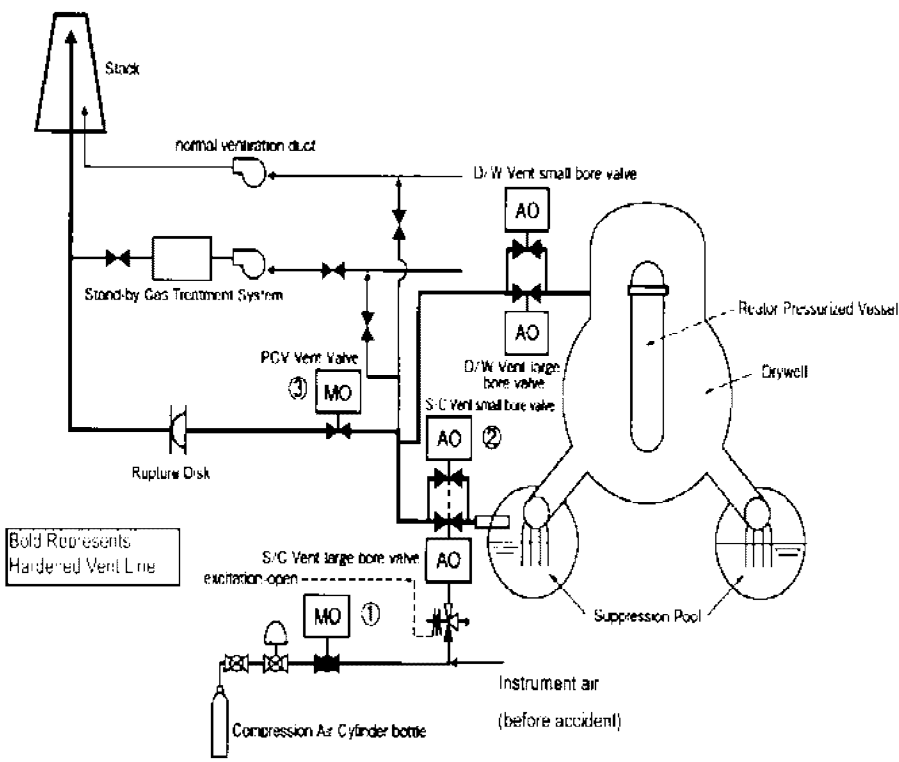
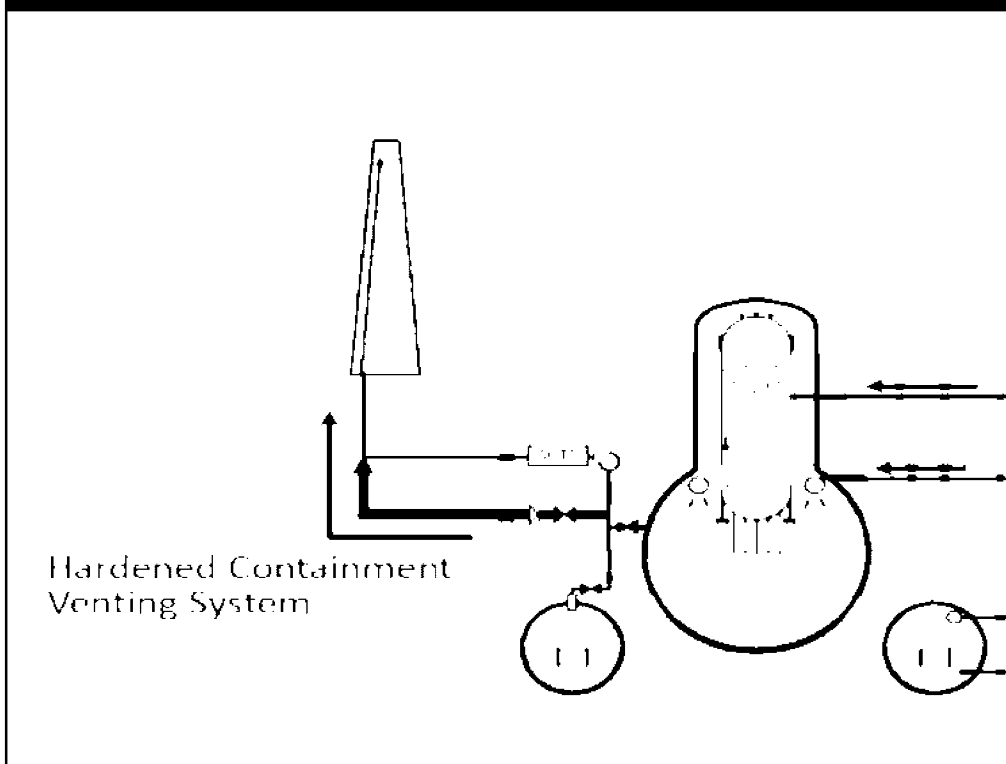
Various vents concepts

[redacted]



PCV Hard Vent





Containment behavior:

Containment integrity challenged by excessive P / T

Mass and Energy released during an accident THE leading factor

Flow distribution affected by multi-compartmentalization

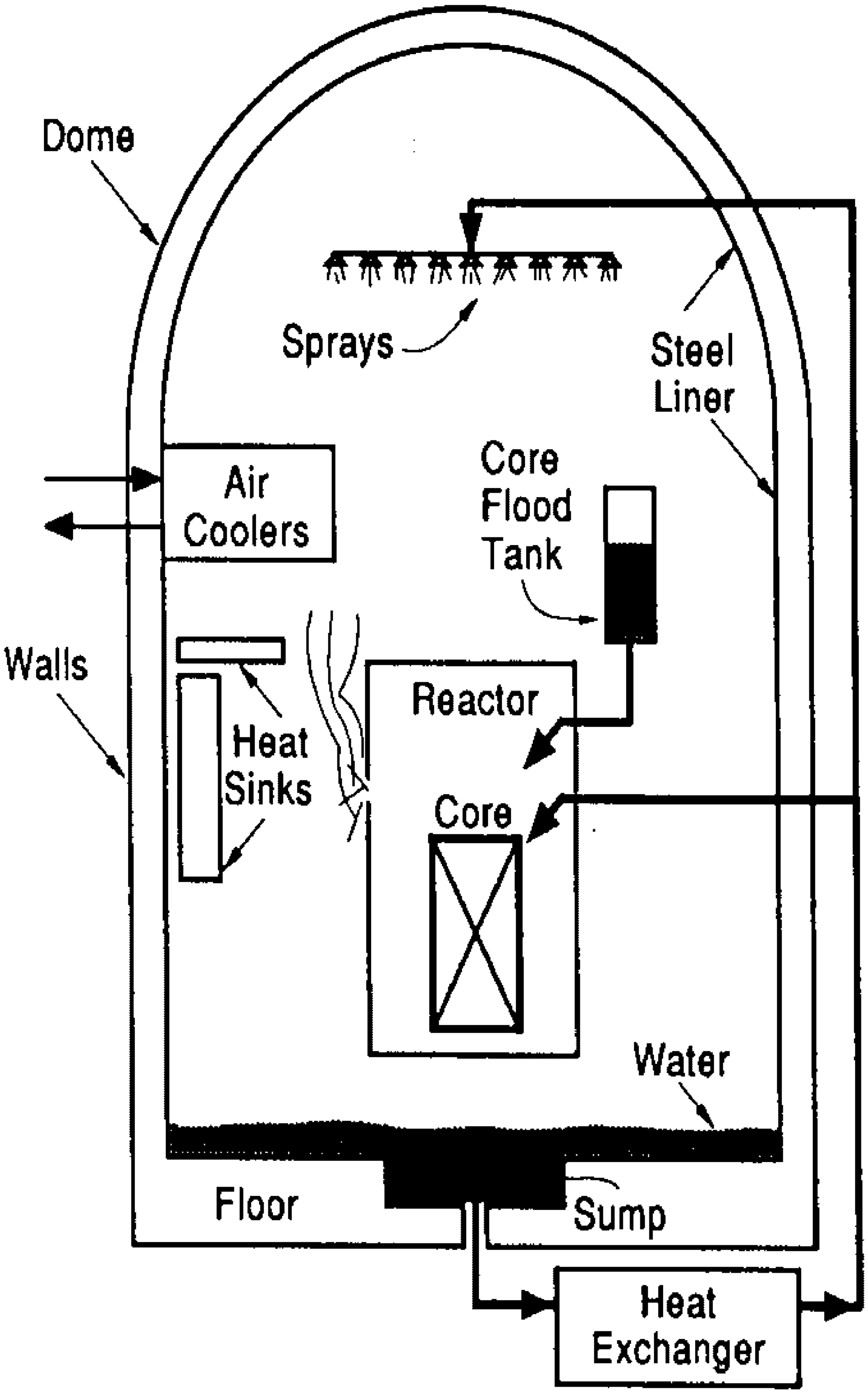
Long(-er) term effects: ESF heat removal and heat structures

Steam condensation major means for decrease of P / T:
- Suppression pools [hydrodynamic loads] and sprays

Passive feature (b)(4) (need active "help" after "peak" pressure at 72 hours)

Containment thermodynamics (and –hydraulics) involve complex phenomena; requires experimental data to support (semi-)empirical models

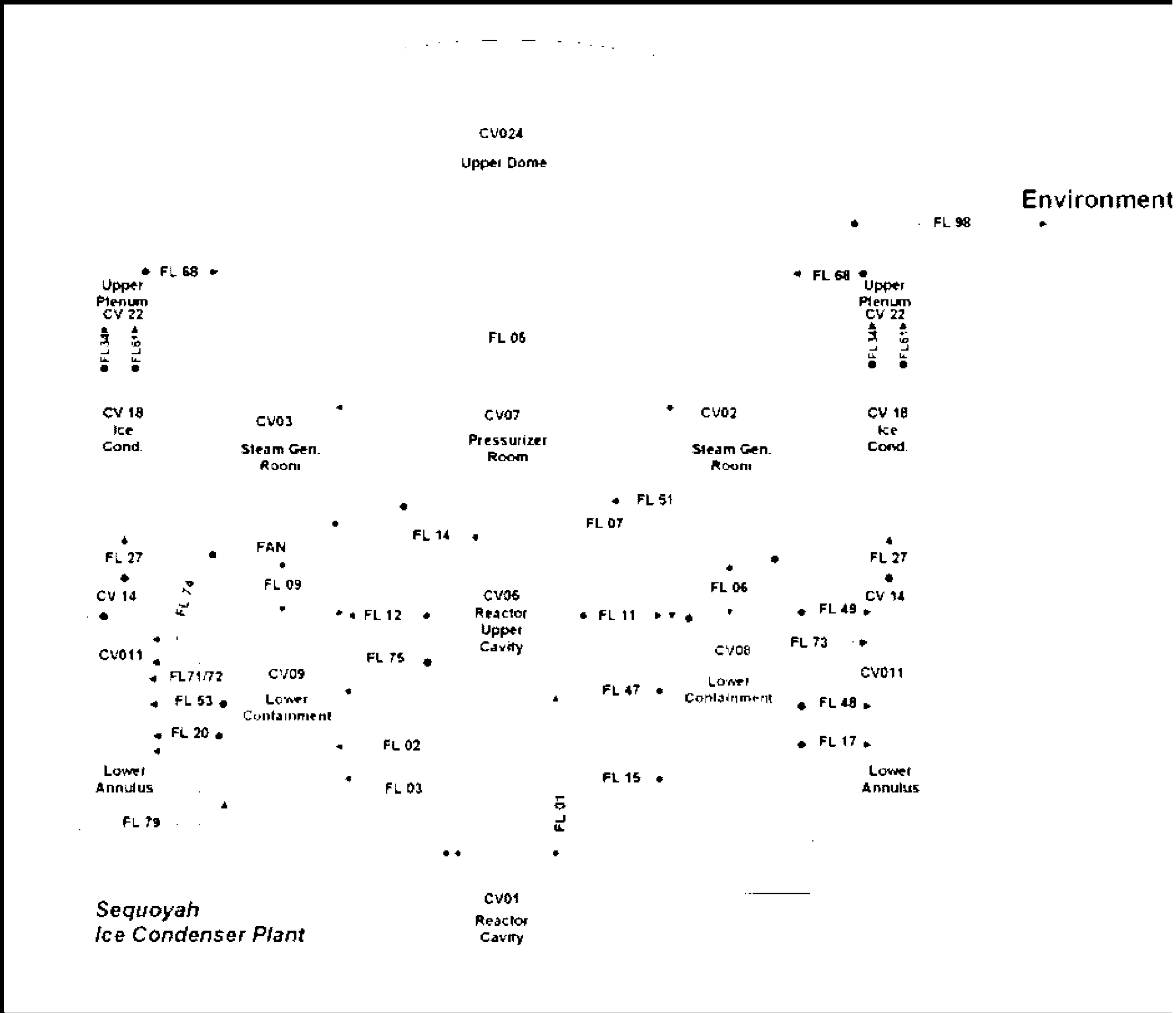
Traditional conservative approach: 1-node, Tagami / Uchida correlations
- acceptable for active system, may not for passive and/or other advanced designs



Dome
Sprays
Steel
Liner
Air
Coolers
Core
Flood
Tank
Walls
Heat
Sinks
Reactor
Core
Water
Floor
Sump
Heat
Exchanger

(b)(4)

Complex Dynamic System Models



- Using control volume elements, dynamic response models of complex systems can be developed
  - Containment
  - Reactor vessel
  - Core internals
  - Steam generators…
- Specialized physics modules account for special features
  - Core heatup
  - Zr oxidation and H_2
  - Fission product release
  - H_2 Burn models
  - Many more

BNWL-1592
UC-41

Battelle

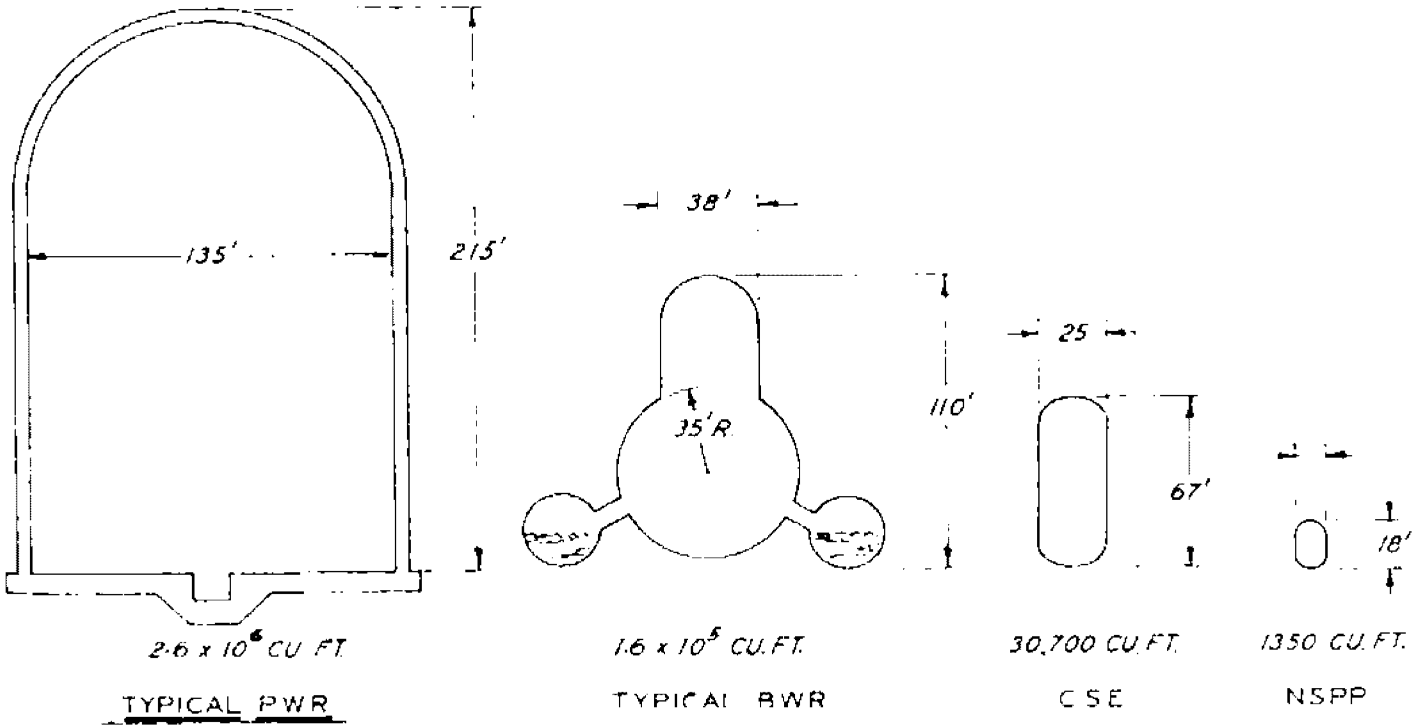
Pacific Northwest Laboratories
Richland, Washington 99352

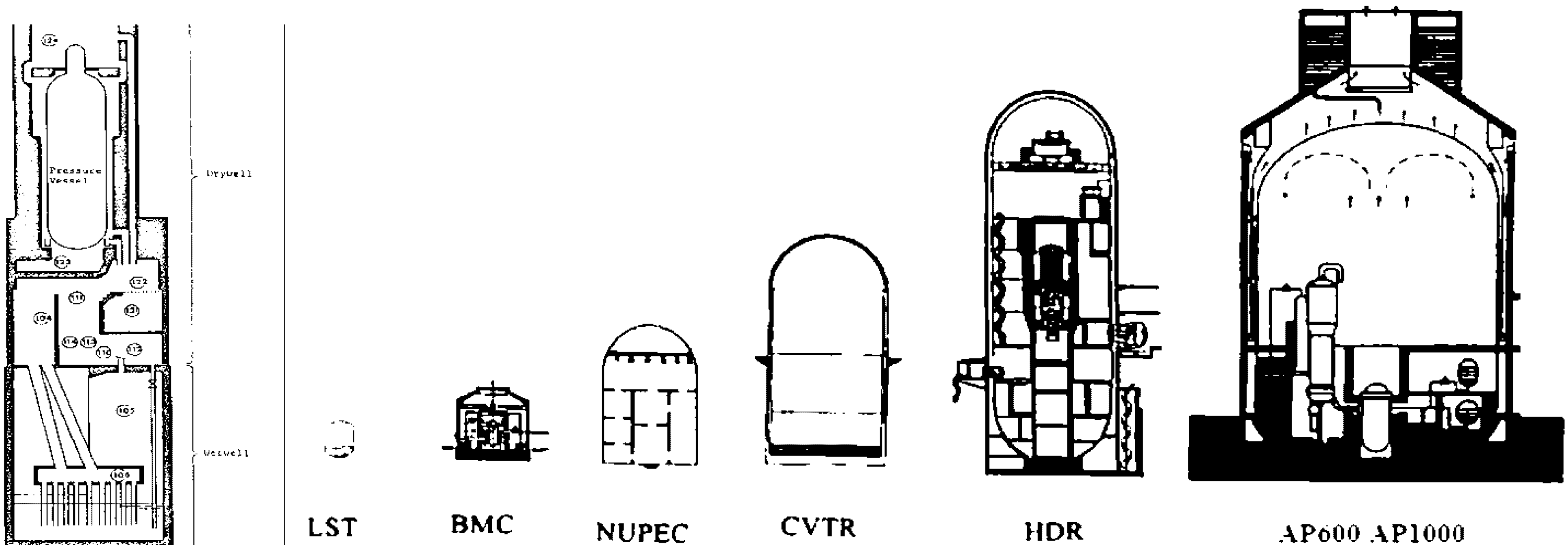
AEC Research and Development Report

CONTAINMENT SYSTEMS EXPERIMENT FINAL PROGRAM SUMMARY

July 1971

Scale of containment test facilities





Marviken (not to scale)

ERCOSAM-SAMARA: Scaled facilities used

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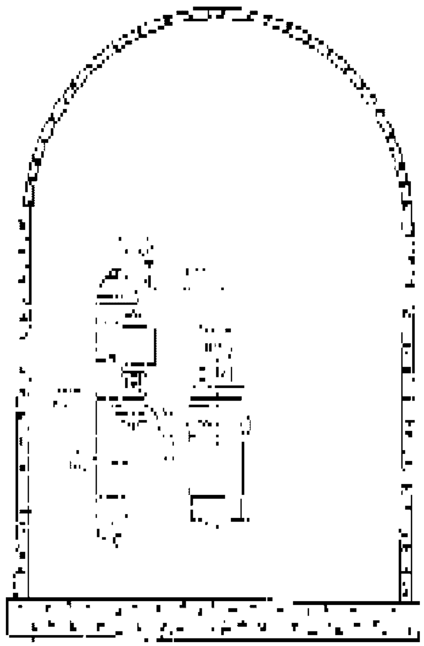
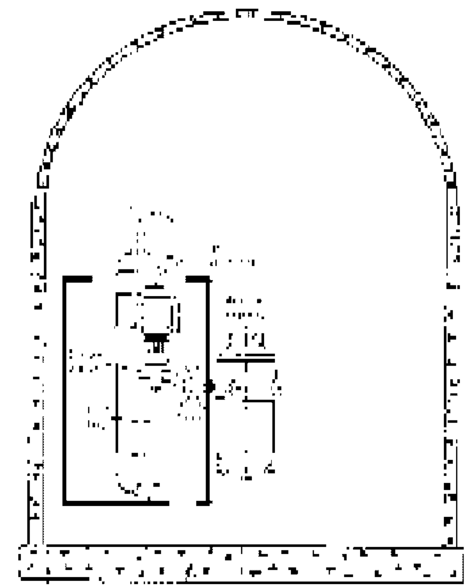
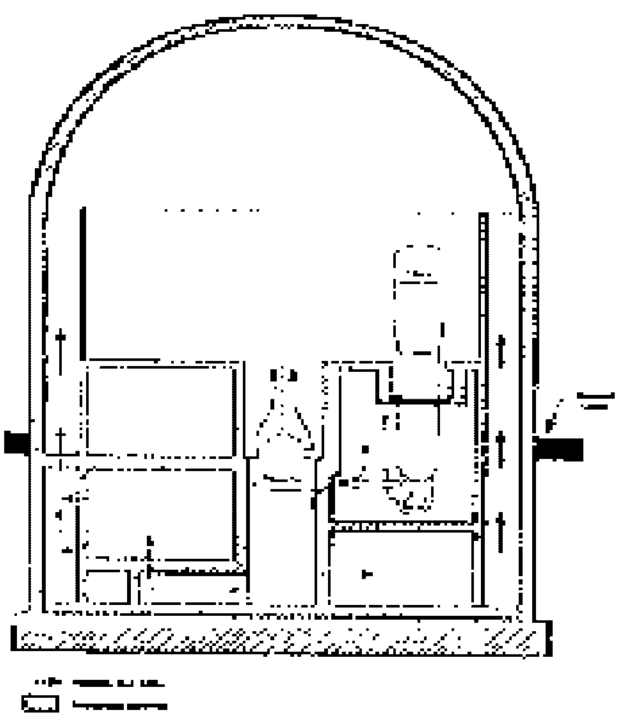


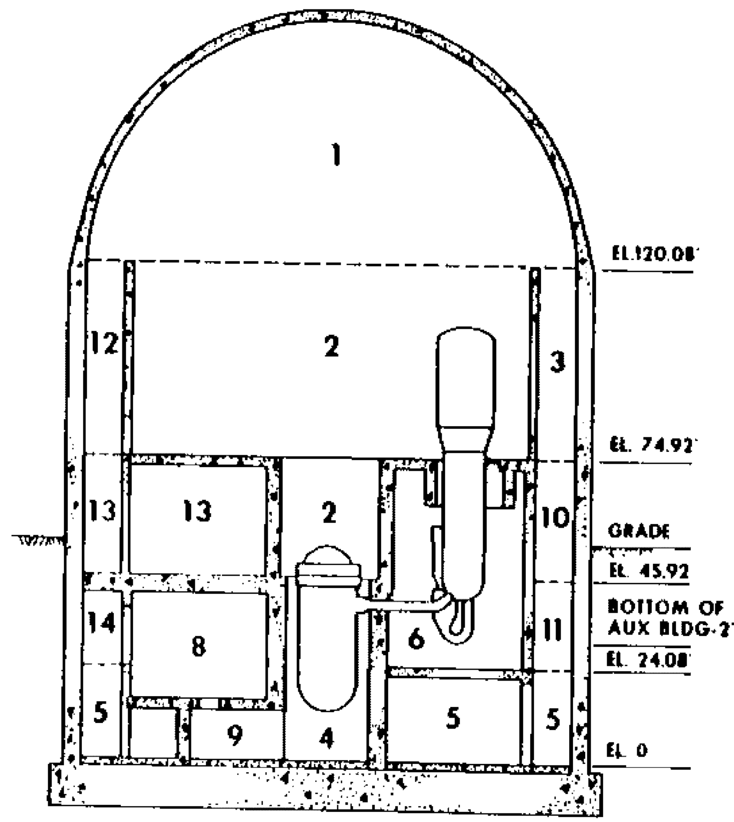
Single node

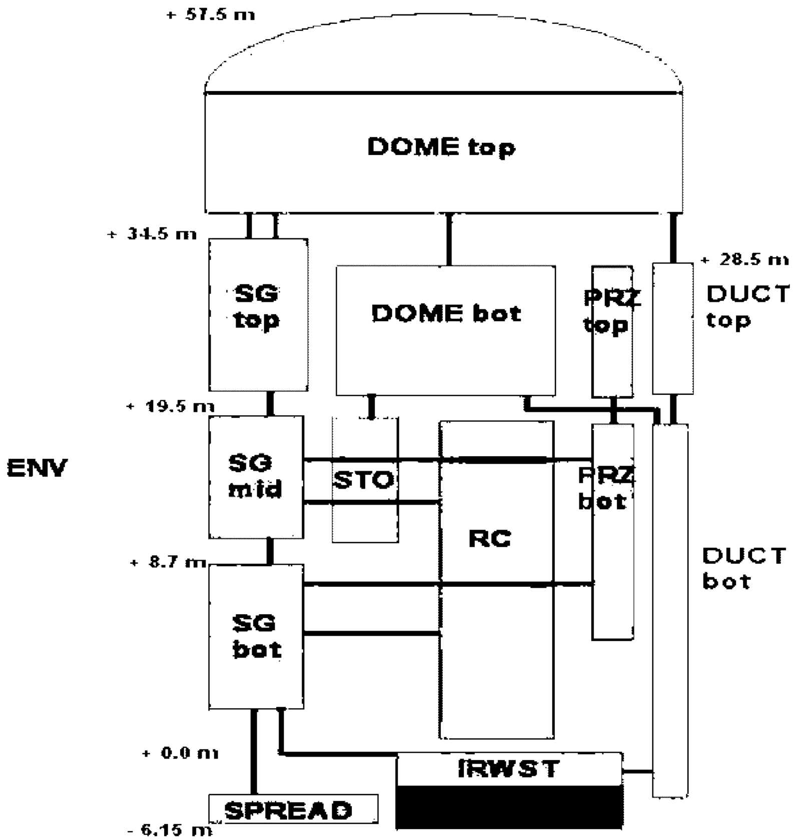


“two”-node

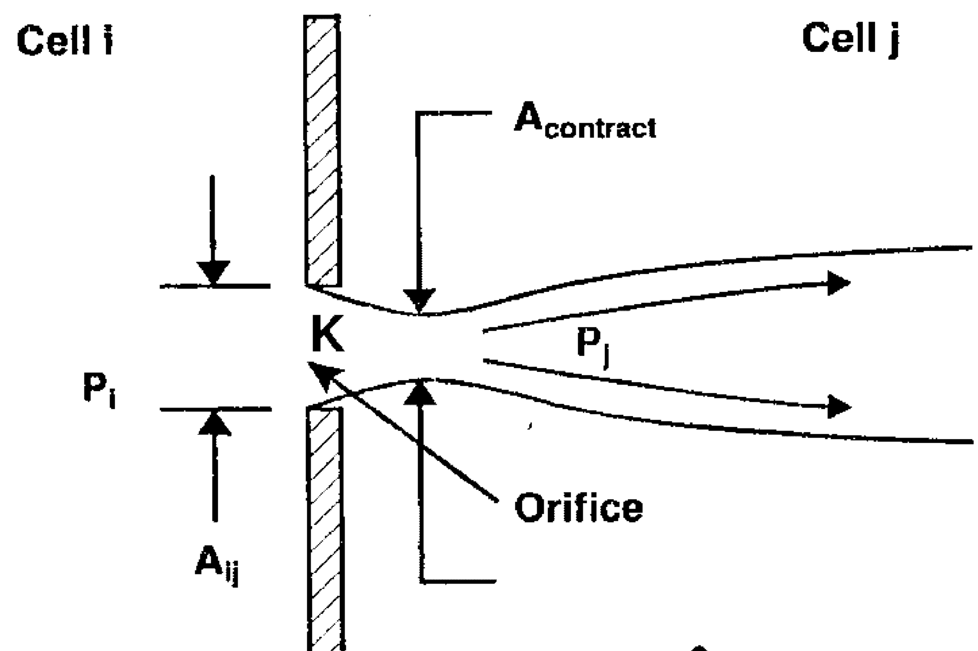


Multi-compartment

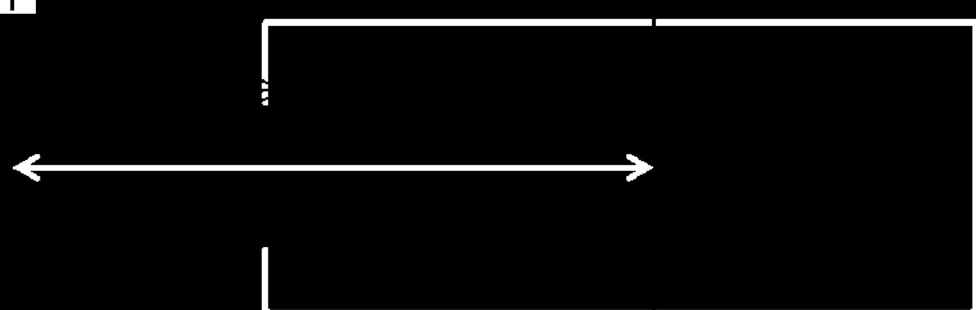




Intercompartment flows



mass and energy



momentum

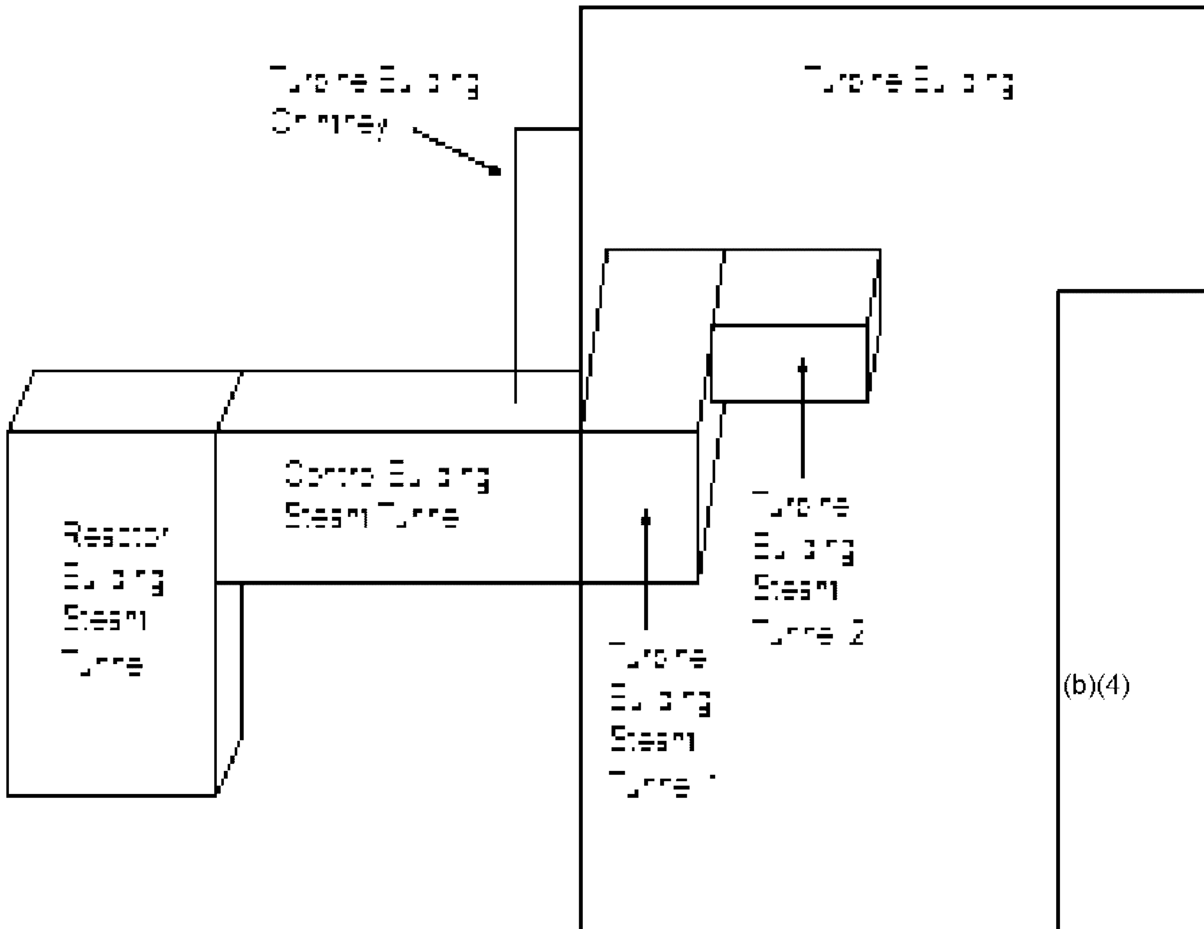
Parameters to review:

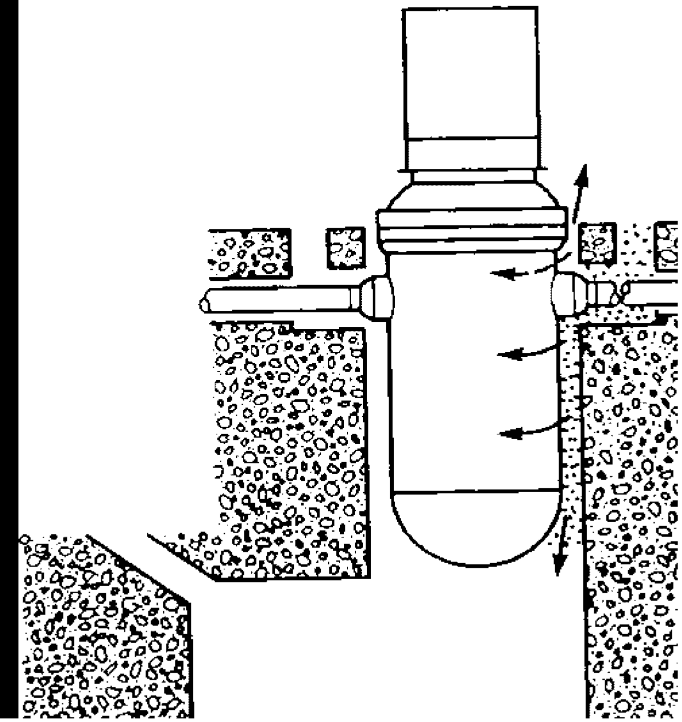
K for orifice

L inertia length

Typical sub-compartment analysis

- **figure of merit: pressure difference across a structure**





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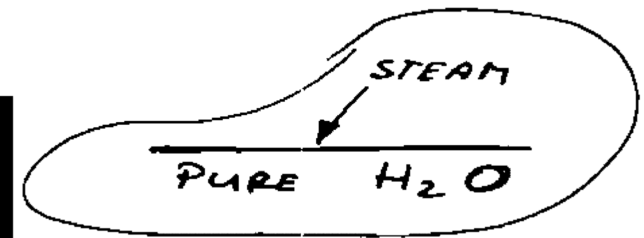
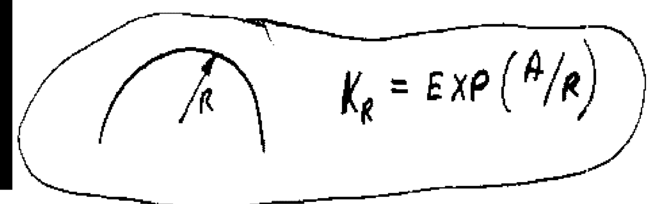
Condensation basics

$$\dot{M} \sim (P_{PS} - P_S(T)) \Rightarrow P_{S,PURE,FLAT}\,[S - K_R * K_{SLN}]$$



$$P_S(T) = P_{S,PURE,FLAT}(T) \bullet K_R(R) \bullet K_{SLN}$$

$P_{S,PURE,FLAT}$: PURE WATER, FLAT SURFACE

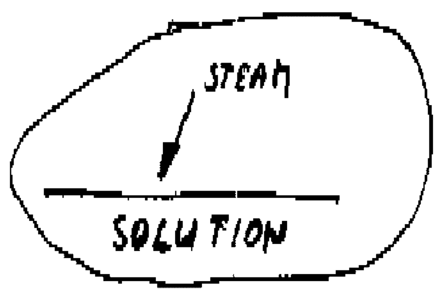


$K_R(R)$: KELVIN CORRECTION FOR CURVATURE

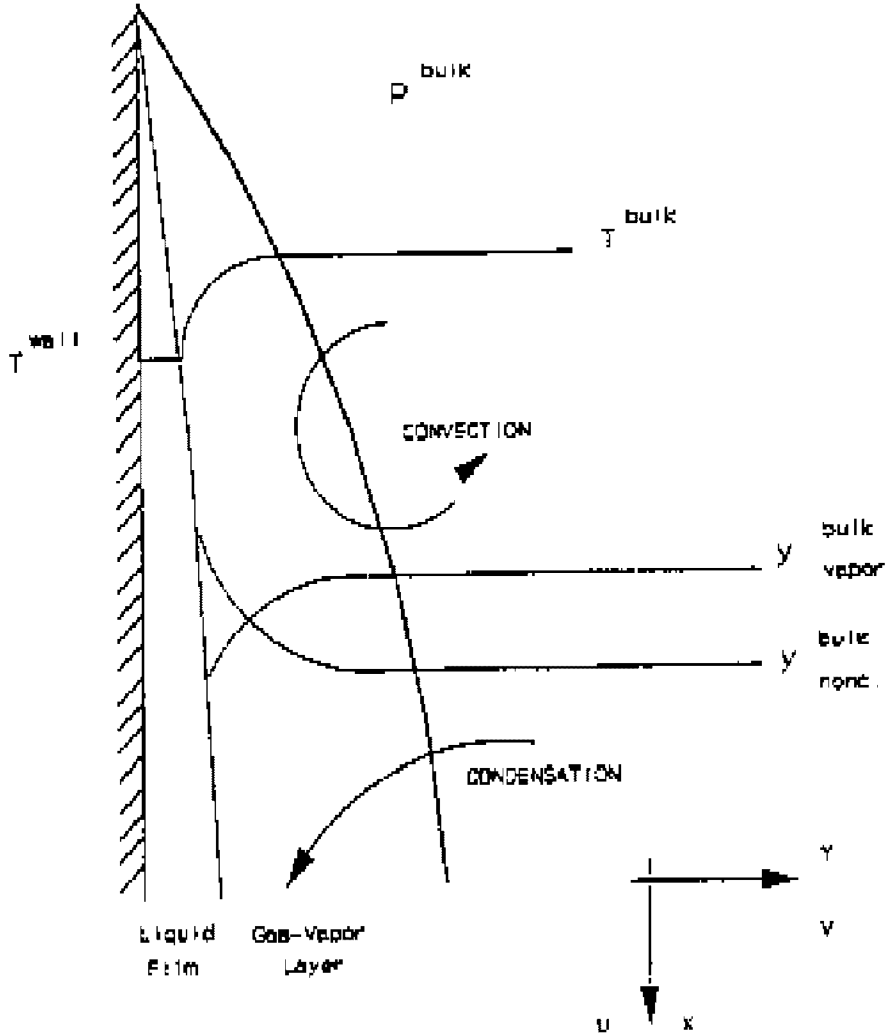
$\dot{M} = 0$ FOR $R \leq R_{CRIT}$ FROM $S - K_R * K_{SLN} = 0$

$K_{SLN} = K_{HYGRO}$: HYGROSCOPIC CORRECTION FOR SOLUTION

(CsI, CsOH)



Solution reduces water vapor pressure, thus effectively reducing value of critical radius



MECHANISMS OF CONDENSING HEAT TRANSFER

NOTE: in reality there is some condensation in the atmosphere

Realistic approach: multi-node, BE condensation HTC

Historical conservative approach: single node, Tagami/Uchida, X% revaporization, minimum heat conductors

T / U condensation models based on Sagawa data (1960's)

- **steady-state (140x300 mm, vertical plate)**
- **LOCA (300, 600, 900 mm cylinder),**
- **problem with interpretation**
- **became known as Uchida and Tagami (Slaughterback paper, 1970's)**

Tagami – "inconvenient", requires iteration (time of Pmax)

Uchida – depends ONLY on air/steam ratio, produces highest peak P, BUT does not apply to superheated conditions

(b)(4)

Total Heat Transfer Coefficient (W/m^2K)

(b)(4)

SPRAYS

Spray [and suppression pool] most effective ESFs

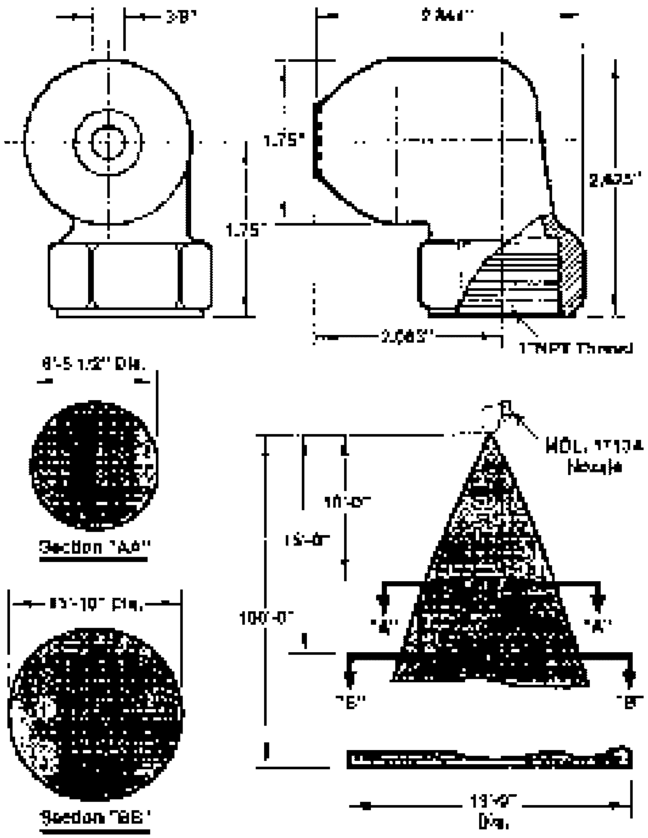
Results in a rapid pressure decrease

Volume coverage depends on headers / nozzles arrangements

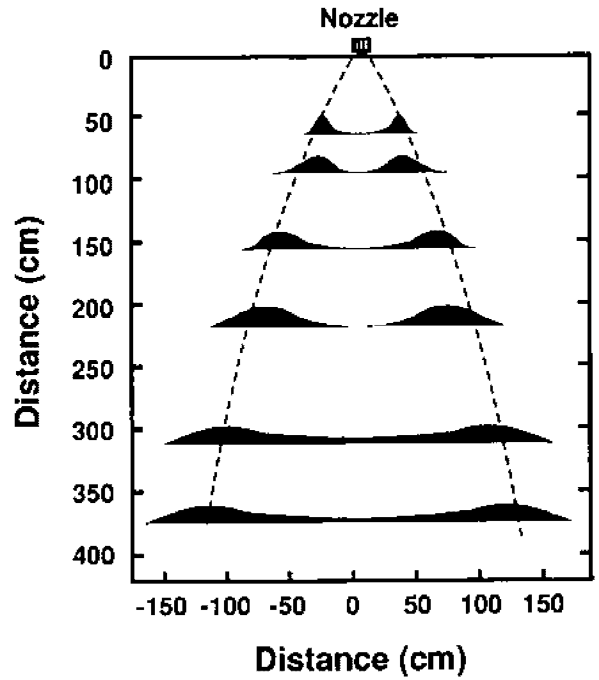
Spray drop size distribution depends on nozzle type [needs test data]

Residence time must be greater than “relaxation time” [i.e. time to reach thermal equilibrium]

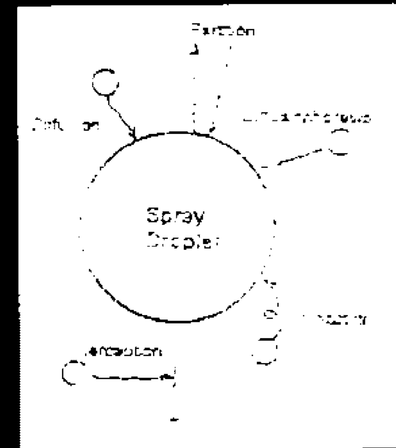
Large spray droplet size [e.g. 1000 microns] is a conservative assumption



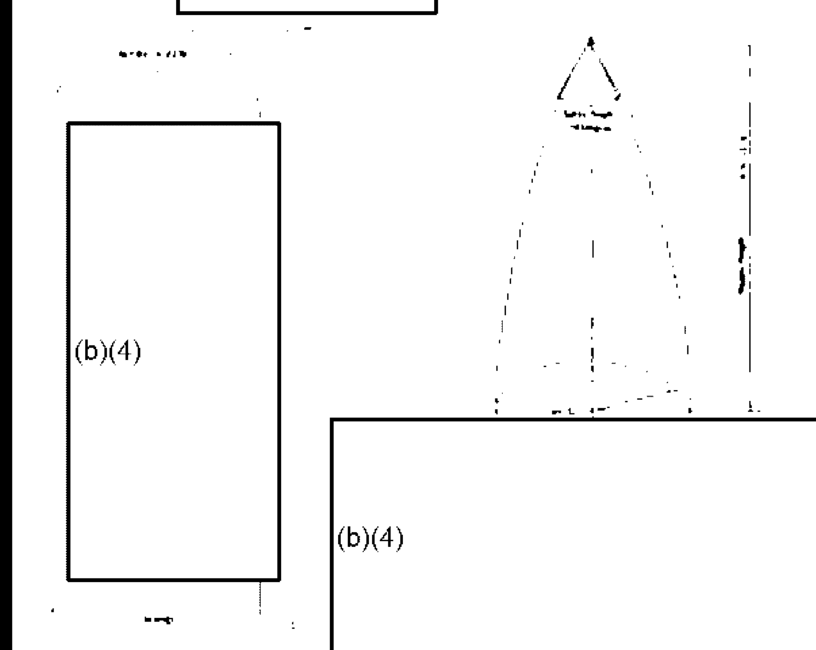
(b)(4) spray nozzle



Spatial variation in water flow



(b)(4) nozzle spray test



Elements of review:
- spray coverage
- vendor size distribution
- initial water temperature
- selected drop size [SRP: 1000 microns]
- height of spray header(s)

Exercise: condensation rate on water droplet

(b)(4)

answer e.g. "Elements of Cloud Physics"
H. R. Byers, p. 122
Univ. of Chicago Press

Regulatory flow rate based spray condensation model: e.g. NUREG-0772

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(b)(4)

of the Freedom of Information and Privacy Act

Importance of parametric studies

GOTHIC 7.2b

Containment Dome Pressure

80

Pressure (psia)

(b)(4)

[redacted]

GOTHIC 7.2b

Containment Dome Pressure

(b)(4)

[redacted]

BWR:

DW similar to PWR

WW very different. Important phenomena:
- Vent clearing
- Pool swell
- Condensation oscillation, chugging
- Associated hydrodynamic loads

- NO established analytical model

- Approval based on various experiments

Hydrodynamic loads

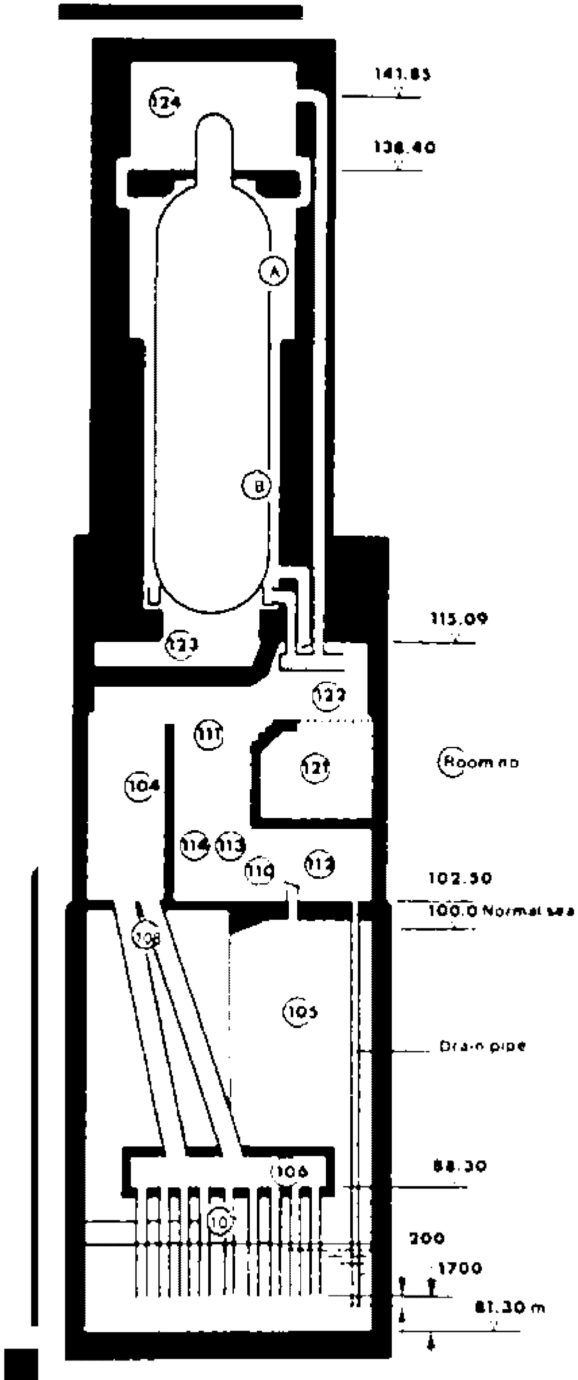
Notes on BWR-related thermal-hydraulic DBA analysis

GE methodology – GESSAR (NUREG-0979):
- basic BWR (MKI, II and III) test data (like PSTF)
- MKIII specific test data (HVT)
- basic models: NUREG-0808 (MKII), and NEDO-20533 (MKIII)
- specific application requires combination of test data and scaling analysis
  - -- direct application of MKII/III models inadequate
  - -- PSAM (NEDO-21061) for H/D loads (approved based on GE/JAERI test data)
  - -- PICSM code with additional correlation for uneven pool slug rise
  - -- subscale (2.5) and partial full scale (full scale vents with 2 horizontal) tests

- Containment P/T:
  - -- GESSAR methodology (M3CPT code for MKIII)
  - -- being replaced by TRACG (ESBWR) with vent clearing correction
  - -- for sub-compartment: SCAM code

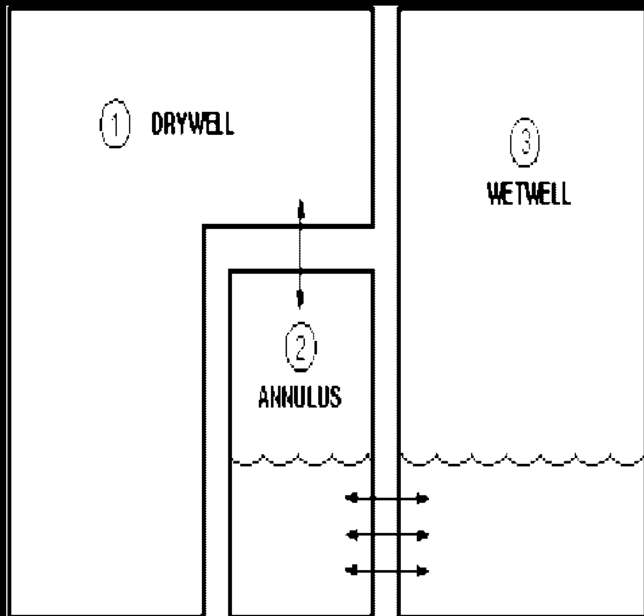
2. NRC independent evaluation based on
- ABWR approved using CONTEMPT-LT28
- currently use of MELCOR, based on CONTAIN models
- CONTAIN qualification report on BWR DBA analysis
- COMPARE / CONTAIN for sub-compartment => MELCOR

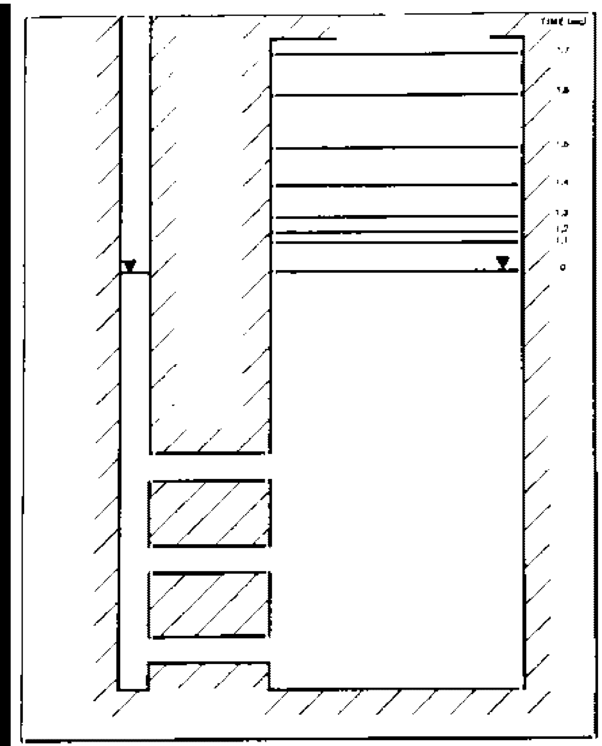
3. Industry approach (STP)
   - use of GOTHIC, benchmarked on other model and/or available test data

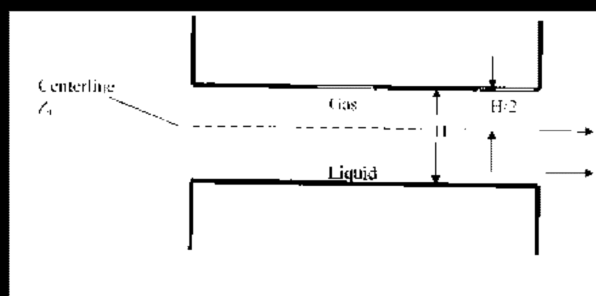

141.85
138.40
124
A
B
115.09
123
122
111
121
Room no
104
114
113
110
112
102.50
100.0 Normal sea
108
105
Drain pipe
106
88.30
107
200
1700
81.30 m

(b)(4)

[redacted]







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[redacted]

(b)(4)

(b)(4) [redacted]

Schematic of the Pool Swell Phenomenon

(b)(4) [redacted]

Important parameters for review:

Timing of vent clearing

Pool swell height

Pool surface velocity

Condensation oscillation loads

Chugging loads

SRV / quencher loads

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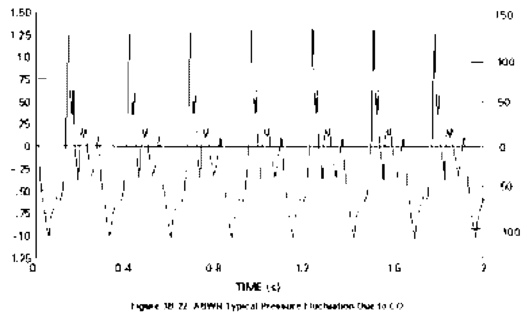
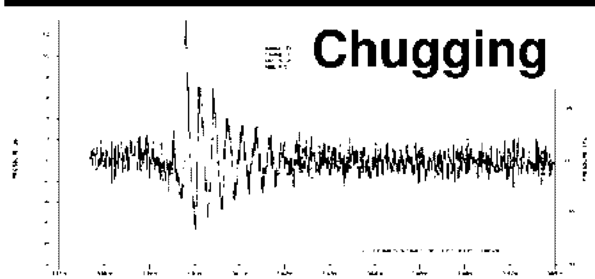
of the Freedom of Information and Privacy Act

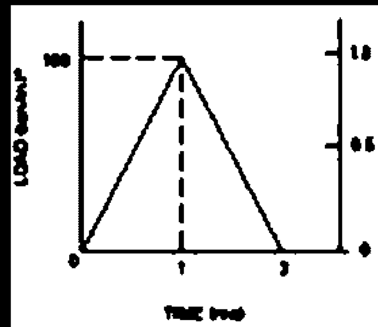


CO

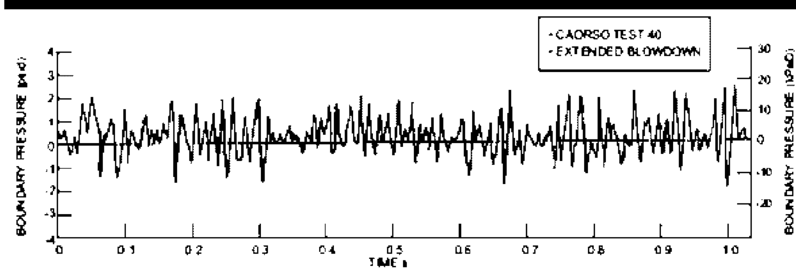


Chugging

Figure 3B-24 Typical Large Chug (021P)



assumed load pulse



Measured pressure at the boundary

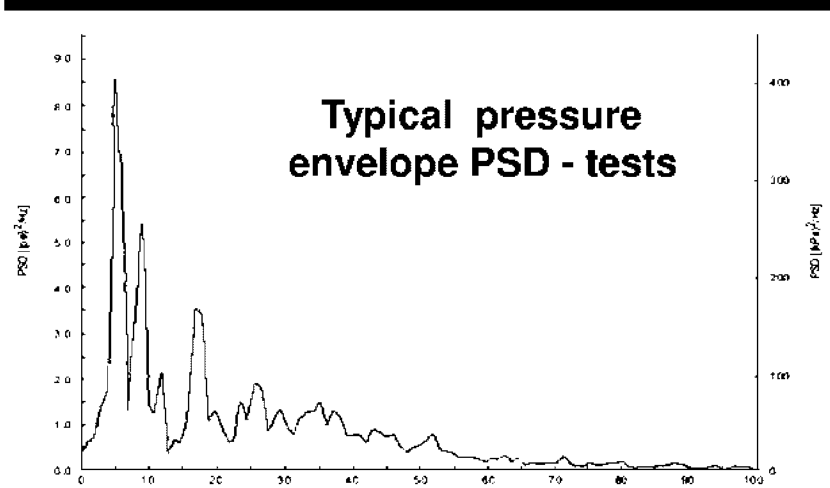
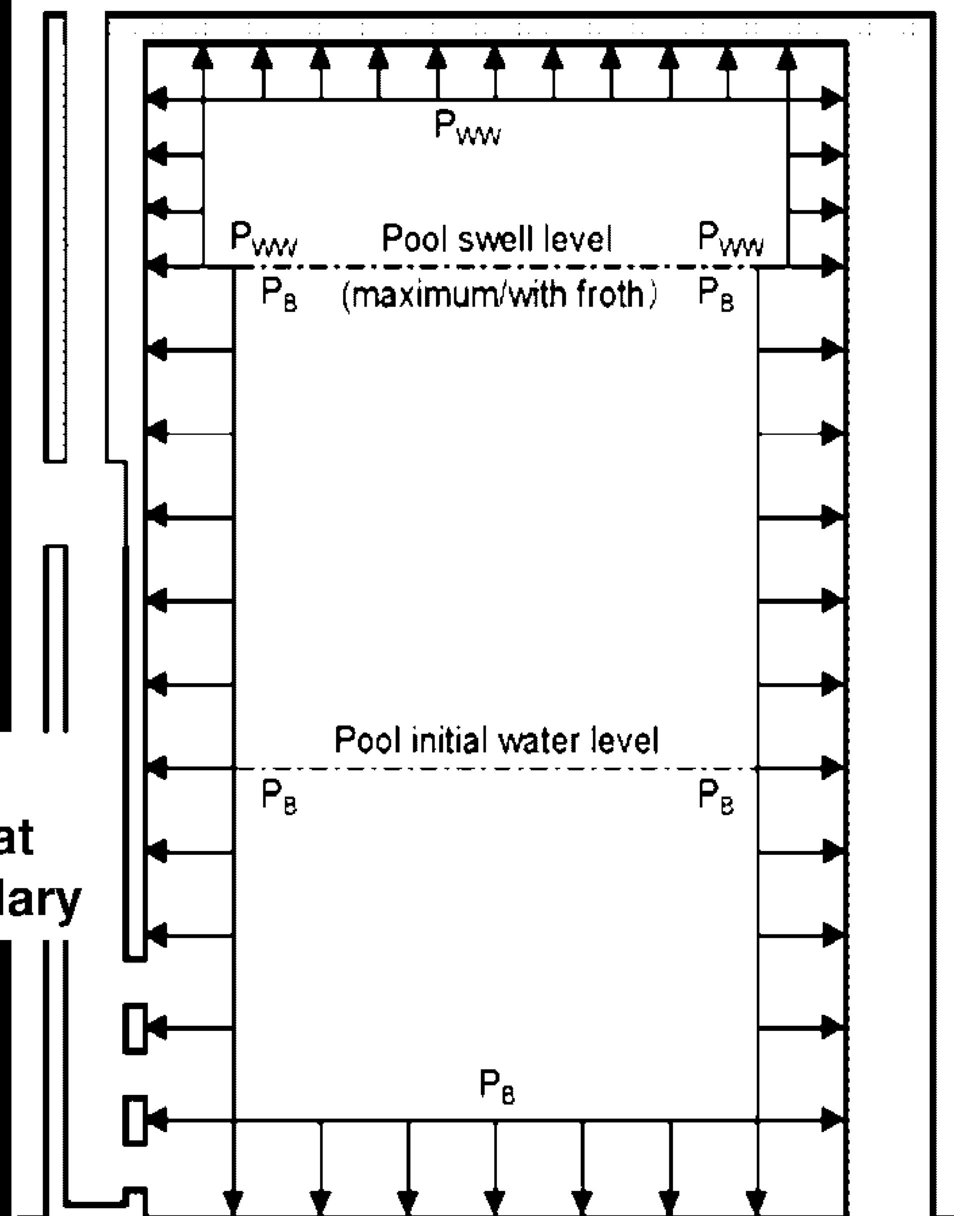


Figure 3B-19 Envelope PSD at 019P for SST-1, 2, 3, 9, 11, and 12



Typical pool boundary load distribution

(b)(4)

Review of testing data base

AND

it’s applicability to a given design

is CRUCIAL

for licensing approval

of hydrodynamic loads

(b)(4)

Beyond DBA Challenges

Beyond DBA Challenges

Phases of Beyond Design Basis Accident

-More that one failure: single failure criteria

-Loss of coolant => core uncovery

-Core heat-up => rapid clad oxidation

-Loss of core coolability => core relocation

-Challenge to plant integrity: RX, containment, and beyond…

Phenomena relevant to containment:

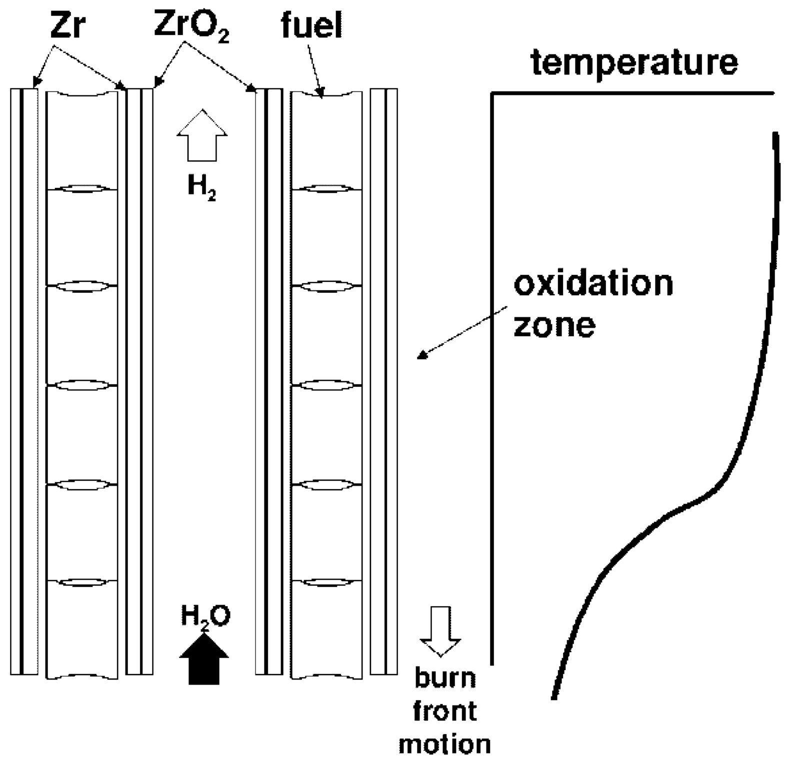
- until core heat-up - similar to DBA

- hydrogen generation leads to deflagration/detonation

- core relocation leads to
- -- steam explosion (in- or ex-vessel)
- -- direct containment heating: dispersion of molten corium
- -- molten core interaction with concrete basemat
- -- vessel missile

- NOTE: for SA phenomena see Dr. Fuller seminar/workshop

Steam Oxidation of Cladding in Fuel Assemblies



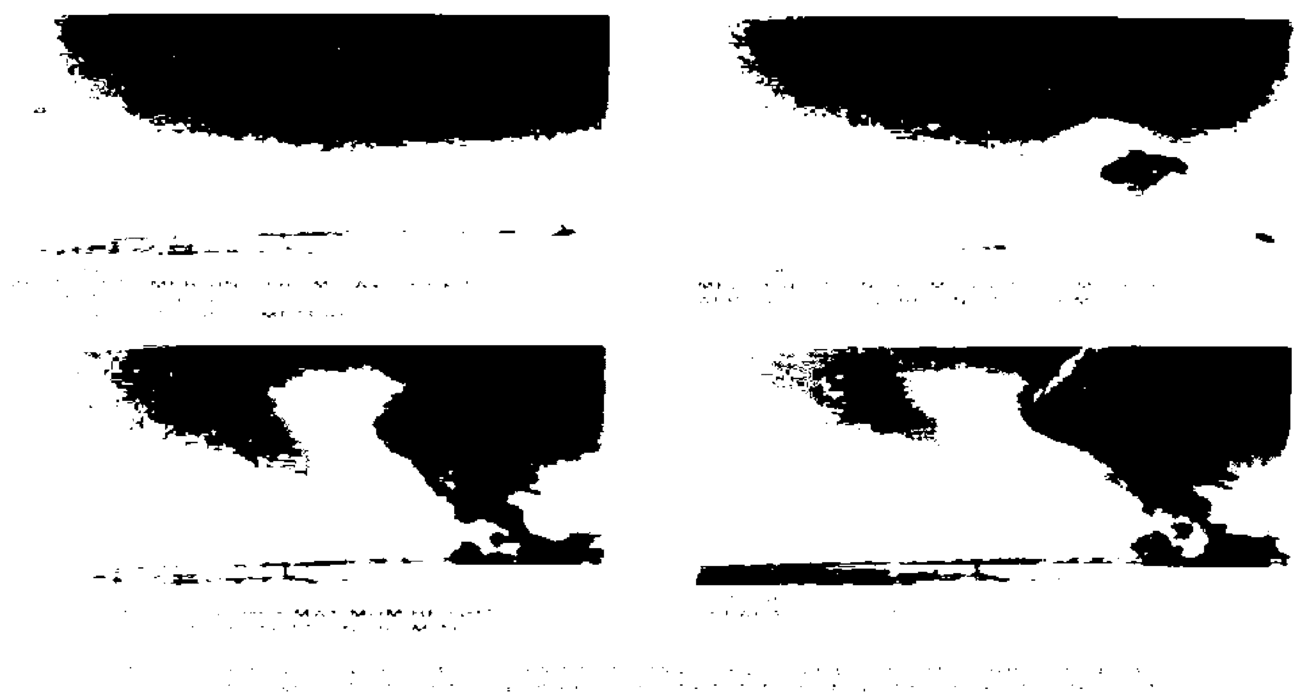
$$Zr + 2H_2O \longrightarrow ZrO_2 + 2H_2 \quad 5.8\text{x}10^6 \text{ joule/kg}$$

$$\frac{dT}{dt} = \frac{1}{mC_p}\left[Q_{ox}(T) - Q_{loss}(T)\right] \quad \text{heatup rate}$$

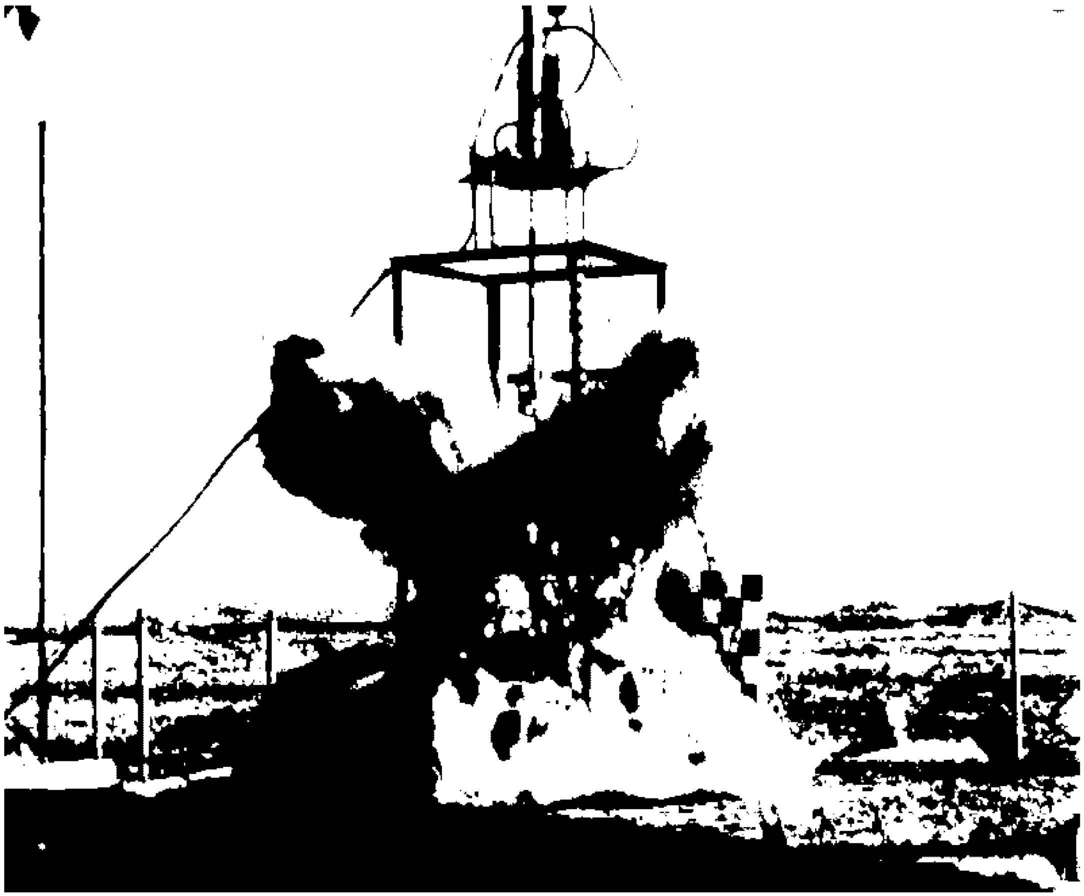
- Steam oxidation actually more studied
- Overall behavior quite similar top air oxidation, except…
- Hydrogen is produced in steam oxidation
- Reaction heat reduced compared to air

Core Debris Penetration of the Reactor Vessel Leads to Ex-vessel Releases of Radionuclides

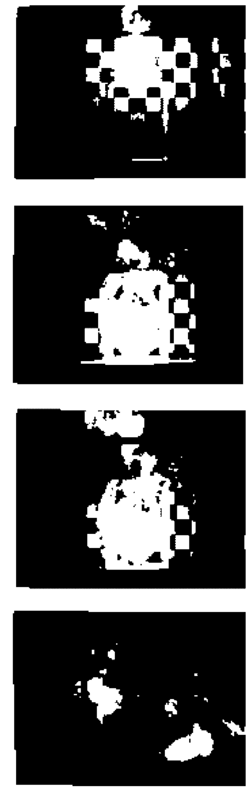
- High Pressure Melt Ejection from Vessel
- Ex-vessel Steam Explosions
- Melt interactions with concrete



Steam Explosion Experiment



THE STEAM EXPLOSION PROCESS



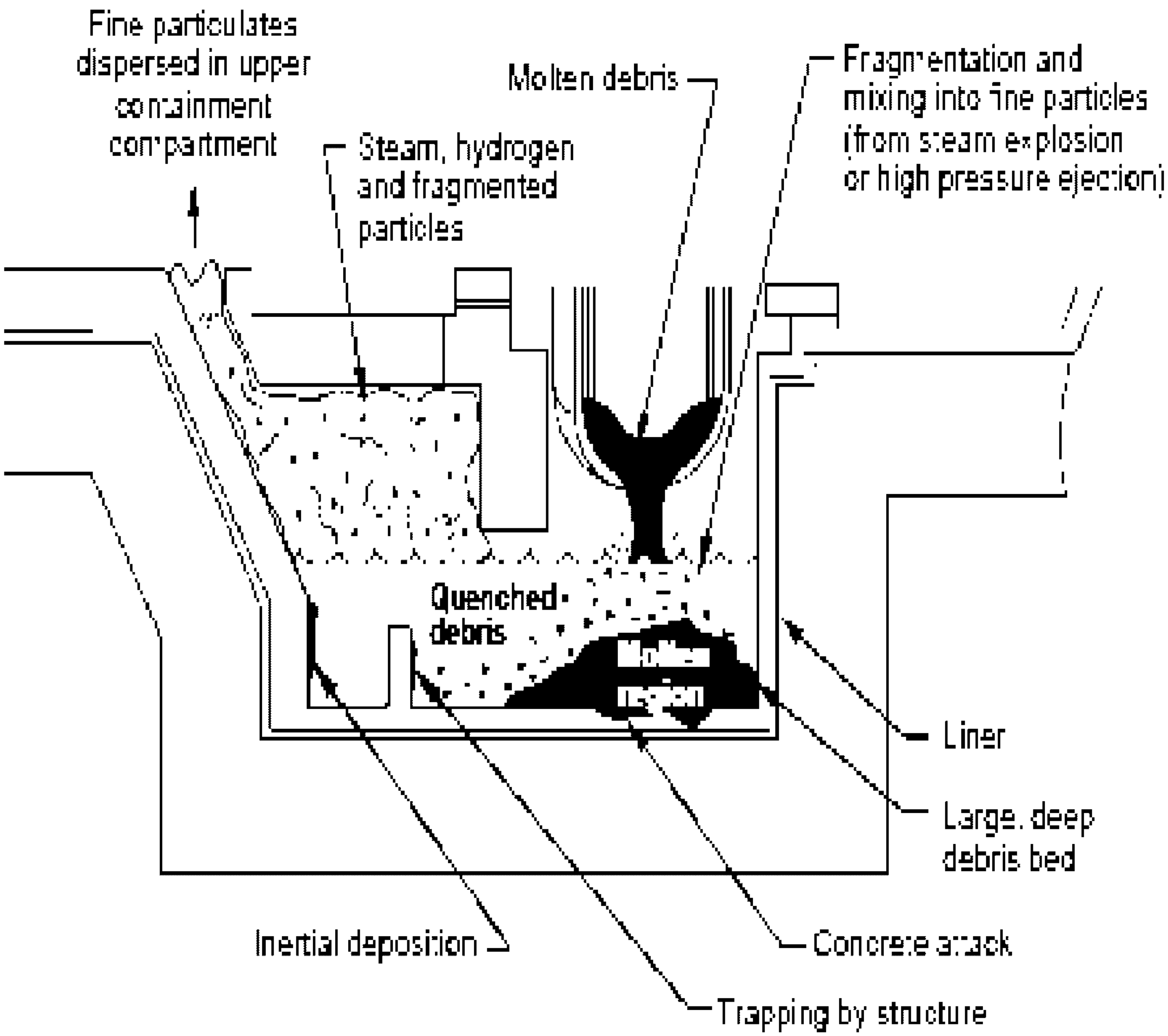
MELT ENTRY INTO WATER
TIME 0

MIXING OF MELT AND WATER
TIME 0.15 (s)

INITIATION OF EXPLOSION
TIME 0.20 (s)

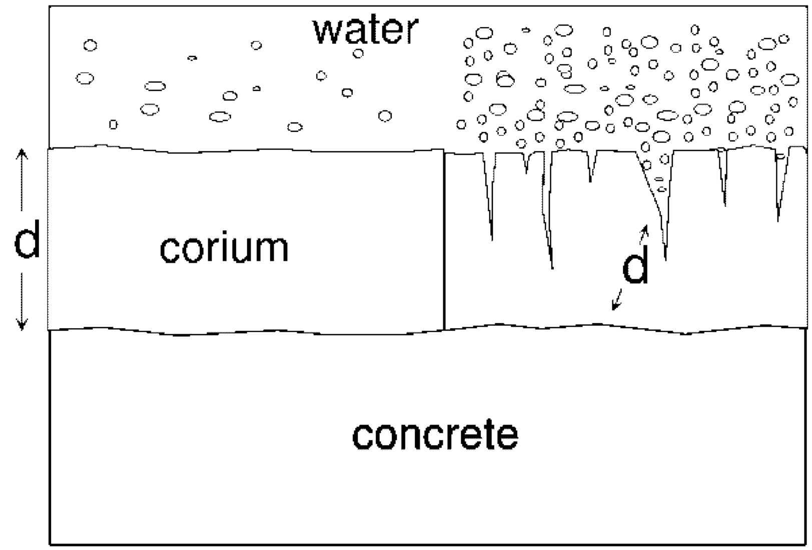
EXPANSION OF PRODUCTS
TIME 0.25 (s)

Direct Containment Heating (DCH) Issues



- Is sufficient melt entrained as vessel depressurizes?
- Does sufficient heat transfer, oxidation, and/or hydrogen combustion occur to threaten containment integrity?

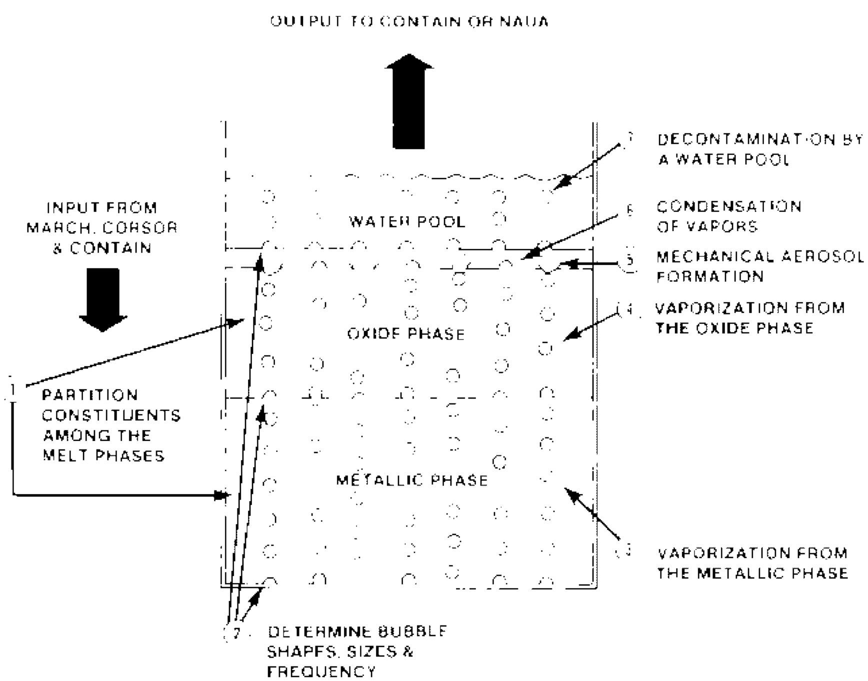
Large-scale High Temperature Melt Interactions with Calcareous Concrete



MCCI modeling

- **Corium assumed to be well mixed (default)**
- **Enhanced effective corium thermal conductivity (10x)**
  - **produces 1 to 5 MW/m^2 heat flux**
  - **Accounts for cracks and fissures**
  - **Consistent with MACE tests**

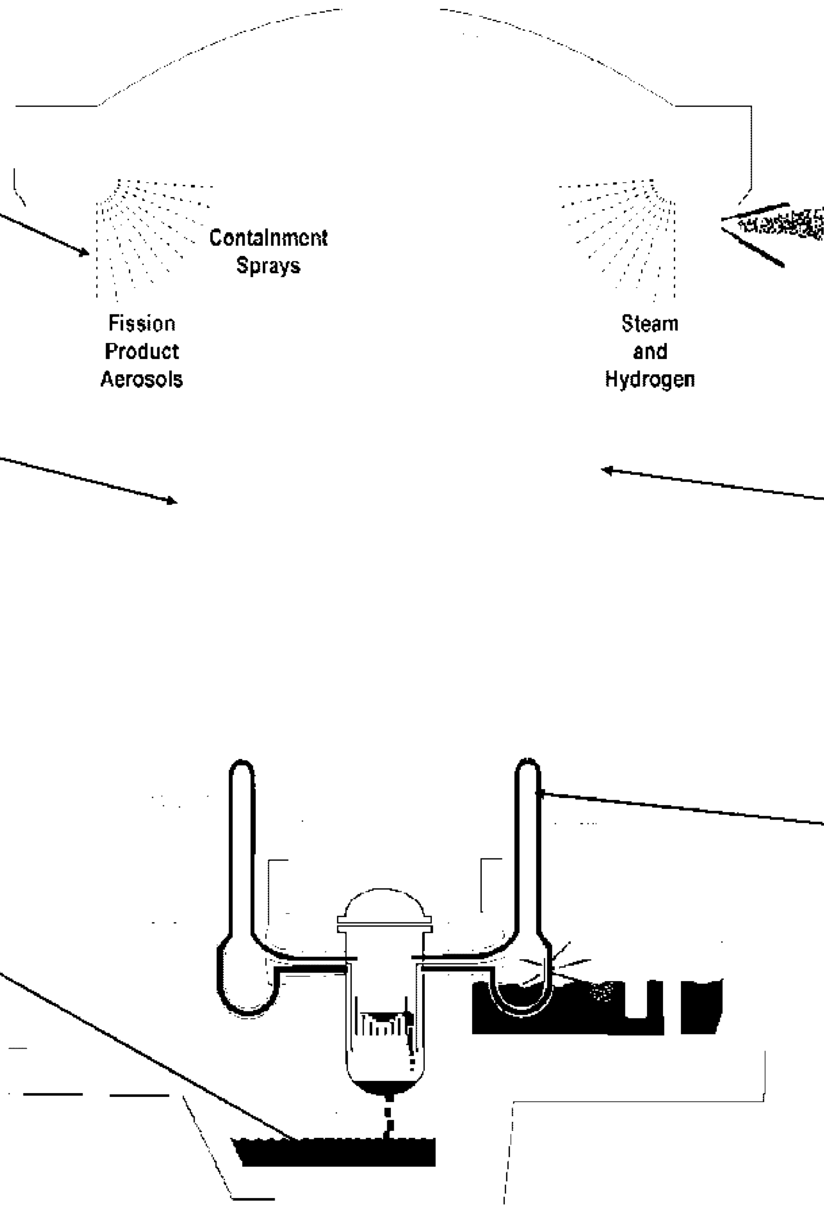
VANESA MCCI model



Containment Phenomena

CSE Spray Tests

Sandia and NUPEC experiments on containment structural response



LACE, DEMONA, VANAM and Phebus Experiments on aerosol depletion

HDR and NUPEC Experiments on H2 distribution

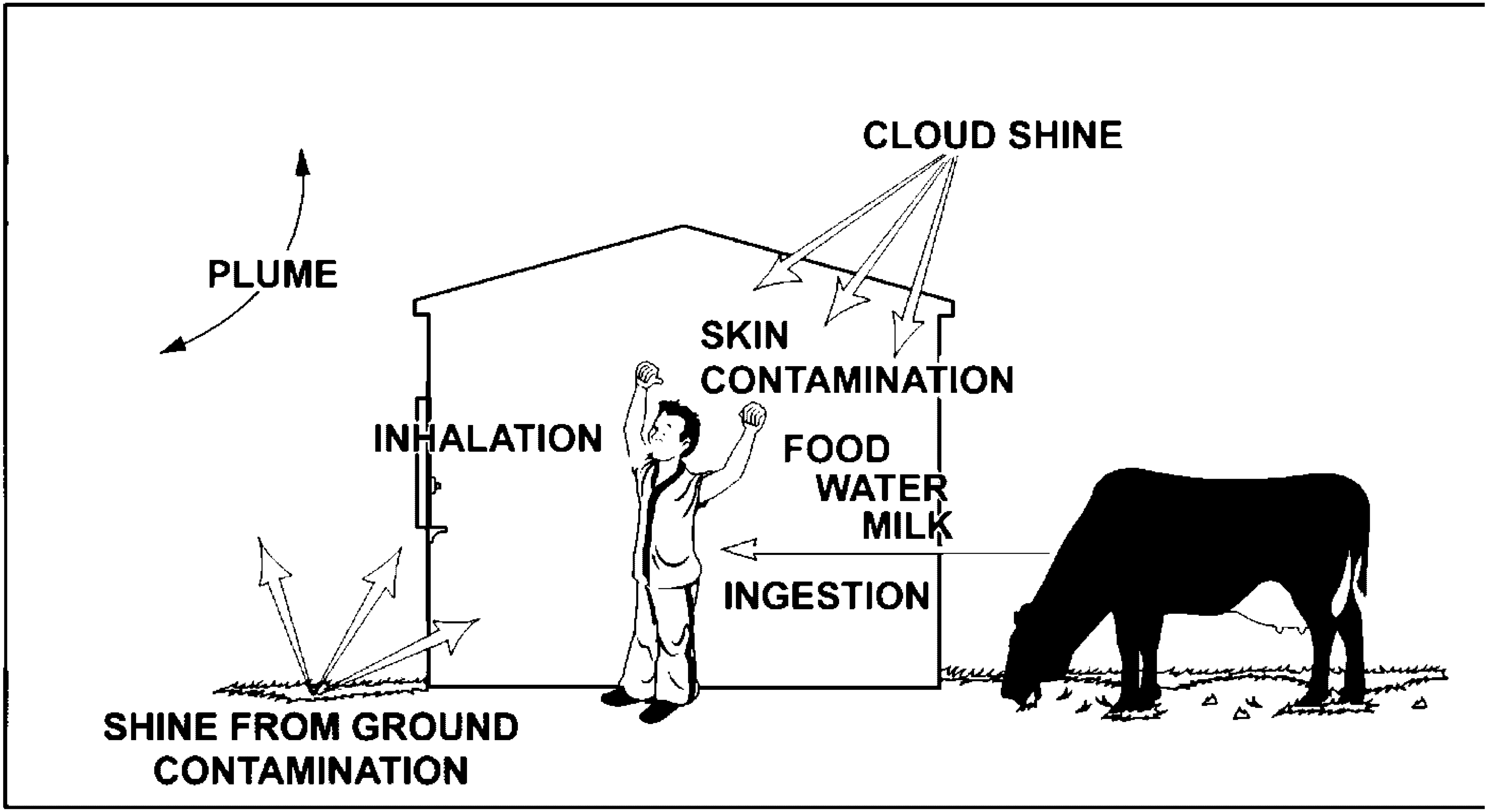
MCCI experiments with enhanced cooling by overlying water (ANL ACE and MACE)

Steam generator retention (Phebus and ARTIST experiments)

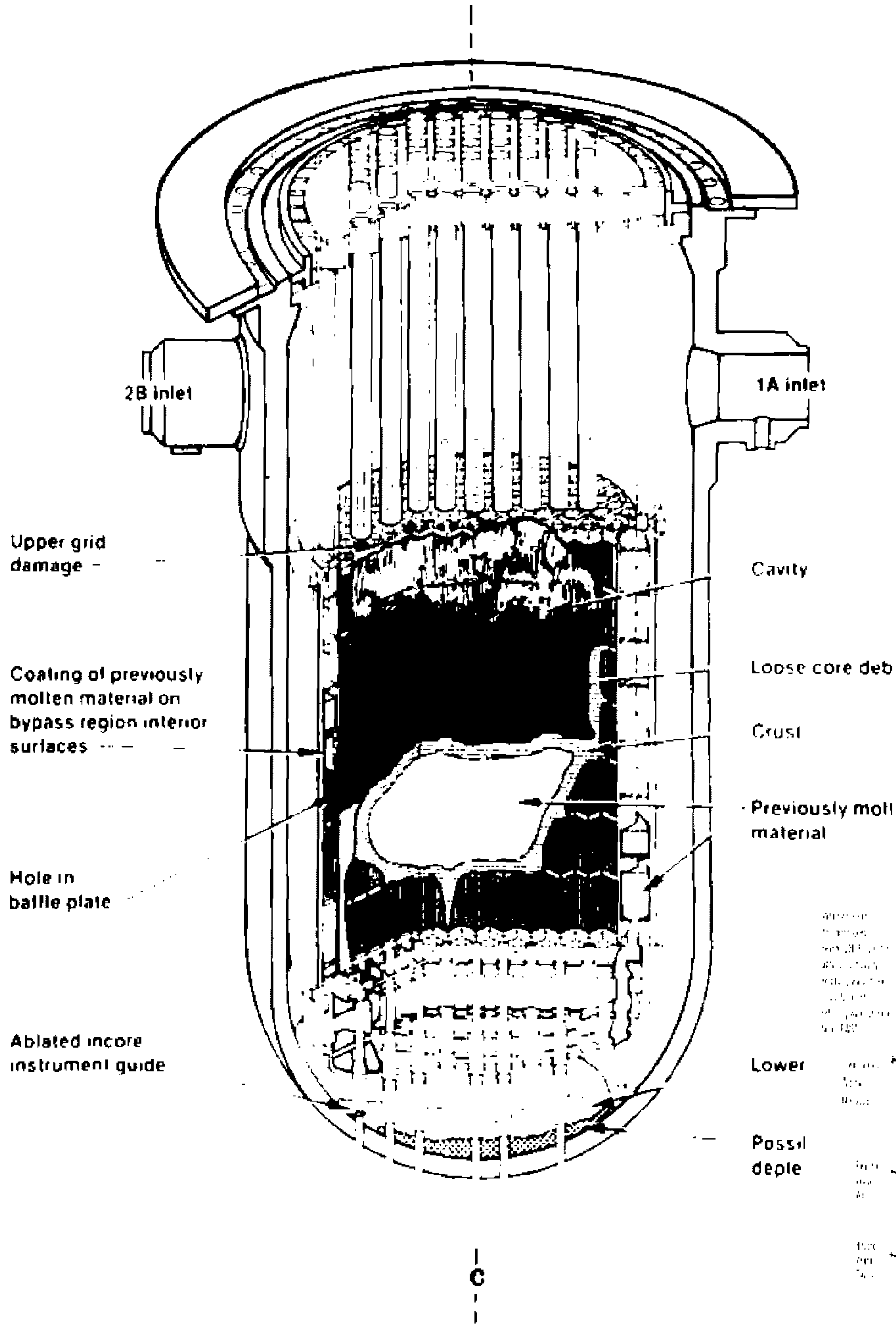
MCCI experiments: ACE, SURC, TURC, BETA

Fission Products Release and Transport

[Source Term Analysis]



CLOUD SHINE
PLUME
SKIN
CONTAMINATION
INHALATION
FOOD
WATER
MILK
INGESTION
SHINE FROM GROUND
CONTAMINATION

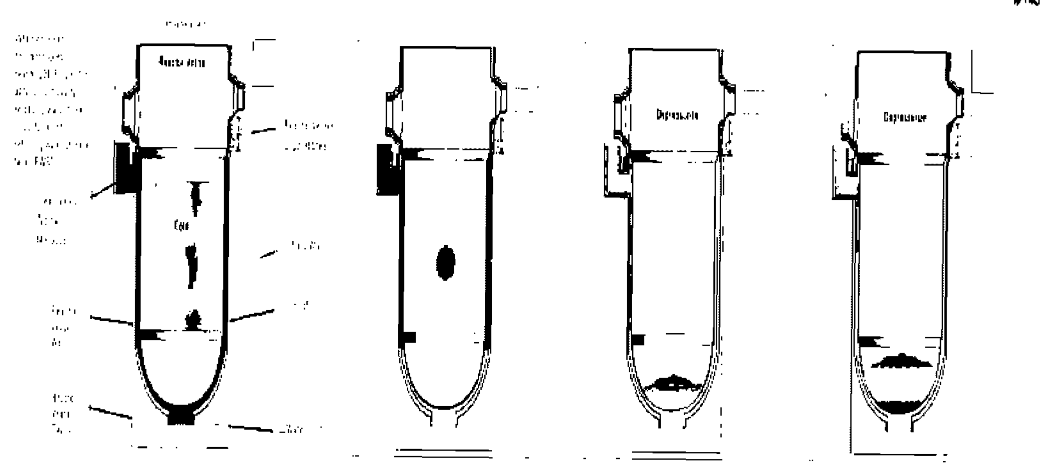


Temperature

(F)	(K)	
5162	3123	Melting of UO_2
4874	2963	Melting of ZrO_2
4712	2873	Formation of ceramic (U,Zr,O) melt
4352	2673	Formation of α-Zr(O)/UO_2 and U/UO_2 monotectics (accelerated liquefaction)
		Melting of B_4C
3722	2323	Melting of Al_2O (burnable poison rod: $Al_2O + B_4C$)
3587	2248	Melting of oxygen-stabilized α-Zr(O)
3200	2033	Melting of as-received Zircaloy-4 (Zry)
2642	1723	Melting of stainless steel or Inconel
2372	1573	Eutectic interactions of Zry with Stainless steel, Inconel and Al_2O_3
2192	1473	
2066	1403	B_4C/Fe eutectics; Formation of liquid U as a result of UO_2/Zry interactions
1724	1213	Formation of first Fe/Zr and Ni/Zr eutectics
1472	1073	Melting of (Ag, In, Cd) alloy also ballooning and rupture

Start of UO_2 Dissolution by molten Zry → formation of metallic (U, Zr, O) melt

Start of rapid Zry oxidation by H_2O → uncontrolled temperature escalation



Elements of Source Analysis

- Release from the Fuel
  - Gap release
  - Fuel degradation release
  - Ex-vessel release
- Transport to the containment
  - Aerosols
  - Vapors
- Behavior within the containment
  - Aerosol physics
  - Iodine chemistry
- Revaporization
- Engineered Safety Features

Core Heatup and Fission Product Release

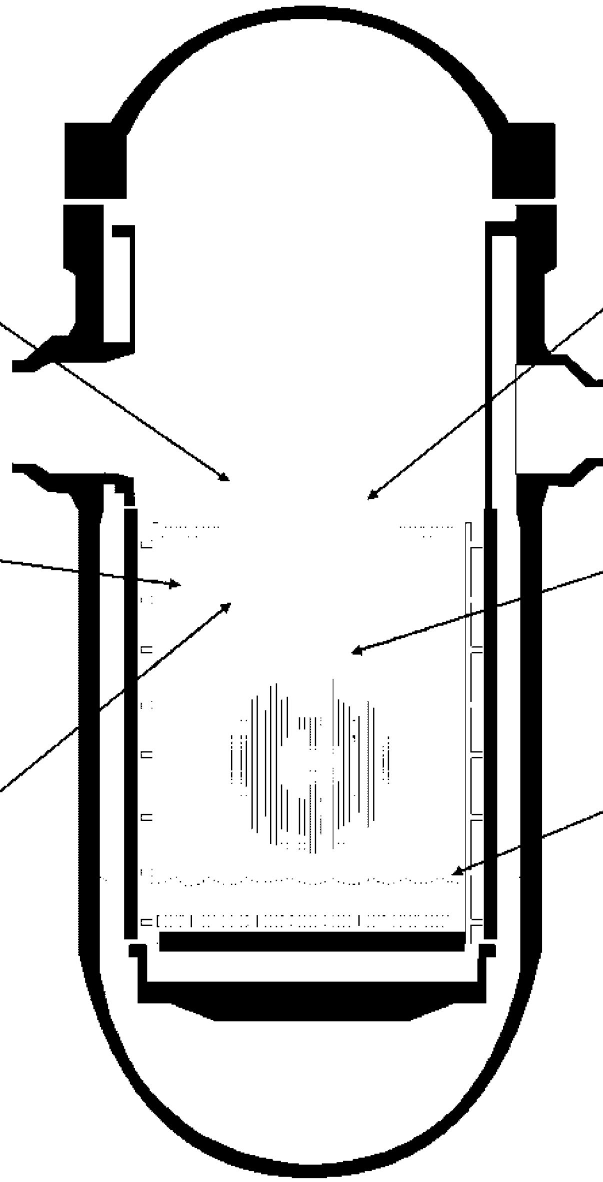
Fission product specie/volatility modified (Cs_2MoO_4) – Phebus Tests **– affects RCS deposition**

RN Package expanded to allow analysis of FP release from mixed MOX/LEU core

(French VERCORS and RT tests)

Fuel failure criteria expanded via control function – Phebus tests – affects **hydrogen generation and melt progression**

BWR failure criteria expanded



Ag release model added – Phebus Tests **– important for iodine** chemistry

- Important to agglomeration

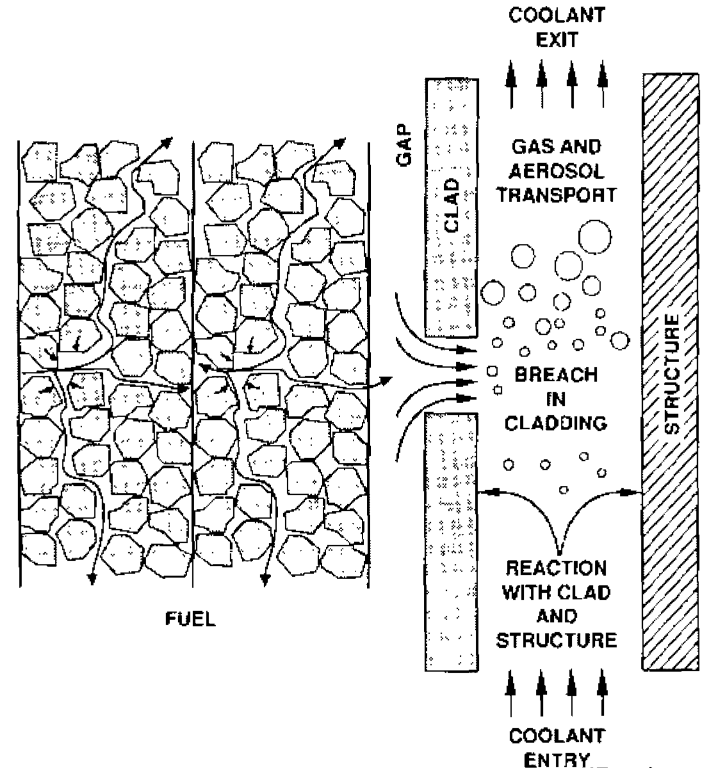
B_4C oxidation model added (PWR) – QUENCH Tests, Phebus FPT-3

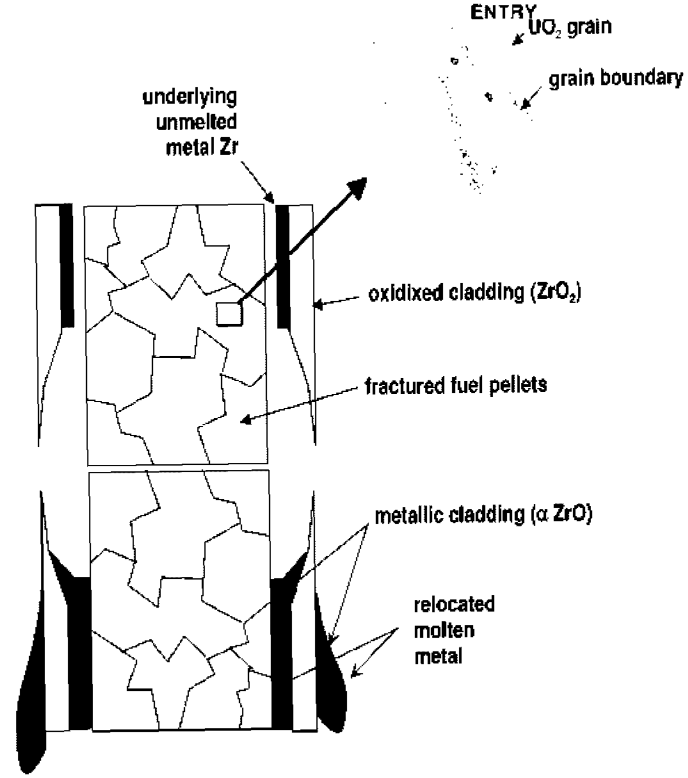
Quench-reflood modeling – QUENCH tests **– quench front not necessarily water level**

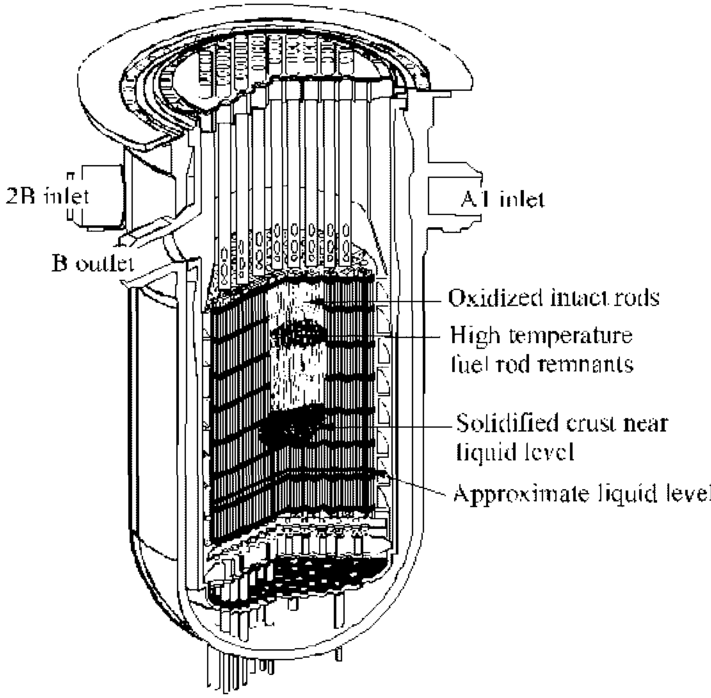
Release from the Fuel

- In-vessel Release
  - Coolant release
  - Gap release
  - Fuel degradation release
  - Air ingression following vessel failure

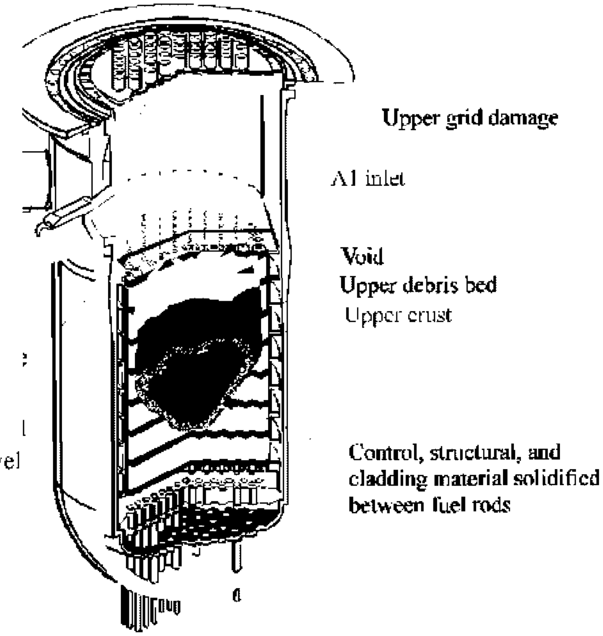
- Ex-vessel Release
  - High pressure melt ejection
  - Melt interactions with concrete
  - Steam Explosions



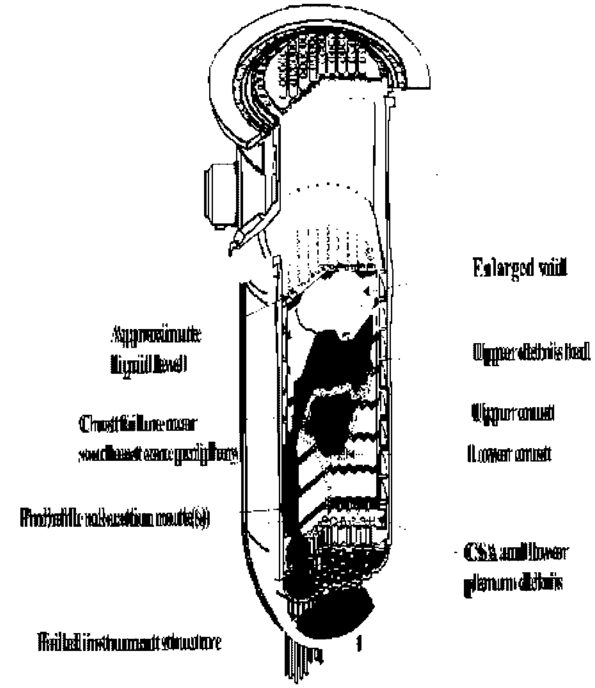




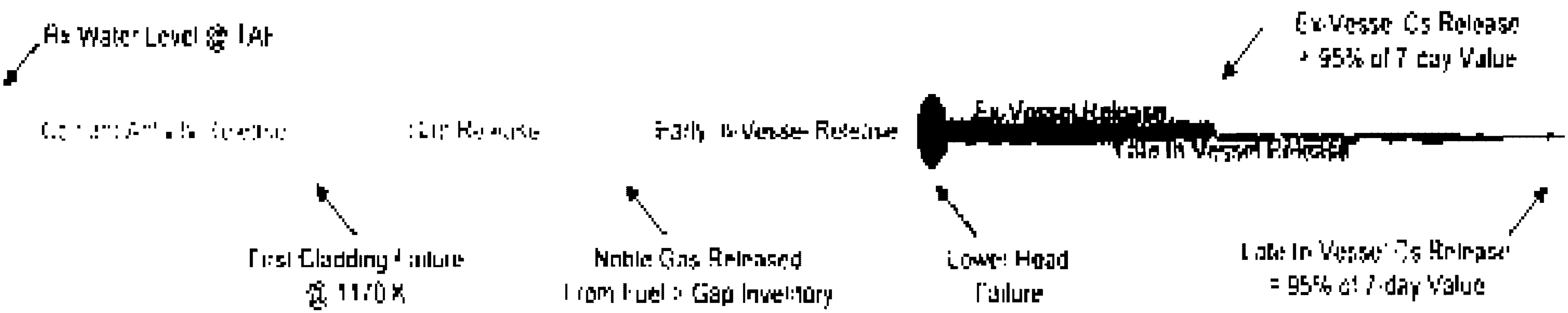
TMI – 150 min



TMI – 244 min



TMI - final



Release Phase Timing Definitions – Tie to Calculated MELCOR Results

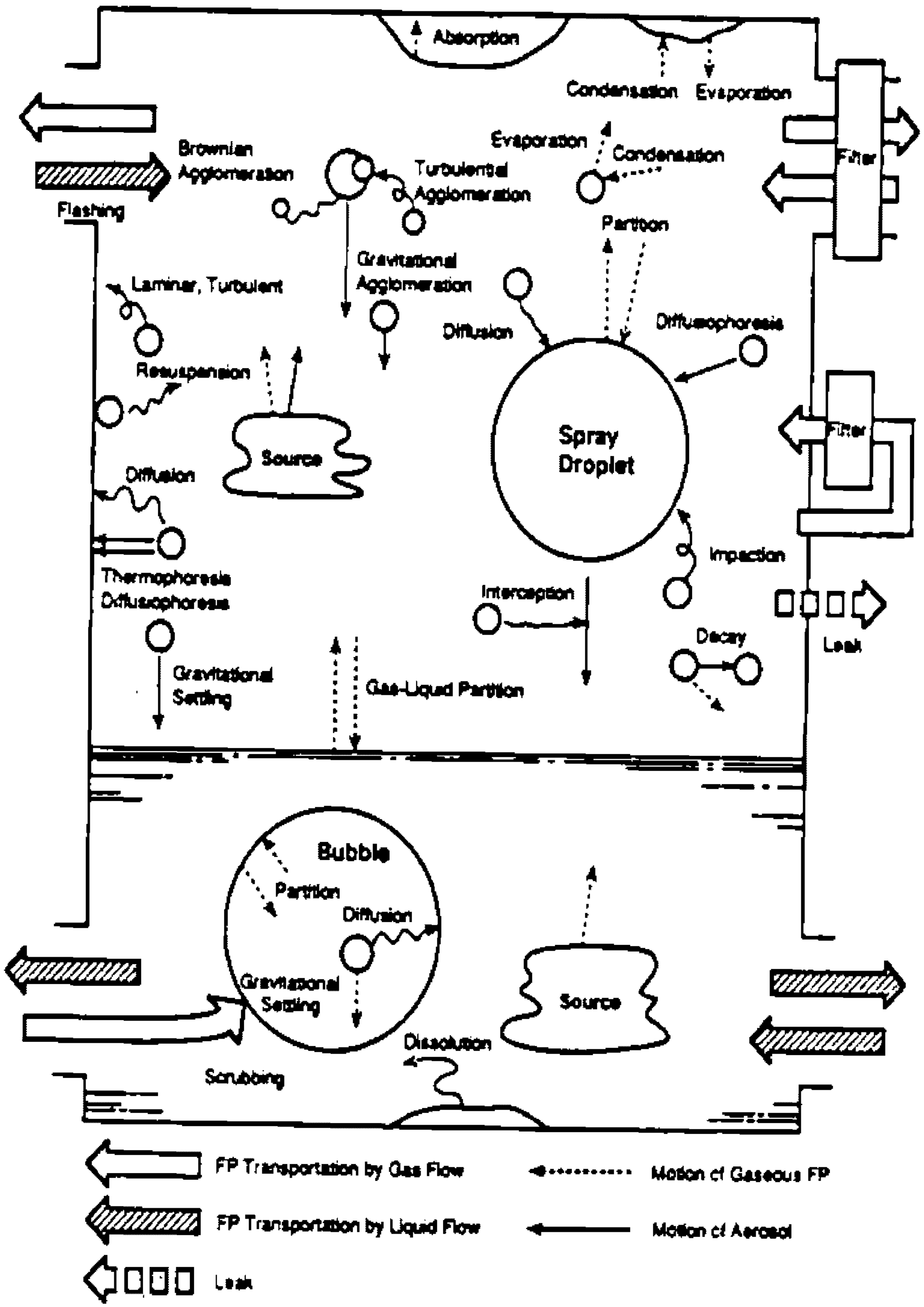
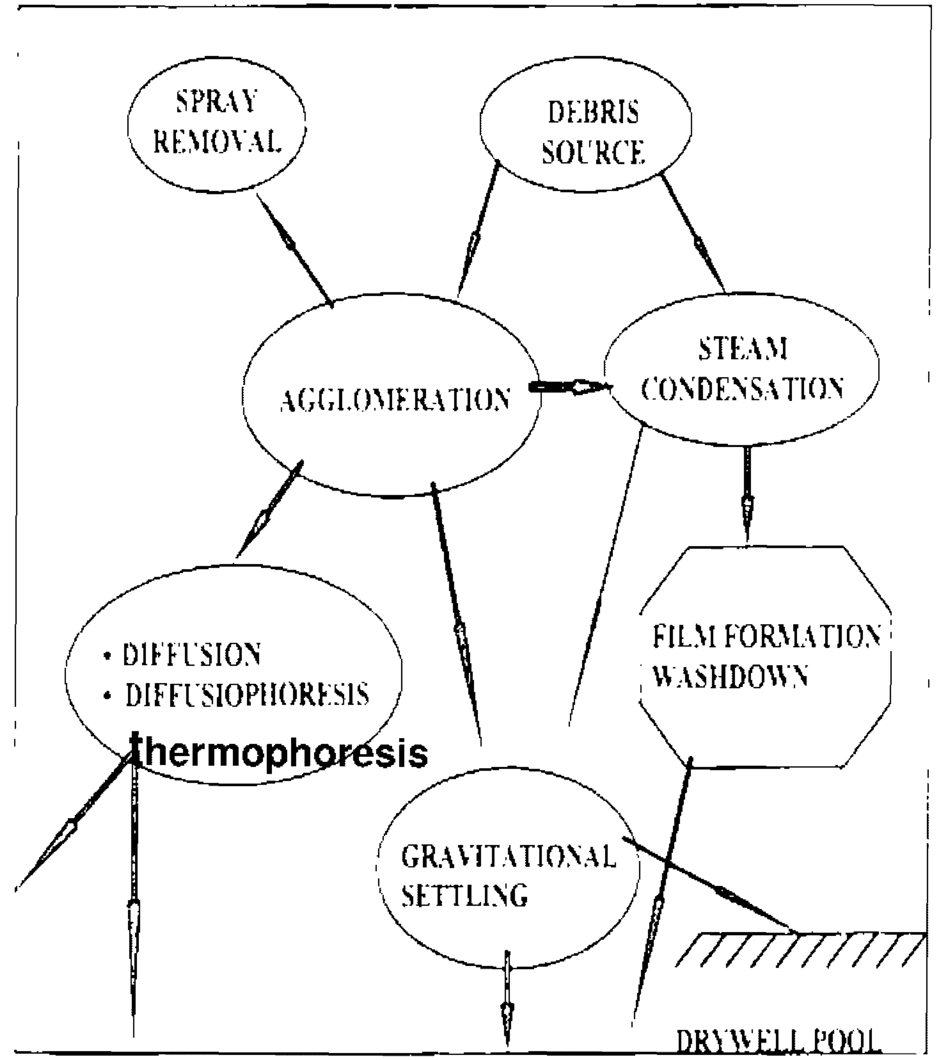
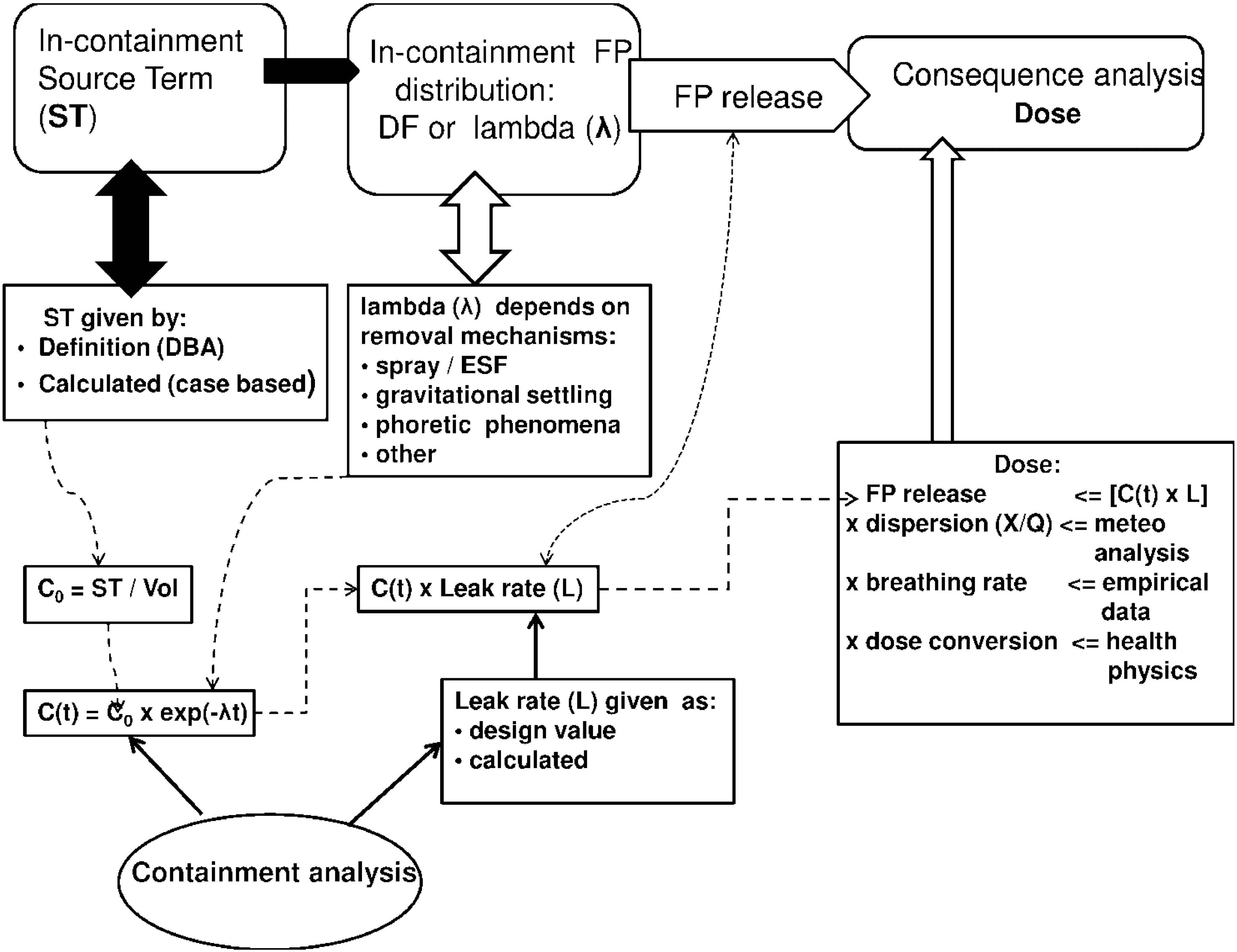


Figure 1 Models of ART for Transport Behavior of Gaseous Fission Products and Aerosol




In-containment Source Term (ST)
In-containment FP distribution: DF or lambda (λ)
FP release
Consequence analysis
Dose
ST given by:
• Definition (DBA)
• Calculated (case based)
lambda (λ) depends on removal mechanisms:
• spray / ESF
• gravitational settling
• phoretic phenomena
• other
Dose:
FP release <= [C(t) x L]
x dispersion (X/Q) <= meteo analysis
x breathing rate <= empirical data
x dose conversion <= health physics
C0 = ST / Vol
C(t) x Leak rate (L)
C(t) = C0 x exp(-λt)
Leak rate (L) given as:
• design value
• calculated
Containment analysis

Source Term (ST) Definition

amount, timing and composition of FP release

TID-14844: instantaneous of release FP
– mostly gaseous iodine

NUREGs -1150 / -1465: Alternative Source Term (AST)
- time dependent release of FP; mostly aerosols

Mechanistic Source Term (MST): to be defined (as of 2011)
- realistic, scenario-based and design-dependent FP release models

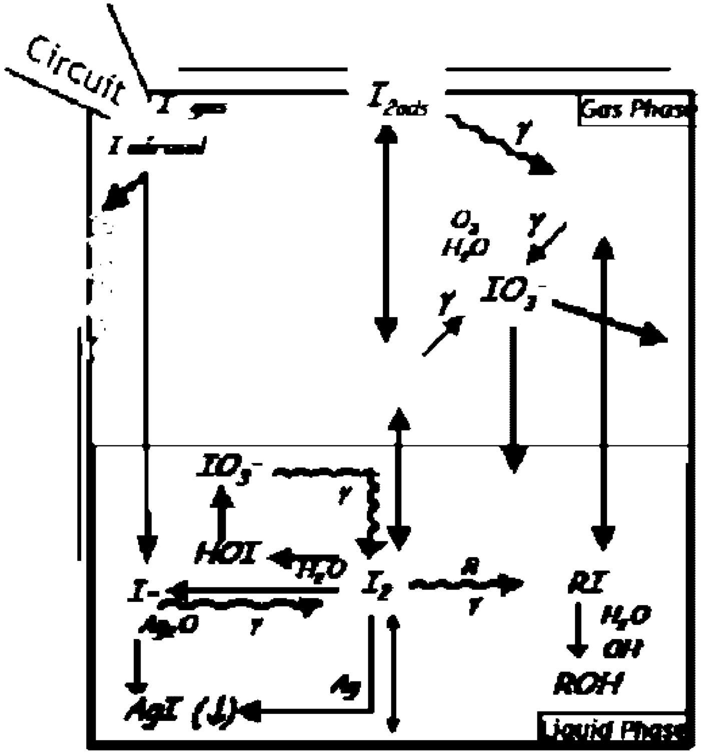
The Iodine Problem

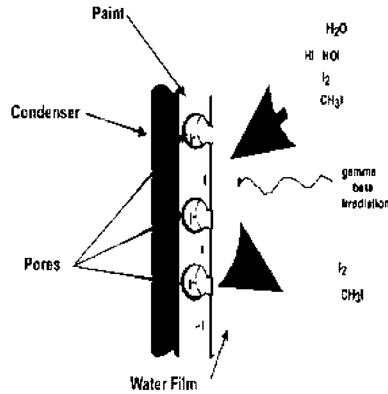
- Iodine removed from containment by sprays or water pools can partition back into the containment atmosphere

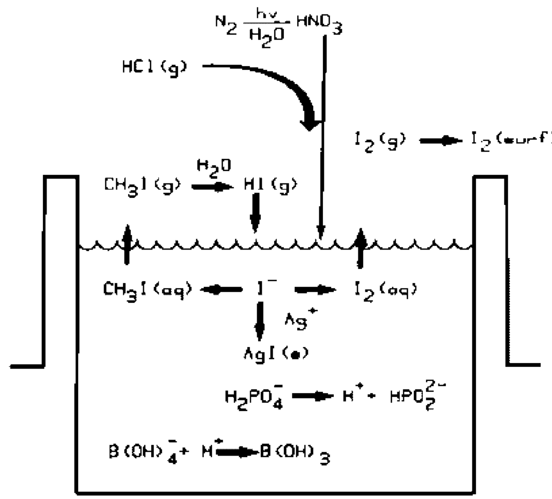
$$I_2(\text{water}) \leftrightarrow I_2(\text{gas})$$

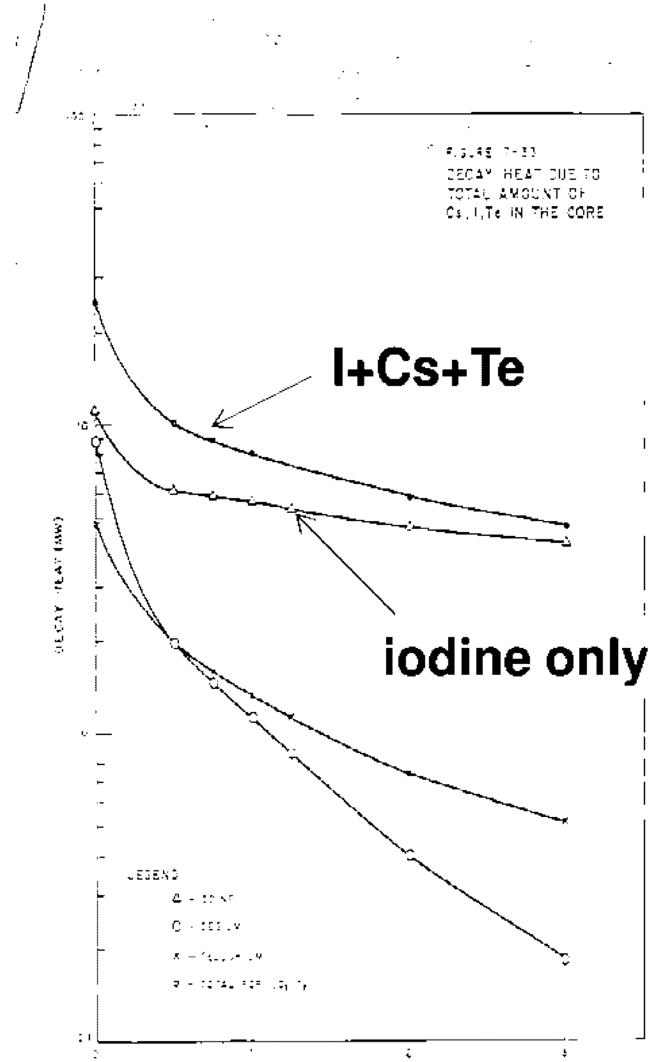
- Complicated chemistry involving radiation dose to water
- Can be suppressed by making sumps basic (pH > 7)

Iodine behavior – complex chemistry









Forms of iodine:

Volatile: I_2, CH_3I, Cs_2MoO_4 (gas)
Aerosol: CsI (soluble)
CsOH (hygroscopic)
IO_x (nano)

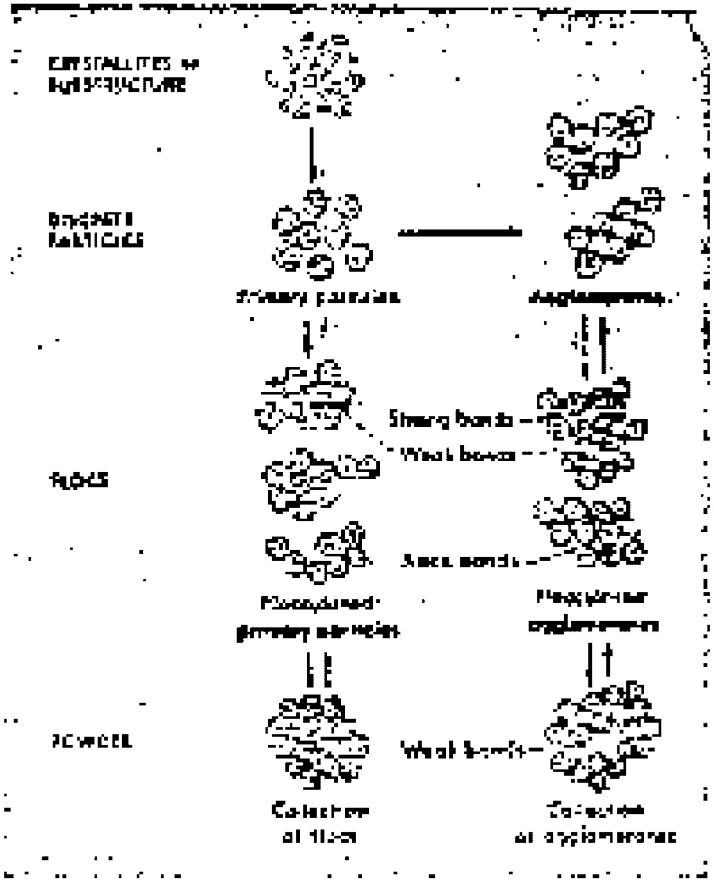
Volatile iodine removal based on regulatory guidance

Note: PHEBUS experiments show CONSTANT presence of airborne Iodine (few percent)

Aerosols removal mechanisms

WHAT IS AEROSOL?

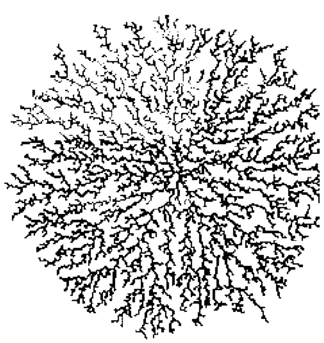
[CHEMICAL ENGINEERING, MAY 20, 1968, P.149]



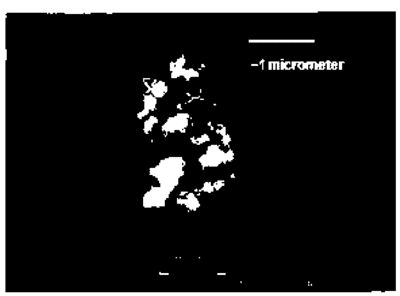
PARTICLE dispersion states and types of bonding—Fig. 1

$$\frac{\rho_{eff}}{\rho_{solid}} \cong 0.5$$

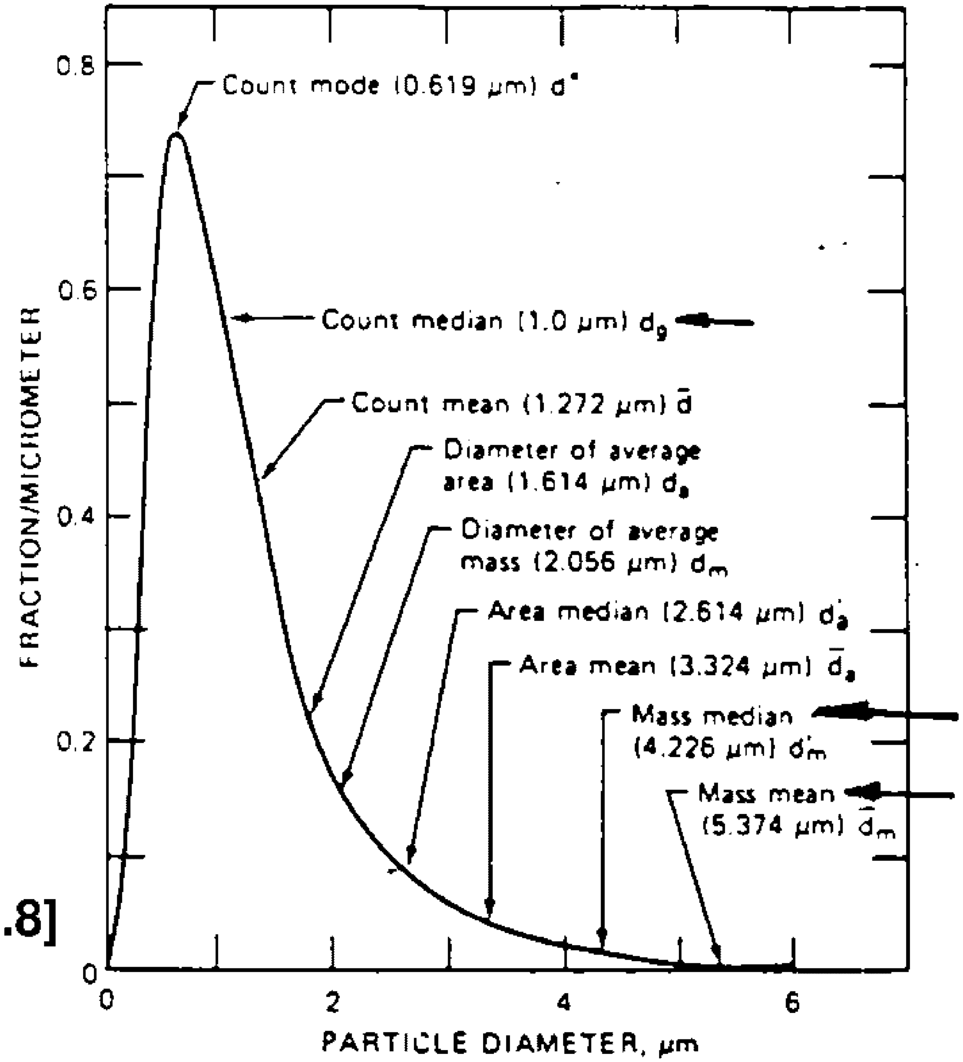
[~0.3÷>0.8]



Mitigation Systems Remove Aerosol Particles



$$y = \frac{1}{(2\pi)^{1/2} \ln \sigma_g} \exp\left[-\frac{(\ln d - \ln d_g)^2}{2 \ln^2 \sigma_g}\right]$$



An example of the log-normal distribution function in generalized linear form for $d_g = 1.0$ and $\sigma_g = 2.0$.

[FROM TID-26608]

Basic equation: Smoluchowski [1916]

[Equation from NUREG/CR-6189]

Aerosol Dynamic Equation

When the homogeneous aerosol assumption has been made, the aerosol dynamic equation is:

$$\frac{\partial n(v,t)}{\partial t} = \frac{1}{2}\int_0^v K[U,v-U]\, n(U,t)\, n(v-U,t)\, dU - n(v,t)\int_0^\infty K[U,v]\, n(U,t)\, dU$$

$$+ \frac{S(v,t)}{V} - \frac{R(v,t)\, n(v,t)}{V} - \frac{\partial I(v,t)\, n(v,t)/V}{\partial v}$$

where:

$n(v,t)$ = number concentration of particles having volumes of v to v + dv,

$\int_0^v K[U,v-U]\, n(U,t)\, n(v-U,t)\, dU$ = the rate of formation of particles of volume v to v + dv by coagulation of smaller particles,

$n(v,t)\int_0^v K[U,v]\, n(v,t)\, dU$ = the rate of coagulation of particles of volume v to v + dv to form larger particles.

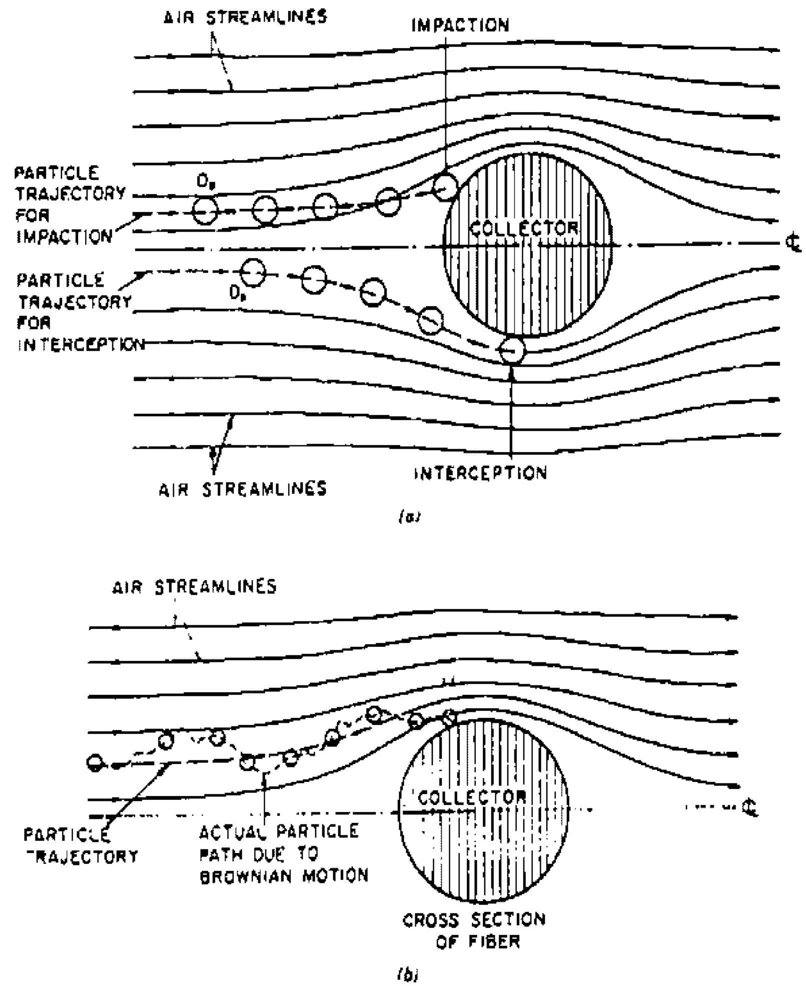
$K[U,v]$ = coagulation "kernel" for particles of volume v with particles of volume U,

$S(v,t)$ = rate at which particles of volume v to v + dv are supplied,

V = containment volume,

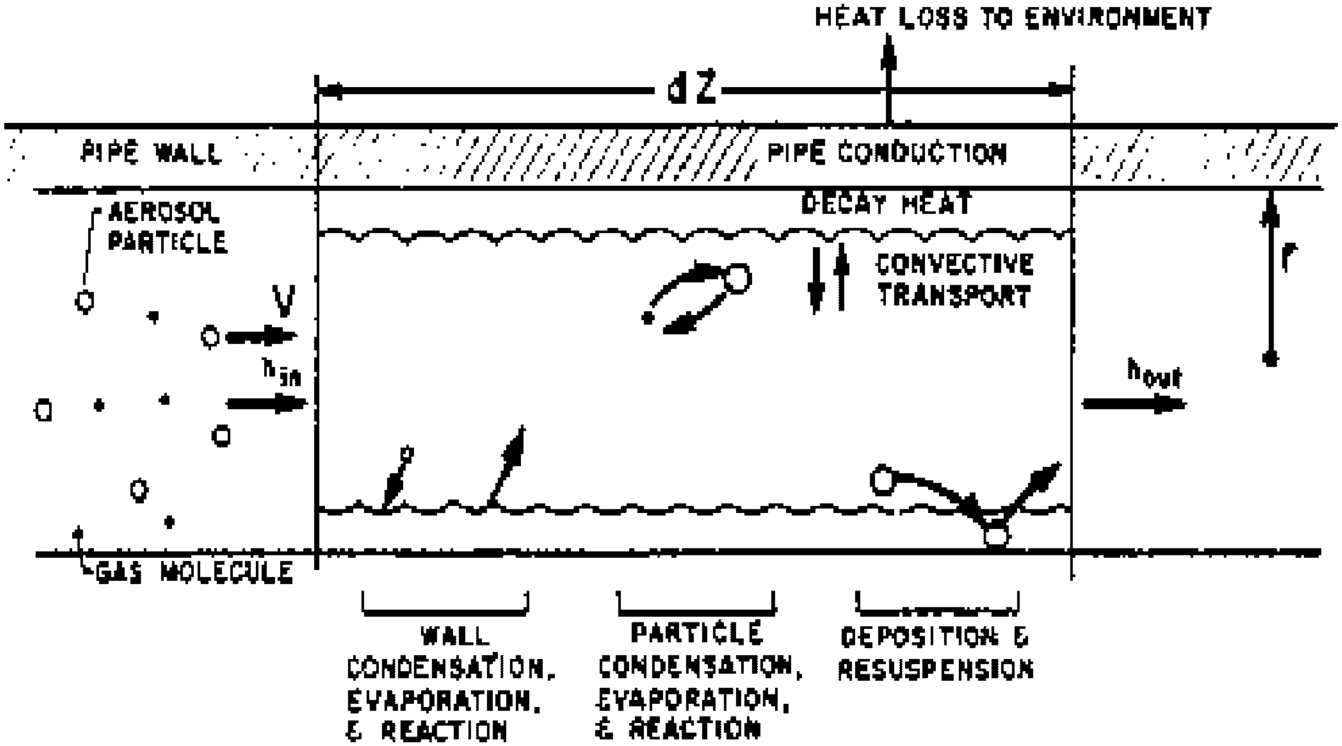
$R(v,t)\, n(v,t)$ = rate of removal of particles from the containment by any of a variety of mechanisms,

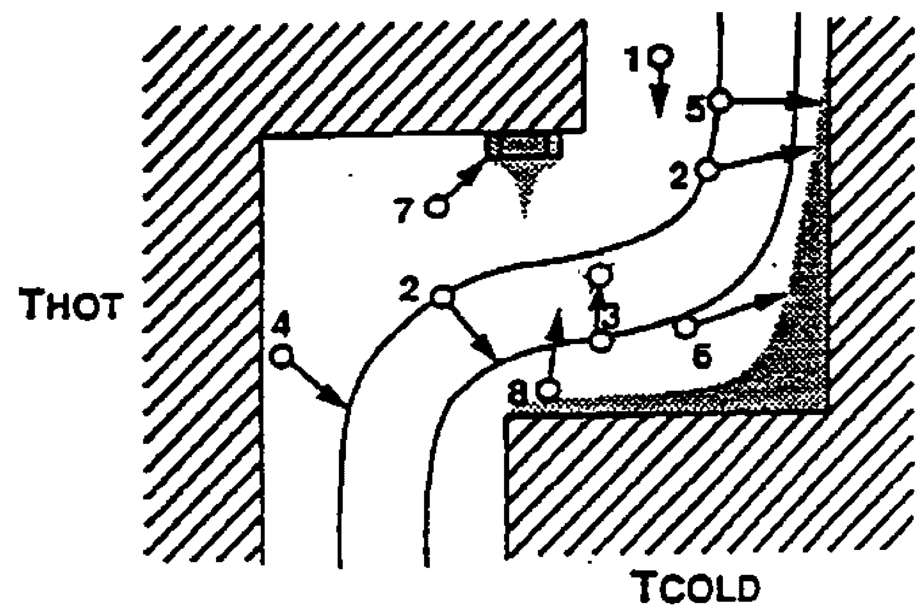
$\frac{\partial I(v,t)\, n(v,t)}{\partial v}$ = rate of growth by condensation of particles from the volume interval of v to v + dv.



Modes of Particle Collection :
(a) Impaction and Interception (b) Diffusion

TRANSPORT PHENOMENA IN PIPE





1. Gravity
2. Turbulence
3. Brownian Motion
4. Thermophoretic
5. Vapor Deposition
6. Inertia(Bends)
7. Irregularities
8. Re-entrainment

Processes affecting aerosol behavior in complex geometry

Aerosol depletion mechanisms

NATURAL PROCESSES

* GRAVITATIONAL SETTLING: STOKES VELOCITY

$$V_D = \frac{\rho_p d_p^2 g}{18 \mu_g \chi} C_u$$

χ SHAPE FACTOR

C_u = CUNNINGHAM SLIP CORRECTION $\Rightarrow f(K_n)$

K_n = KNUDSEN NUMBER $= \left[\frac{\text{MEAN FREE PATH}}{\text{RADIUS}}\right]$

* BROWNIAN DIFFUSION

* DIFFUSIOPHORESIS

  PROCESS ASSOCIATED WITH MOLAR STEAM FLUX TOWARDS CONDENSING SURFACE.

* THERMOPHORESIS

  PROCESS ASSOCIATED WITH TEMPERATURE GRADIENT. PARTICLES TEND TO MIGRATE FROM HOTTER TO COLDER REGION.

ENGINEERED SAFETY FEATURES

* SPRAY
* SUPPRESSION POOL
* FILTERS

Parameters to review:

Aerosol size distribution

Assumed aerosol density

Condensation rate AND steam density for (DhP) (based on total - NOT partial - pressure)

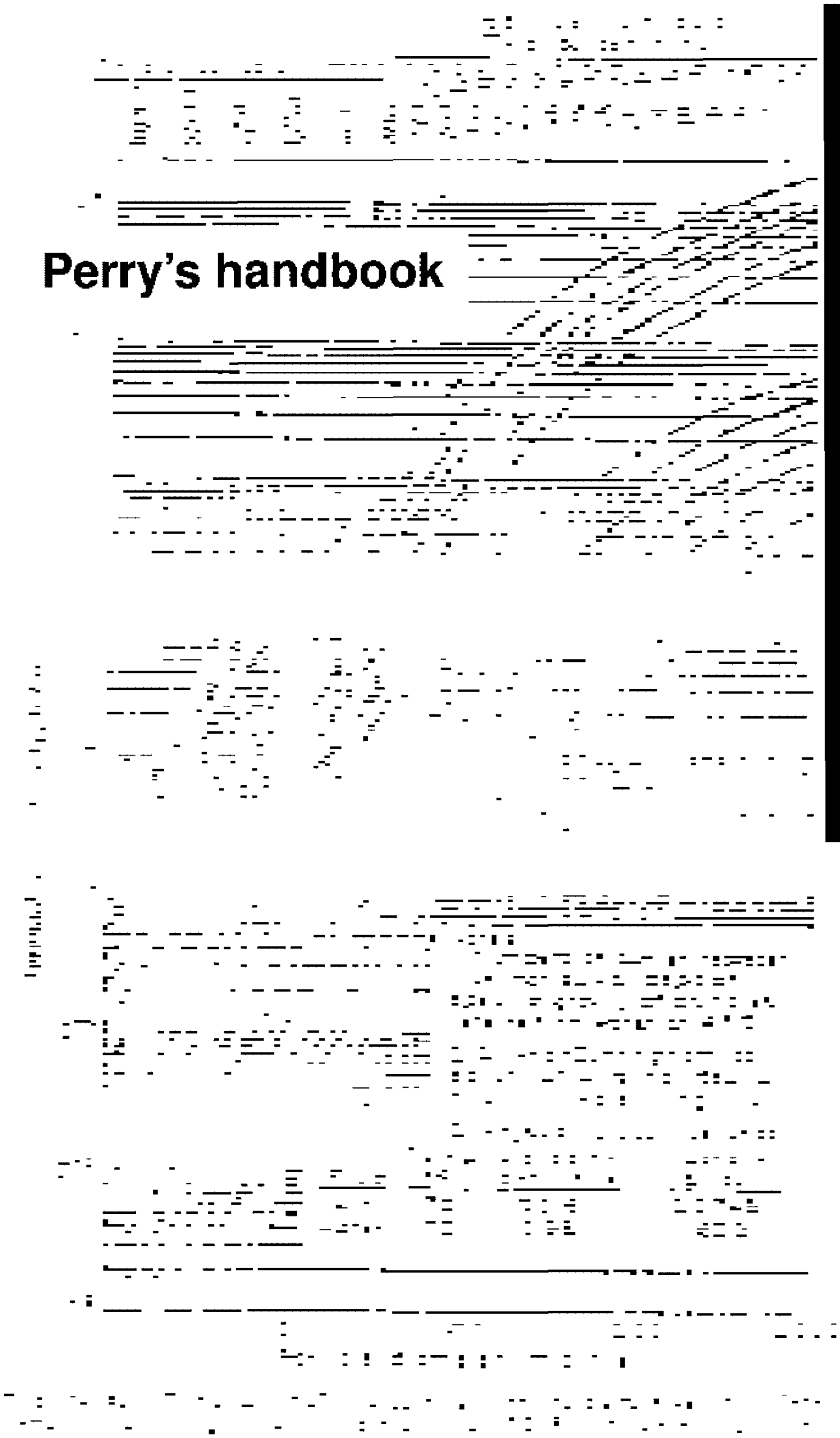
Convective heat flux ONLY (ThP)

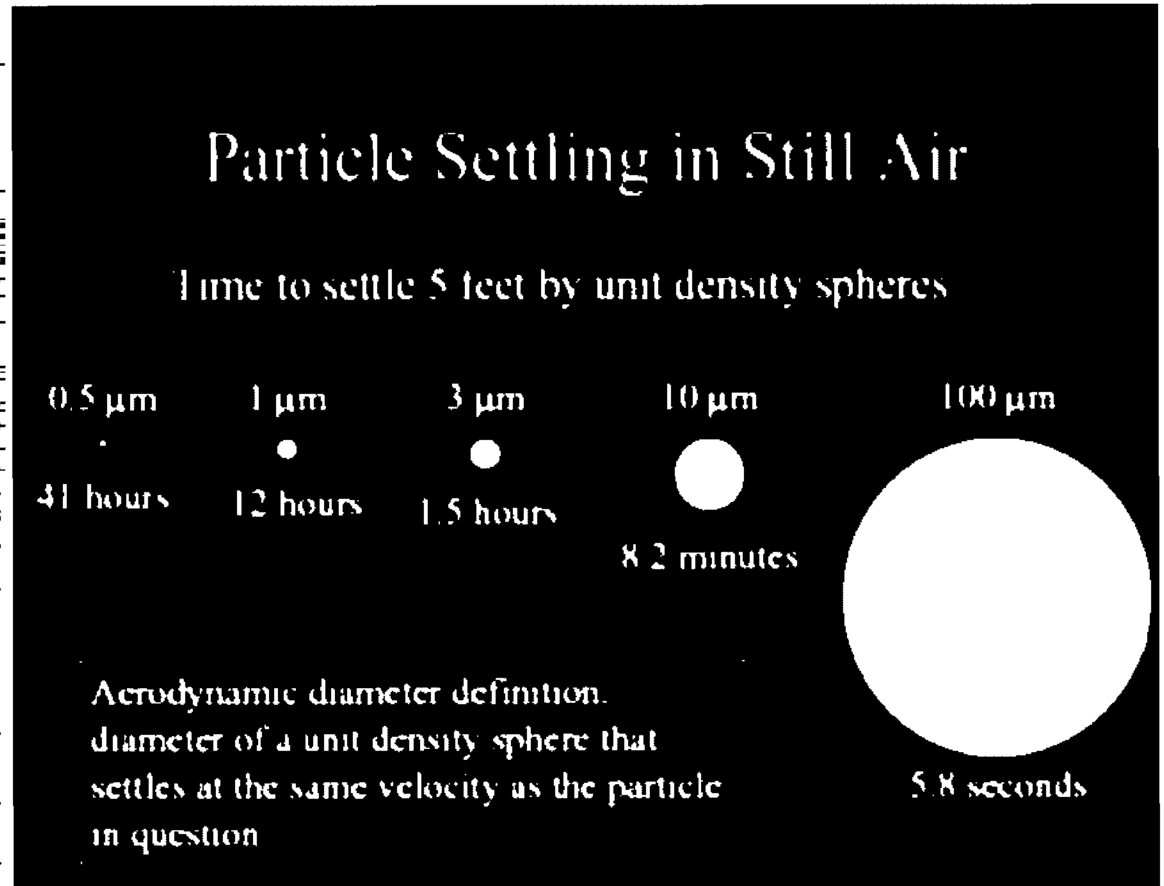
Spray drop size

Suppression pool DF

Filter DF

Perry's handbook





$$V_D = \frac{\rho_p d_p^2 g}{18 \mu_g \chi} C_u$$

SHAPE FACTOR

C_u = CUNNINGHAM SLIP CORRECTION $\Rightarrow f(K_n)$

K_n = KNUDSEN NUMBER $= \left[\frac{\text{MEAN FREE PATH}}{\text{RADIUS}}\right]$

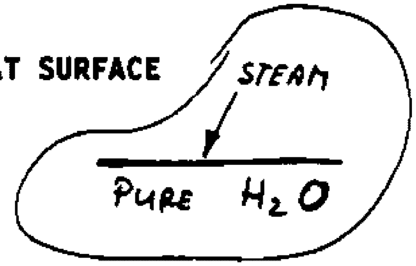
Parameter to review: particle density

Gravitational settling enhanced by condensation on aerosol

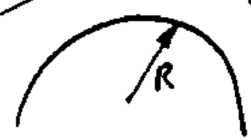
$$\dot{M} \sim \left(P_{PS} - P_S(T)\right) \Rightarrow P_{S,PURE,FLAT}\left[S - K_R * K_{SLN}\right]$$

$P_S(T) = P_{S,PURE,FLAT}(T) \bullet K_R(R) \bullet K_{SLN}$

$P_{S,PURE,FLAT}$: PURE WATER, FLAT SURFACE



$K_R(R)$: KELVIN CORRECTION FOR CURVATURE

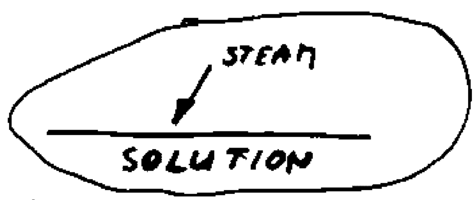


$$K_R = EXP\left(A/R\right)$$

$\dot{M} = 0$ FOR $R \leq R_{CRIT}$ FROM $\underline{S - K_R * K_{SLN} = 0}$

$K_{SLN} = K_{HYGRO}$: HYGROSCOPIC CORRECTION FOR SOLUTION

(CsI, CsOH)



SOLUTION REDUCES WATER VAPOR PRESSURE, THUS EFFECTIVELY REDUCING VALUE OF CRITICAL RADIUS.

MASON EQUATION:

SOLN... KELVIN EFFECT

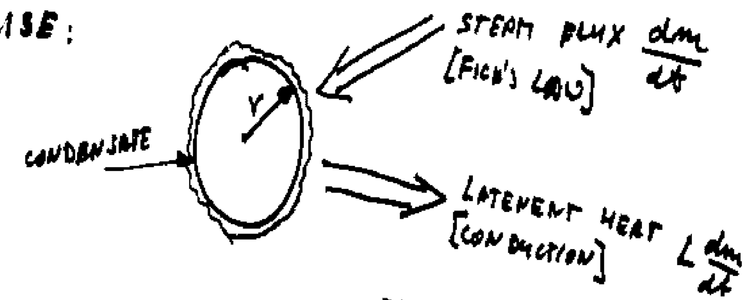
$$\frac{dm}{dt} = \frac{2\pi d_p\left[\frac{Q_s}{Q_{eq}} - a_w \boxed{\exp\left[\frac{4\sigma M_w}{RT\rho_w d_p}\right]}\right]}{\frac{1}{Q_{eq}Đ} + \frac{M_w h_{fg}^2}{Rk_{gas}T^2} - \frac{h_{fg}}{k_{gas}T}}$$

$Q_S \equiv P_{STEAM}$

$Q_{EQ} \equiv P_{SAT}(T)$

$\frac{Q_S}{Q_{EQ}} \Rightarrow$ SUPERSATURATION RATIO (RELATIVE HUMIDITY)

EXERCISE:



HINTS: ① ASSUME DROPLET $T = T_\infty$

② USE CLAPEYRON EQ FOR $\left(\frac{dP_s}{dT}\right)$

ANSWER: e.g. "ELEMENTS OF CLOUD PHYSICS"
H. R. BYERS, p. 122
UNIV. OF CHICAGO PRESS

HYGROSCOPIC CORRECTION

$$r \frac{dr}{dt} = [S - K_H{*}\mathrm{Exp}(A)]/B$$

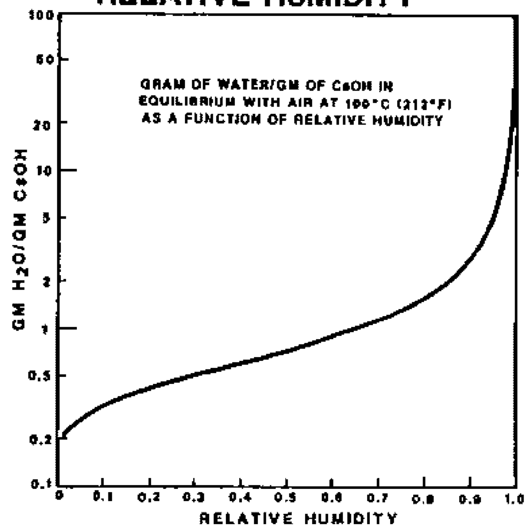
S = SATURATION INDEX = Pp/Ps (t)

A,B = THERMODYNAMIC FUNCTIONS

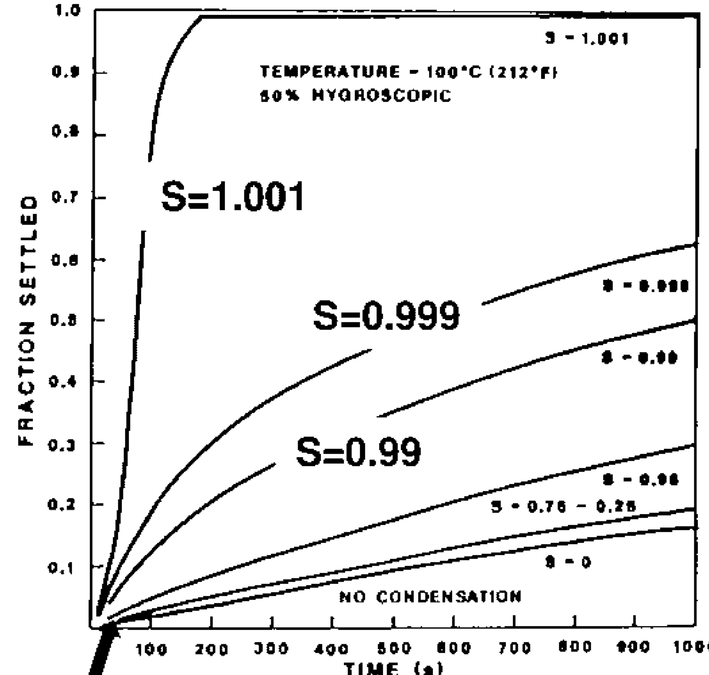
K_H = HYGROSCOPIC CORRECTION

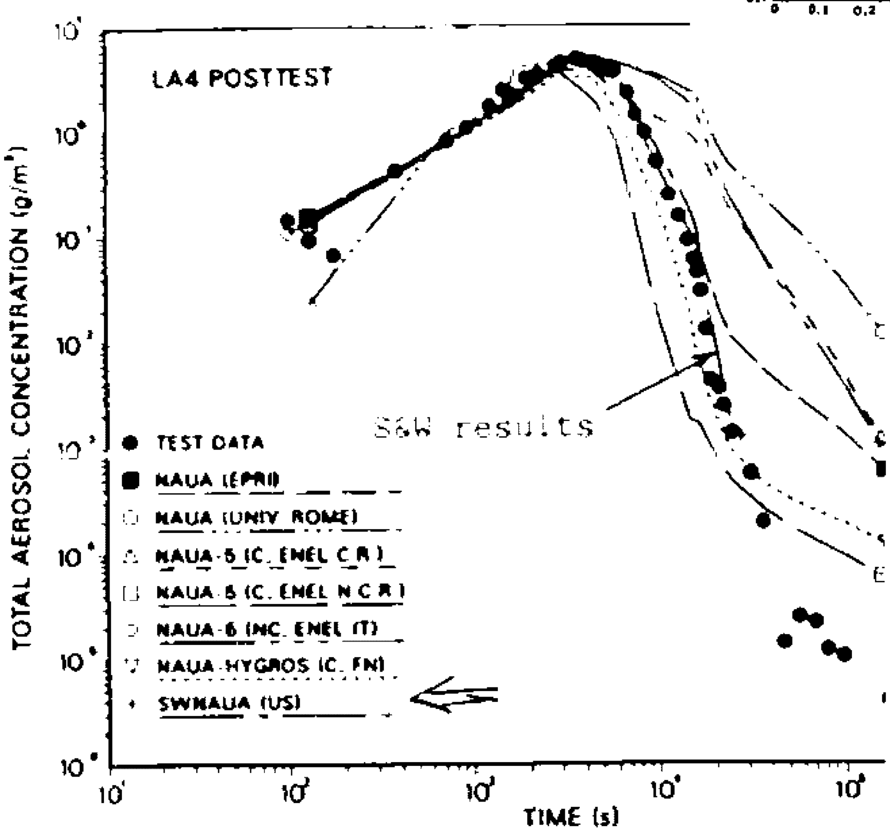
$$= \frac{\text{Ps, SOLUTION, FLAT}}{\text{Ps, PURE WATER, FLAT}}$$

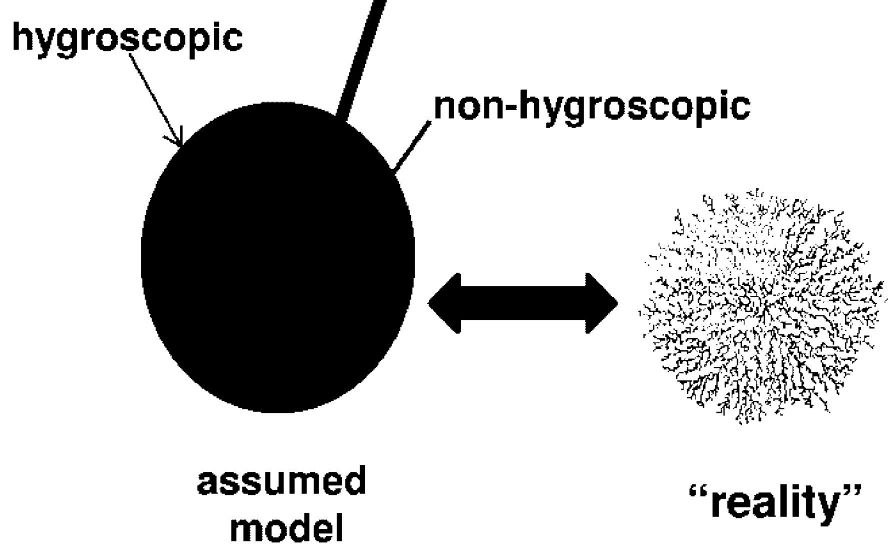
GRAM OF H_2O/GRAM OF CsOH VS RELATIVE HUMIDITY



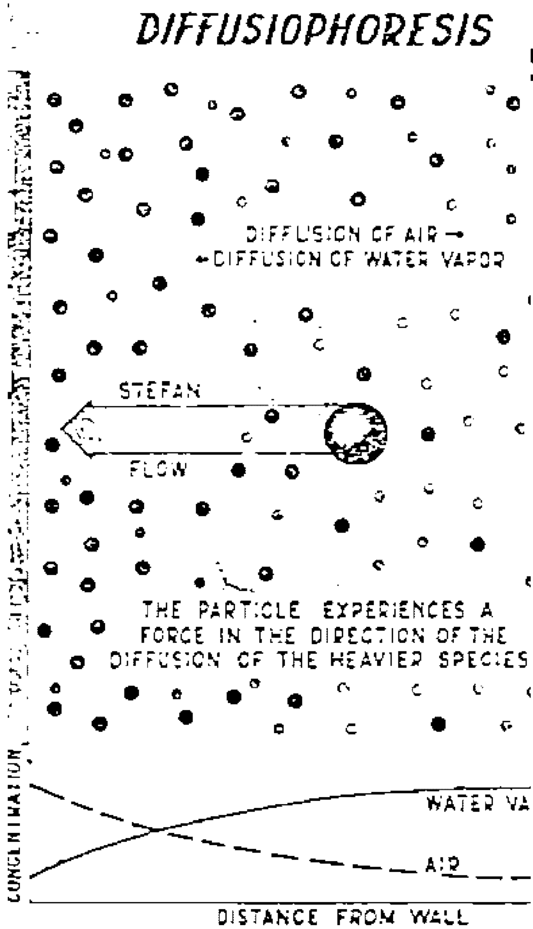
FRACTION SETTLED VS TIME







SIMPLE MODEL: STEFAN FLOW



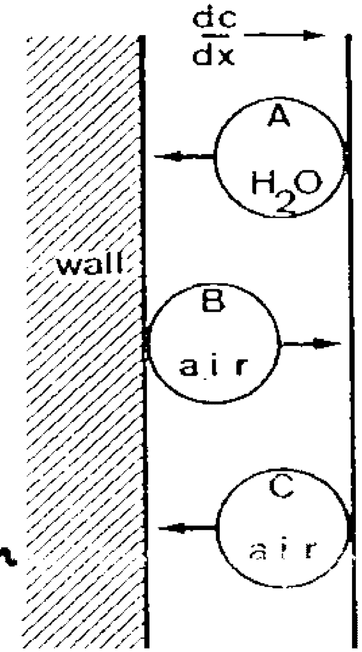
$$V_{SEP} = K \left(\frac{\dot{m}_{COND}}{S_v \; A_{SURF}} \right)$$

STEAM DENSITY

SLIP CORRECTION

e.g. $\left[\frac{M_v^{1/2}}{X_v M_v^{1/2} + X_g M_g^{1/2}} \right]$

v → vapor
g → gas
X → mole fraction



Stetter's "back flow" model

Model (large particles) rate constant, K

STEFAN FLOW (SF)
mean molar velocity of fluid

$\frac{\dot{m}_{COND}/\rho_{STEAM}}{V_{CONT}}$

SCHMITT/WALDMANN (SW)

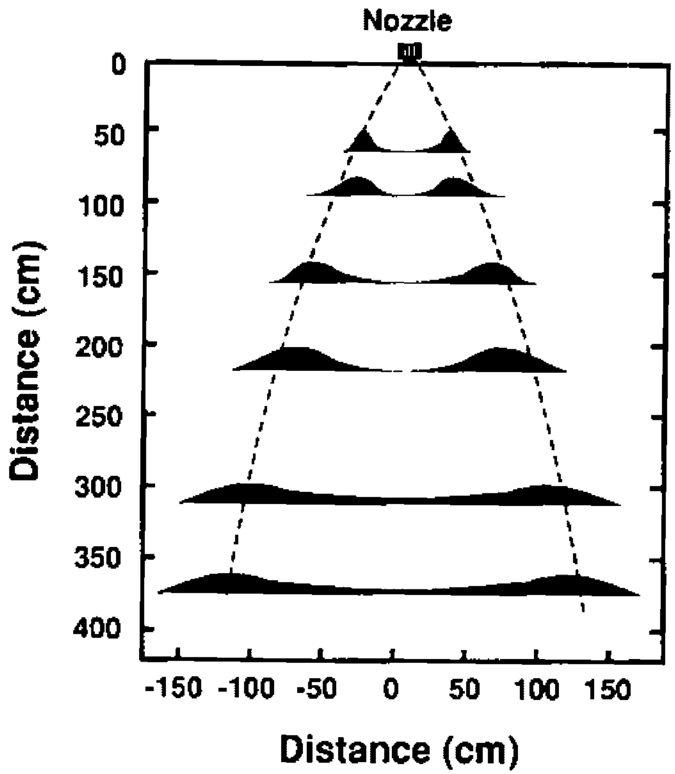
WHITMORE/MEISEN (WM)
mean mass velocity of fluid

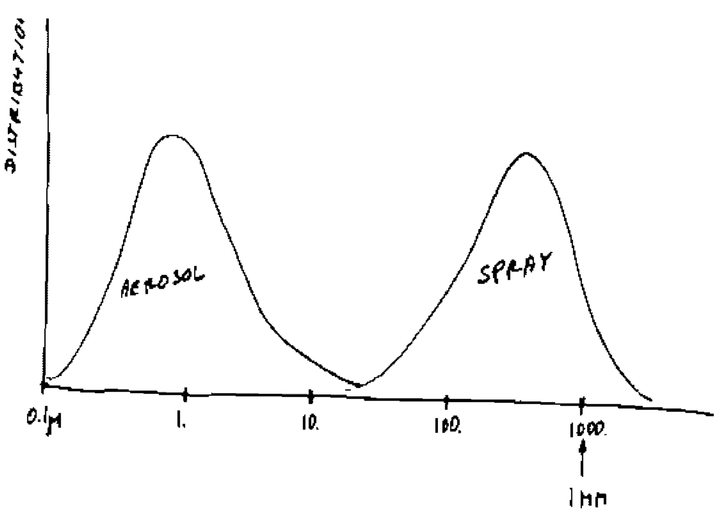
$\frac{M_2}{x_1 M_1 + x_2 M_2}$

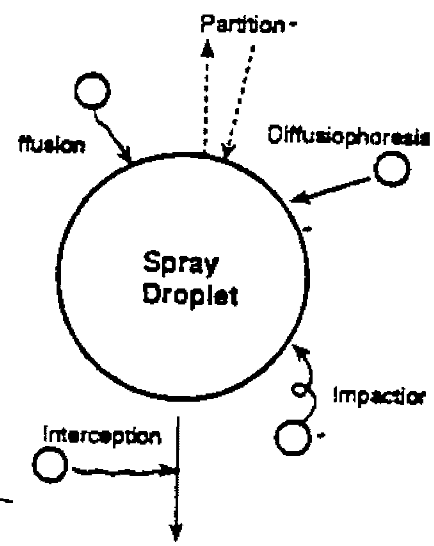
Parameter to review: steam density

Conservative assumption: steam density based on total and NOT partial pressure

Aerosol removal by spray







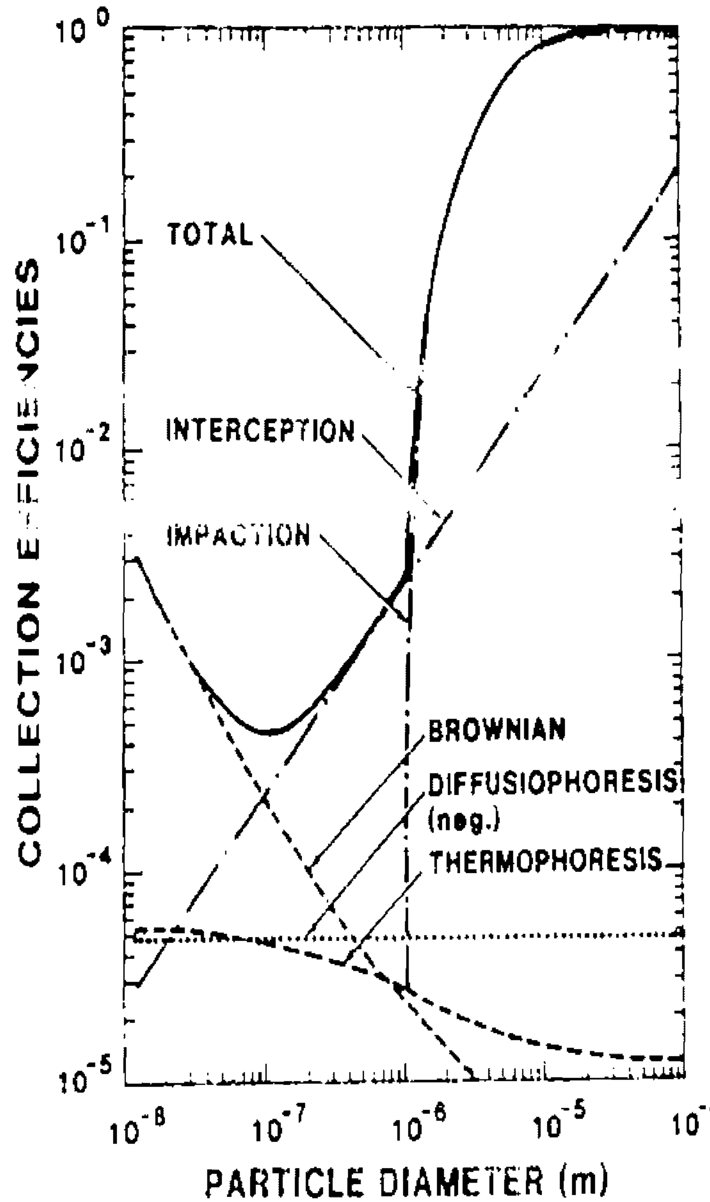
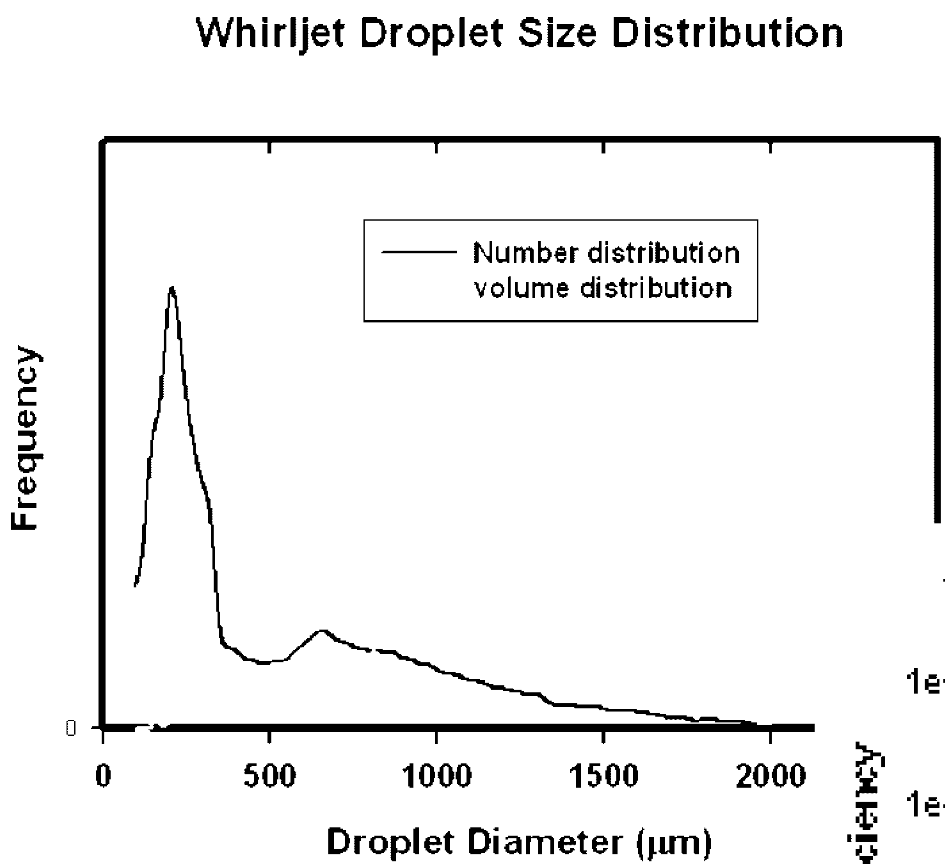
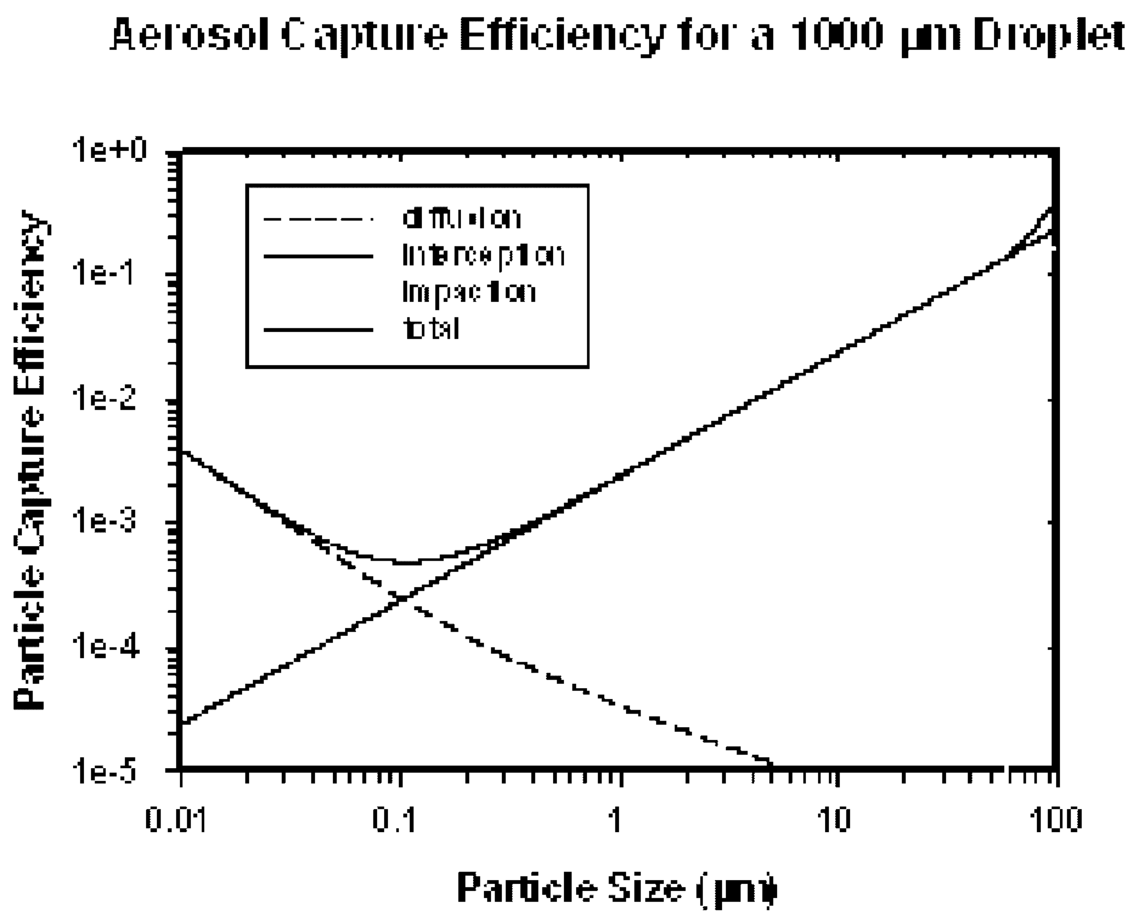
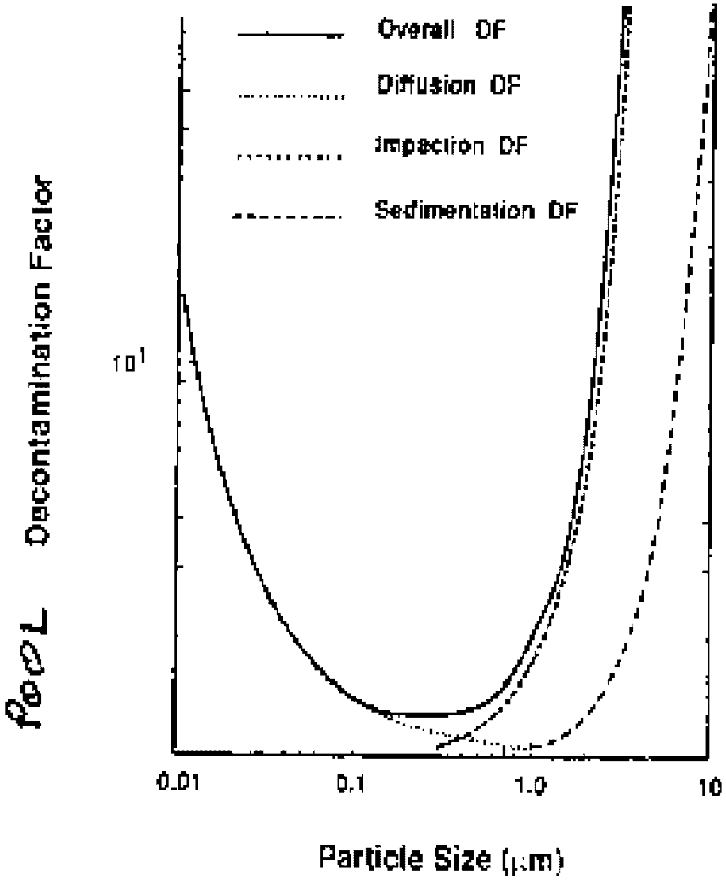


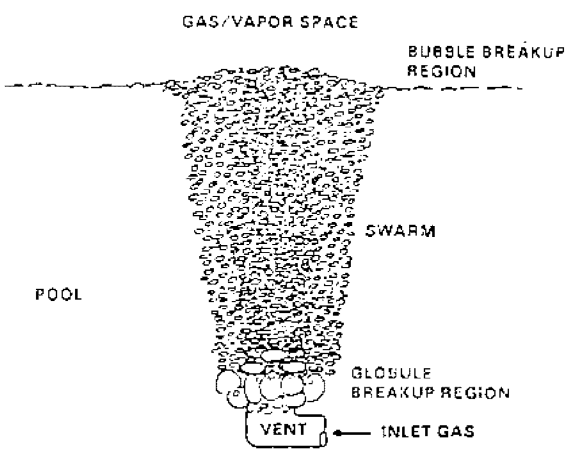
Fig. 13. Aerosol collection efficiencies for a 1000-μm spra drop as a function of particle size for the collectio mechanisms treated in the CONTAIN spray model. Th diffusiophoretic effect is negative for the containment cond


Whirljet Droplet Size Distribution
Number distribution
volume distribution
Frequency
0
0
500
1000
1500
2000
Droplet Diameter (µm)

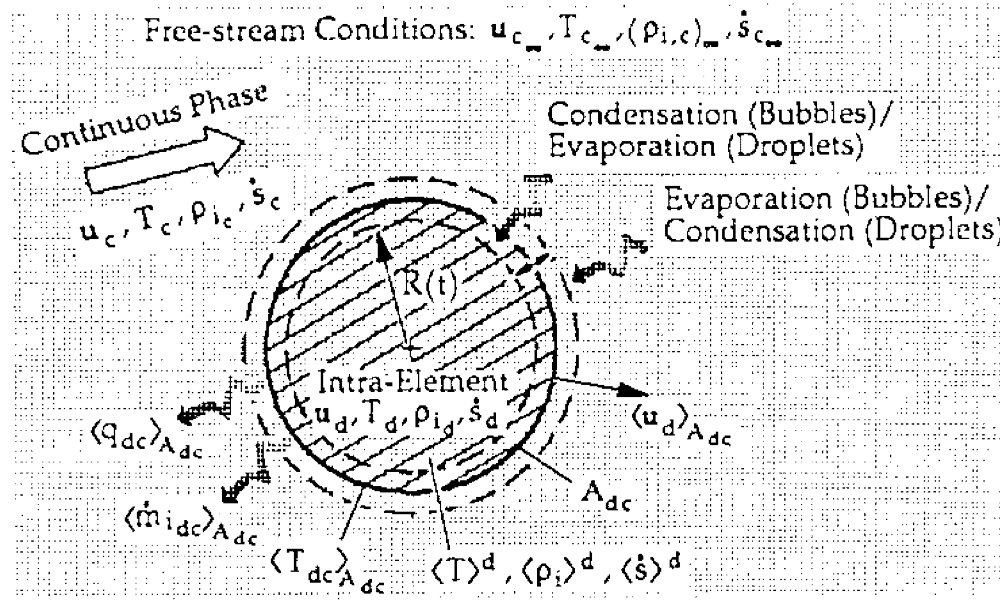

Aerosol Capture Efficiency for a 1000 µm Droplet
diffusion
interception
impaction
total
Particle Capture Efficiency
1e+0
1e-1
1e-2
1e-3
1e-4
1e-5
0.01
0.1
1
10
100
Particle Size (µm)

Pool scrubbing

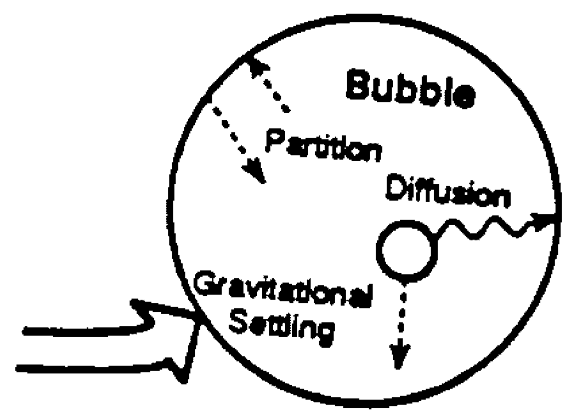




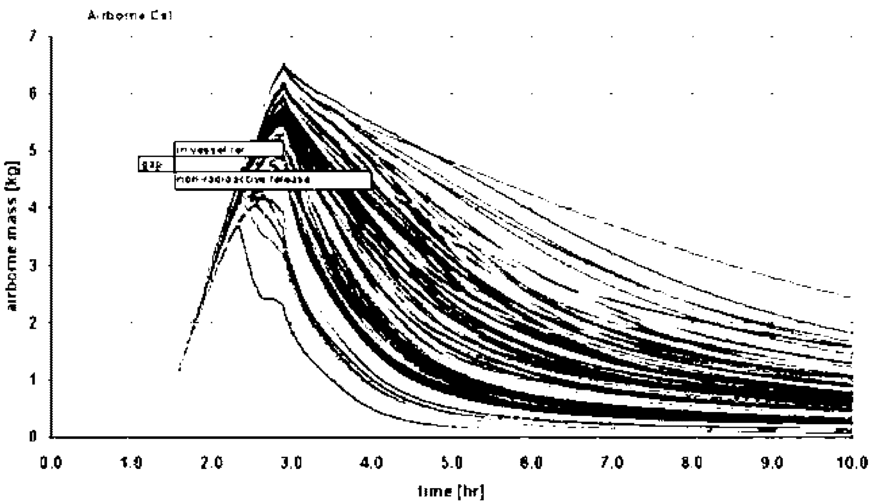
Schematic of Suppression Pool During Scrubbing of Inlet Gases



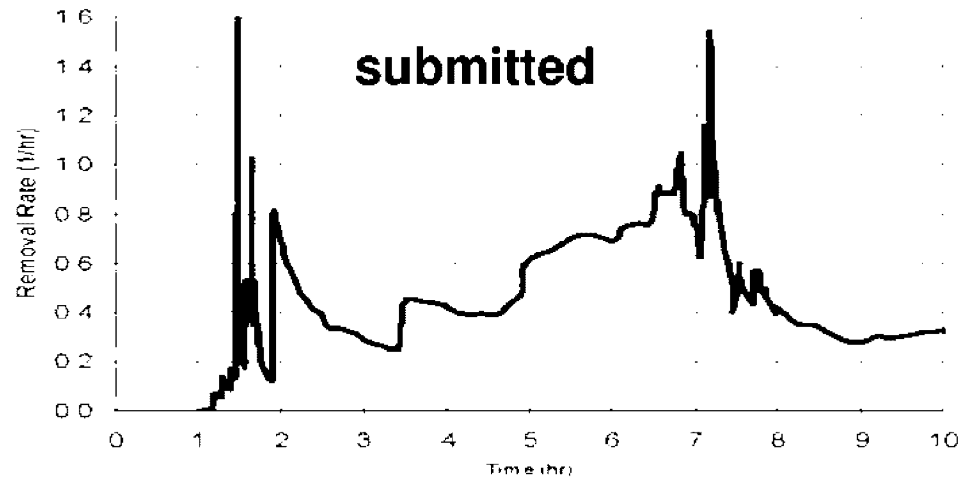
Growth or decay of dispersed bubble or droplet in the steam of continuous phase (liquid or gas)

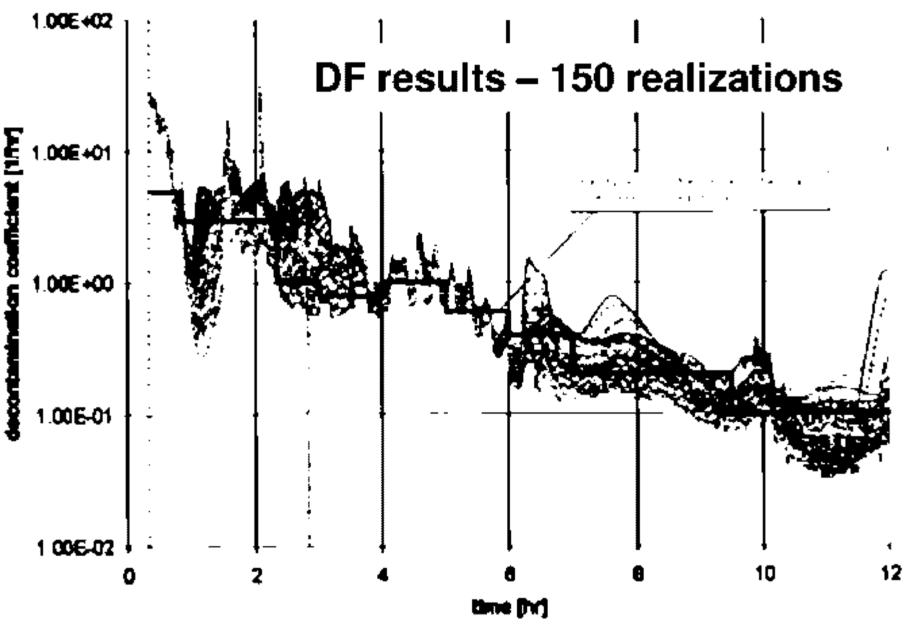


Uncertainty analysis ESSENTIAL



Airborne CsI for all 200 samples of the distribution.





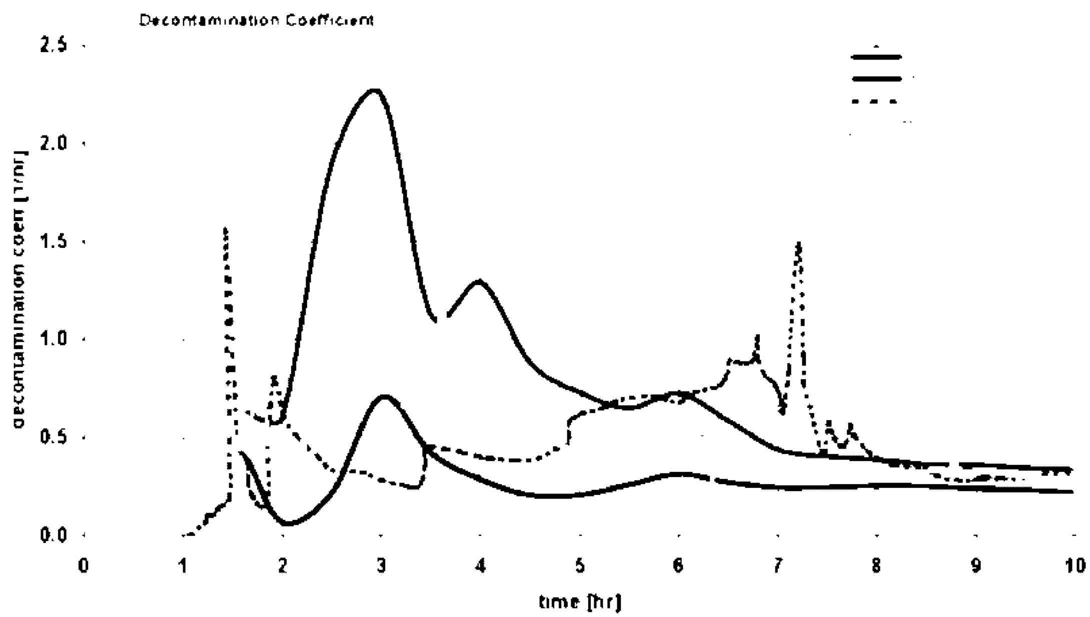
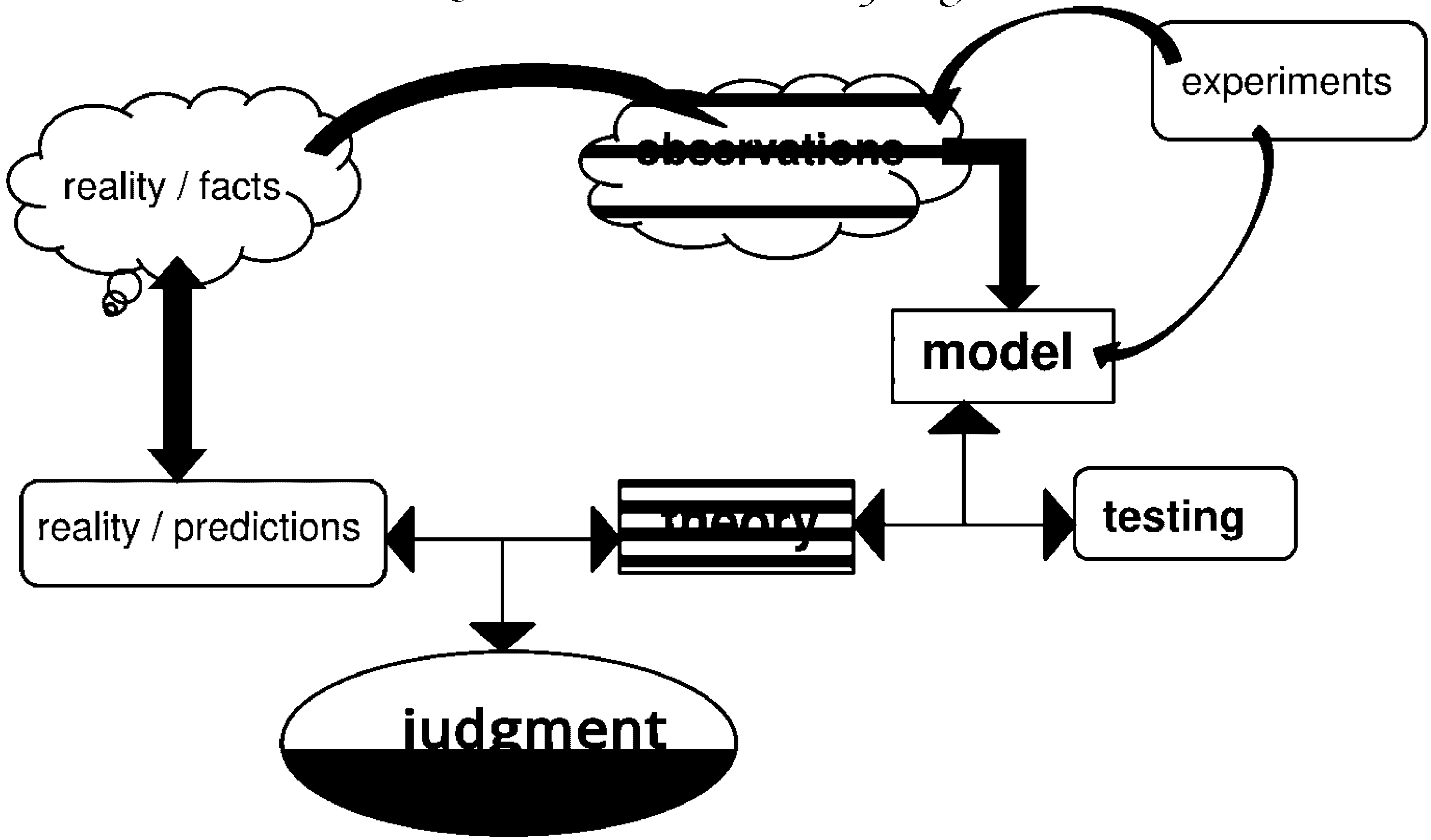


Figure 2 Decontamination coefficient: 5%/95%, mean value and ERI-predicted values.

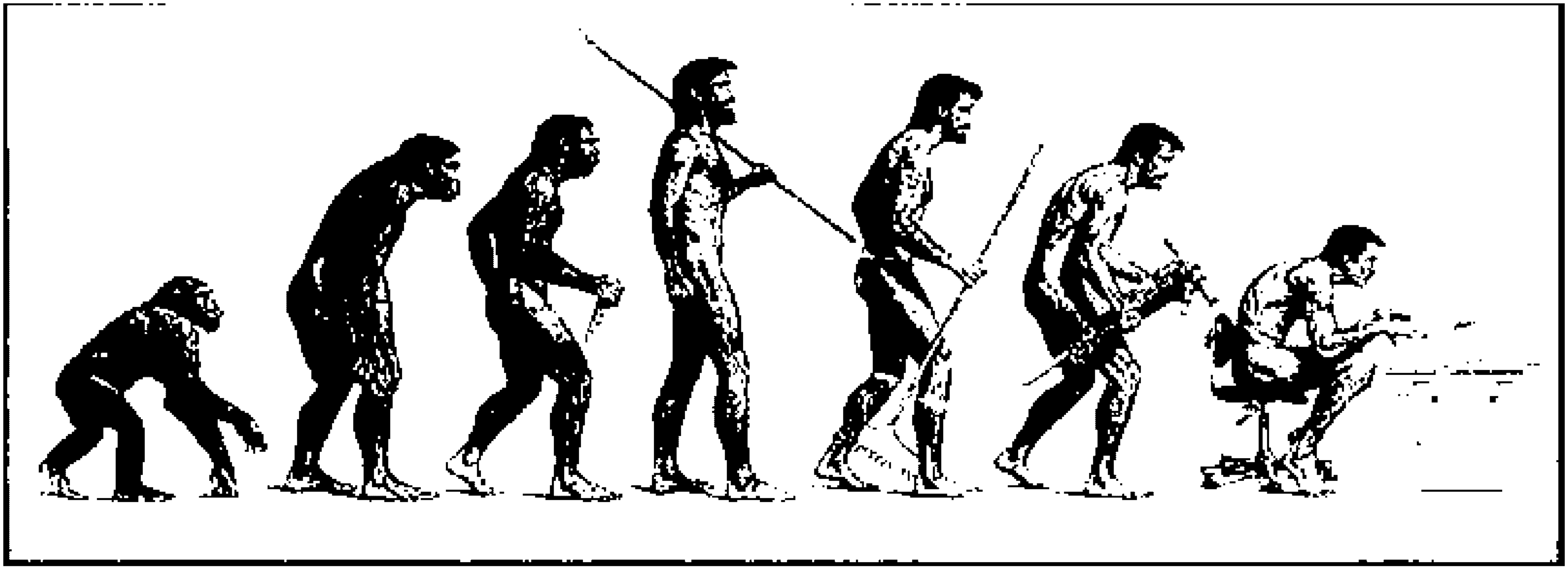
"Engineering is the art of modeling materials we do not wholly understand, into shapes we cannot precisely analyze so as to withstand forces we cannot properly assess, in such a way that the public has no reason to suspect the extend of our ignorance."

Dr A.R. Dykes, British Institution of Structural Engineers, 1976

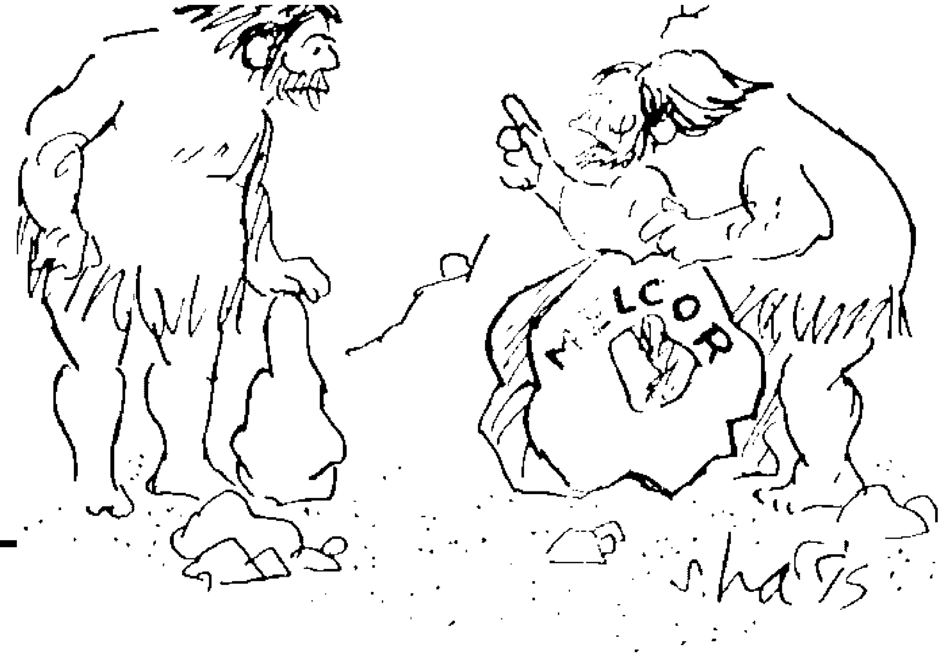
It ALWAYS comes down to judgment…



and HOW we get there is a mystery…



Somewhere, something went terribly wrong



"IT MAY NOT BE A PERFECT WHEEL, BUT IT'S A STATE-OF-THE-ART WHE....."

MELCOR ACCIDENT SIMULATION USING SNAP (MASS)

Corrosion in Nuclear Power Plants

Joel Jenkins
NRO/DE/CIB
June 20, 2012

Faces of Corrosion

Rusty Chain

Faces of Corrosion

Chemical tank with SCC

Source: National Transportation Safety Board
Accident Brief NTSB/HZB-0401

What is Corrosion?

Degradation of a metal due to electrochemical interaction with its environment

Corrosion Mechanisms

- General Corrosion
- Pitting
- Crevice Corrosion
- Intergranular Attack

Corrosion Mechanisms

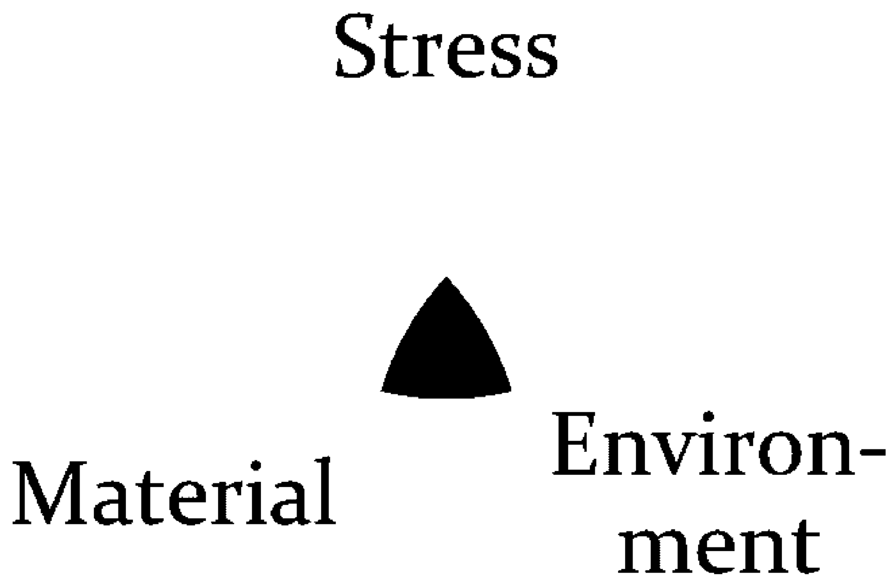
Galvanic Corrosion

Flow Accelerated Corrosion

Stress Corrosion Cracking (SCC) includes but not limited to

- PWSCC (Primary Water SCC)
- IGSCC (Inter-Granular SCC)
- IASCC (Irradiation-Assisted SCC)

SCC Factors



SCC Factors

Problem Material-Environment Combinations

Steel and caustic solutions

Stainless steel and halogen salt solutions

Alloy 600 and reactor primary water

Design Considerations

Is corrosion of the component likely considering material and operating conditions?

If corrosion is likely, at what point is it detrimental to operation?

What methods can be used to detect presence and extent of corrosion?

Corrosion in the Nuclear Power Plant

Boric Acid Corrosion (BAC)

- Responsible for corrosion of bolts and/or vessel head (e.g. Davis Besse 2002)

Flow Accelerated Corrosion (FAC)

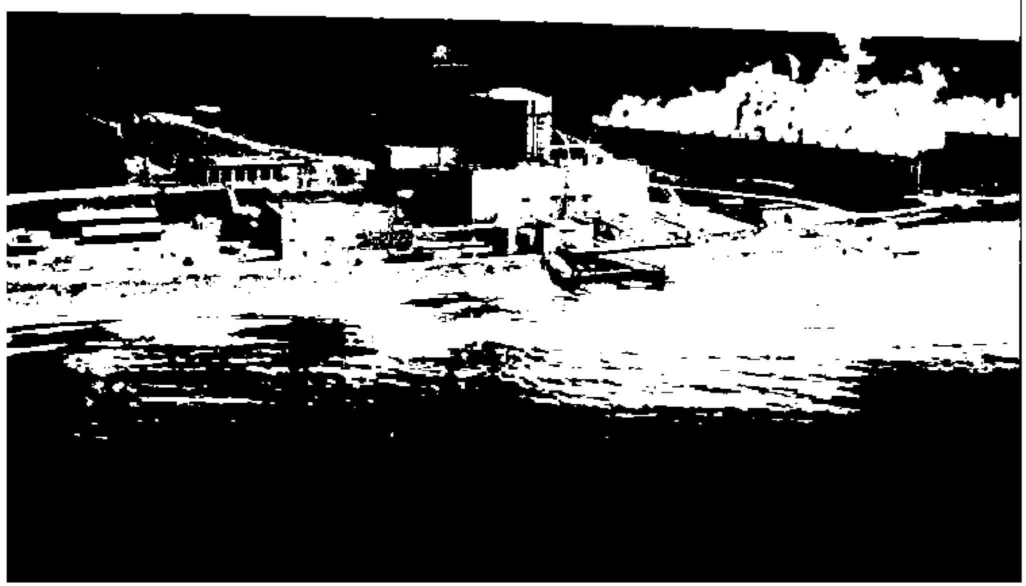
- A synergistic wear/corrosion mechanism responsible for failure of pipe (e.g Surry 1986)

Stress Corrosion Cracking (SCC)

- A common corrosion problem for nickel alloys and stainless steel. Can happen in primary and non-primary systems. Sometimes aggravated by neutron radiation.

Case Study: SCC at Pallisades

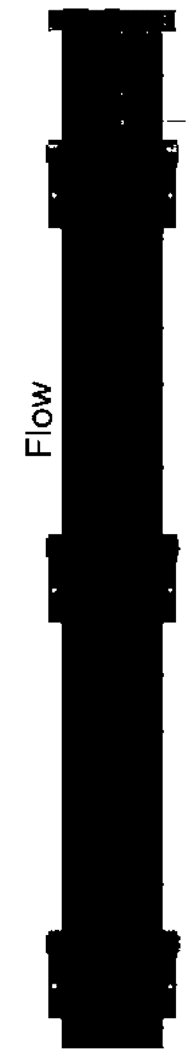
(IGSCC of Service Water Pumps)



SWPs at Pallisades

Service Water Pumps (SWPs)

- Major components: motor, hollow casing, rotating shaft, impeller
- Originally designed with carbon steel shaft couplings
- Redesigned in 2007 to use corrosion resistant stainless steel couplings (Type 416)



Pallisades-2009

Redesigned stainless steel coupling failed in 2009

Cursory evaluation found hardness to be out-of-spec

Failed coupling replaced with like material, and back to operation

Susceptibility of design to further IGSCC not considered

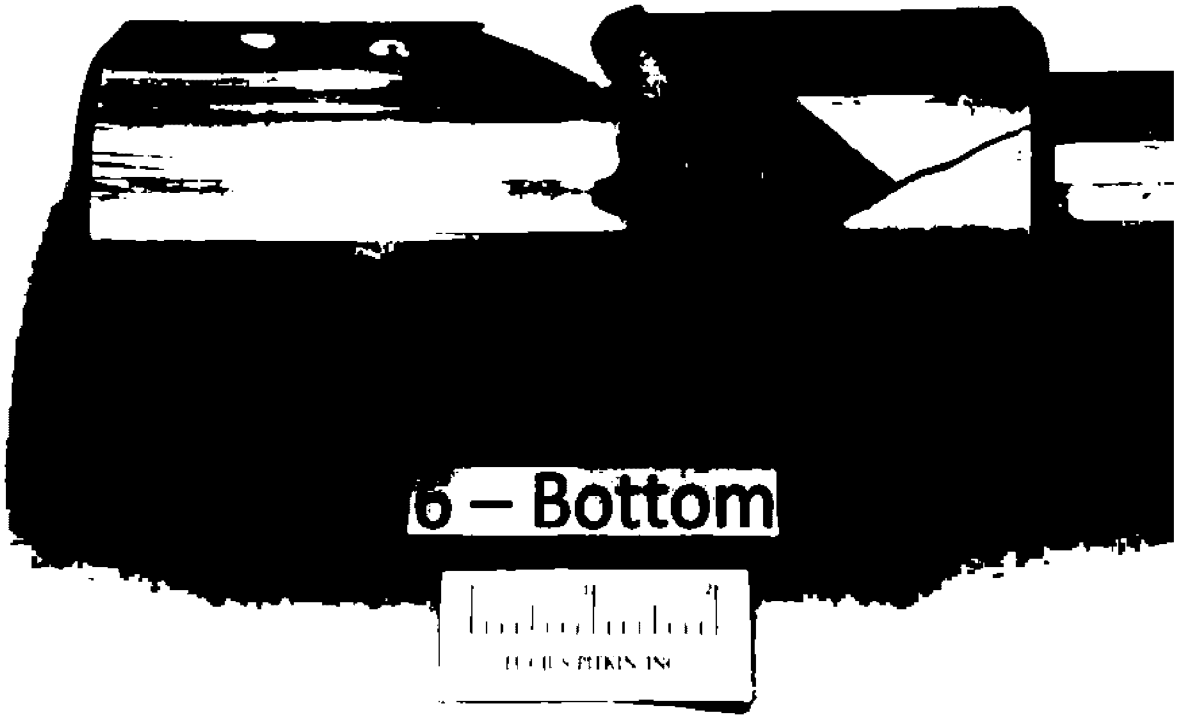
Pallisades-2011

Stainless steel coupling fails during operation, disabling one of the three SWPs

Per tech specs, failure of a single pump activates an LCO action (fix in 72h or shut down)

Utility performs root cause analysis and links failure to IGSCC

Failed Coupling, Pallisades-2011



Source: Entergy Operations, Pallisades (ADAMS ML12006A049)

Failure Analysis, Pallisades-2011

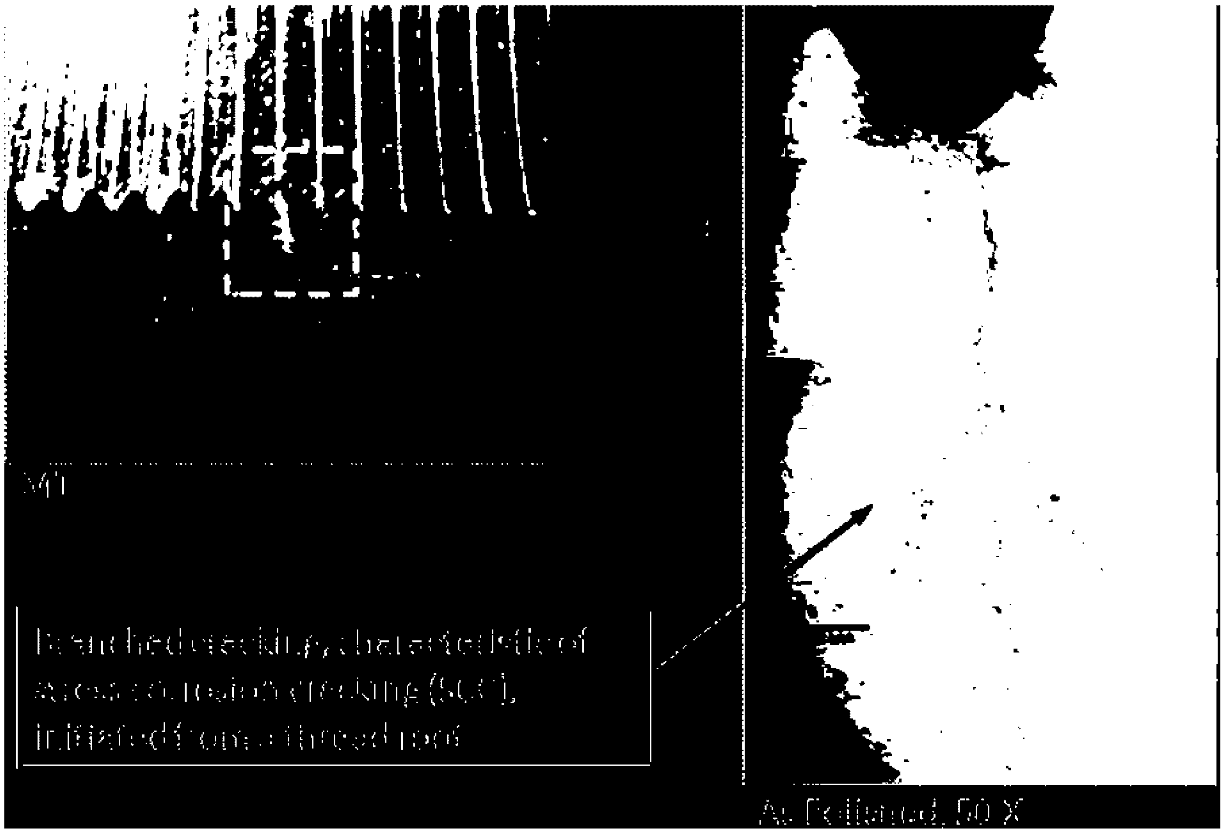
Material was found to be improperly heat treated and thus susceptible to IGSCC

Sufficient tensile stress in shaft assembly to initiate SCC in couplings

Chlorination of lake water and concentration of ions due to wet/dry cycles increased corrosiveness of environment

Failure Analysis, Pallisades-2011

Evaluation of Unbroken Coupling



Source: Entergy Operations, Pallisades (ADAMS ML12006A049)

Solving the Corrosion Problem at Pallisades

How did Pallisades solve the problem?

Was the Pallisades solution the only option?

Are type 410 and 416 stainless steel “bad” materials?

Solving the Corrosion Problem in General

Preventing Problems through Good Design (the best solution!)

Effective Inspection at Proper Intervals

Learning Lessons from Operating Experience

Where to Get More Info

Subject Matter Experts

Training

- NACE General Corrosion Course
- NRC Course E116 (Corrosion and Corrosion Control in LWRs)
- NRC Effects of Corrosion Web-Based (basically a short discussion of Davis Besse)

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Digital I&C
Operating Experience Insights

NRC Office of Research: How Things Fail Seminar Series

Daniel Santos
David Garmon

"The only source of knowledge is experience"
- Albert Einstein

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Purpose and Agenda

- Challenges
- Operating Experience (OpE)
  - Selected Industry Reports
  - Specific Events
- Agency Efforts
  - Domestic
  - International

Digital System OpE Knowledge Management

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Why is Digital System OpE Important?

- Effect on Plant Safety
- Increasing Integration of Plant Systems
  - Operating Plants
  - New Reactors

Learn from Our Experience

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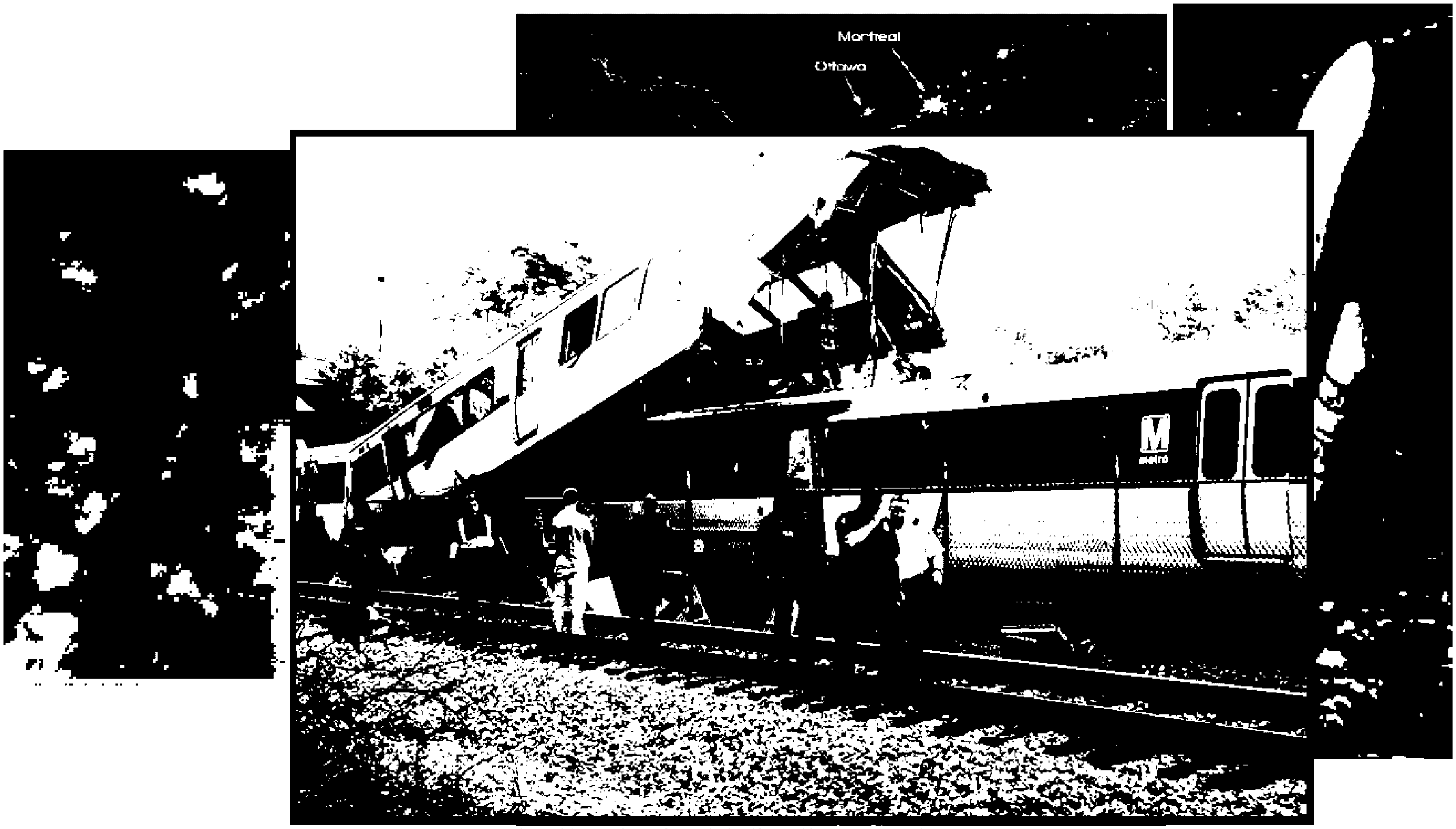
Non-Nuclear Events



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Challenges to OpE Collection

- Small event population
  - Novelty of digital safety systems
- Reporting and analysis quality
  - Improve quality
  - Consider standardization

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Industry Reports

- Electric Power Research Institute (EPRI)
  - Operating Experience Insights on Common-Cause Failures in Digital Instrumentation and Control Systems (2008)
- Institute of Nuclear Power Operations (INPO)
  - Topical Report (TR) 8-63 Software Events (2008)
  - TR 8-64 Microprocessor-Based Digital-Hardware-Related Events (2008)

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EPRI Study: Intro

- Study Objective
  - Identify potential digital I&C related common cause failures
- Approach
  - Primary distinction between safety and non-safety events

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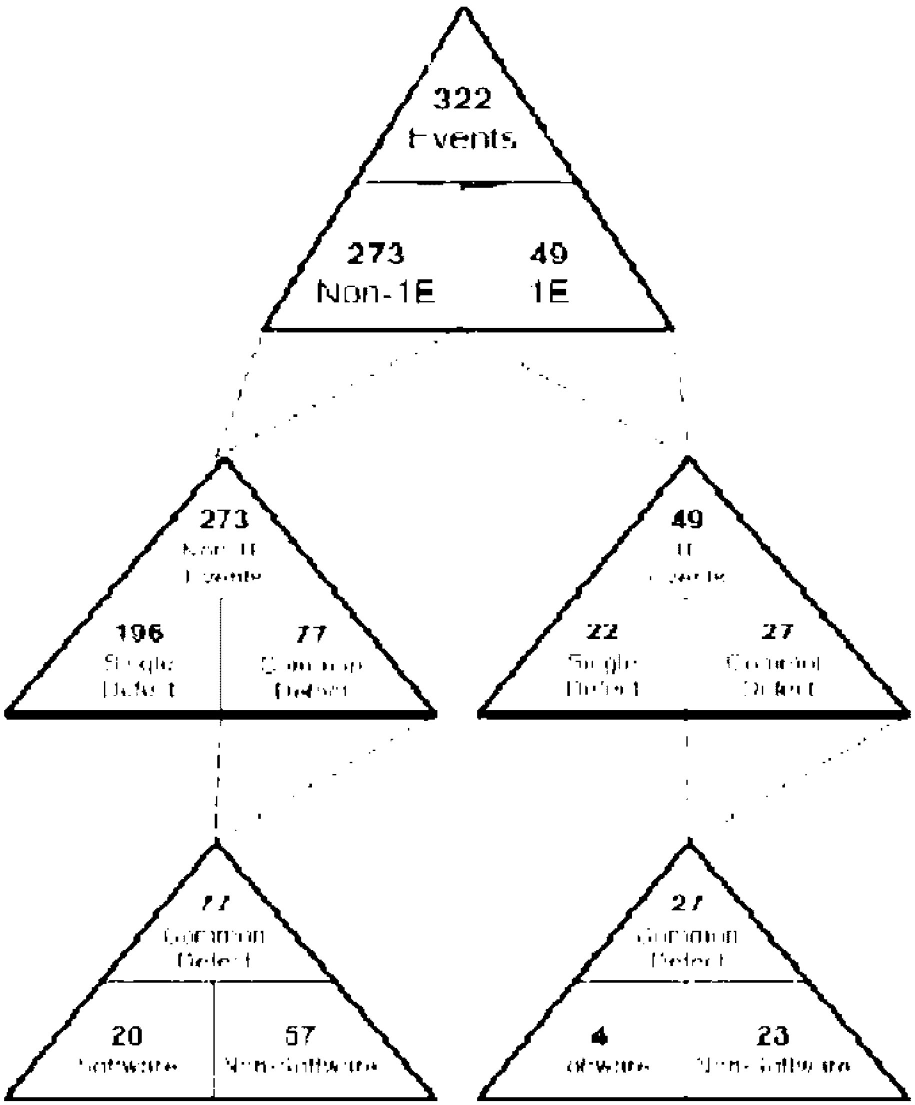
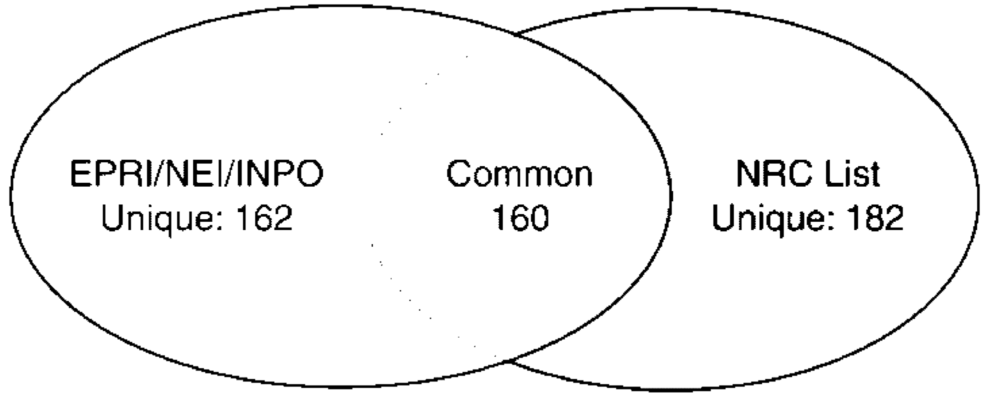
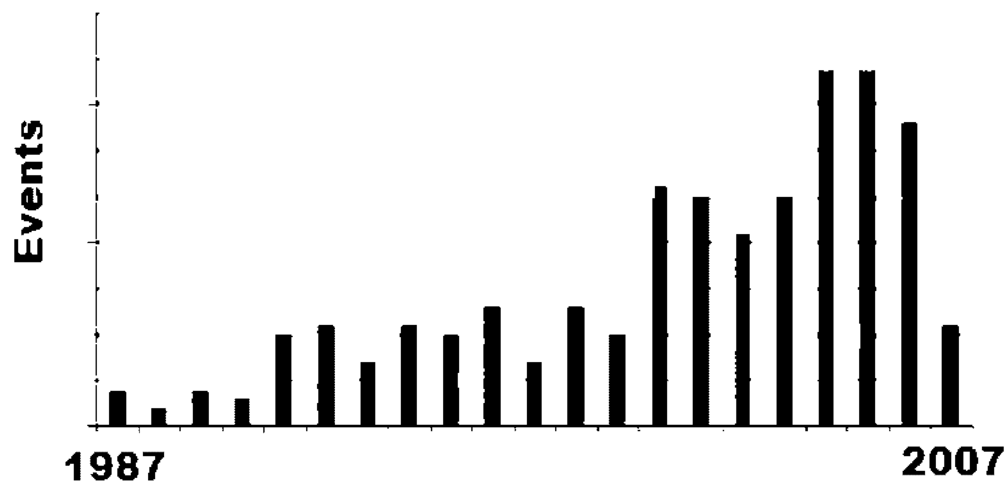
EPRI Study: Data





Study Data Sources



Event Distribution

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EPRI Study: Selected conclusions

- Number of events increases with installations
- Systemic Failures
  - Inadequate ___________
    - Requirements definition
    - Testing programs
    - Vendor oversight
    - Design compatibility, etc
- Software based changes to address non-software problems

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INPO: TR 8-63 and TR 8-64

- Objective
  - Analyze digital hardware/software related events and their impact on power production.

- Approach
  - Significant Events Evaluation and Information Network (SEE-IN)
  - Equipment Performance and Information Exchange System (EPIX)
  - Plant Events Database (PED)
  - Licensee Event Reports, 10 CFR 50.73 (LER)
  - International events (World Association of Nuclear Events, WANO)

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TR 8-64: Hardware Data

- 55 Domestic events (2003-2007)
  - SCRAMS (24), power reductions (20), misc impact on production (11)
  - Most Prevalent System
    - Balance of Plant systems (37)
  - Most Prevalent Causes
    - Circuit cards, power supplies failures and component "design deficiencies" (39)

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TR 8-64: Results

- Inadequate system designs
  - Inadequate design compatibility
  - Insufficient annunciation of internal failures
- Unique attributes of digital systems
  - Sensitivity to EMI and signal noise
- Inadequate preventive maintenance
  - Aging and environmental effects not monitored

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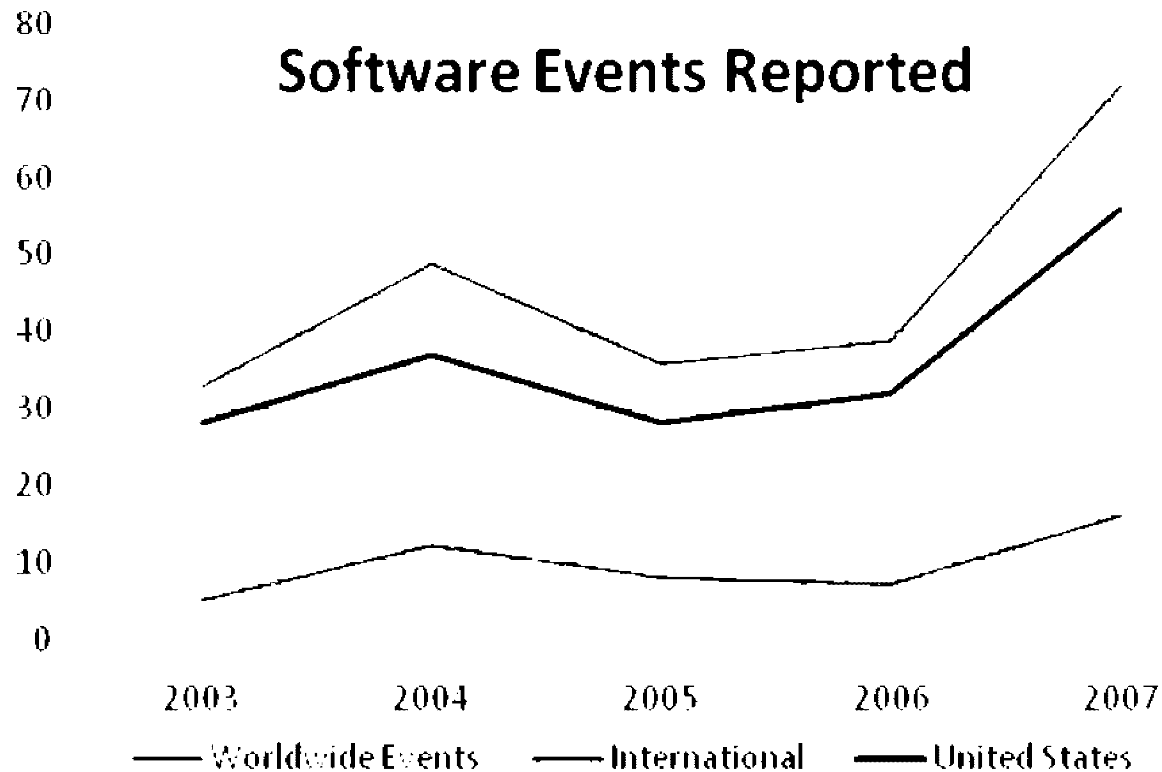
TR 8-63: Software Events



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TR 8-63: Key Observations

- Complexity of software requirements definition
- Organic skill set
- V&V issues are common
- Consideration of effects of software upgrades
- Vendor control issues

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TR 8-63: Software Event Contributors

Software Event Causes

Inadequate Verification and Validation

Vendor Involvement

Design Deficiency

Inadequate Configuration Control

Programming Errors

Incompatible Design Modification

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TR 8-63: Results

- Verification and Validation (V&V)
- Software Design
- Software Modifications/Upgrades/ Configuration Control

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Recap of Studies

- Number of events follow increases in installations
- Design issues (software and hardware)
- Verification and Validation
- Configuration control programs
- Software used as workaround for hardware failures

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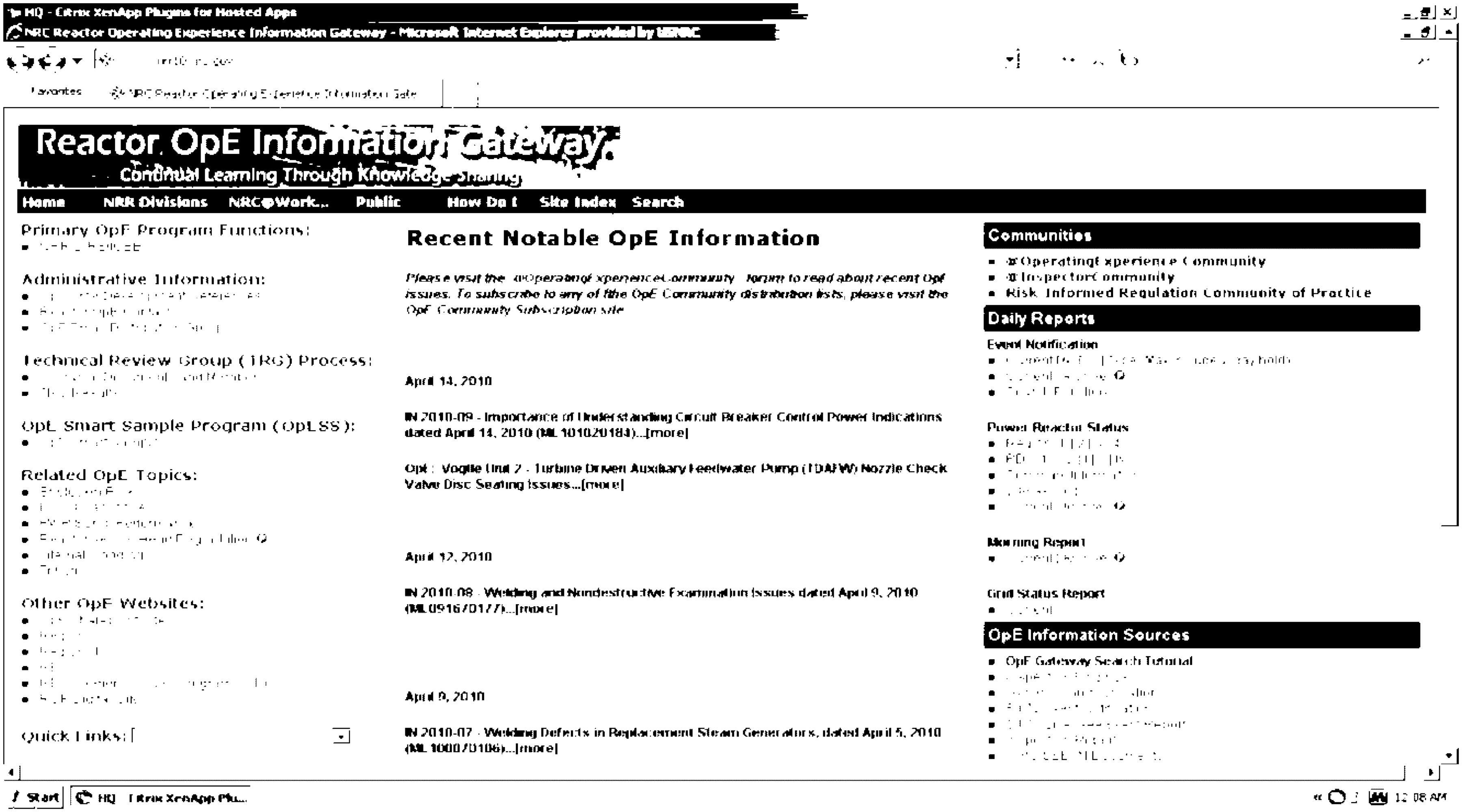
NRR Reactor OpE Program



http://nrr10.nrc.gov/ope-info-gateway/index.html

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Plant Events

- **2005** **Palo Verde 1 – Reactor Trip due to Incorrect Operation of DFWCS**
- **2007** **Perry – Reactor Scram due to Failure of DFWCS Power Supplies**
- **2009** **Columbia – Reactor Scram with Complications**

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Human Performance (2005, Palo Verde 1)

- Operator "not comfortable" with DFWCS shifted from manual to automatic feed control and overfed the S/G resulting in reactor trip
- Operator consideration as part of implementation program (V&V, training, qualifications program etc.)
- Refs:
  - IFR 2008-06; SIT Report ML080280499

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Hardware (2007, Perry)

- Reactor trip due to loss of feedwater
  - Complications associated with level control
- Insufficient reliability of power supplies compounded by insufficient error annunciation
- Refs
  - IFR 2008-06; SIT Report ML080280499

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Inadequate Software V&V (2009, Columbia)

- Fault on electrical bus results in reactor trip
- During generator load reject a digital electro-hydraulic control system failure results in generator bypass valves being held open
- Following the trip a feedwater control system failure results in low suction pressure trips of each feedpump
- Insufficient design review and testing
- Refs
  - IFR 2010-04 (and attachments); SIT Report ML093280158

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Recap of Individual Event Review

- Weakness in V&V programs
  - Not including operators
  - Not fully testing system
- Preventive maintenance
  - May not be a priority
- Quality of hardware and software
  - Remains at the foundation of system performance and plant safety.

Learning from Our Experience Depends on Quality of Reporting

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Agency Efforts

- Office of Research
  - NRC DI&C System Research Plan FY 2010-2014
  - COMPSIS Cooperation
- NRR
  - Update NEI 01-01
  - ISG-6
  - IN 2010-10
- NRO
  - Design Acceptance Criteria/Inspection, Tests, Analysis and Acceptance Criteria Procedures

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Takeaways

- Systemic failures
- Latent Defects
  - No event too small to be considered
- Improve event documentation
- Need a well established KM effort

Operating Experience Supports Technical Basis for Regulations

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Questions?

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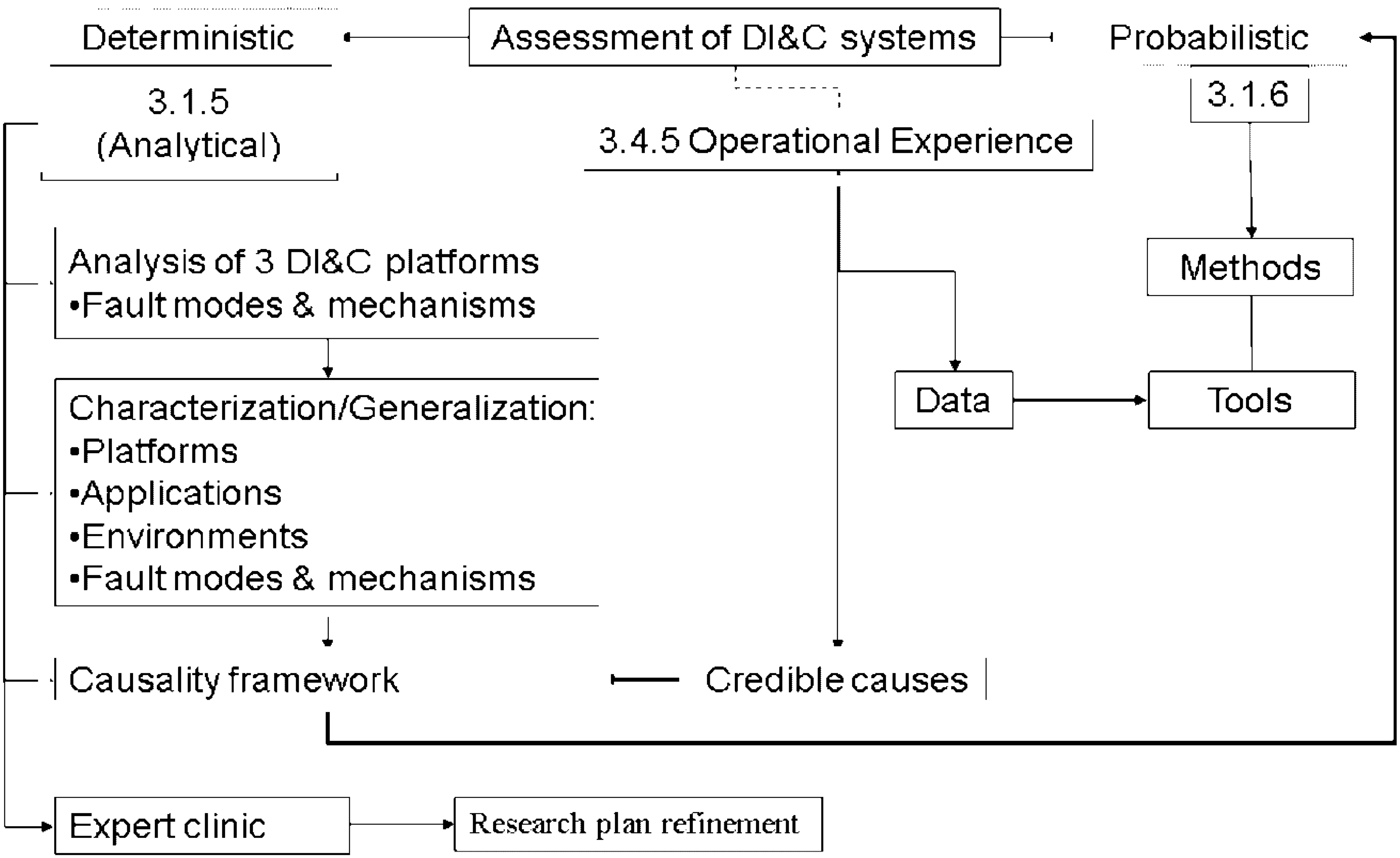
Digital System Research Plan



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Preventive Maintenance (2007, North Anna 2)

- Spurious safety injection due to protection system circuit card failure
  - Signal could not be immediately reset
  - PORVs lifted
  - Rupture of relief tank rupture discs
- Running circuit cards to failure vice program to replace based on age.
- Refs
  - IFR 2007-030; SIT Report ML072410359; INPO SEN268

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Manas Chakravorty
Sunwoo Park

Structural Engineering Branch 2

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Outline

- Basics of the earthquake
- Recorded ground motions
- Seismic impact to the plant
- NRC inspection findings
- Licensee short-term and long-term actions

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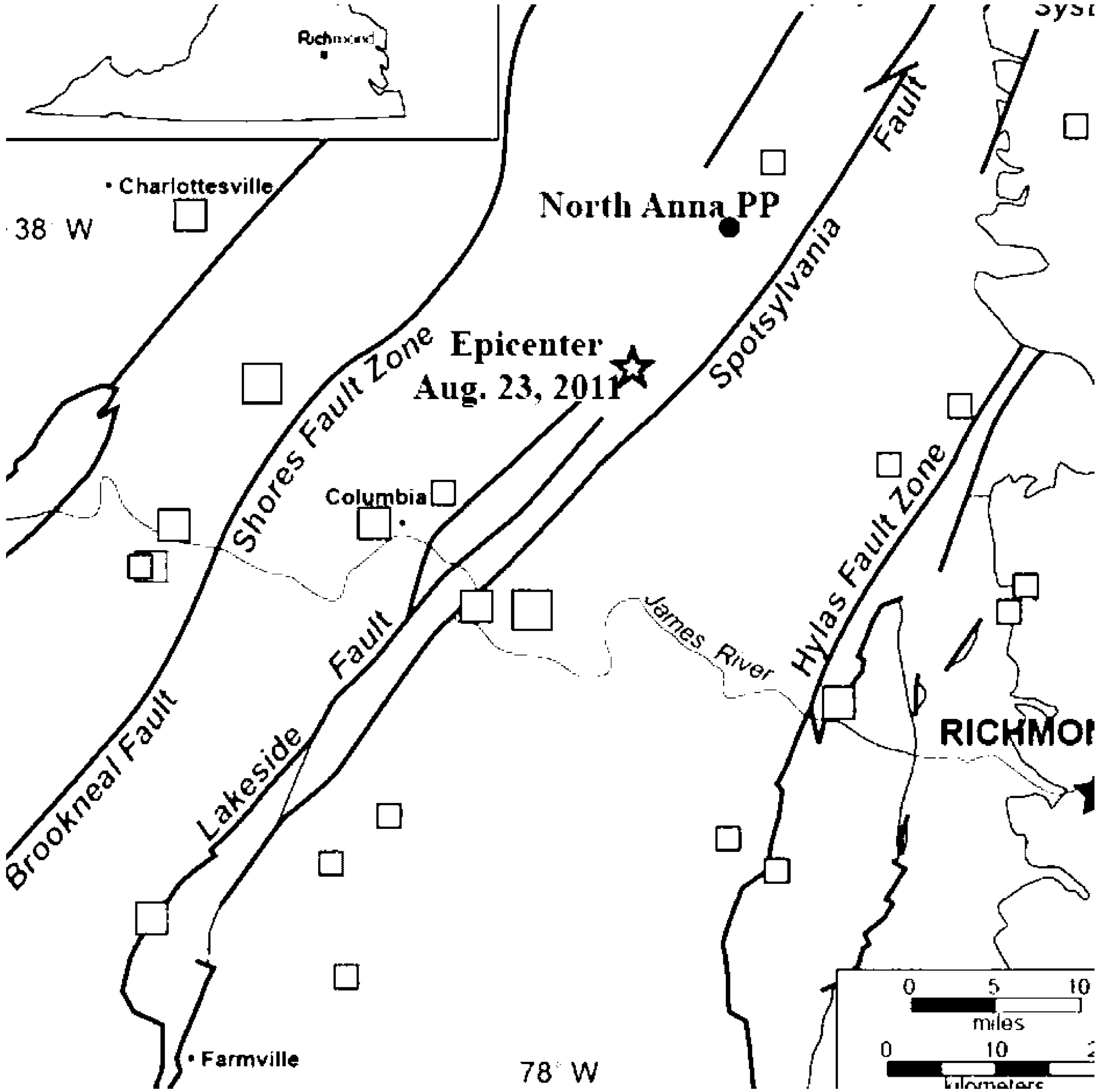
The 2011 Mineral, VA Earthquake

- Magnitude : 5.8
- Depth: 6 km
- 11 miles SE from NAPS
- Fault types: reverse
- Largest recorded quake east of the Rocky Mountains since 1897
- Widely-felt earthquake in U.S/Canada (Alabama to Canada and the East Coast to Illinois), according to USGS

Main Shock and Aftershocks



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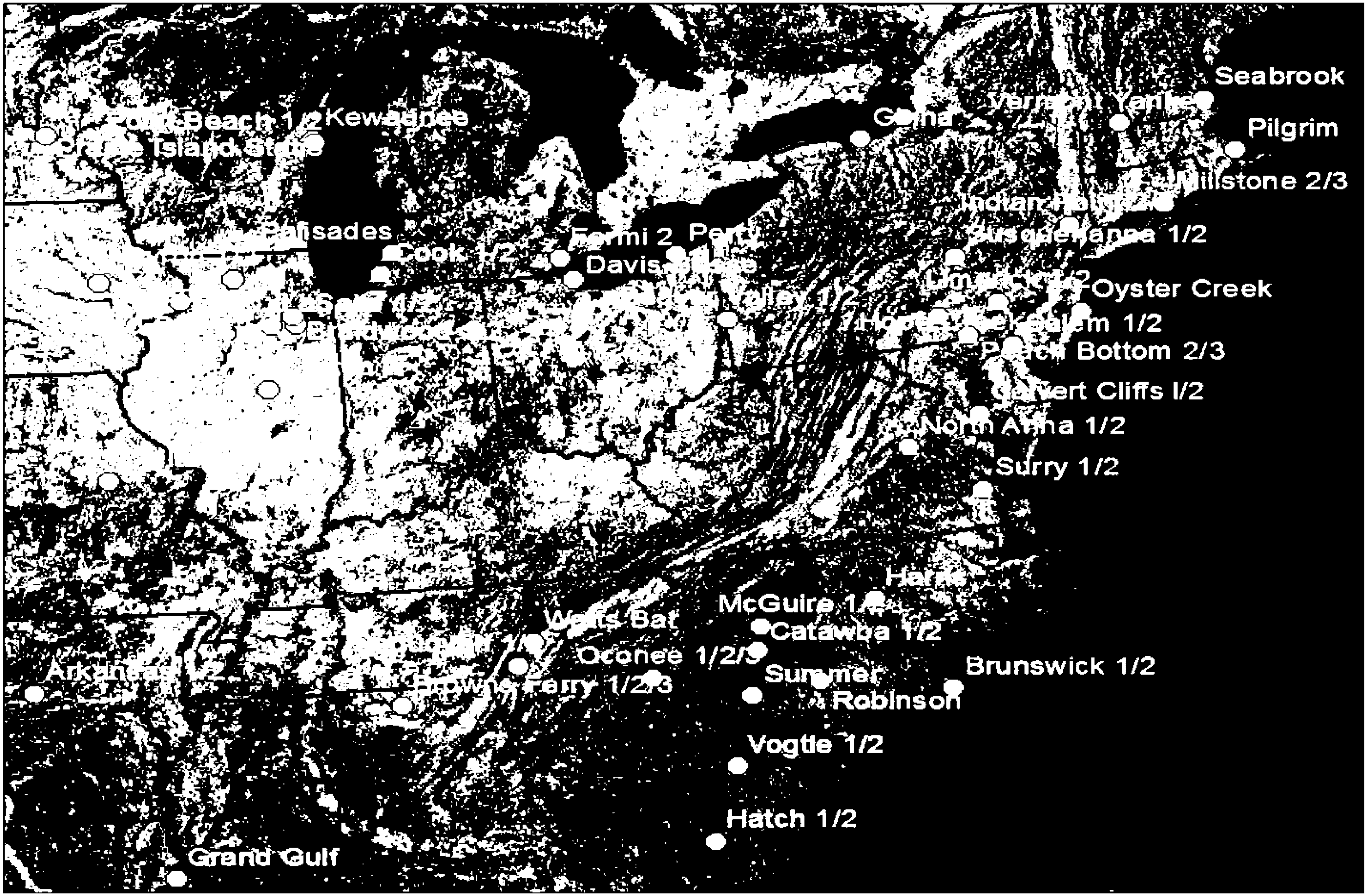
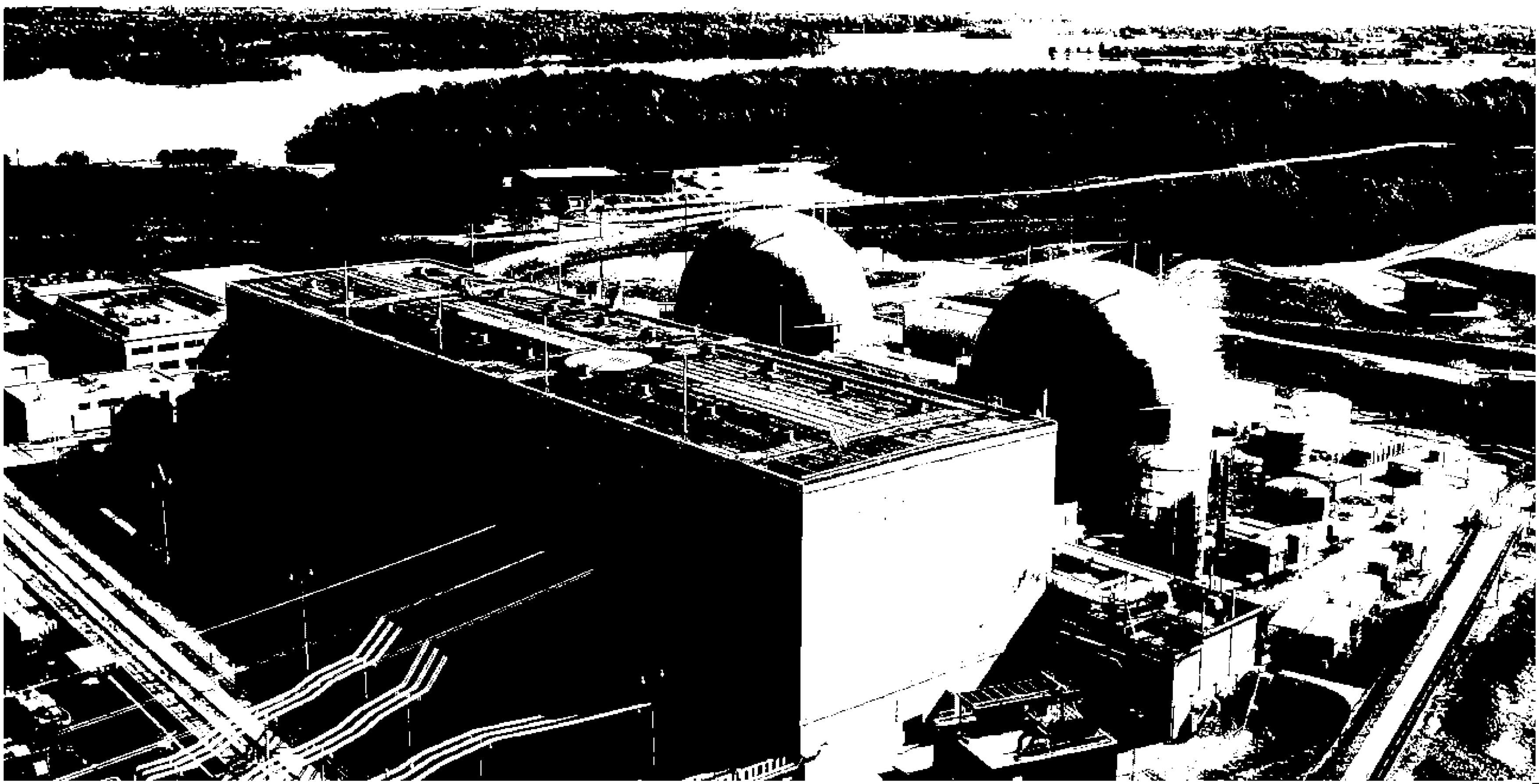
Impact on Local Community

- More than 900 homes damaged
- Pre-Civil War house in the county was damaged (foundations, chimneys)
- 2 of 6 public schools in the county suffered structural damage
- Estimated costs for repairs in Louisa County exceed $18 million, including damage to public buildings and roadways

Ten Nuclear Power Plants Declared "Unusual Event"



North Anna Nuclear Power Station



Impact to North Anna Nuclear Power Plant

- Epicenter was ~18km (11mi) from the plant
- PGA estimated at the site is ~0.26g
  - 13:51:00 – Earthquake occurs with both units at 100% power
  - 13:51:11 - Reactor Trip Breakers open for both reactors on negative flux rate trip. All control rods inserted
  - 13:51:12 – Loss of offsite power due to sudden pressure trips on offsite power transformers
  - 13:51:20 – All four diesel generators and the SBO DG start

Impact to North Anna Nuclear Power Plant (cont'd)

- 14:03:00 – An Alert was declared based on judgment because LOOP prevented the seismic panel from reporting the earthquake
- 14:40:00 – 2H EDG was tripped due to coolant leak. Subsequently, SBO DG was aligned to 2H bus
- 22:58:00 Offsite power was restored
- On August 30, NRC Implemented AIT to assess the total LOOP and dual unit trip, failure of 2H EDG, and other equipment issues following the seismic event

Seismic Design at North Anna Nuclear Power Plant

The North Anna Plant has two Safe Shutdown Earthquake ground motions (SSE),

- for structures, systems, and components (SSCs) located on top of <u>rock</u>, it anchored at a peak horizontal ground acceleration (PGA) of 0.12 g
- for SSCs located on top of <u>soil</u>, it anchored at a PGA of 0.18 g

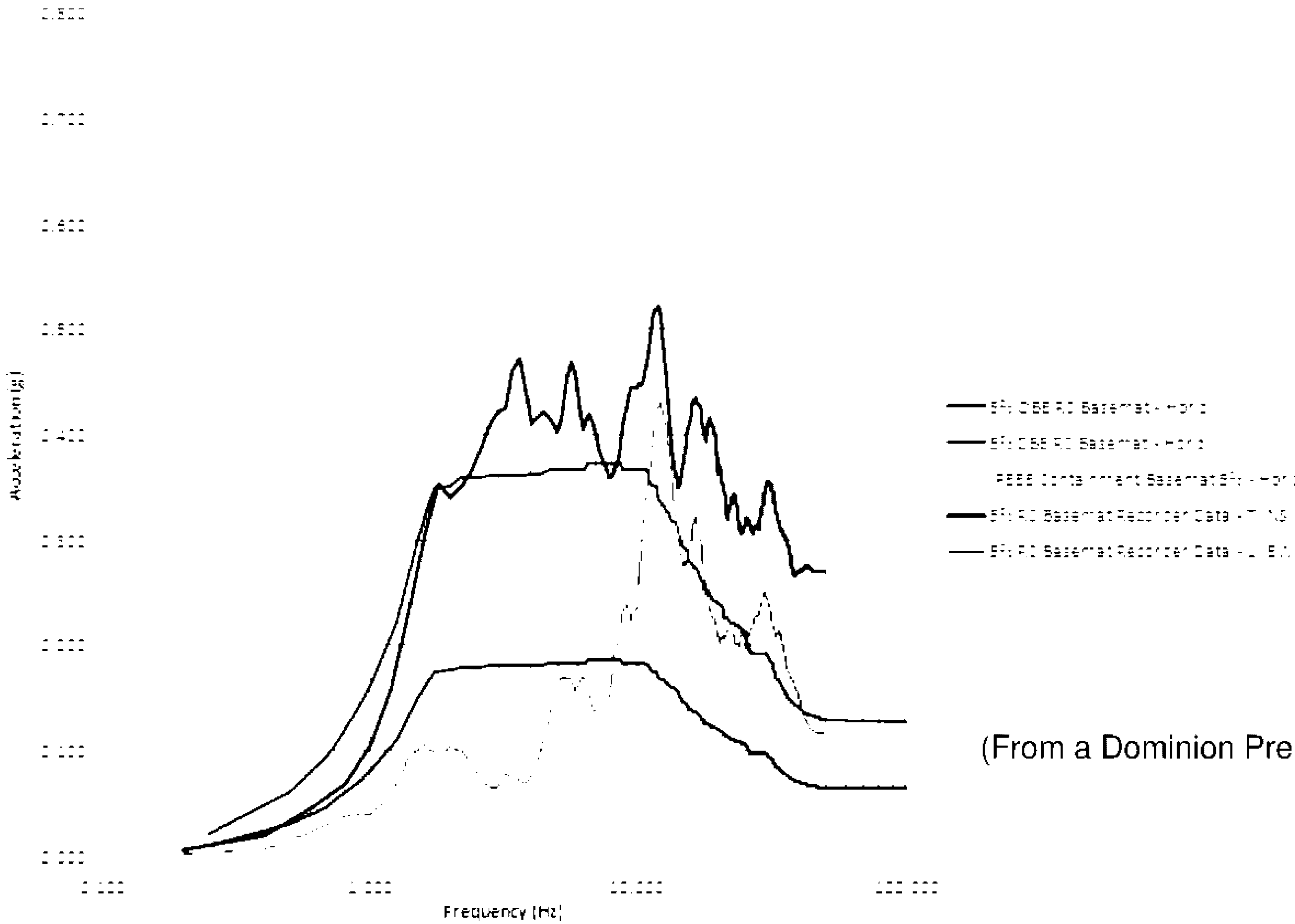
OBE and DBE (SSE) – Peak Ground Accelerations

OBE	0.06	0.04	0.09	0.06
DBE	0.12	0.08	0.18	0.12

(From a Dominion Presentation)

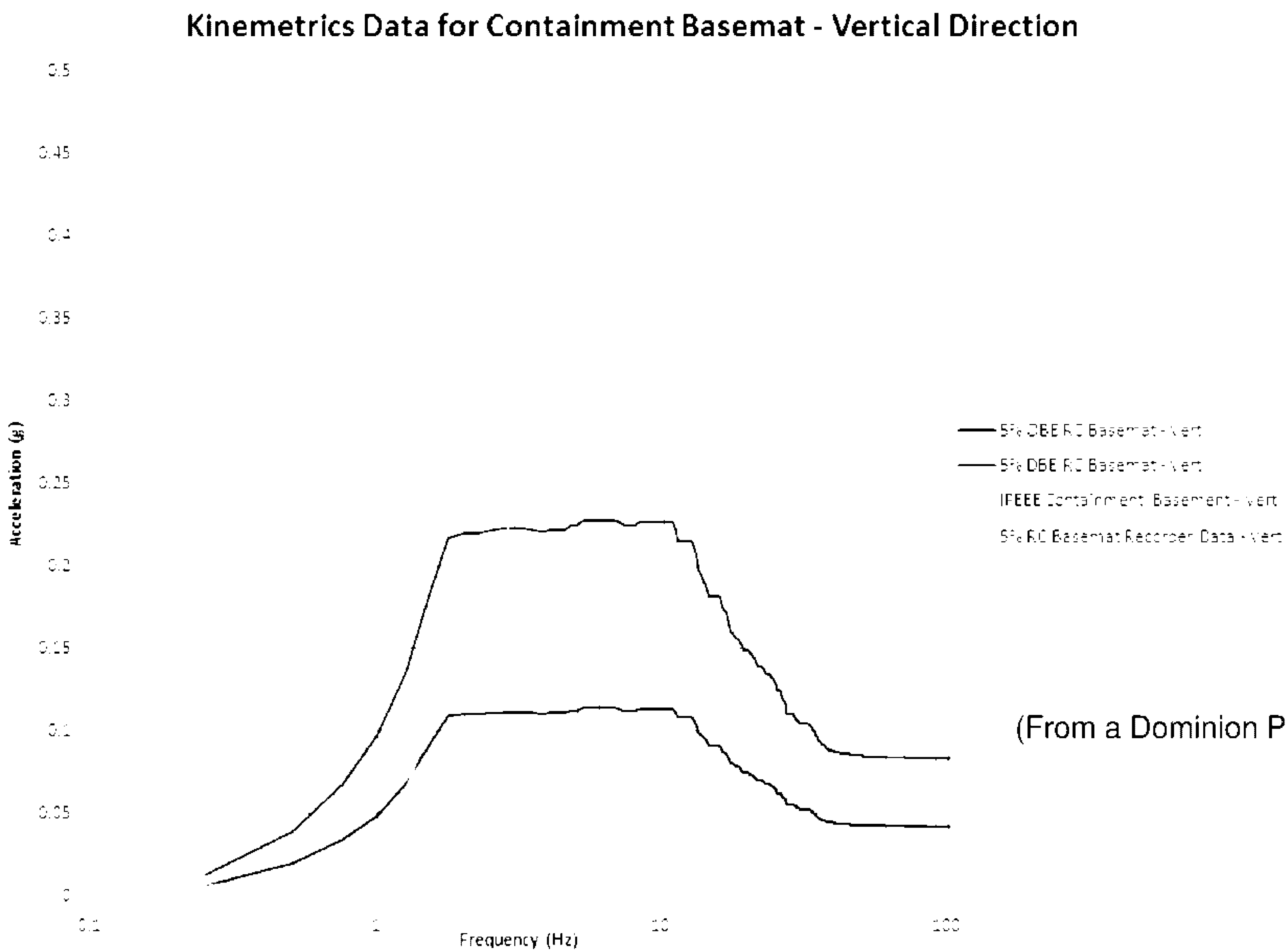
Response Spectra Comparison
(Horizontal, Rock Site)

Kinemetrics Data for Containment Basemat - Horizontal Direction



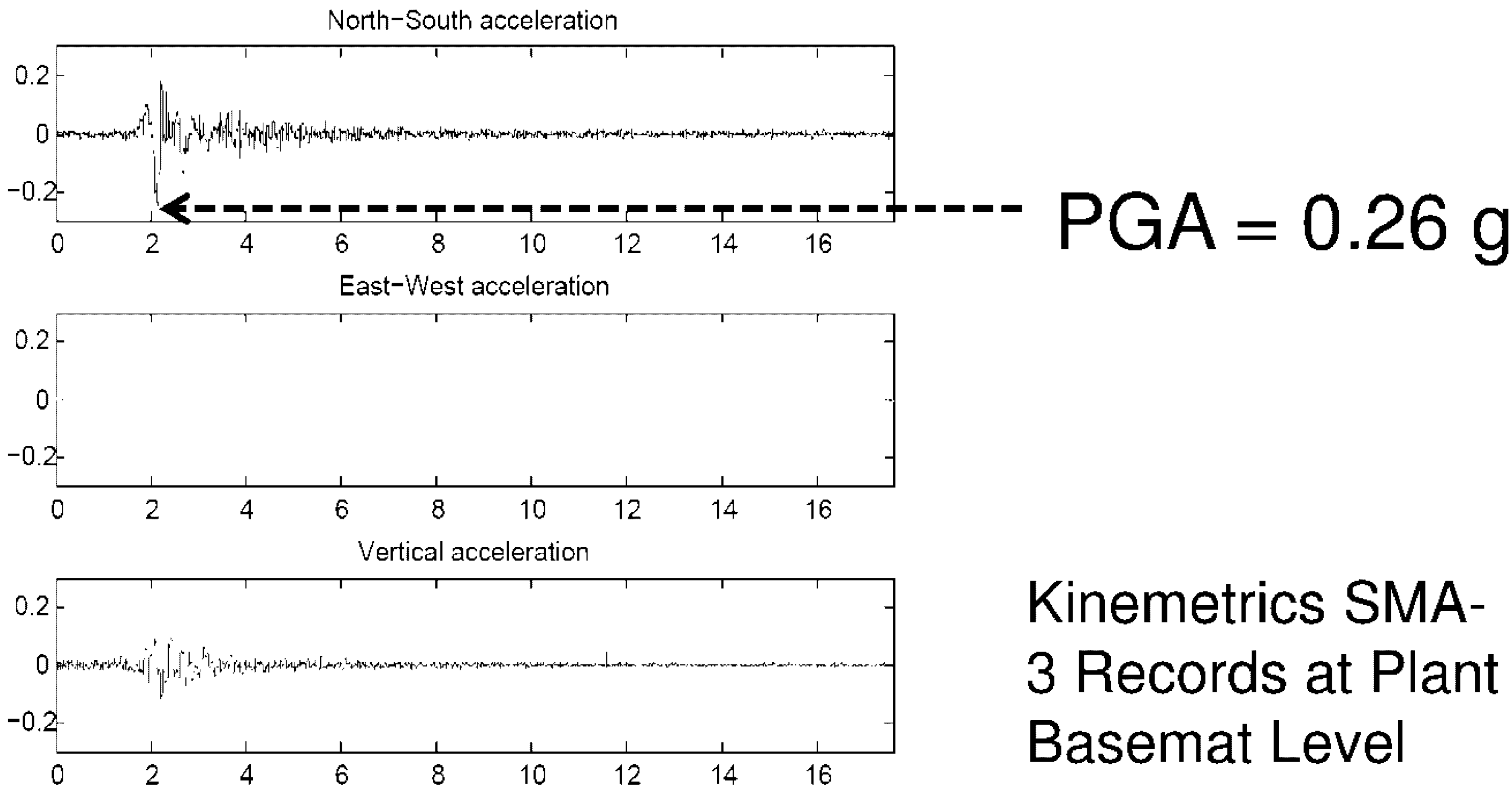
(From a Dominion Presentation)

Response Spectra Comparison
(Vertical, Rock Site)



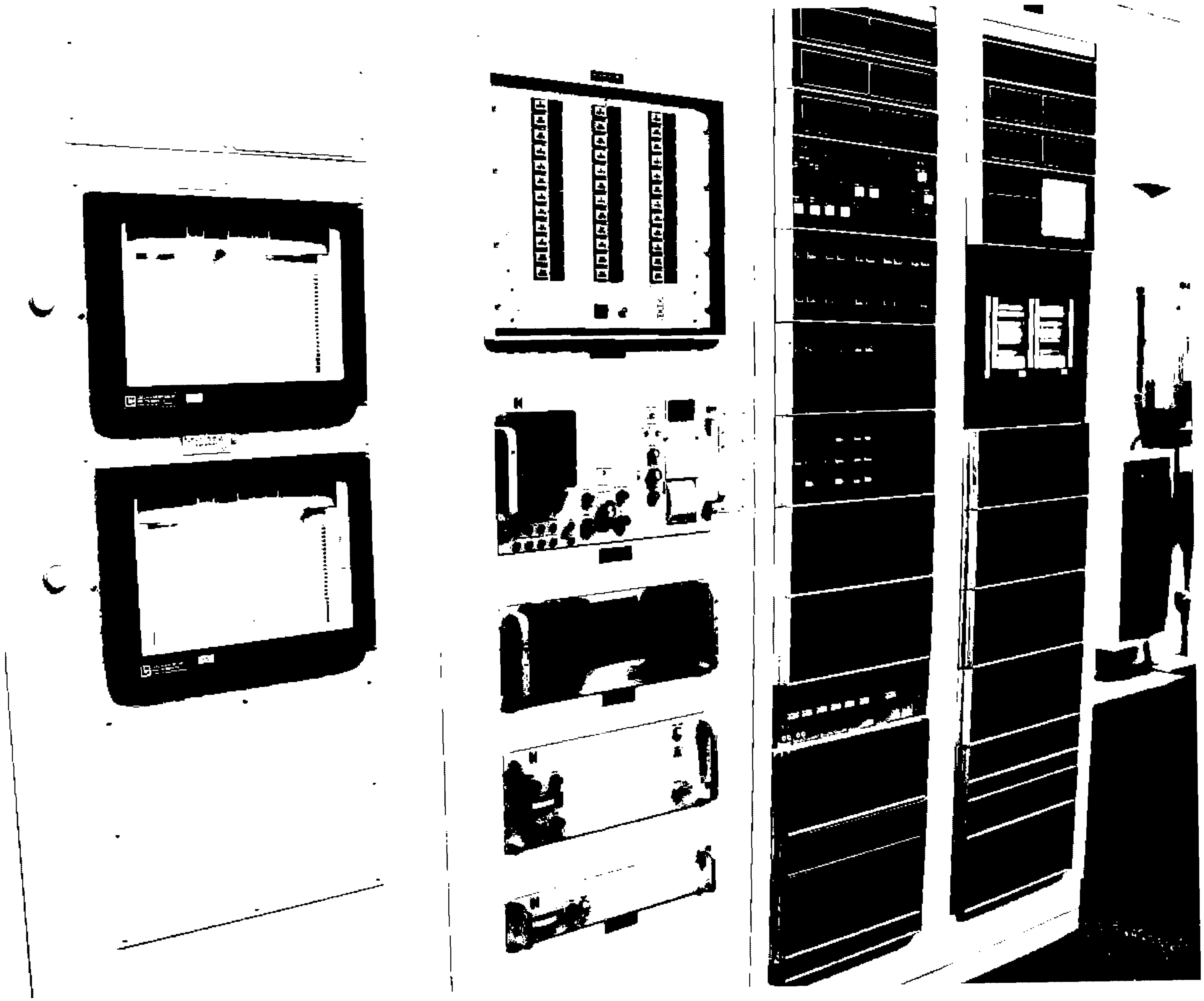
(From a Dominion Presentation)

Recorded Motion at North Anna Plant from the Mineral, Virginia Earthquake (M5.8)

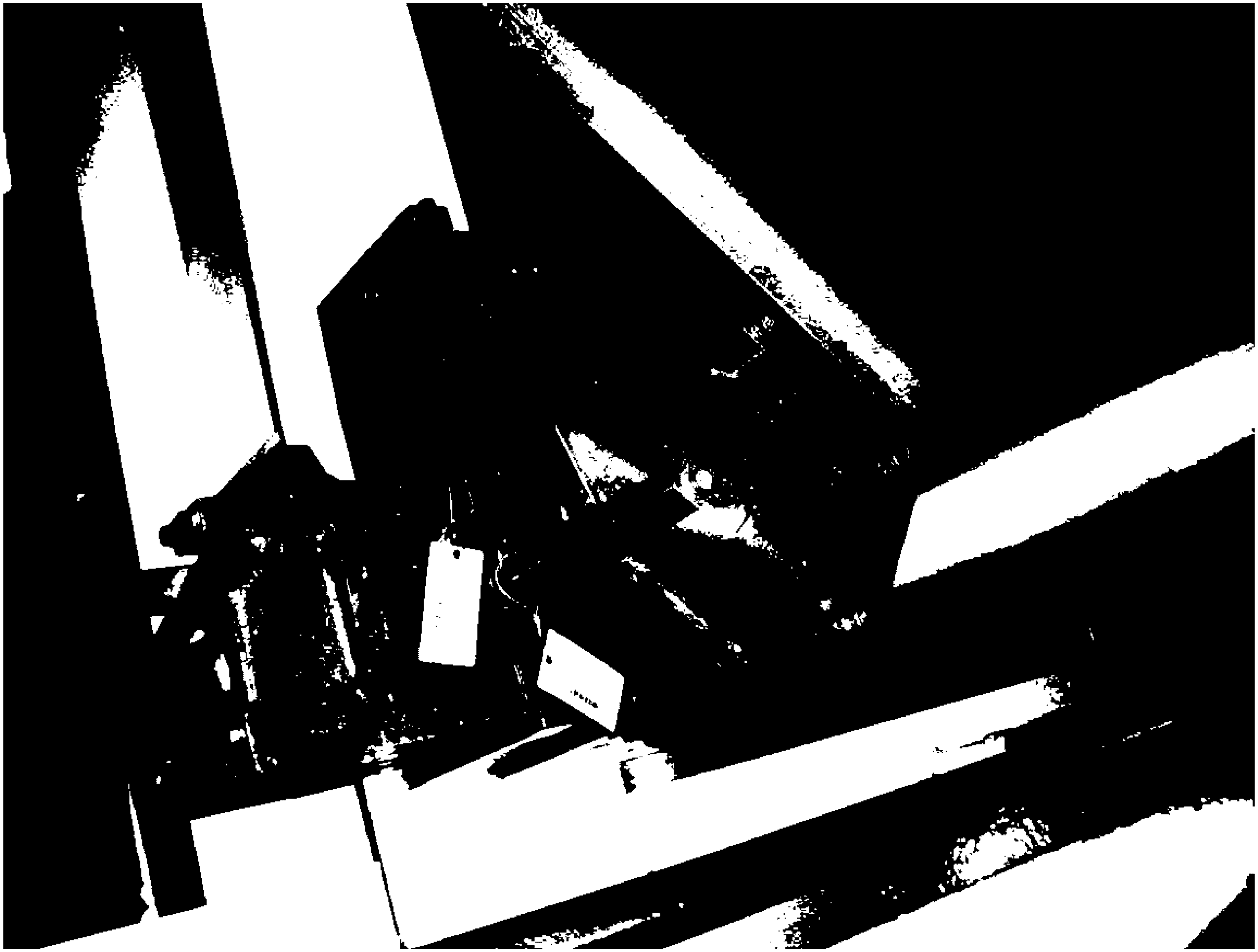


Kinemetrics SMA-3 Records at Plant Basemat Level

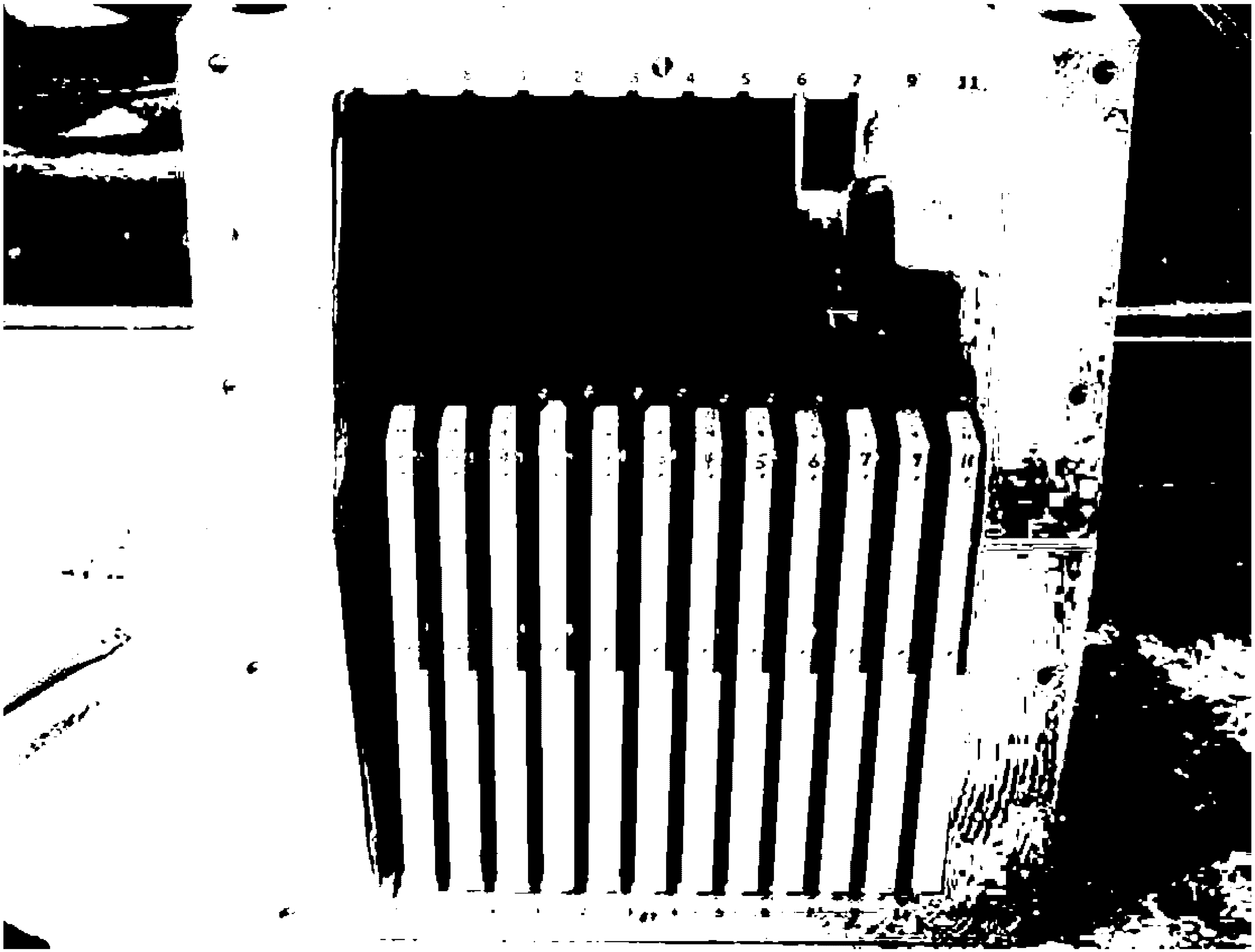
Main Control Room
Seismic Instrumentation Panel



Kinemetrics Triaxial Accelerometers

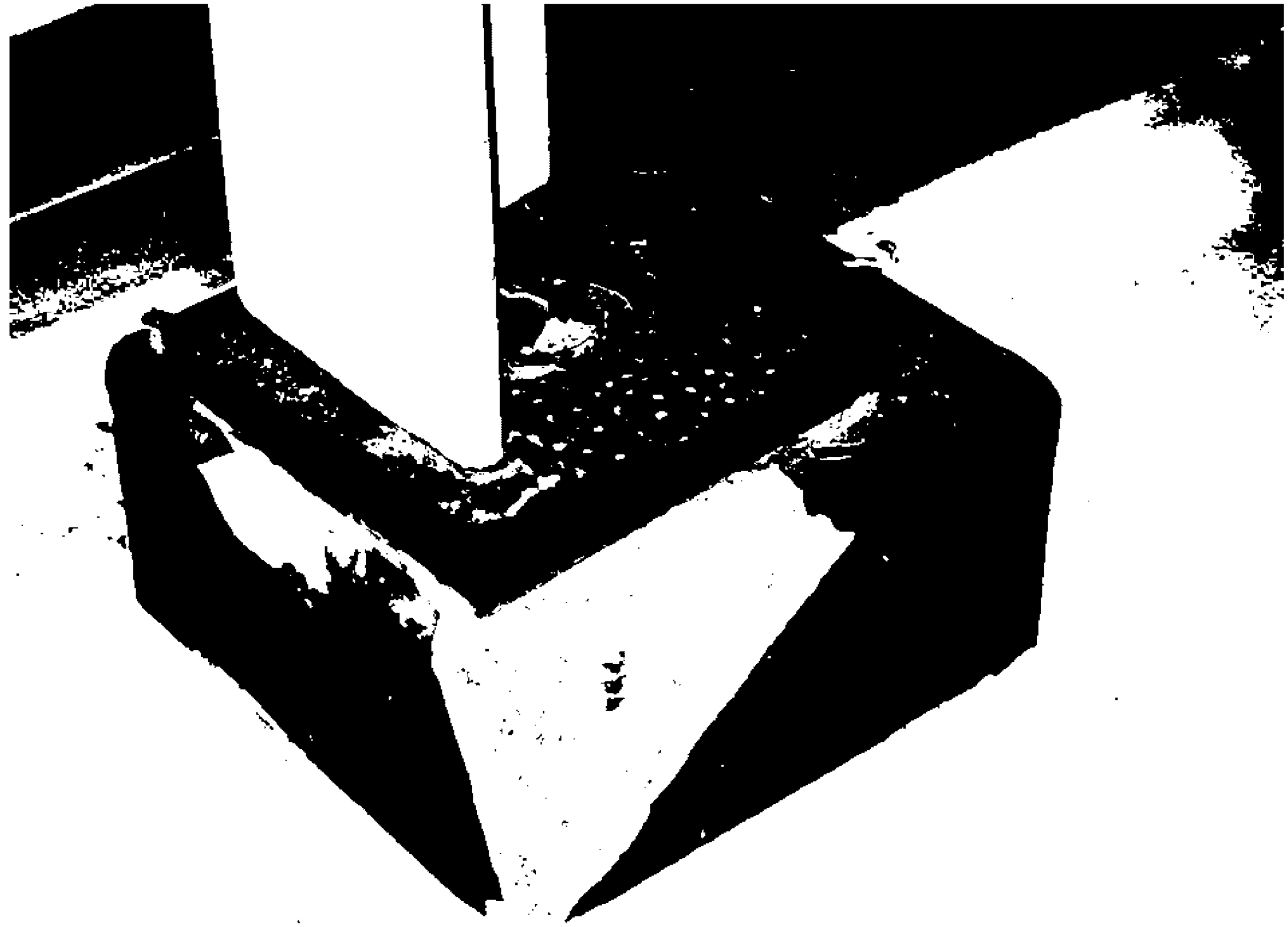


Engdahl Scratch Plates
(Response Spectrum Recorder)



U2 Turbine Building

Powdex Demineralizer Tanks Base Pedestal (non-safety related)



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Turbine Building Hallway

Crack In
Unreinforced
Non–Safety
Related
Block
Wall

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Unit 1 Containment

Surface Hairline Crack In Interior Containment Wall

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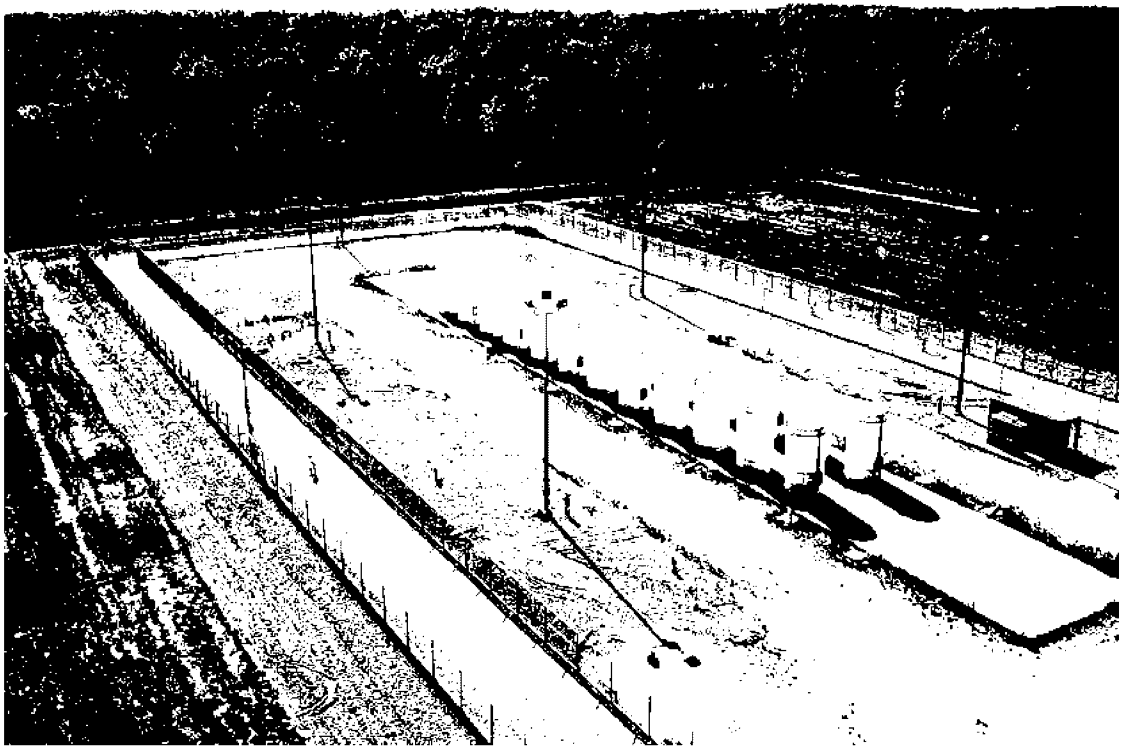
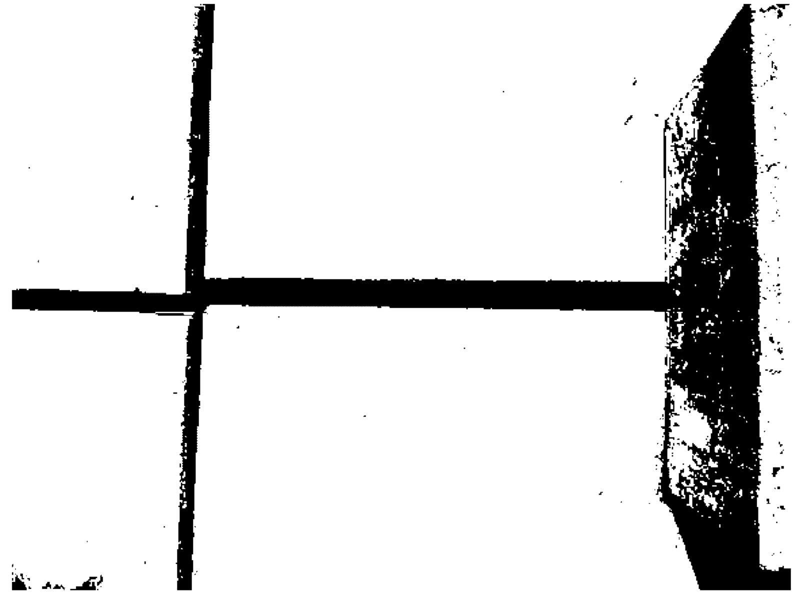
ISFSI - Dry Cask Storage Pad #1 (TN-32 Units)



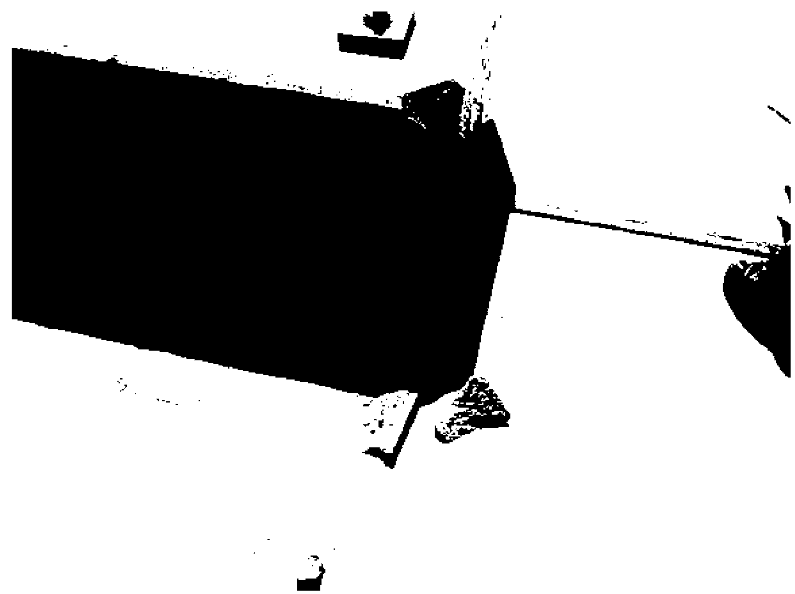
All radiology and temperatures normal

25 of 27 TN-32 vertical casks moved between 1 and 4 ½ inches

ISFSI - Dry Cask Storage Pad #2 (NUHOMS HD System)



NuHoms horizontal modules had small gaps and corners cracked



Augmented Inspection Findings

- Operators responded properly
- Ground motion exceeded licensing design basis
- No significant plant damage
- Safety systems functioned properly
- Some equipment issues were revealed (seismic monitoring equipment performance, failure of 2H EDG, etc)
- The event did not adversely impact the health and safety

Short-Term Actions

- ✓ Installed Temporary Free Field Seismic Monitor
- ✓ Installed Qualified UPS to Seismic Monitoring Panel in Main Control Room
- ✓ Revised Abnormal Operating Procedure
- ✓ Complete Start-Up Surveillances

(From a Dominion Presentation)

- NRC performed readiness restart inspections from 10/5/11 – 11/7/11.
- NRC determined licensee performed adequate inspections, walkdowns and testing to ensure that SSCs were not adversely affected by the earthquake.
- NRC approved restart on 11/11/11.

Long-Term Actions

- Install permanent free-field seismic monitoring instrumentation
- Permanently re-power seismic monitoring panel in the main control room
- Re-evaluate safe shutdown equipment (components with identified lower margins)
- Perform seismic analysis of recorded event consistent with EPRI guidance
- Maintain seismic margins in future modifications
- Revise the North Anna Safety Analysis Report
- Coordinate update of seismic design and licensing basis with GI-199 resolution effort

Summary

- Significant beyond DBE occurred
- RG 1.167 and EPRI NP-6695 were used by licensee and staff
- No significant damage to SSCs necessary for operation
- NRC staff are reviewing lessons learned

(More information at: http://www.nrc.gov/about-nrc/emerg-preparedness/virginia-quake-info.html)

Thank you!

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NUREG-0800
U.S. NUCLEAR REGULATORY COMMISSION
Standard Review Section 3.10

SEISMIC AND DYNAMIC QUALIFICATION OF MECHANICAL AND ELECTRICAL EQUIPMENT

Regulation

1. 10 CFR Part 50, Appendix A, General Design Criterion 1, "Quality Standards and Records."
2. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."
3. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Dynamic Effects Design Bases."
4. 10 CFR Part 50, Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants."
5. 10 CFR Part 52, "Licenses, Certification, and Approval for Nuclear Power Plants."
6. 10 CFR Part 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants."

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Guidance Documents

1. SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," April 2, 1993; Staff Requirements Memorandum 93-087 issued on July 21, 1993.
2. NRC Regulatory Guide 1.100, Revision 3, "Seismic Qualification of Electric and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants."
3. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 3.10.
4. Interim Staff Guidance COL/DC-ISG-1, "Interim Staff Guidance on Seismic Issues Associated with High Frequency Ground Motion in Design Certification and Combined License Applications.
5. NRC Regulatory Guide 1.60, Revision 1, "Design Response Spectra for Seismic Design of Nuclear Power Plants."
6. NRC Regulatory Guide 1.206, " Combined License Applications for Nuclear Power Plants (LWR Edition)."

Industry Standards

1. IEEE Std 344-1987, "IEEE Recommended Practice for Seismic Qualification of Class IE Equipment for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers.

- IEEE Std. 344-2004, "IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations."
- IEEE Std. 323-2003, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations."

3. ASME QME-1-2007, "Qualification of Active Mechanical Equipment Used in Nuclear Power Plants."

ISG (Internal Staff Guideline)

- DC/COL-ISG-1, Internal Staff Guidance on Seismic Issues of High Frequency Ground Motion.

AREAS OF REVIEW

- Seismic and dynamic qualification criteria
- Methods and procedures for qualifying electrical equipment, instrumentation, and mechanical components
- Methods and procedures for qualifying supports of electrical equipment, instrumentation, and mechanical components
- Documentation
- COL Action Items

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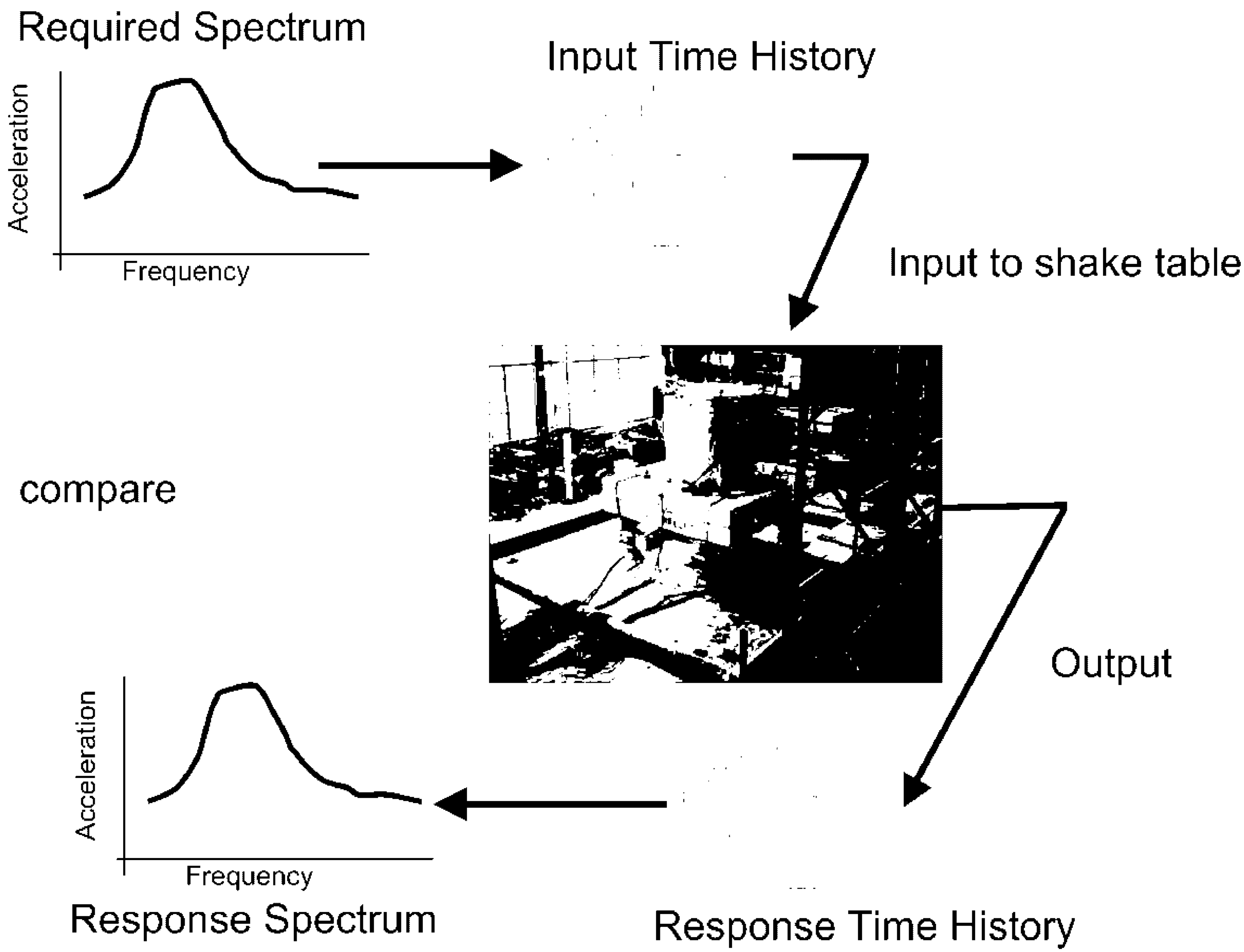
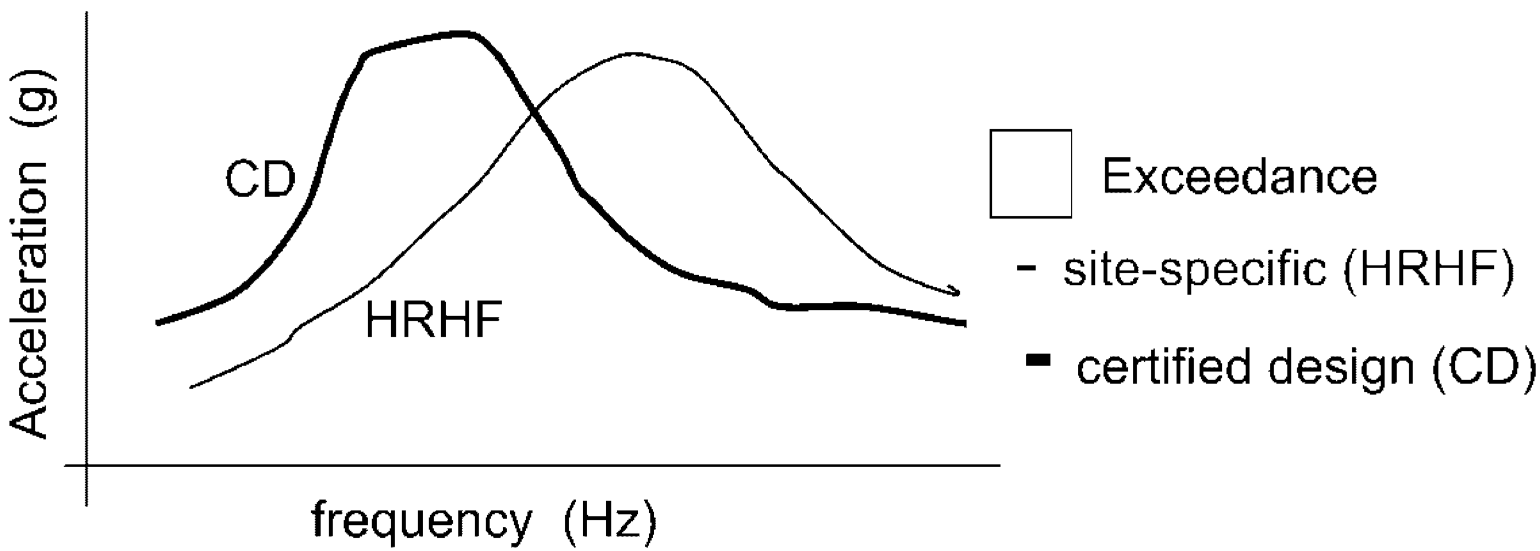
Information Reviewed

- Deciding factors for choosing between tests or analyses.
- Considerations in defining the seismic and other relevant dynamic load input motions.
- Demonstration of adequacy of the qualification program
- Methods and Procedures used to ensure structural integrity and the functionality of Equipment in the event of a SSE after a number of OBEs.
- Methods and Procedures of analysis or testing of supports for equipment.
- Seismic qualification report or similar equipment data documentation.


Required Spectrum
Acceleration
Frequency
Input Time History
Input to shake table
compare
Output
Acceleration
Frequency
Response Spectrum
Response Time History

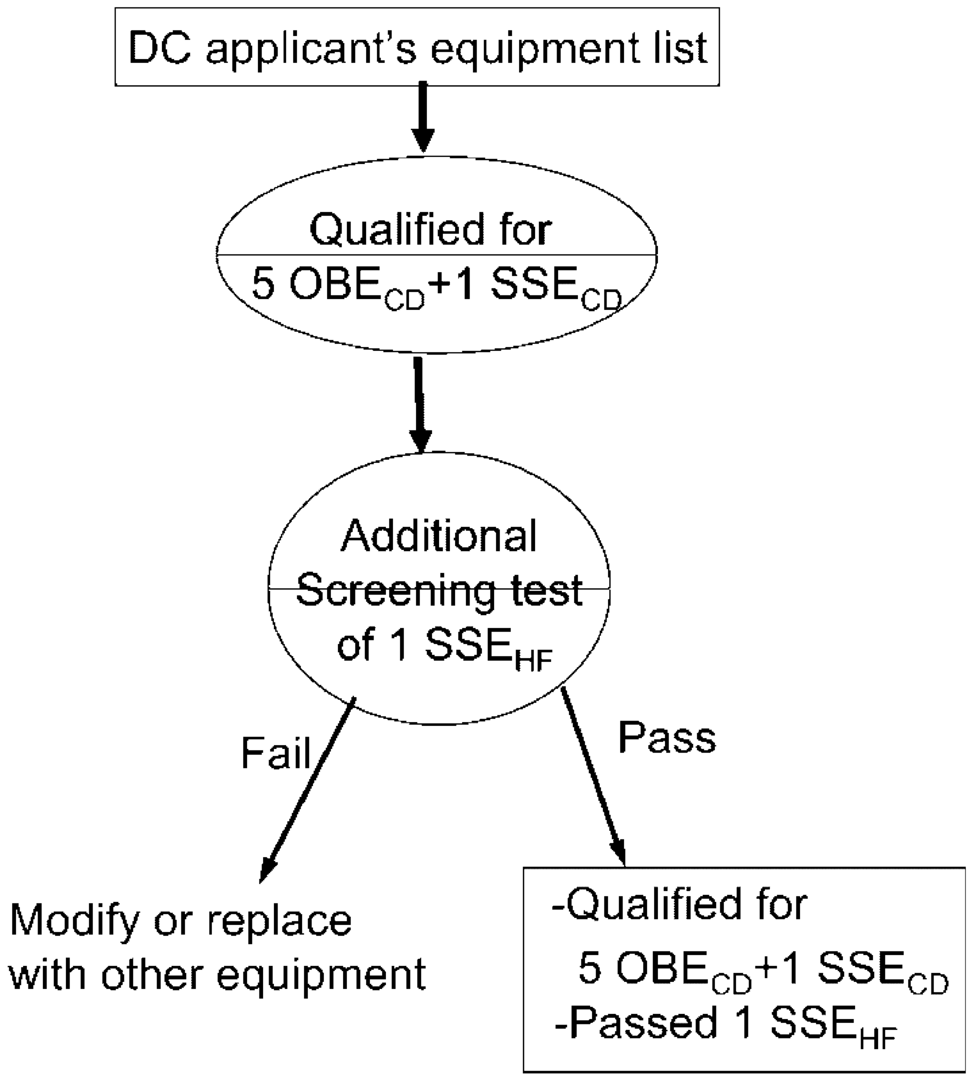
U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

High-frequency Exceedance



U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Staff Concern on Industry's Approach for Case I



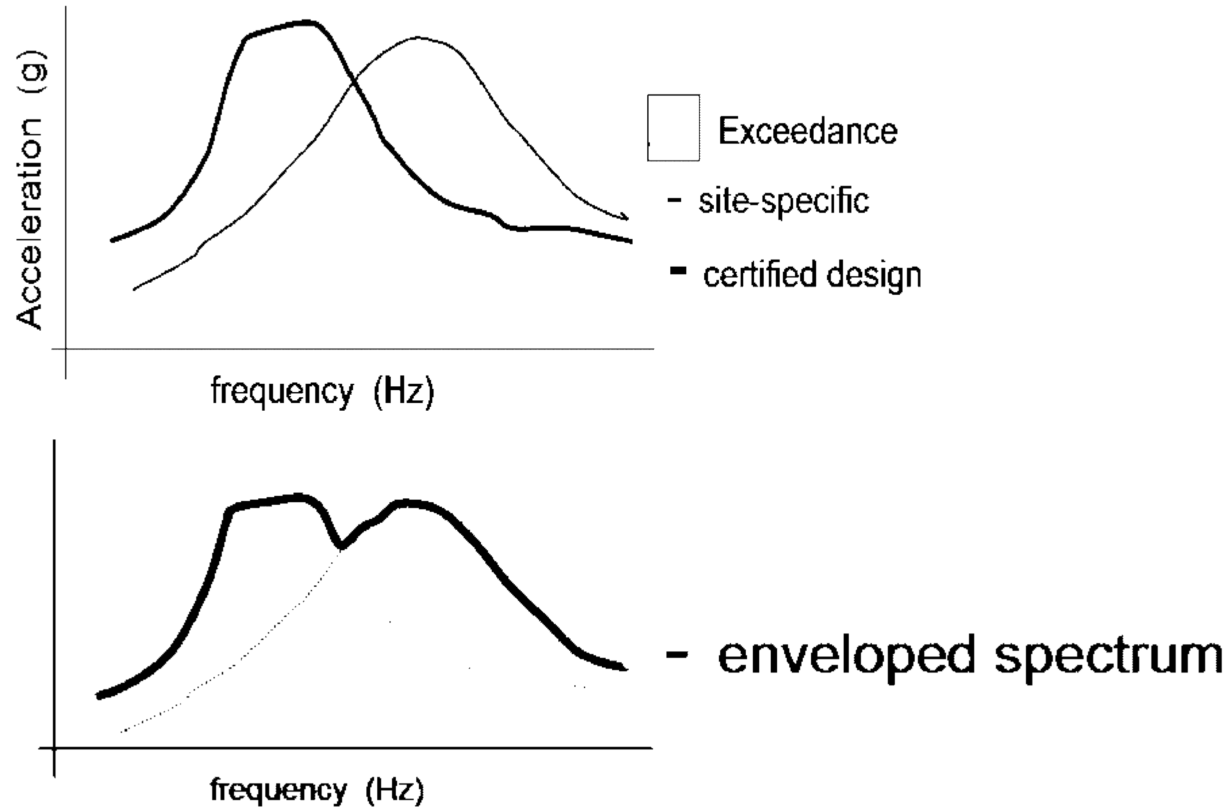
Resolution

$5\ OBE_{CD} + 1\ SSE_{CD}$ & $1\ SSE_{HF}$

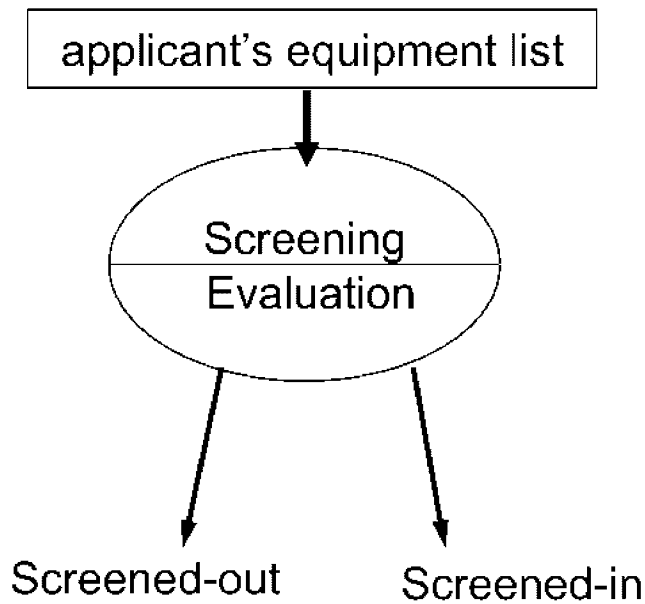
Show that it is equivalent to/more than

$5\ OBE_{HF} + 1\ SSE_{HF}$

Using IEEE Std 344 Annex D to compute equivalent peak stress cycles


Acceleration (g)
frequency (Hz)
Exceedance
- site-specific
- certified design
- enveloped spectrum
frequency (Hz)

Interim Staff Guidance on HF issues for DC/COL (ISG-1)



U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Staff Guidance

Staff Requirements Memorandum of SECY-93-087:

For nuclear power plants that were designed and/or licensed with the elimination of the OBE (plants with OBE defined as equal or less than 1/3 of SSE), electric and mechanical equipment qualified by testing should be qualified with

5 x OBE + 1 x SSE

5 x ½SSE + 1 x SSE

U.S.NRC
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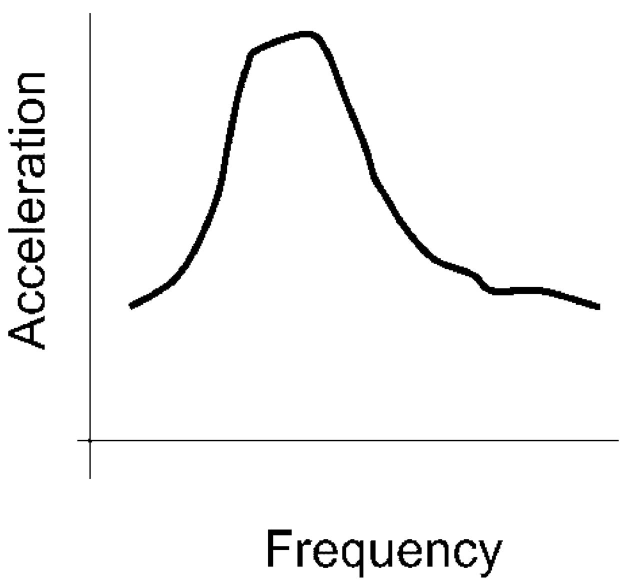
Protecting People and the Environment

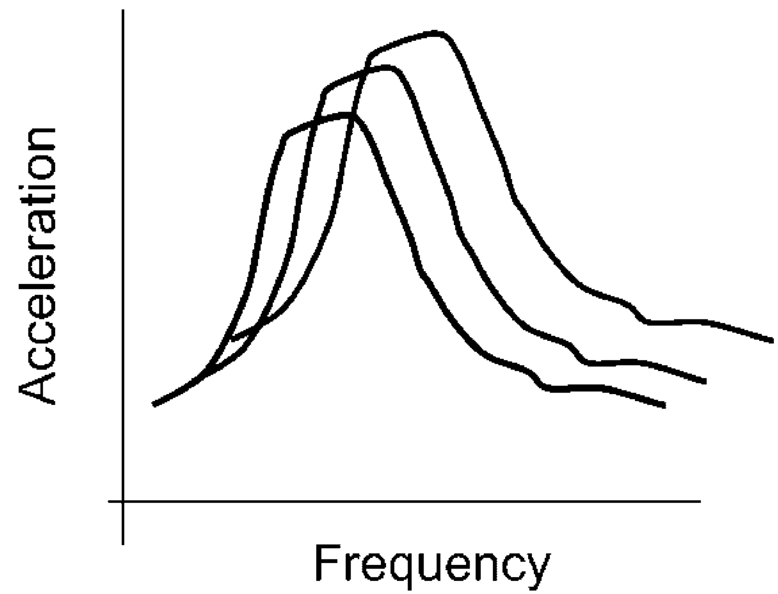
Response Spectrum

(an example)



Response Spectrum

(soft, medium, hard)



ABC's Of Welding

John Honcharik
Office of New Reactors

June 12, 2013

Agenda

- What is a weld and where is it used.
- Welding processes.
- Weld joint design.
- Weld Procedures.
- Welder Qualification.
- Issues with welds.
  - Steels
  - Stainless steels
  - Weld defects
  - Dissimilar metal weld degradation
  - Weld residual stresses

Welding

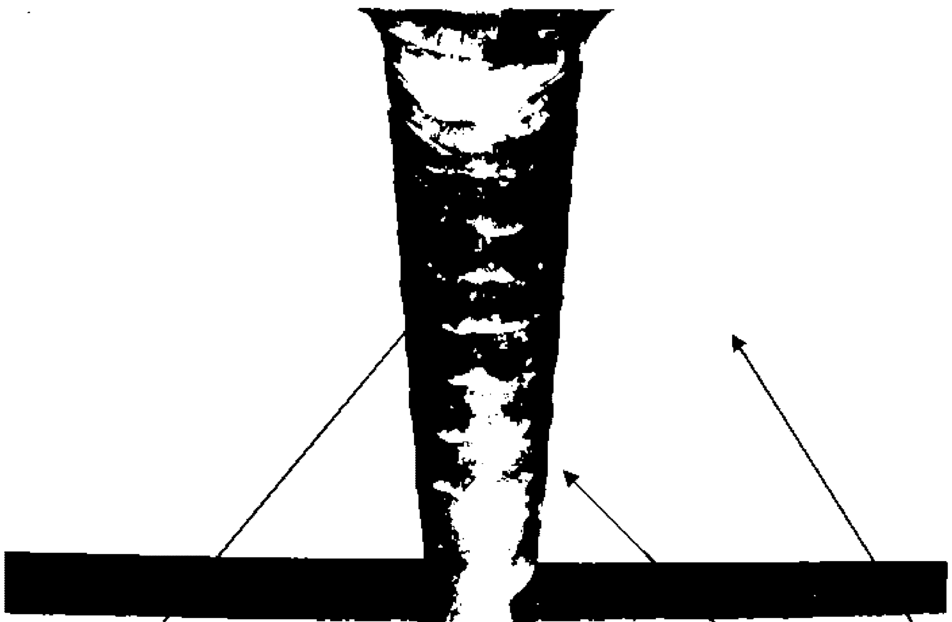
- A weld:
  - A localized coalescence of metals or nonmetals produced by heating the materials to the welding temperature, with or without the application of pressure, or by the application of pressure alone and with or without the use of filler metal.
- Is it an art, or is it science?
- A bit of both (automation tries to take out some of the art/skill)

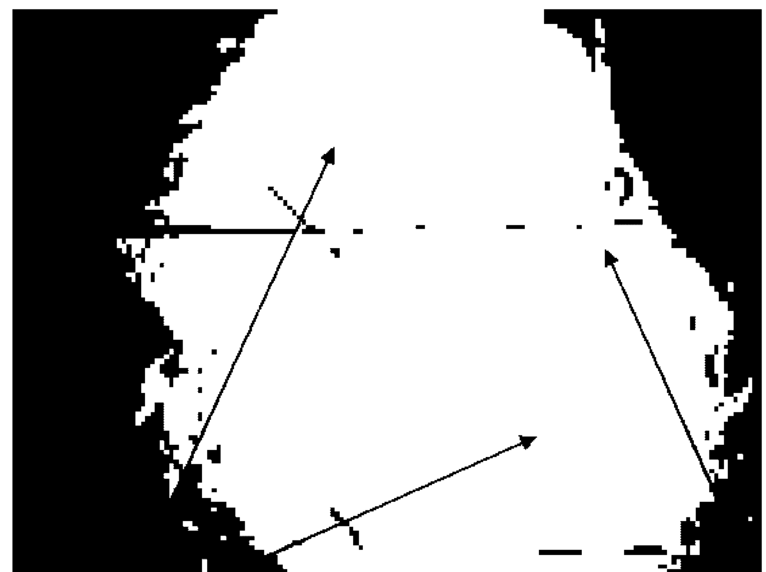
Welding

- Welding versus brazing/soldering
  - An atomic bond between the atoms at that interface is predominant, and the process that produces that joint is called welding.
  - Brazing or Soldering
    - A mechanical bond is predominant at the interface created by the process (brazing, soldering and thermal spraying)

Welding

- Weld Joint Brazed Joint



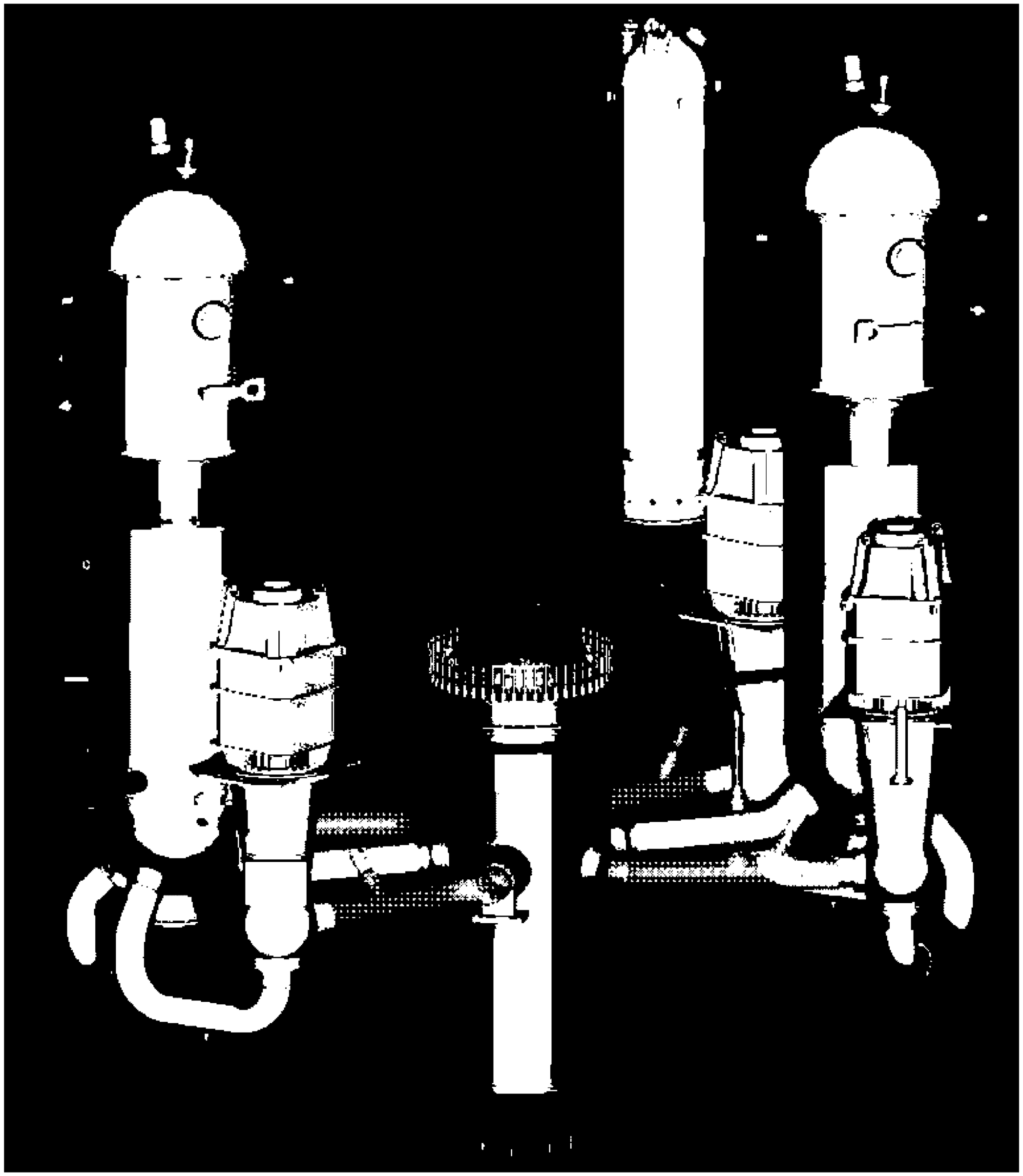


Weld metal

Heat affected zone (HAZ)- base metal which has not been melted, but whose mechanical properties or microstructures have been altered by the heat of welding

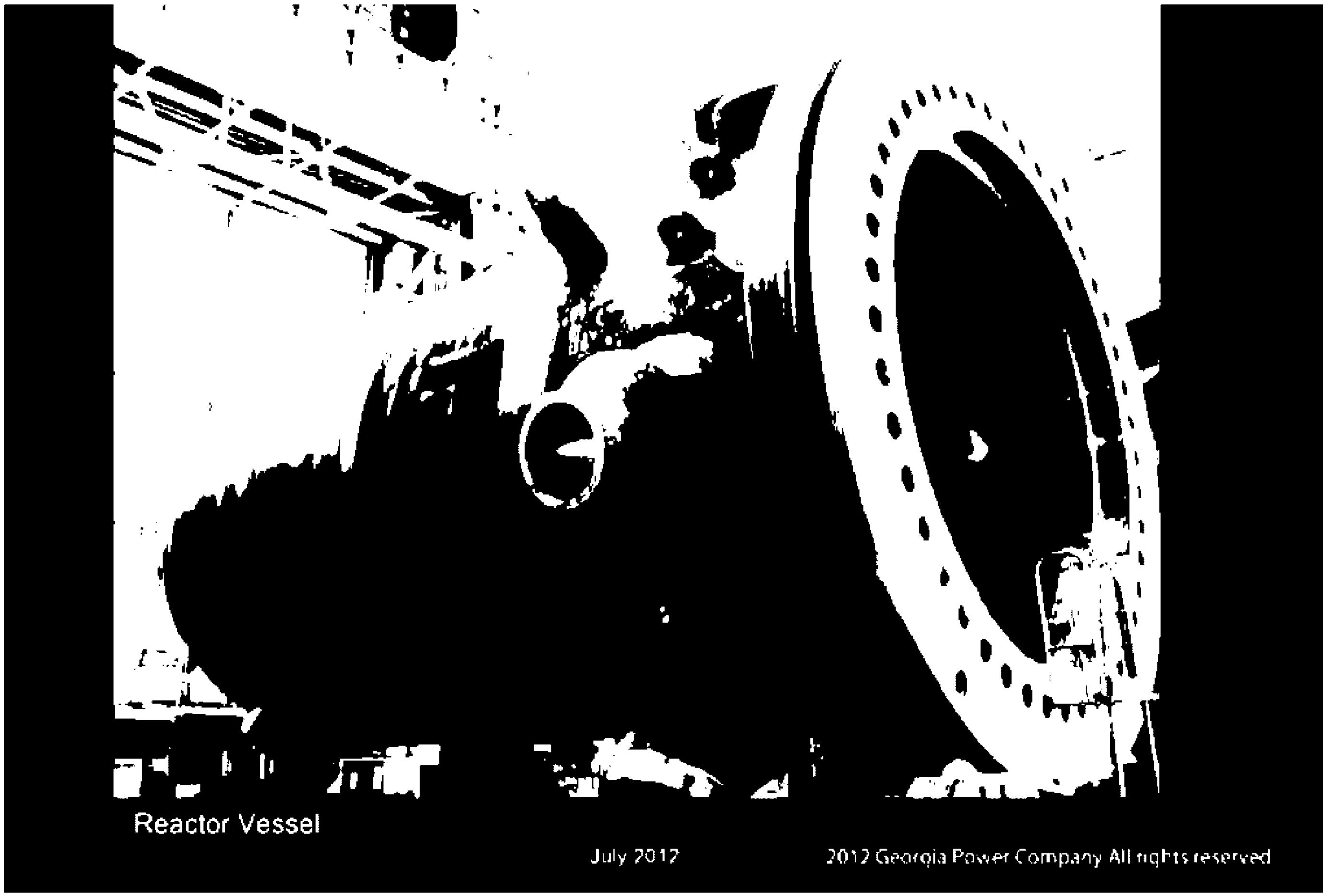
.

Where do you use welding?

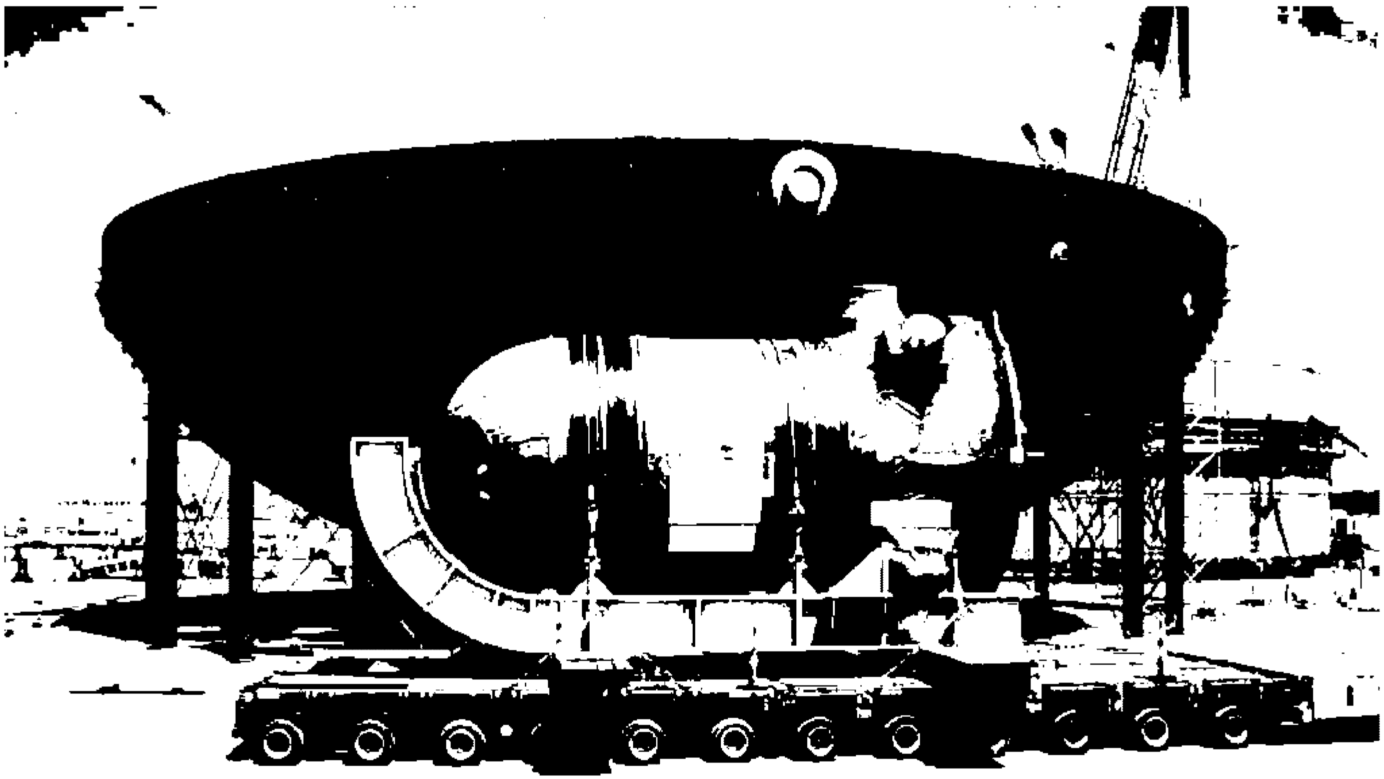


- Pressure vessels
- Piping
- Components
  - CRD
  - Pumps
  - Valves
  - Internals
  - Core supports
- Just about everywhere you do not use mechanical joints (bolting)

Weld- Integral Part of Components



Weld - Reactor to Structural

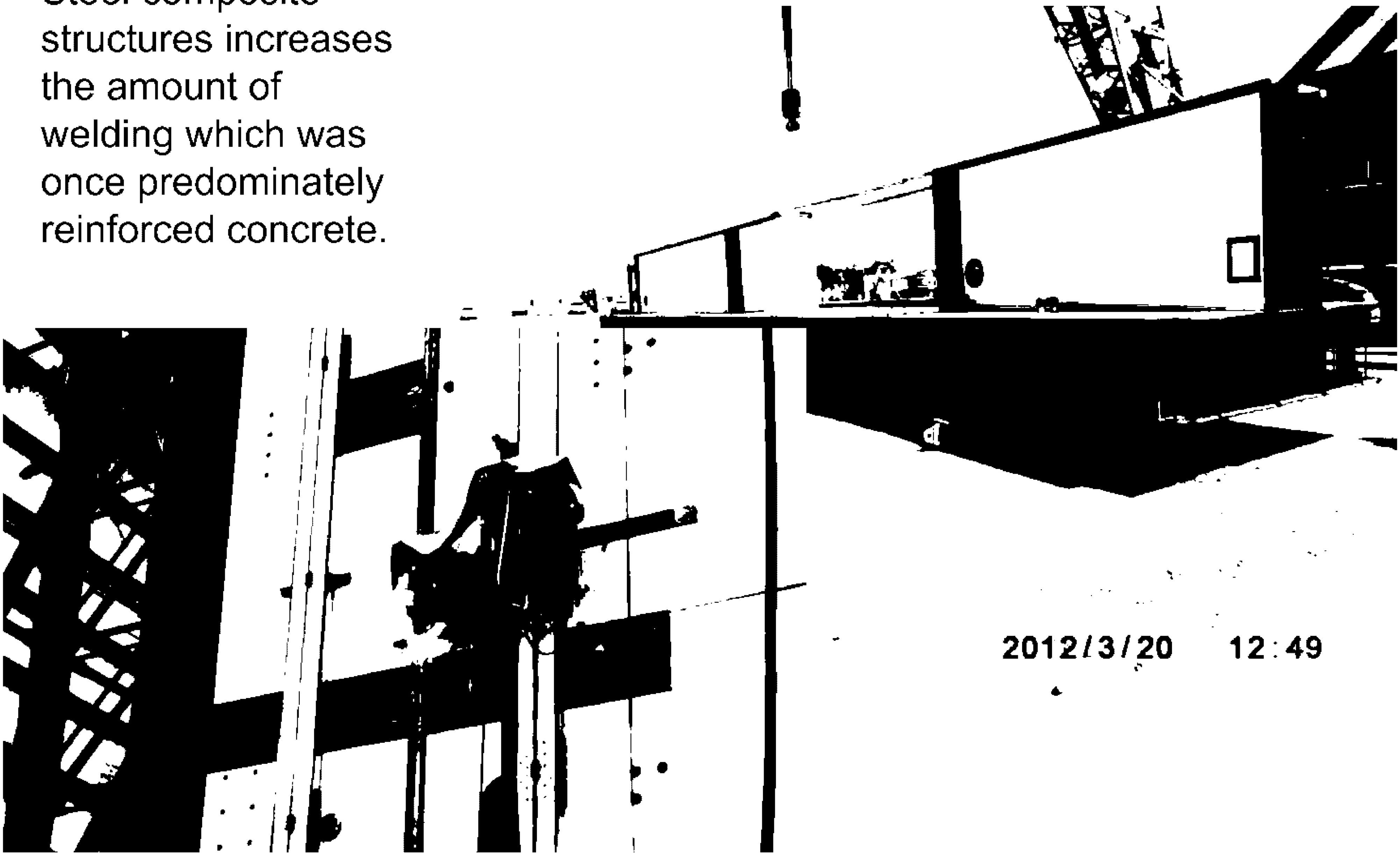


Vogtle Unit 3 reactor vessel is transported in front of Unit 4 containment vessel bottom head

May 2013

Modular Construction

Steel composite structures increases the amount of welding which was once predominately reinforced concrete.



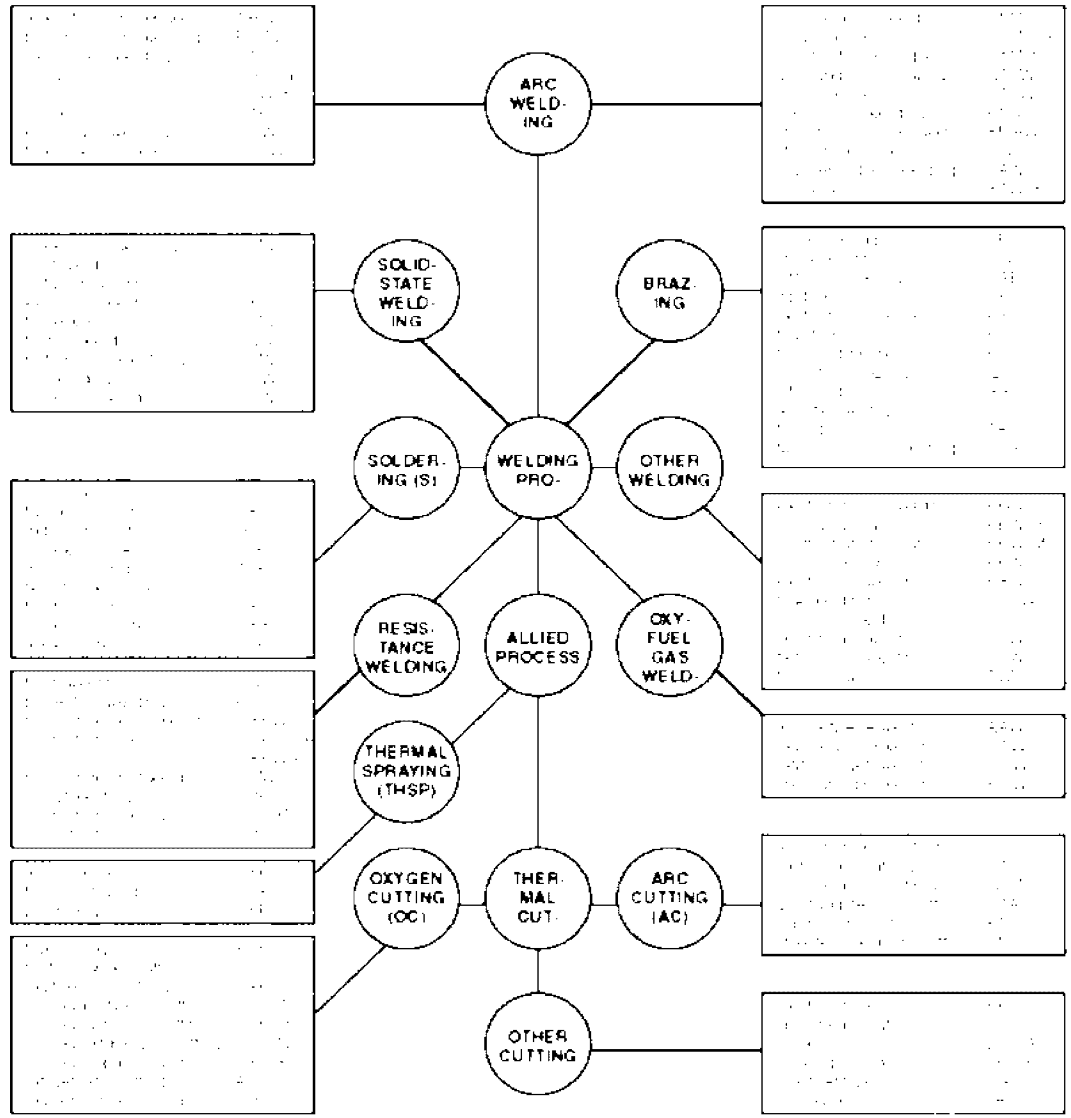
Welding Processes

- Processes most commonly used in the nuclear industry is Arc welding.

Annex D

American Welding Society Master Chart of Welding and Allied Processes

(This guide is not a part of ANSI Z49.1:1999, *Safety in Welding, Cutting, and Allied Processes*, but is included for information purposes only.)

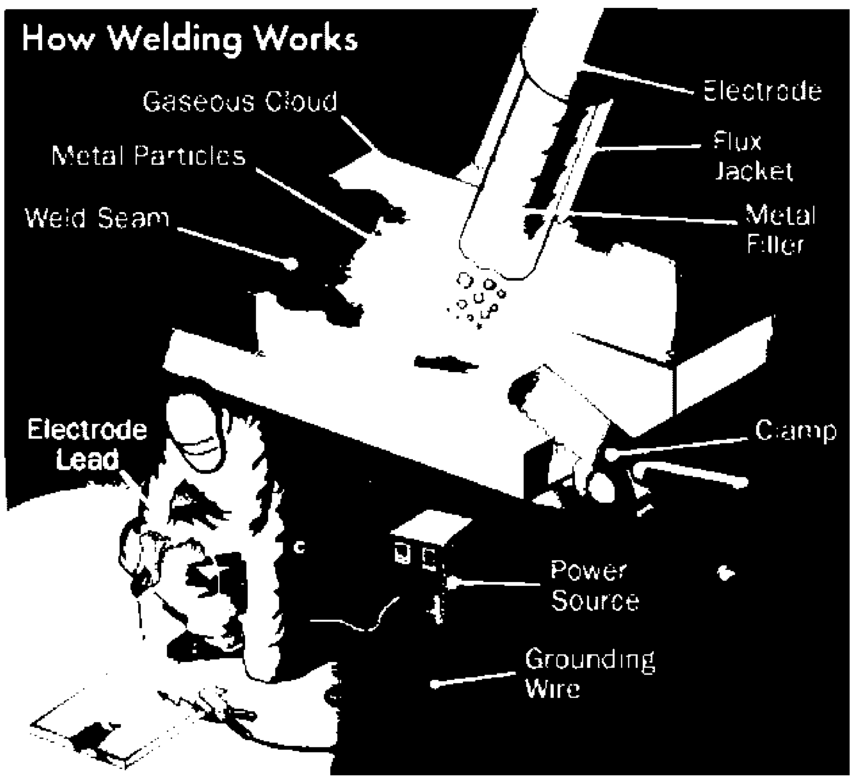


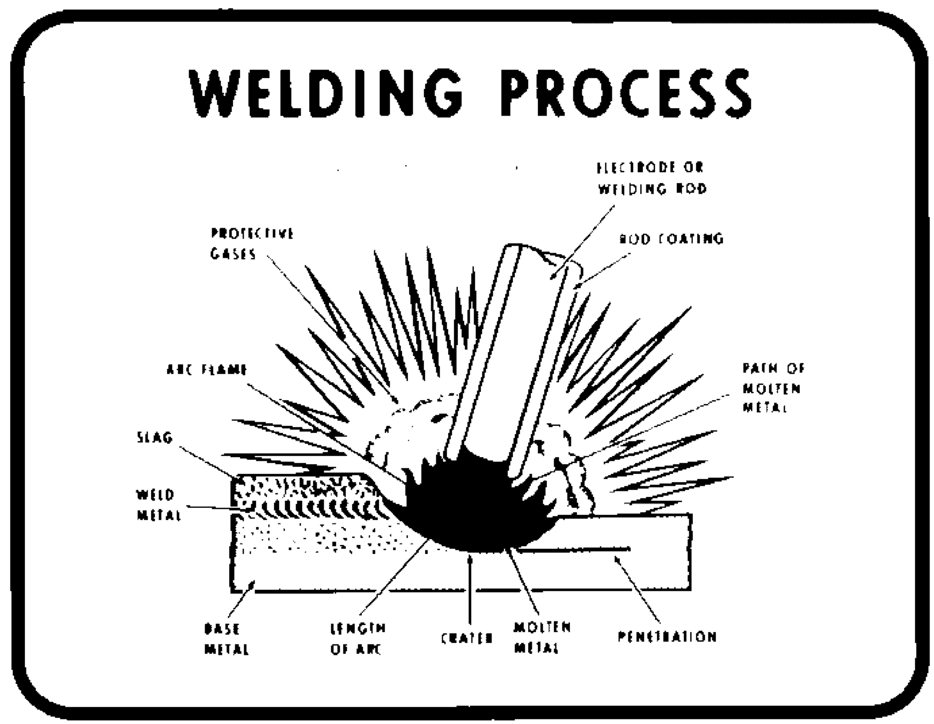
Welding Processes

- Shielded Metal Arc Welding (SMAW)
- Gas Metal Arc Welding/Flux Cored Arc Welding (GMAW/FCAW)
- Gas Tungsten Arc Welding (GTAW)
- Plasma Arc Welding (PAW)
- Submerged Arc Welding (SAW)
- Electroslag Welding (ES)
- Stud welding (SW)
- Friction Welding (FRW)
- Laser Beam Welding (LBW)

Welding Process - SMAW

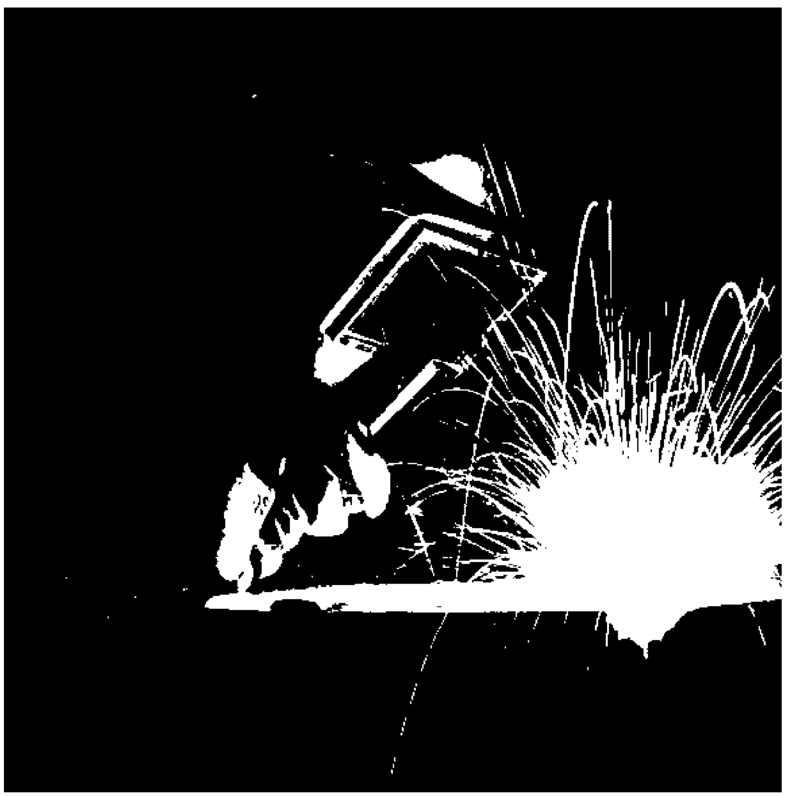
- Shielded Metal Arc Welding
  - Welding arc produced by completing electrical circuit
  - Uses flux to shield arc and weld metal from contaminants and oxidization

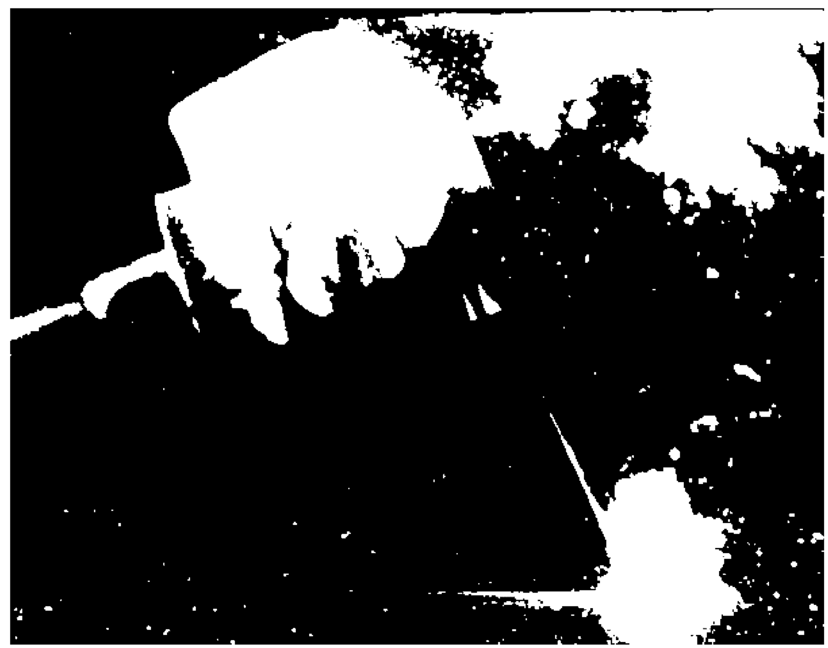




Welding Process - SMAW

- Flux can also add alloying elements to weld
- Filler metal designation example : E7018 (E-electrode, 70 minimum tensile strength, 1-position (all), 8 – usability (DCEP, low hydrogen –iron powder flux)

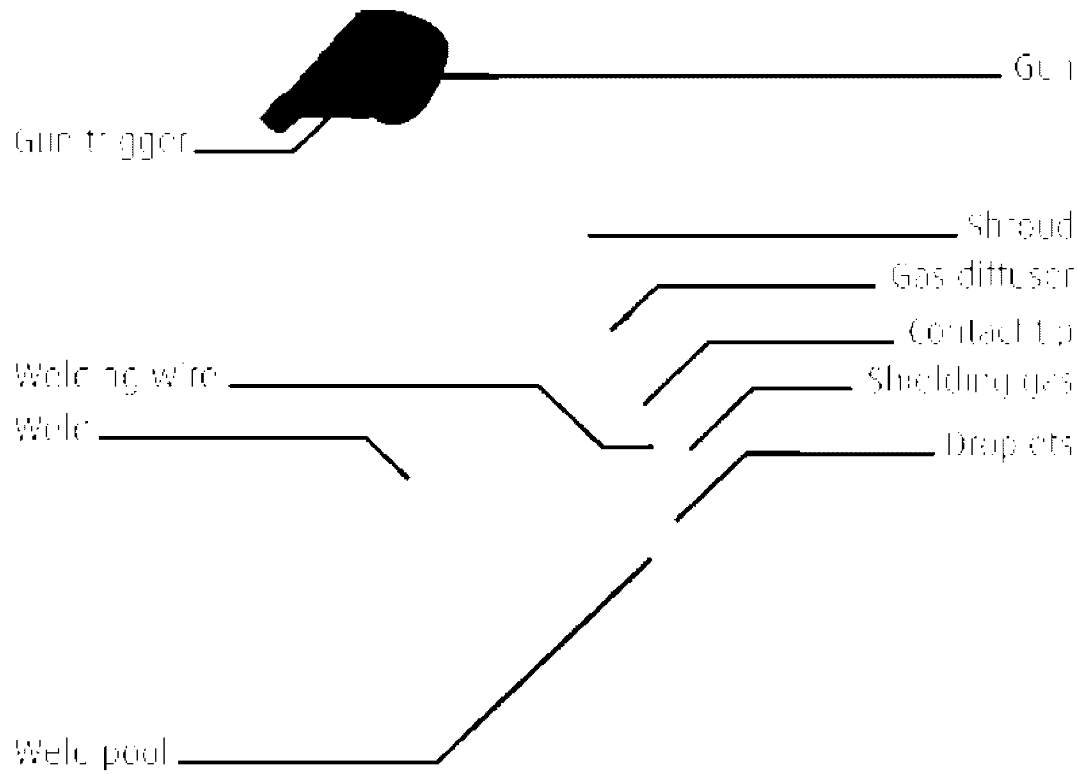




Welding Process-GMAW/FCAW

Gas Metal Arc Welding/ Flux Cored Arc Welding

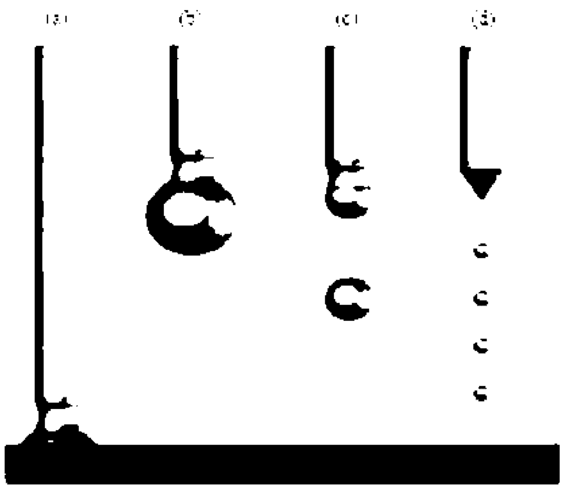
- Uses inert gas to shield arc and weld metal from contaminants and oxidizing
- FCAW uses flux (in core of wire) and sometimes inert gas shielding also
- Semi-automatic (auto wire feeder)

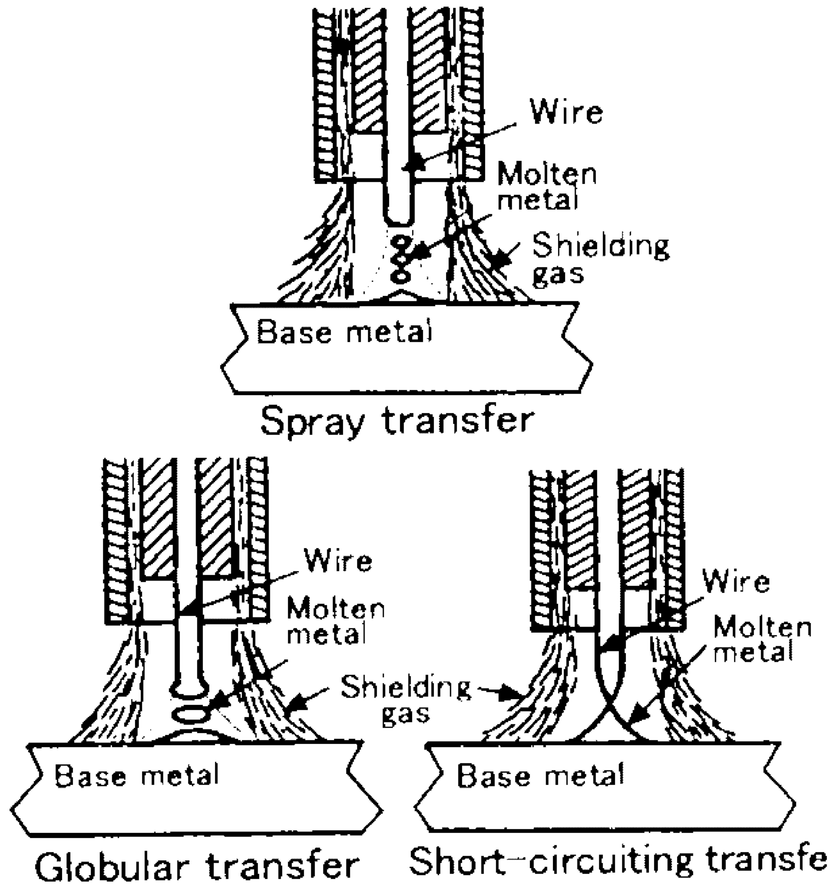


The welding circuit consists essentially of the following elements:

Welding Process-GMAW/FCAW

Different Modes of Weld Transfer

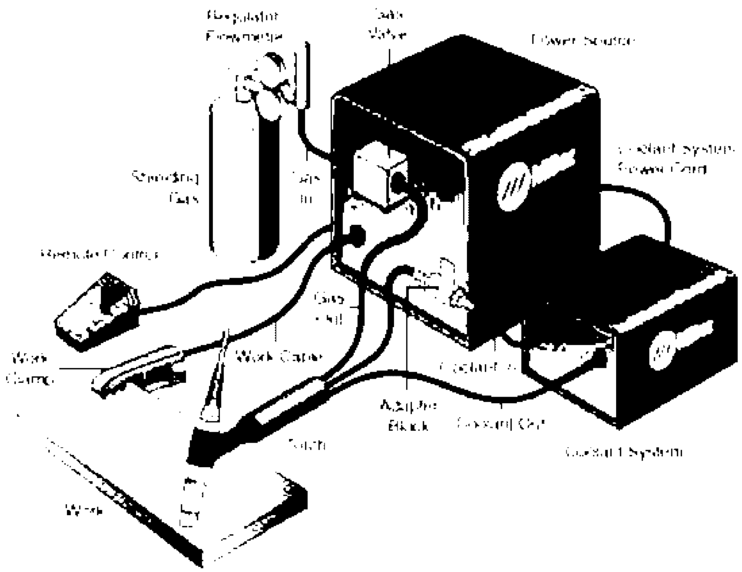


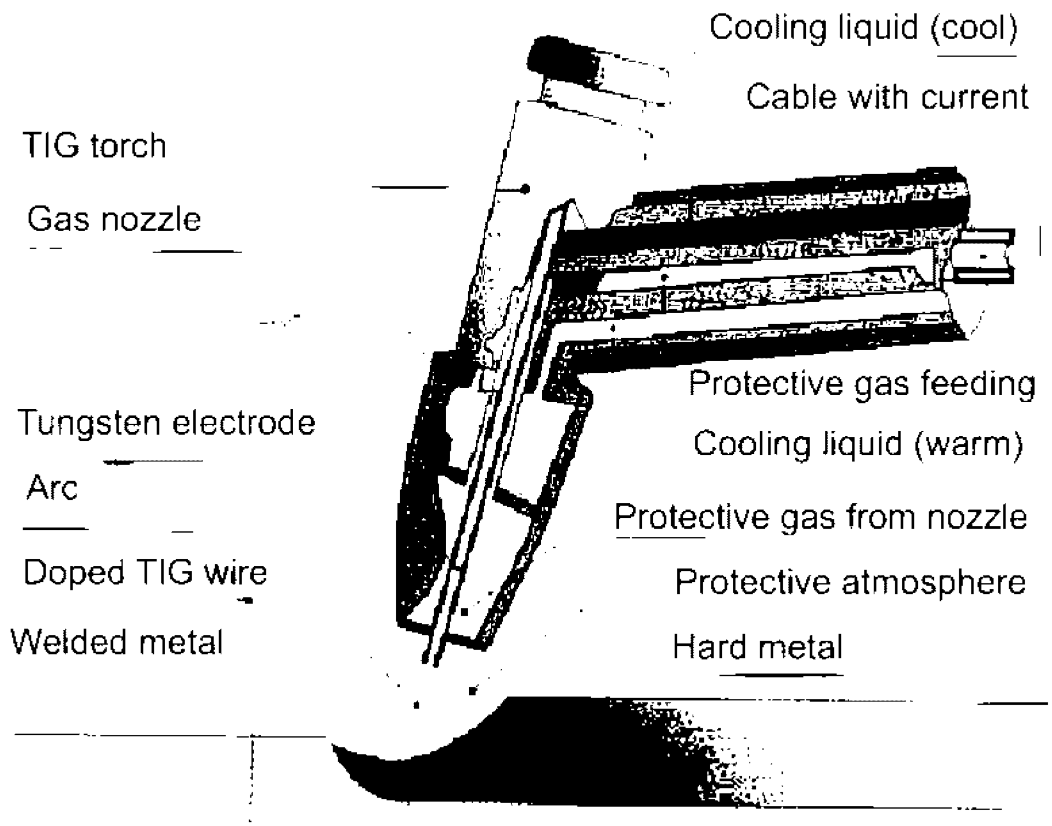


- SC $75Ar/25CO_2$
- Globular CO_2
- Spray $95Ar/5CO_2$ and higher voltage/amps
- Pulsed –power source pulses (smaller weld puddle-more control)

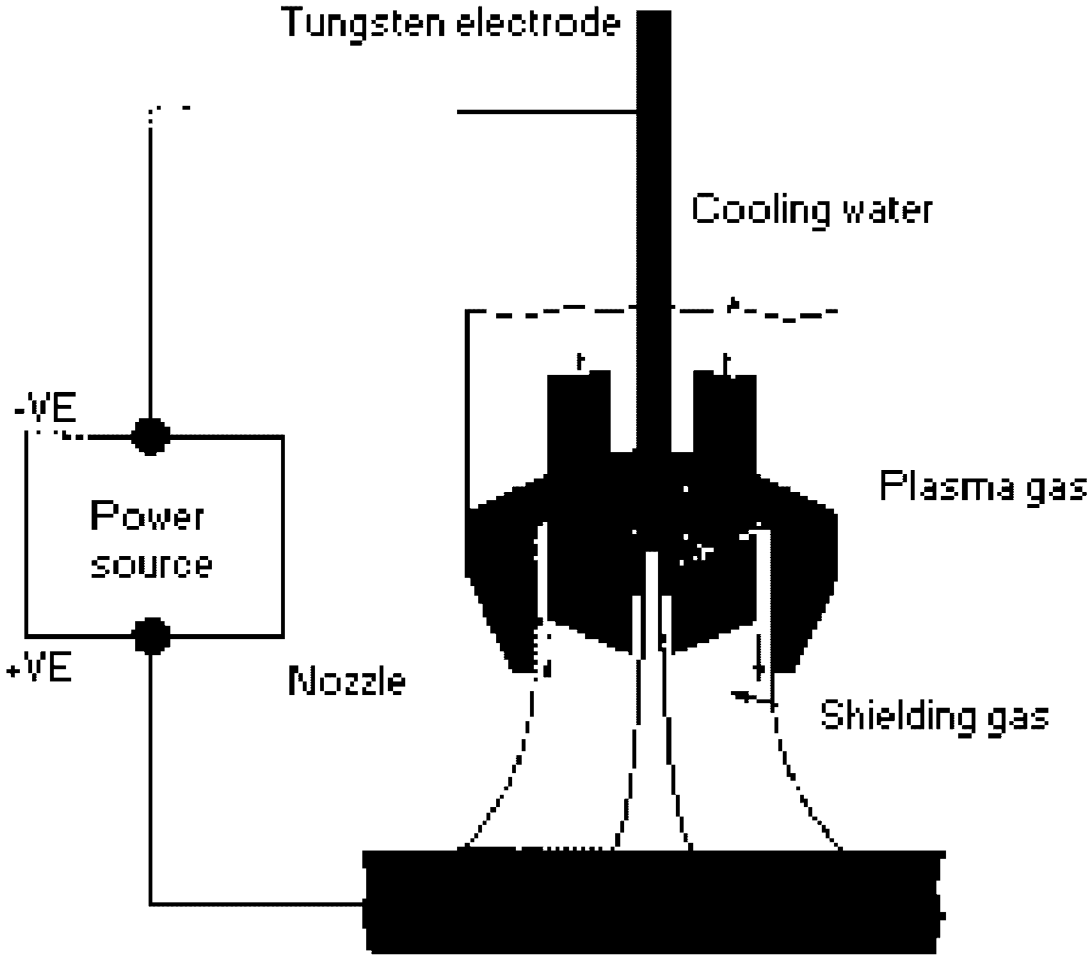
Welding Process – GTAW (Gas Tungsten Arc Welding)

GTAW produces high quality weld, but low deposition rate

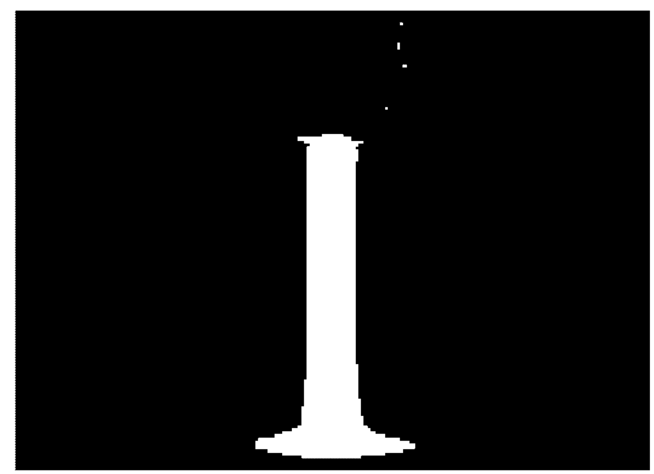




Welding Process – PAW (Plasma Arc Welding)

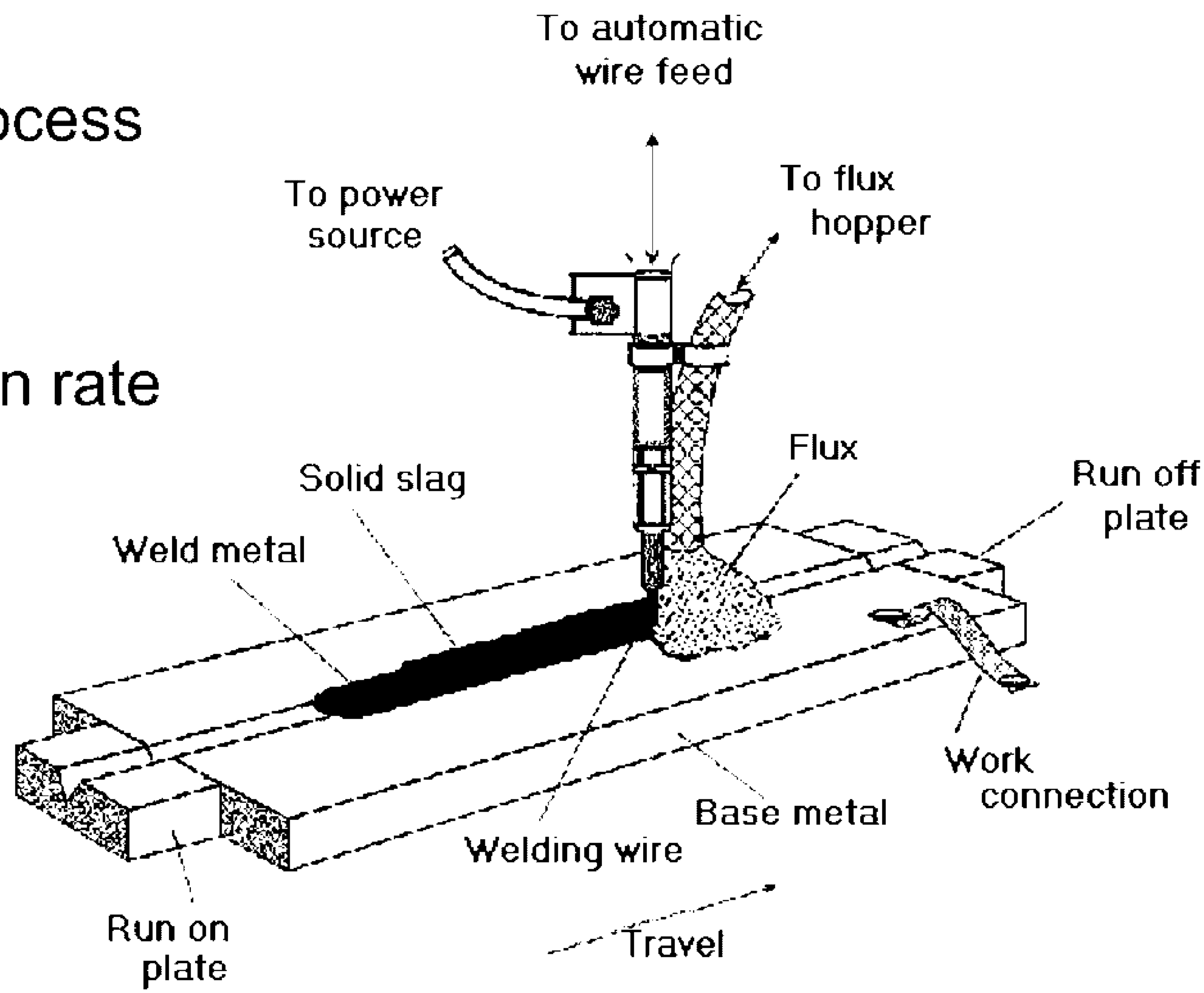


- Similar to GTAW but uses additional gas to concentrate the arc.
- Can be used for hardfacing.

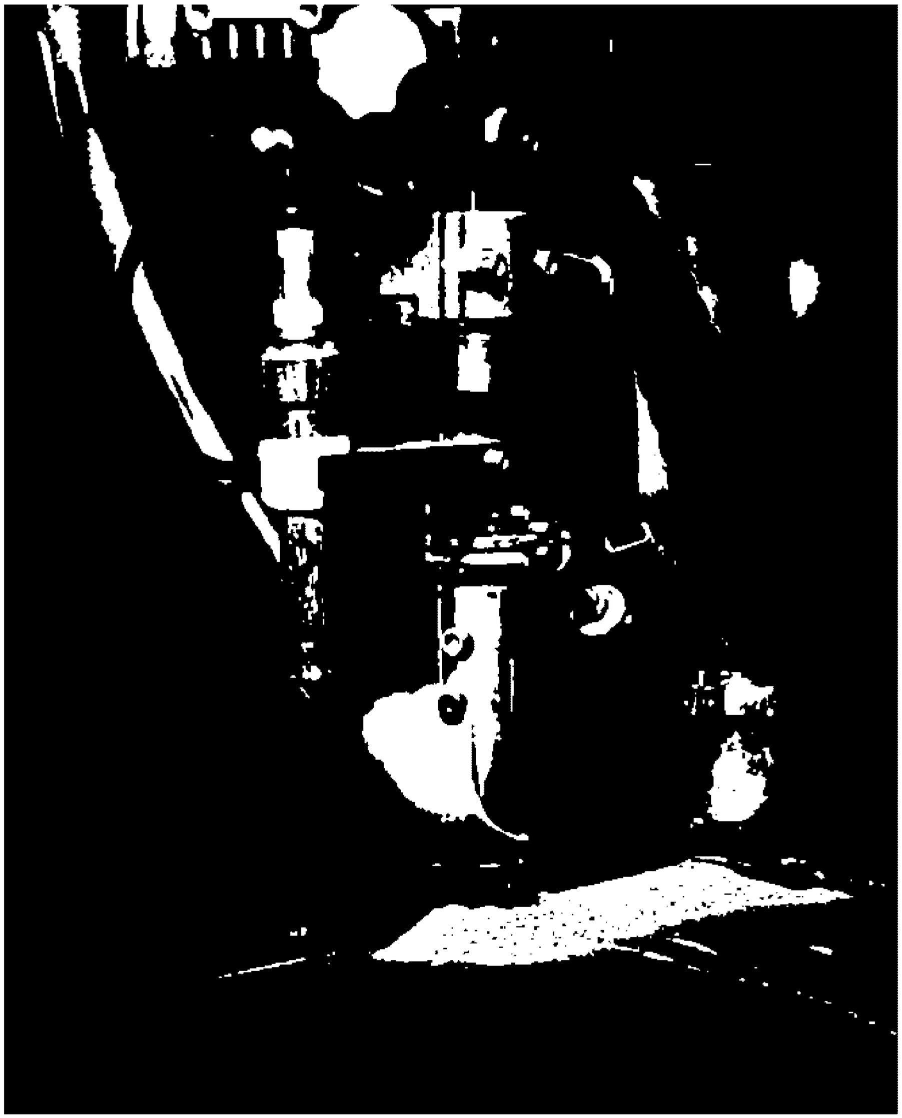


Welding Process - SAW
Submerged Arc Welding

- Automated process
- Uses flux for shielding
- High deposition rate



Welding Process - SAW

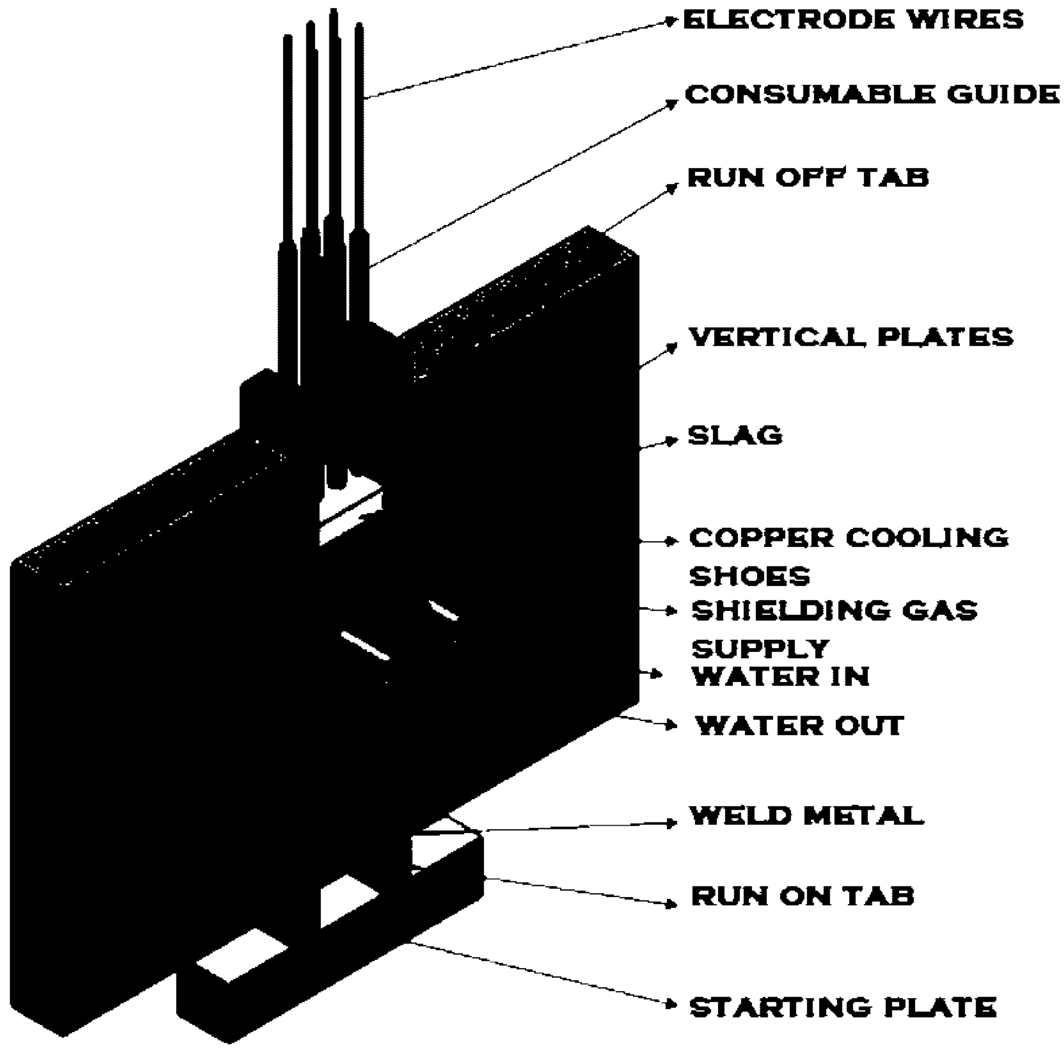


Welding Process - SAW

Strip cladding – form of SAW with different shape filler metal (primarily for cladding).

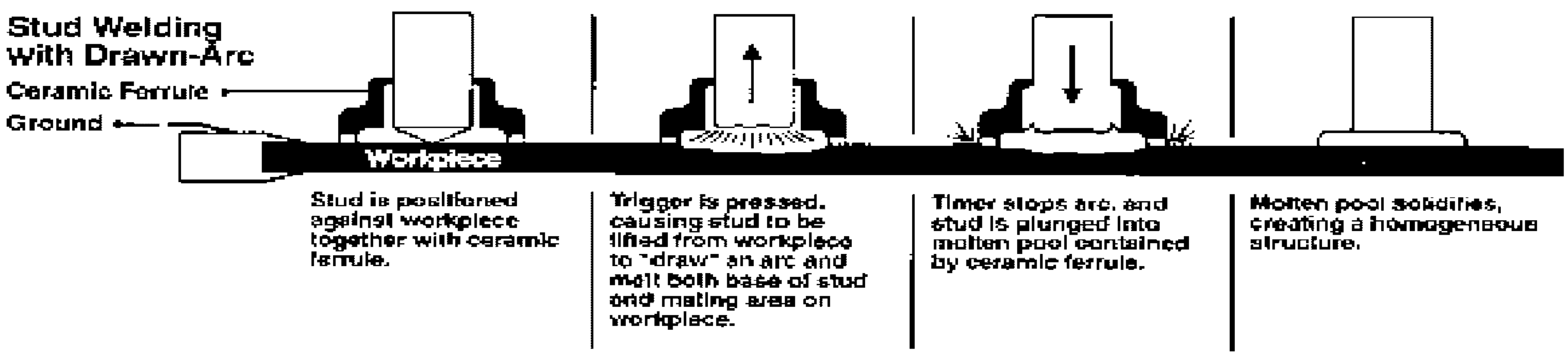
Welding Process - Electroslag

- Such high deposition rates that it is similar to casting material
- Can affect material properties based on orientation of solidified weld metal
- Metal crystal solidification with dendritic growth from weld sides to center of weld having an acute angle results in stronger centerline bond (RG 1.34).



ELECTROGAS WELDING

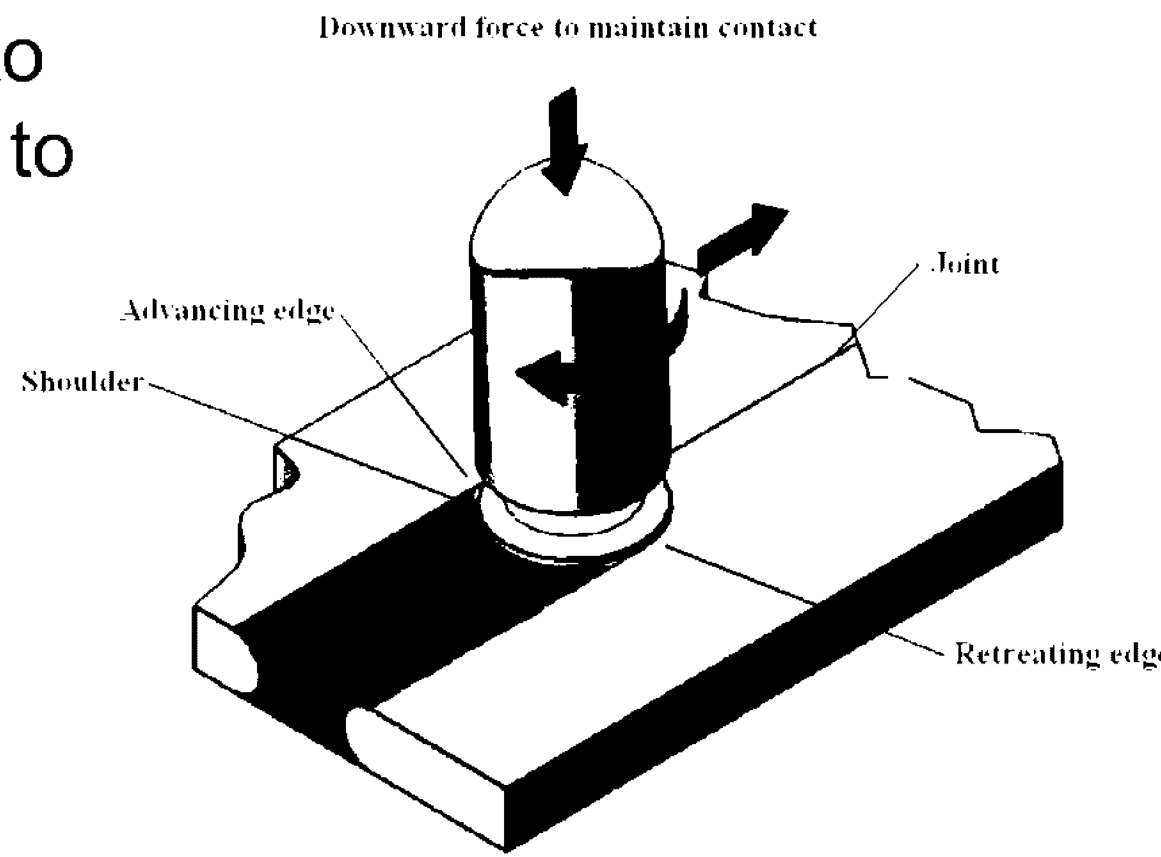
Welding Process – Stud Welding



- Stud welding used in structural applications (temporary or permanent).
- Mostly mechanized (Stud welding gun).

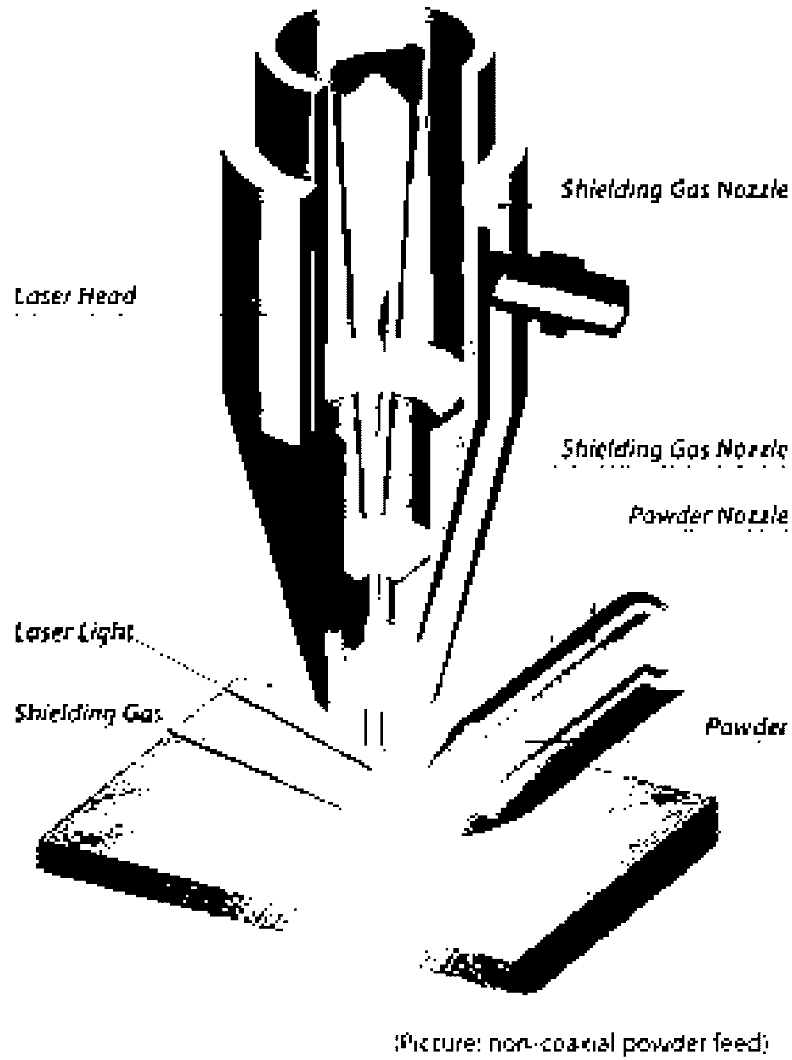
Friction Welding

- Uses non-consumable tool to melt material due to pressure.
- No addition of filler metal.



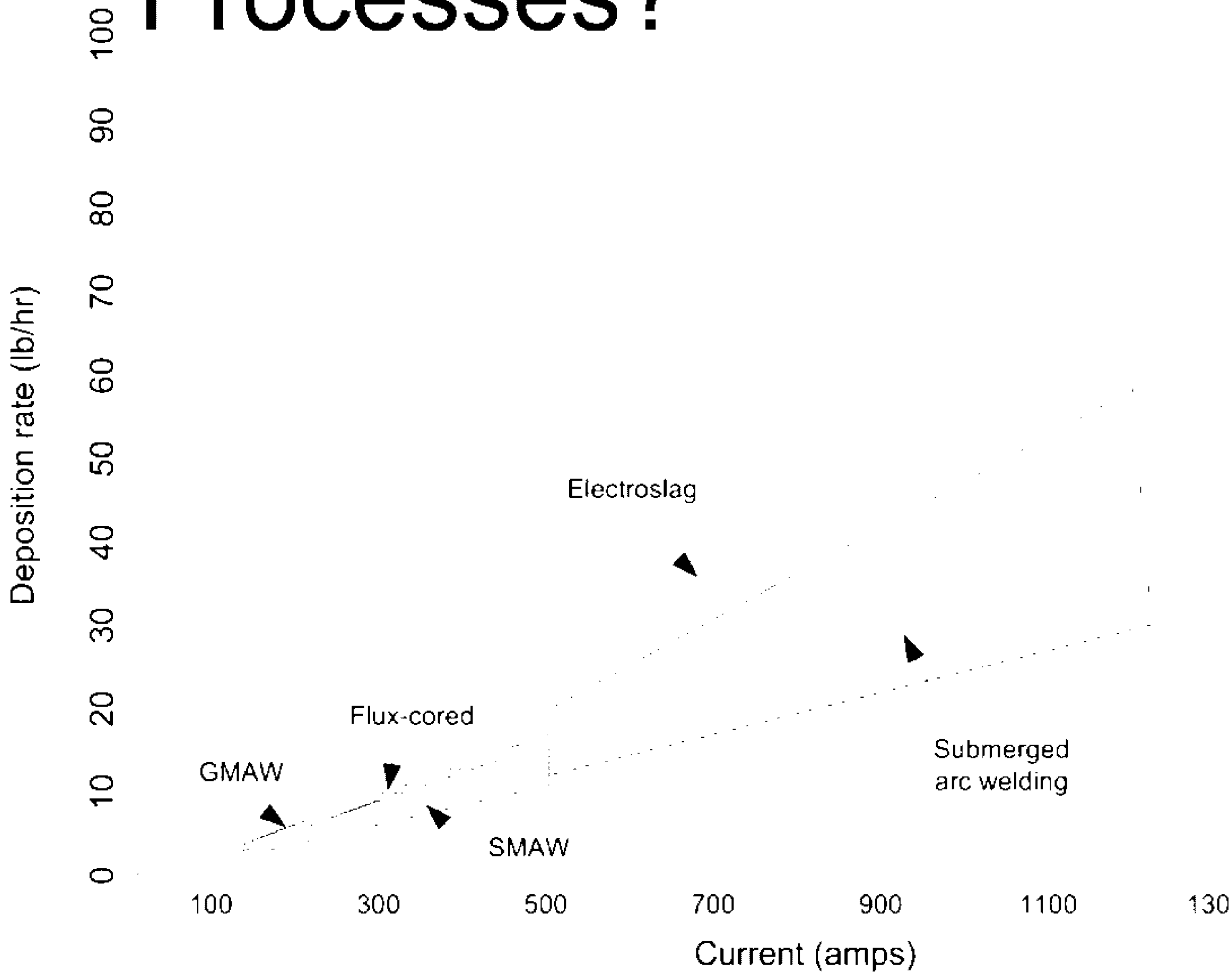
Laser Beam Welding

- Uses laser to melt material in lieu electrical arc.
- Higher cost.
- Better quality.
- Piping inlays.



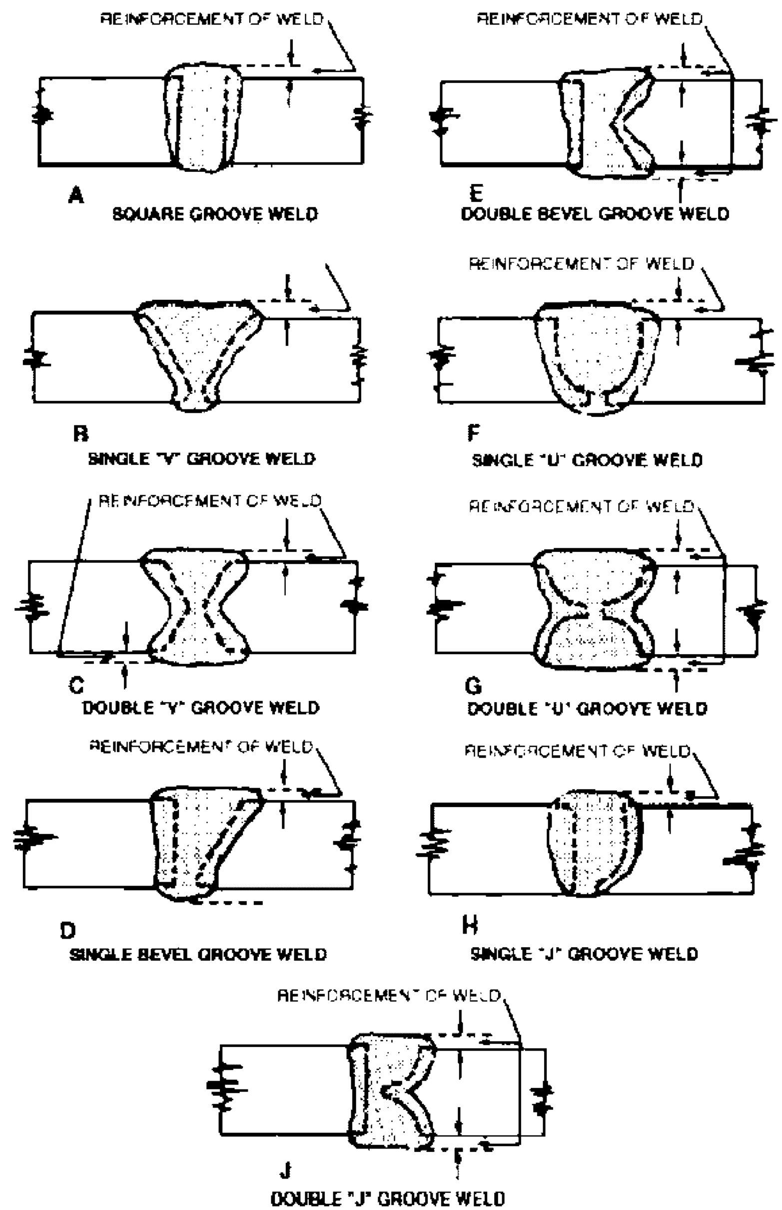
Welding – Why Different Processes?

- Productivity (related to deposition rate) is major reason.
- Quality
- Access
- Training/skill
- Type of weld joint
- Position



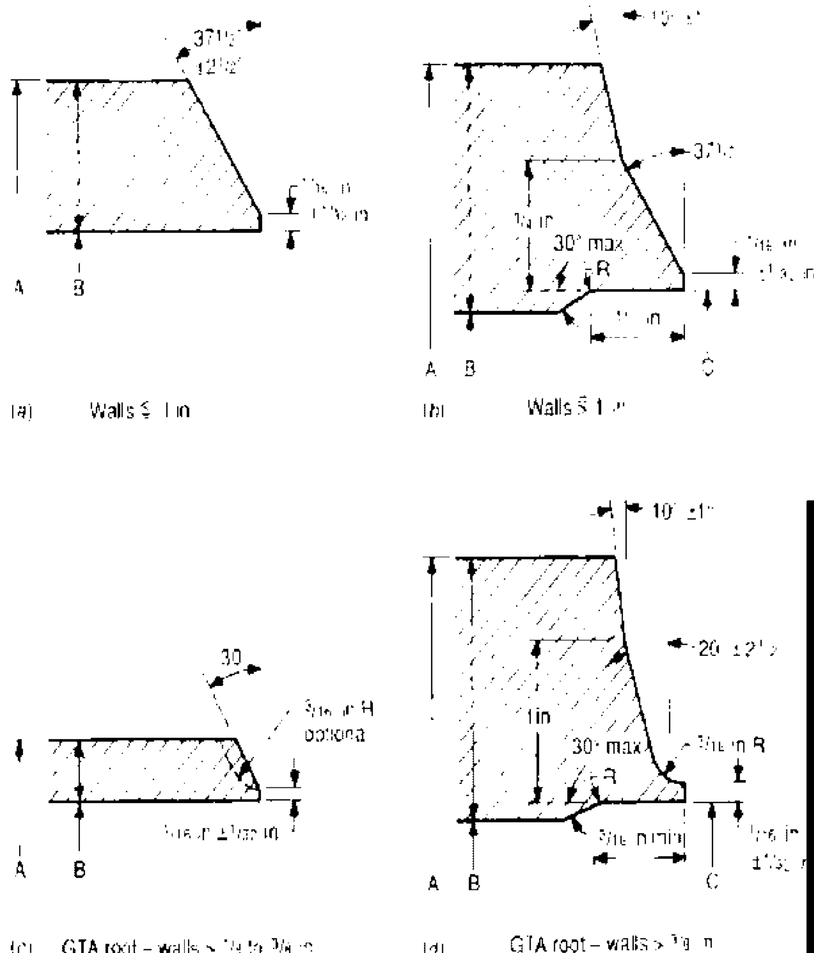
Weld Joint Design

- Different Joint types depends on:
  - Access
  - Efficiency/Strength
  - Ease of welding
  - Ease of machining
  - Welding process
  - Thickness
  - Distortion

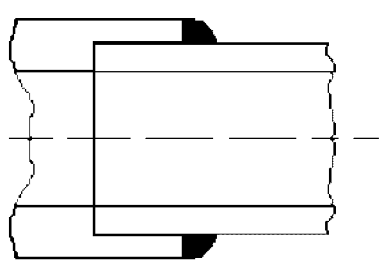


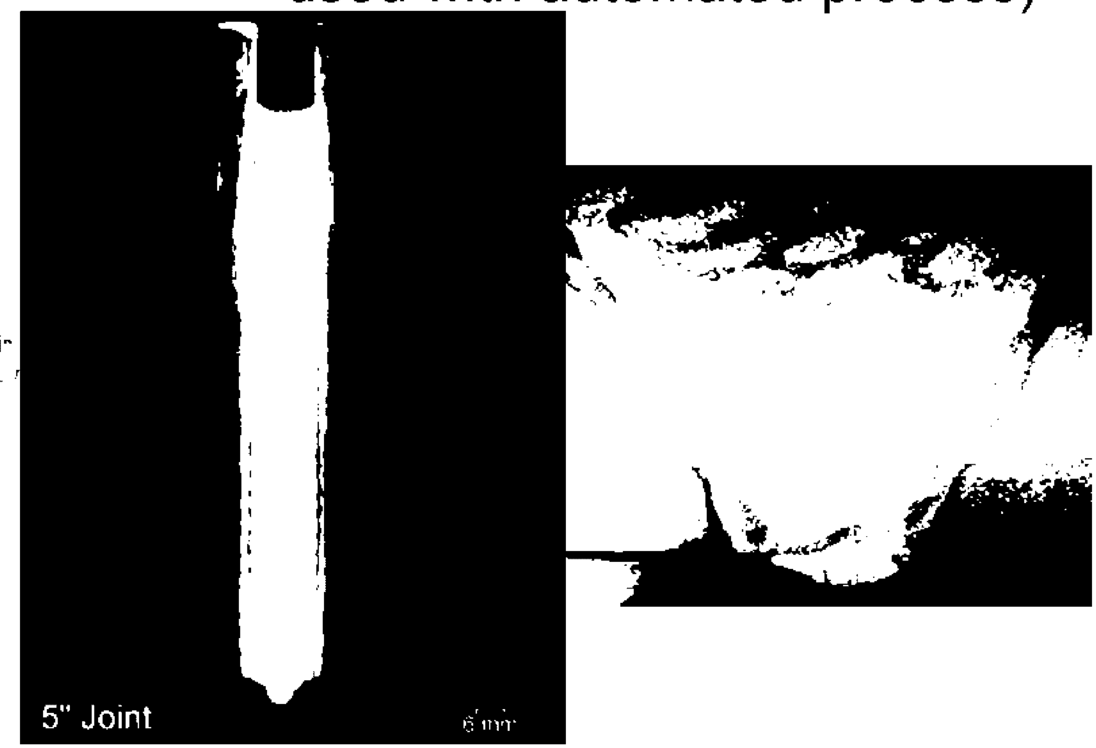
Weld Joint Design

- Simple to Complex
- Balance quality and productivity
- Narrow groove weld uses less weld metal therefore faster welding (however, must be used with automated process)



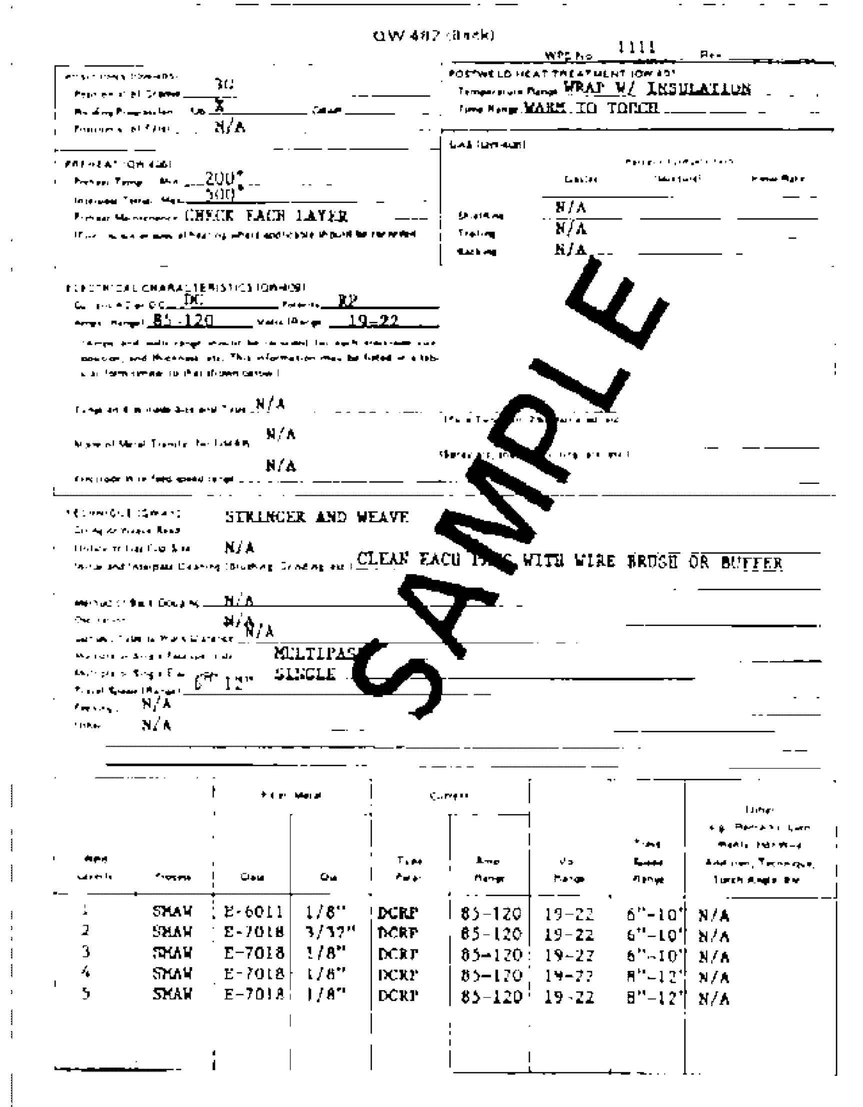
Socket weld end





Weld Procedures

- Common/essential parameters (variables)
  - Depends on weld process and applicable code used
  - Typical parameters:
    - Amps/polarity
    - Volts
    - Travel speed
    - Wire speed
    - Base metal/filler metal
    - Base metal thickness
    - Joint (backing ring, EB ring)
    - Shielding
    - Preheat/interpass


QW-482 (Back)

WPS No. 1111

3G

X

N/A

WRAP W/ INSULATION

WARM TO TOUCH

200°

500°

CHECK EACH LAYER

N/A

N/A

N/A

DC

RP

85-120

19-22

N/A

N/A

N/A

STRINGER AND WEAVE

N/A

CLEAN EACH PASS WITH WIRE BRUSH OR BUFFER

N/A

N/A

N/A

MULTIPASS

SINGLE

6"-12"

N/A

N/A

SAMPLE

1	SMAW	E-6011	1/8"	DCRP	85-120	19-22	6"-10"	N/A
2	SMAW	E-7018	3/32"	DCRP	85-120	19-22	6"-10"	N/A
3	SMAW	E-7018	1/8"	DCRP	85-120	19-22	6"-10"	N/A
4	SMAW	E-7018	1/8"	DCRP	85-120	19-22	8"-12"	N/A
5	SMAW	E-7018	1/8"	DCRP	85-120	19-22	8"-12"	N/A

Weld Procedures

- Preheat - the minimum temperature in the section of the previously deposited weld metal, immediately prior to welding
- Interpass temperature - the highest temperature in the weld joint immediately prior to welding, or in the case of multiple pass welds, the highest temperature in the section of the previously deposited weld metal, immediately before the next pass is started
- Preheat and interpass
  - Prevent unwanted material phase/properties microstructure
    - Untempered martensite
    - Carbide precipitation
    - Cracking, stresses
  - Good toughness
    - Typically, increase in strength will decrease toughness (brittle vs. ductile)
    - Need balance for each application

Weld Procedures

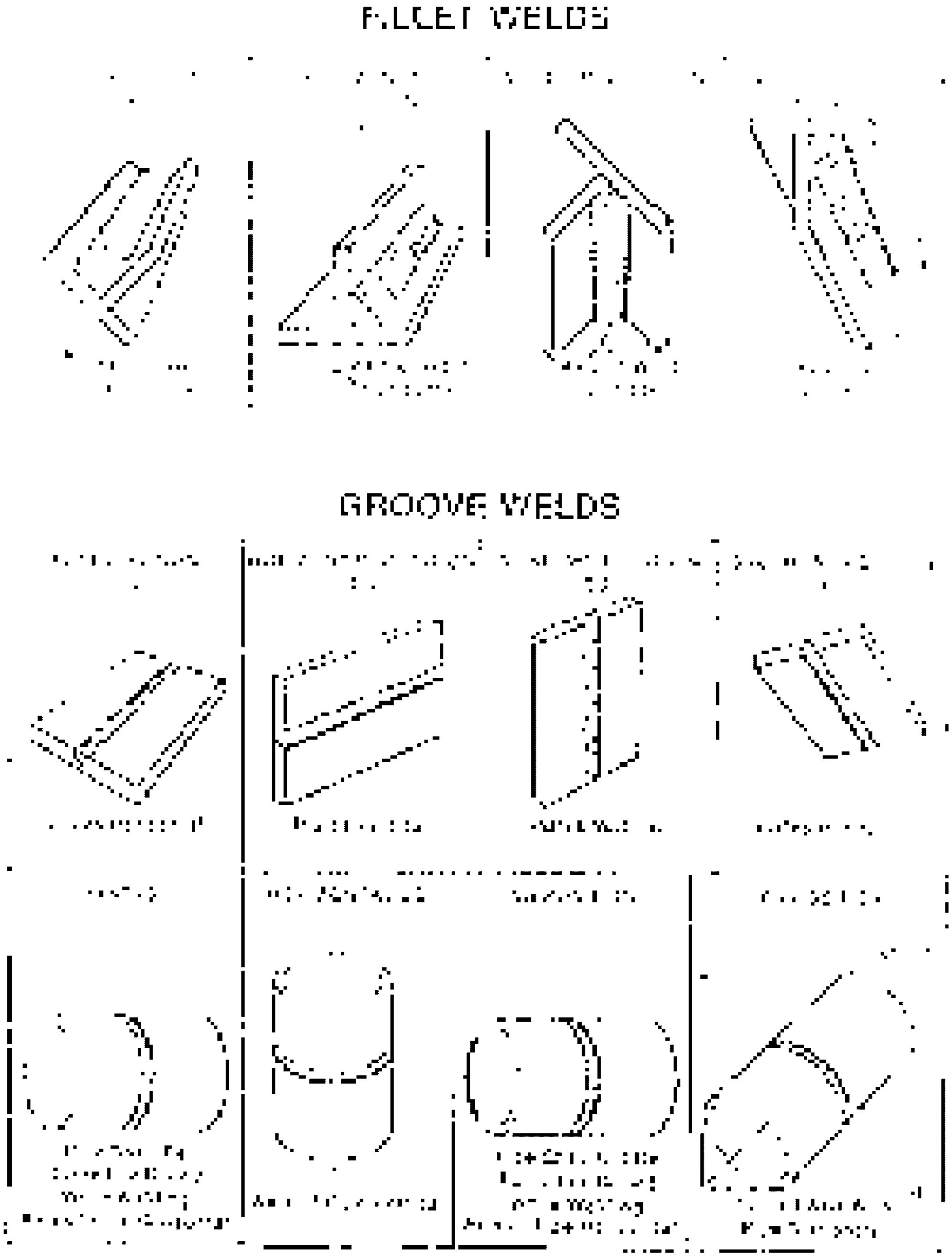
- Changes to essential parameters (variables) require requalification of procedure
- Changes to supplemental variables may require partial requalification
- Changes to non-essential variables do not require requalification
- Weld procedures are qualified in accordance with the applicable code and documented in a procedure qualification record (PQR).
- Note – specific codes only provide requirements on qualifying weld procedures and welders (does not state how to weld).

Procedure Qualification Record

- Base material are assigned to groupings (P-numbers) to group similar metals (properties) to reduce amount of procedure qualifications.
- Filler metals are also assigned to groupings (F-numbers) based on compatibility with base material characteristics.
- Joint designs, such as full penetration (groove welds- backing rings, double sided, EB ring), partial penetration (fillet welds).
- Base material thickness
- Preheat and interpass temperature

Procedure Qualification Record

- Positions.
- Shielding (flux and/or shielding gas)
- Electrical Characteristics (volts, amps, current, etc.)
- Technique and other requirements
  - Stringer vs. weave bead
  - Travel speed
  - Multipass to single pass, etc.
  - Cleaning
- Typically, NDE and tensile and bend tests performed on test assemblies to determine acceptability.



Positions

- Groove qualifies fillet
- Plate qualifies pipe

For Plate

QUALIFICATION POSITION RANGE FOR THE SINGLE VEE GROOVE WEL

Groove Test Position	Welding Positions Qualified For	
Groove Position	Groove Positions	Fillet Positions
FLAT 1G	F	F, H
HORIZONTAL 2G	F, H	F, H
VERTICAL 3G	F, H, V	F, H, V
OVERHEAD 4G	F, H, OH	F, H, OH
3G AND 4G	ALL	ALL

NOTE: Also Qualifies for pipe over 24" diameter.
if backing is used qualification is with backing

Welder Qualification

- Typically, a base material and filler metal combination using a given process is used to qualify a weld procedure for that specific material combination and process.
- Base material and filler metal combinations for welder qualifications are broader.
- Generally, welders qualify based on:
  - Filler material
  - Thickness
  - Process
  - Position

Welder Qualification

- Bend tests or volumetric examines are performed on welder qualification test assemblies
- Generally, a groove weld qualifies also qualifies fillet welds.
- However, this does not mean welders are not trained for different joint types, material combinations (welder qual. tests may use only one type of material), access, etc.

Issues With Welding

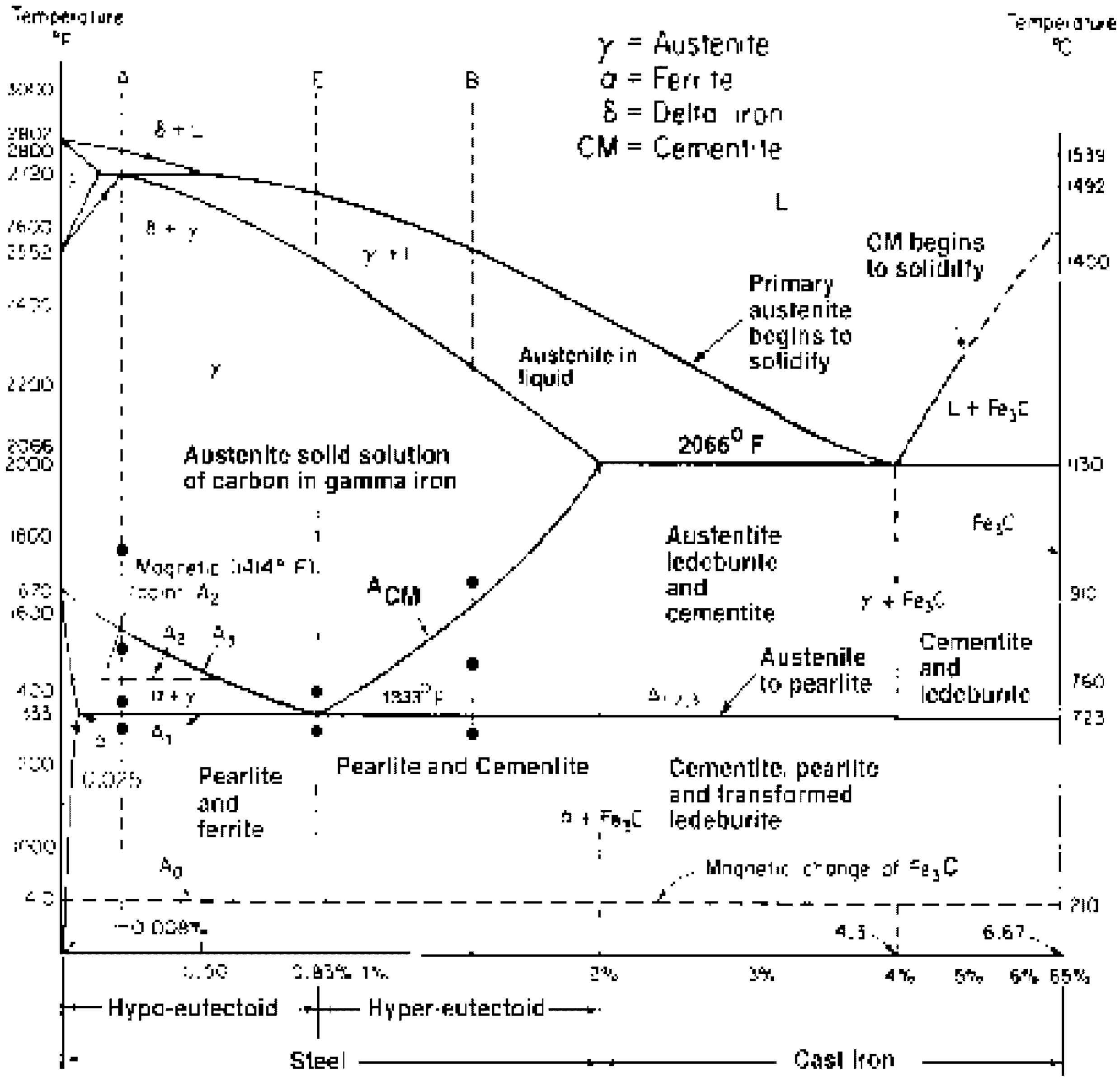
- Changes in microstructure
- Changes in material properties
- Introduce defects
- Introduce stresses

Issues With Welding-Steels

- Steels are affected by welding causing
  - Hydrogen cracking
  - Brittle/hard heat affected zones (i.e., Martensite)
  - Increase stresses (residual) during cooling
- Methods to minimize these effects (i.e., tempered martensite):
  - Preheating base metal before welding minimizes these effects
  - Post weld heat treatment also minimizes these effects (i.e., tempered martensite).
  - Temper bead welding

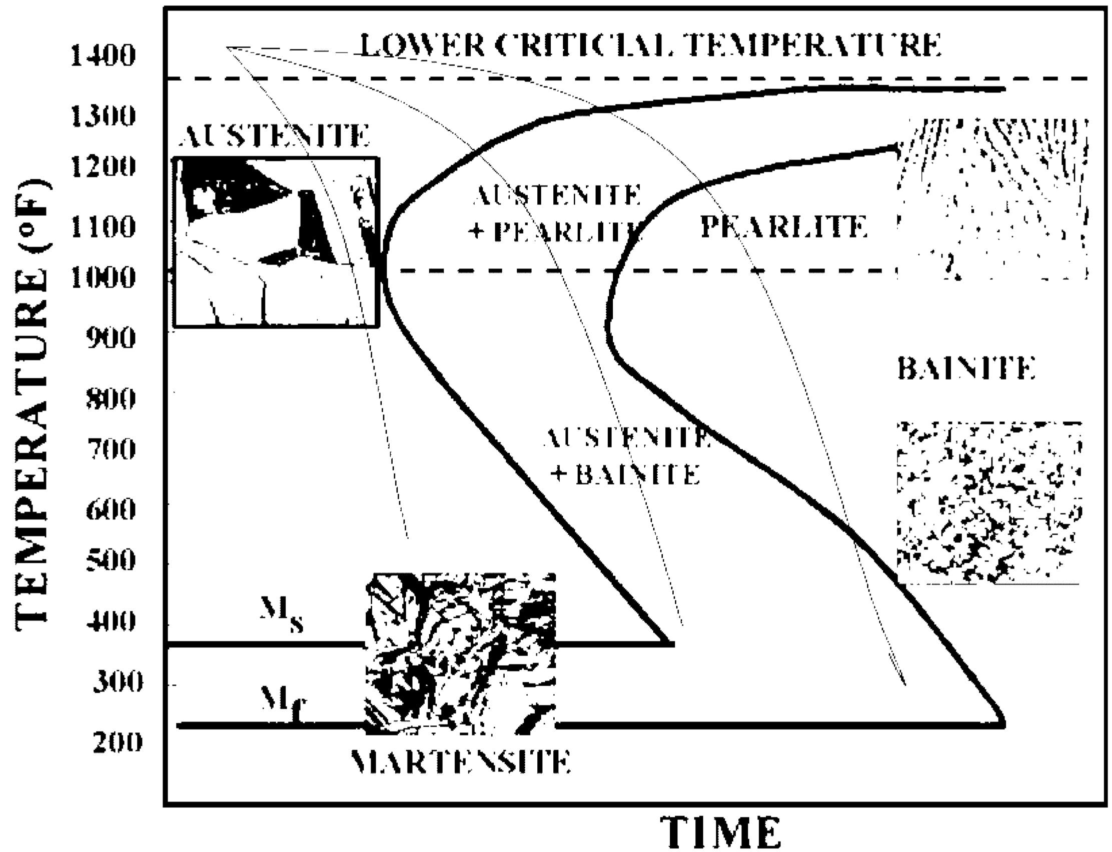
Issues With Welding-Steels

- Depending on composition and heat treatment can change material microstructure/phase.
- Change material properties



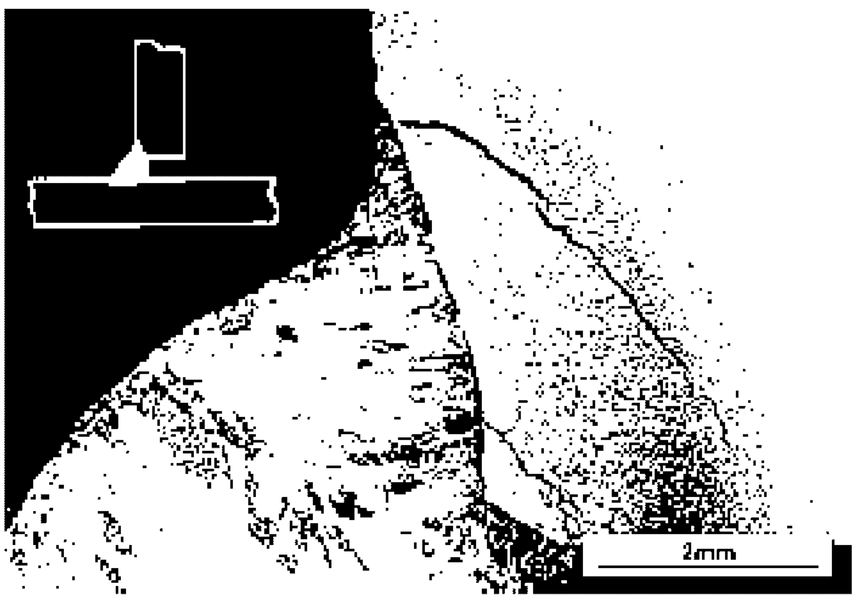
Issues With Welding-Steels

- Different microstructure affects toughness
- Verify toughness of welds/adjacent base metal by Charpy impact tests

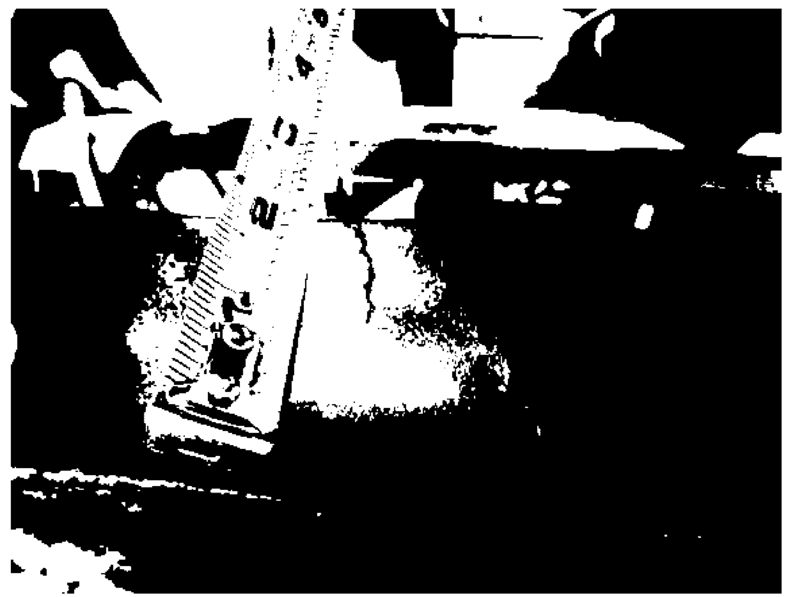


Issues With Welding-Steels

Underbead cracking/hydrogen cracking

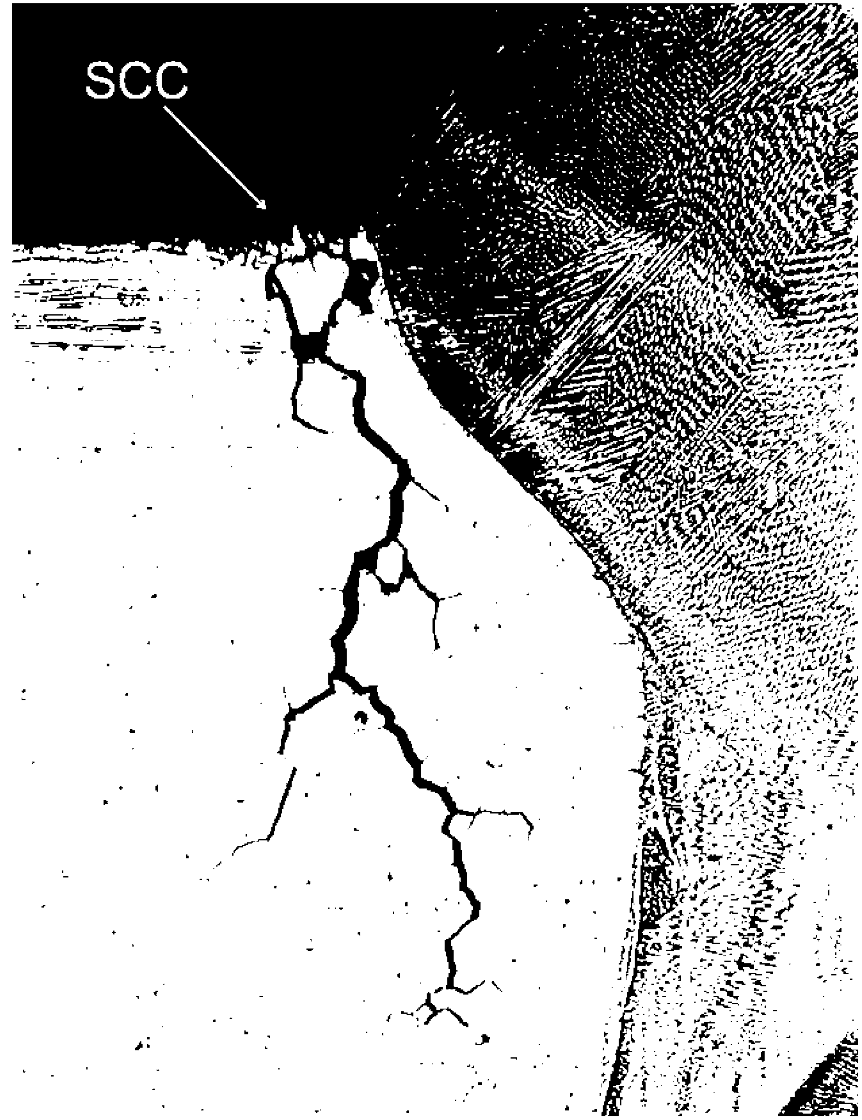


Stress induced/improper post weld heat treatment



Issues with Welding-SS

- Sensitization (800°F to 1500°F.
- Cr carbides precipitate out depleting the grain boundary areas of Cr making SS susceptible to SCC.
- Methods to minimize sensitization (RG 1.44):
  - Material composition(low carbon)
  - Heat treatment
  - Weld heat input
  - Interpass temperature

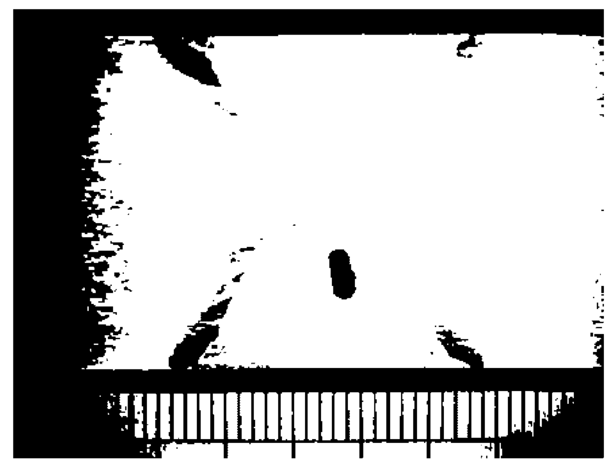


Issues With Welding-Defects

- Welding defects affect the integrity of the weld and mechanical properties.
- Some defects (depending on size) are acceptable, while others are never acceptable.

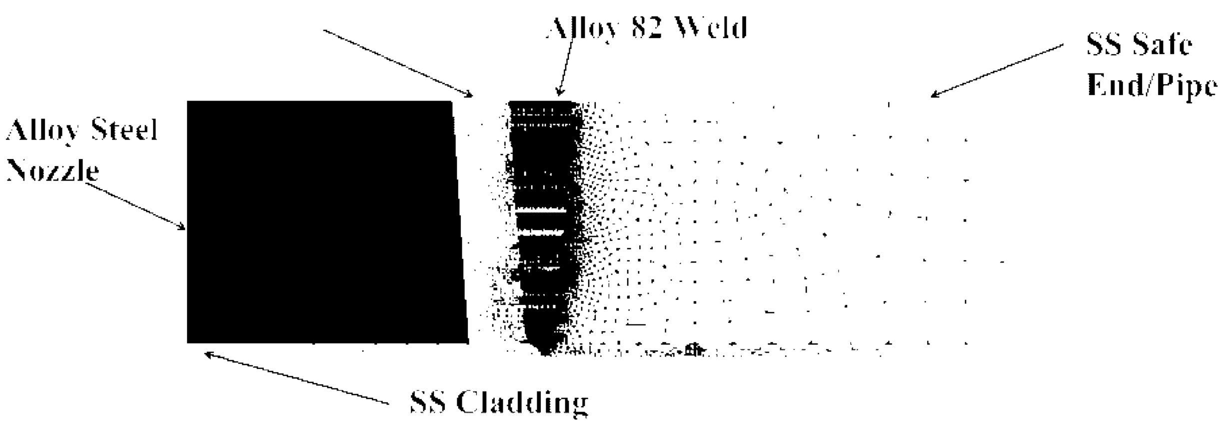
(a)

(b)



Issues With Welding-Dissimilar Metal Welds

- Design use different materials for various components (due to conditions/environment), and eventually these components need to be connected.
- Welding different material can create totally new alloys/microstructure (dilution) and affect material properties or resistance to aging degradation.



Issues With Welding-Dissimilar Metal Welds

- Develop materials with similar properties (strength, thermal expansion, etc.) and still be weldable.
- Some alloys can be welded only in a particular way
  - Weld Ni based alloy to stainless steel, but can not weld stainless steel onto Ni based alloy.
- New degradation mechanism may occur
  - Primary stress corrosion cracking (PWSCC)
    - Ni based alloy (600) in PWR water previously was thought immune to SCC
- Need to develop new alloys and how to weld them.

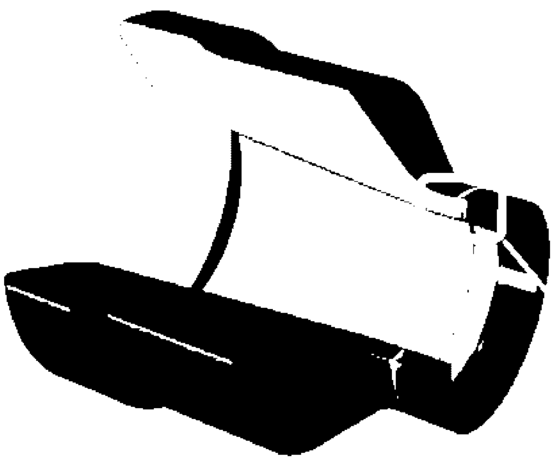
Issues with Welding – Crack Growth

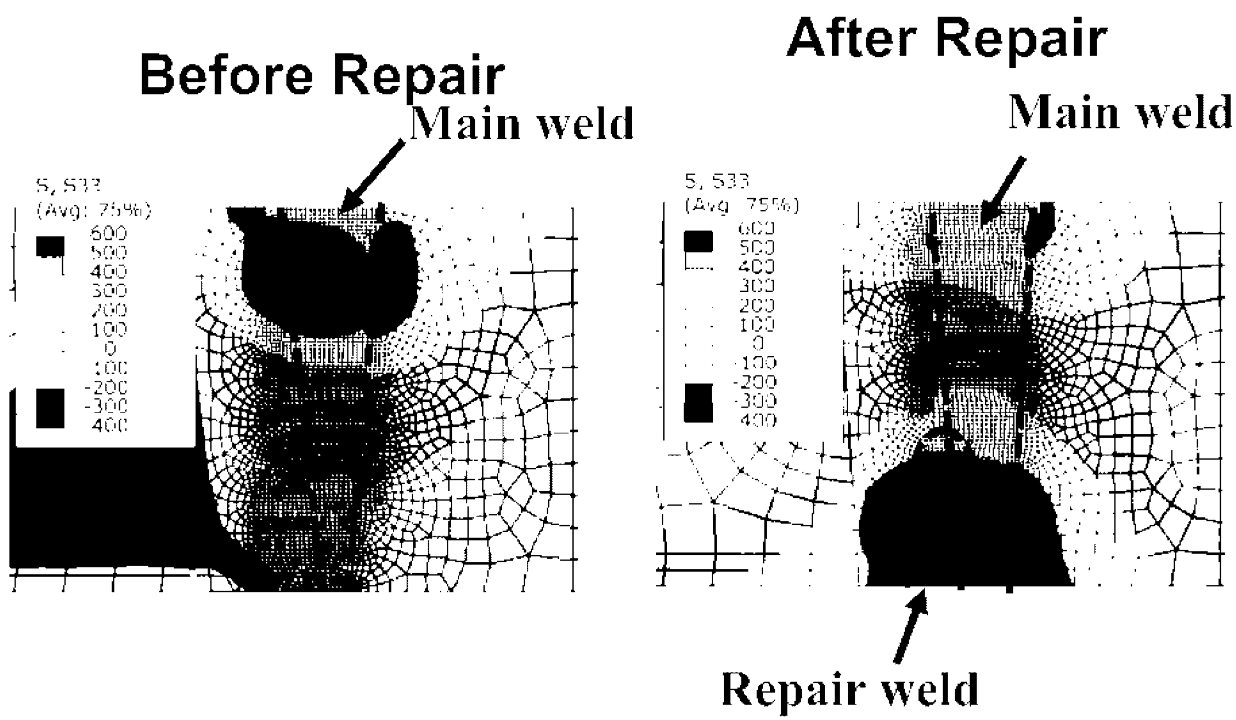
- New alloys or welding processes
- Learn from past experiences
- Tests to simulate aging degradation in evaluating long term integrity

Issues with Welding - Residual Stresses

•**Stresses can lead to long term issues/aging degradation (i.e., SCC)**

Hoop stresses before and after repair weld





Summary

- Welding improves productivity but can change material properties or microstructure of material being welded.
- Need balance of productivity, cost and level of quality.
- Weld procedures are qualified by the fabricator to specific codes (each code may have different requirements).
- Welders are trained and qualified to the specific codes.
- Codes used to qualify weld procedures and welder; does not provide instructions on how to weld.
- Issues with welding including long term degradation have to be taken into account.

Application of Lessons Learned from Flow-Induced Vibration to New Reactors

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Office of New Reactors
U.S. Nuclear Regulatory Commission

May 18, 2010

Introduction

- Some operating nuclear power plants have experienced failure of safety-related and non-safety related components from flow-induced vibration (FIV).
- FIV significance caused by acoustic resonance was not recognized during previous Design Certifications.
- Lessons learned from FIV operating experience is currently being considered during design, qualification, and surveillance planning for new reactors.
- NRC staff reviewing potential FIV in Design Certification and Combined Operating License (COL) applications.

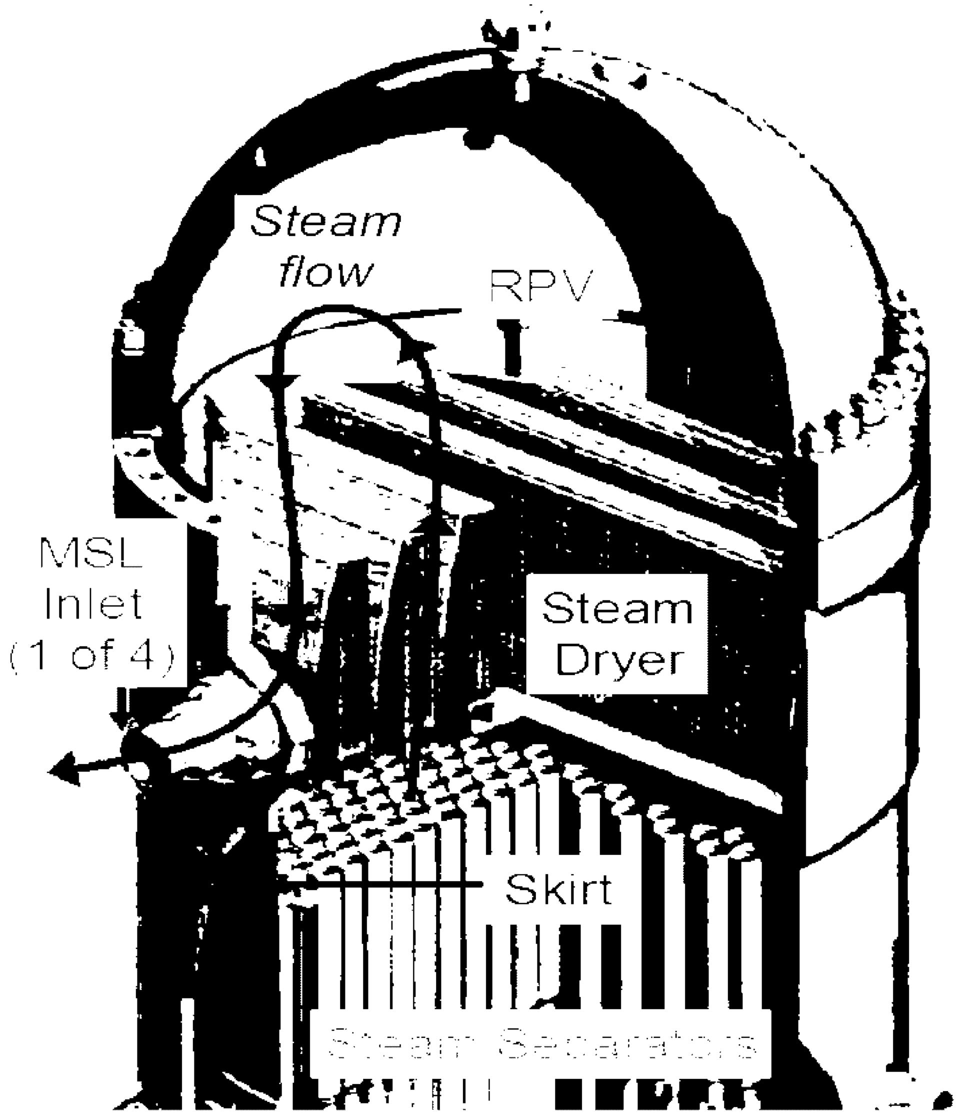
FIV Safety Significance

- Severe hydrodynamic and acoustic resonance loads can cause failure of safety-related and non-safety related components in reactor, steam, and feedwater systems.
- Failure of safety-related or non-safety related components might cause sudden reactor transient.
- Inoperability of safety-related components (such as safety relief valves) might not be revealed until component is signaled to perform its safety function.
- Failure of non-safety related components can cause small pieces to interfere with plant operation, or safe shutdown.
- High steam moisture content from steam dryer failure could damage reactor turbine if shutdown not initiated.

FIV Operating Experience

- Quad Cities Unit 2: Steam Dryer (June 2002/June 2003)
- Quad Cities Unit 1: Steam Dryer (November 2003)
- Quad Cities Unit 2: Steam Dryer (March 2004)
- Quad Cities Units 1 and 2: Relief Valves (January 2006)
- Palo Verde Unit 1: Shutdown Cooling Piping (Dec. 2005)
- Waterford Unit 3: Steam Generator Internals (April 2005)

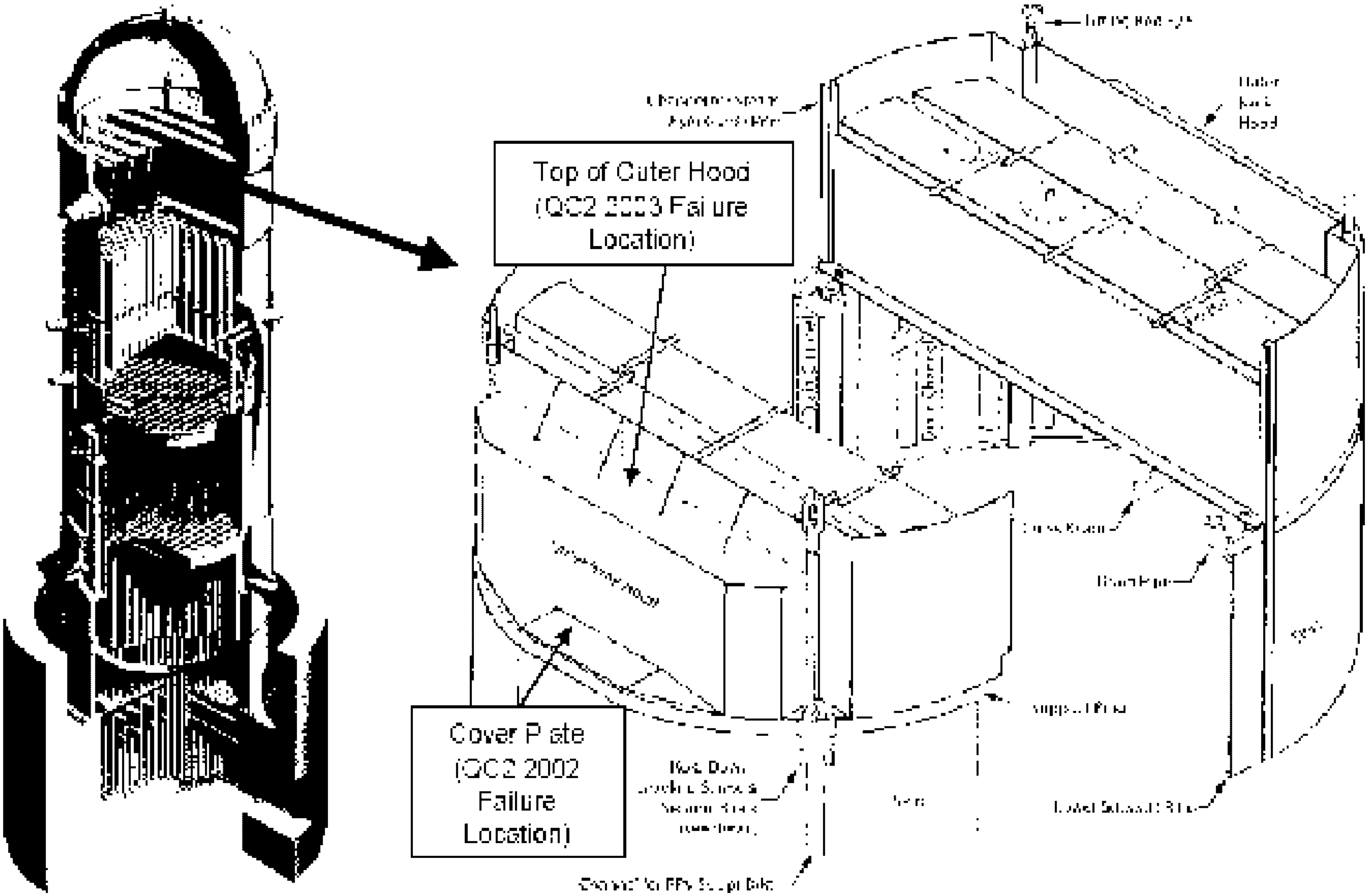
Boiling Water Reactor Pressure Vessel



Quad Cities Unit 2 Power Uprate

- Quad Cities Unit 2 – June 2002
  - After 90 days of Extended Power Uprate (EPU) operation, steam dryer cover plate fails with pieces found on steam separators and in main steam line (MSL).
- Quad Cities Unit 2 – June 2003
  - After additional 300 days of EPU operation, steam dryer experiences failure of hood, internal braces, and tie bars.

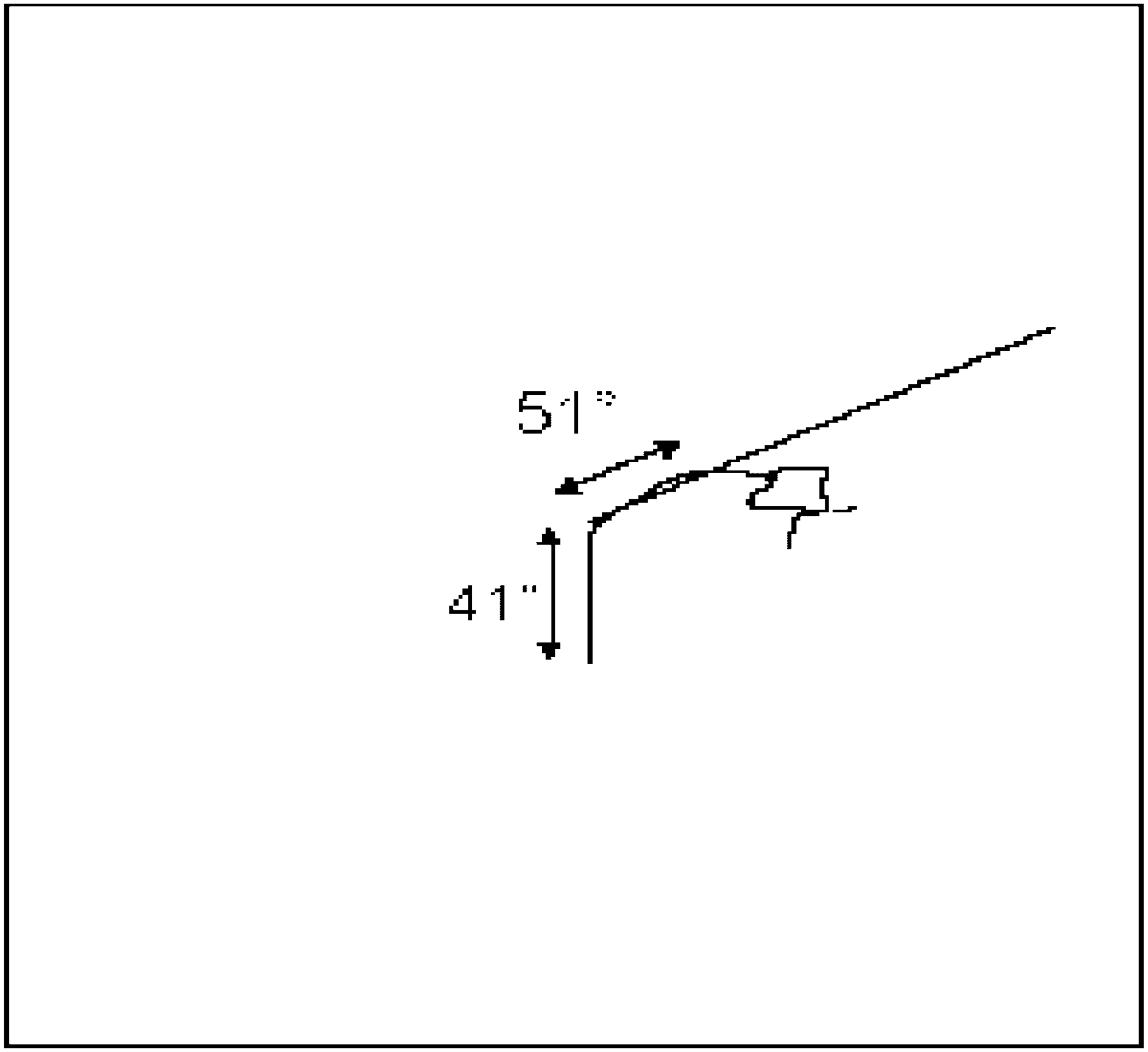
QC2 Steam Dryer Failures
2002 and 2003



Quad Cities Unit 1 Power Uprate

- Quad Cities Unit 1 – November 2003:
  - After about 1 year of EPU operation, steam dryer hood experiences significant cracking with 6x9 inch piece of outer bank vertical plate missing.
  - Damage also found to
    - Main steam electromatic relief valve (ERV)
    - Steam line supports, and
    - High Pressure Coolant Injection (HPCI) steam supply motor-operated valve.

QC1 Steam Dryer Failure
November 2003



270° Side

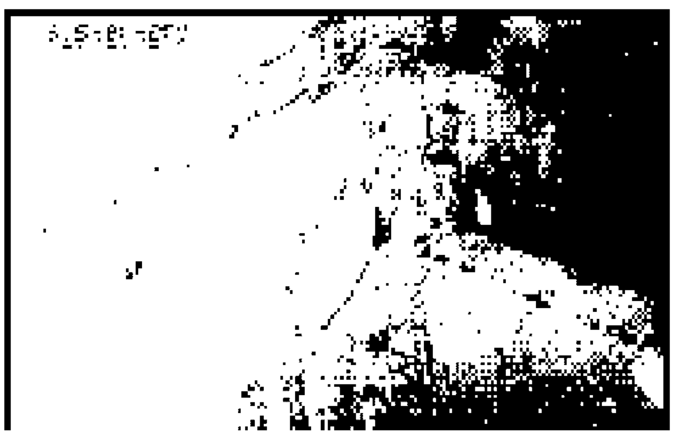
QC1 Steam Dryer Failure
November 2003

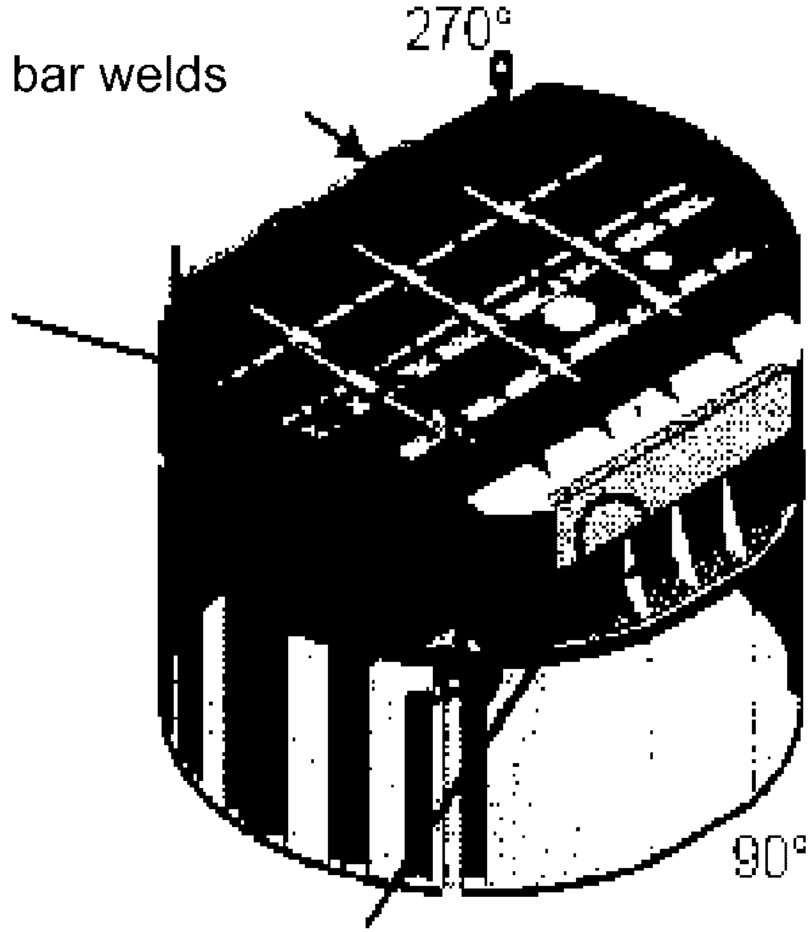
Outer bank vertical plate 6x9 inch hole

Quad Cities Unit 2 Power Uprate

- Quad Cities Unit 2 – March 2004
  - After about 8 months of EPU operation, numerous steam dryer indications identified during refueling outage inspection including
    - Cracking near gussets installed in 2003,
    - Broken tie bar welds, and
    - Damaged stiffener plate weld.

QC2 Steam Dryer Failure
March 2004

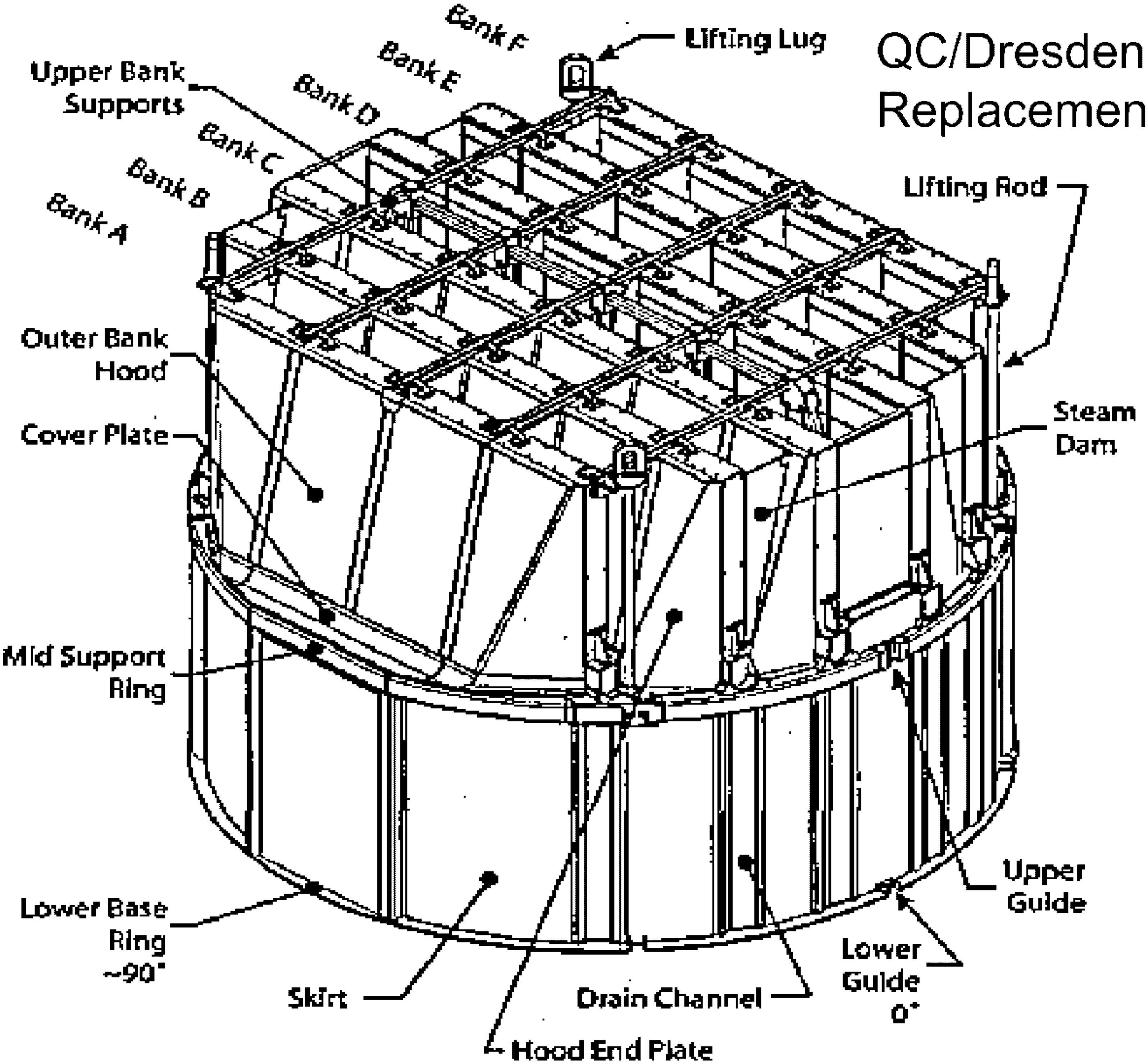


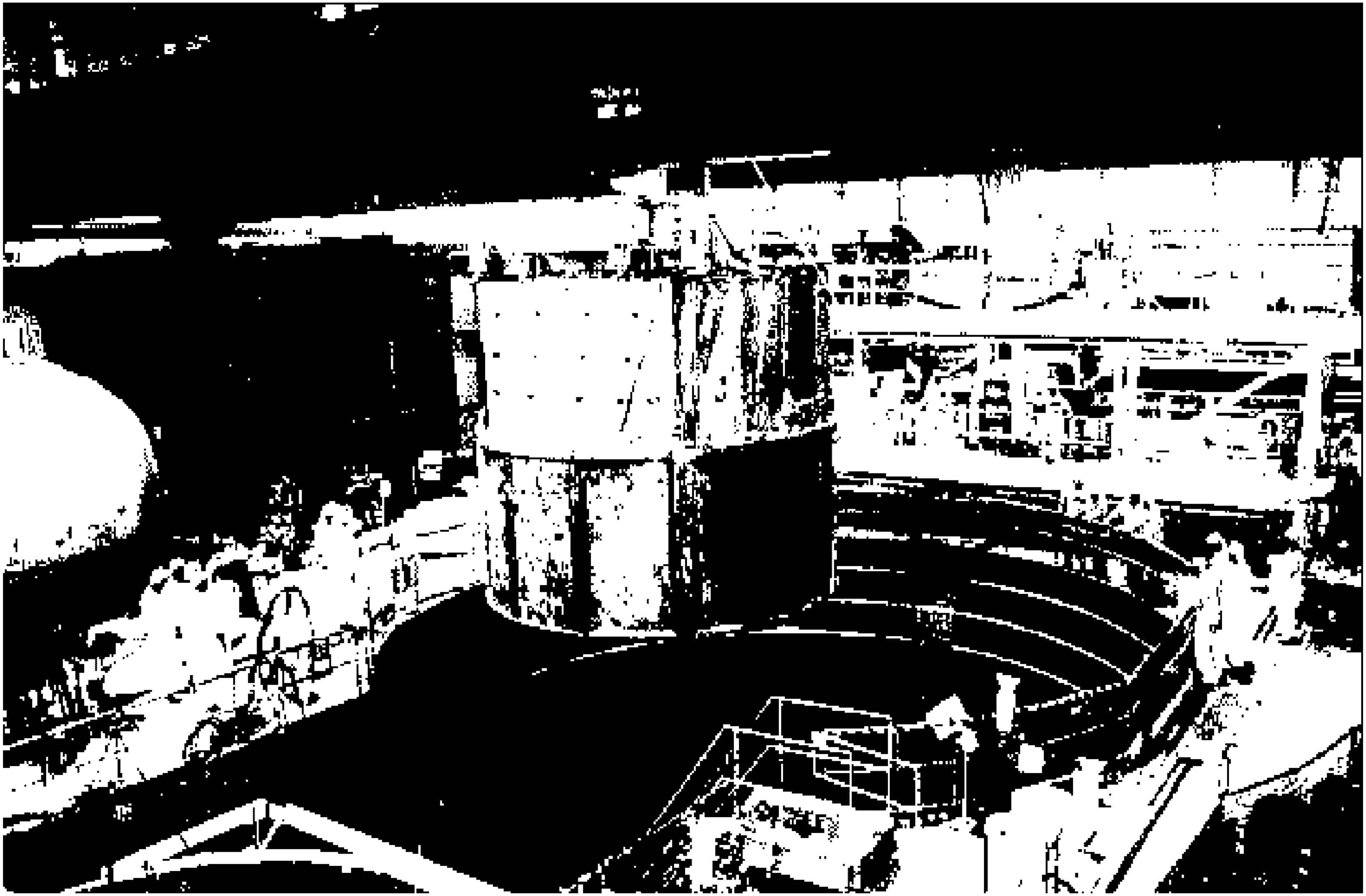


Quad Cities and Dresden Steam Dryer Replacement

- Replacement steam dryers for Quad Cities and Dresden are stronger and more streamlined than their previous dryers.
- Exelon replaced Quad Cities steam dryers in Spring 2005.
- Exelon installed pressure, strain, and acceleration instrumentation directly on Unit 2 steam dryer to determine steam dryer loading and to calibrate acoustic circuit analysis.
- Exelon installed strain gages on main steam lines to measure pressure fluctuations as input to acoustic circuit analysis to calculate steam dryer loading for QC1 and other plants.
- Exelon replaced steam dryer in Dresden Unit 3 in 2006, and Unit 2 in 2007.

QC/Dresden Replacement Dryer

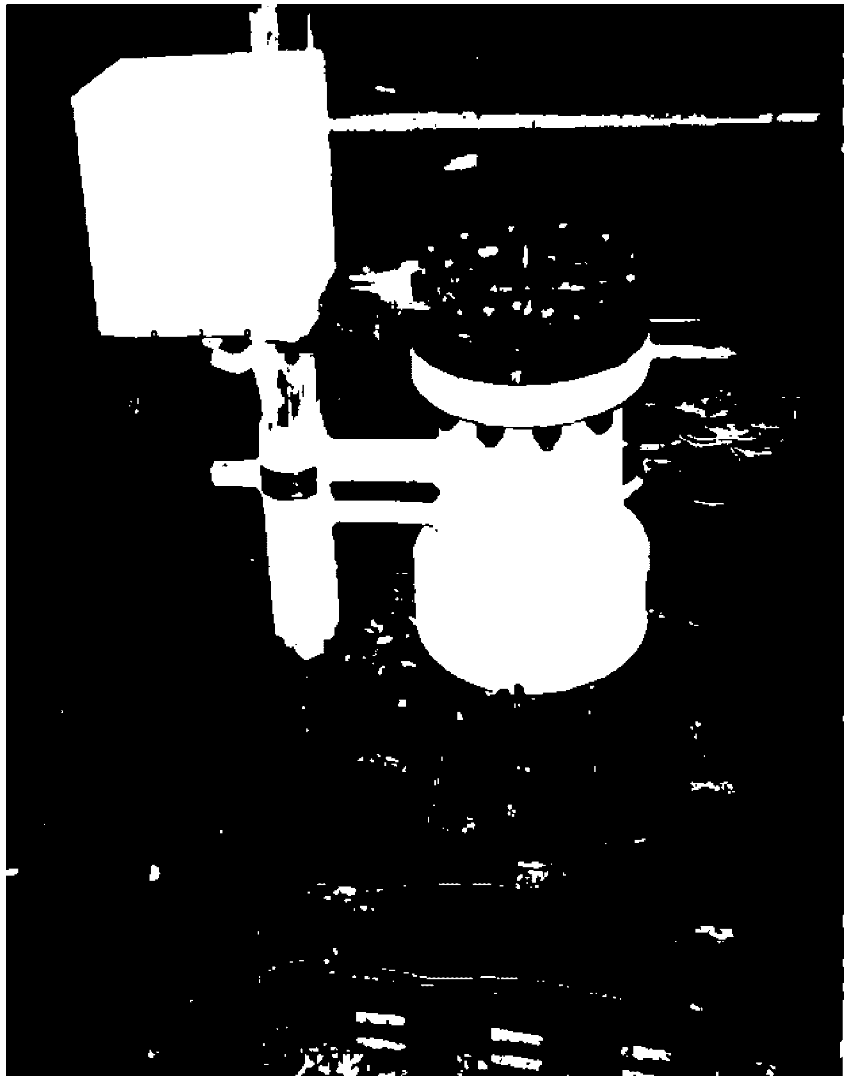




ERV Damage at Quad Cities

- In late 2005, Exelon identified intermittent short circuiting of safety-related Electromatic Relief Valve (ERV) at Quad Cities Unit 2.
- Exelon reduced power in QC2 to inspect ERV actuator and found broken internal parts.
- Exelon shut down QC2 and found damage to other ERVs, and performed repairs.
- Exelon shut down QC1 in early January 2006 and found damage to its ERVs, and performed repairs.
- If ERV short circuiting had not occurred, both Quad Cities units might have operated with multiple ERVs inoperable or experienced spurious ERV opening.

Electromatic Relief Valve



Palo Verde Unit 1 FIV

- Long history of vibration and leakage of shutdown cooling (SDC) valve and piping at Palo Verde Unit 1.
- Vibration cause hypothesized as pressure pulsations in suction line from coupling between fundamental frequency of SDC suction line and vortex shedding due to RCS flow over SDC suction line.
- Licensee initially attempted an SDC nozzle modification but obtained unacceptable results.
- Licensee subsequently relocated SDC valve to increase acoustic frequency away from vortex shedding modes.
- Acceptable vibration results obtained.

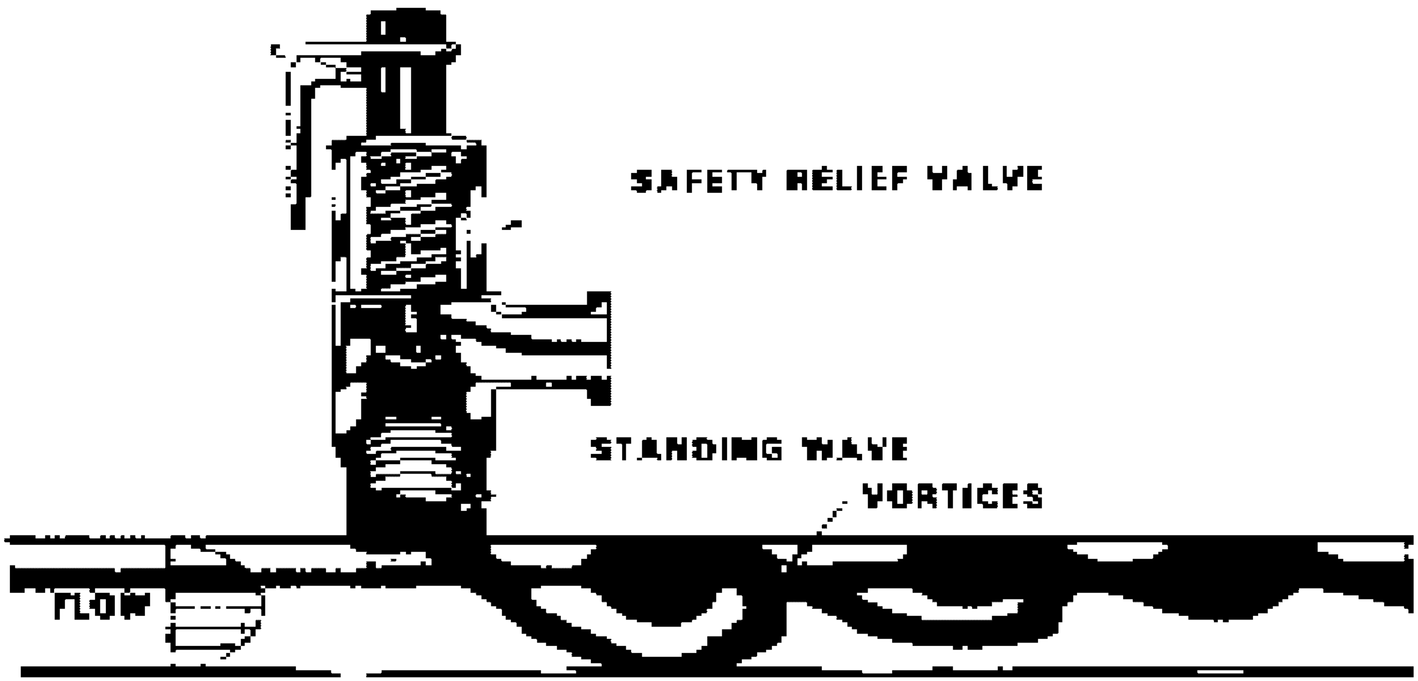
Waterford Unit 3 FIV

- Failure of batwing supports in Steam Generator (SG) #2 internals found during inspection in April 2005.
- Most probable cause determined to be fatigue due to FIV.
- Nov. 2006 inspection found additional batwing damage in SG #2, but no damage in SG #1.
- Batwing weld repairs performed and selected SG tubes plugged to mitigate potential impact of SG tube vibration.

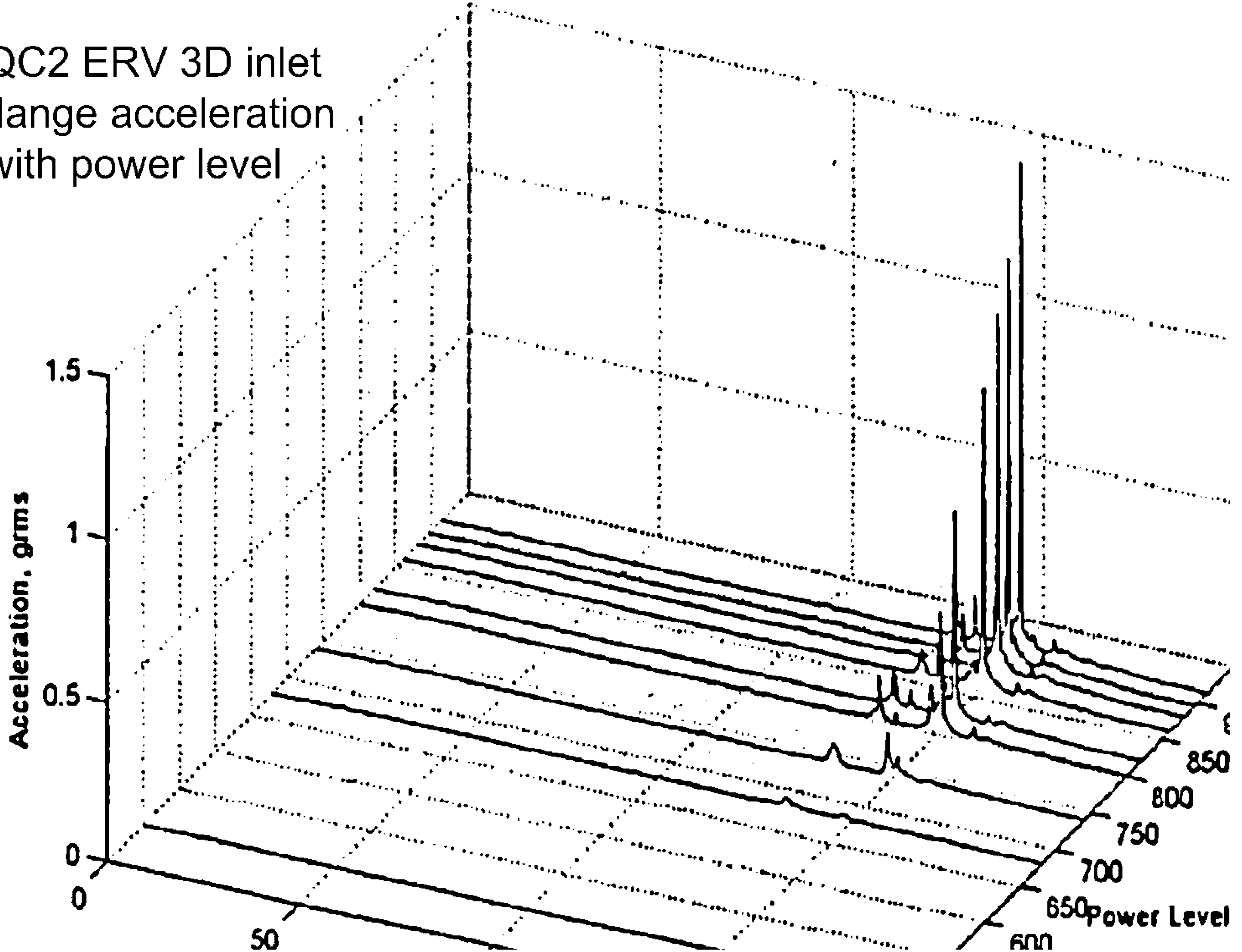
Acoustic Resonance FIV Cause

- MSL flow creates vortices when passing over branch lines.
- At specific flow velocities, vortices couple with acoustic mode of branch lines.
- Pressure fluctuations in MSLs can cause significant pressure loading on steam dryer.
- Severe vibration can occur in MSL piping and components, including relief valves.
- Acoustic resonance is difficult to predict and quantify prior to its occurrence.

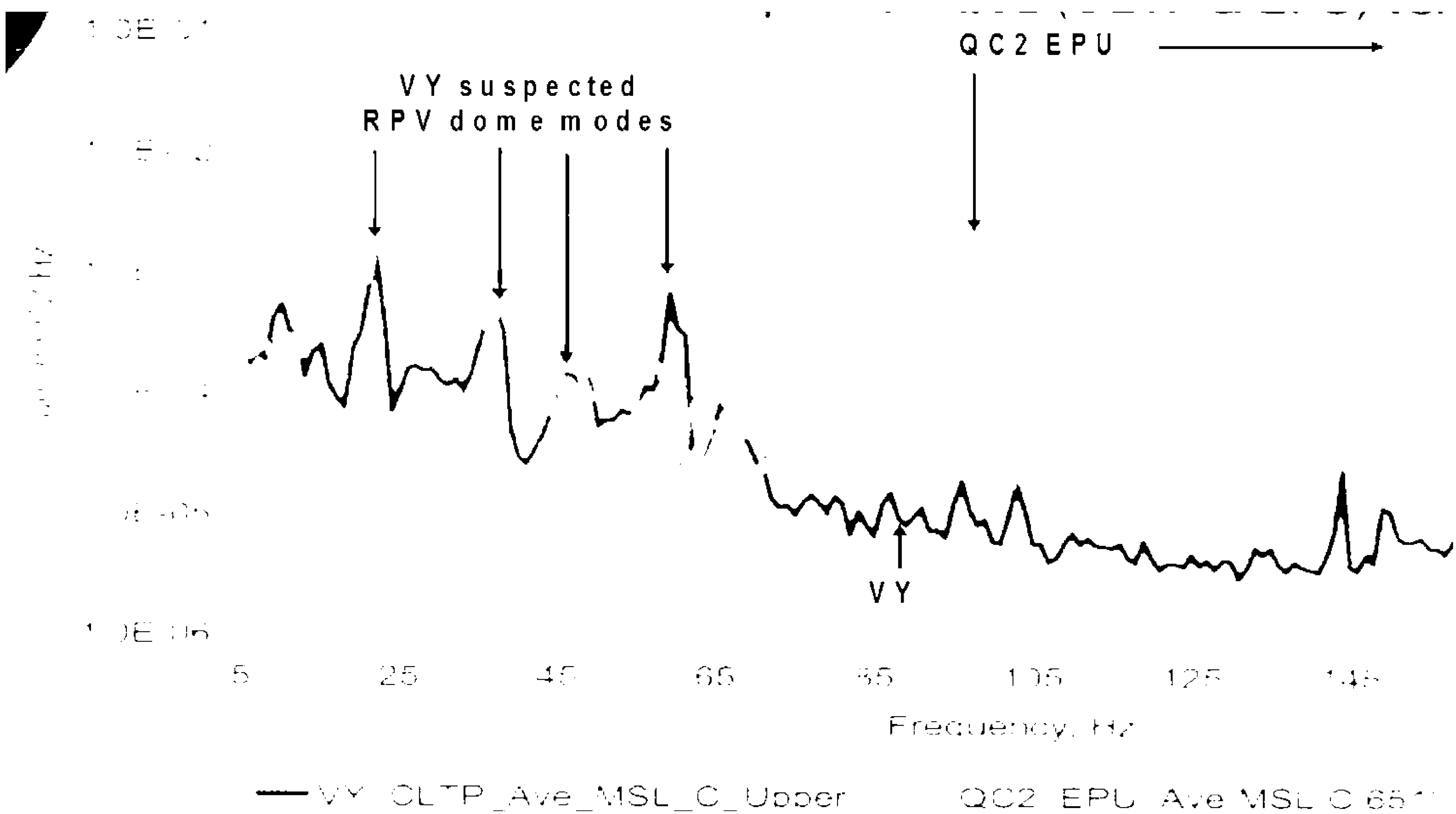
Singing Safety-Relief Valve



QC2 ERV 3D inlet flange acceleration with power level



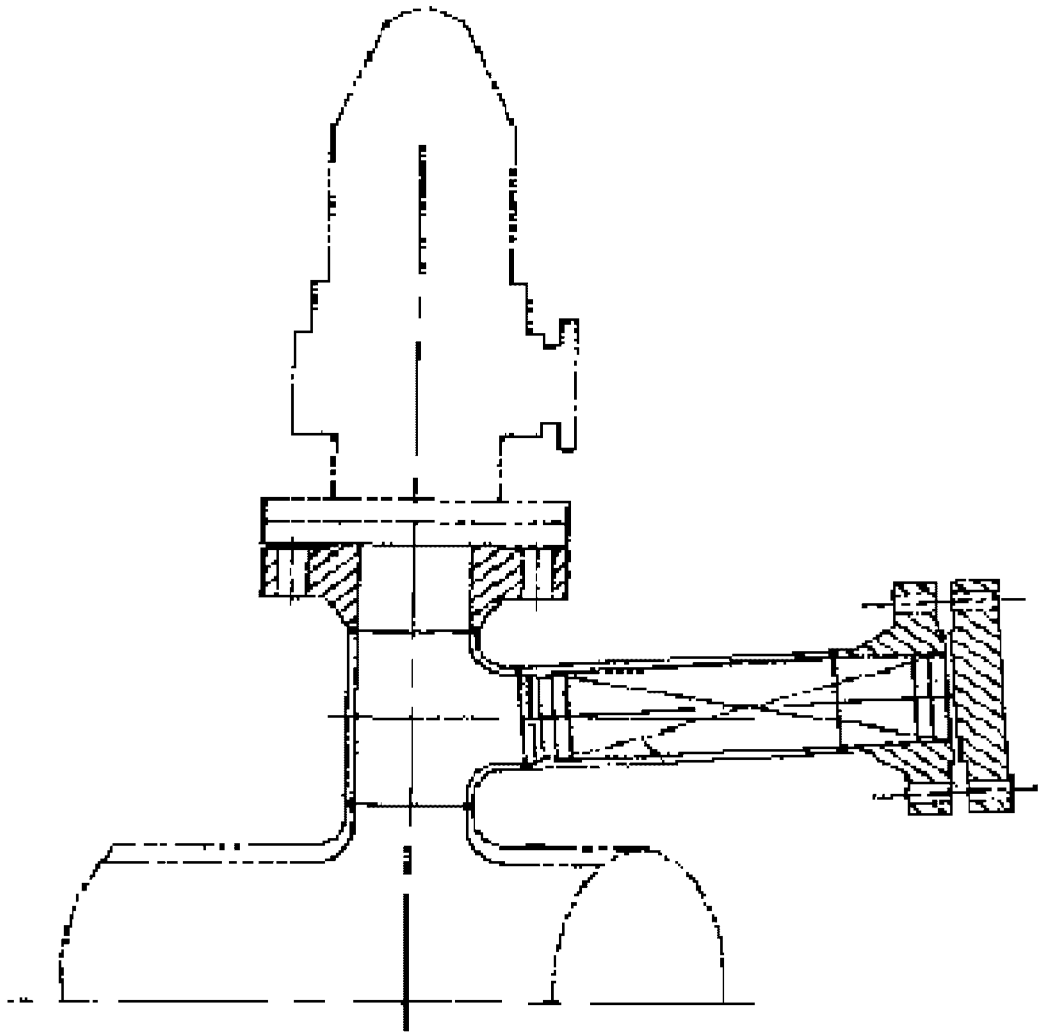
MSL Strain Gage Readings
for Quad Cities and Vermont Yankee



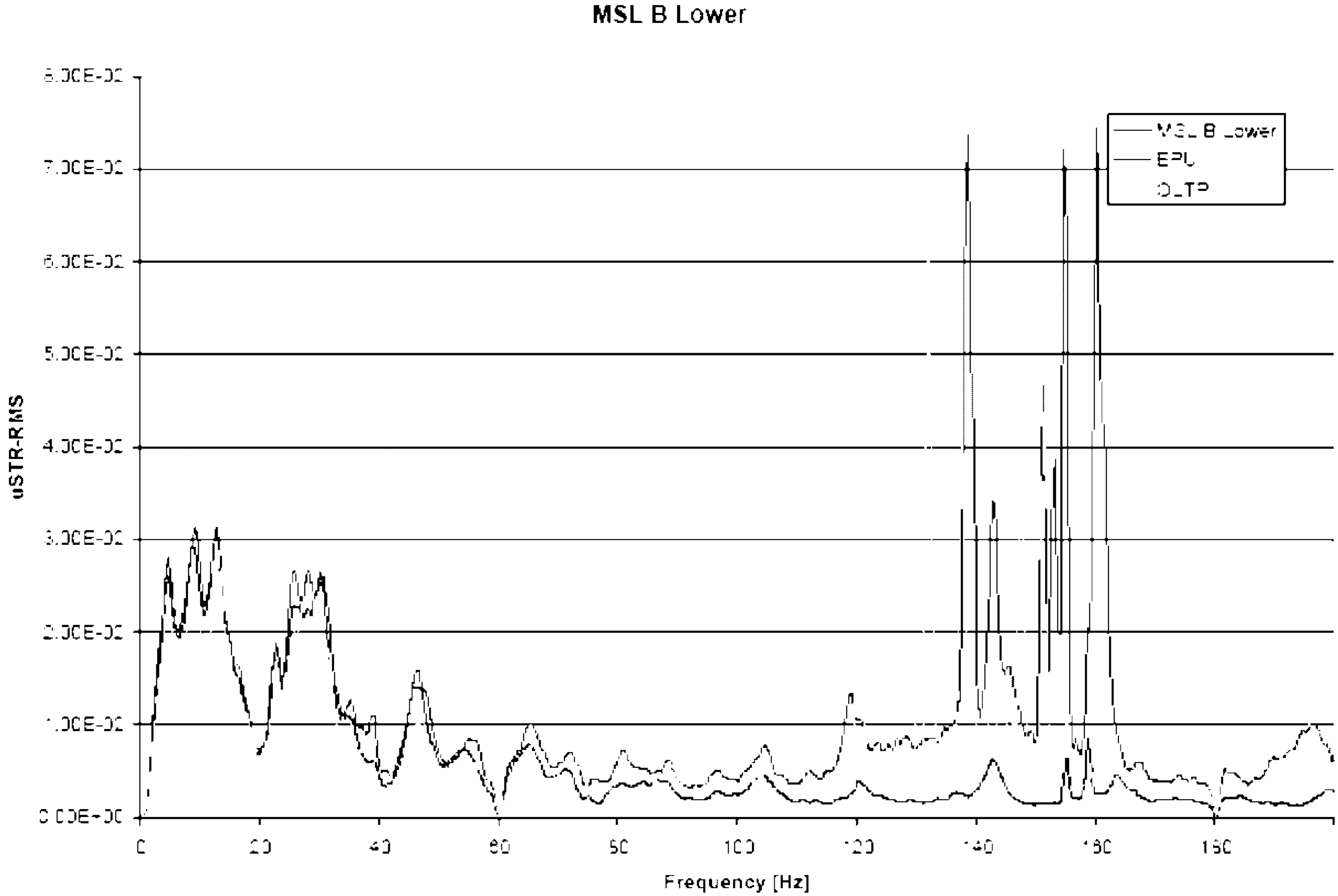
Quad Cities Modifications

- In spring 2006, Exelon modified branch lines to 8 main steam safety valves and 4 ERVs in each QC unit by installing Acoustic Side Branch (ASB) consisting of 6-inch diameter pipe filled with screen mesh.
- ASB increases effective length of branch line that decreases frequency of acoustic standing wave, and lowers steam velocity at which vortex shedding will excite acoustic standing wave.
- ASB screen mesh dampens pressure fluctuation.
- MSL strain gate data collected after modification reveal pressure fluctuations and vibrations reduced to pre-EPU levels.

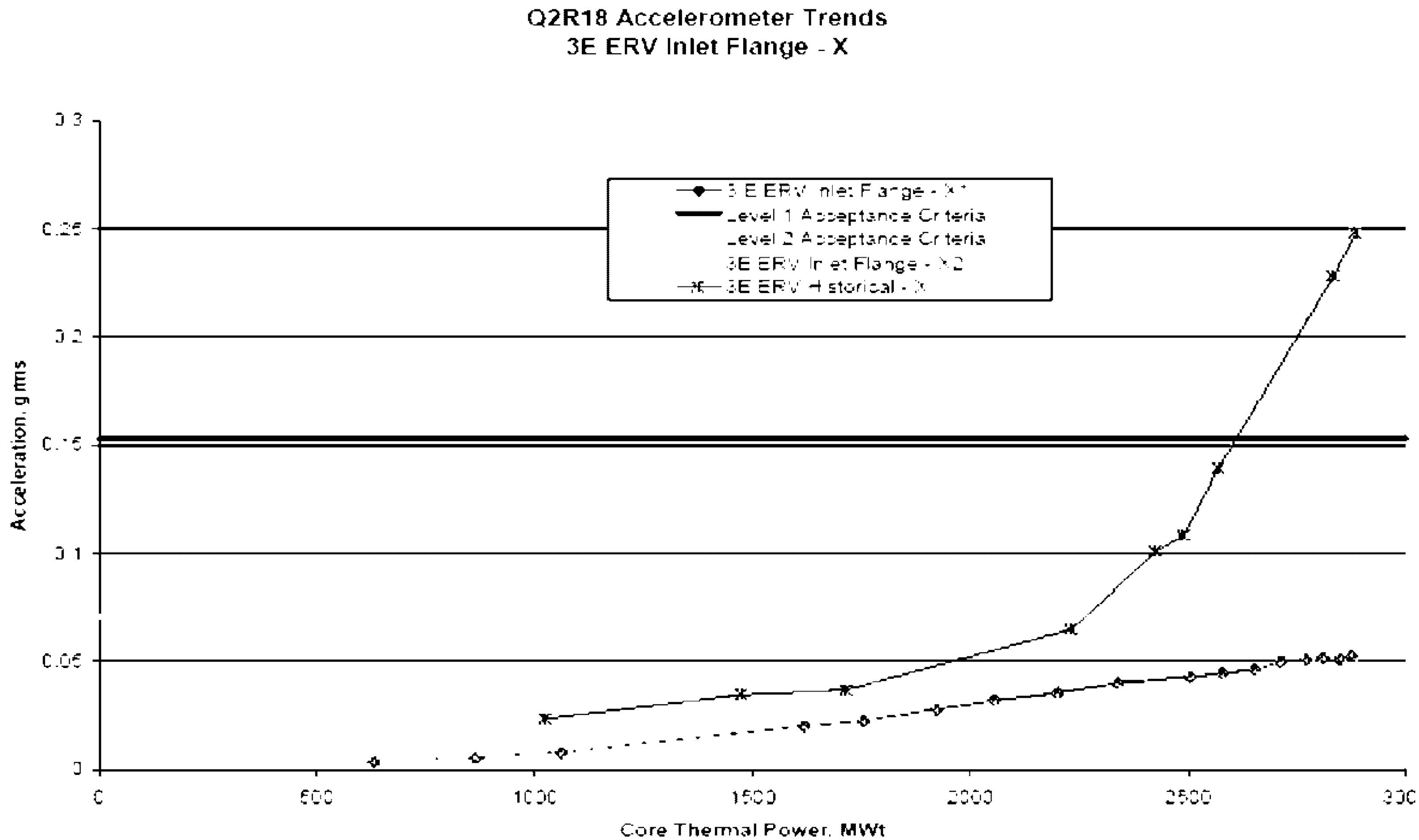
ASB Modification



QC2 Spring 2006 MSL B Strain Gage Data compared to pre-ASB Data



QC2 Spring 2006 3E ERV Accelerometer Data compared to pre-ASB Data



Industry Response

- Scale model testing and acoustic analysis methodology developed to evaluate acoustic resonance in MSLs.
- GE updated its steam dryer inspection guidance (SIL 644).
- BWR Vessel Internals Project prepared generic guidance:
  - BWRVIP-139 on steam dryer inspections
  - BWRVIP-182 on demonstrating steam dryer integrity
  - BWRVIP-194 on methodologies for demonstrating dryer integrity
- BWR Owners Group prepared lessons learned report on power uprates.
- GEH developing Plant Based Load Evaluation (PBLE) Methodology for evaluation of acoustic load on ESBWR steam dryers.

Operating Plant FIV Status

- Vermont Yankee Power Uprate (March 2006): MSL data collected during power uprate ascension without significant FIV. Inspected dryer following power uprate and found Intergranular Stress Corrosion Cracking (IGSCC).
- Susquehanna 1 and 2 Power Uprate (Jan. 2008): Steam dryers replaced with upgraded design. Dryer data collected without significant FIV. Unit 1 dryer inspection found IGSCC after 2-year power uprate operation.
- Hope Creek Power Uprate (May 2008): Steam dryer upgraded prior to initial startup. MSL data collected without significant FIV.
- Browns Ferry, Monticello, and NMP2 Power Uprates: under review

NRC Staff Response

- Information Notices 2002-26 (S1 and 2) and 2004-06 on Quad Cities and Dresden FIV events.
- Evaluation of Exelon activities on QC and Dresden FIV events including plant inspections and observation of replacement steam dryers, MSL modifications, small scale testing, and EPU restart monitoring with support from Argonne National Laboratory, Penn State, and McMaster University.
- SRP Sections 3.9.2 and 3.9.5 and RG 1.20 updated to incorporate FIV lessons learned for BWRs and PWRs.
- Evaluation of generic industry FIV guidance.
- Power uprate safety evaluations with monitoring of power ascension for acoustic resonance and FIV.

Power Ascension Program

- License condition provides slow and deliberate power ascension with lengthy hold points and data evaluation.
- Monitor and trend data (e.g., pressure transducers, strain gages and accelerometers) hourly with 96-hour hold point for data evaluation and walkdown every 5% power when approaching full power.
- If data exceed limit curve, return to acceptable power level, re-evaluate dryer loads, re-establish limit curve, and perform assessment before continuing power ascension.
- Power ascension report to be submitted within 60 days.
- Conduct visual dryer inspection of all accessible, susceptible locations at first 3 RFOs, then long-term BWRVIP-139 plan.

NRO Activities

- Monitoring FIV operating experience for lessons learned applicable to new reactor review.
- Reviewing Design Certification applications for evaluation of potential adverse flow effects.
- Reviewing COL applications (e.g., STP Units 3 and 4) for consideration of potential FIV in design, testing, and monitoring programs.
- Assisting in planning ITAAC inspections for design, quality assurance, fabrication, and testing of plant components that can be adversely affected by FIV.

Summary

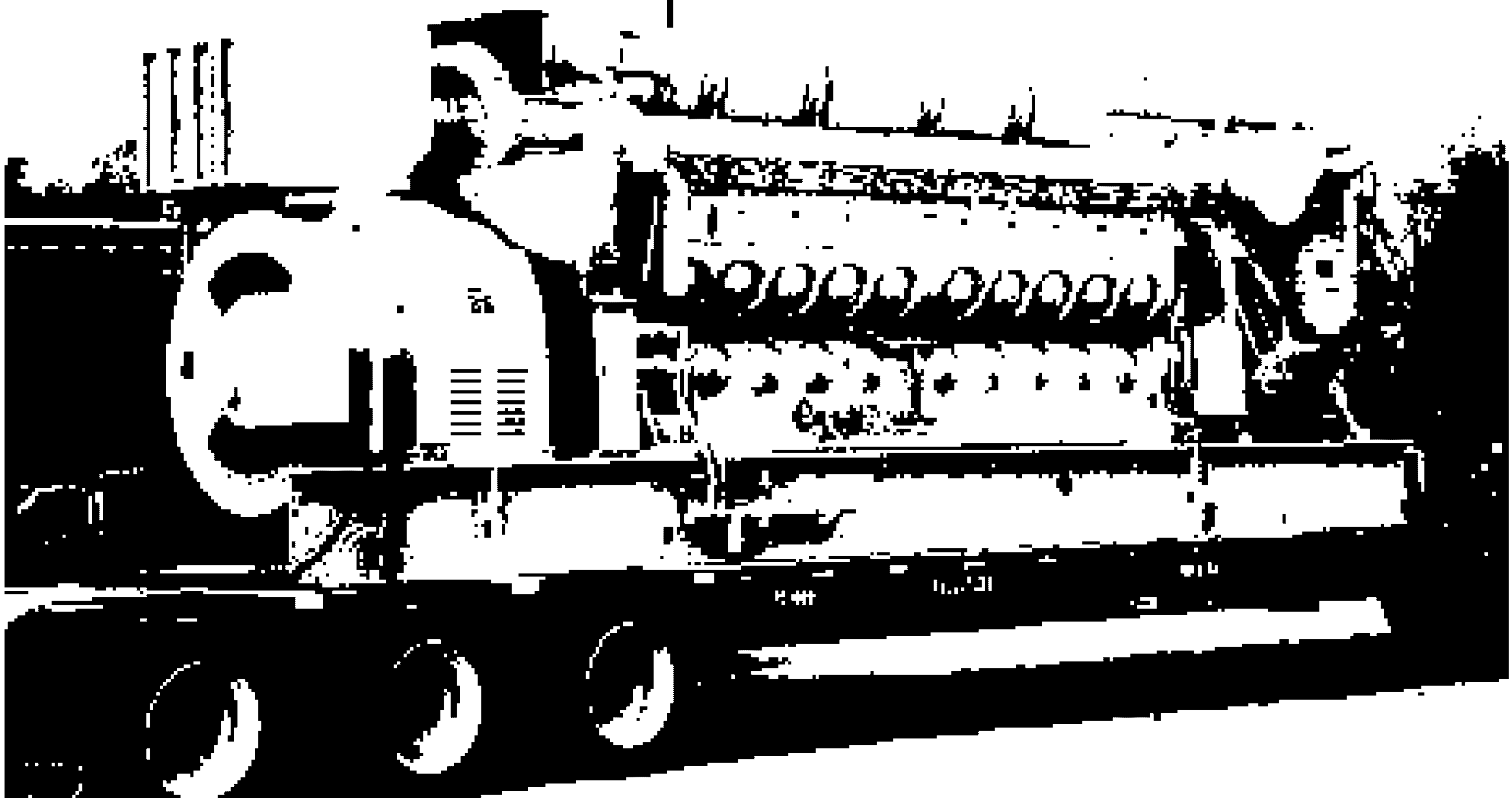
- Acoustic resonance in reactor and steam systems has led to failure of safety-related and non-safety related components at BWR and PWR nuclear power plants.
- Potential for severe hydrodynamic and acoustic resonance loads was not recognized during previous Design Certification reviews.
- Nuclear industry addressing FIV for new reactor designs, new reactor license applications, and operating reactors requesting power uprate.
- NRO staff evaluating FIV in Design Certification and COL applications to ensure lessons learned addressed in reactor design, testing, and monitoring.

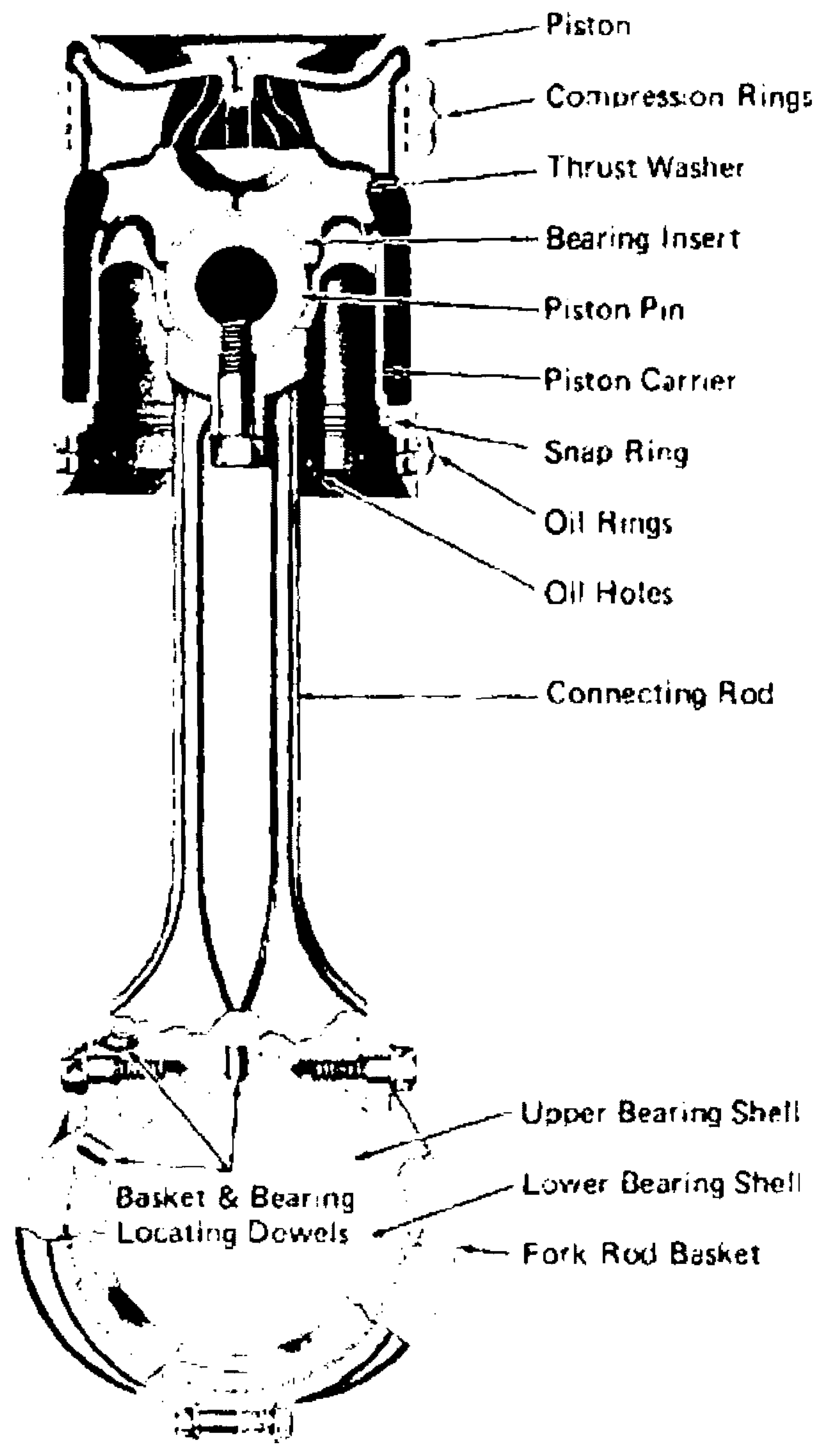
Surry EMD Diesel Failure, Notice of Enforcement Discretion, and Generic Implications

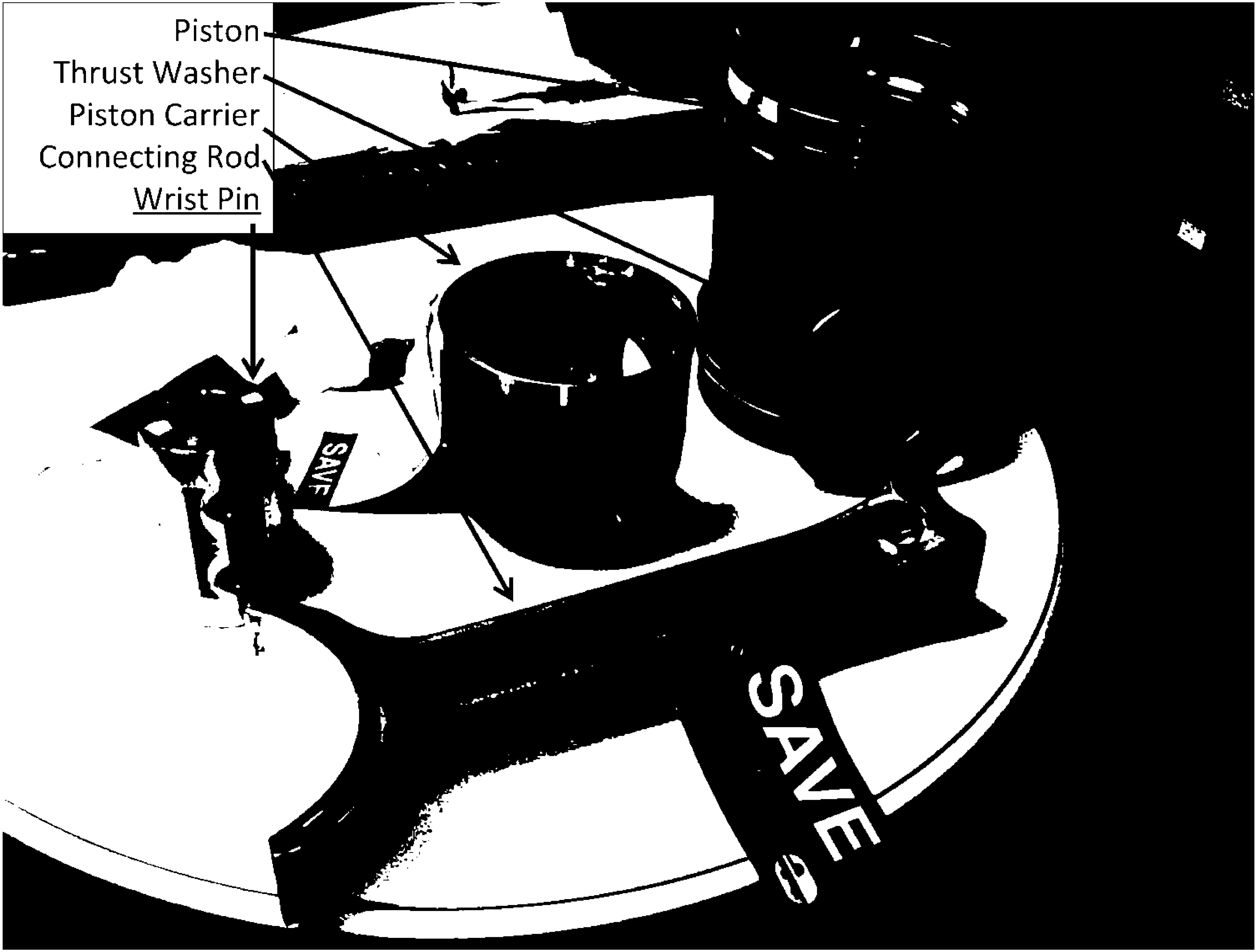
Contents

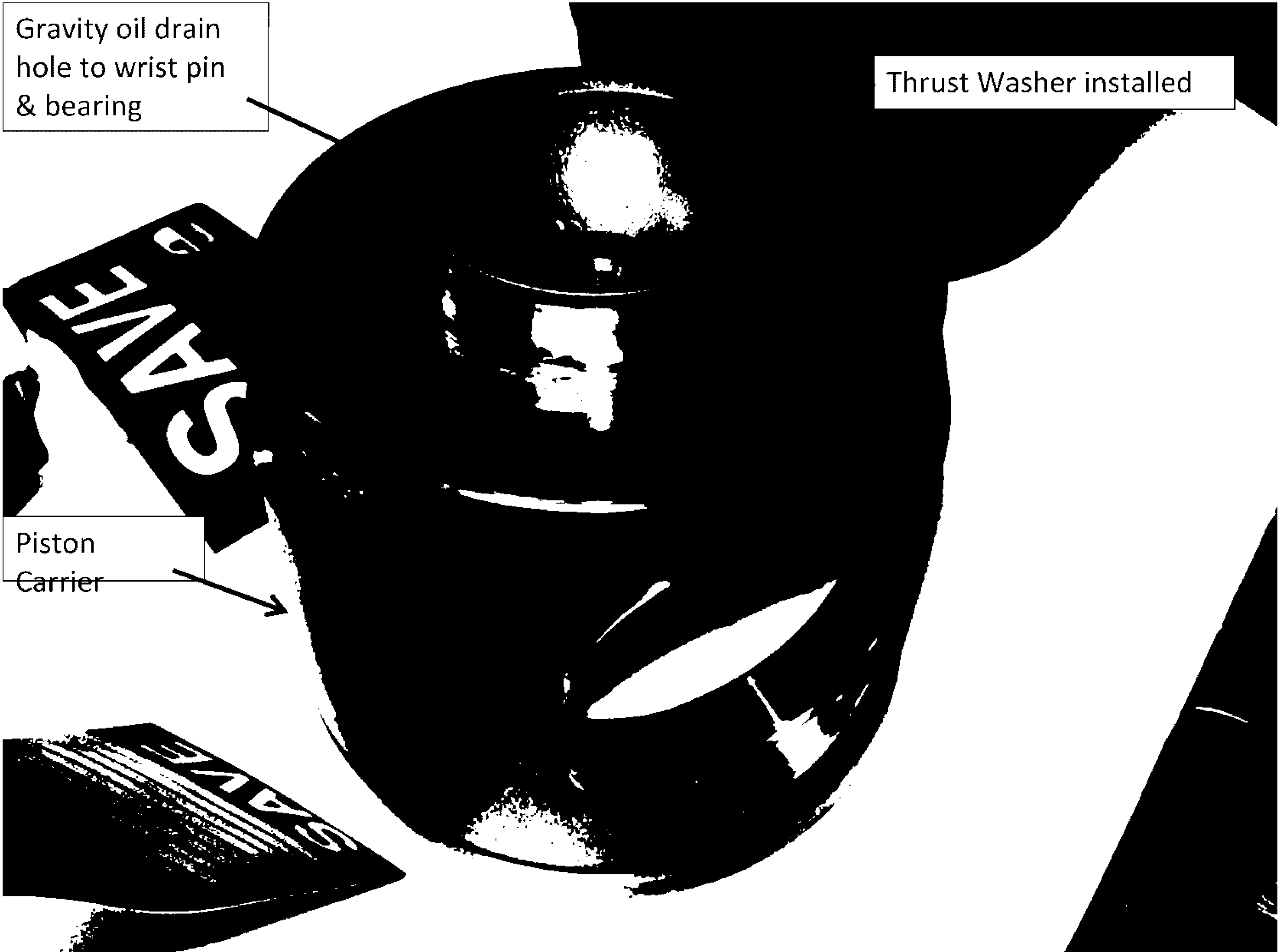
- Surry diesel wrist pin bearing failure
- Notice of Enforcement Discretion
- Industry OE
- Generic Implications
- Lessons Learned

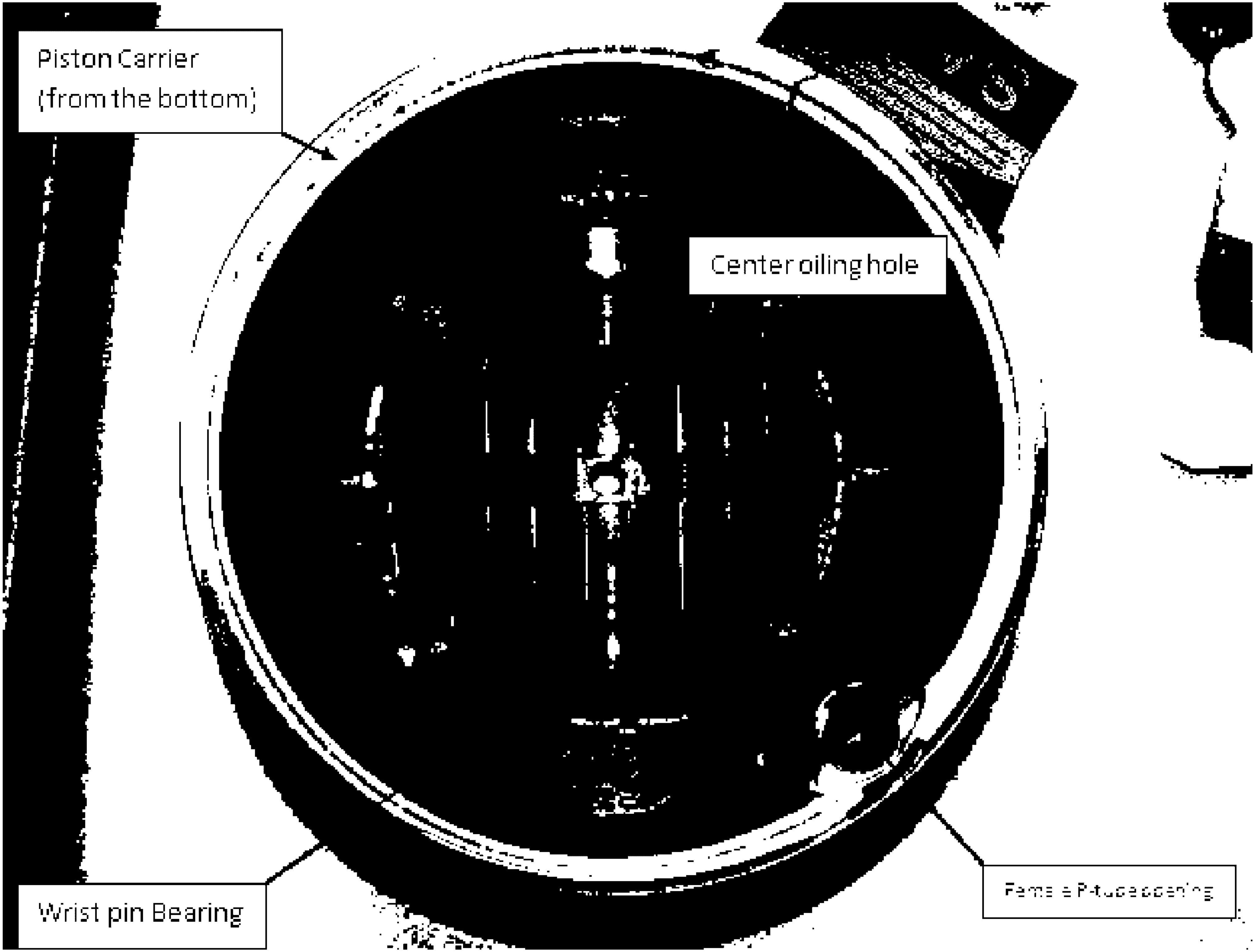
Surry EMD 645 Emergency Diesel Generators



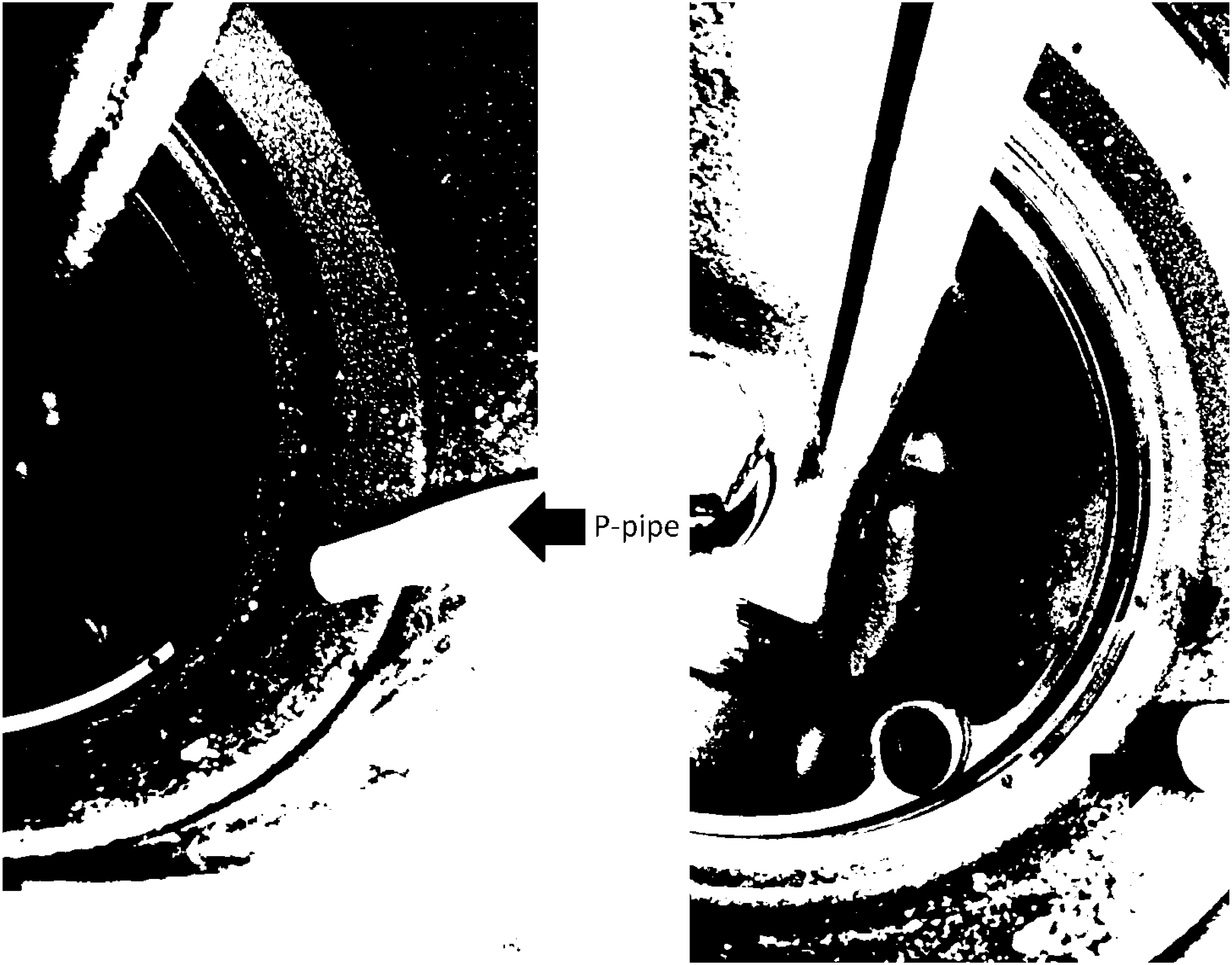

Piston
Compression Rings
Thrust Washer
Bearing Insert
Piston Pin
Piston Carrier
Snap Ring
Oil Rings
Oil Holes
Connecting Rod
Upper Bearing Shell
Lower Bearing Shell
Fork Rod Basket
Basket & Bearing
Locating Dowels


Piston
Thrust Washer
Piston Carrier
Connecting Rod
Wrist Pin
SAVE
SAVE


Gravity oil drain hole to wrist pin & bearing
Thrust Washer installed
Piston Carrier

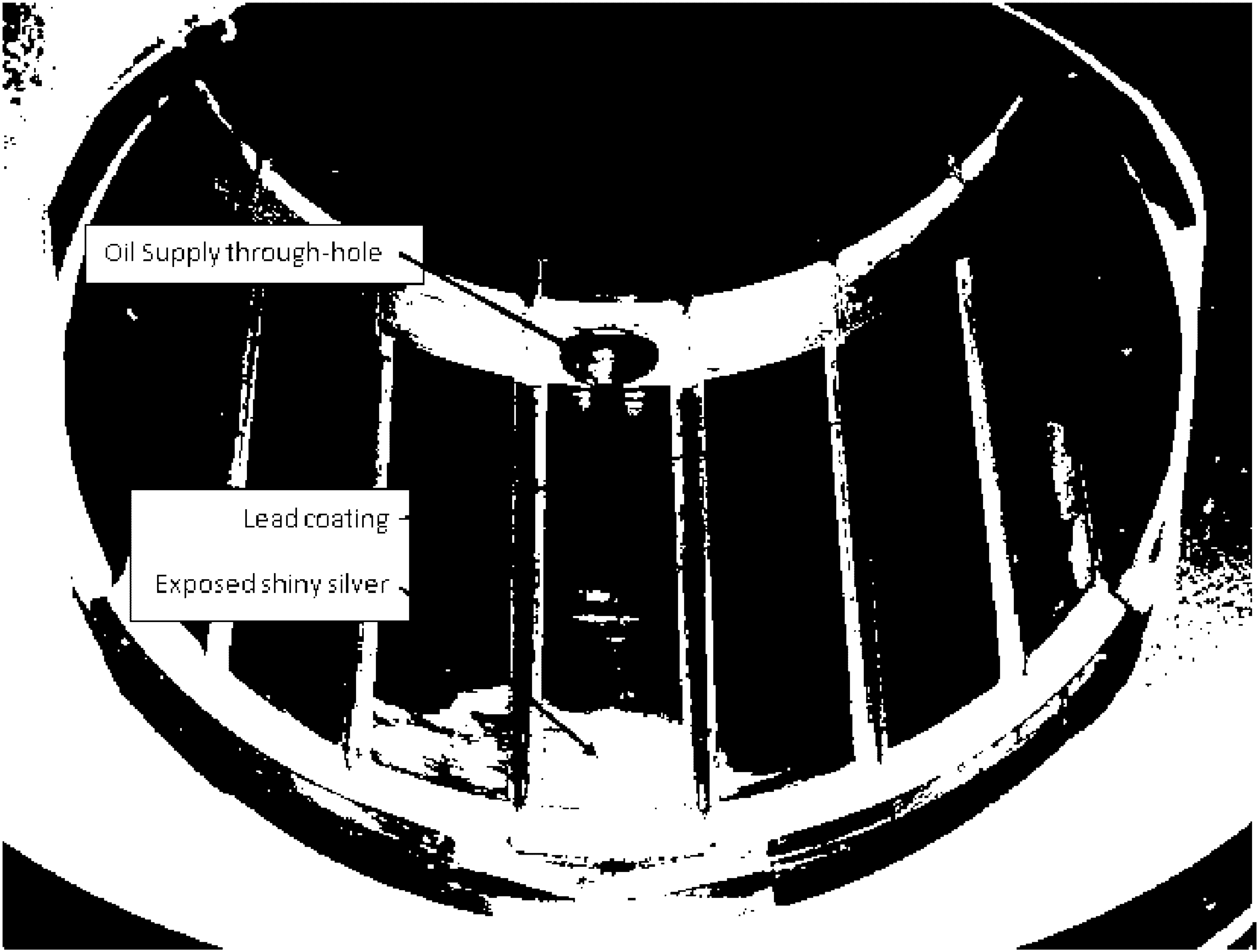


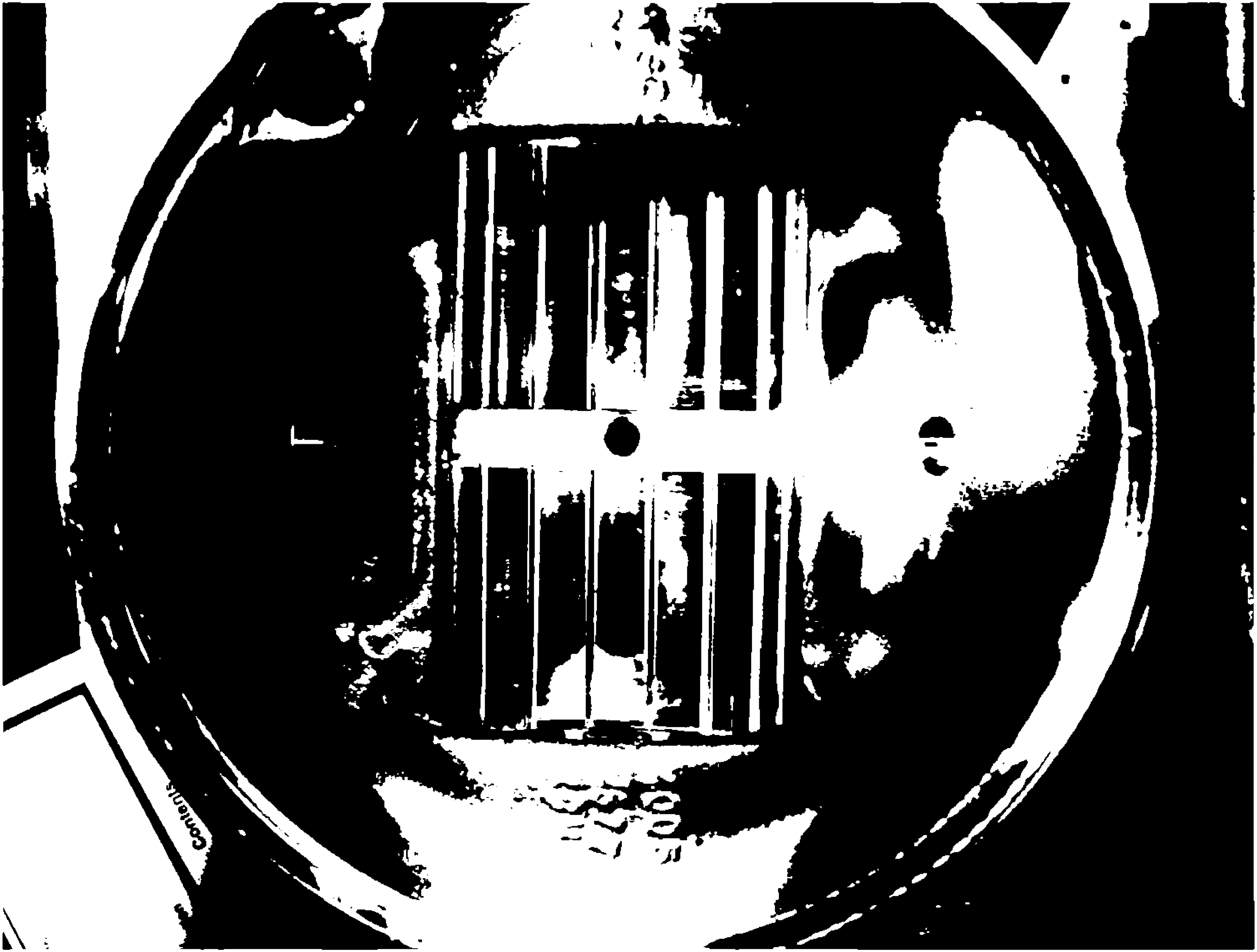
Piston Carrier
(from the bottom)
Center oiling hole
Wrist pin Bearing


P-pipe

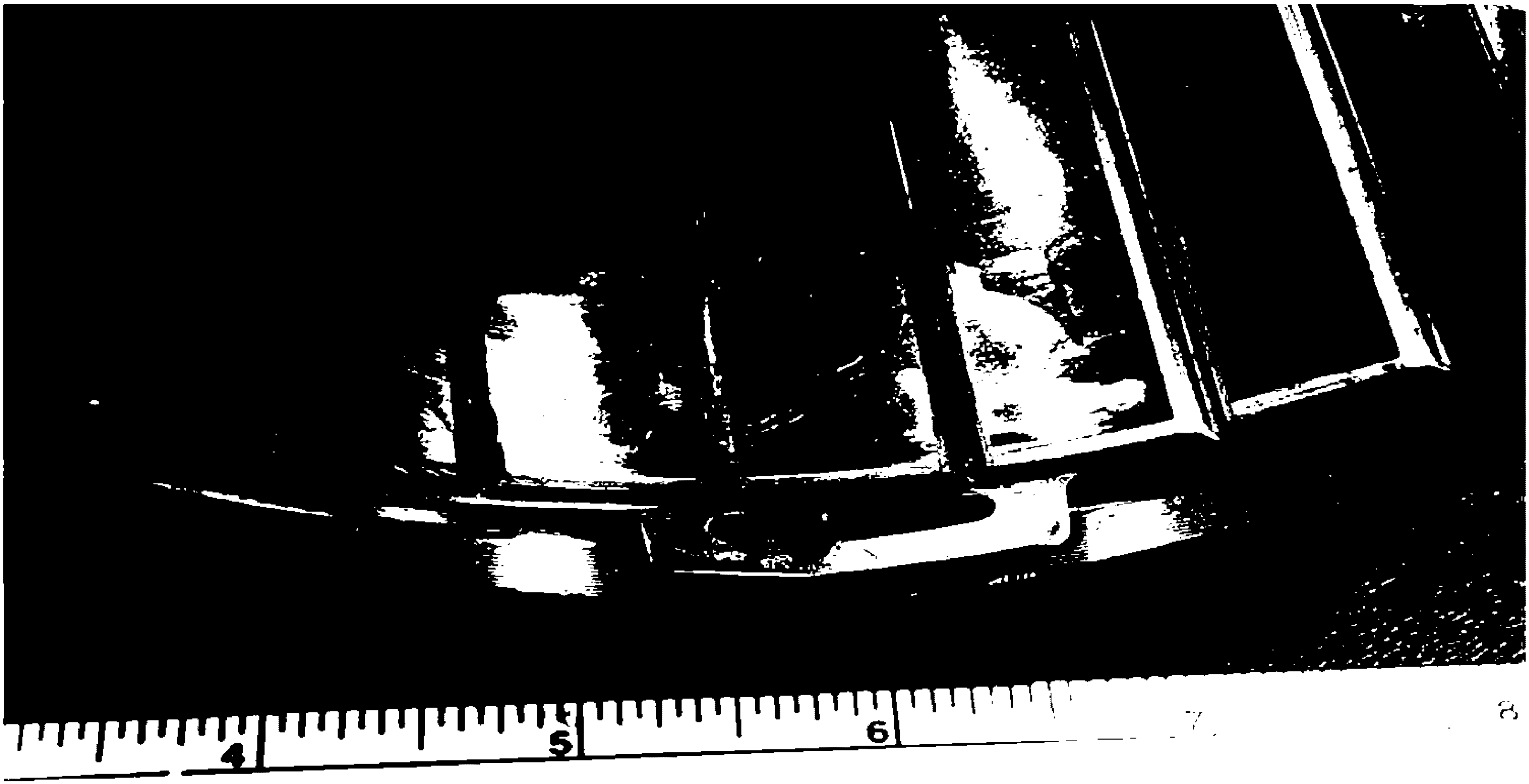
Wrist Pin Bearing Failure Timeline

Date	Time	Event
8/9/2012	14:26	EDG slow started for beginning of PMTs.
8/9/2012	19:08	After ~2hr loaded run, EDG load reject test performed. An oil sample taken during loaded run showed silver level as less than the detectable threshold (0.1 ppm)
8/9/2012	23:43	EDG fast start test (minimum starting air pressure and fuel rack held closed during first crank cycle). EDG was shutdown 15 min later with out being loaded.
8/10/2012	00:05	A second fast start was performed and the EDG was loaded.
8/10/2012	01:46	The oil sample tube fell into sump during sample attempt. When lube oil samples obtained, 0.23 ppm silver. EDG continues to run loaded for ~ 2 hrs.
8/10/2012	11:46	Silver flakes found on bottom of oil sump after it was drained to retrieve sample tube.
8/11/2012	09:45	Cylinder #5 wrist pin bearing found damaged with silver material blocking the oil hole.


Oil Supply through-hole
Lead coating
Exposed shiny silver


Contents

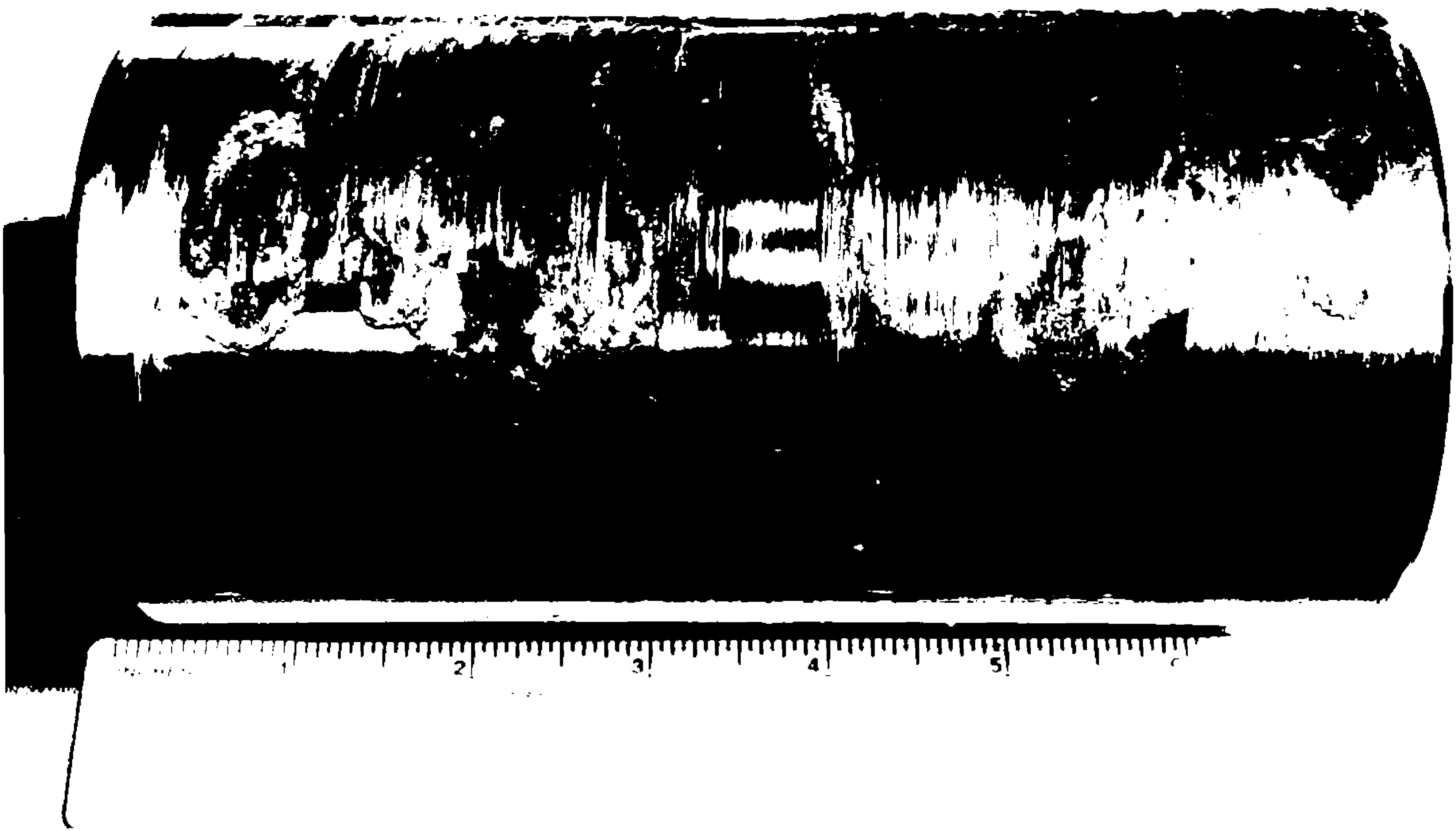
Wrist Pin Bearing Oil Grooves



Cylinder #8

Cylinder #5 Wrist Pin Bearing

Cylinder #5 Wrist Pin



P-Pipe Port Discharge
Wrist Pin Oil Feed Hole
16
P-Pipe Port Discharge
SAVE

NOED

- EDG #2 maintenance package started on 08/06 , damage identified on 08/11
- Surry EDG Tech Spec allowed outage time is 7 days
- All power packs must be replaced
- NOED discussions begin
- Several internal calls were held
- NOED granted verbally on 08/12 for additional 7 days AOT with comp measures in place

Operational Experience

- 1986 - ANO crankcase explosion (unknown cause)
- 2000 - Point Beach EDG wrist pin failure (FME)
- 2001 - Sequoyah EDG wrist pin failures (FME)
- 2001 - Surry #1 and #3 EDG wrist pin failures (oil change?)
- 2012 - Surry #2 EDG wrist pin failure (design)
- 2012 - Laguna Verde crankcase explosion (unknown cause)
- 2012 - Point Beach elevated silver

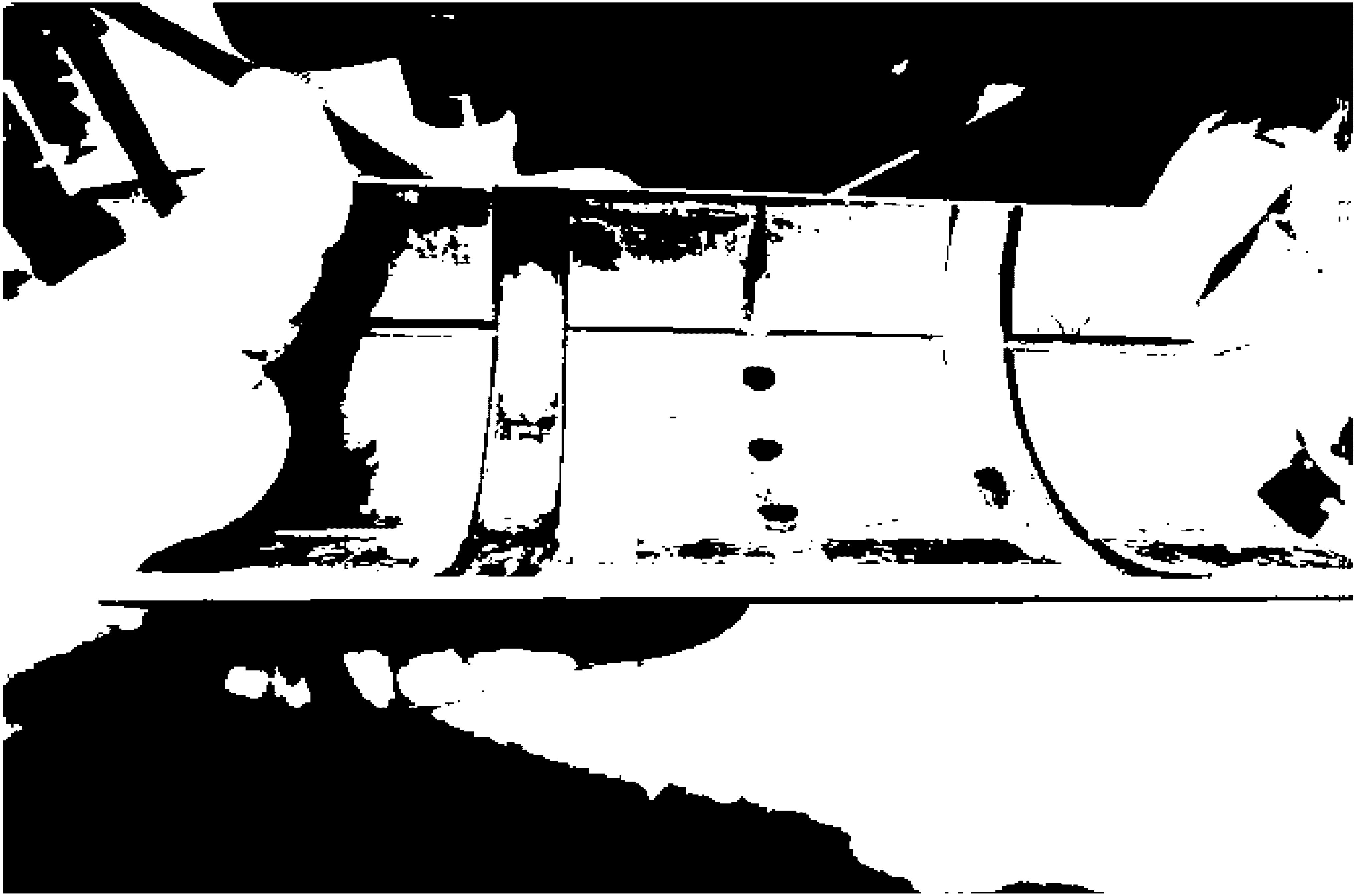
Generic Implications

- EMD diesels may be susceptible to excessive startup wear in nuclear service
- OE may not support current industry practice
- Oil analysis not necessarily a predictive tool
  - EMD OG action level of 0.3 ppm not sufficient to predict bearing failure
  - Most silver ends up on the bottom of the crankcase and not in oil
- Lead wire readings not a definitive diagnostic procedure either
- Internal engine component inconsistency

Lessons Learned

- Even supposedly well proven equipment can have unexpected failures
- Closely following licensee activities will pay dividends
- NRC engagement with licensee and collaboration across offices enhances industry safety
  - Contact HQ and other regions/residents with any information that may have generic implications
- Use of Confirmatory Action Letter for one licensee allows NRC to push other licensee to take early measures
  - R2 CAL gave R3 leverage with Point Beach

“New” Bronze Style Bearing



“New” Rocking Style Pin

U.S.NRC

United States Nuclear Regulatory Commission

Protecting People and the Environment

Key Principles of I&C Systems Architecture

Objective

- To inform staff on the key principles of I&C systems architecture to ensure that plant safety is maintained and to provide lessons learned from recent I&C systems reviews.

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I&C Systems and Their Purpose

- Monitoring of plant parameters
- Control of plant processes
- Protection of the plant during and following any anticipated operational occurrences or postulated accidents
- Auxiliary functions (e.g. control of HVAC systems)

Key Principles of I&C Systems Design

- Redundancy
- Independence
- Diversity
- Determinism
- Simplicity

Redundancy

- Redundancy in I&C safety systems can be used to achieve system reliability goals.
- I&C safety systems should have sufficient redundancy to meet the single failure criterion and provide for maintenance and testability.
- Redundancy should not be compromised through a dependency or interference.

Independence

- Independence between redundant portions of safety I&C system and between safety system and non-safety I&C systems
  - Physical Separation
  - Electrical Isolation
  - Functional Independence
  - Communications Independence

Diversity

- Diversity is the use of different means including function, design, principles of operation, and organizational and development strategies to compensate for failures within a safety system.
- Diversity is used to address common cause failures (CCFs) of the safety system.
- To mitigate against CCFs, diversity can be provided within the safety system or it can be provided through a diverse back up system.

Determinism

- Safety systems should be designed to operate deterministically.
  - Predictable: having a known system output at any time in which a given set of input signals will always produce the same output signals.
  - Repeatable: having the output of a system being consistently achieved given the same input and system properties.

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Simplicity

- Simplicity is considered to be a cross-cutting principle that affects the fundamental design principles.
- Simplicity of I&C systems design supports demonstration of conformance to other key principles such as independence and defense-in-depth.
- Given several design options on how to implement a function, the more simple design options are those that accomplish the function and address potential hazards with the most confidence and clarity.

Challenges of Implementing Digital Technology in I&C Systems

- New failure modes in digital I&C systems may challenge defense in depth measures.
- Some digital I&C systems designs are highly integrated and unnecessarily complex, making demonstration of compliance to key principles difficult.

Lessons Learned From Recent Digital I&C Systems Reviews

- Several applications had highly integrated systems without an understanding of the potential adverse effects on safety due to this integration.
- Applicants did not provide adequate justification for including additional functionality in I&C systems that are not necessary to perform the safety function but may challenge plant safety.
- Not all digital I&C failure modes were addressed by the applicant.
  - Control systems failures that can adversely impact safety.
  - Communications failures and functional dependencies were often not adequately addressed.
- Applicants often times provided claims of safety but did not provide sufficient evidence to support these claims.

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Summary

- Safety I&C systems should be as simple as possible to ensure that failures within safety systems and of connected systems do not adversely impact the safety system's ability to perform the safety function.
  - Unnecessary functions should be avoided so that failure of such functions do not adversely impact the safety system.
- Sufficient diversity should exist either within the safety I&C system or between the safety I&C system and the diverse backup system to address the potential for common cause failures.
- Designers of digital I&C systems should be cognizant of the new failure modes introduced by such systems.

High Level Presentation to DE and DSRA

Sodium-cooled Fast Reactors (SFRs) and LWRs

September 24, 2013

Imtiaz K. Madni
Sr. Reactor Systems Engineer
NRO/DSRA/SCVB

U.S.NRC
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Why Sodium-Cooled Fast Reactors (SFRs)?

- SFRs can be used for breeding or transmutation of transuranic waste products (minimizing need for permanent repositories)
- Fast spectrum reactors need to be compact to minimize moderation of fast neutrons, require higher fuel enrichment, hence high power density
- Places significant heat transfer requirement on reactor coolant
- This and other requirements have been met by the use of liquid sodium

SFR Experience in the US

EBR-I	Idaho	R&D	1951-1963	1.4 / 0.2	Pool	NaK
Fermi 1	Michigan	Power	1963-1972	200 / 61	Loop	Na
EBR-II	Idaho	Test	1963-1994	62.5 / 20	Pool	Na
SEFOR	Arkansas	Test	1969-1972	20 / 0		Na
FFTF	Washington	Test	1980-1992	400 / 0	Loop	Na

(CRBR, SAFR, PRISM, 4S designed, not constructed)

- Five facilities have operated from 1951 to 1994. Combined over 30 years operating experience
- The first electrical power production was generated by EBR-I, which fed into the power system for Arco, Idaho
- Most of the initial designs were intended to support development of breeder reactors

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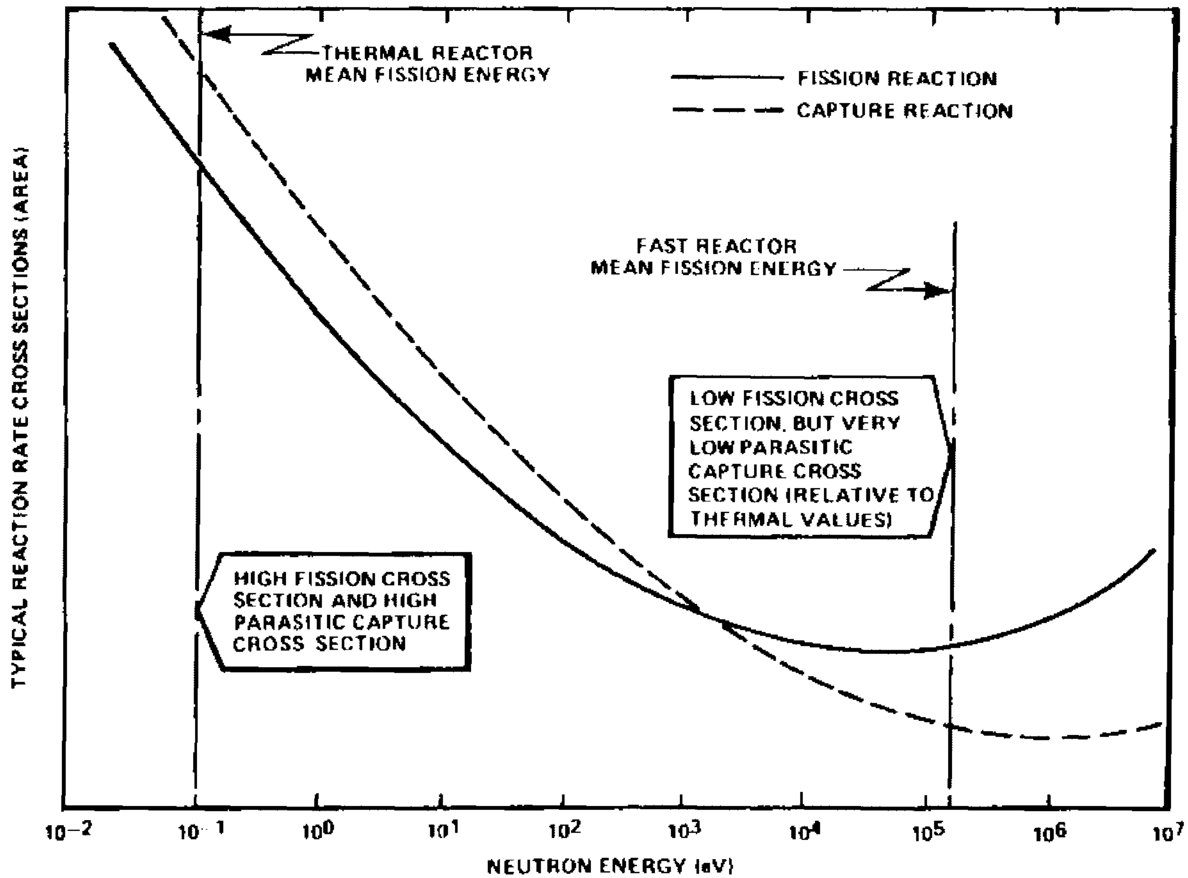
Neutronics: Thermal and Fast Spectrum Impacts



Impact of Coolant on SFR and LWR Differences

- Selected Properties of Sodium and Water

	Sodium	Water
Atomic Weight	22.997	18
Optical Properties	Opaque	Transparent
Melting Point (°C)	97.8	0
Boiling Point (°C)	>892	100
Density (kg/m3)	*880*	*713*
Specific Heat (J/kg-K)	*1300*	*5600*
Thermal Conductivity (W/m-K)	*76*	*0.54*
Viscosity (cP)	*0.34*	*0.1*
Values at STP. *Italic = Evaluated at ~300°C (and 2000 psi for water)*		

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Impact of Coolant on SFR and LWR Differences

- High BP of Sodium Provides Large Margin to Boiling

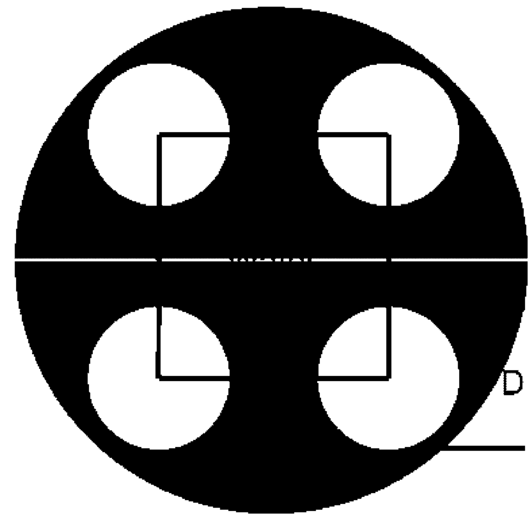
	PWR (2200 psi)	SFR
Inlet Temperature (°C)	300	355
Core DT (°C)	30	155
Outlet Temperature (°C)	330	510
Boiling Temperature (°C)	345	>892
Margin to Boiling (°C)	15	>380

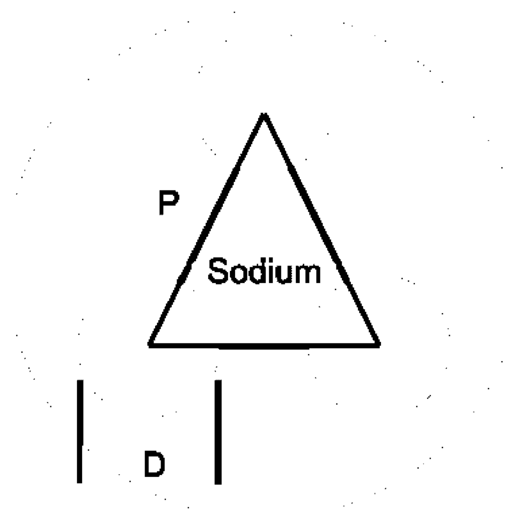
- Some nucleate boiling in a PWR is allowable under accident conditions, & margin to boiling is not real limit. Limit is defined by departure from nucleate boiling, which can result in clad burnout
- Boiling in an SFR significantly impairs heat transfer and must be avoided.

Impact of Coolant on SFR and LWR Differences

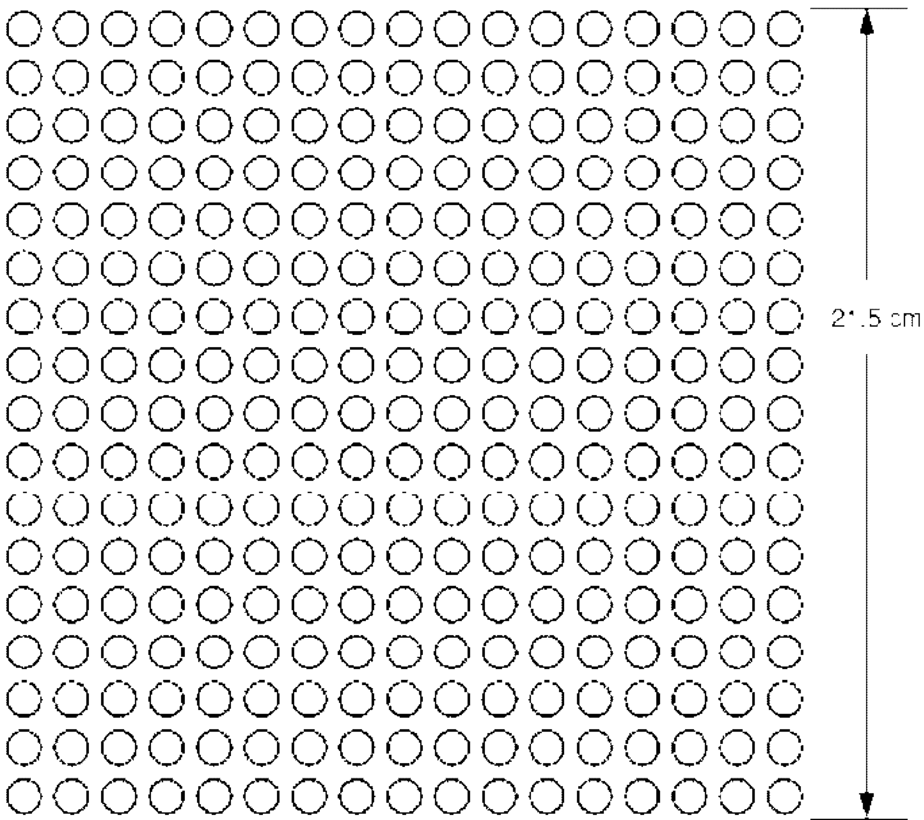
- High BP of Sodium allows Operation at Low System Pressure (near atmospheric)
- Impact on Design Features
  - Vessel thickness: PWR ~ 10-12 inches, SFR ~ 1-2 inches
  - No need for pressurization of SFR fuel pins
- Safety Advantages of low system pressure
  - Minimal pressure loading on coolant boundary
  - Coolant leaks are unlikely to propagate to a large-scale failure
  - In comparison, in a high-pressure system, coolant pipe breaks are a concern
  - No need for high pressure injection or ECCS.

- In an LWR, water acts as both a coolant and a moderator. An optimal P/D ratio is adjusted so that
    1. Adequate moderation is obtained
    2. Generated nuclear heat is affectively removed by the coolant
- In an LMR, no moderation is needed. Sodium acts only as a coolant. Because of excellent cooling properties of sodium, fuel pins can be placed much closer on a triangular pitch

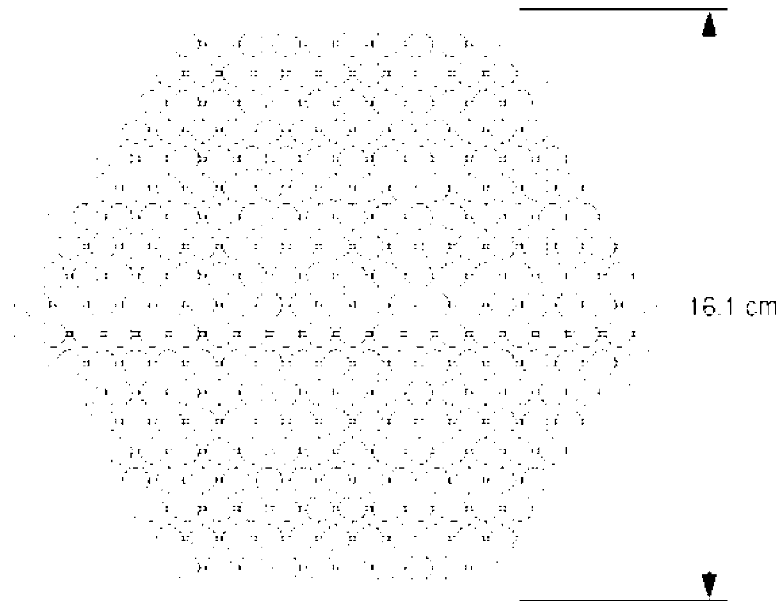




Impact of Coolant on SFR and LWR Differences



Typical PWR Assembly (289 pin locations)
Pin Diameter = 9.4 mm
Pin Pitch = 12.5 mm



"Typical" SFR Assembly (271 pins)
Pin Diameter = 7.4 mm
Pin Pitch = 8.9 mm

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Impact of Coolant: Sodium Interactions

- Sodium is inherently compatible with stainless steel
  - Does not corrode structural materials
  - Experience with EBR-II after 30 years of operation
- Fuel-coolant interactions are benign for metallic fuel
- Many fission products are soluble in sodium, hence
  - can be filtered out in the cold trap, this contributes to scrubbing
- Sodium Reaction with Air
  - Characterized by small flames at interface, formation of Na_2O on surface, and vigorous emission of oxide fumes
  - Hence sodium systems need sealed guard vessels and inert cover gas (Argon)
- Sodium Reaction with Water
  - Vigorous, exothermic, and releases hydrogen
- ^{23}Na Activation
  - Results in radioactive isotope ^{24}Na circulating in primary system

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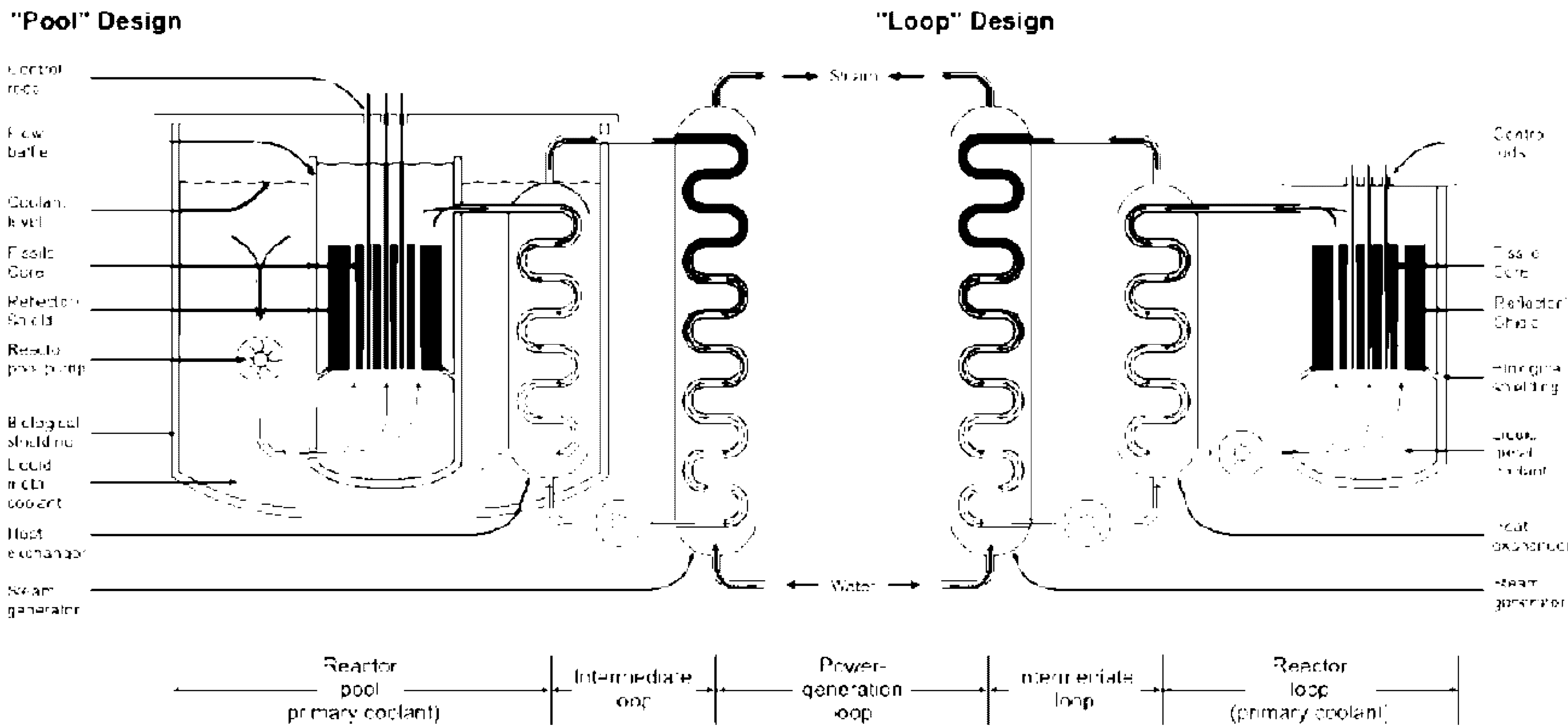
Impact of Coolant: SFR needs an Intermediate Coolant Loop



- Na activation and reaction with water
- Requires separation of high-pressure steam cycle from radioactive primary system; hence intermediate heat transfer system (IHTS) used
- Two design choices: Pool and Loop

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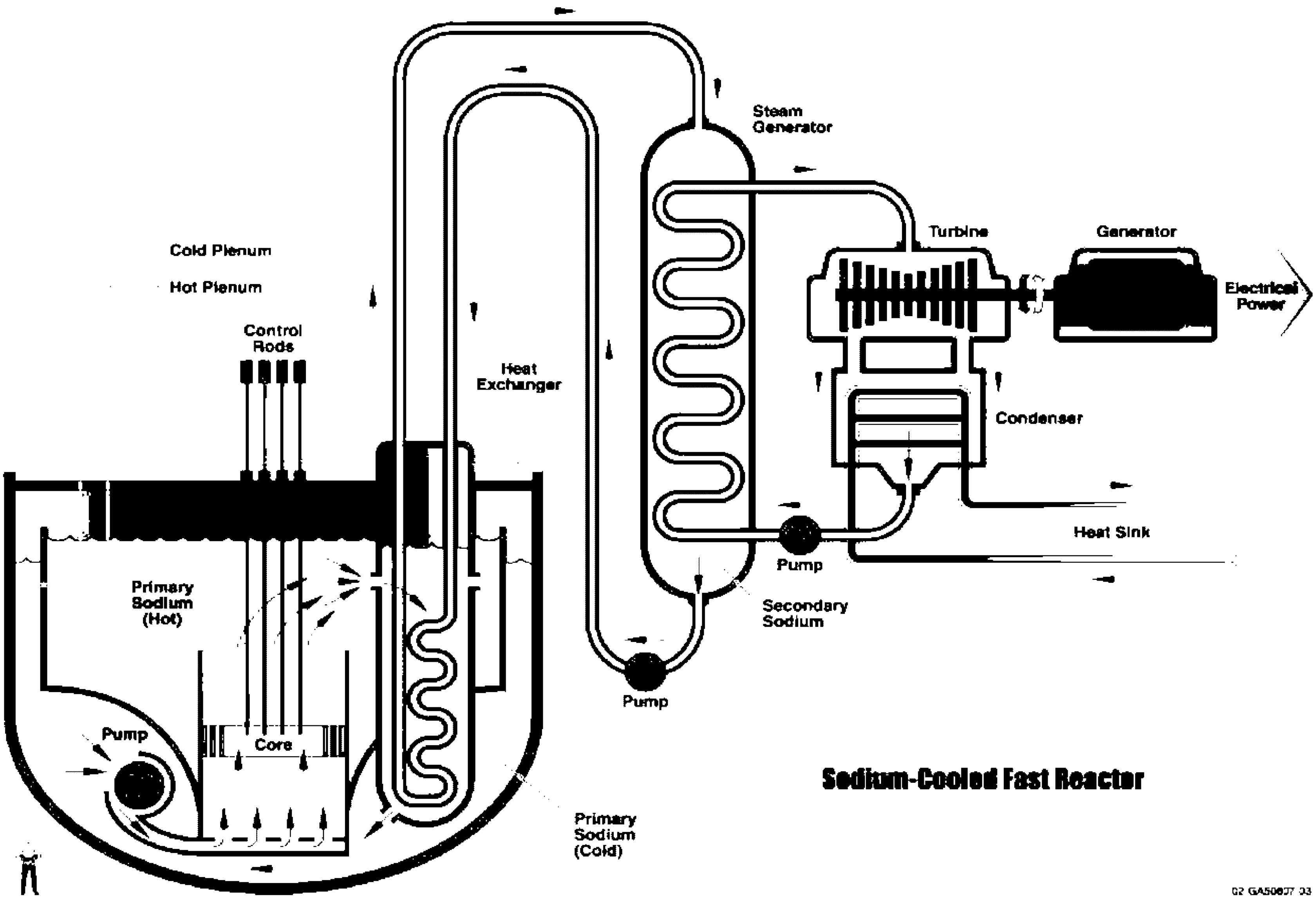
Impact of Coolant: Example of Pool Configuration



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SFR Safety Issues: Neutronics

- Power variation across core due to neutron leakage at core boundaries plus high power density
  - Need to use ducted assemblies to control location of fuel and core temperatures (lesson learned from EBR-I)
  - Flow rate in each ducted assembly is adjusted via adjusted nozzle openings in order to have uniform outlet temp from the core
  - Core reactivity is very sensitive to core geometry
- Core sodium void worth is typically positive in center of larger reactor cores (e.g. +ve for PRISM, -ve for 4S)
  - With proper design, overall core void could be made negative. How?
    - High leakage cores
    - Large L/D
- Fuel is not in most neutronically reactive configuration in reactor core
  - Relocation of fuel has the potential to significantly increase reactivity, including exceeding prompt critical conditions

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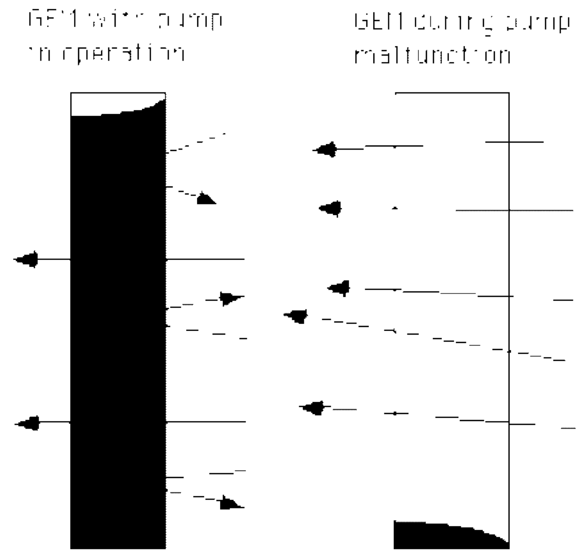

- Na allowed to enter at bottom & fill tube when pumps are operating
- GEM's designed to lower core power level if main coolant pumps malfunction
  - Helium pressure causes Na level in pins to drop & allow more neutrons to leak out of core
  - Lower number of neutrons causes fewer fissions to occur and power level drops

SFR Safety Issues: Sodium Reactivity with Air

- In event of a leak from Na-containing systems/components
  - Na may emerge as a jet (and impinge on other structures) or may spill
- If air is available
  - Leaking sodium will react with air. This can lead to:
  - Spray fire (starts at 120 °C, high burning rate, high aerosol production) or
  - Pool fire (starts at 250 °C, low burning rate, low aerosol production)
  - Na fires produce aerosols (NaO and Na_2O) which react with air to produce NaOH (in a few sec) and Na_2CO_3 (in several min)
  - Aerosol generation may increase thermal loading and raise gas pressure
  - Aerosols deposit on floor, walls, and ceiling, can cause equipment damage (electrical, instrumentation)

SFR Safety Issues: Sodium Reactivity with Air

- Significant Na Leaks
  - BN-600, October 7,1993, Na leak on pipeline for cold trap. 800 kg escaped. May 1994, leak from secondary, 30 kg burned.
  - Monju (spill and burn of several hundred kg of secondary Na, December 8, 1995)
  - Super Phenix (Na leak from used fuel storage tank. Inerted GV, hence no fire, no casualties.) Led to shutdown in July 1996, too expensive.
  - No reported adverse effects to plant personnel or surrounding environment

SFR Safety Issues: Sodium Reactivity with Air

- Prevention
  - Surround sodium pipes/vessels with inert-gas filled and sealed guard pipes/guard vessels (not including intermediate piping) so any Na leakage enters an inert volume (multiple barriers).
  - Steel-lined confinement cells to contain secondary Na from a leak and avoid core-concrete reactions. (Also, concrete selection to minimize interactions).
  - Direct leaking Na through catch pan systems into a oxygen starved recovery tank to avoid pool fire.

•

SFR Safety Issues: Sodium Reactivity with Air

- Detection
  - Objective is to prevent large leaks, corrosion, and fires by early detection
- Mitigation
  - Fire extinguishing powders for quick and effective extinction of Na fire
  - Limit duration of fires (~15 min) to avoid serious damage to structures
  - Preclude pressurization of cells by relief valve openings or rupture disks
  - Limit drop height of spray fire by arranging catch pans every few meters vertically – makes spray fire into pool fire.
  - Special filters in ventilation to remove Na aerosols from containment atmosphere

SFR Safety Issues: Na Reactivity with Water

- SG tubes are boundary between sodium in IHTS & HP steam
- SG tube leaks have occurred in many SFRs (e.g. PFR, Phenix, BN-600) and are an important safety issue with new designs
  - IHX is designed to withstand full steam pressure if a tube leak occurs
  - However, contact of water with sodium leads to exothermic reaction that could rapidly pressurize IHTS piping or IHX (2nd safety barrier)
  - Need to develop and qualify early leak detection systems to prevent propagation of tube leaks and ruptures
  - In PRISM, both large Na leak into air and steam generator tube rupture are included by GE as bounding events to be considered in the design for licensing

SFR Safety Issues: Na Reactivity with Water

- Design options
  - Use successful approach used in EBR-II, Toshiba 4S designs i.e. use double-walled SG tubes
    - EBR-II experience: No tube leaks occurred during 30 years of operation, Na and water never came in contact during operating lifetime of plant
  - Use better SG tube materials, or new tertiary fluid, or a new fluid in IHTS
- Should SG tube leaks be considered as DBAs?
  - Not settled yet, because no final application has been submitted

Commercial Grade Item Dedication and 10CFR21: Application to Digital I&C Systems and Software

Milton Concepcion, MSc (Eng)
Sr. Digital I&C Engineer
NRO/DE/ICE2

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This presentation is intended for information purposes only and does not replace independent professional judgment. Statements of fact and opinions expressed are those of the presenter individually and, unless expressly stated to the contrary, are not the opinion or position of the NRC, EPRI, or ASME NQA-1 Committee. The presenter assumes no responsibility for the content, accuracy or completeness of the information presented.

Agenda

- Commercial-grade item (CGI) dedication process requirements & guidance
- CGI dedication of digital I&C equipment
- Latest developments affecting CGI dedication of software

CGI Dedication References

- 10 CFR Part 21
- Generic Letter 89-02
- Generic Letter 91-05
- EPRI NP-5652, "Guideline for the Utilization of Commercial Grade Items in Nuclear Safety Related Applications (1988)."

Additional Inspection Guidance

- Inspection Procedure 38703 (issued 1993)
- New Inspection Procedure 43004 (issued October 2007)

CGI Dedication Process(cont'd)

- Commercial-grade dedication is an **acceptance** process by which a CGI is designated for use as a **basic component.**

- This acceptance process is undertaken to provide reasonable assurance that a CGI to be used as a basic component will perform its intended safety function and, in this respect, is deemed equivalent to an item designed and manufactured under a 10 CFR Part 50, Appendix B, quality assurance program.

CGI Dedication Process

- An acceptable dedication program involves:
  - Review for suitability of application per Criterion III, "Design Control," of Appendix B
    - (i.e., Technical Evaluation)
  - Acceptance controls per Criterion VII, "Control of Purchased Material, Equipment, and Services," of Appendix B
    - (i.e., Four Acceptance Methods)

CGI Dedication Process (cont'd)

- **Technical Evaluations**
  - Determine item's safety function and service conditions
  - Functional classification of items and components
  - Review of vendor's technical/development data
  - **Identification and selection of item's critical characteristics**
  - Determine appropriate acceptance criteria

- **Acceptance Process**
  - Method 1: Special tests and inspections
  - Method 2: Commercial-grade survey of supplier
  - Method 3: Source verifications
  - Method 4: Acceptable supplier/item performance record

Software Used in Nuclear Industry

- Process Control (Digital I&C Equipment)
- Design & Analysis
- Operations (Management/Administration)
  - Not addressed in this presentation

CGI Dedication of Digital I&C Equipment

- NRC conditionally accepted of the following EPRI Guidance Documents for Dedication of Digital I&C including Programmable Logic Controllers (PLC):
  - EPRI TR-106439, "Guideline on Evaluation and Acceptance of Commercial Grade Digital Equipment for Nuclear Safety Applications," October 1996
  - EPRI TR-107330, "Generic Requirements Specification for Qualifying a Commercially Available PLC for Safety-Related Applications in Nuclear Power Plants," December 1996

CGI Dedication of Digital I&C Equipment (cont'd)

- Digital I&C equipment introduces additional challenges
  - Complexity of the device – including its internal architecture, external interfaces, communication links, etc.
  - Access to detailed information/documentation (design, development, testing, verification/validation, configuration control)
  - Proper identification and verification of critical characteristics
    - Hardware+software (operating/application)
    - Extent/thoroughness of Critical Design Review (CDR)*
  - Use of software tools
  - Cybersecurity
  - Crediting relevant operating history
  - Engineering judgment
- Not all commercial digital I&C equipment can be successfully dedicated*

*EPRI TR-106439, "Guideline on Evaluation and Acceptance of Commercial Grade Digital Equipment for Nuclear Safety Applications."

Implementation and Examples

(with the invaluable collaboration from Rossnyev Alvarado -- NRR)

- HFC-6000 platform
- SPINLINE 3 platform

- Review of the OS software QA program was limited to assessment of the process, plans, and procedures as they relate to maintaining the commercially dedicated system
- Verification of the critical characteristics – Methods 1, 2 and 4

Note Method 3 would require the licensee to observe FAT for a purchased system

Latest Developments Impacting Software Dedication

- NQA-1-2008/NQA-1a-2009 changes
  - Reaffirmed endorsement in RG 1.28, Rev. 4 – June 2010
  - Subpart 2.14, "Quality Assurance Requirements for Commercial Grade Items and Services."
    - Provides amplified requirements for CGI Dedication
  - Subpart 2.7, Section 302
    - For acquisition of software that has not been previously approved under a program consistent with NQA-1 for use in its intended application
    - Changed from an "evaluation" (i.a.w. SP 2.7) to a dedication process (i.a.w. Part I, Req. VII and SP 2.14)
    - Application in the context of SP 2.7 includes ALL software (e.g., process control, design & analysis)

Latest Developments Impacting Software Dedication (cont'd)

- EPRI 1025243, "Plant Engineering: Guideline for the Acceptance of Commercial-Grade Design and Analysis Computer Programs Used in Nuclear Safety-Related Applications."
  - Generic technical evaluation process overview
  - Functional safety classification of computer programs
  - Acceptance of commercial-grade computer programs using the dedication process
  - Currently under review by NRC QA staff – potential RG endorsement
  - Impact of EPRI report on IEEE 7-4.3.2 guidance related to software tools?

Latest Developments Impacting Software Dedication (cont'd)

- IEEE 7-4.3.2 and software tools
  - At this time, there is no direct relationship between EPRI-1025243 and IEEE Std. 7-4.3.2.
  - It is worth noting that IEEE 7-4.3.2-2010 added CGI dedication (Clause 5.17) as an alternative to establish suitability of software tools for use in safety related systems.
  - Although the scope of EPRI-1025243 does not directly address software tools used to support the development of operating and/or application software in digital I&C systems, the dedication guidance provided in EPRI-1025243 may be considered by an applicant/licensee.

Latest Developments Impacting Software Dedication (cont'd)

- Embedded digital devices
  - Register Notice on Embedded Digital Device RIS (requesting public comment) issued May 20, 2013.
  - Commercial-grade replacement products containing embedded digital devices that include software, software-developed firmware, or software-developed logic that may not have been developed in accordance with guidance and acceptable industry standards.
  - Requirements to identify the use of embedded digital devices and sufficiently document the quality of the embedded digital devices to support commercial-grade item dedication.

Latest Developments Impacting Software Dedication(cont'd)

- 10CFR21 rulemaking efforts
  - Part 21 and the philosophy of dedication apply to all safety-related items and services, including software. However, Part 21 and its associated guidance do not provide contemporary requirements for software dedication.
  - While the staff notes that software can be safety-related and can be dedicated, some stakeholders have interpreted Part 21 to the contrary. Part 21 provides an area for potential improvement in defining the requirements for software dedication.
  - A regulatory guide to address commercial grade dedication will be essential in providing clear expectations to Part 21 stakeholders. The regulatory guide would include implementation guidance for software.

Summary

- Licensees/vendors must know, define, and control CGI dedication process
  - Establish and maintain product/design suitability (hardware+software)
  - Determine safety function(s)
  - Identify and select critical characteristics
  - Utilize defined acceptance methods

- Licensees/vendors must establish and maintain complete documentation/records
  - Evaluations
  - Acceptance tests & inspections
  - Supplier controls

Questions/Comments

Graphite: Properties and Behavior

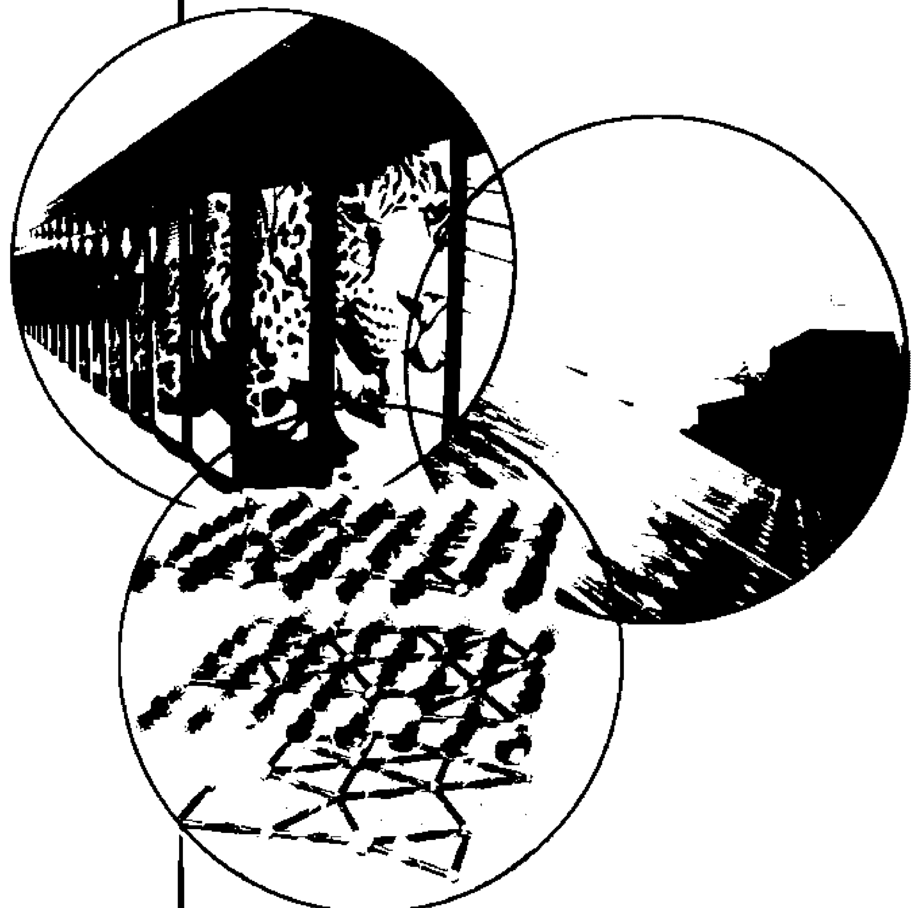
Tim Burchell

Materials Science & Technology Division

Presented to

US Nuclear Regulatory Commission

January 12th 2011



U.S. DEPARTMENT OF ENERGY

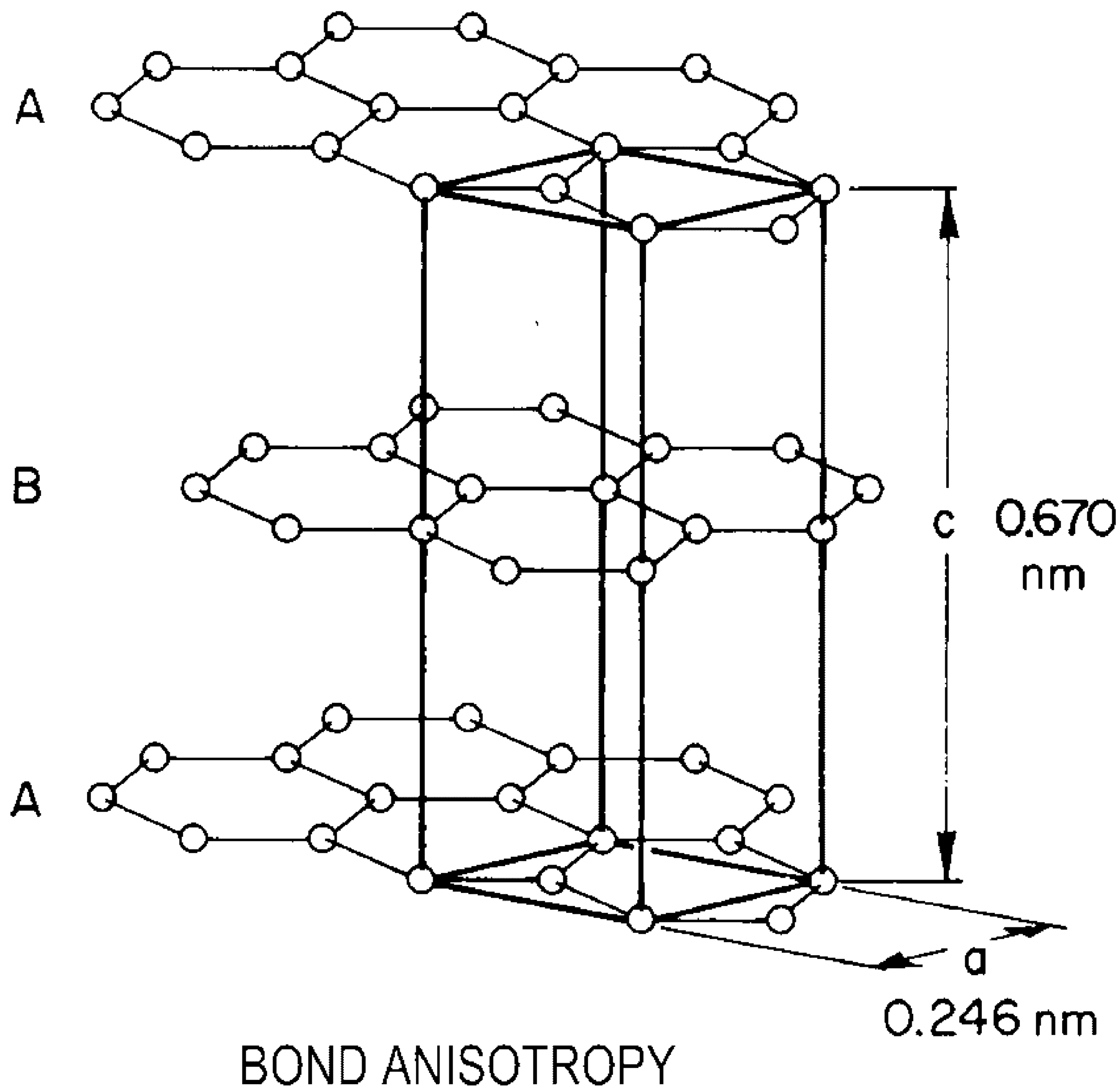
OAK RIDGE NATIONAL LABORATORY

MANAGED BY UT-BATTELLE FOR THE DEPARTMENT OF ENERGY

Overview of Presentation

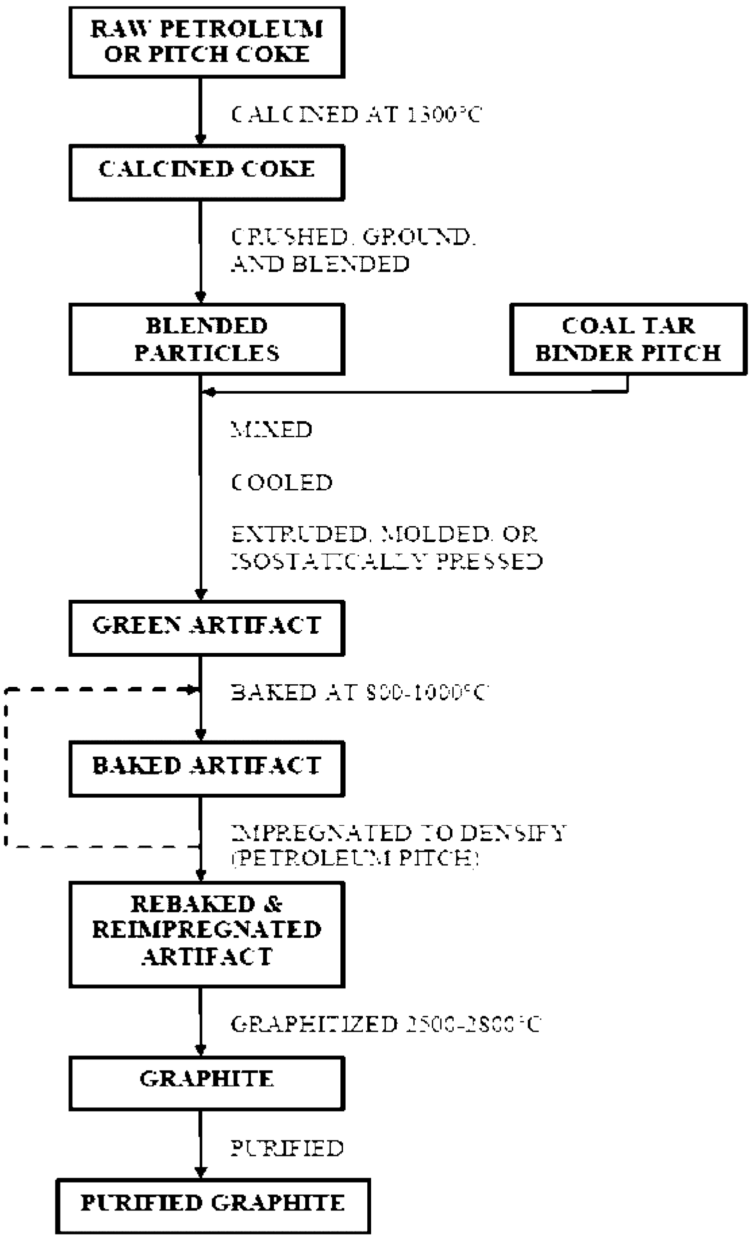
- Manufacture
- Structure
  - Single crystal and polycrystalline synthetic graphite
- Porosity and texture
- Physical Properties
  - Thermal
  - Electrical
- Mechanical Properties
  - Elastic constants
  - Strength and fracture
- Applications
- Summary

Graphite Single Crystal Structure



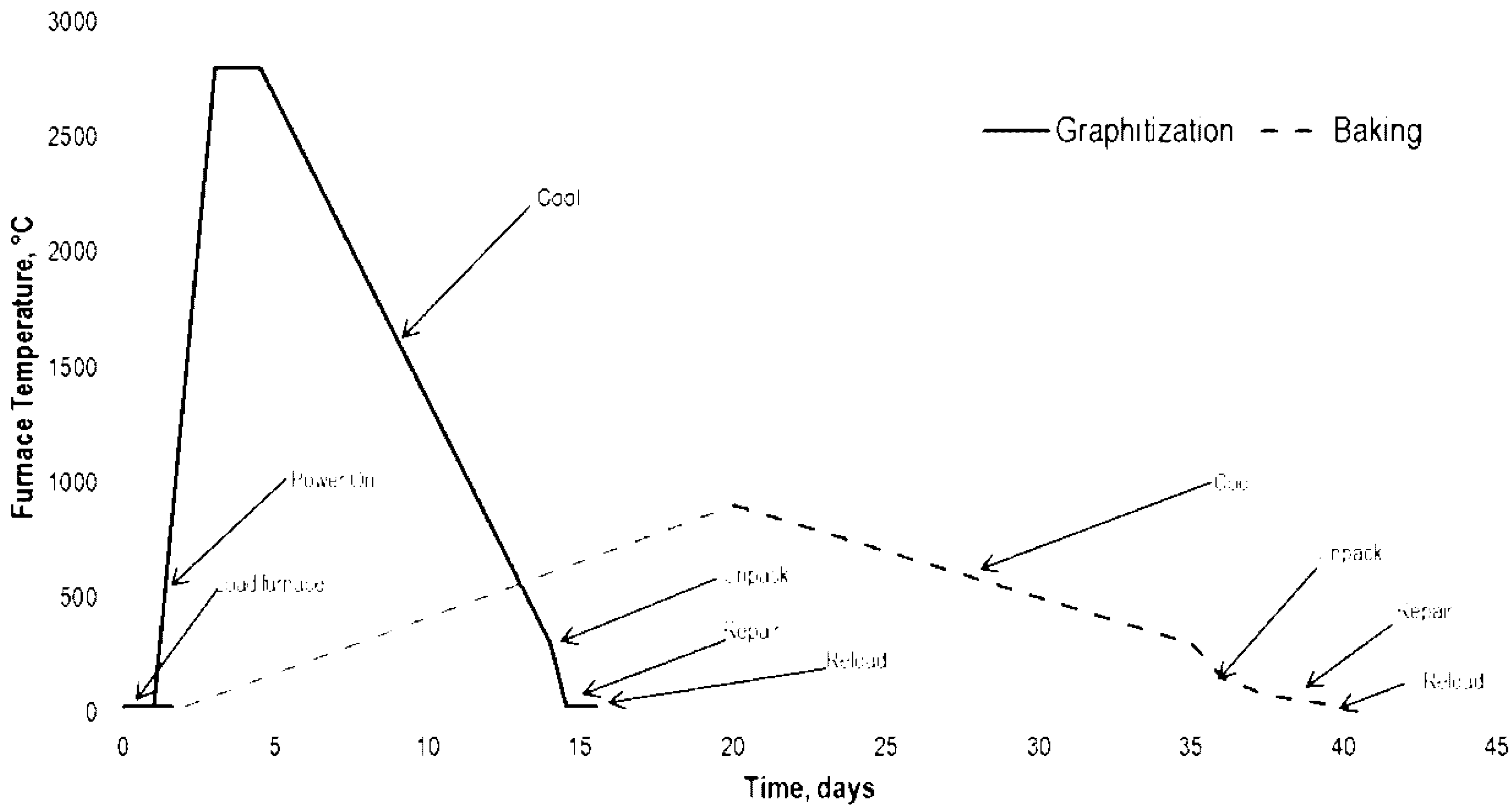
- Strong, stiff covalent bond in-plane
- Weak bonds of attraction between graphene planes
- ABA repeat stacking (can get ABC…)
- Crystal unit cell size:
- <a> = 0.246 nm
- <c> = 0.670 nm
- Coherence lengths, l_a and l_c are measures of crystal size

Synthetic Graphite Manufacture



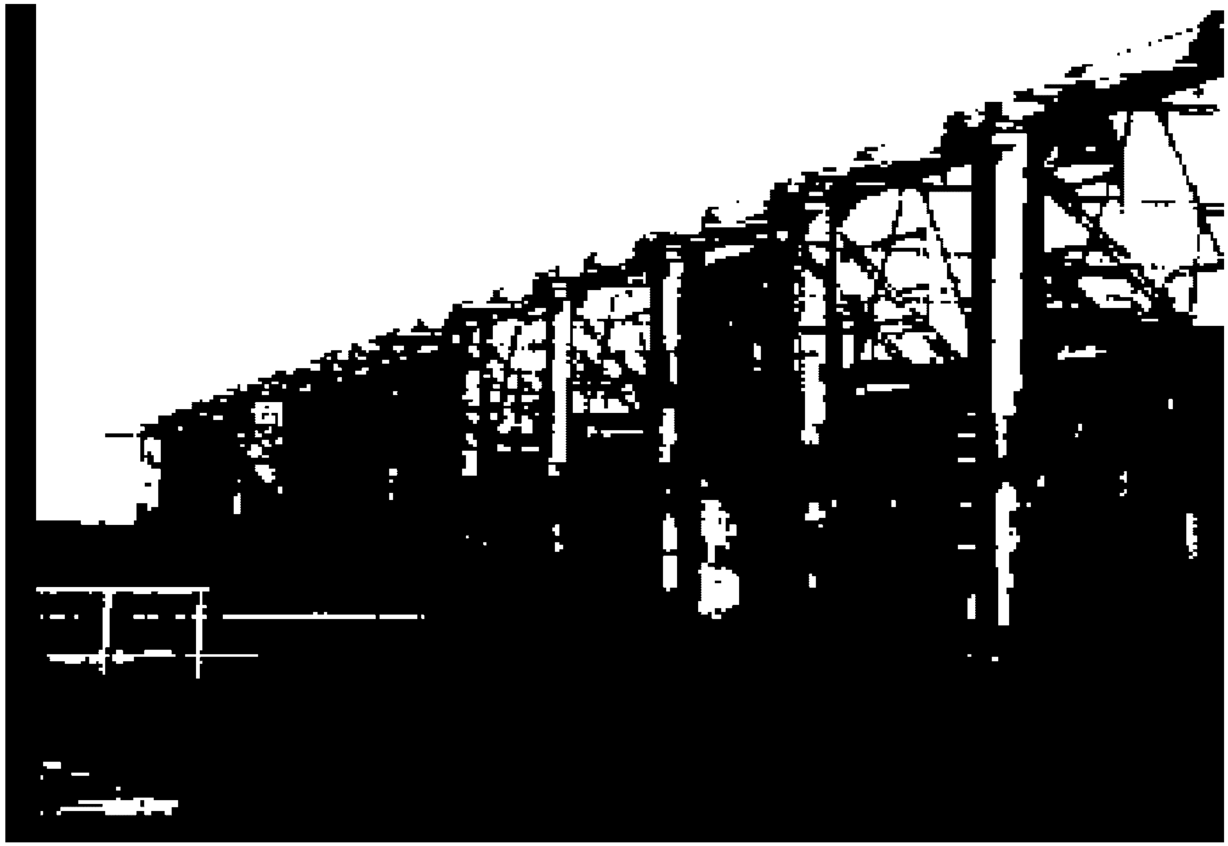
- Petroleum coke from calcination of heavy oil distillates
- Pitch coke from calcination of coal -tar pitch
- Coke filler particle morphology and green artifact forming method affect texture and properties
- First bake is a critical stage, - controlled binder pyrolysis
- Acheson or longitudinal graphitization
- Long cycle times ~ 9 months

Manufacture – Baking and Graphitizing



Slow heating and cooling during baking allows escape of pyrolysis gasses and minimizes thermal gradients

Manufacture - Baking



Modern car bottom carbon/graphite baking furnace

Green bodies packed in coke and placed in steel saggers

Manufacture – Acheson Graphitization

Baked artifacts surrounded by a coke pack and covered with sand to exclude air. Electric current flow through the coke-pack & artifacts

Manufacture – Longitudinal Graphitization

Furnace covered with sand to exclude air, current flows though the baked artifact

Carbon atoms migrate to thermodynamically more stable graphitic lattice structure and 3D ordering achieved (degree of ordering depends on feedstock type)

Manufacture - Purification

Post graphitization halogen gas process

Can be performed with solid fluoride additives to Acheson graphitization furnace or solid fluoride additives to formulation

Manufacture - Purification

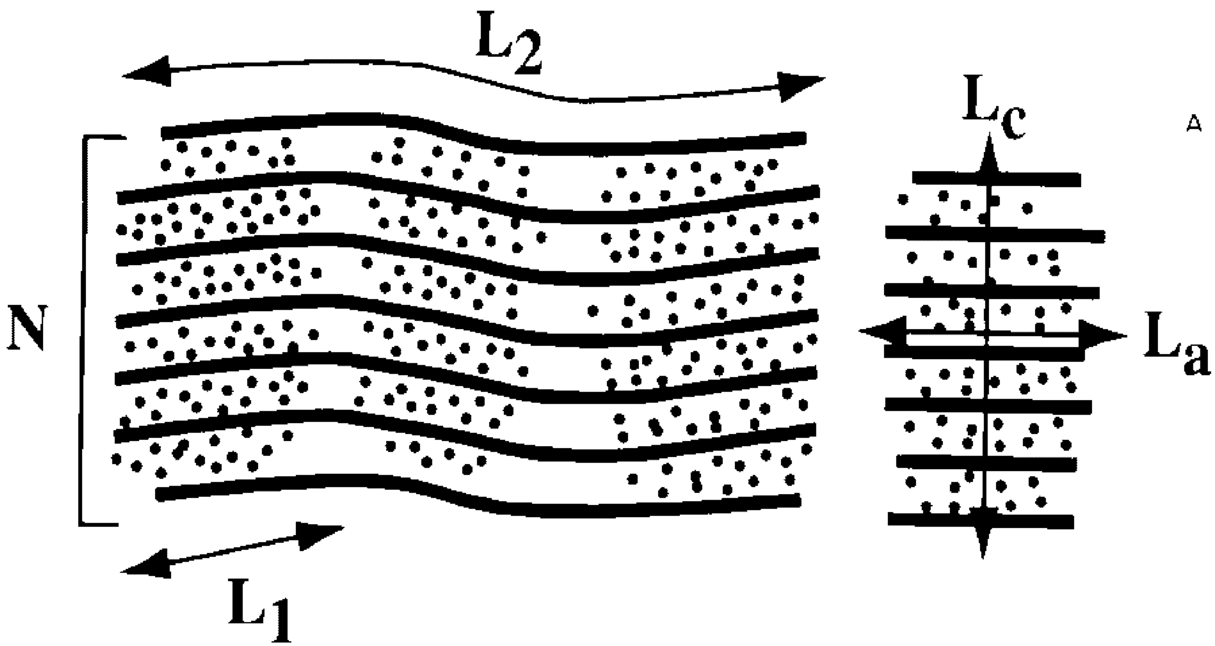
Unpurified graphite

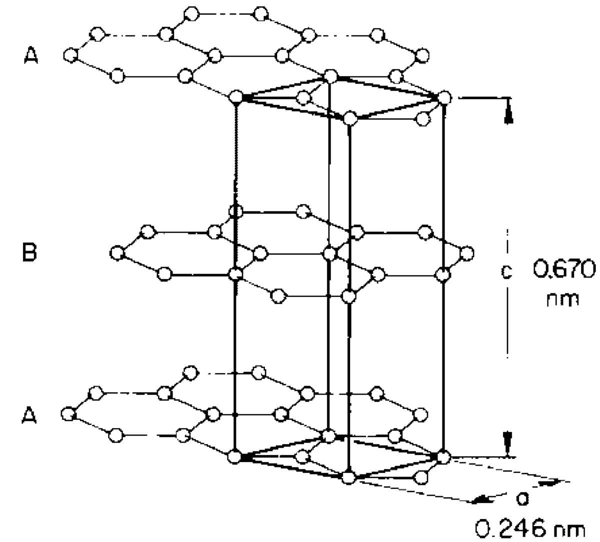
Thermally purified graphite

Chlorine purified graphite

Fluorine Purification

Crystallites & Optical Domain





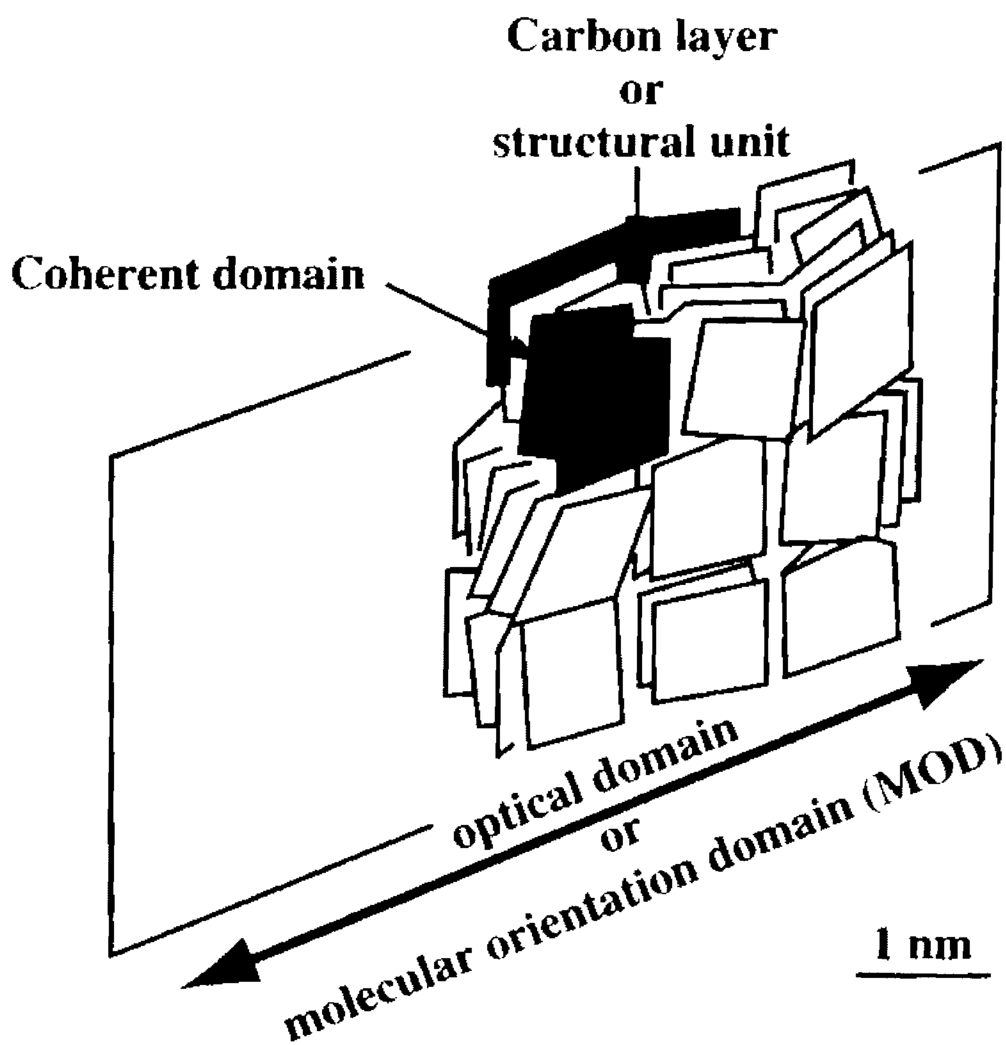
Sciences of Carbon Materials, Marsh & Reinoso

Synthetic Graphite Structure

Nuclear graphite XRD crystal parameters

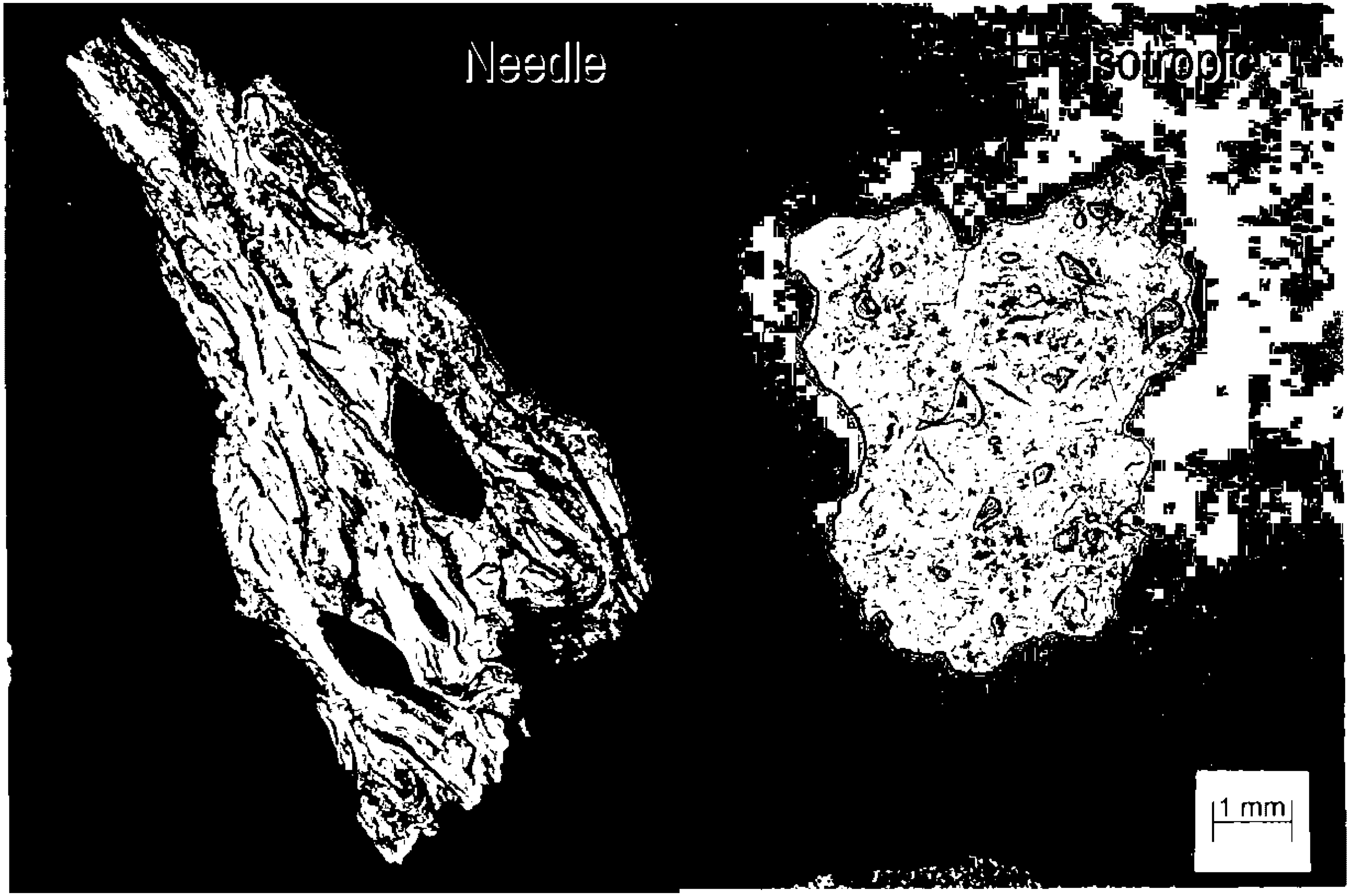
Graphite grade	Coke type	<c> (A)	lc (A)	<a> (A)	la (A)
HOPG	n/a	3.345	424	1.229	138
H-451	Petroleum	3.353	269	1.230	300
IG-430	Pitch	3.361	218	1.231	298
BAN	Petroleum	3.365	250	1.231	322
NBG-17	Pitch	3.364	186	1.231	286
IG-110	Petroleum	3.366	190	1.231	256
PCEA	Petroleum	3.367	238	1.231	250
NBG-18	Pitch	3.370	191	1.232	294

Crystallites & Optical Domain



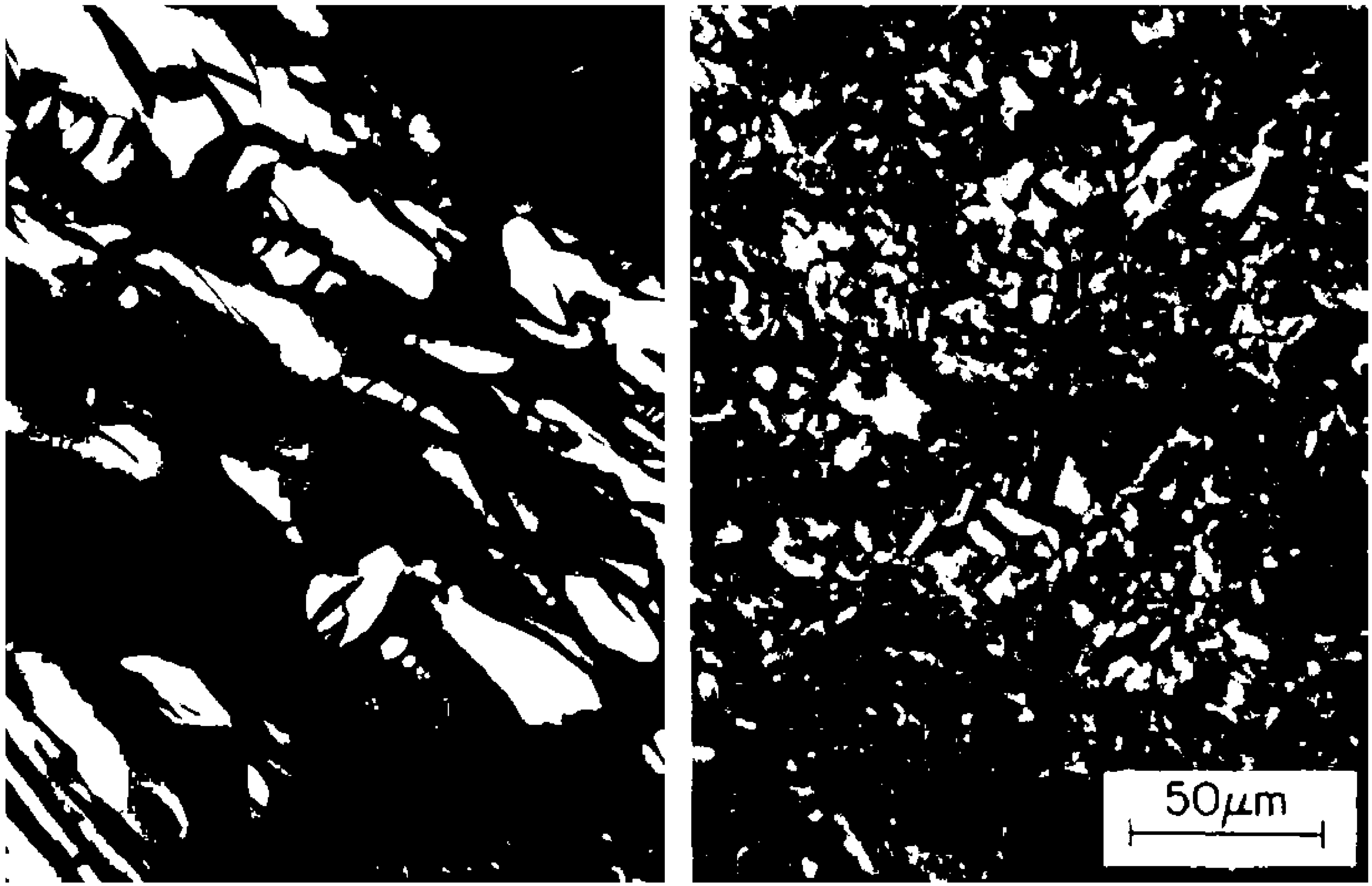
Sciences of Carbon Materials, Marsh & Reinoso

Coke Particles



Shape Indicates Domain Orientation

Optical Domain- Coke Particle



Needle **500x** Isotropic

Optical Domain

Individual Particle
500 X

Manufactured Graphite
400 X

Porosity in Graphite

Graphite single crystal density = 2.26 g/cc
Synthetic graphite bulk density = 1.6-1.9 g/cc
Most graphite contains ~>20% porosity
> 60% of porosity is open

Three classes of porosity may be identified in synthetic graphite:

1. Those formed by incomplete filling of voids in the green body by the impregnant pitch, the voids originally occur during mixing and forming;

2. Gas entrapment pores formed from binder phase pyrolysis gases during the baking stage of manufacture;

3. Thermal cracks formed by the anisotropic shrinkage of the crystals in the filler coke and binder.

Graphite Structural Features

Lattice (a= 2.45 A, c= 6.7 Å)

Crystallite "Coherent Domain"

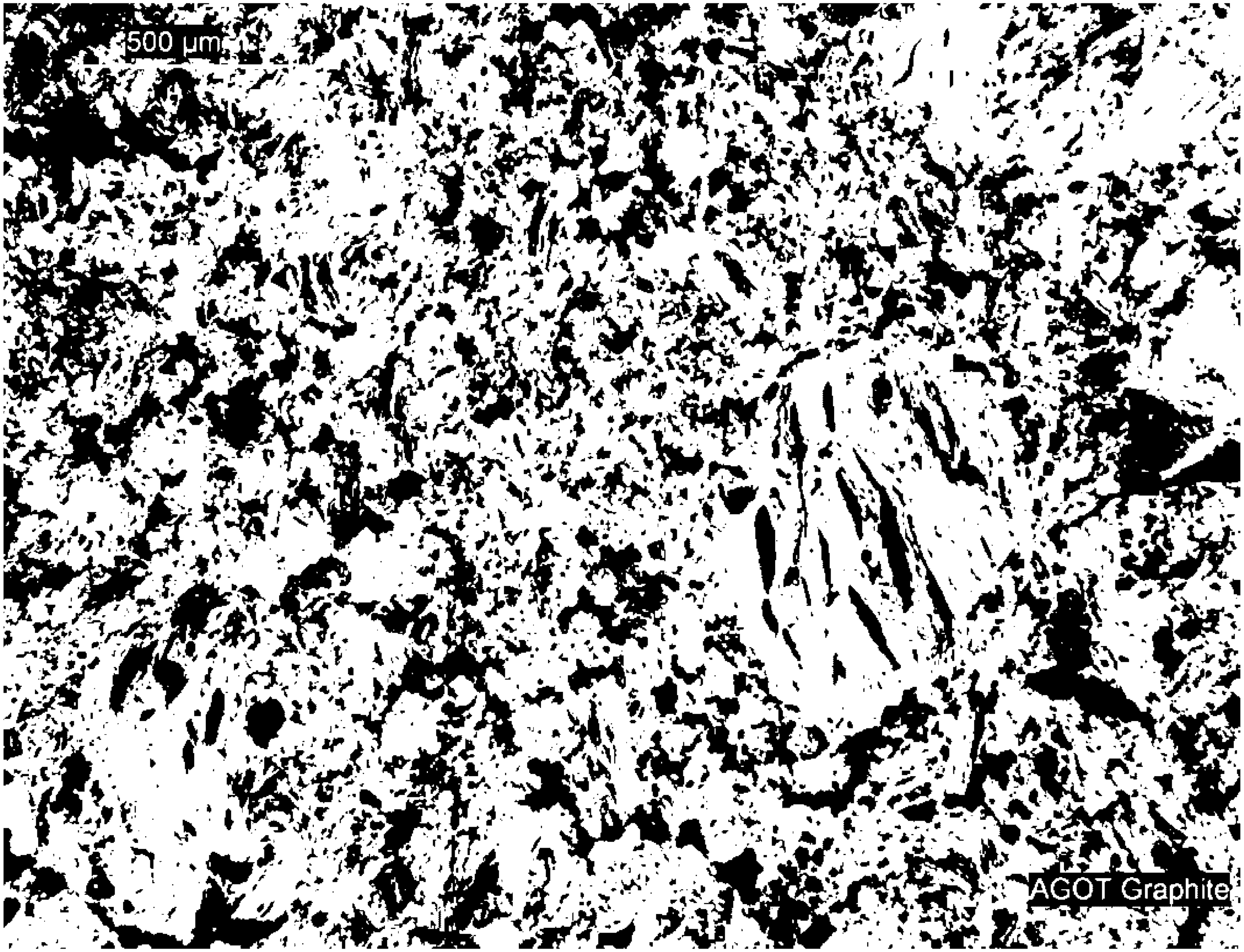
Micro-crack (between planes, about the size of crystallite)

Optical Domain (extended orientation of crystallites)

Grain Size

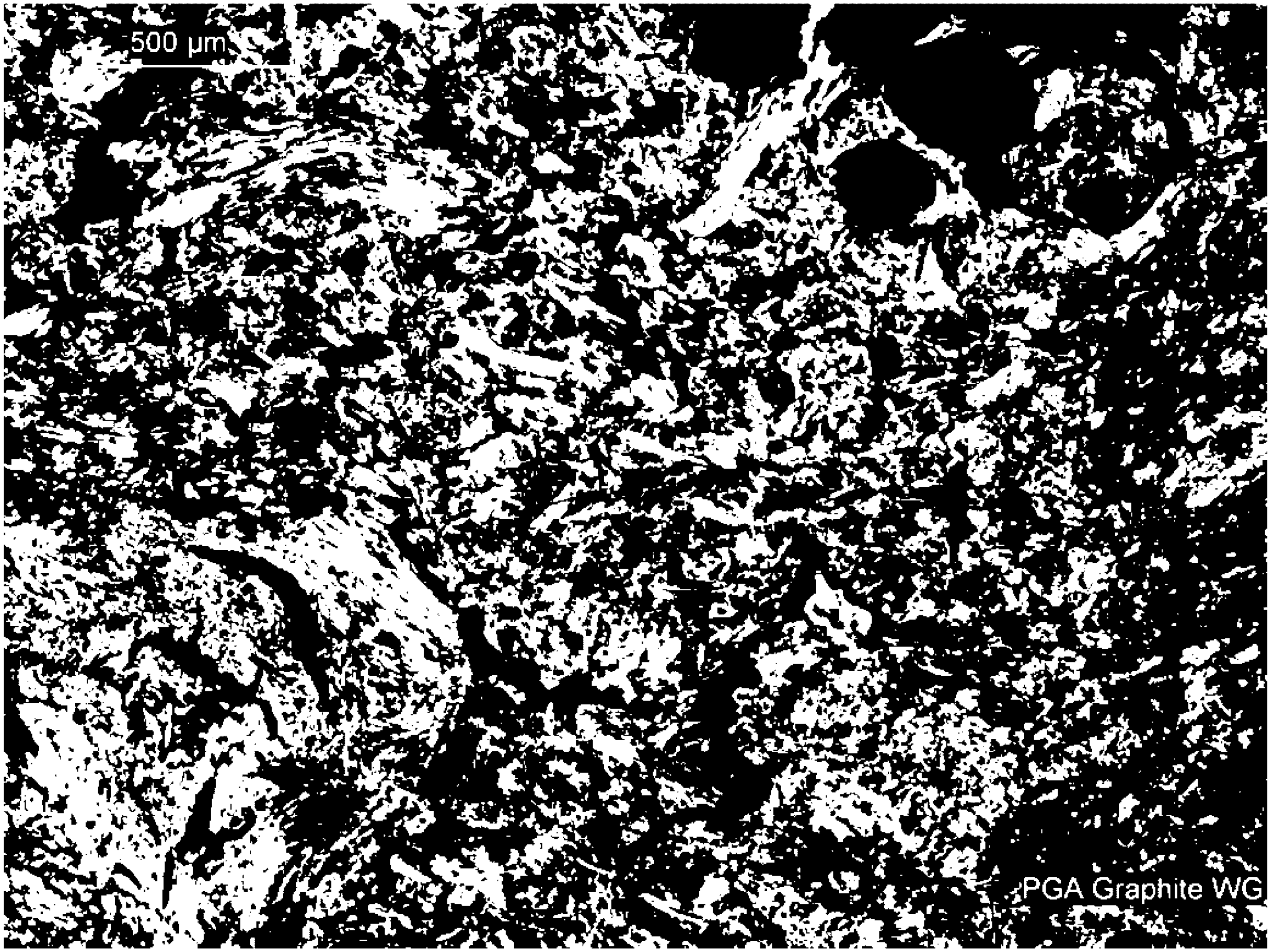
Pore Size

Synthetic Graphite Microstructure



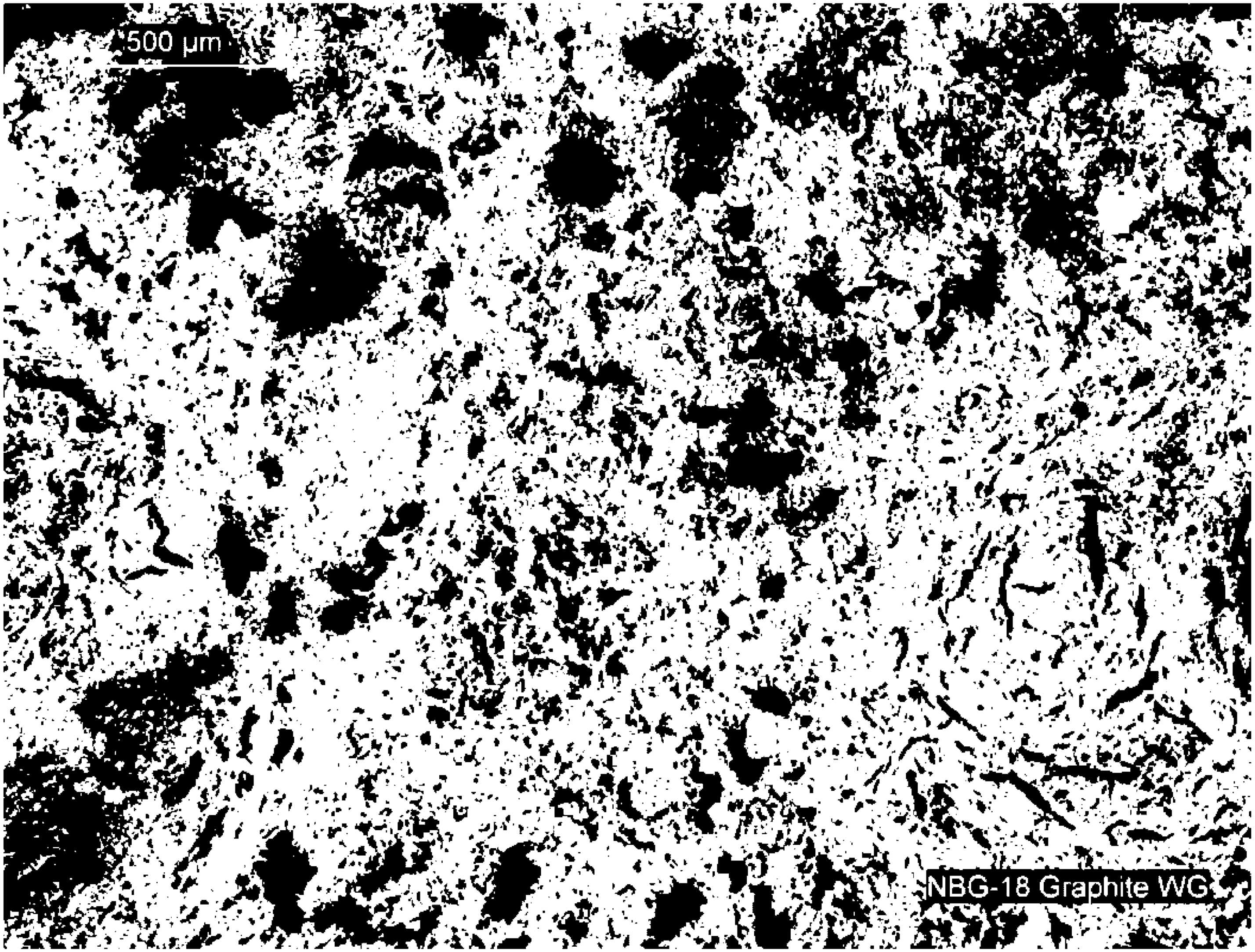
Grade AGOT graphite microstructure (viewed under polarized light)

Synthetic Graphite Microstructure



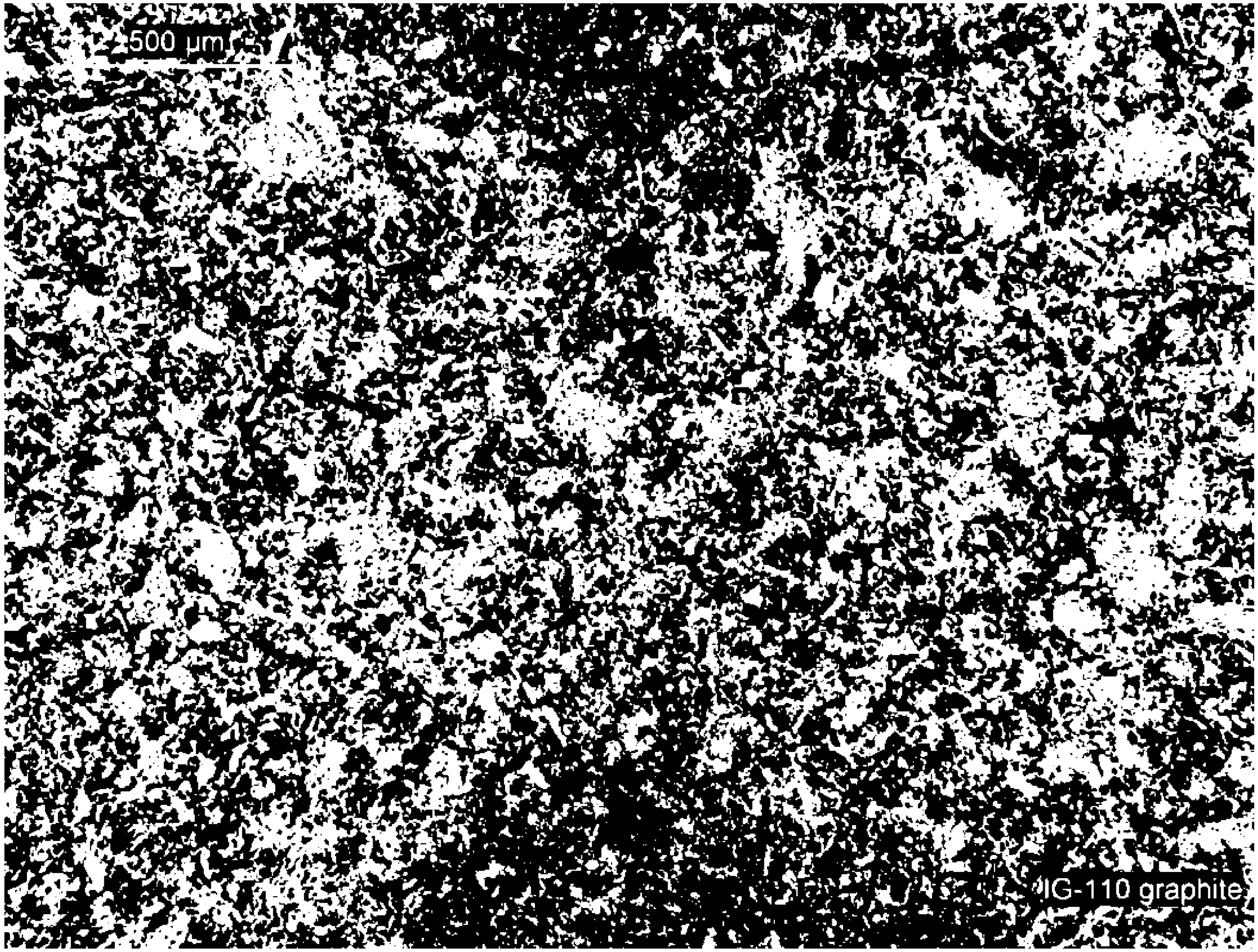
Grade PGA graphite (with-grain) microstructure (viewed under polarized light)

Synthetic Graphite Microstructure



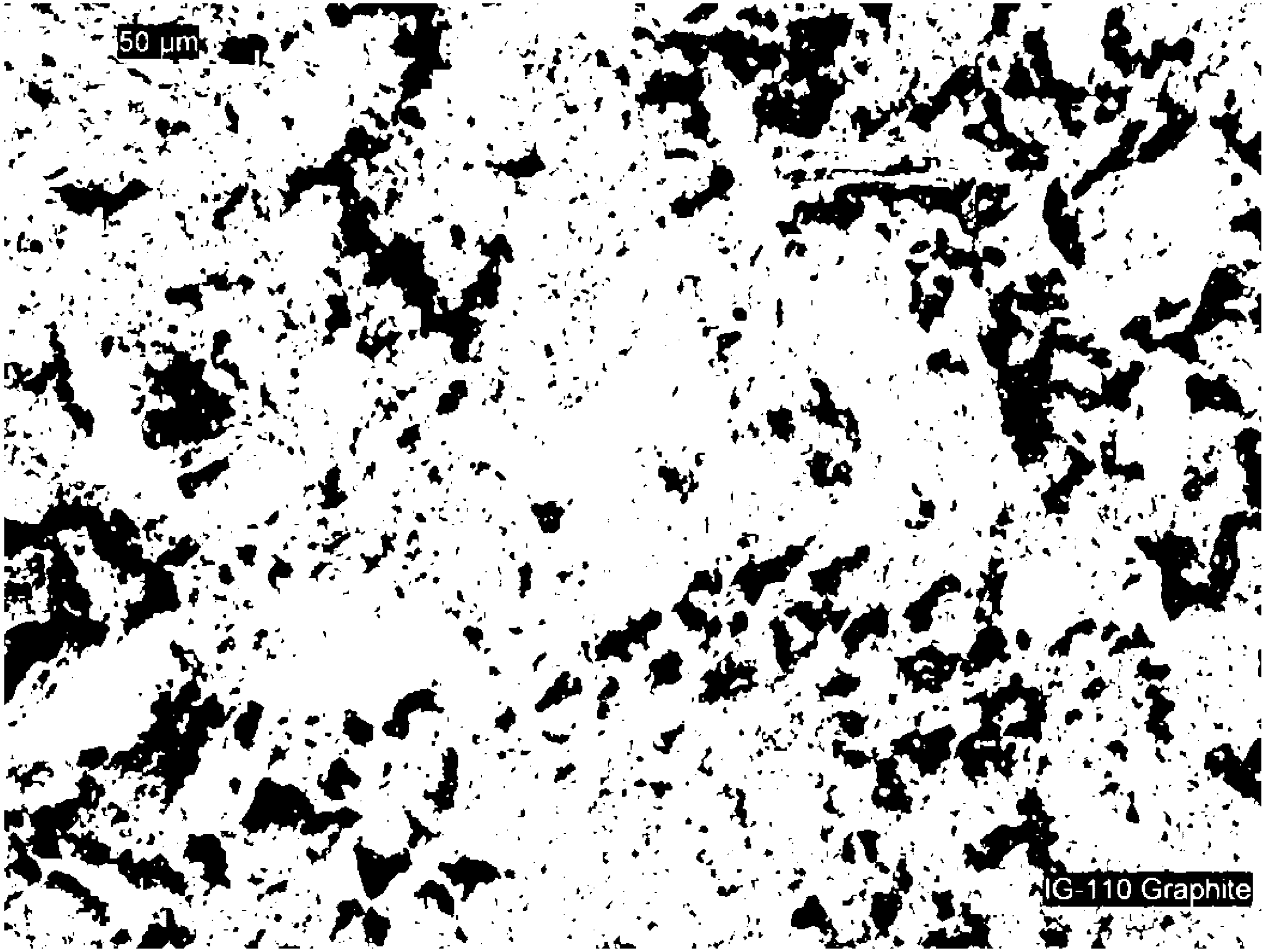
Grade NBG-18 graphite (with-grain) microstructure (viewed under polarized light)

Synthetic Graphite Microstructure



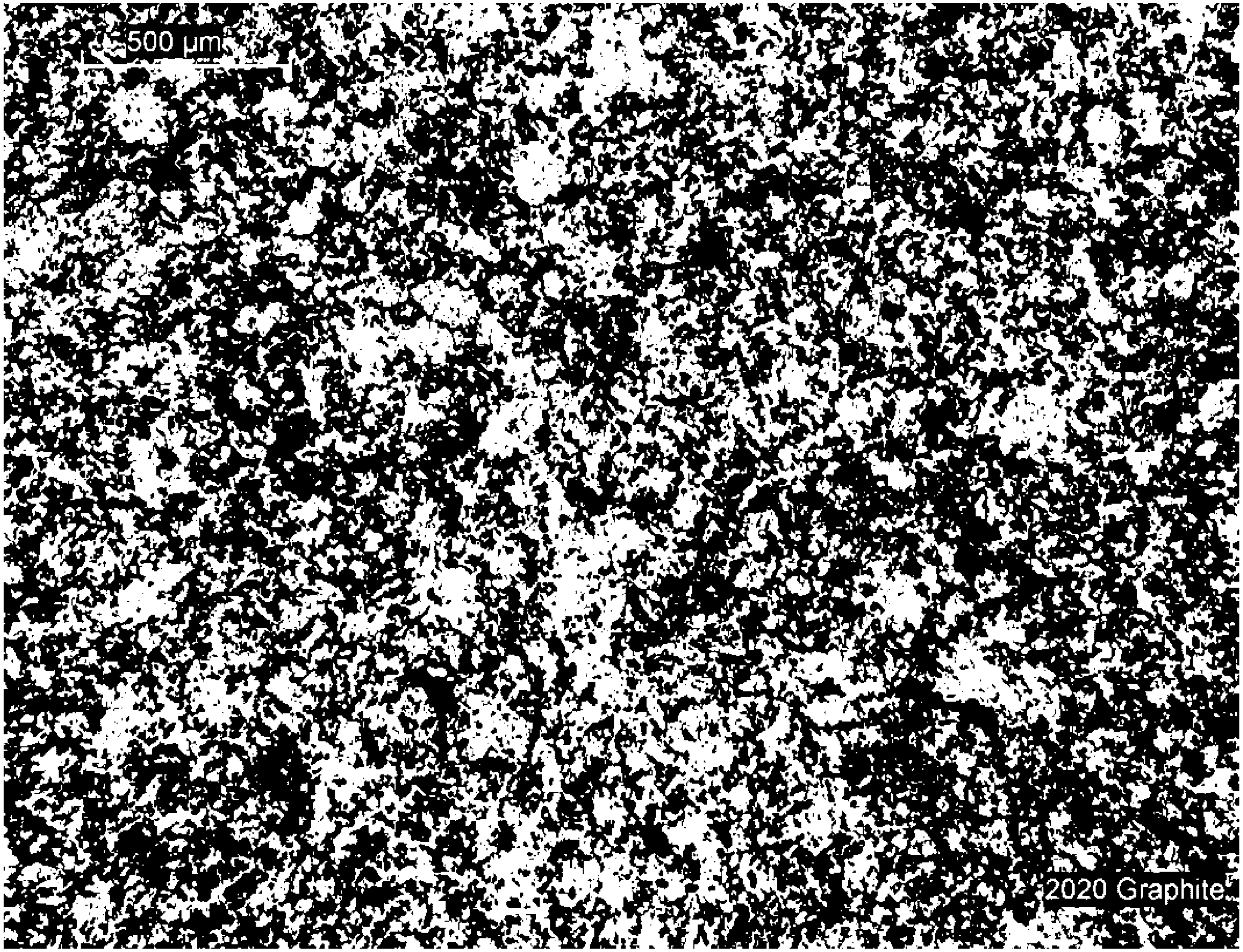
Grade IG-110 graphite microstructure (viewed under polarized light)

Synthetic Graphite Microstructure



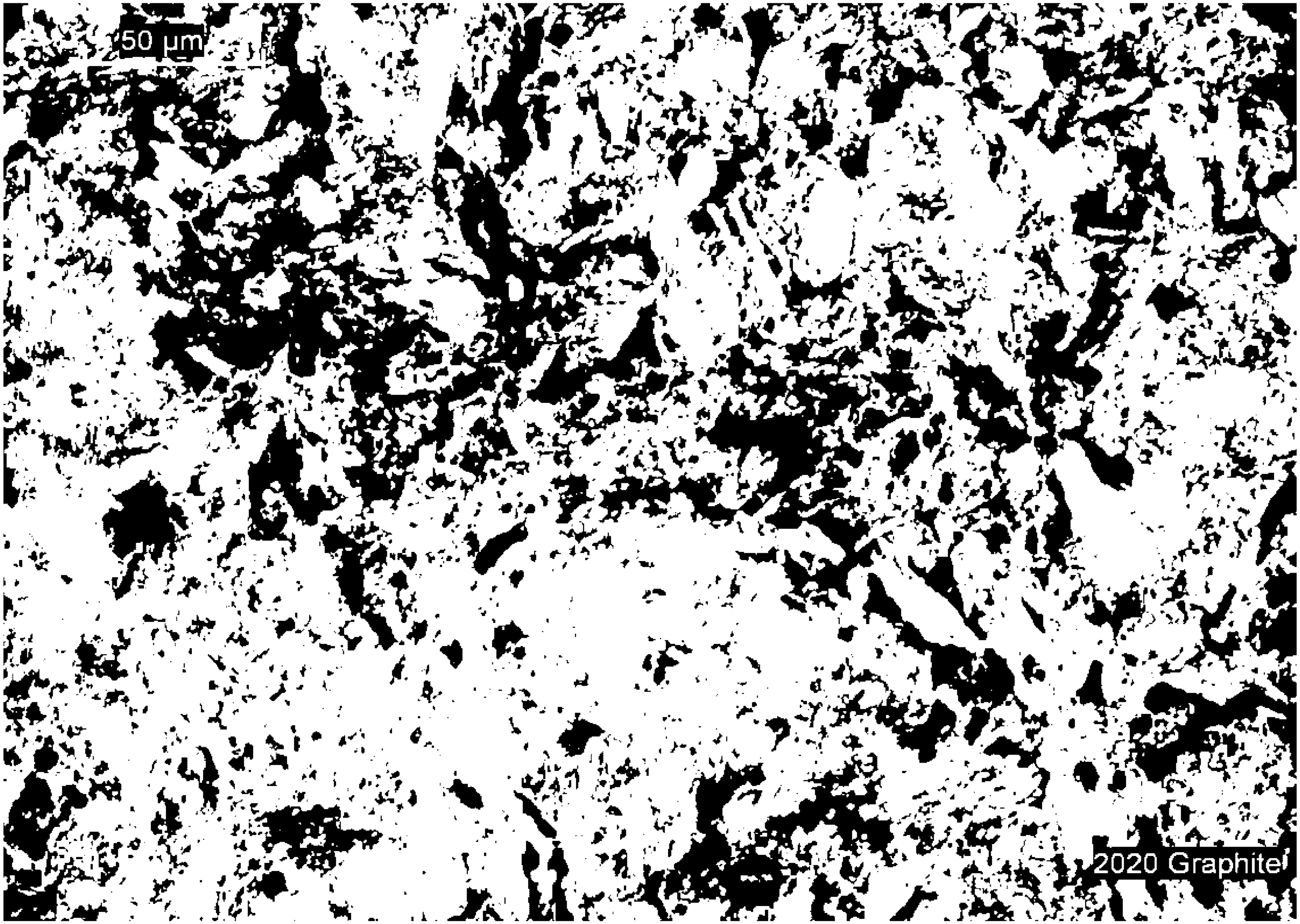
Grade IG-110 graphite microstructure (viewed under polarized light)

Synthetic Graphite Microstructure



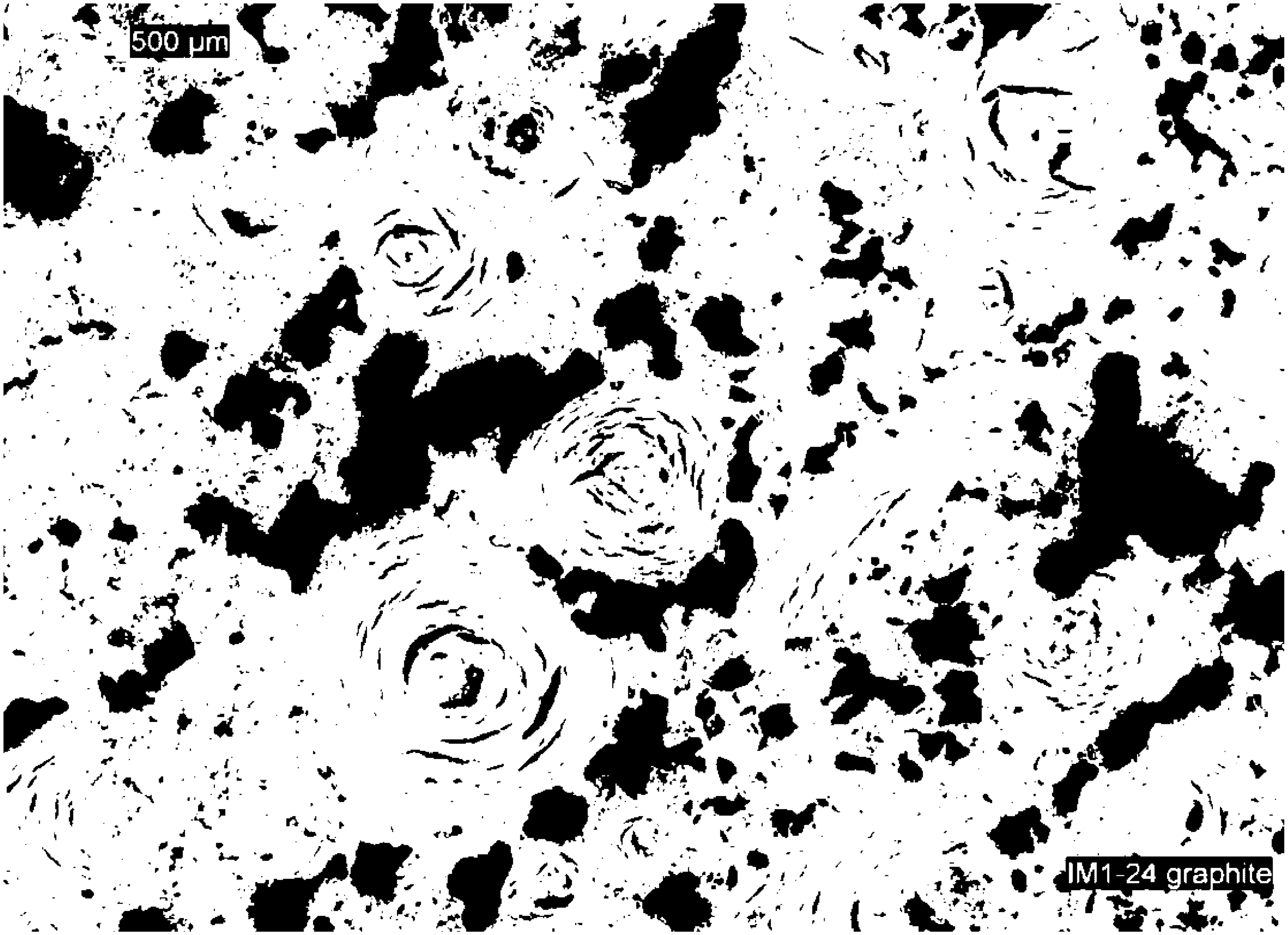
Grade 2020 graphite microstructure (viewed under polarized light)

Synthetic Graphite Microstructure



Grade 2020 graphite microstructure (viewed under polarized light)

Synthetic Graphite Microstructure



Grade IM1-24 graphite microstructure (viewed under polarized light)

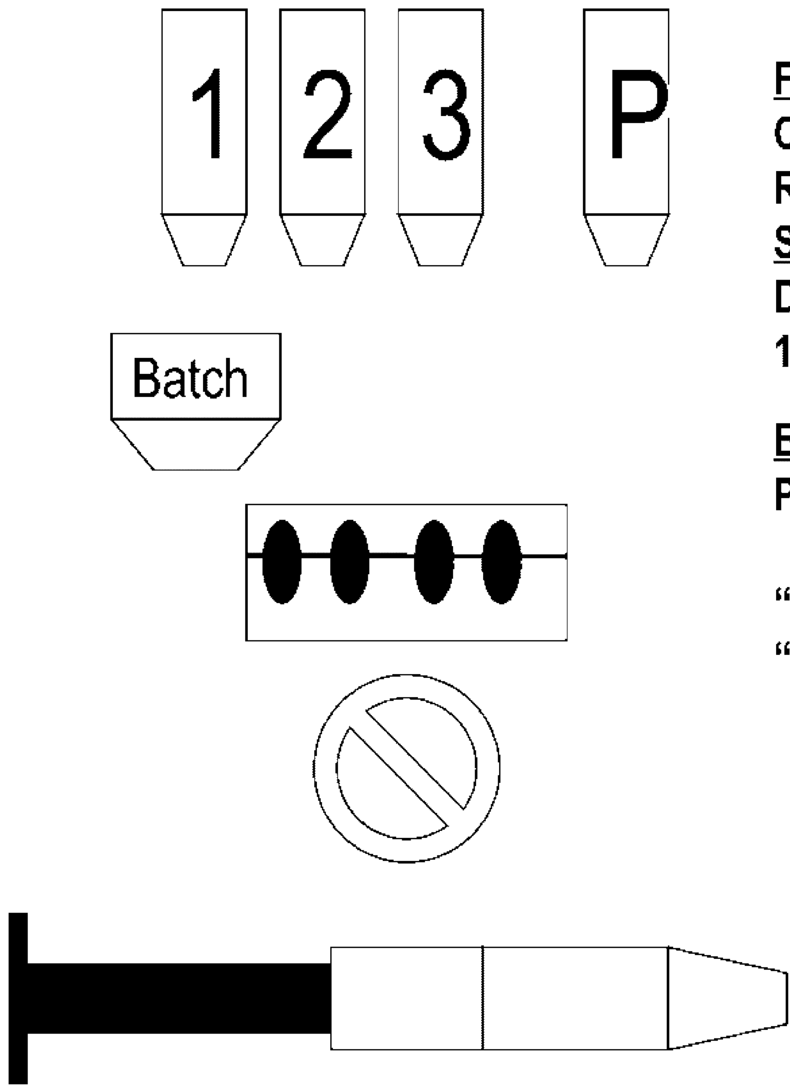
Synthetic Graphite Texture

Texture in synthetic graphite arises because of:

- •Crystal anisotropy, coke and binder domain size
- •Filler cokes and binder cracks
- •Size and shape distribution of filler particle
- •Filler coke type
- •Recycle fraction and morphology
- •Porosity
- •Forming method (preferential orientation of filler coke and binder porosity)

Texture imparts anisotropy!

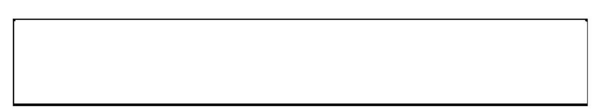
Graphite Production -Extrusion



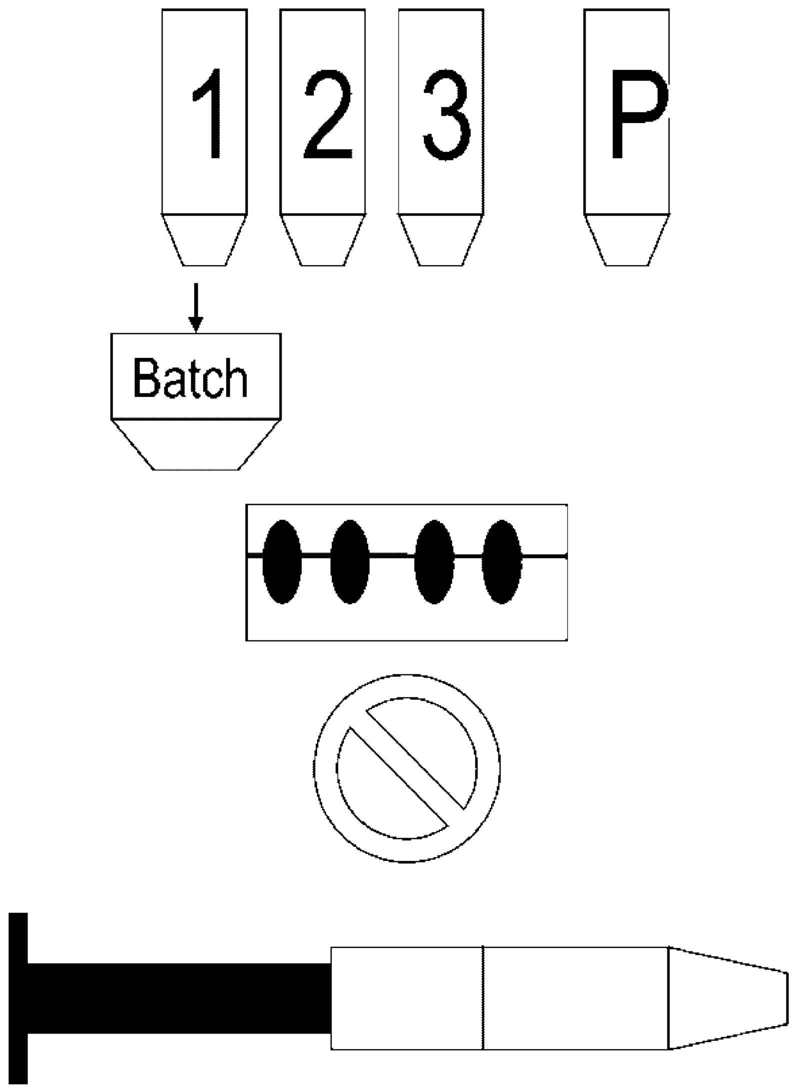
<u>Filler Materials:</u>
Calcined Coke, (Raw Coke)
Recycle Graphite
<u>Sizing:</u>
Designed Combination of Discrete Fractions
10-1,000 microns

<u>Binder:</u>
Pitch (20-45 parts/hundred)

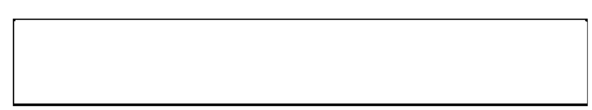
"Remix"
"Additives"



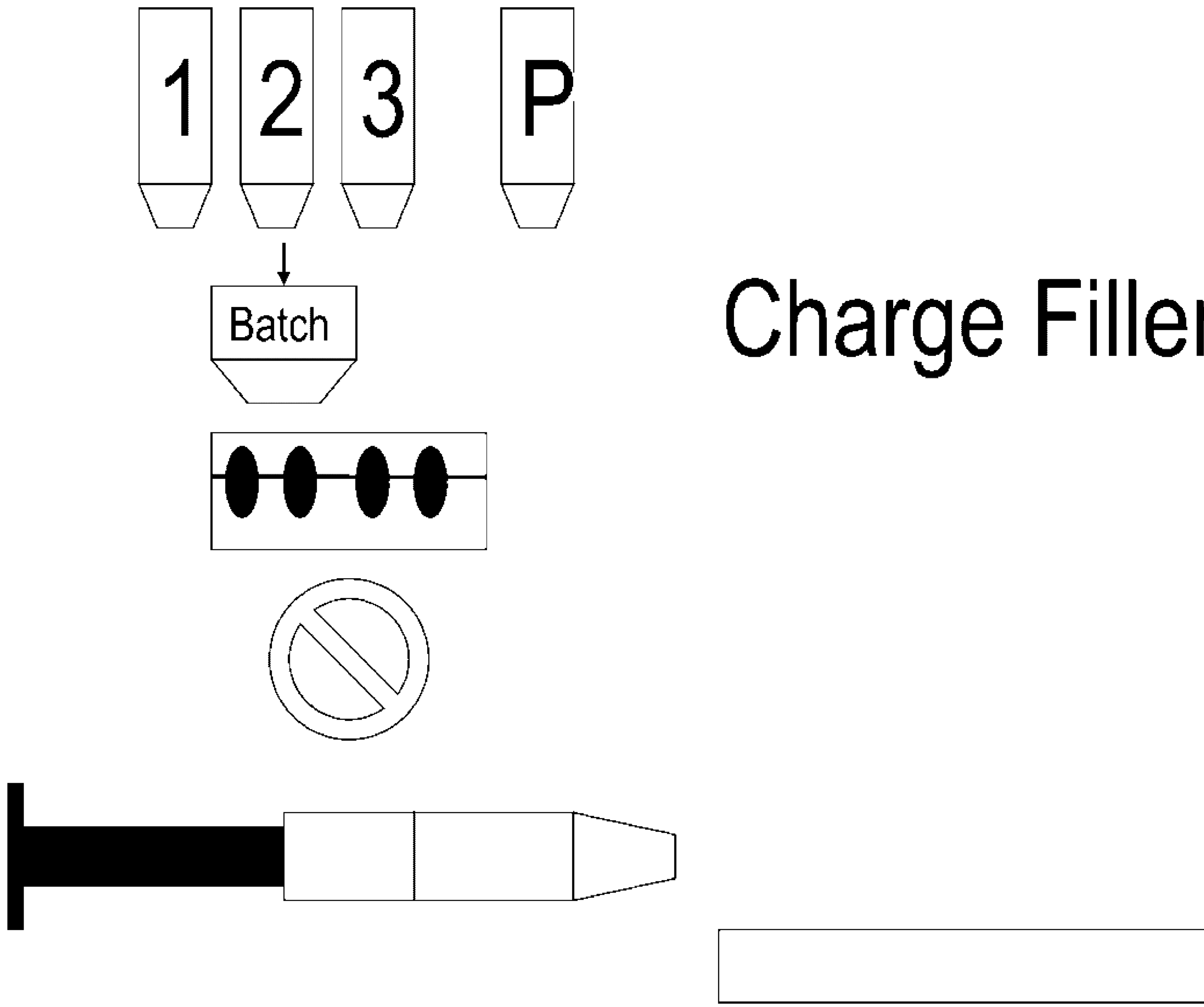
Graphite Production -Extrusion



Charge Filler

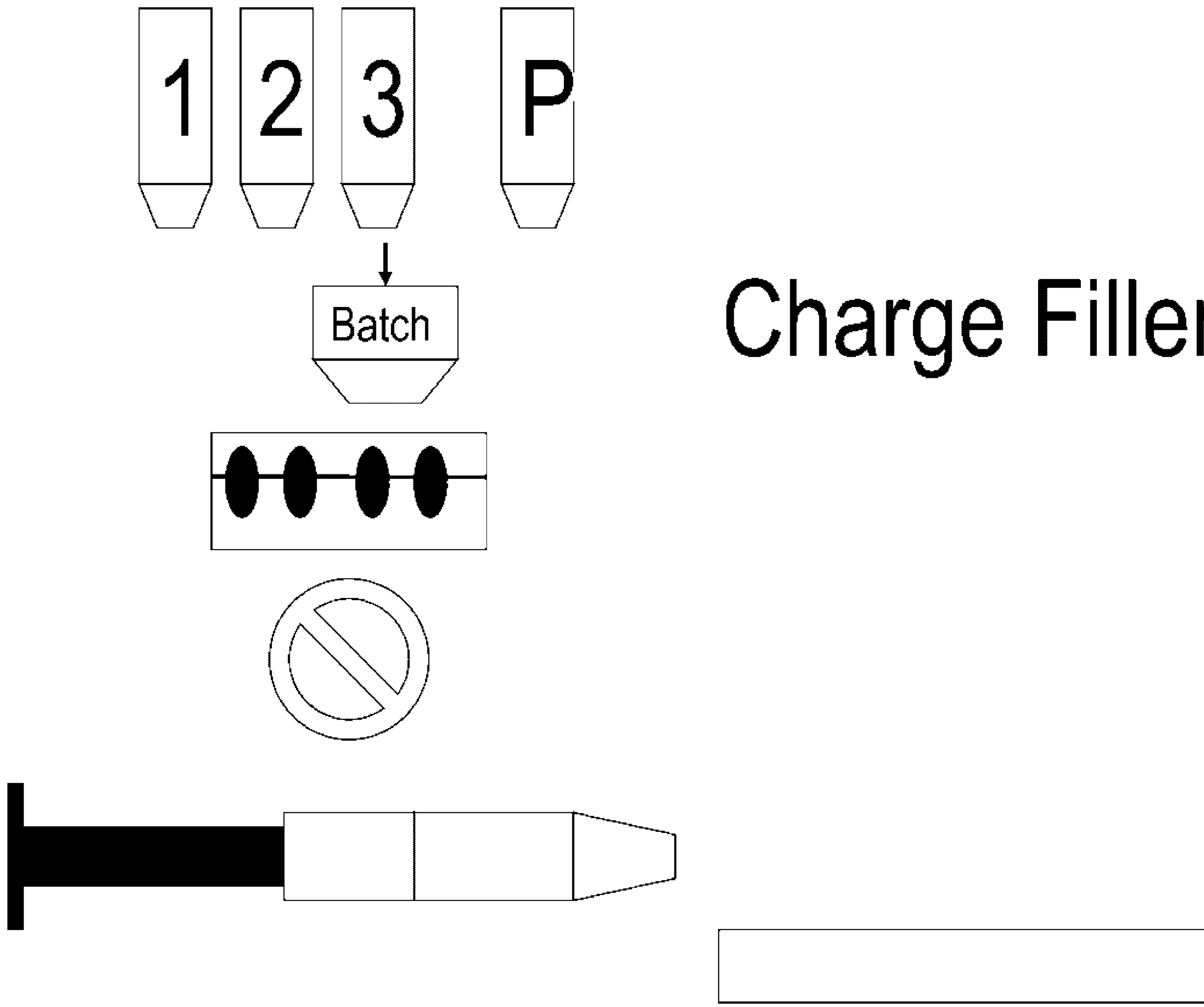


Graphite Production -Extrusion



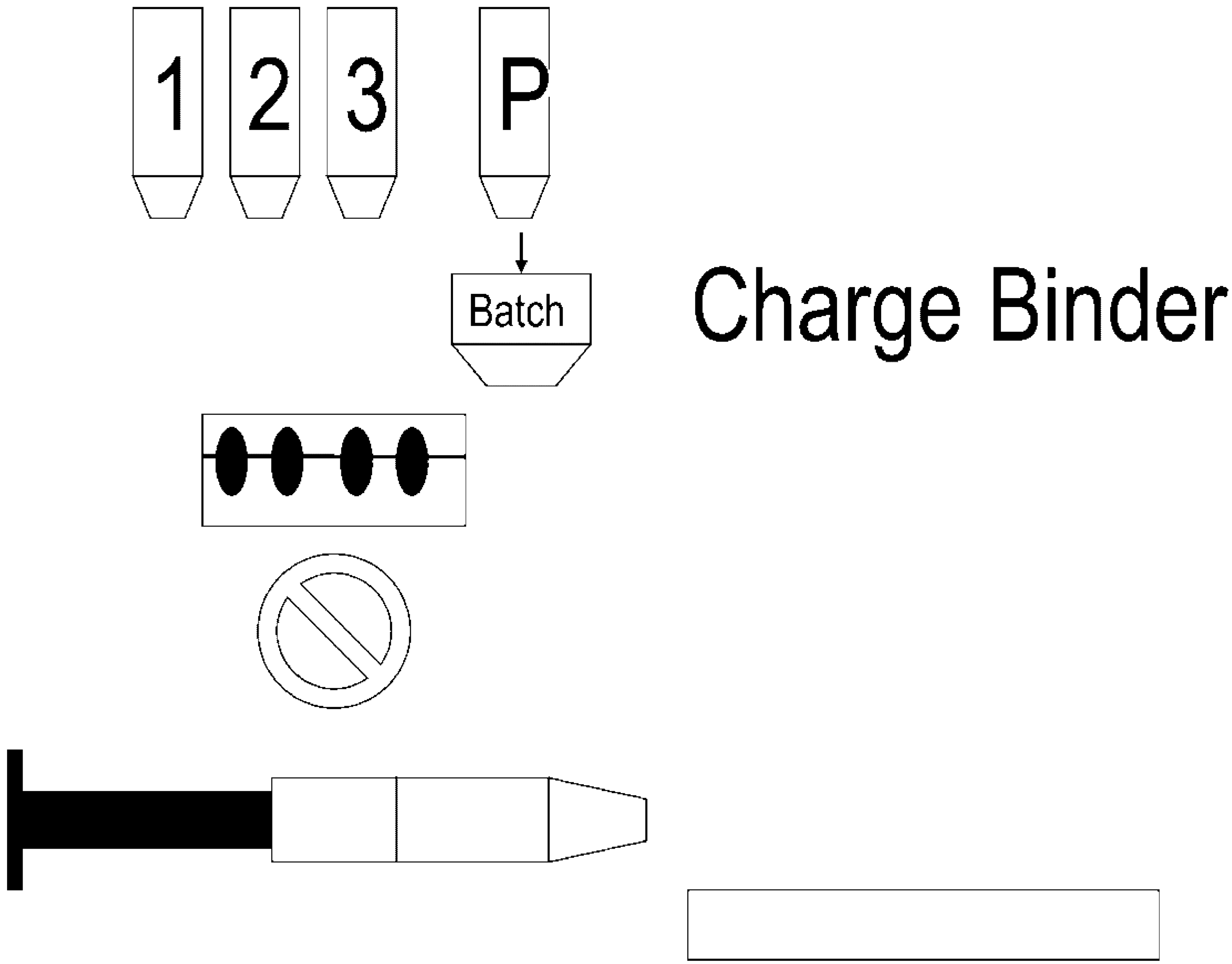
Charge Filler

Graphite Production -Extrusion

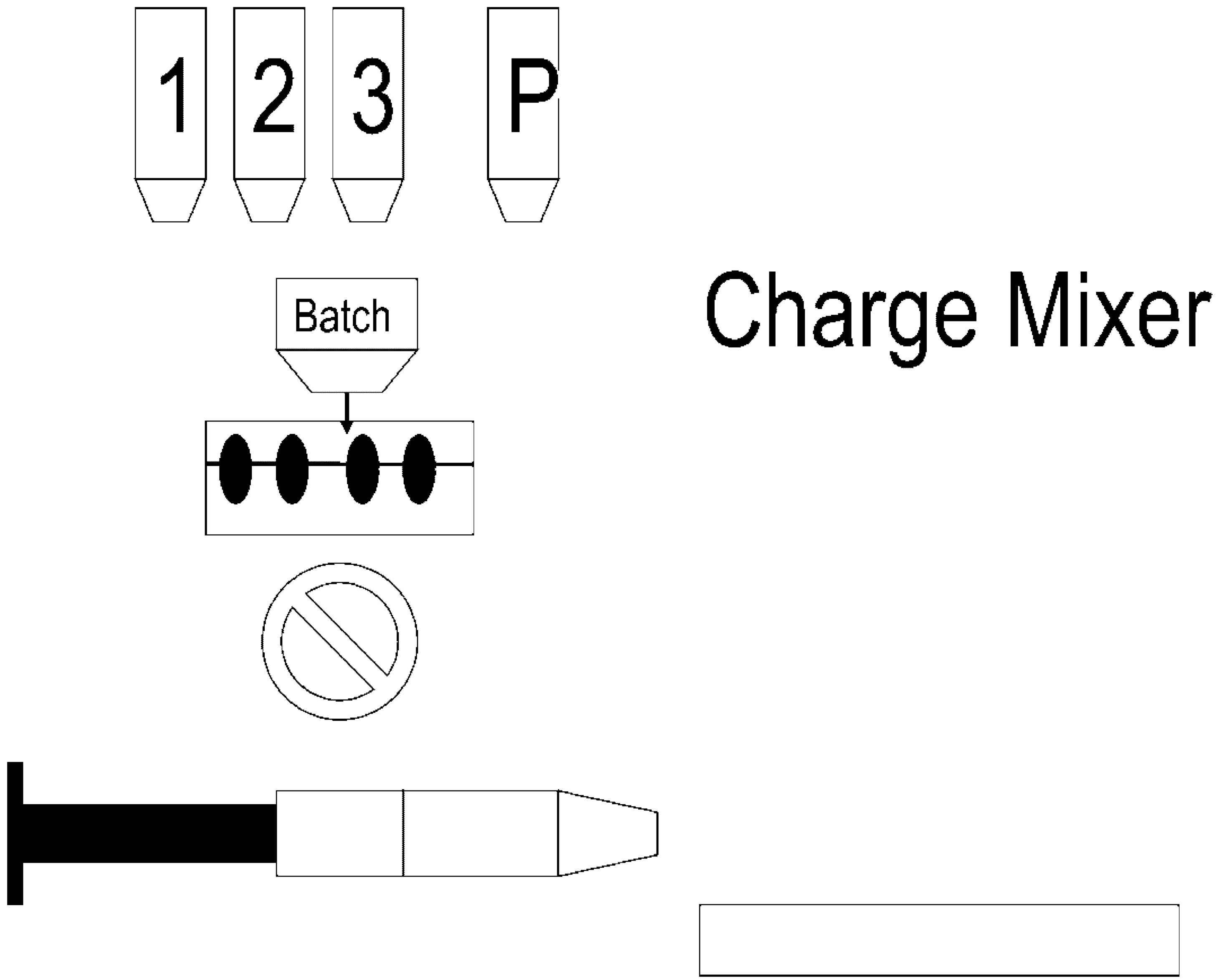


Charge Filler

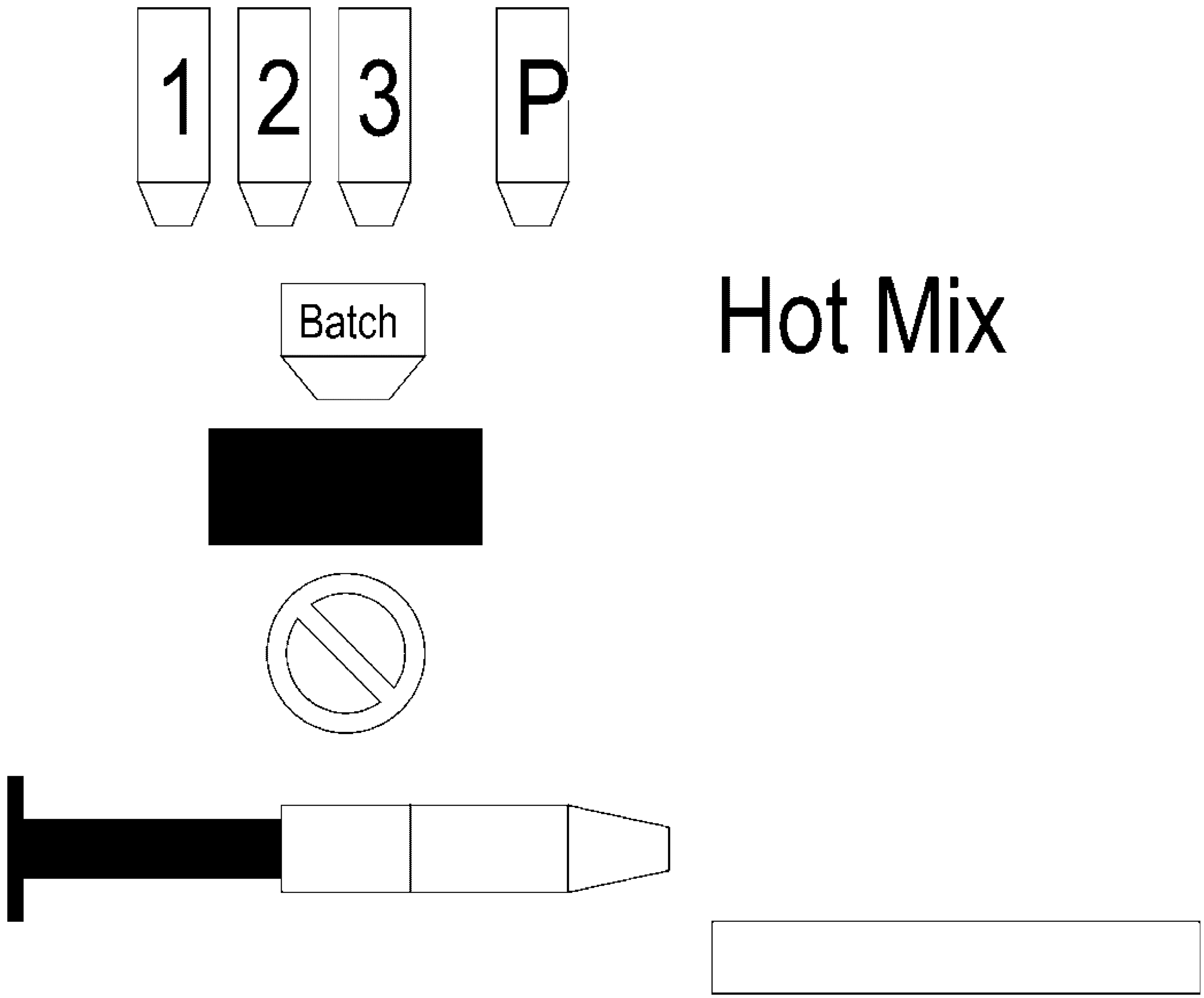
Graphite Production -Extrusion



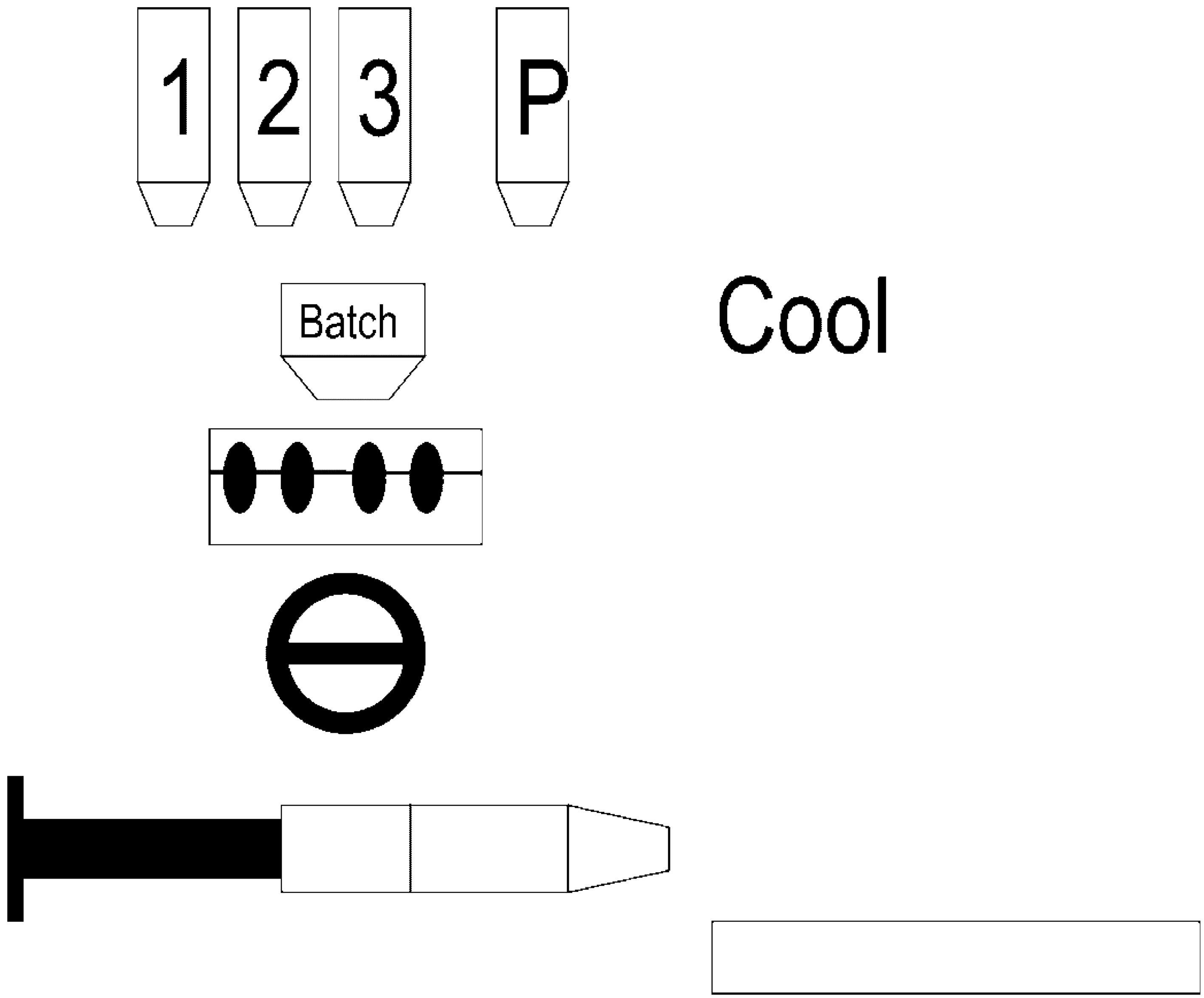
Graphite Production -Extrusion



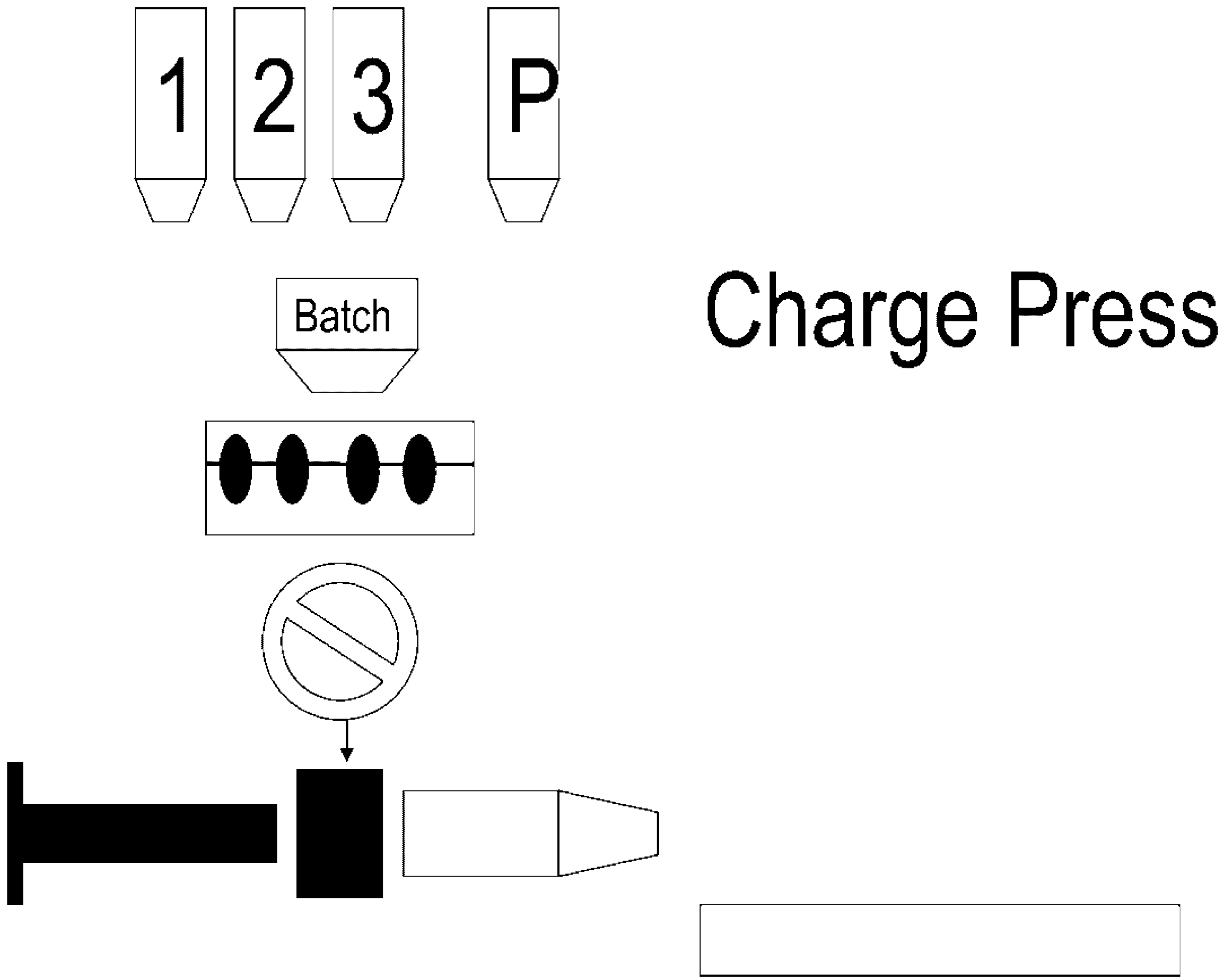
Graphite Production -Extrusion



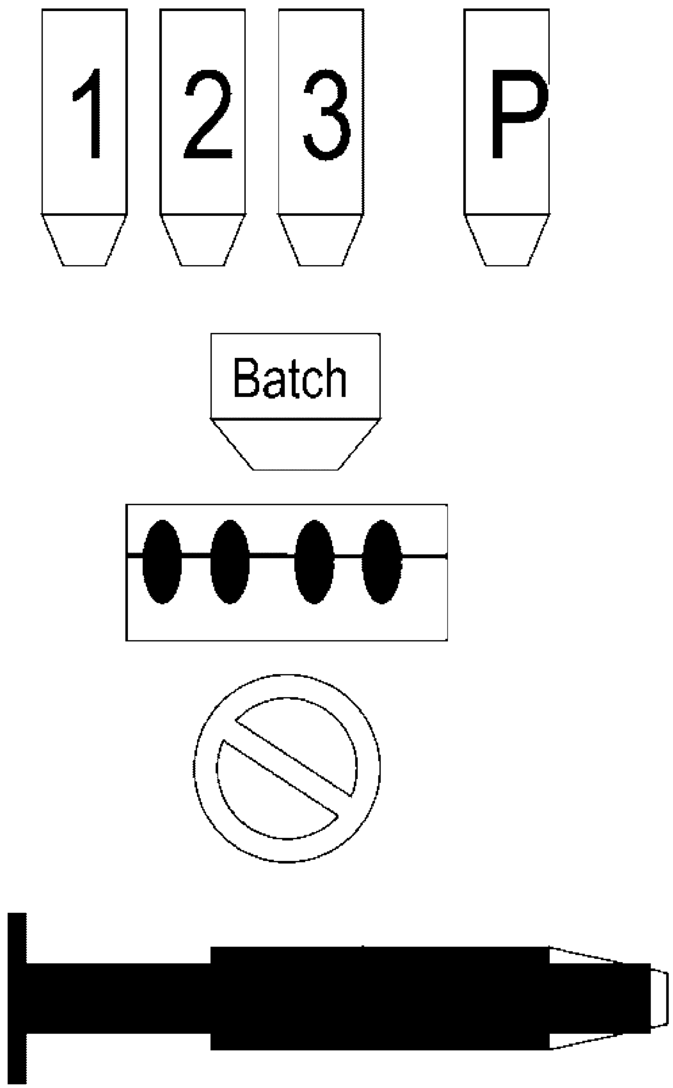
Graphite Production -Extrusion



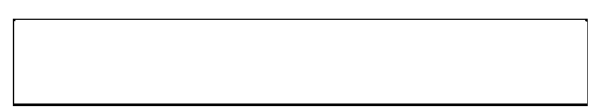
Graphite Production -Extrusion



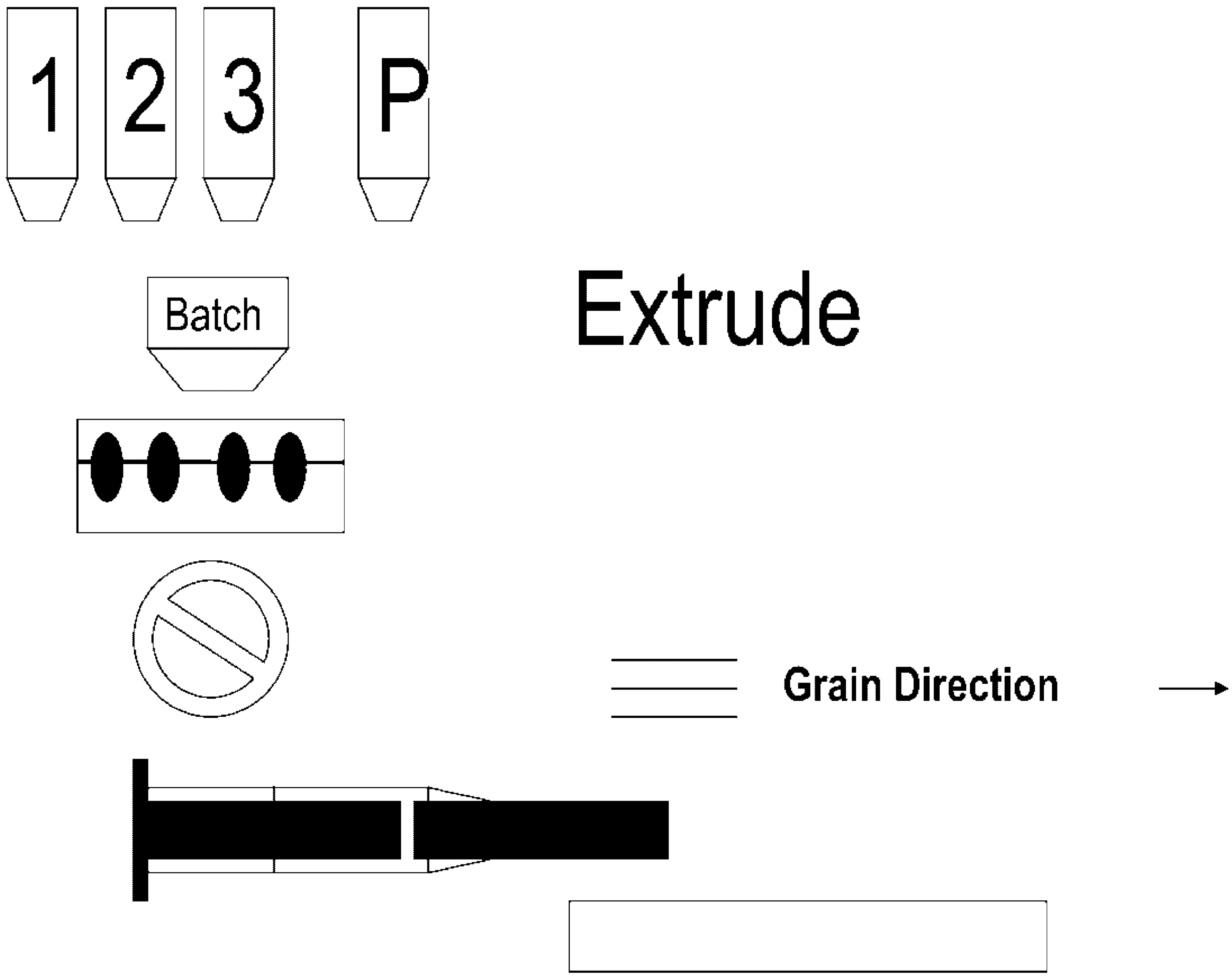
Graphite Production -Extrusion



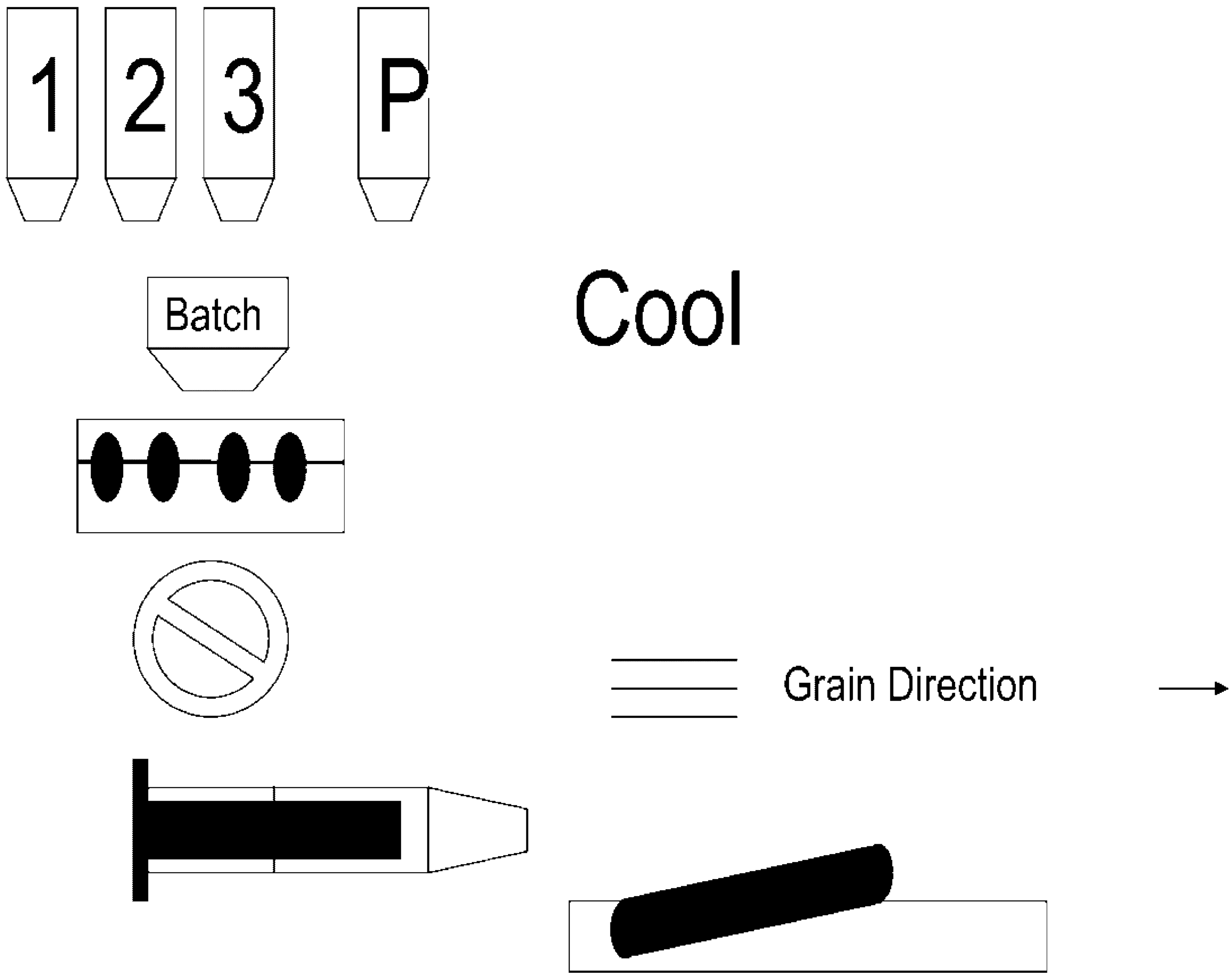
Extrude



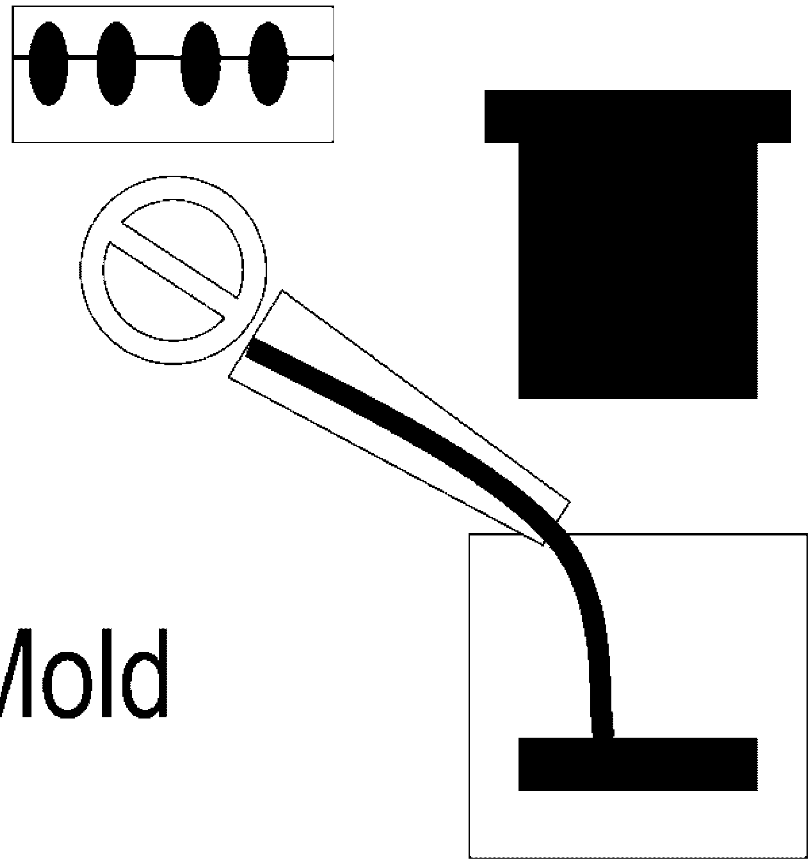
Graphite Production -Extrusion



Graphite Production -Extrusion

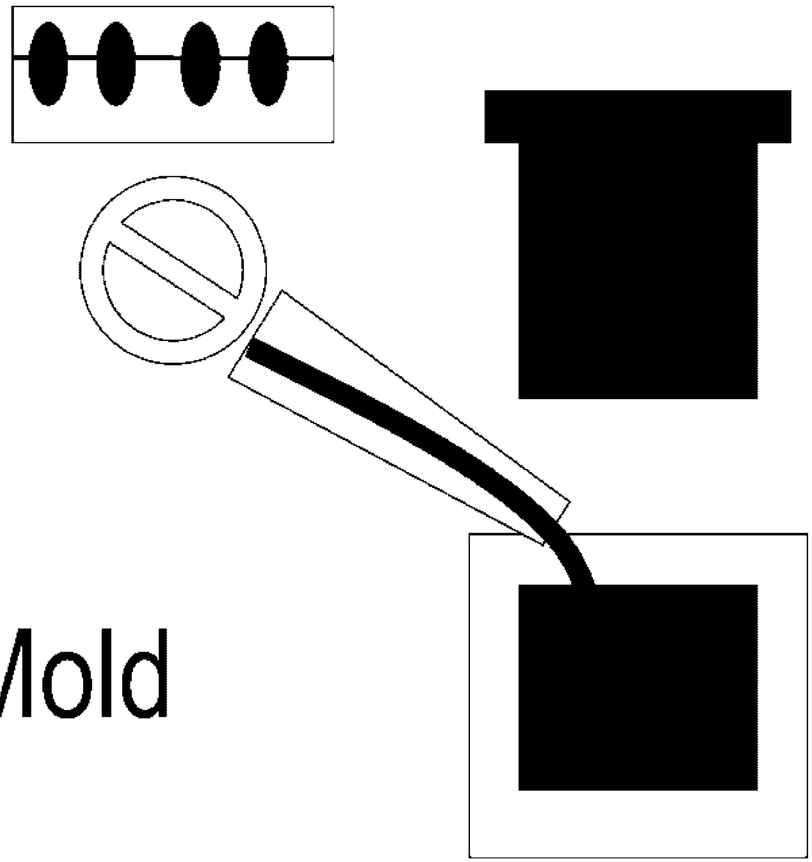


Graphite Production -Die Molding



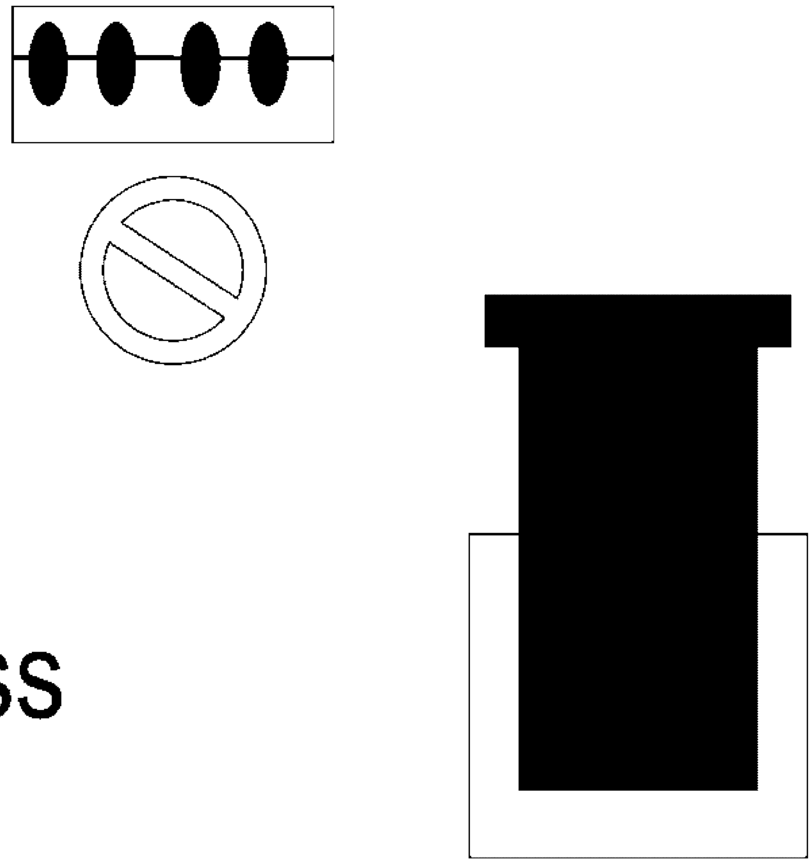
Charge Mold

Graphite Production -Die Molding



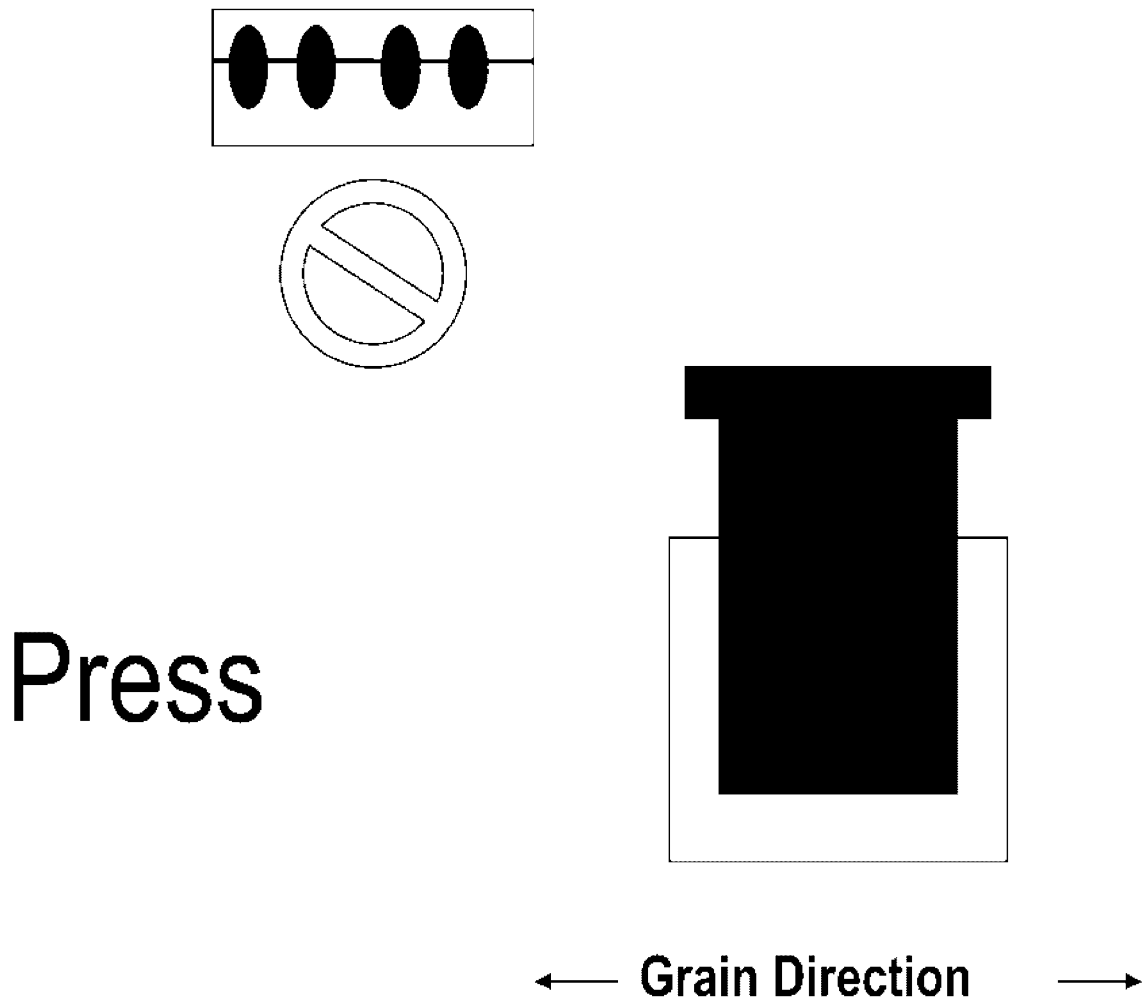
Charge Mold

Graphite Production -Die Molding

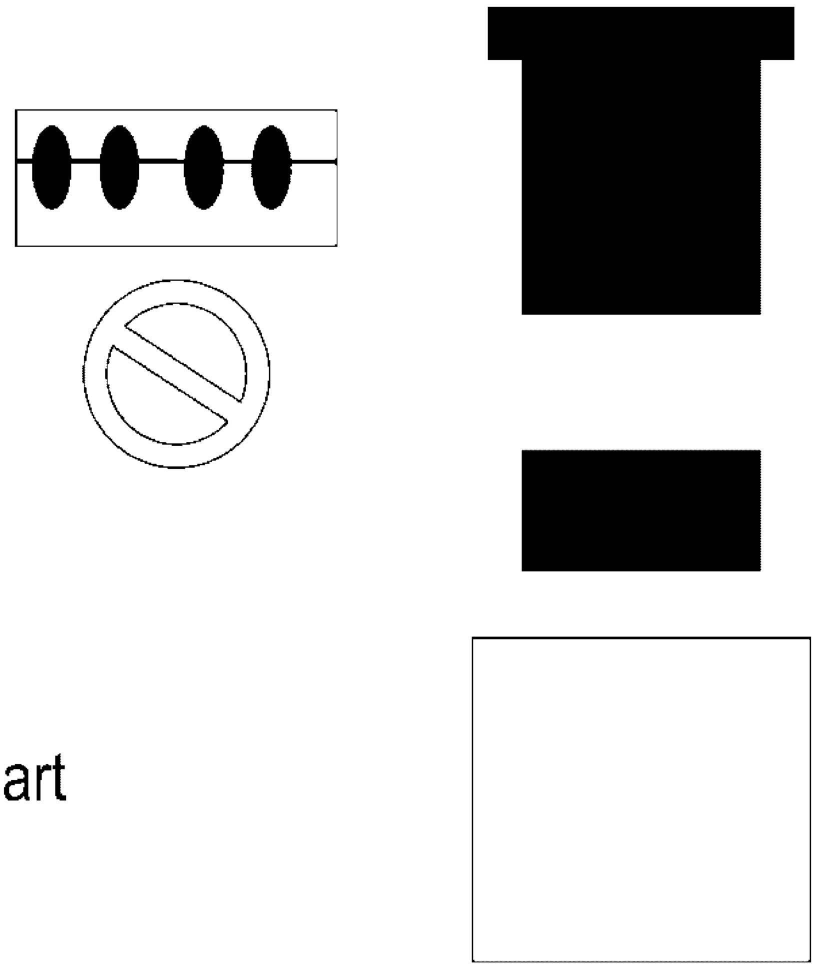


Press

Graphite Production -Die Molding



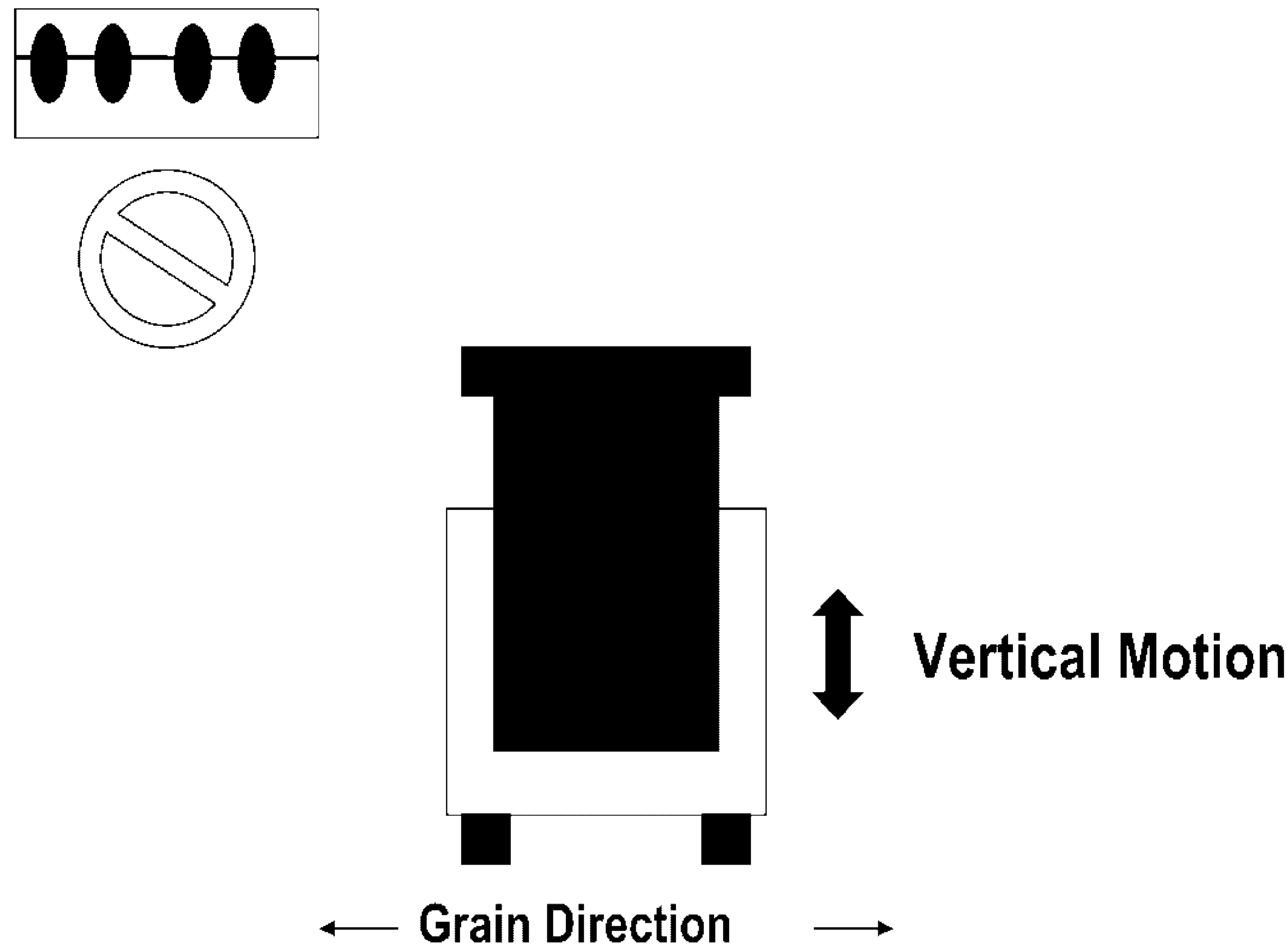
Graphite Production -Die Molding



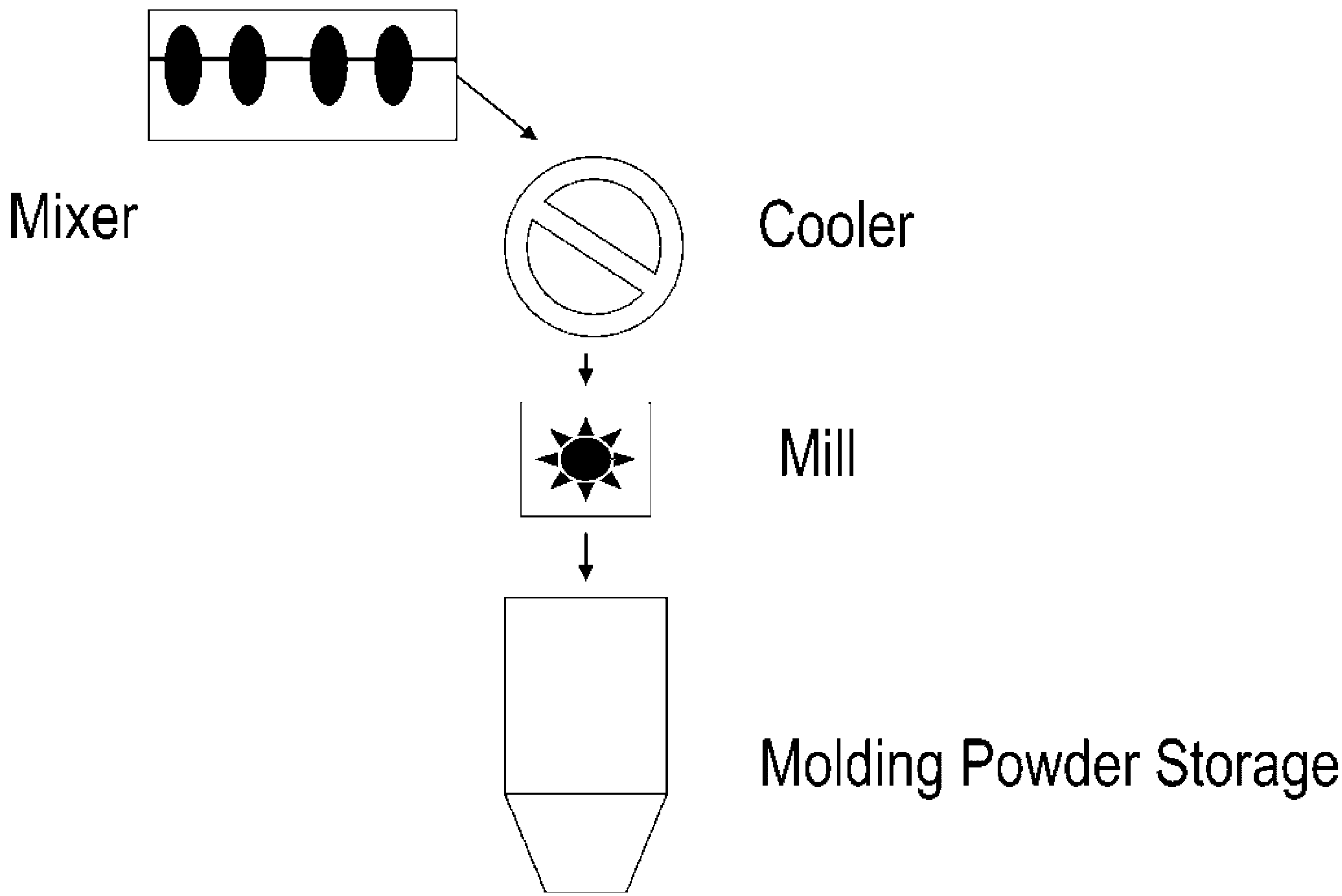
Unload Part

← Grain Direction →

Graphite Production - Vibration Molding

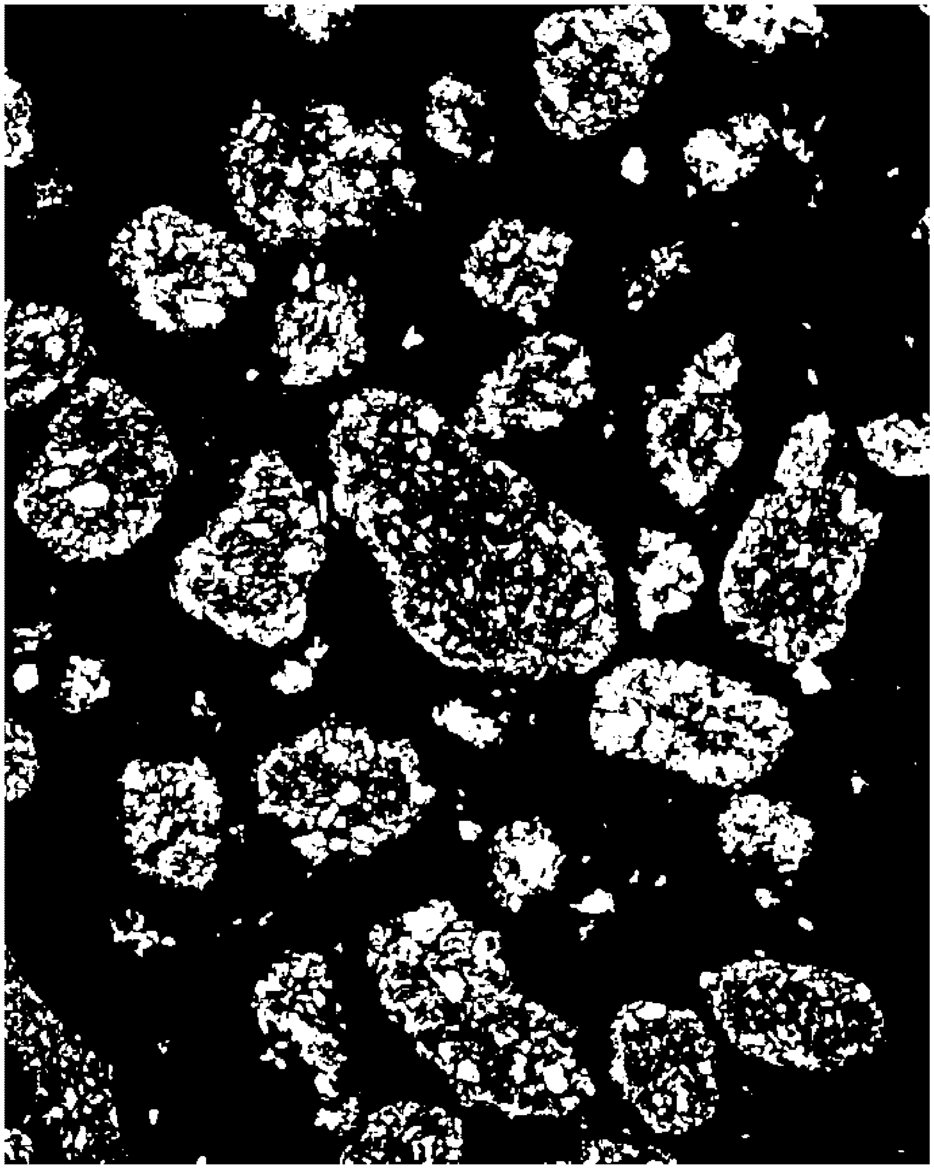


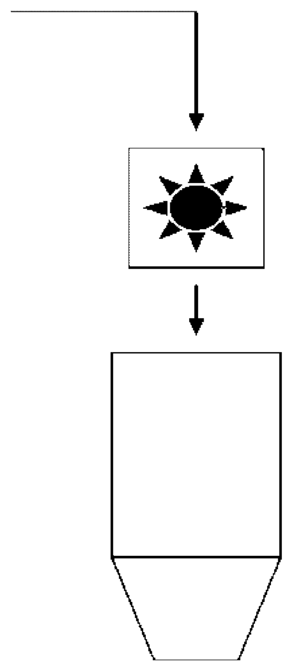
Graphite Production - Isostatic Molding



Graphite Production - Isostatic Molding

Mix Agglomerates
are Milled to
Produce Molding Powder



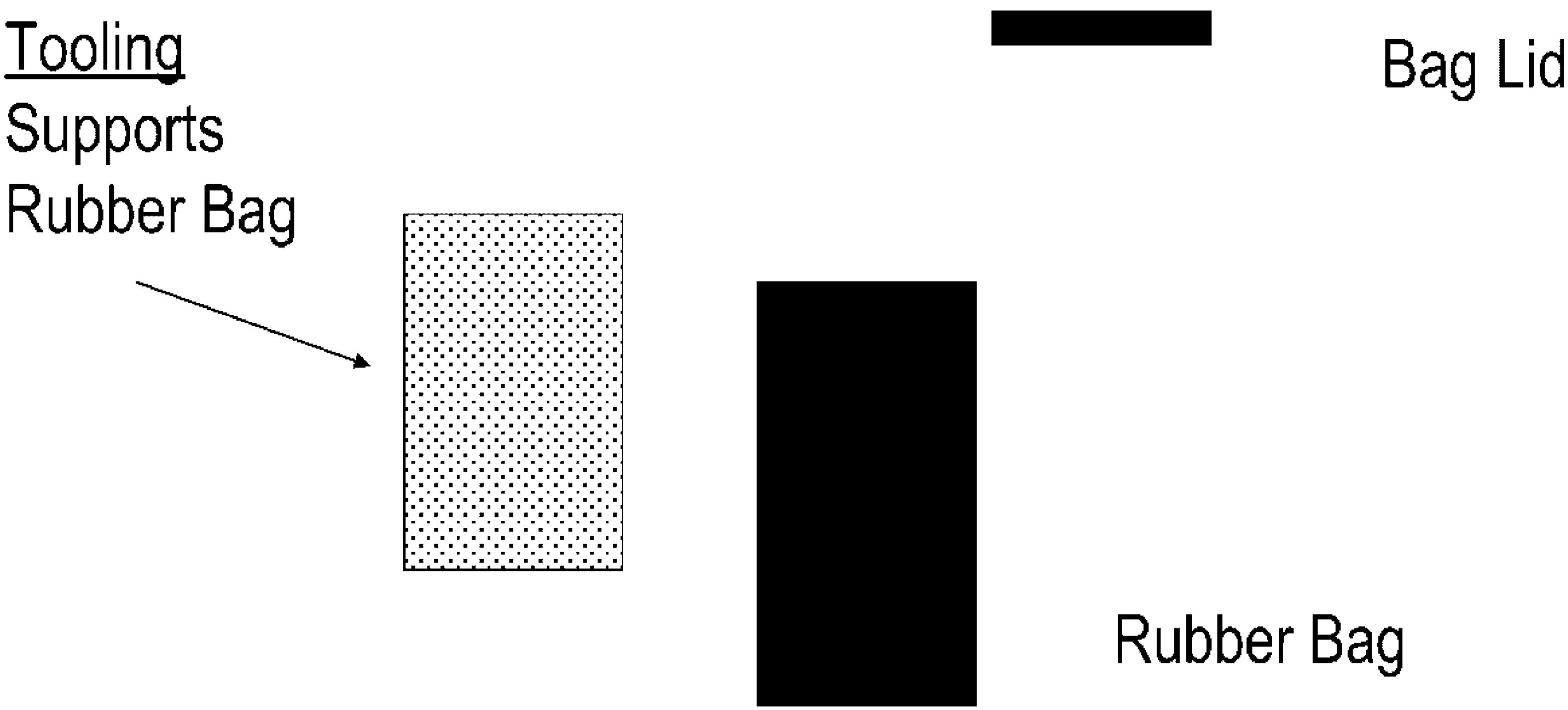


Mill

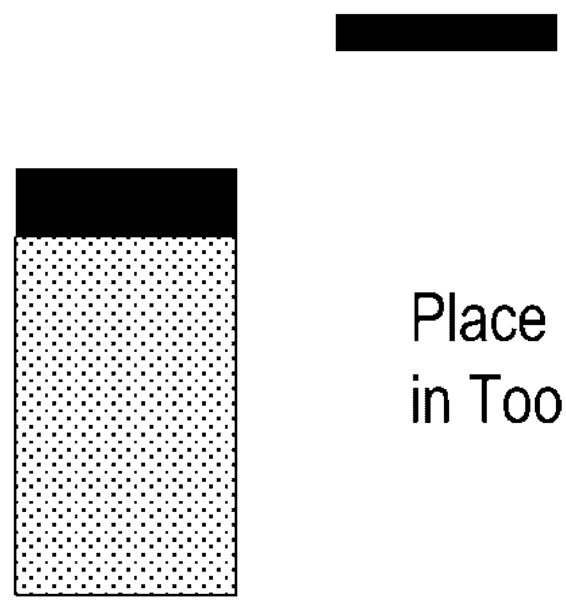
Molding Powder

Storage

Graphite Production - Isostatic Molding

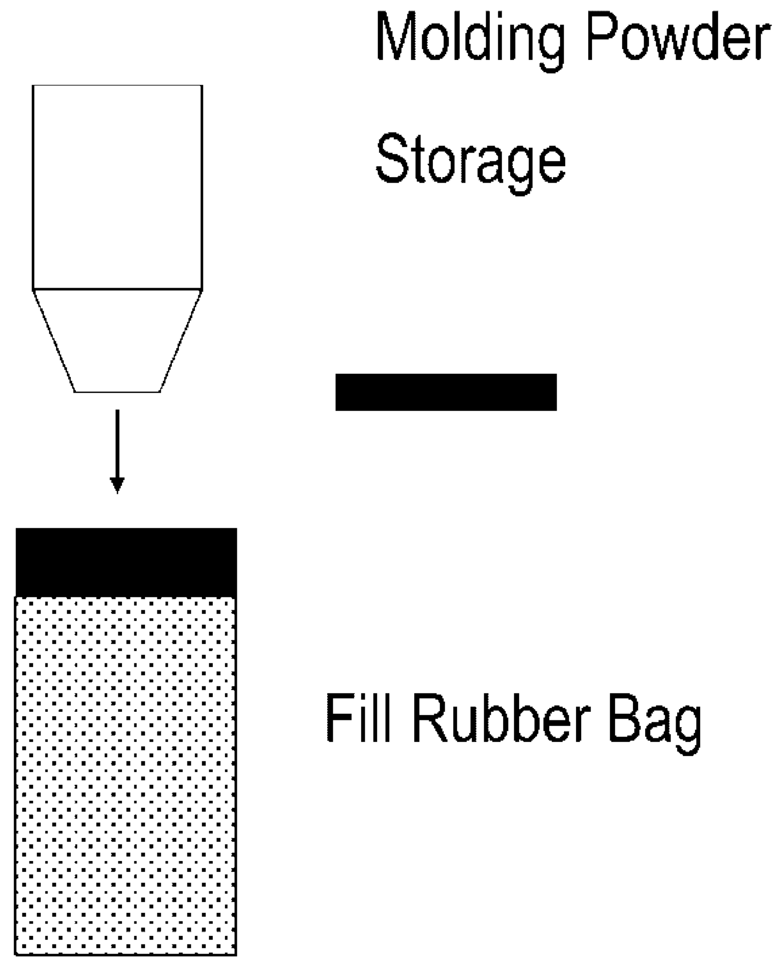


Graphite Production - Isostatic Molding

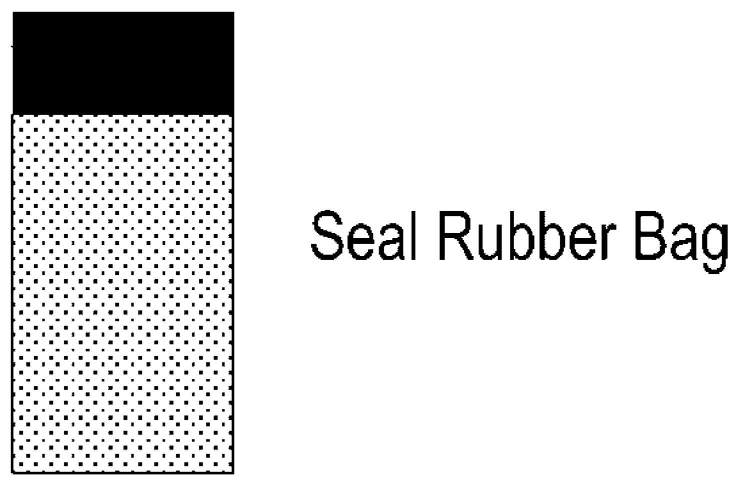


Place Bag
in Tooling

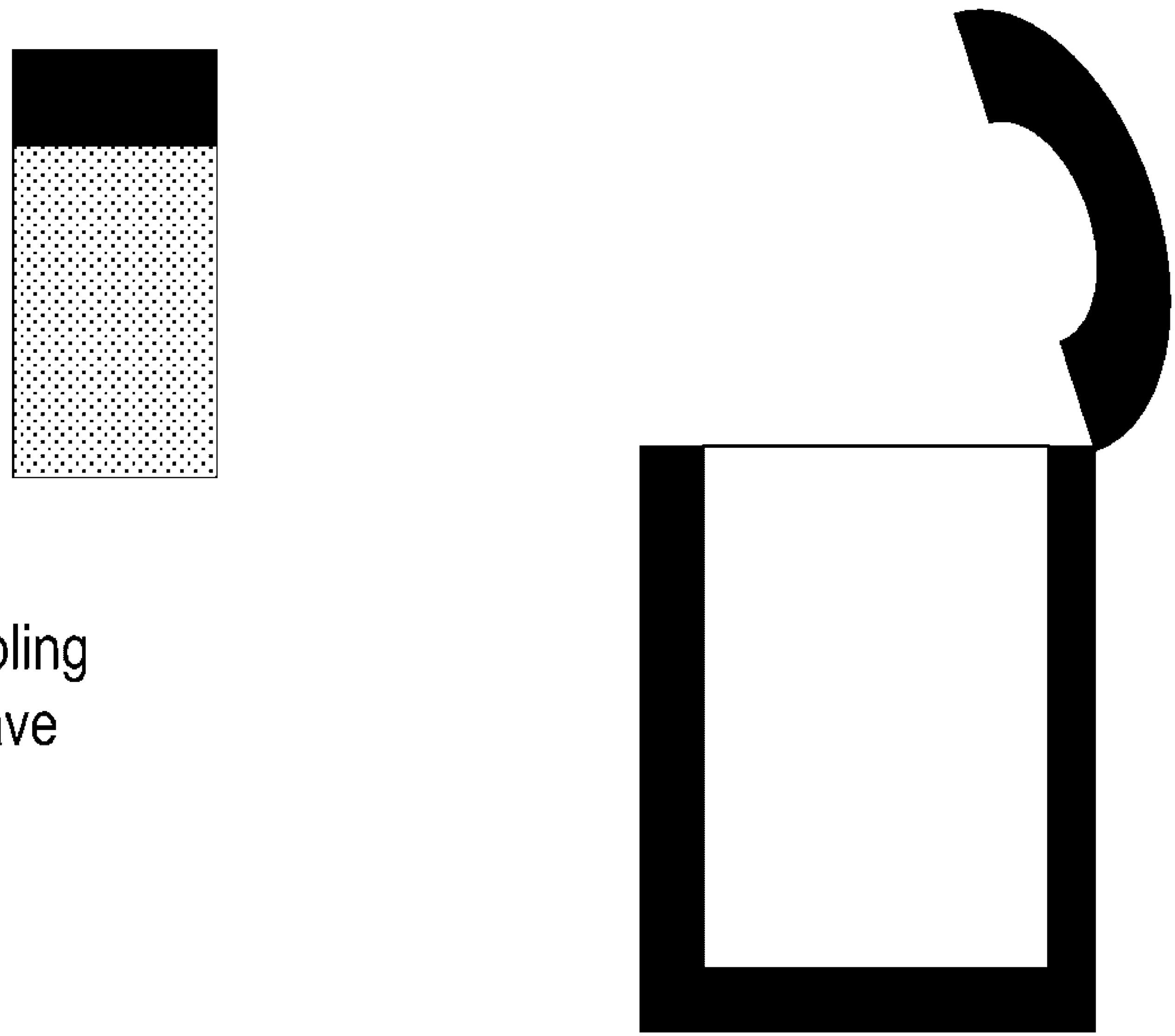
Graphite Production - Isostatic Molding



Graphite Production - Isostatic Molding



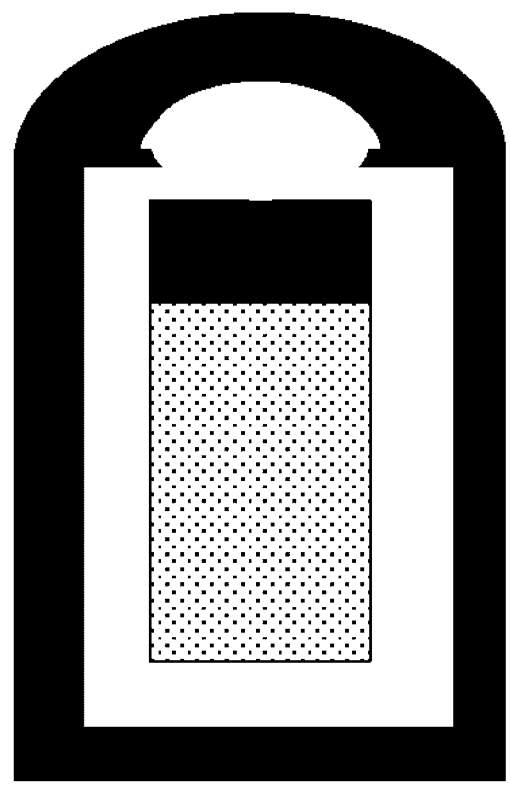
Graphite Production - Isostatic Molding



Place Tooling
in Autoclave

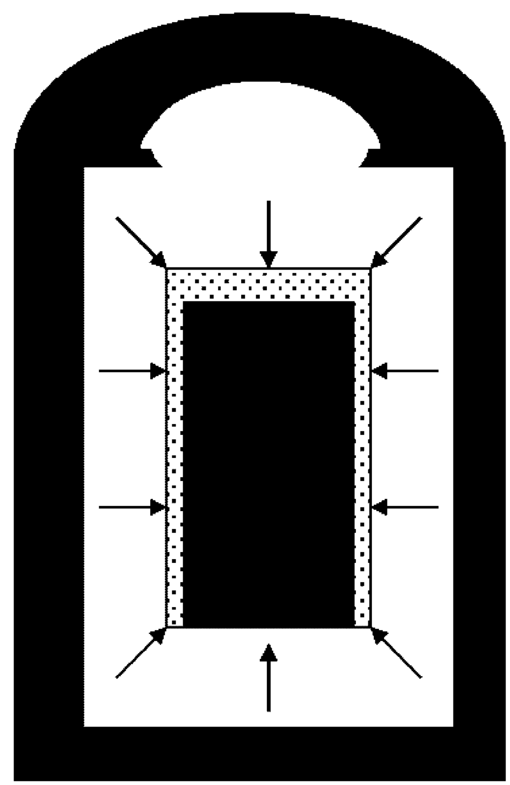
Graphite Production - Isostatic Molding

Seal
Autoclave

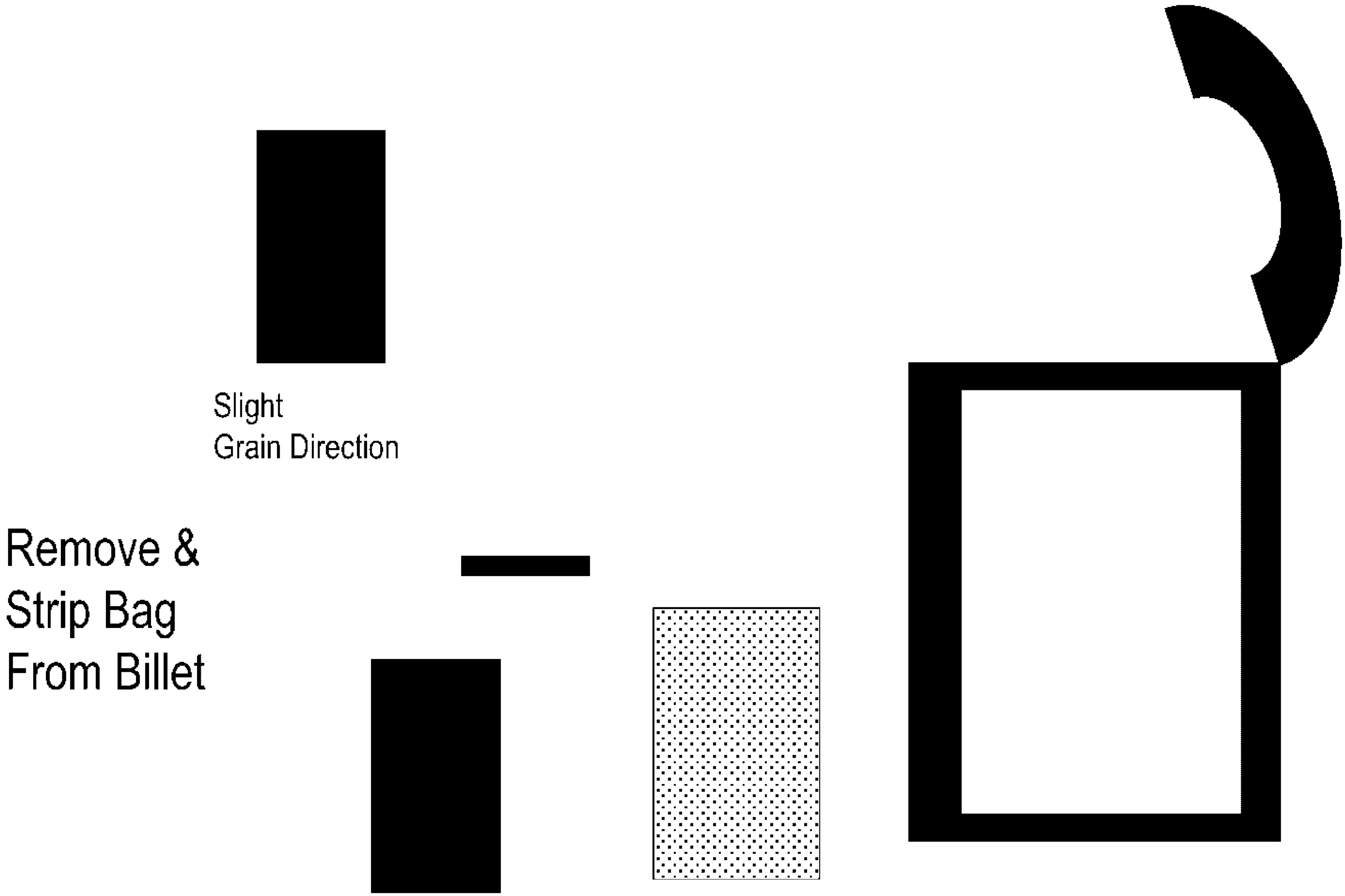


Graphite Production - Isostatic Molding

Pressurize



Graphite Production - Isostatic Molding



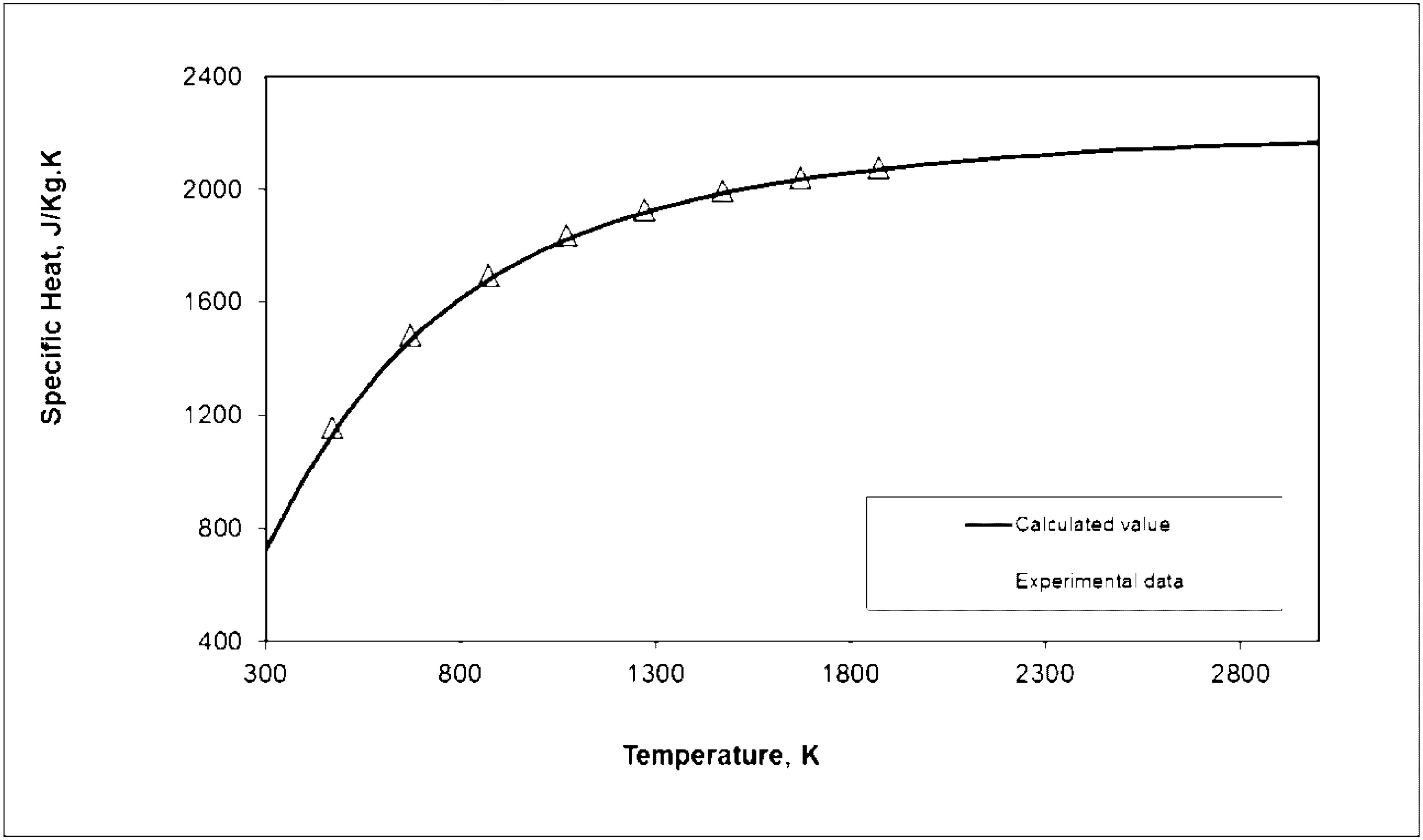
Physical Properties

Synthetic graphite properties

Typical Properties	Graphite Grade and Manufacturer					
	AXF-5Q POCO	IG-43 Toyo-Tanso	2020 Mercen	ATJ GTI	NBG-18 SGL Carbon	AGR GTI
Forming Method	Isomolded	Isomolded	Isomolded	Isomolded	vibro-molded	Extruded
Maximum Particle Size, μm	5	10 (mean)	15	25 (mean)	1600	3000
Bulk Density, g/cm^3	1.8	1.82	1.77	1.76	1.88	1.6
Thermal Conductivity, W/m.K (Measured at ambient temperature)	85	140	85	125(WG) 112 (AG)	156 (WG) 150 (AG)	152 (WG) 107 (AG)
Coefficient of Thermal Expansion, 10^{-6}/K (over given temperature range)	7.4 (20-500°C)	4.8 (350-450°C)	4.3 (20 -500°C)	3.0 (WG) 3.6 (AG) (@500°C)	4.5 (WG) 4.7 (AG) (20-200°C)	2.1 (WG) 3.2 (AG) (@500°C)
Electrical Resistivity, μΩ.m	14	9.2	15.5	10.1 (WG) 11.7 (AG)	8.9 (WG) 9.0 (AG)	8.5 (WG) 12.1 (AG)
Young's Modulus, GPa	11	10.8	9.3	9.7 (WG) 9.7 (AG)	11.2 (WG) 11.0 (AG)	6.9 (WG) 4.1 (AG)
Tensile Strength, MPa	65	37	30	27.2 (WG) 23.1 (AG)	21.5 (WG) 20.5 (AG)	4.9 (WG) 4.3 (AG)
Compressive Strength, MPa	145	90	80	66.4 (WG) 67.4 (AG)	72 (WG) 72.5 (AG)	19.8(WG) 19.3 (AG)
Flexural Strength, MPa	90	54	45	30.8 (WG) 27.9 (AG)	28 (WG) 26(AG)	8.9 (WG) 6.9 (AG)

WG-with grain, AG-against grain

Temperature Dependence of Specific Heat



Calculated from ASTM C781

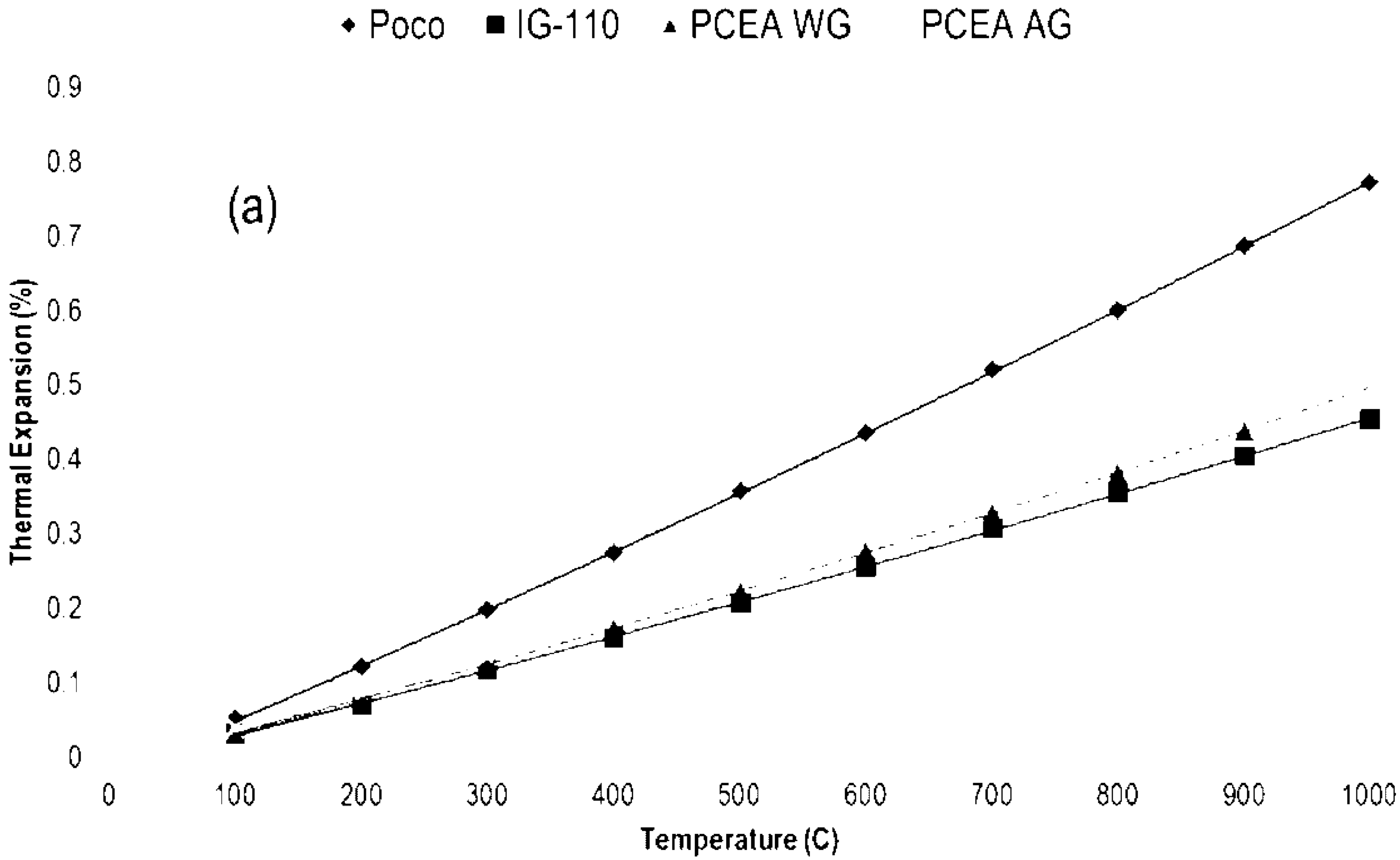
Single crystal thermal expansion behavior

Hexagonal graphite lattice has two principal thermal expansion coefficients; α_c, the thermal expansion coefficient parallel to the hexagonal <c>-axis and α_a, the thermal expansion coefficient of the crystal parallel to the basal plane (<a>-axis). The thermal expansion coefficient in any direction at an angle φ to the <c> axis of the crystal given by:

$$\alpha(\varphi) = \alpha_c cos^2\varphi + \alpha_a sin^2\varphi$$

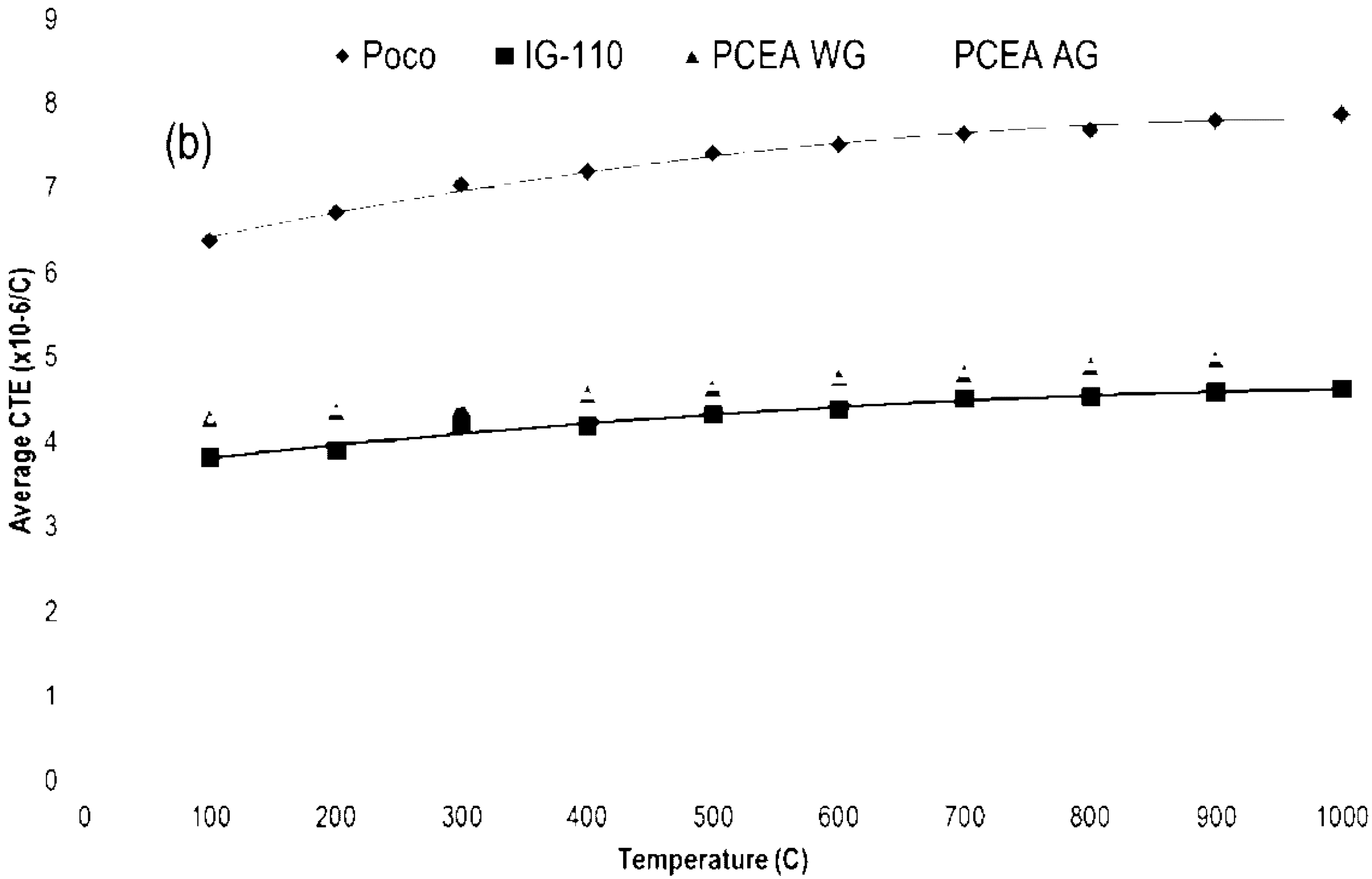
- α_c varies linearly with temperature from ~25 × $10^{-6}K^{-1}$ at 300K to ~35 × 10^{-6} K^{-1} at 2500K.
- α_a is much smaller and increases rapidly from -1.5 × 10^{-6} K^{-1} at ~300K to approximately 1 × 10^{-6} K^{-1} at 1000K, and remains relatively constant at temperatures up to 2500K.

Temperature Dependence of Thermal Expansion



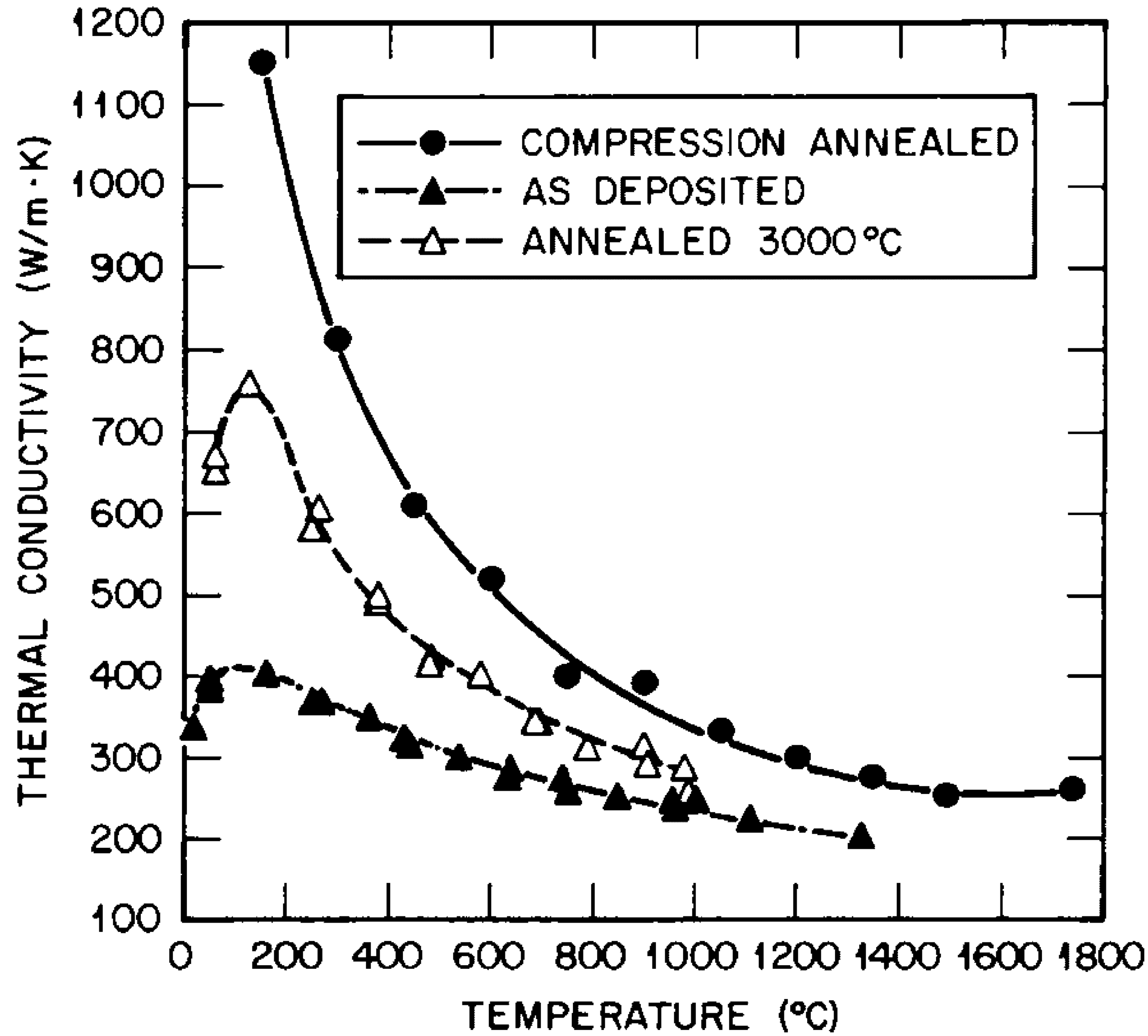
Thermal closure of aligned porosity

Temperature Dependence of Coefficient of Thermal Expansion



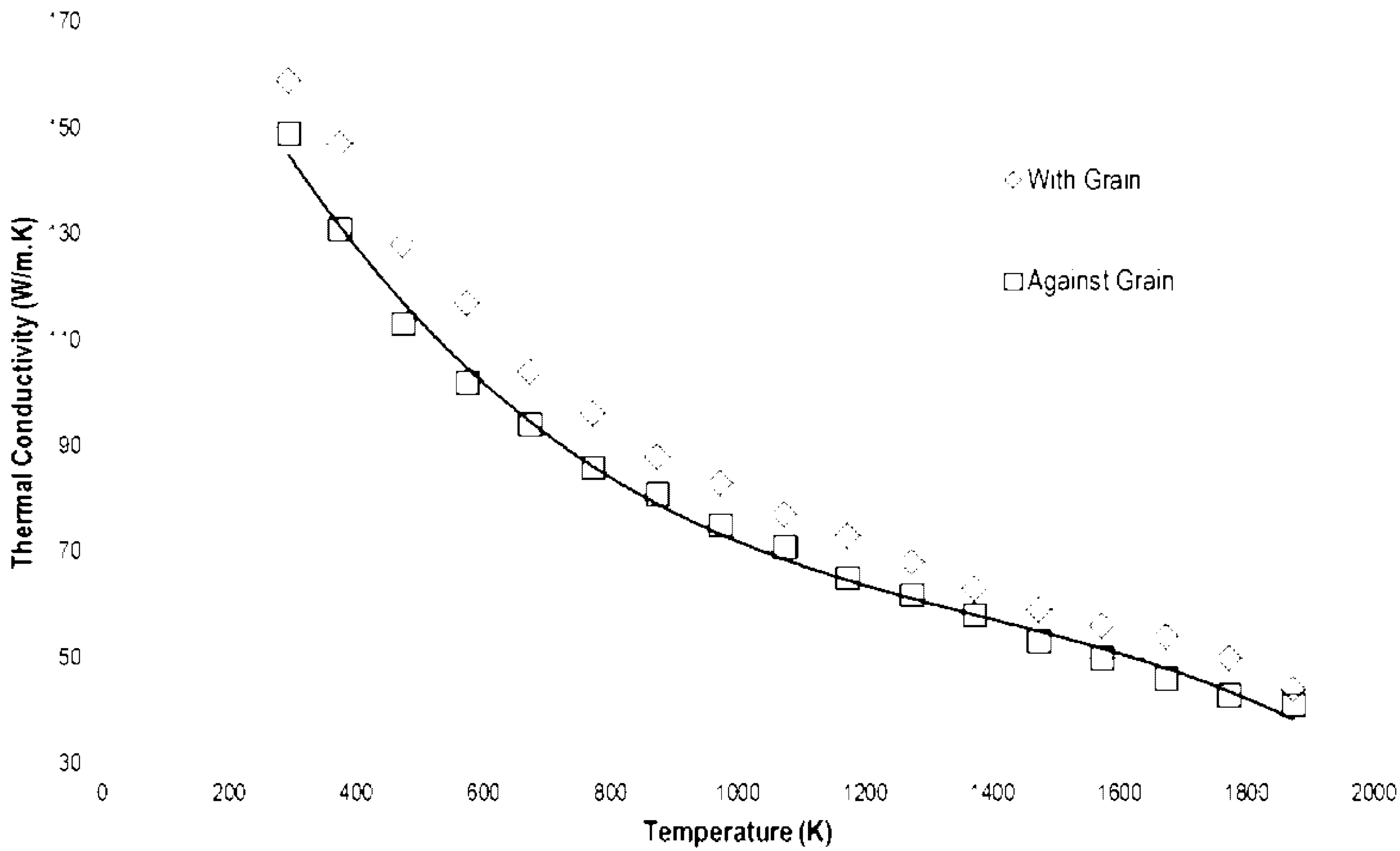
Thermal closure of aligned porosity

Temperature Dependence of the Thermal Conductivity



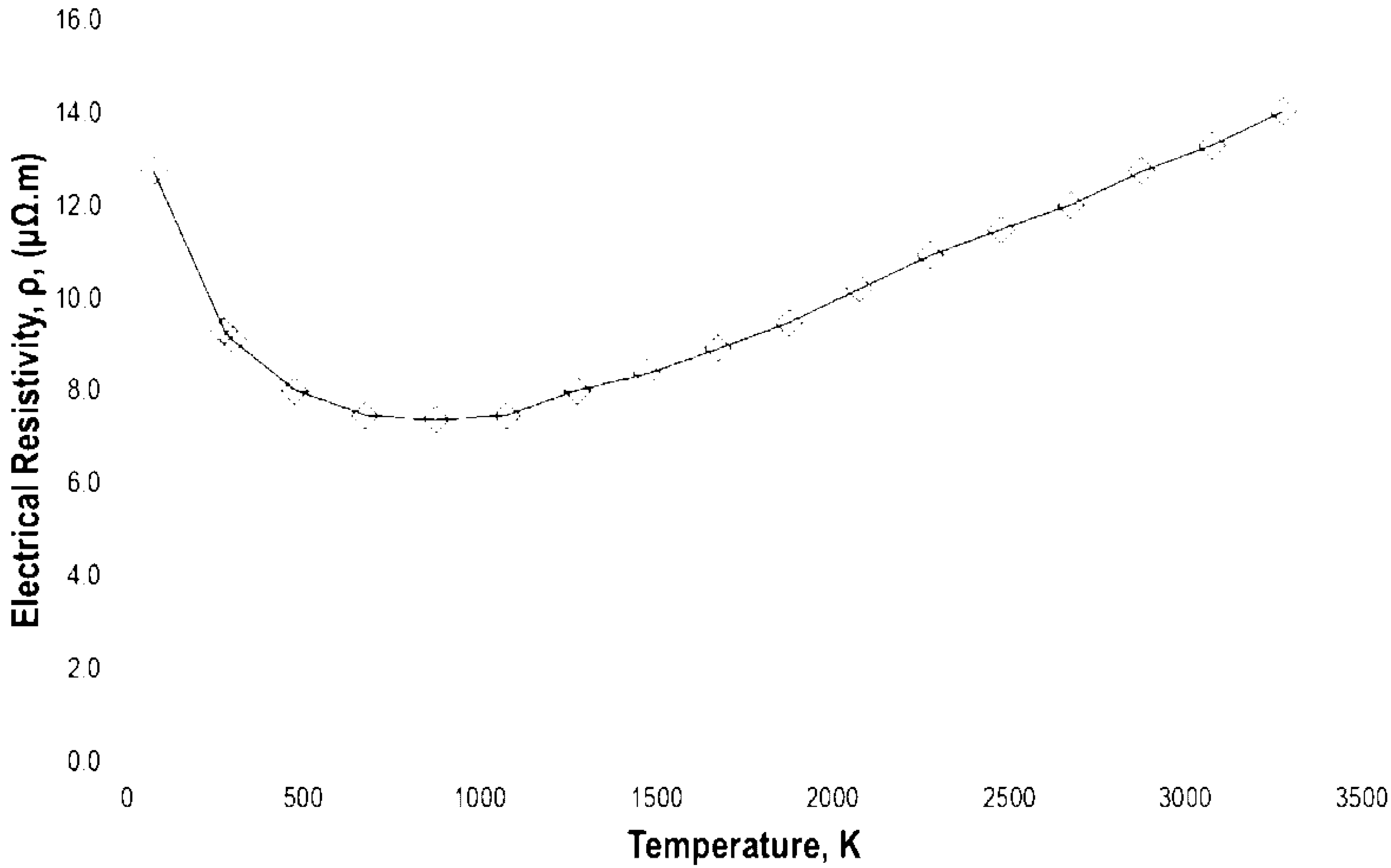
Phonon scattering, effect of intrinsic defects

Temperature Dependence of the Thermal Conductivity



Anisotropy in extruded thermal conductivity

Temperature Dependence of the Electrical Resistivity



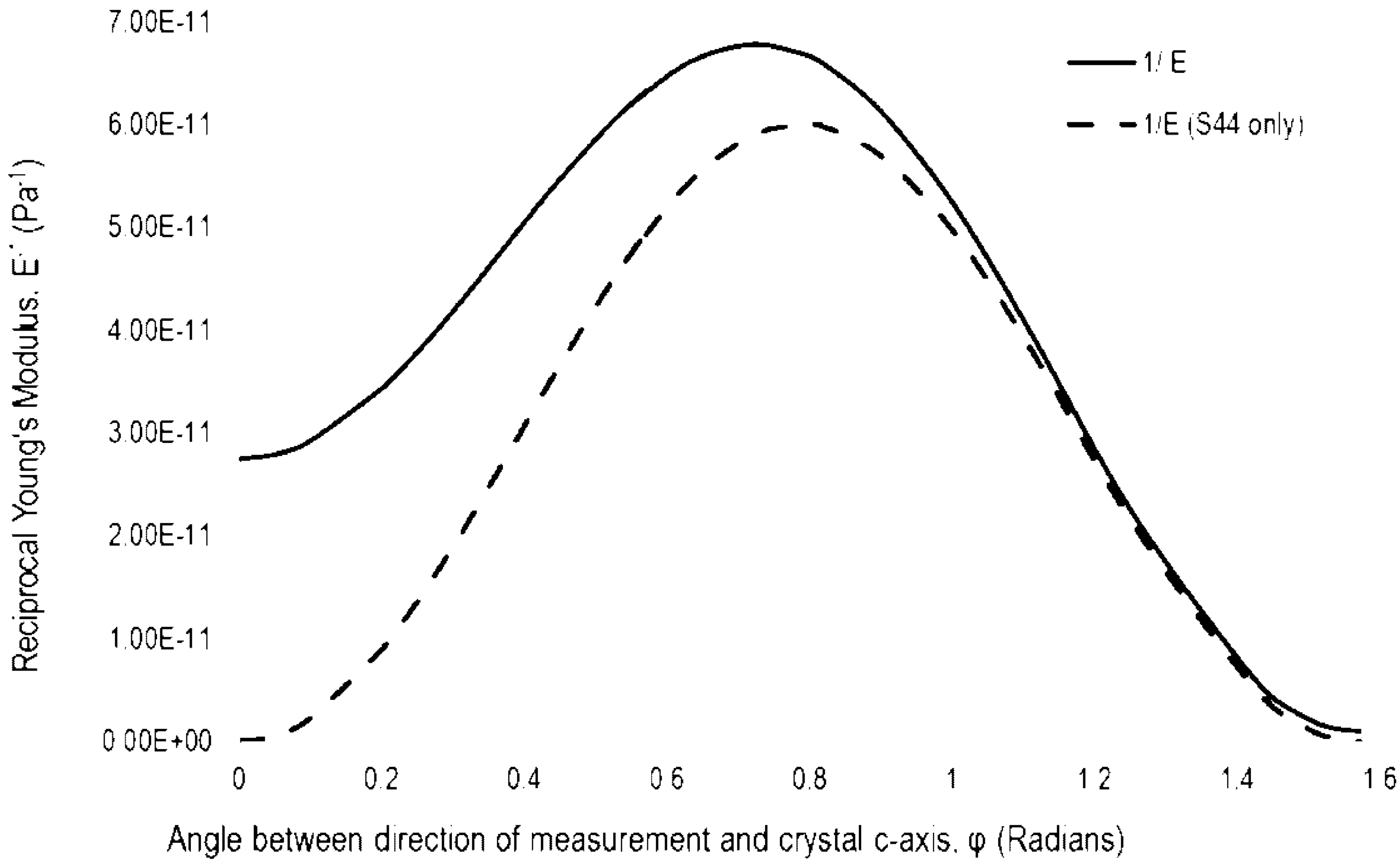
Electron transport mechanism

Elastic Behavior

Elastic constants of single crystal graphite (Kelly)

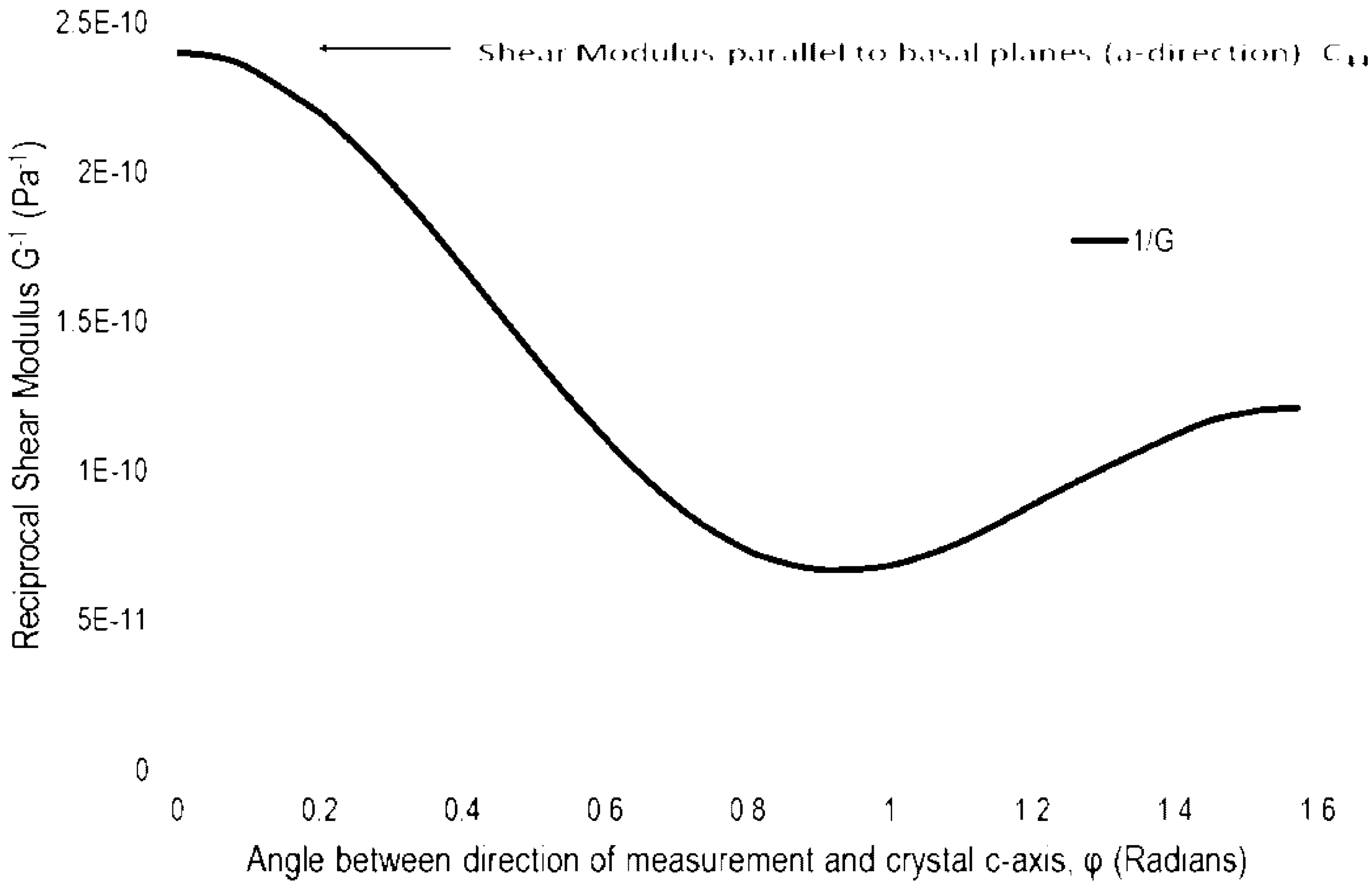
Elastic moduli, GPa		Elastic compliances, 10^{-13} Pa^{-1}	
C_{11}	1060 ± 20	S_{11}	9.8 ± 0.3
C_{12}	180 ± 20	S_{12}	-1.6 ± 0.6
C_{13}	15 ± 5	S_{13}	-3.3 ± 0.8
C_{33}	36.5 ± 1	S_{33}	275 ± 10
C_{44}	4.0 - 4.5	S_{44}	2222 - 2500

Variation of the reciprocal Young's modulus with angle of miss-orientation between the c-axis and measurement axis



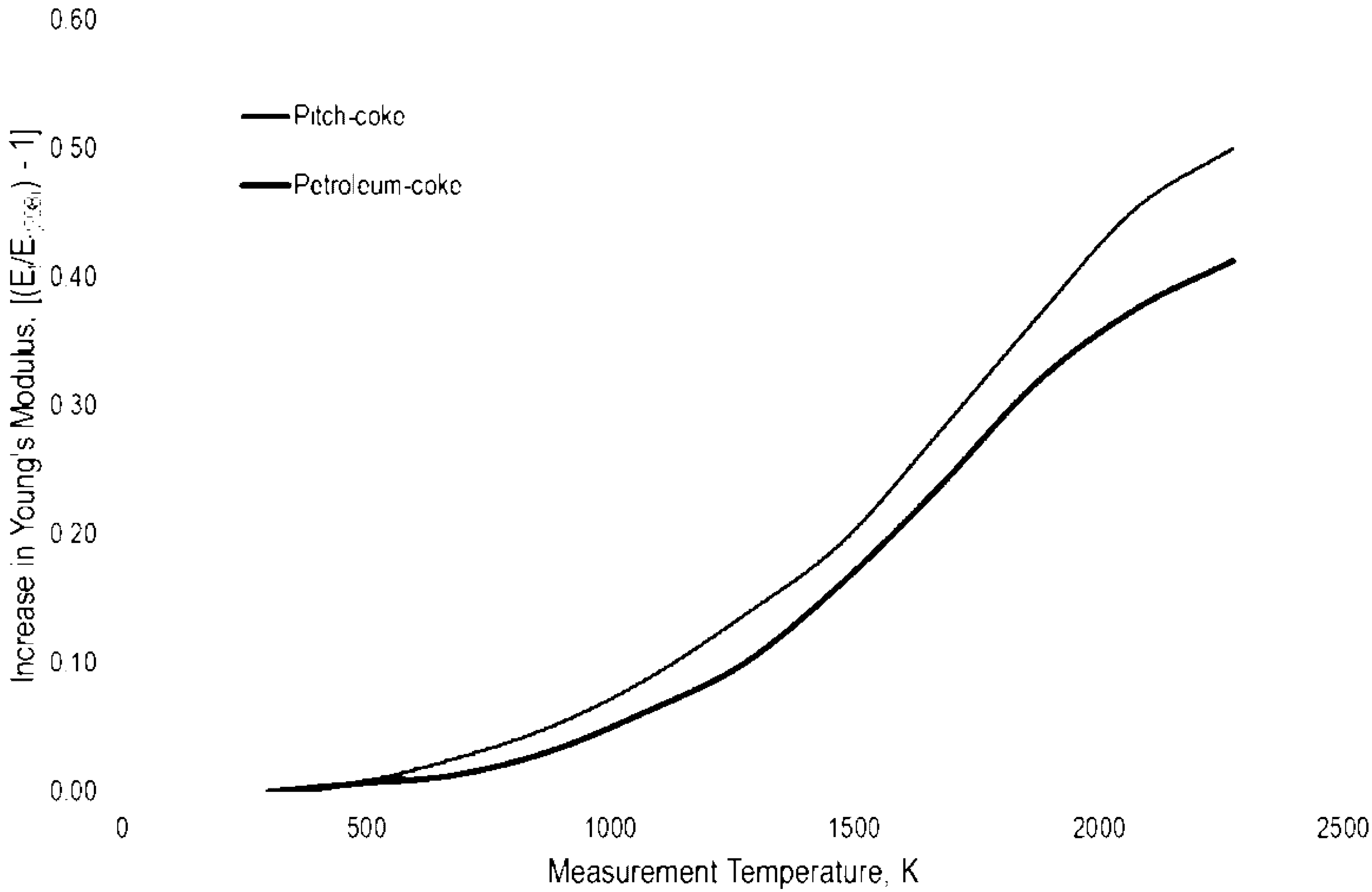
$$E^{-1} = S_{11}(1-\gamma^2)^2 + S_{33}\gamma^4 + (2S_{13} + S_{44})\gamma^2(1-\gamma^2)$$

Variation of the reciprocal Shear modulus with angle of miss-orientation between the c-axis and measurement axis



$$G^{-1} = S_{44} + \left(S_{11} - S_{12} - \frac{S_{44}}{2}\right)(1-\gamma^2) + 2(S_{11} + S_{33} - 2S_{13} - S_{44})\gamma^2(1-\gamma^2)$$

Typical Young's Modulus increase with temperature for pitch-coke and petroleum coke synthetic graphite



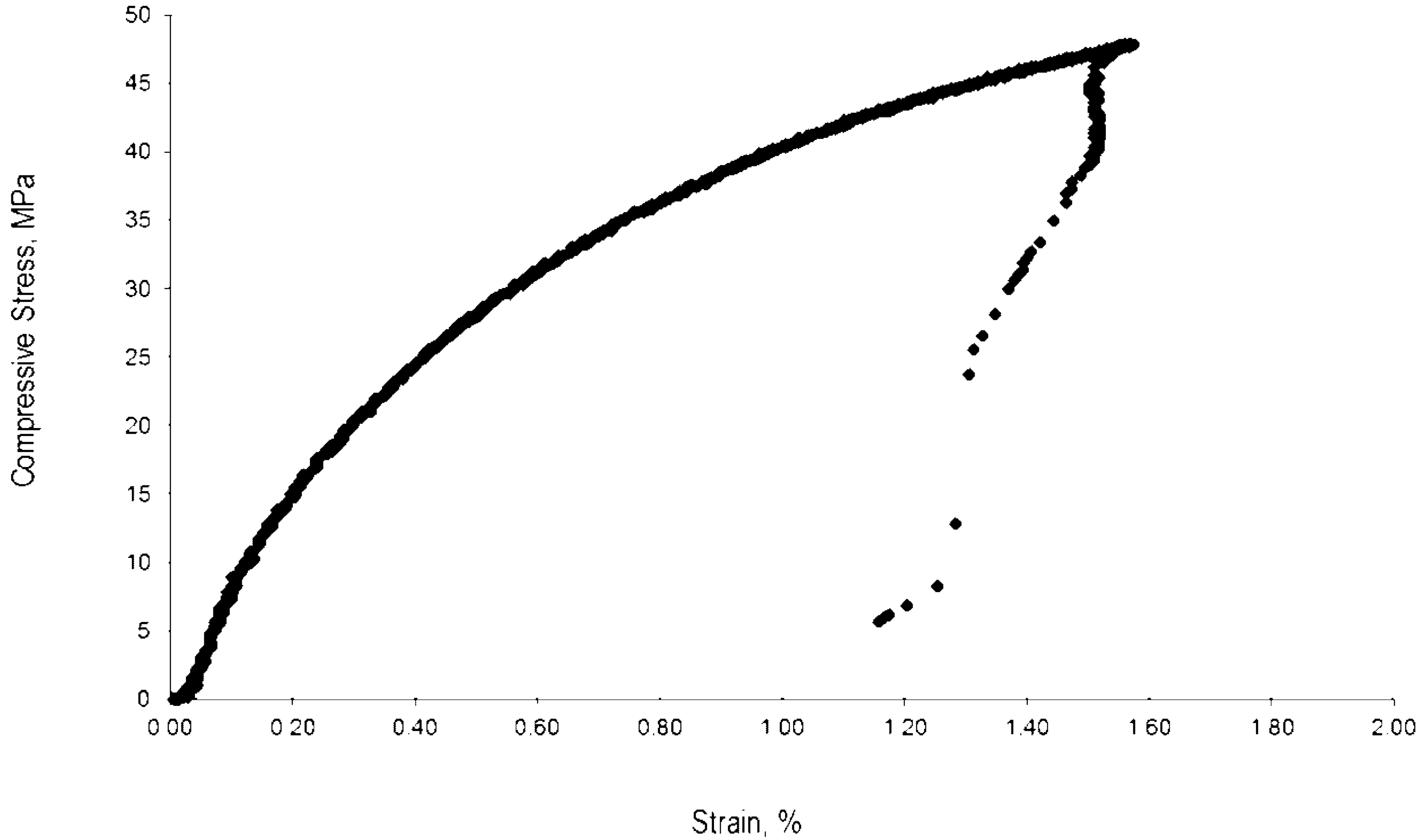
Stress-strain behavior of synthetic graphite

There are two major factors that control the stress-strain behavior of synthetic graphite:

- The magnitude of the constant C_{44}, which dictates how the crystals respond to an applied stress,
- The defect/crack morphology and distribution, which controls the distribution of stresses within the body and thus the stress that each crystallite experiences.

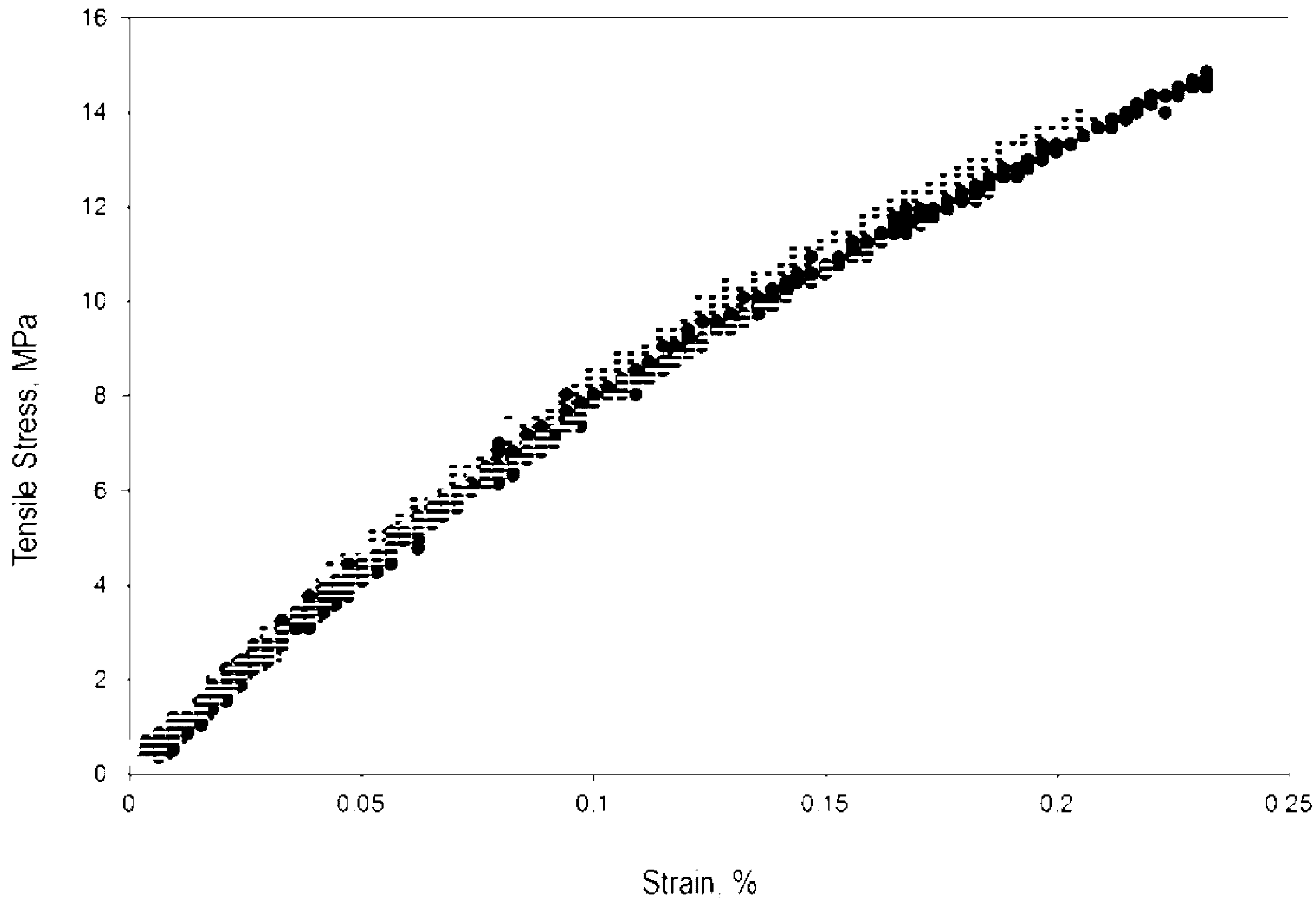
Strength
and
Fracture

Typical compressive stress-strain curve for medium-grain extruded graphite (WG)

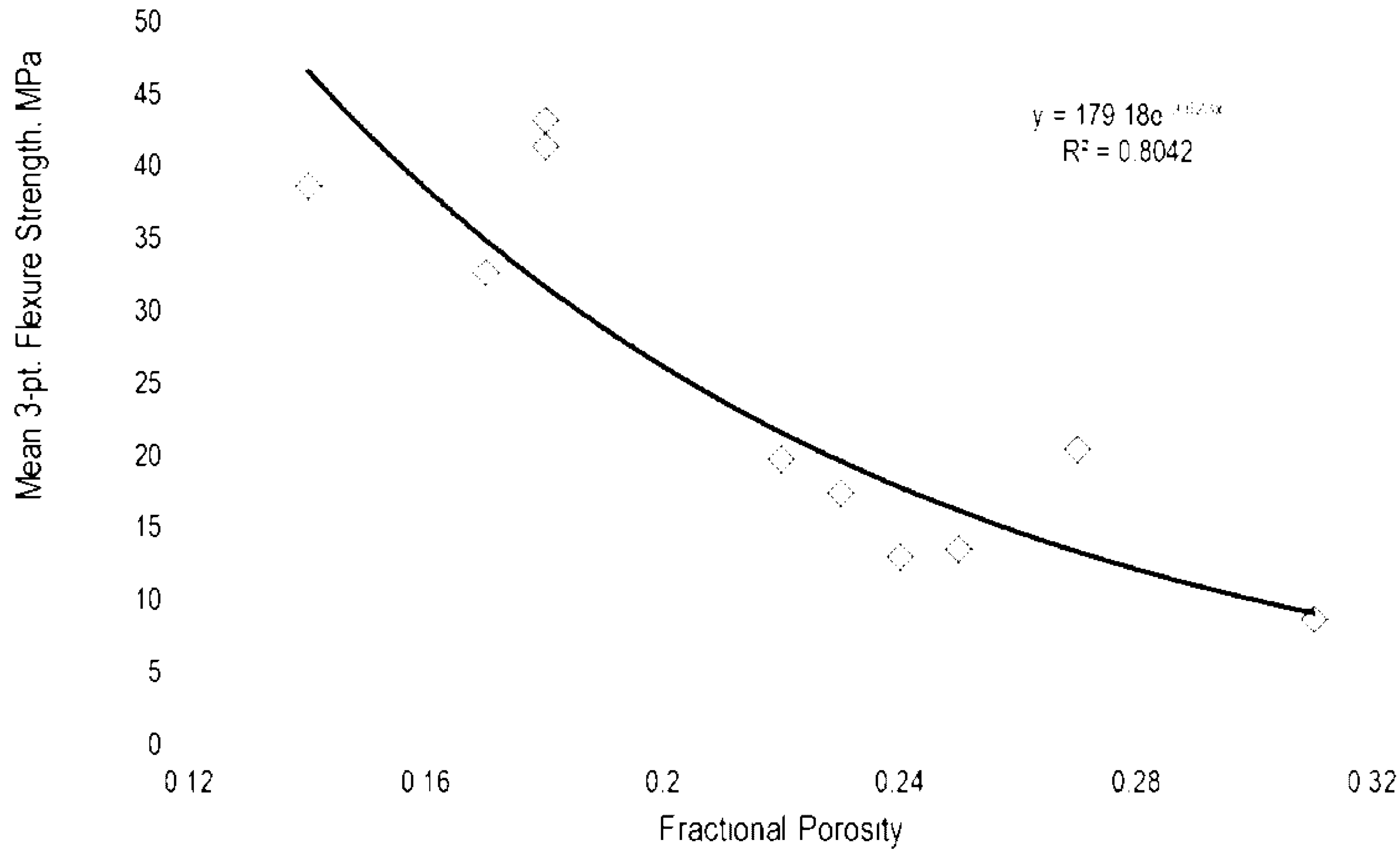


Non-linear stress strain curve

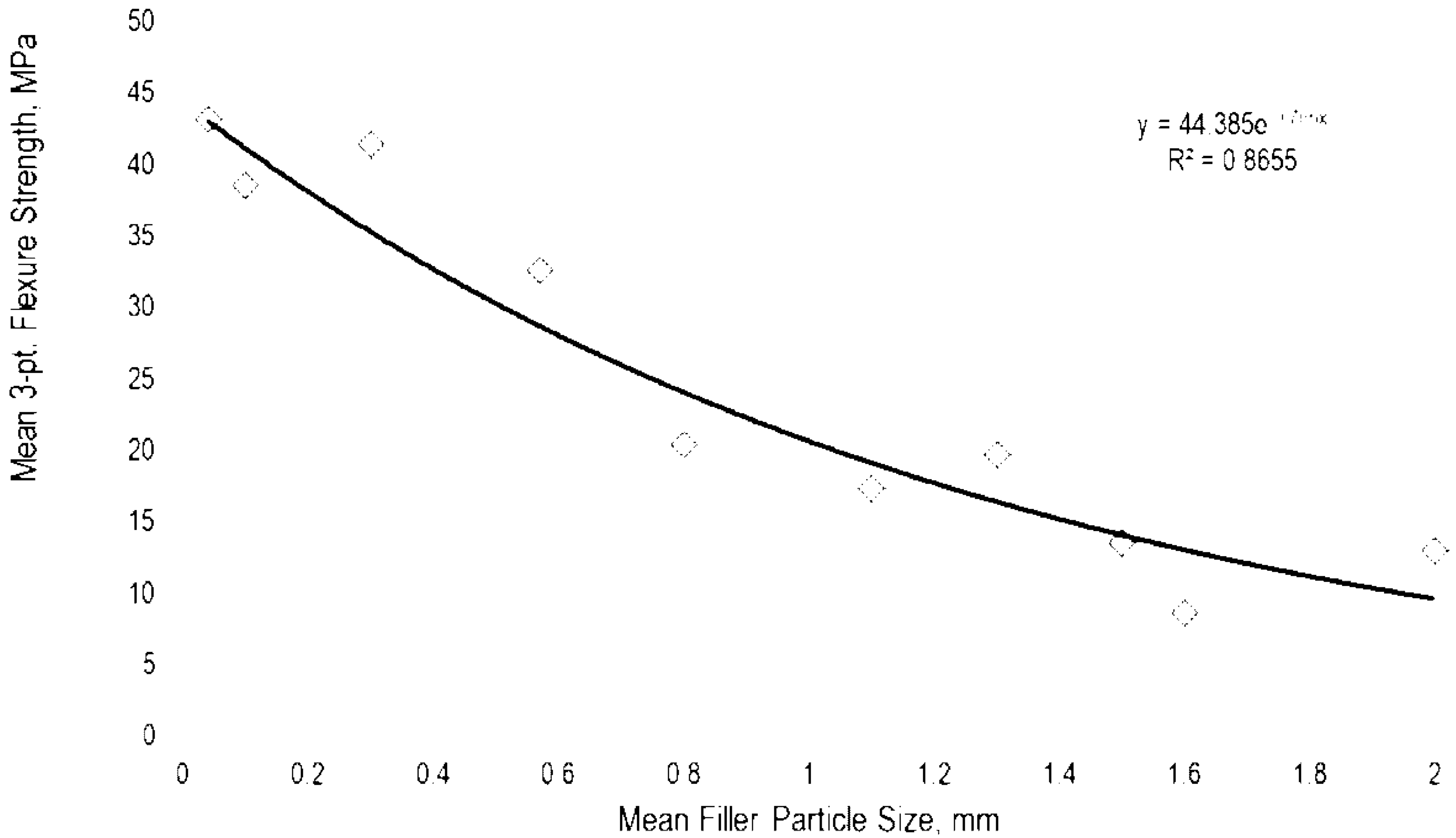
Typical tensile stress-strain curves for medium-grain extruded graphite (WG)



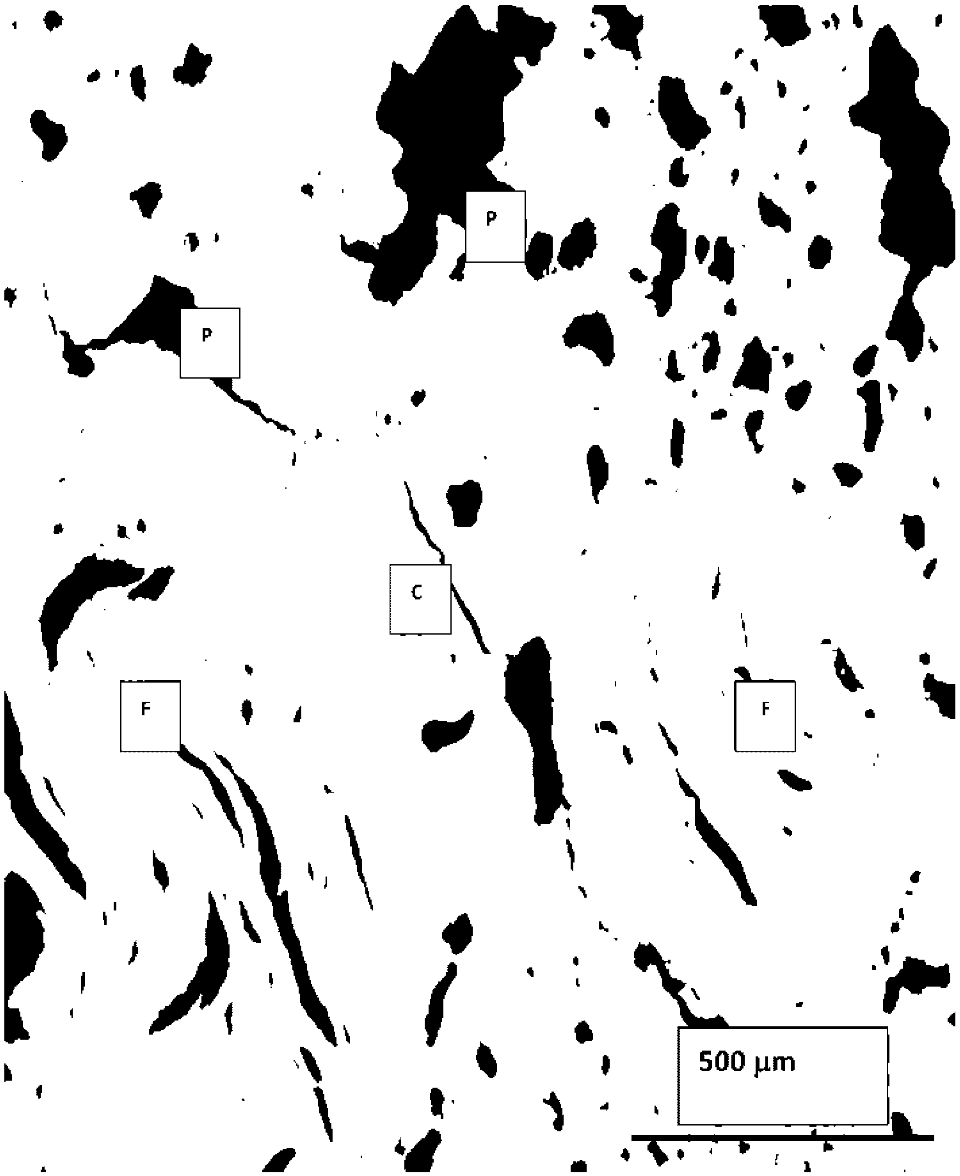
The correlation between mean 3-pt flexure strength and fractional porosity for a wide range of synthetic graphite representing the variation of textures



The correlation between mean 3-pt flexure strength and mean filler coke particle size for a wide range of synthetic graphite representing the variation of textures



Crack Propagation in Synthetic Graphite



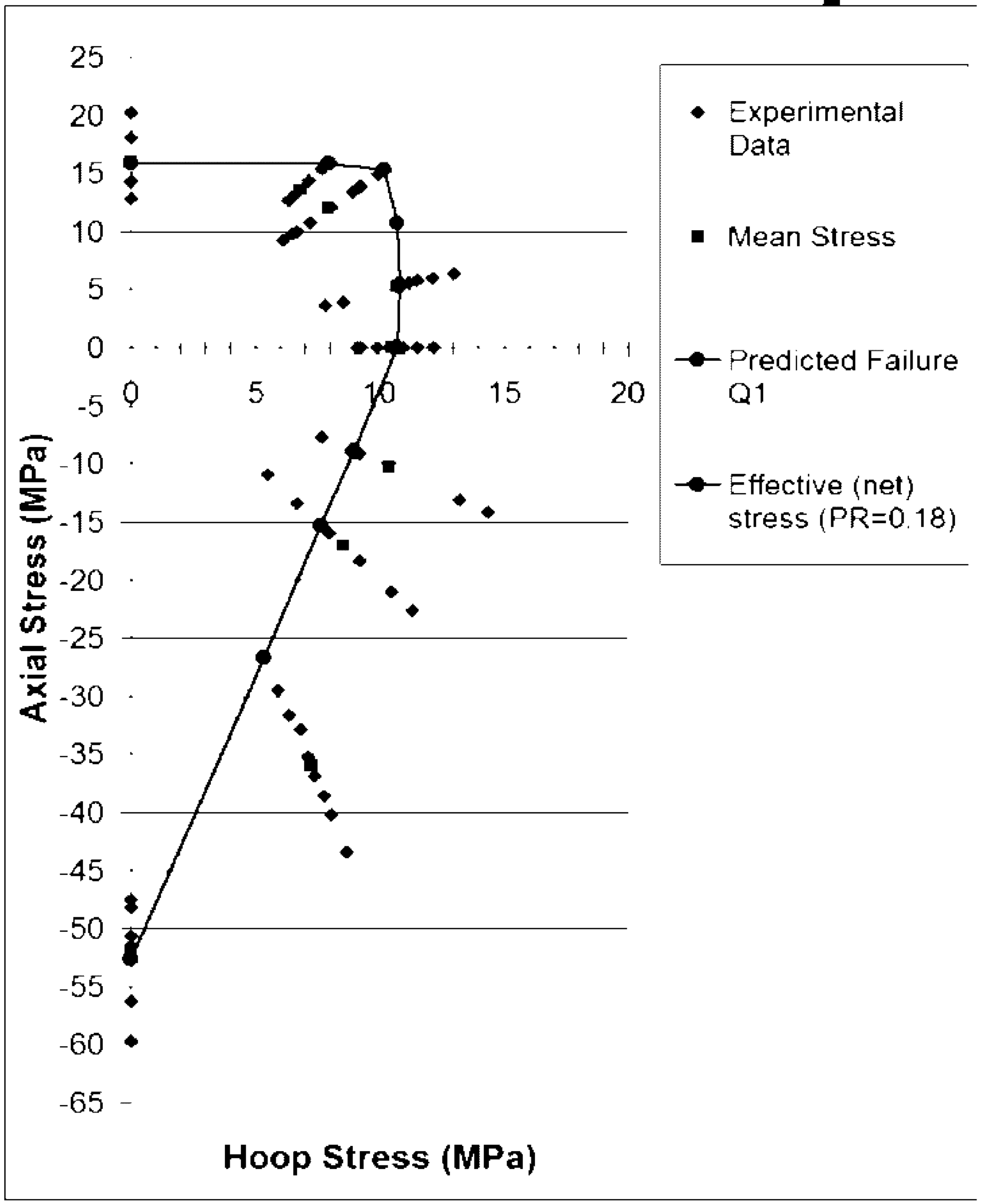
An optical photomicrograph of the microstructure of grade H-451 graphite revealing the presence of pores [P], coke filler particles [F] and cracks [C] which have propagated through the pores presumably under the influence of their stress fields

Burchell fracture model probability predictions for different graphite

Model Inputs:

- Mean filler particle size
- Particle K_{Ic}
- Specimen breadth
- Stressed volume
- Pore size distribution mean and st. dev.
- Density
- Number of pores per unit volume

Graphite multiaxial strength behavior & model predictions



- Previous 1st and 2nd stress quadrant testing of H-451 & IG -110
- Extended to NBG-18 1st & 2nd stress quadrants
- NBG-18 modeled with Burchell model incorporating Shetty mixed mode fracture criterion
- NBG-18 3rd and 4th stress quadrant testing initiated at ORNL

Graphite thermal shock resistance

Thermal Shock FOM, $$\Delta_{th} = \frac{K\sigma}{\alpha E(1-\upsilon)}$$

K is the thermal conductivity, σ_y the yield strength, α the thermal expansion coefficient, E the Young's modulus, and υ is Poisson's ratio

Material	FOM
Graphite, AXF-5Q	124,904
Graphite, IG-110	84,844
Wrought beryllium	~1×10^4
Pure tungsten	~0.5×10^5
Carbon-carbon composite	~1×10^6

Graphite does not melt but rather sublimes at T>3300K

Applications

- Metal processing
  -
  -
  -
- Semiconductor manufacture
  -
- Electrical and electronic
  -
  -
- Mechanical
  -
- Aerospace
  -
- Nuclear
  -

Graphite flaw detection & NDE

- Ultrasonic inspection used in graphite manufacture for inspection after bake stage and after graphitization for dry core and gross flaw detection
- But, ultrasonic has limited usefulness/applicability
- Flaw resolution cannot exceed wavelength of signal, i.e., to resolve critical flaws need to use high frequency
- Attenuation increases (range decreases) as frequency increases
- Thus for engineering components may not have sufficient range or resolution, depending on structure/texture and critical defect size
- X-ray tomography promising technique for small artifacts
- Resolution down to micron level or better

Summary

- Synthetic graphite is a unique high temperature material
- Crystals have strong in-plane covalent bonds, weak van der Waal bond between planes
- Complicated processing route with many variables
- Properties are controlled by bond anisotropy, structure and texture
- Domain size (extended order) in filler coke and binder directly affects isotropy
- Manufacturing process imparts texture which influences isotropy
- Porosity controls fracture behavior & strength
- Phonon conductor of heat, electron conduction mechanism

Nuclear Graphite

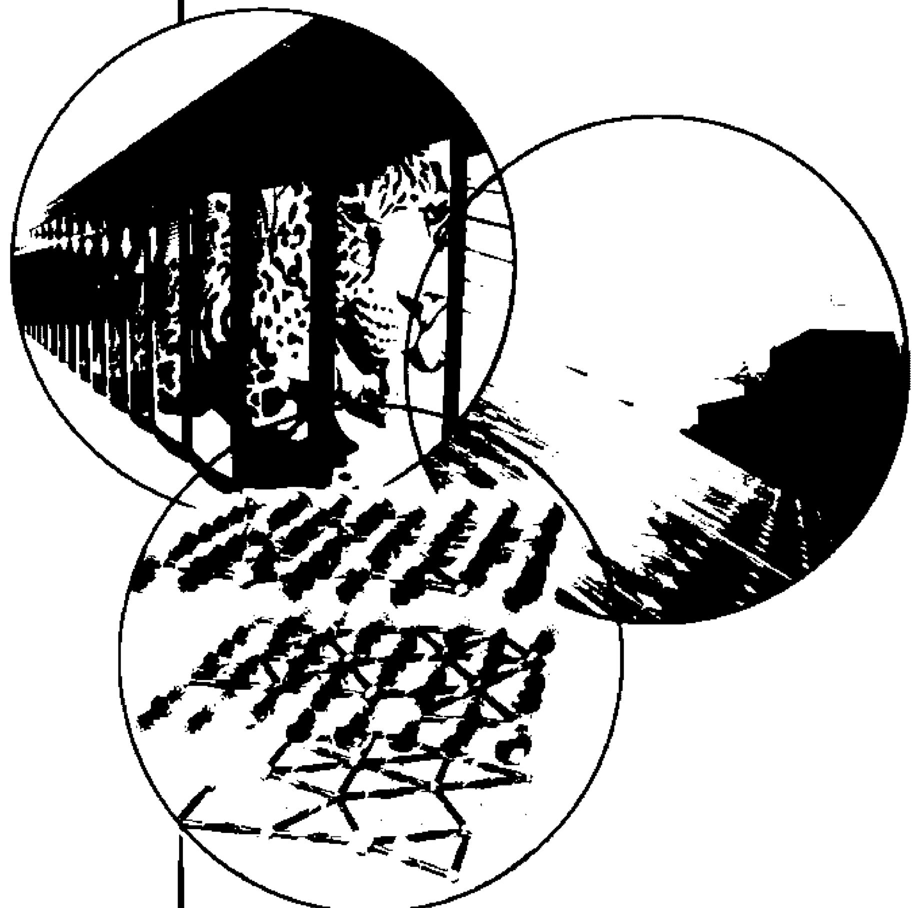
Tim Burchell

Materials Science & Technology Division

Presented to

US Nuclear Regulatory Commission

January 12th 2011



U.S. DEPARTMENT OF ENERGY

OAK RIDGE NATIONAL LABORATORY

MANAGED BY UT-BATTELLE FOR THE DEPARTMENT OF ENERGY

Outline

Graphite

Role in HTGRs

Role of Graphite in a Nuclear Reactor

Neutron moderator (carbon & graphite)

Neutron reflector – returns neutrons to the active core

Graphite (nuclear grade) has a low neutron capture cross section

High temperature material

Role of Graphite in a Nuclear Reactor

Graphite is the reactor core structural material

HTGR cores are constructed from graphite blocks and do not form a pressure boundary

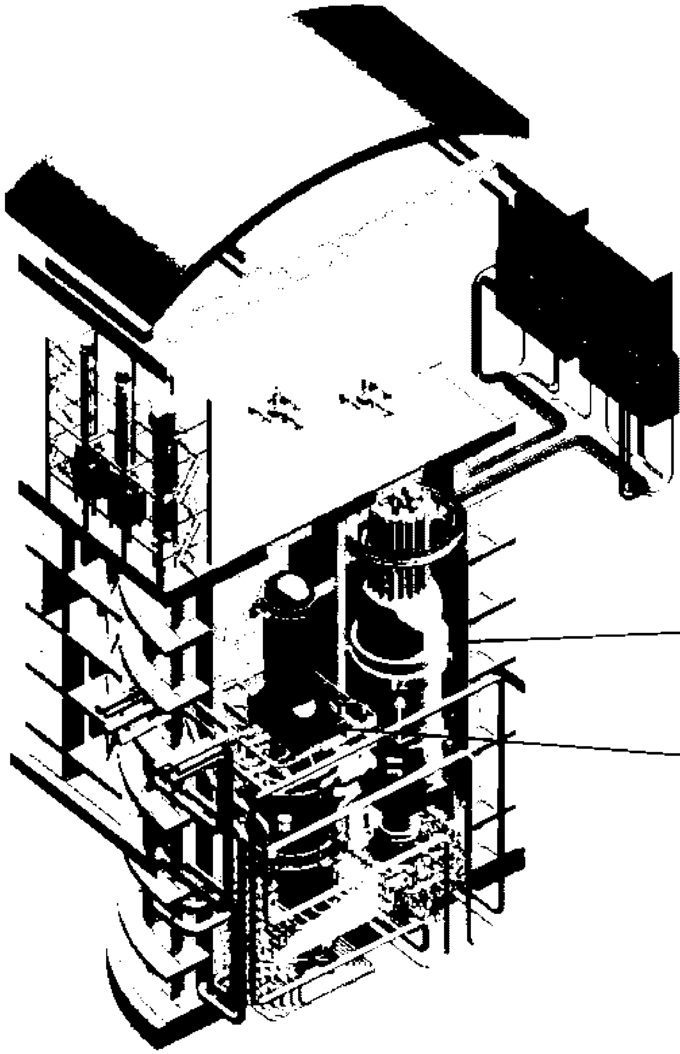
In prismatic cores the graphite fuel elements retain the nuclear fuel

In a pebble bed the graphite structure retains the fuel pebbles

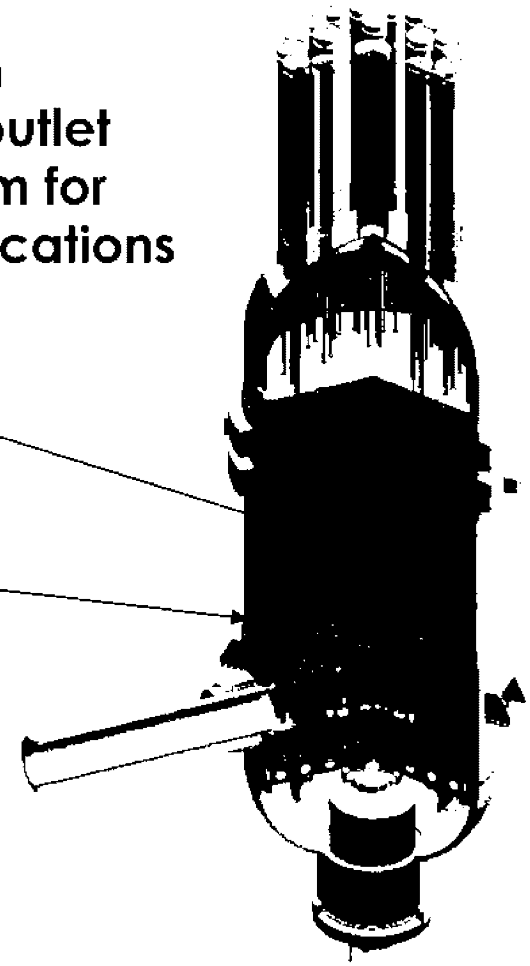
The graphite reflector structure contains vertical penetrations for reactivity control

Reactivity control in also graphite fuel elements

Gas Turbine-Modular Helium Reactor (GT -MHR)

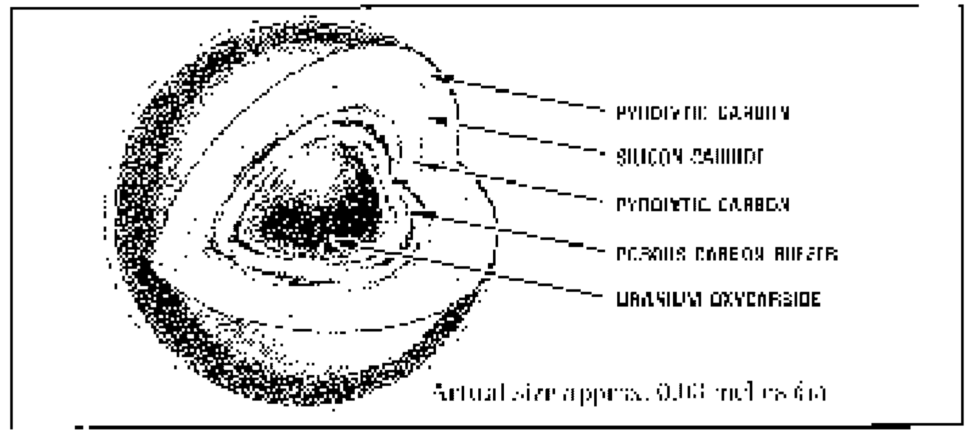


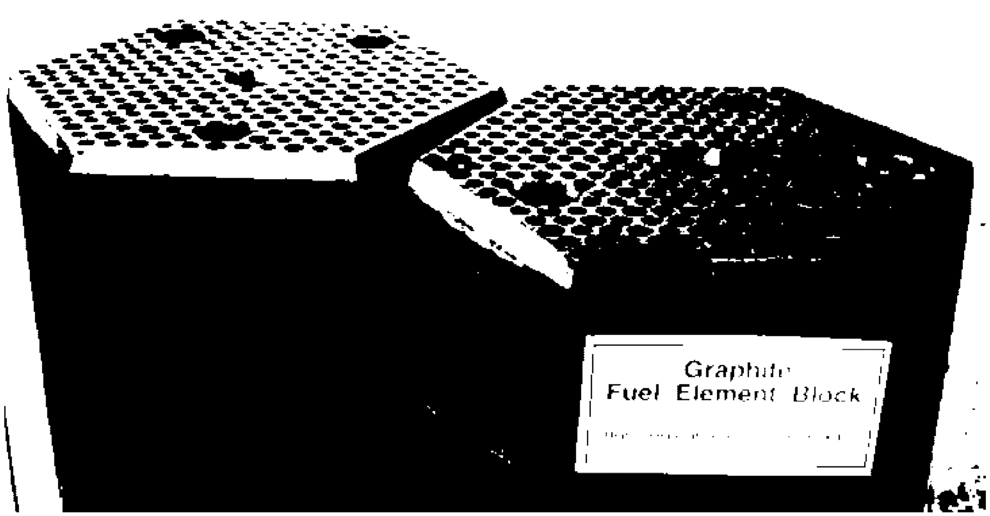
> 50% conversion efficiency (Brayton Cycle) or 1000°C outlet temperature Helium for process heat applications (H_2 production)

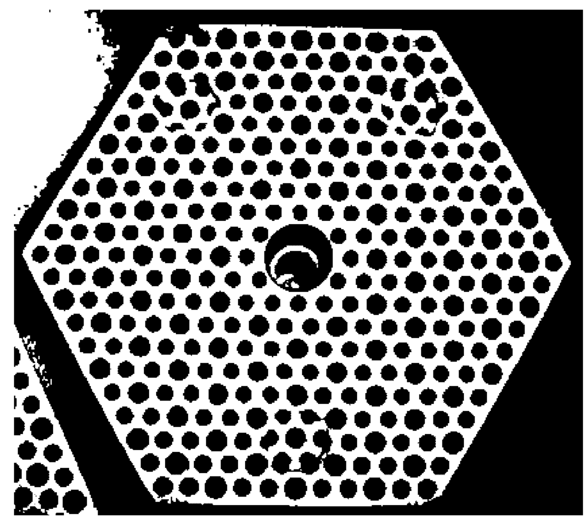


The GT-MHR Utilizes Ceramic Coated Particle Fuel

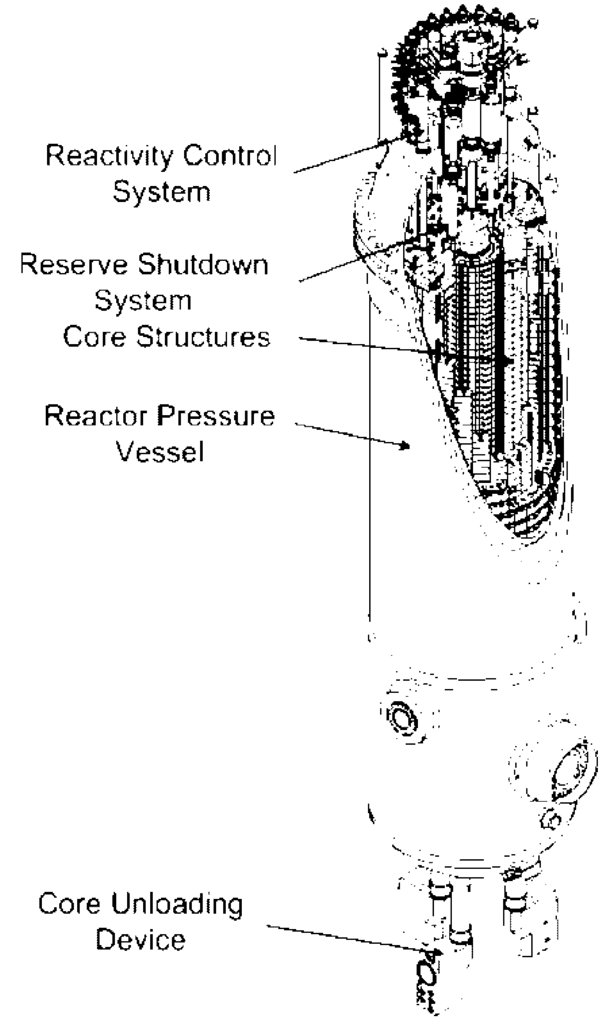
The TRISO fuel particles are formed into 12 mm diameter graphite (carbon) fuel sticks and inserted into graphite fuel blocks





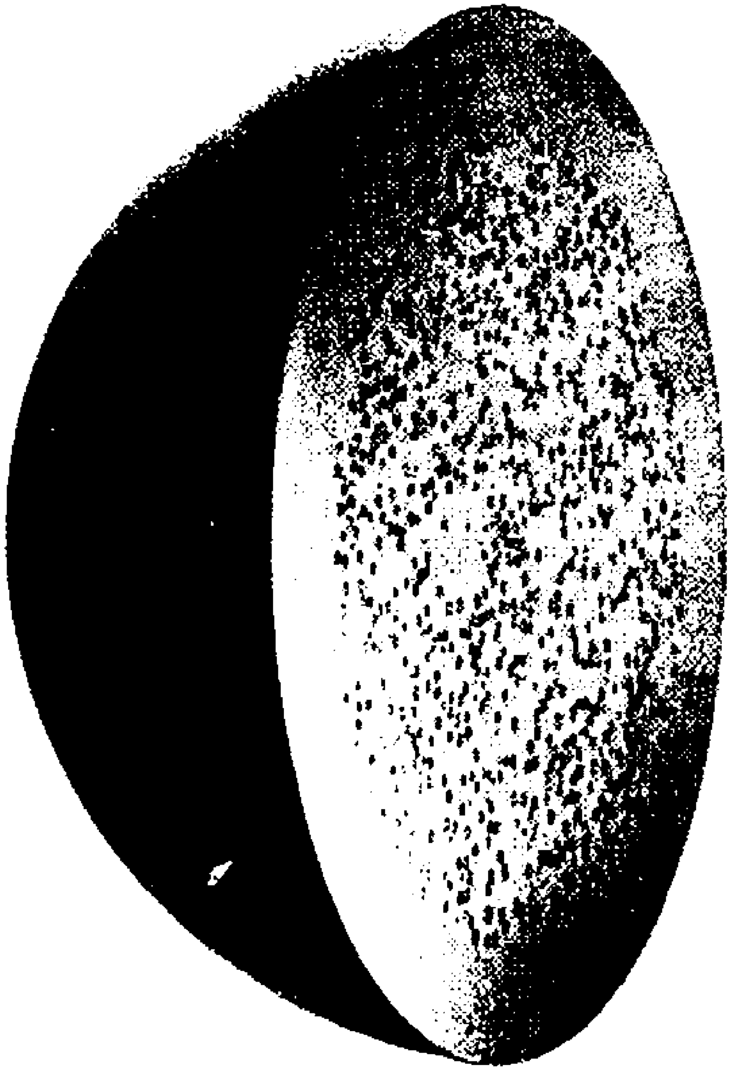


Graphite Core Components – Pebble Type HTR (PBMR)

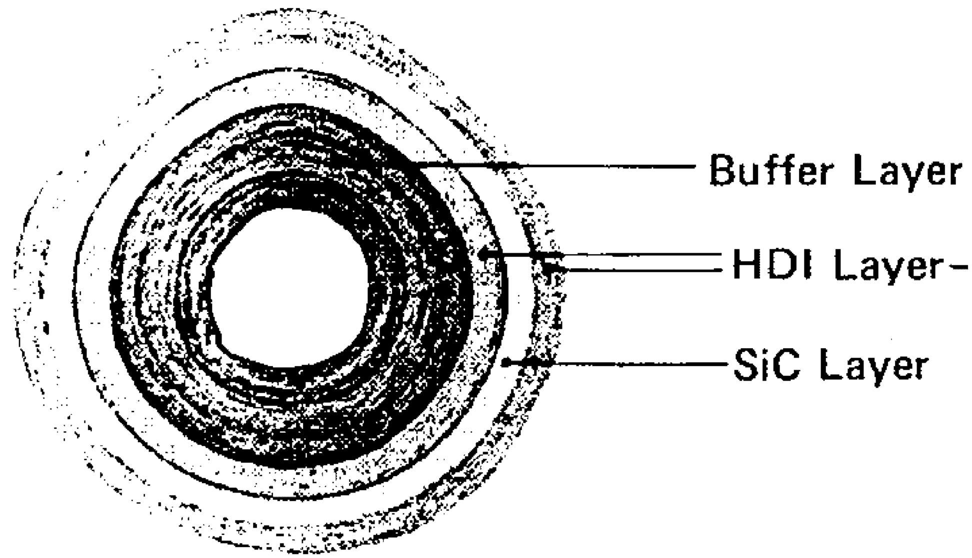


- NBG-18 Graphite blocks form the PBMR outer reflector
- Reflector penetrations are for the control rods and reserve shutdown system

The Pebble Type HTR Utilizes Ceramic Coated Particle Fuel

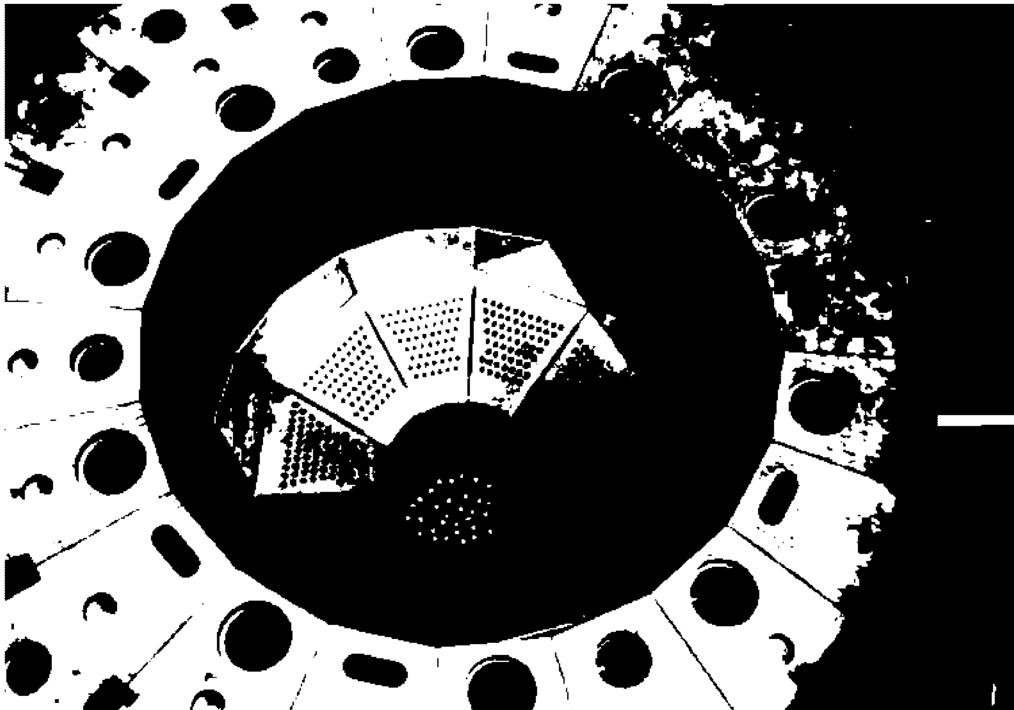


The TRISO fuel particles are combined into a graphite (carbon) fuel ball (pebble) 6 cm in diameter

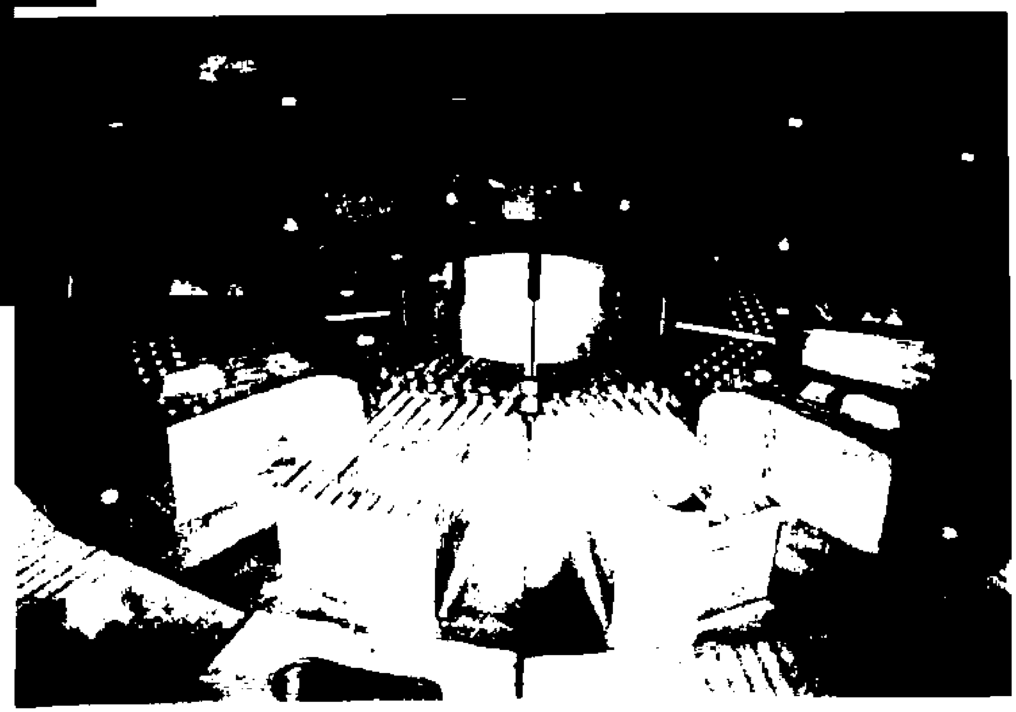


UC_2 Kernel, 200-μm diam
FEED PARTICLE

HTR-10 Graphite Reactor Internal Structures (Grade IG-110)



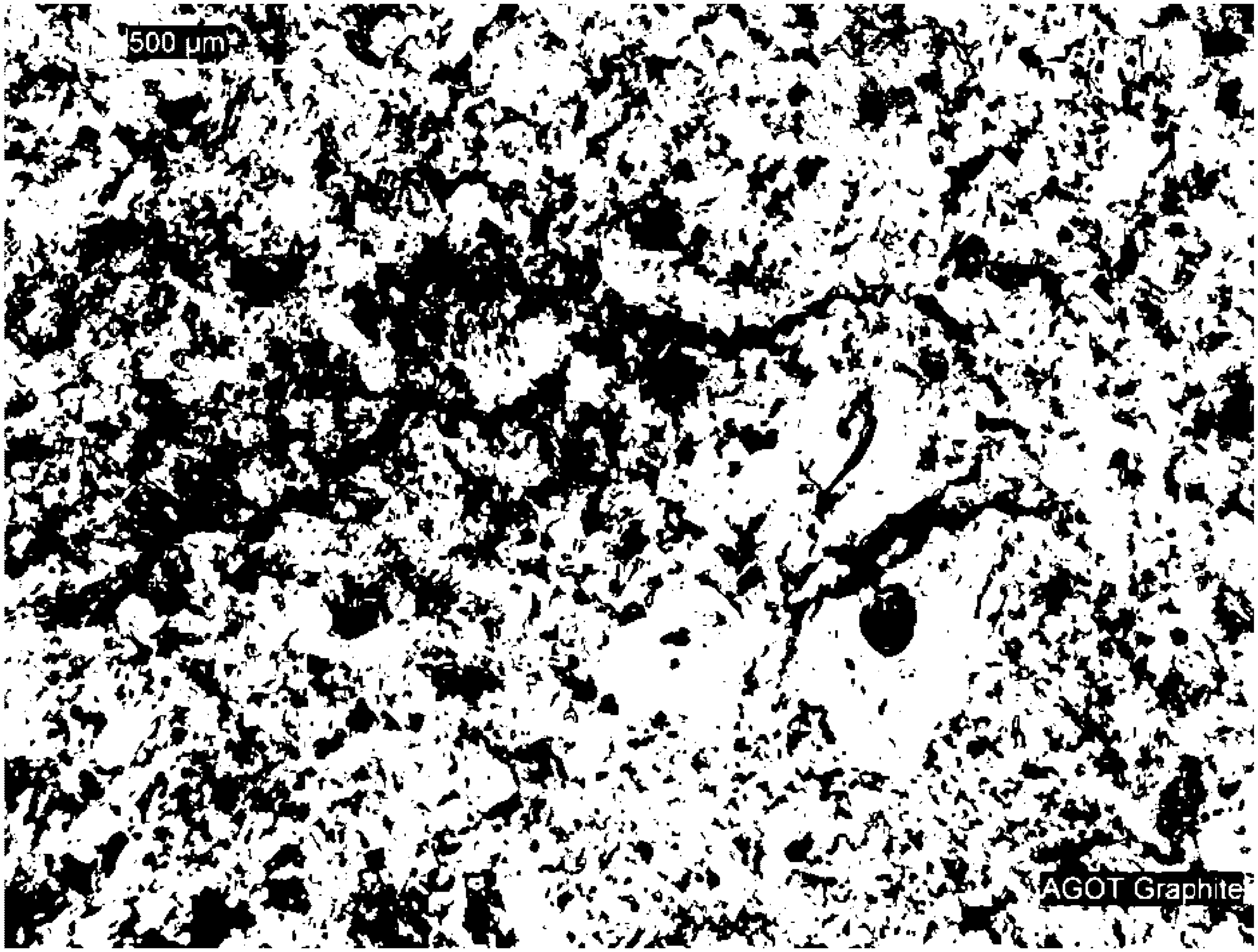
Top of the graphite core of HTR-10



Core bottom of the HTR-10 showing the fuel pebble collection area

Design of nuclear graphite for HTGRs

In The USA - AGOT Graphite



Very anisotropic irradiation induced dimensional changes

- Coarse texture
- Anisotropic needle coke
- Extruded (faster, lower cost)
- High Purity (low Boron and Sulfur content)
- Low Strength
- Nuclear Graphite – The First Years, W. P. Eatherly, J. Nucl. Mater. 100 (1981) 55 -63

Factors Controlling The Neutron Irradiation Damage Response Of Graphite

- Crystallinity (degree of graphitization): More graphitic crystals retain less displacement damage. Crystallinity is a function of precursor (pitch/coke) and graphitization temperature.
- Small crystallite sizes promotes higher strength and retardation of pore generation.
- Structural isotropy (both coke isotropy and final product isotropy). Isotropic irradiation behavior is much preferred. CTE ratio is used as an indication of isotropy. Higher coke CTE and graphite CTE preferred.
- Forming technique – structural and property anisotropy is introduced by extrusion and molding. Isostatic molding produces an isotropic graphite.

Developments in Nuclear Graphite – Process Improvements

- Purity
  - Advent of in-graphitization furnace purification
- Crystallinity
  - High crystallinity retains less radiation damage
- Filler coke size
  - Small size preferable (stronger) but larger block sizes requires coarser particles size
- Forming method
  - Isostatic pressing & vibrational molding yields less anisotropy than extrusion or molding
- Higher strength
  - Resists pore generation
- Near-isotropic (isotropic filler coke and graphite artifact)
  - Minimizes crystal strains

What Was Learned Over The Years Flowed Down To Improved Graphites:-

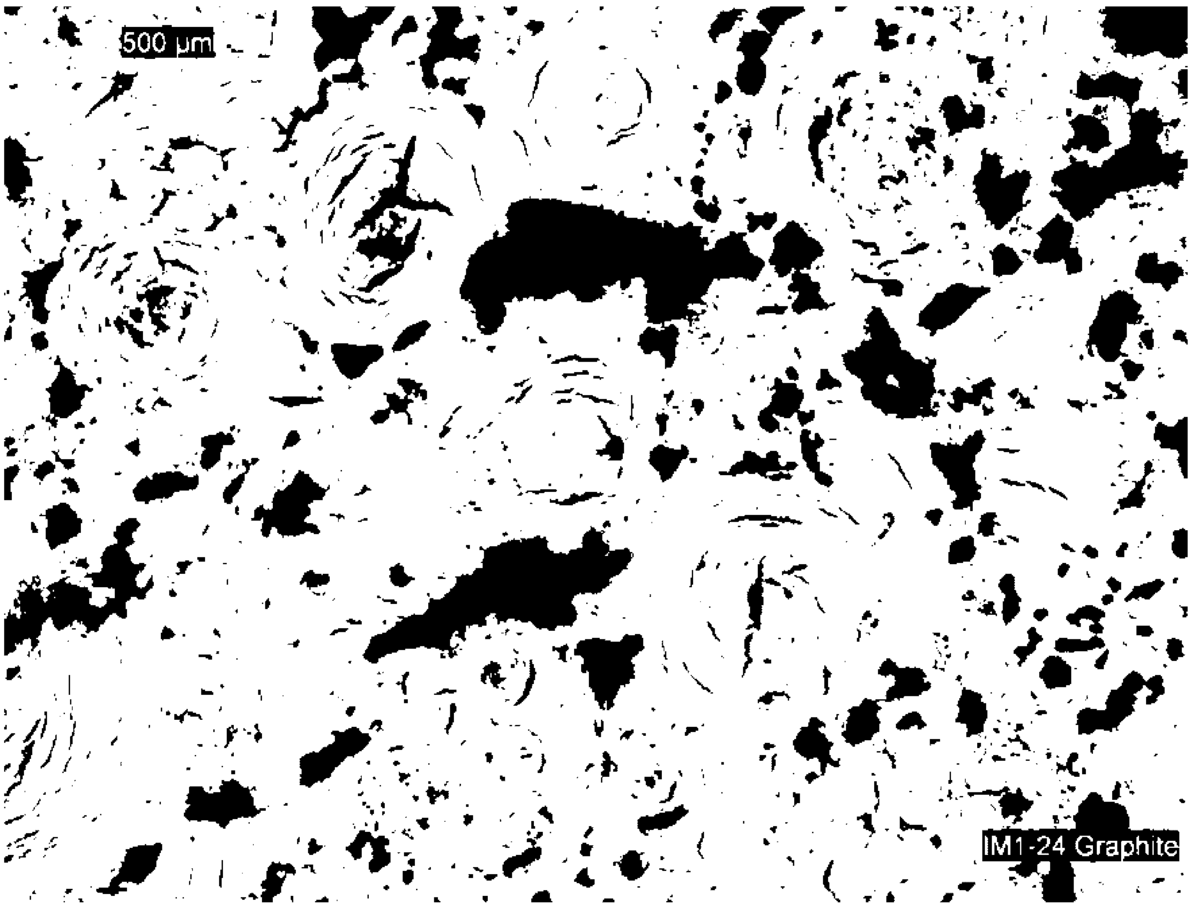
- Halogen purification (allowed alternate feedstock sources)
- Understanding of damage mechanism and role of graphite crystallite size
- Need for isotropic cokes - high CTE which yield isotropic properties in the final artifact
- Thus second generation graphites were born
  - USA, H-451 – extruded, isotropic pet coke
  - UK, IM1-24 – molded, Gilsonite coke

Near-isotropic Graphites – H-451

- Extruded, isotropic petroleum coke (NO LONGER AVAILABLE)
- 500 µm mean filler particle size
- Near-isotropic physical properties
- High CTE & reasonable strength
- Replaced H-327

Fuel elements & replaceable reflectors in the FSV HTGR (GA)

Near-isotropic Graphites – IM1-24 (UK)



- Molded, isotropic Gilsonite coke (NO LONGER AVAILABLE)
- ~500μm filler particle size
- Isotropic physical properties
- High CTE and reasonable strength
- Replaced Pile Grade A (Magnox)

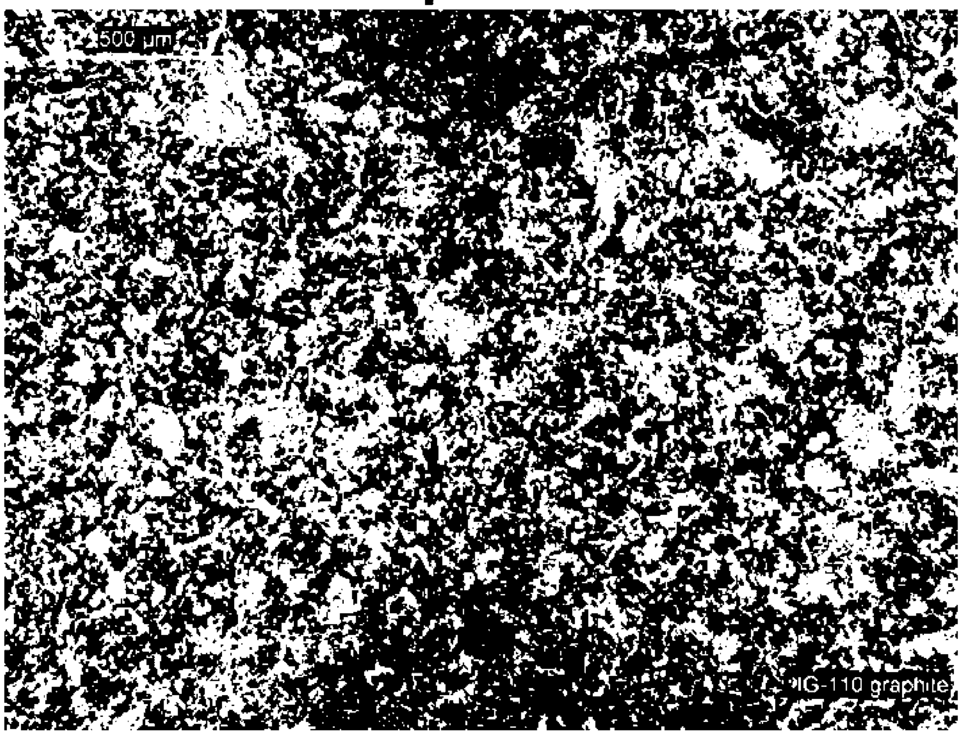
Advanced Gas-Cooled Reactor (CO_2 cooled) permanent core structure (lifetime component)

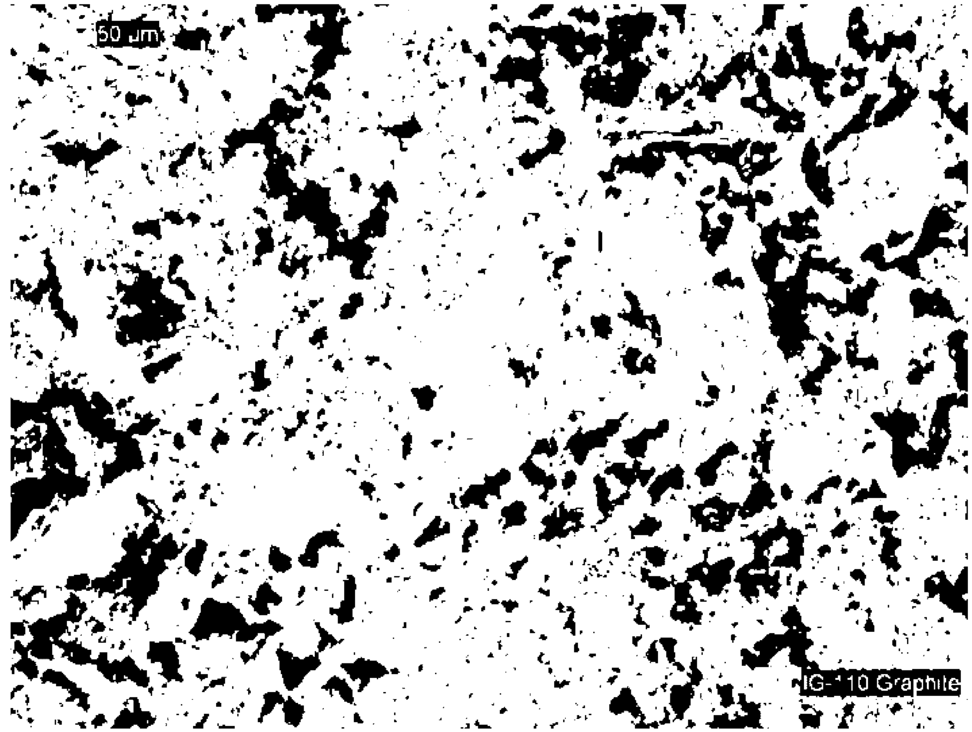
Developments In Nuclear Graphite- Near Isotropic Graphites

- Crystallinity
- Smaller particle size
- forming method (Isostatic molding)
- green coke technology
- high strength
- Isotropic
  - Properties
  - Irradiation induced dimensional change
- Third generation graphites are born

Developments in Nuclear Graphite - isotropic graphites - IG-110

- Fine grain (~20 μm)
- High CTE 4-5 x $10^{-6}\,{}^{0}C^{-1}$
- High strength
- isotropic properties and irradiation response

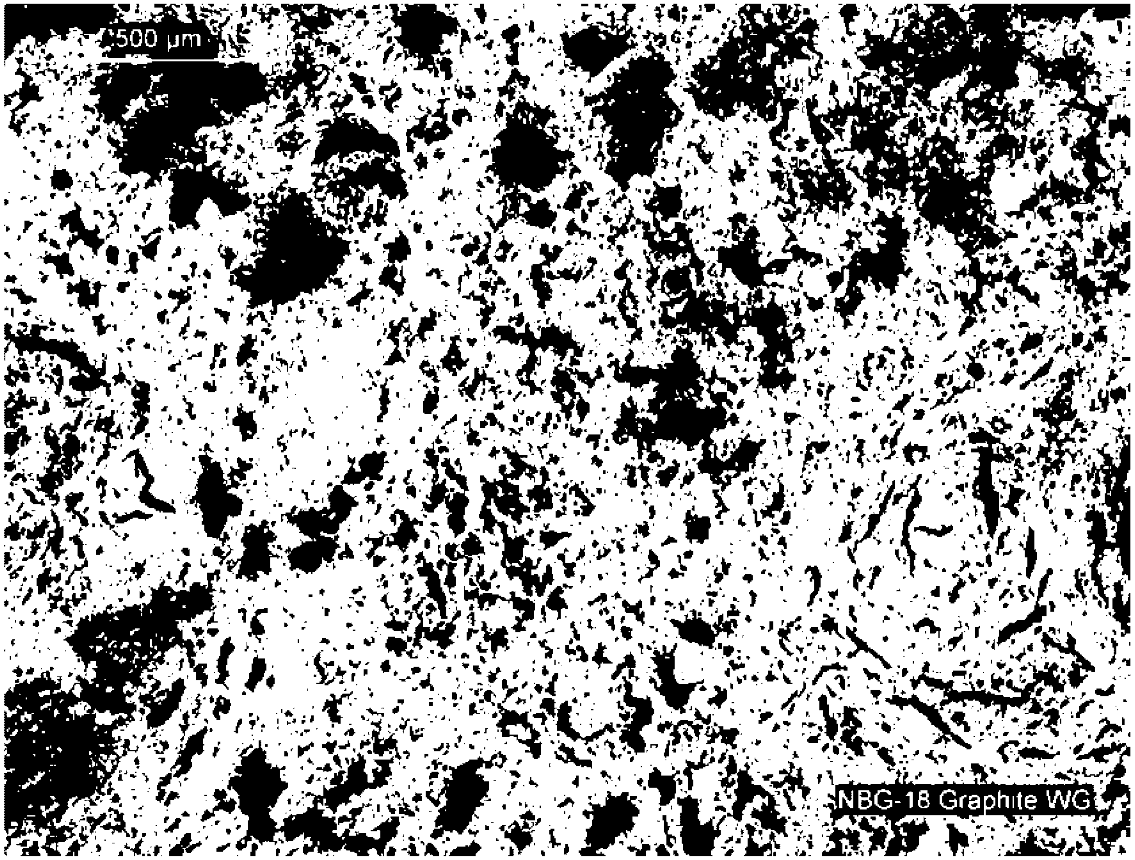




High Temperature Test Reactor (Japan), Fuel Blocks and Replaceable Reflector Blocks

HTR-10 & HTR-PM, Permanent Core Structure

Developments in Nuclear Graphite - isotropic graphites – NBG-18



- Vibrationally molded graphite
- Isotropic Pitch coke
- Medium grain (1.6 mm max)
- High CTE 5-5.5 x 10^{-6} $^0C^{-1}$
- isotropic properties and irradiation response

Permanent and replaceable core structures in the Pebble Bed Modular Reactor

Graphite and graphite testing standards

ASTM Standard Specifications

- D7219-08 Standard Specification for Isotropic and Near-isotropic Nuclear graphites

- D7301-08 Standard Specification for Nuclear Graphite Suitable for Components Subjected to Low Neutron Irradiation Dose

What is Specified by The ASTM?

- Coke type and isotropy (CTE)
- Method of determining coke CTE
- Maximum filler particle size
- Green mix recycle
- Graphitization temperature (2700°C)
- Method of determining graphitization temperature
- Isotropy ratio and chemical purity
- Properties: density, strength (tensile, compressive, flexural), CTE, E
- Marking and traceability
- Quality assurance (NQA-1)

ASTM Standard Practices

- C625 Reporting Irradiation Results on Graphite
- C781 Testing Graphite and Boronated Graphite Materials for High-Temperature Gas-Cooled Nuclear Reactor Components
- C783 Core Sampling of Graphite Electrodes
- C709 Standard Terminology Relating to Manufactured Carbon and Graphite

ASTM Standard Test Methods

- C559 Bulk Density by Physical Measurement of Manufactures Carbon and Graphite Articles
- C560 Chemical Analysis of Graphite
- C561 Ash in a Graphite Sample
- C562 Moisture in a Graphite Sample
- C565 Tension testing of Carbon and Graphite Mechanical Materials
- C611 Electrical Resistivity of Manufactured Carbon and Graphite Articles at Room Temperature

ASTM Standard Test Methods (continued)

- C651 Flexural Strength of Manufactured Carbon and Graphite Articles Using Four-Point Loading at Room Temperature
- C695 Compressive Strength of Carbon and Graphite
- C714 Thermal Diffusivity of Carbon and Graphite by Thermal Pulse Method
- C747 Moduli of Elasticity and Fundamental Frequencies of Carbon and Graphite by Sonic Resonance
- C748 Rockwell Hardness of Graphite Materials

ASTM Standard Test Methods (continued)

- C749 Tensile Stress Strain of Carbon and Graphite
- C769 Sonic Velocity in Manufactured Carbon and Graphite for Use in Obtaining Young's Modulus
- C816 Sulfur in Graphite by Combustion-Iodometric Titration Method
- C838 Bulk Density of As-Manufactured Carbon and Graphite Shapes
- C886 Scleroscope Hardness Testing of Carbon and Graphite Materials

ASTM Standard Test Methods (continued)

- C1025 Modulus of Rupture in Bending of Electrode Graphite
- C1039 Apparent Porosity, Apparent Specific Gravity, and Bulk Density of Graphite Electrodes
- C1179 Oxidation Mass Loss of Manufactured Carbon and Graphite Materials in Air
- Dxxxx Oxidation Rate and Threshold Oxidation Temperature for Manufactured Carbon and Graphite in Air

New ASTM Test Methods Currently in Development

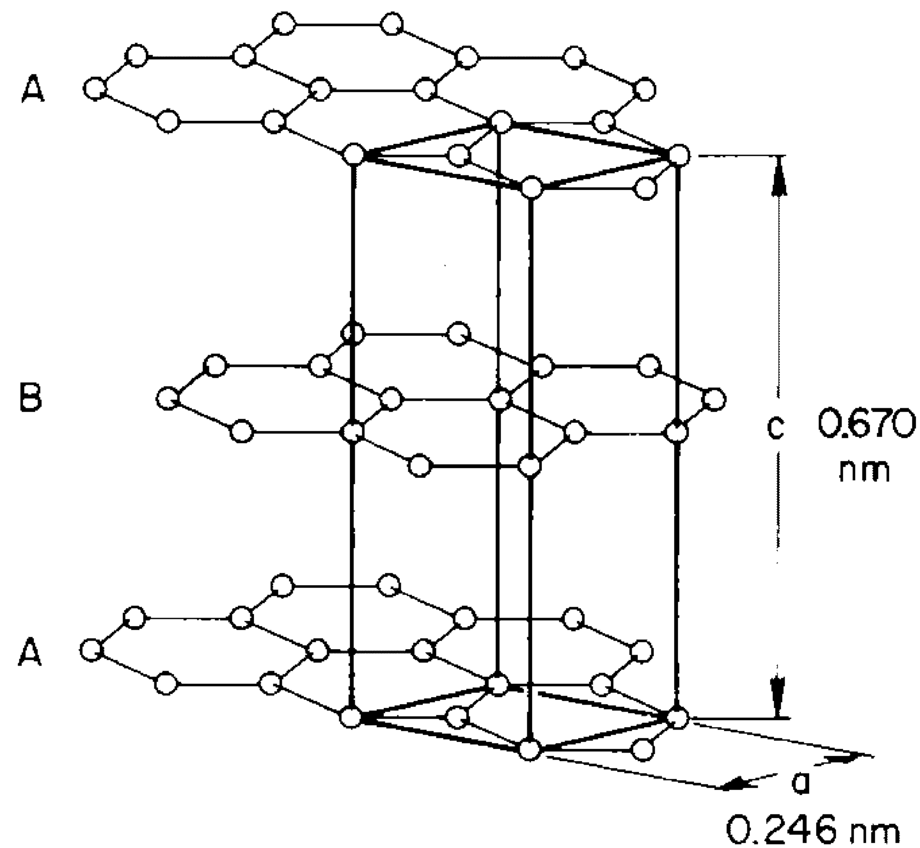
- ASTM D02.F on manufactured carbons and graphites has several test methods in development
  - Critical stress intensity factor
  - Shear modulus and Poisson's ratio from sonic velocity
  - Flexural strength by three point bend
  - Chemical purity by ICP- OES and GDMS
  - Small (irradiation) specimen best practice
  - Non-destructive test and evaluation
  - X-Ray diffraction analysis

Physical properties and irradiation effects

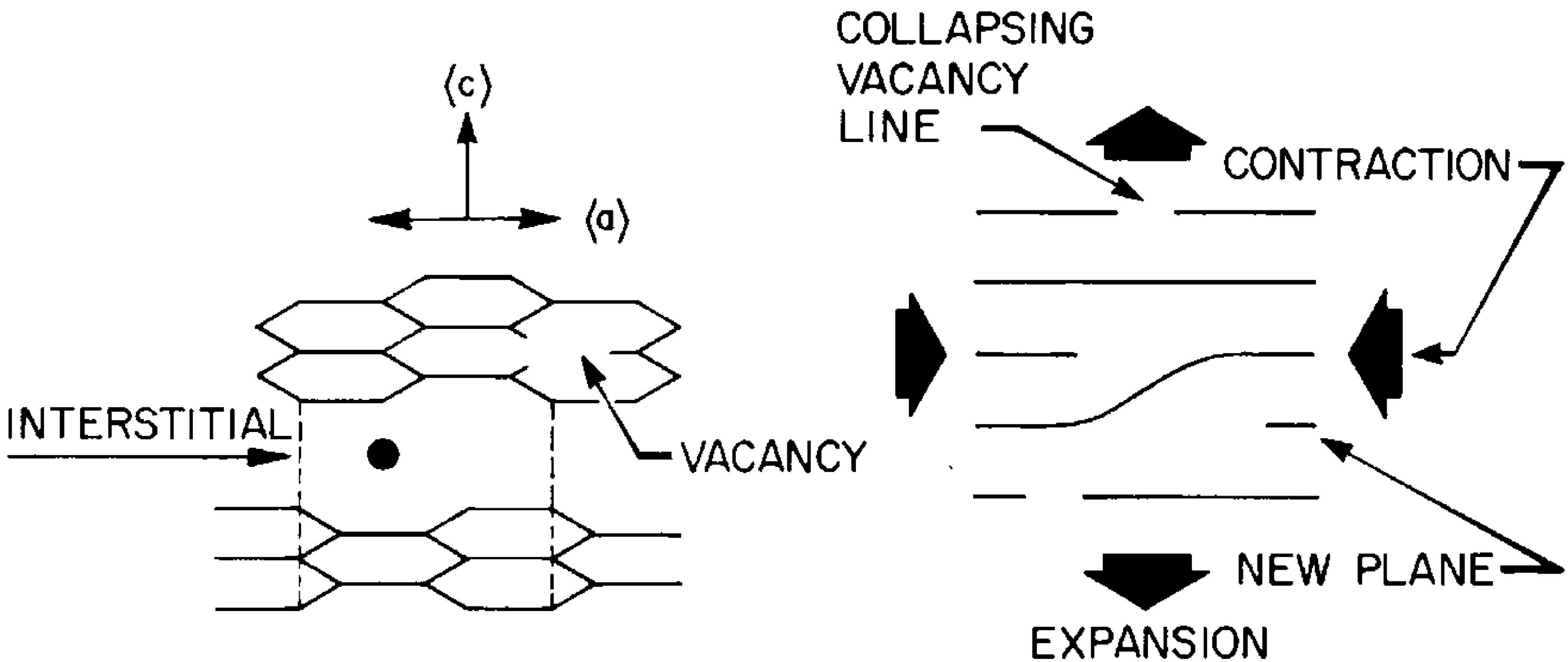
Neutron Irradiation Damage

- Neutron irradiation causes
- carbon atom displacement
- Dimensional and physical property changes result
- Damage mechanism well understood
- Key physical properties are: irradiation dimensional stability, strength, thermal expansion coefficient, thermal conductivity, radiation creep behavior, fracture behavior, oxidation behavior.

GRAPHITE CRYSTAL STRUCTURE



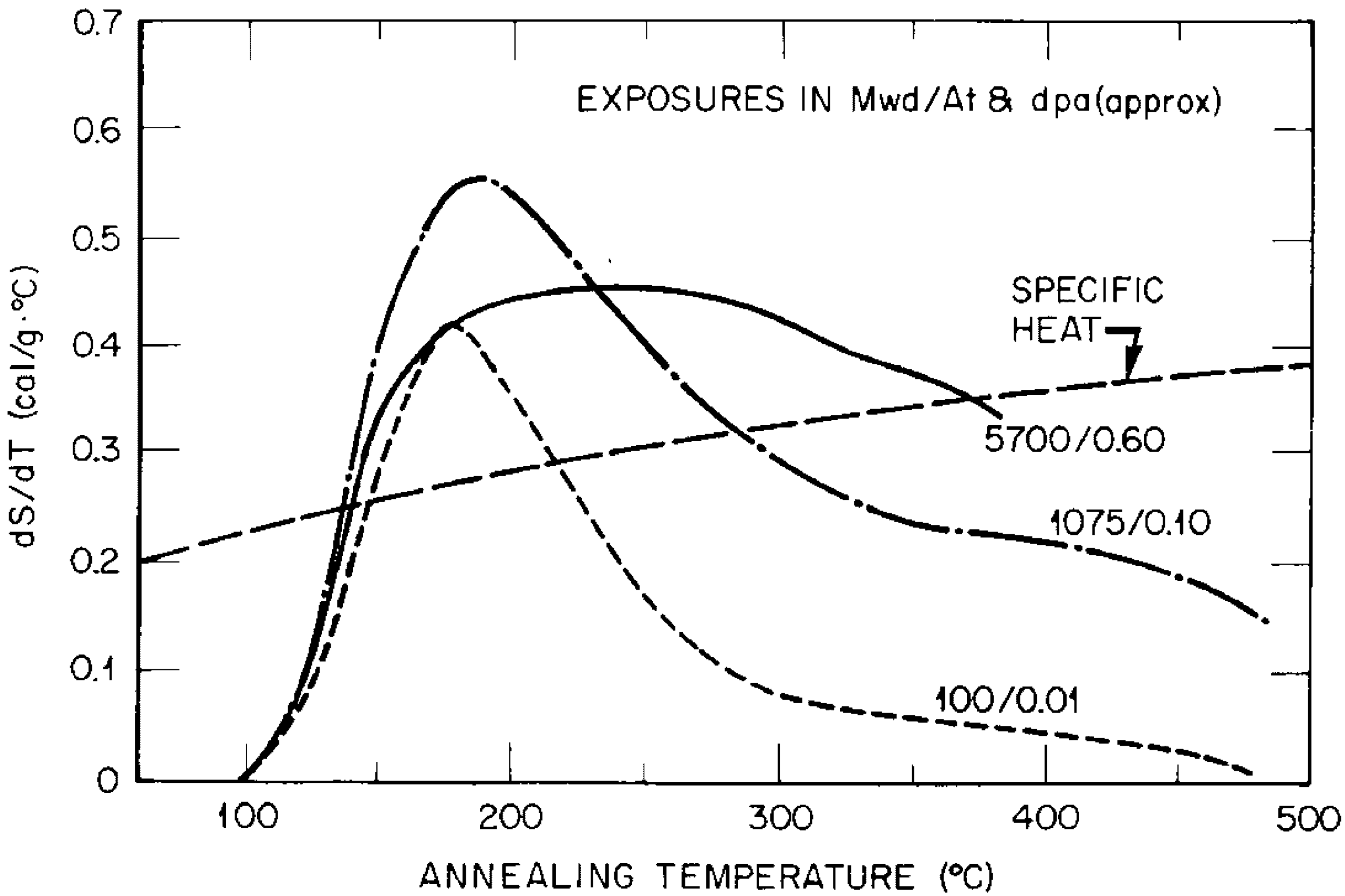
The Radiation Damage Mechanism In Graphite



CARBON ATOM BINDING ENERGY IN GRAPHITE LATTICE IS 7 eV

DISPLACEMENT ENERGY FOR CARBON ATOM IS APPROX. 30 eV

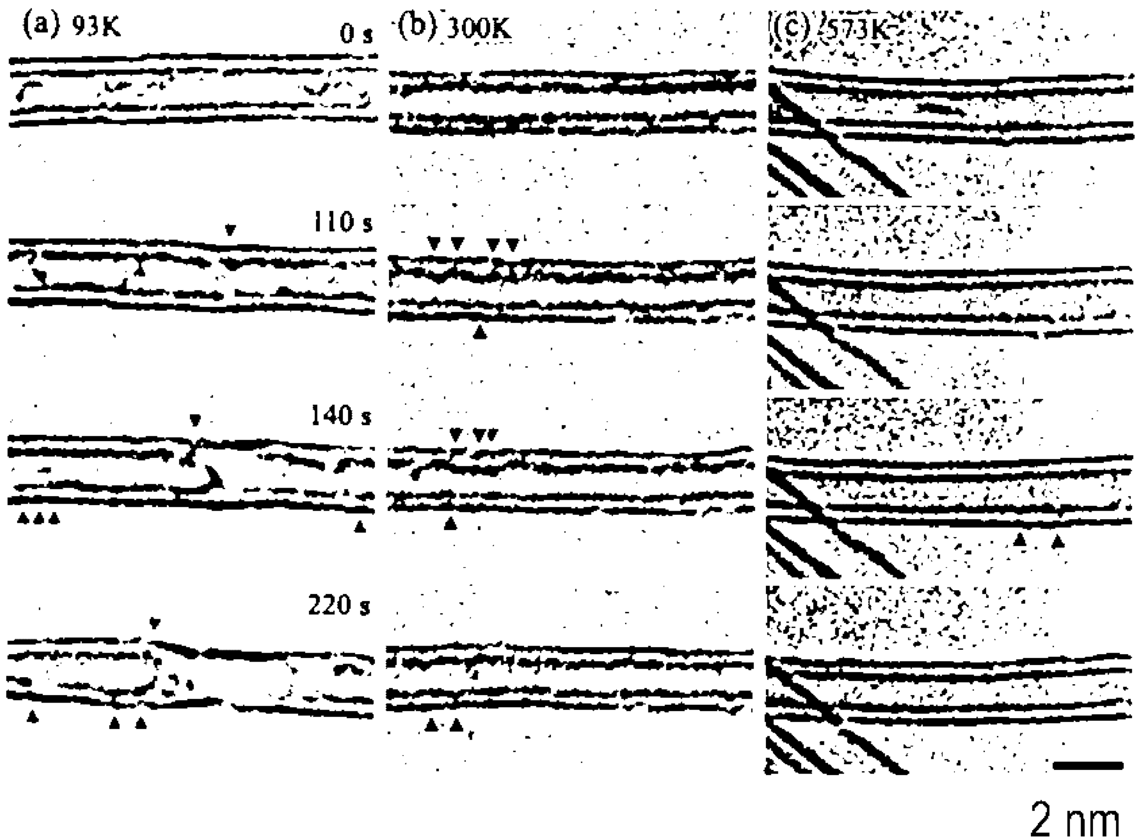
Low Temperature Stored Energy Release



- T_{irr} ~ 30°C
- Hanford K
- reactor test
- Data
- Traditionally associated with Frenkel pair recombination

Burchell T, Carbon Materials for Advanced Technologies, Chpt. 13 (1999) p. 429

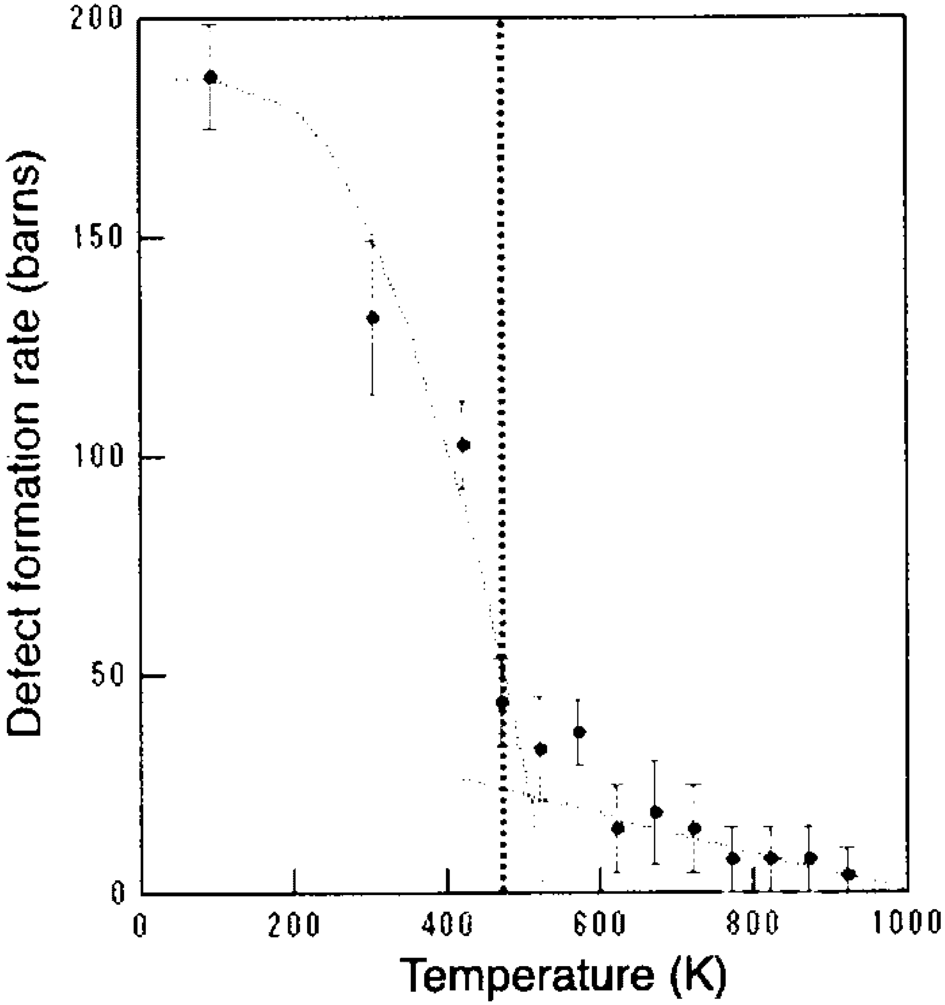
Displacement Damage in Layered Graphitic Structures



- **Sequential HRTEM images illustrating the formation rates of interlayer defects at different temperatures with the same time scale (0 to 220 seconds). (a) 93K, (b) 300K, (c) 573K, in double-wall carbon nanotubes.**
- **The arrows indicate possible interlayer defects.**

Urita, K.; Suenaga, K.; Sugai, T.; Shinohara, H.; Iijima, S. *Physical Review Letters* **2005**, 94, 155502.

Displacement Damage in Layered Graphitic Structures

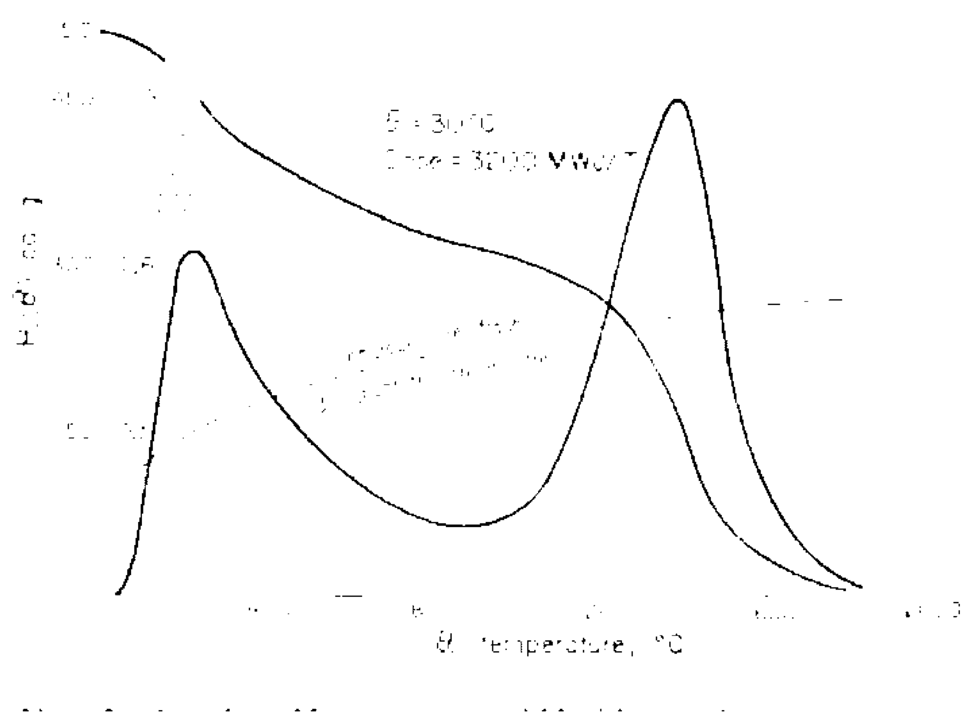


- Normalized formation rate of the clusters of *I-V* pair defects per unit area of bilayer estimated in HRTEM images recorded at different temperatures
- The dotted line shows the known temperature for Wigner-energy release (~473 K)

Urita, K.; Suenaga, K.; Sugai, T.; Shinohara, H.; Iijima, S. *Physical Review Letters* **2005**, 94, 155502.

High Temperature Stored Energy Release

Stored Energy Release Curve for Graphite Irradiated at 30°C Compared with Unirradiated Graphite Cp Curve

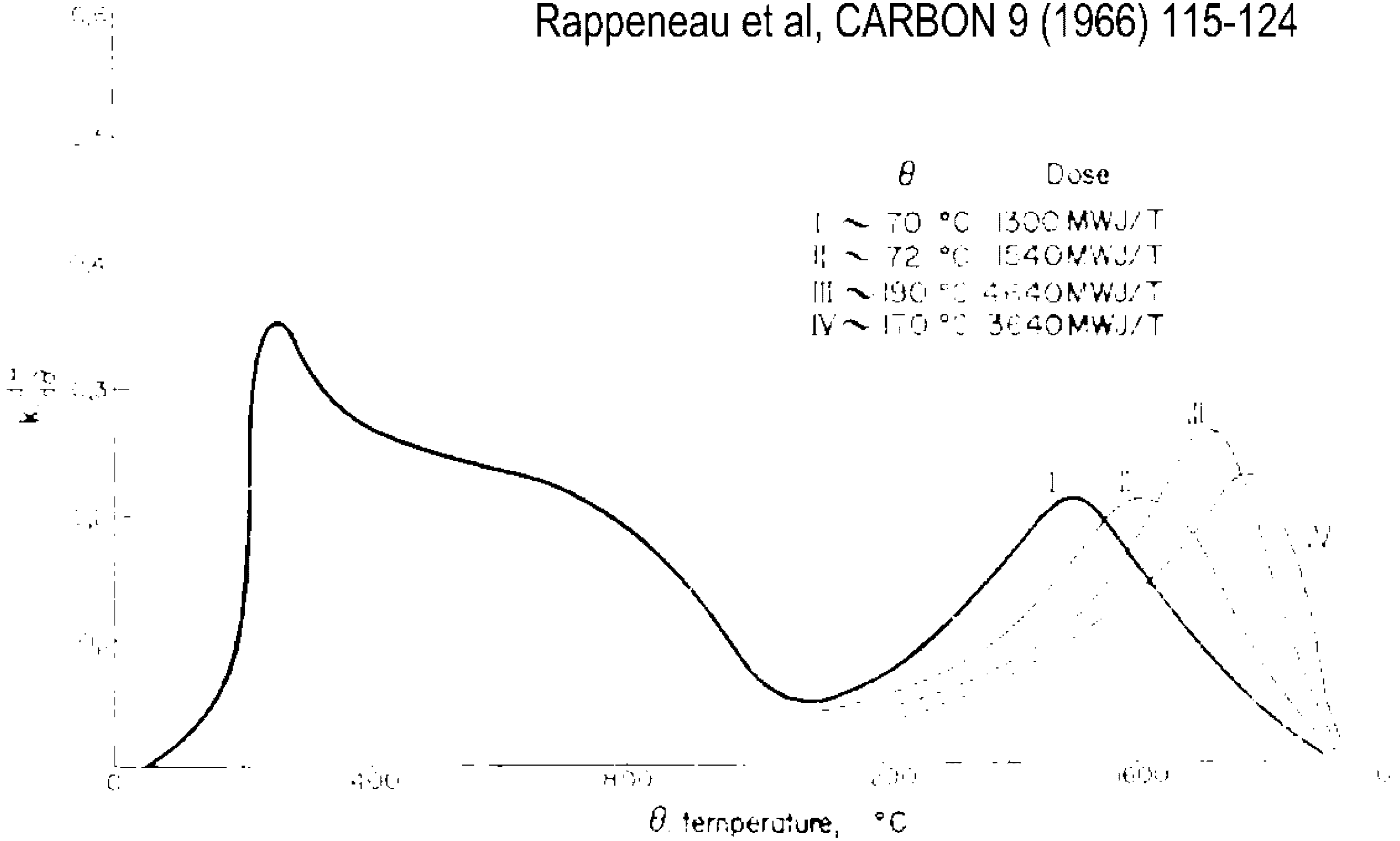


Rappeneau et al, CARBON 9 (1966) 115-124

- A second release peak is observed at ~1400°C in graphite irradiated at LOW temperatures
- Associated with annealing of small interstitial clusters
- Immobile vacancies can coalesce at high temperature
- Release rates > Cp NOT seen in graphite irradiated at higher temperatures

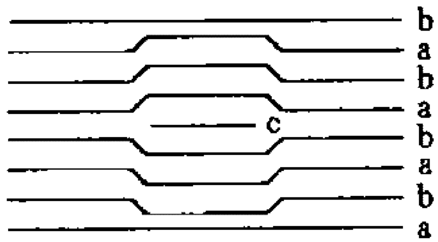
High Temperature Stored Energy Release

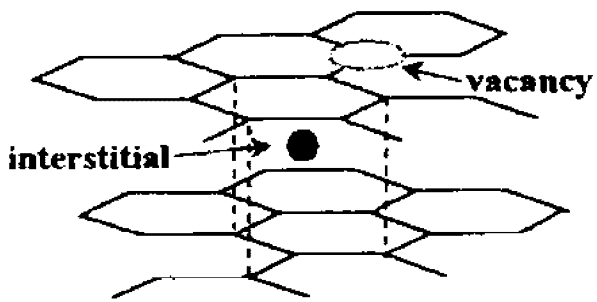
Rappeneau et al, CARBON 9 (1966) 115-124

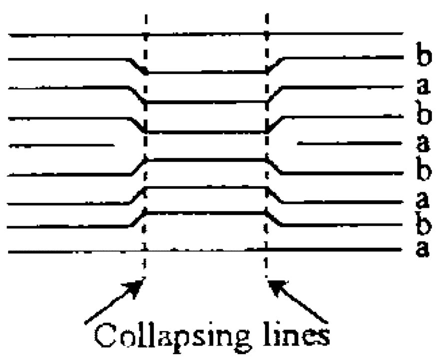


- High temperature release is due to a separate mechanism
- Release rate does NOT exceed Cp

Radiation Damage In Graphite Is Temperature Dependent







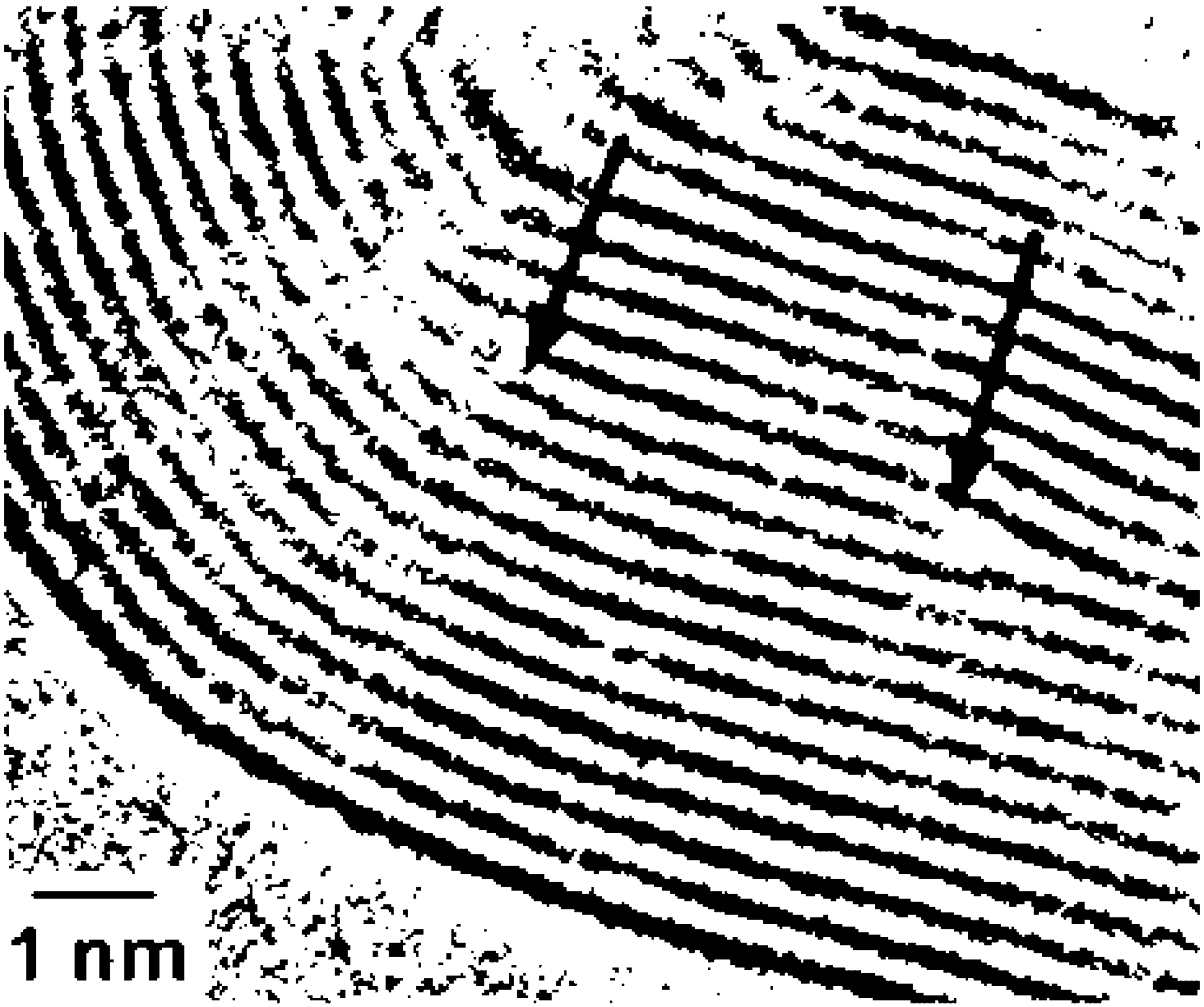
INTERSTITIALS

Mobile at room temperature.
Above ~200°C form into clusters of 2 to 4 interstitials.
Above 300°C form new basal planes which continue to grow at temperatures up to 1400°C.

VACANCIES

Immobile below 300°C.
300-400°C formation of clusters of 2-4 vacancies which diffuse in the basal planes and can be annihilated at crystallite boundaries (function of lattice strain and crystal perfection).
Above 650°C formation of vacancy loops.
Above 900°C loops induce collapsing vacancy lines.

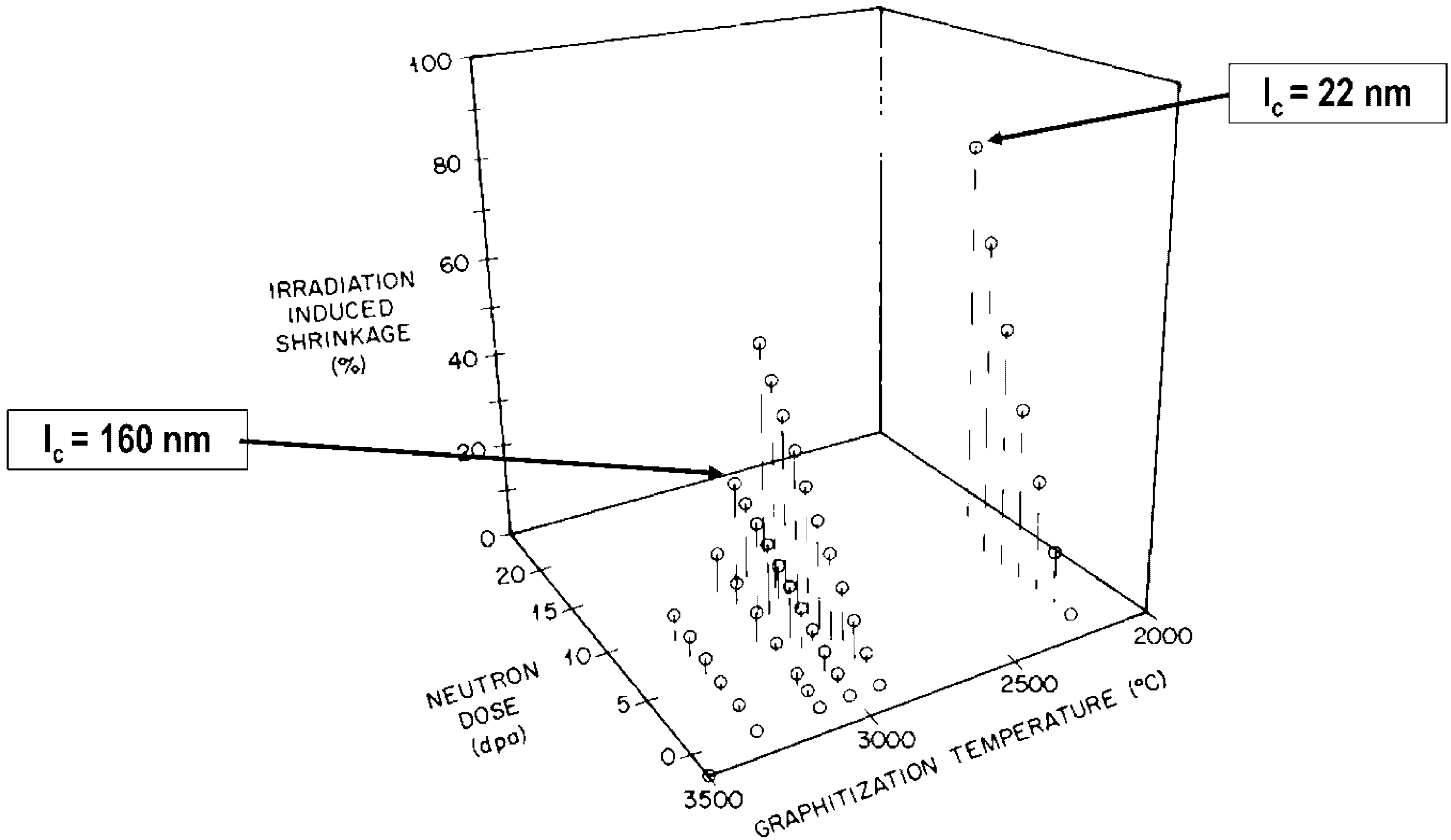
The Creation of New Basal Planes in Layered Graphitic Structures



Banhart, F. *Rep. Prog. Phys.* **1999**, 62, 1181–1221.

A high-resolution electron micrograph showing the basal planes of a graphitic nano-particle with an interstitial loop between two basal planes, the ends of the inserted plane are indicated with arrows.

The Influence of Crystallinity on the <a>-axis Shrinkage of Pyrolytic Graphite



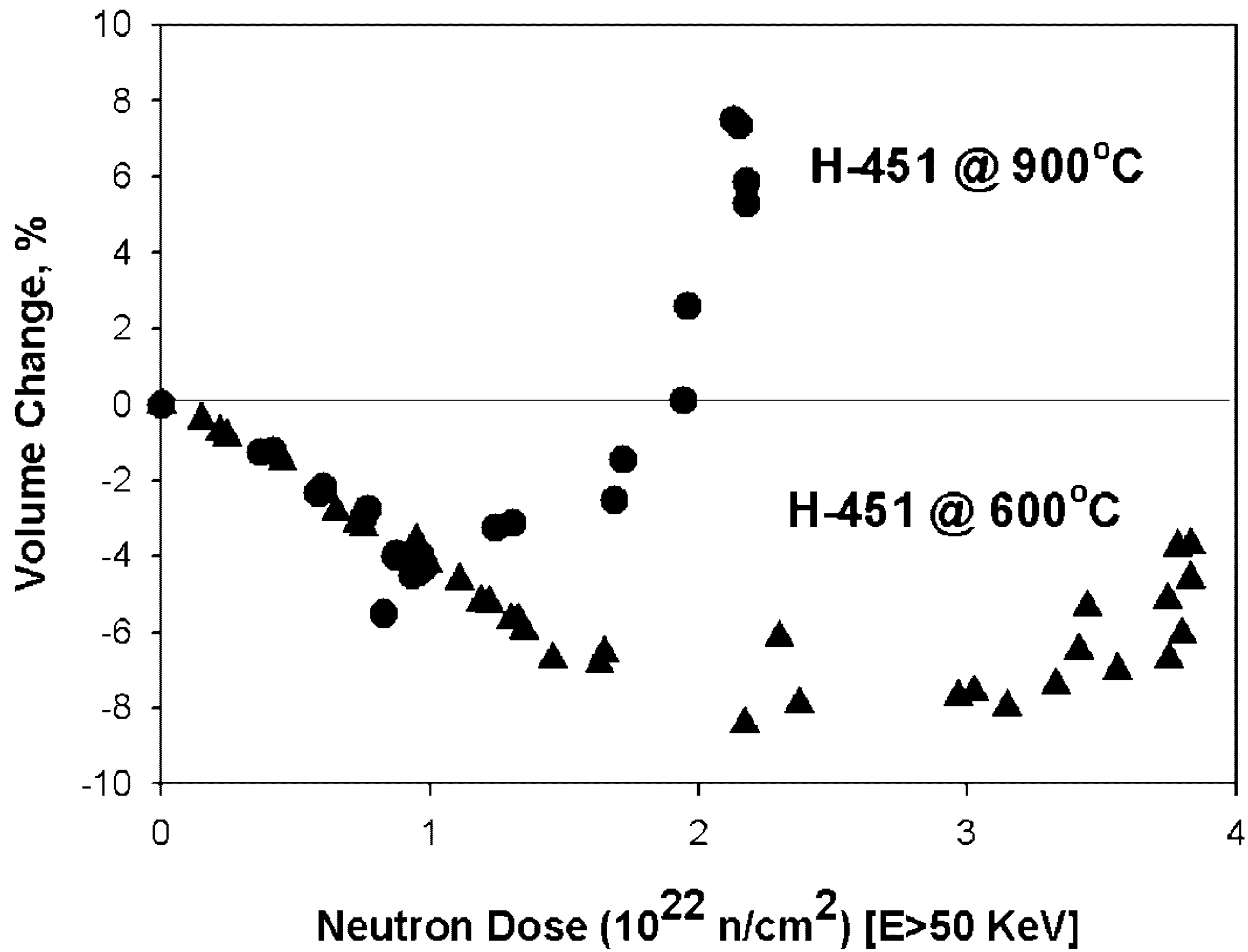
Neutron Irradiation Induced Dimensional Change

- Graphite dimensional changes are a result of crystallite dimensional change and graphite texture.
- Swelling in c-direction is initially accommodated by aligned microcracks that form on cooling during manufacture.
- Therefore, the a-axis shrinkage initially dominates and the bulk graphite exhibits net volume shrinkage.
- With further irradiation, incompatibilities in crystallite strains causes the generation of new porosity and the volume shrinkage rate falls eventually reaching zero.

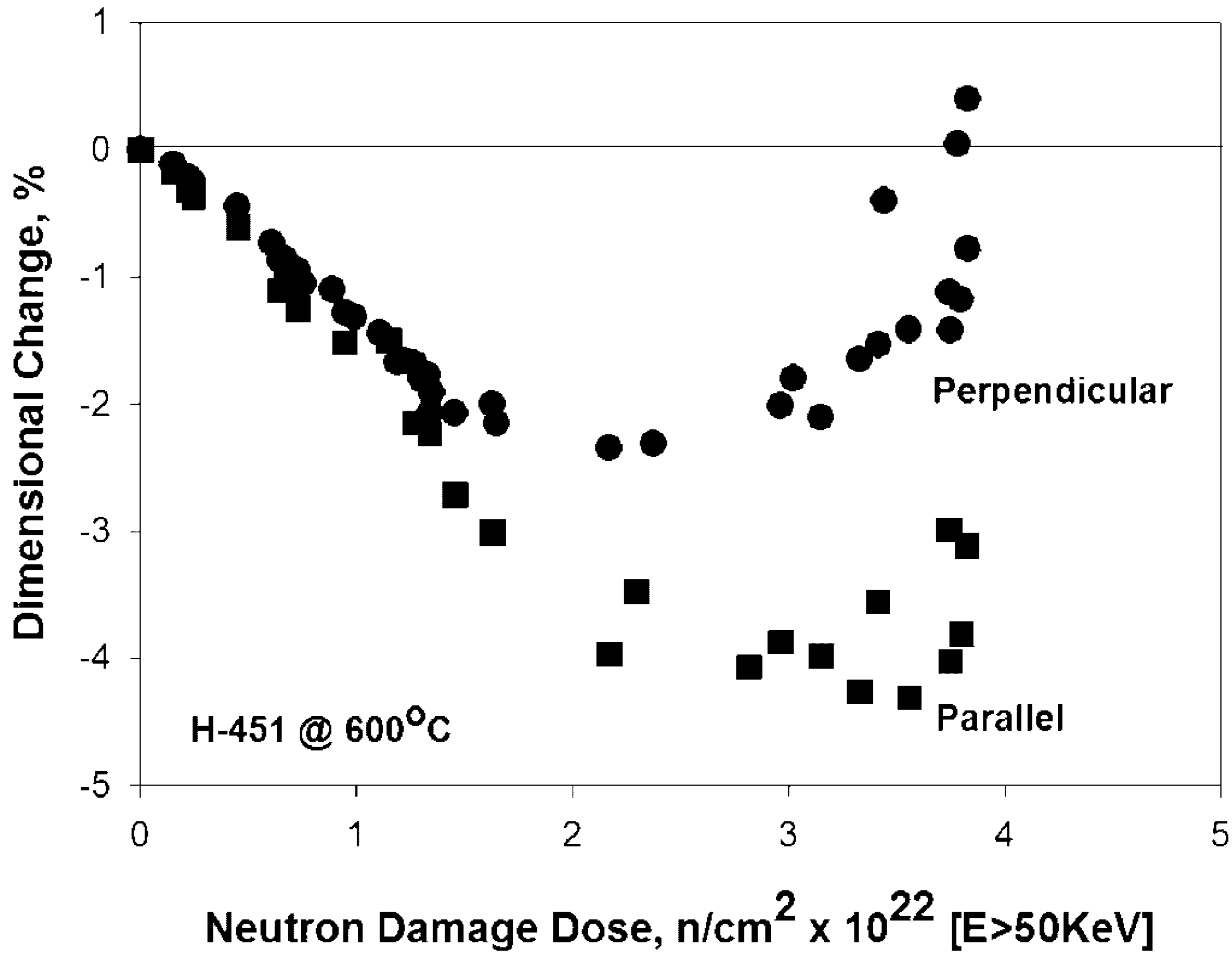
Neutron Irradiation Induced Dimensional Change (Continued)

- The graphite begins to swell at an increasing rate with increasing damage dose due to c-axis growth and new pore generation.
- The graphite thus exhibits volume “turnaround” behavior from initial shrinkage to growth.
- Eventually disintegration occurs due to excessive pore/crack generation.

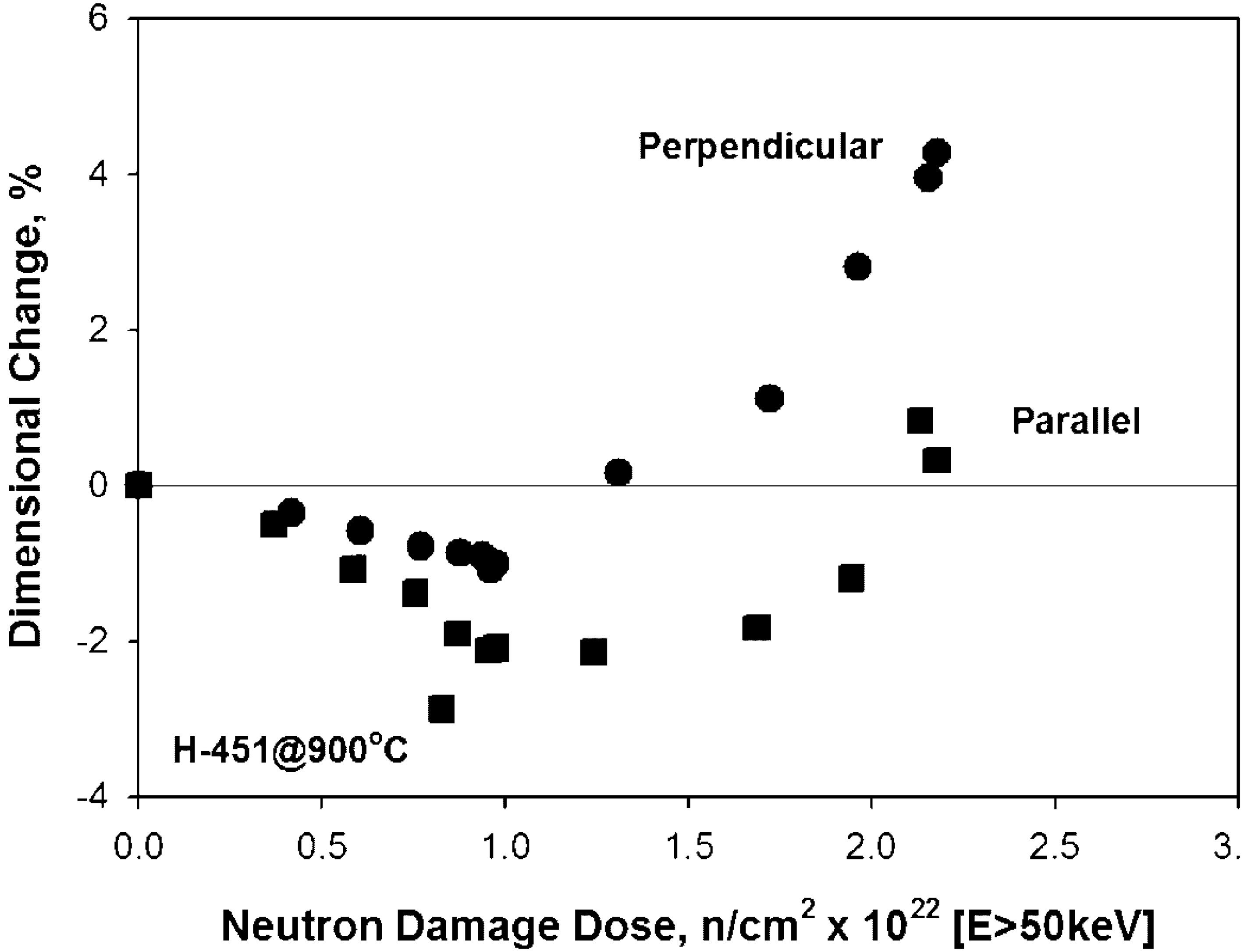
Radiation Induced Dimensional Changes in H-451 (Effect of Temperature)



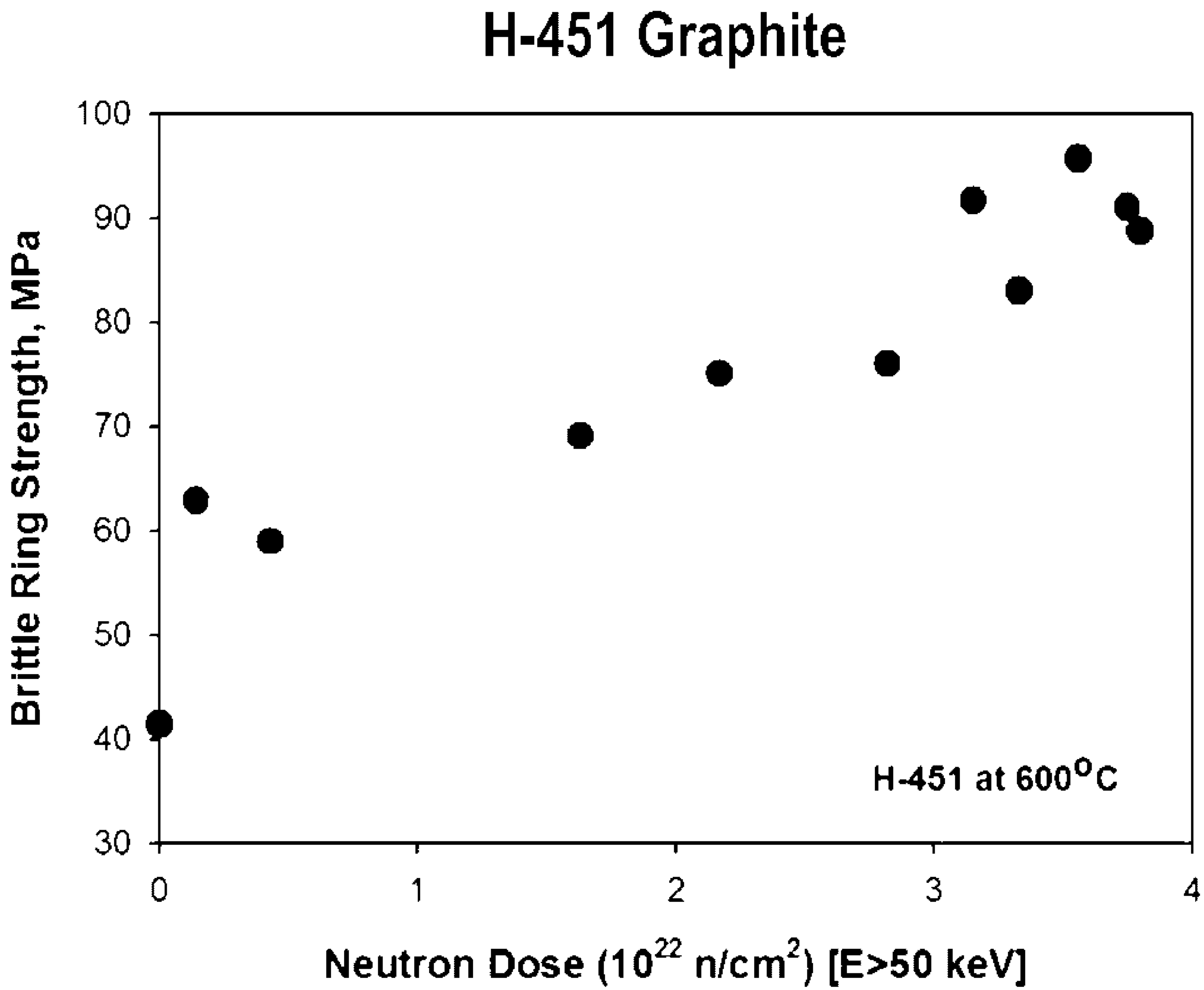
Radiation Induced Dimensional Changes in H-451 (Effect of Texture)



Radiation Induced Dimensional Changes in H-451 (Effect of Texture)

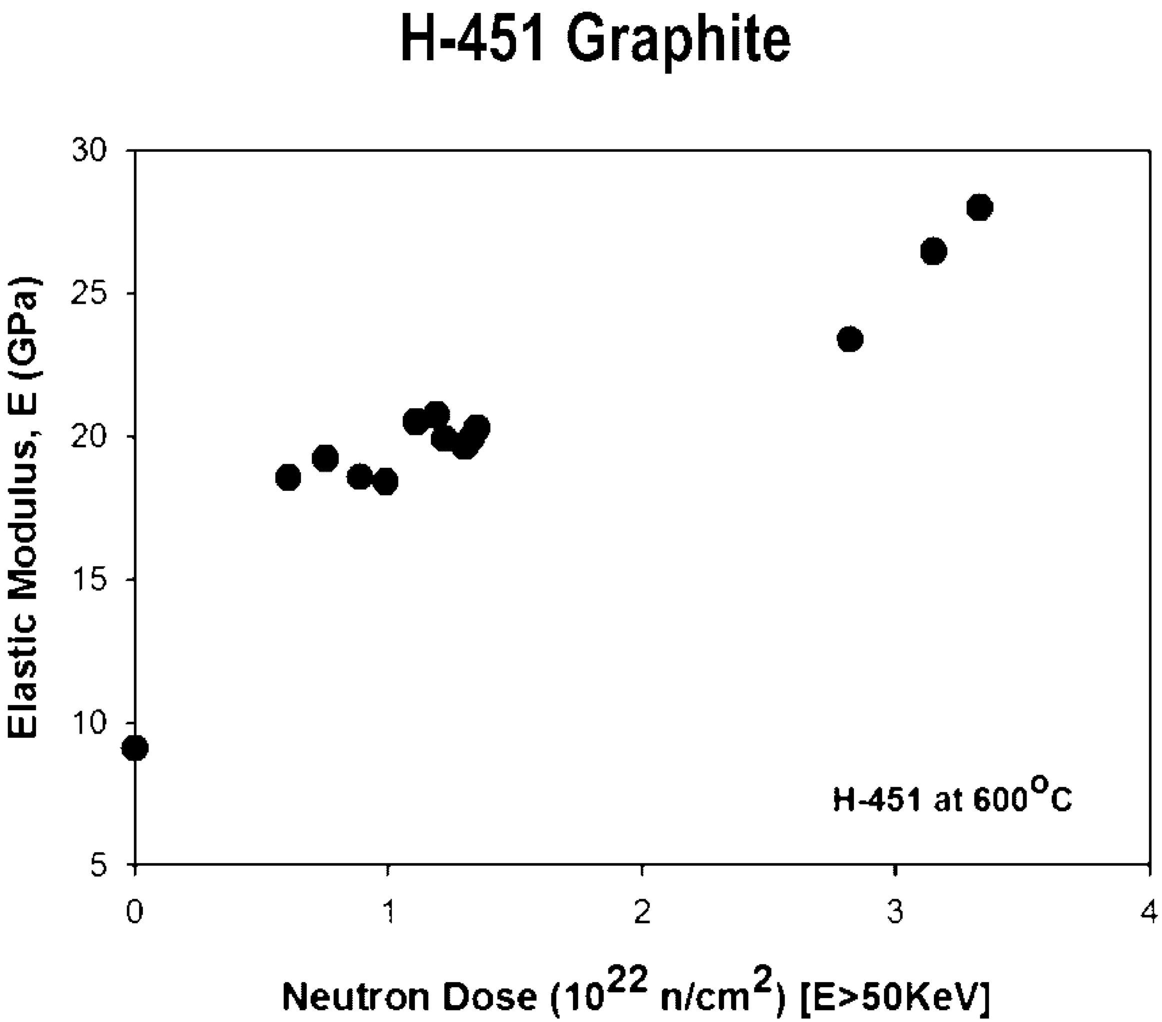


Neutron Irradiation Induced Changes in Fracture Strength



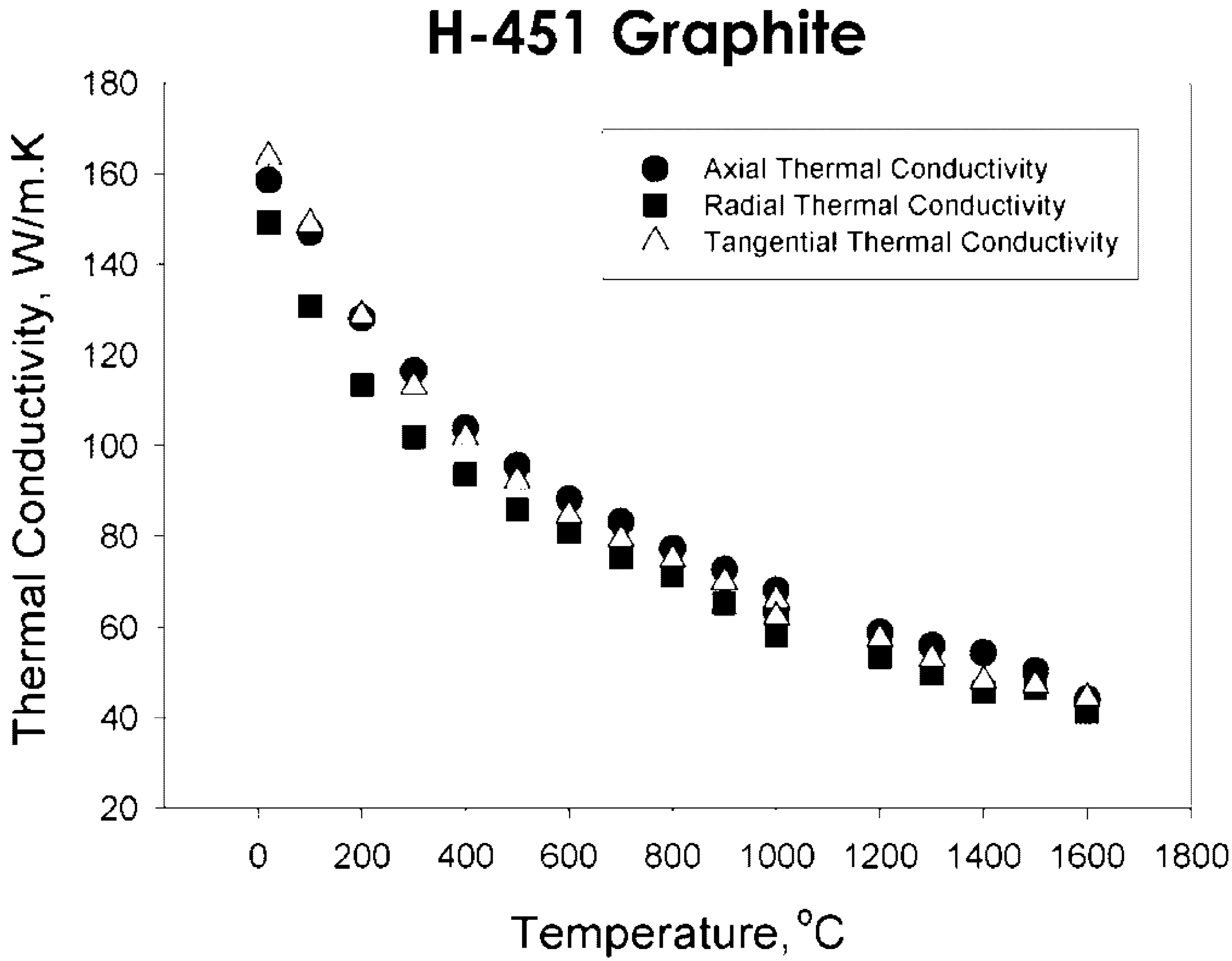
- Initial increase due to dislocation pinning
- Subsequent changes due to pore closure and new pore generation
- $K=s[pc]^{1/2}$
- Critical flaw (unirradiated) approximately 1 mm

Neutron Irradiation Induced Changes in Young's Modulus



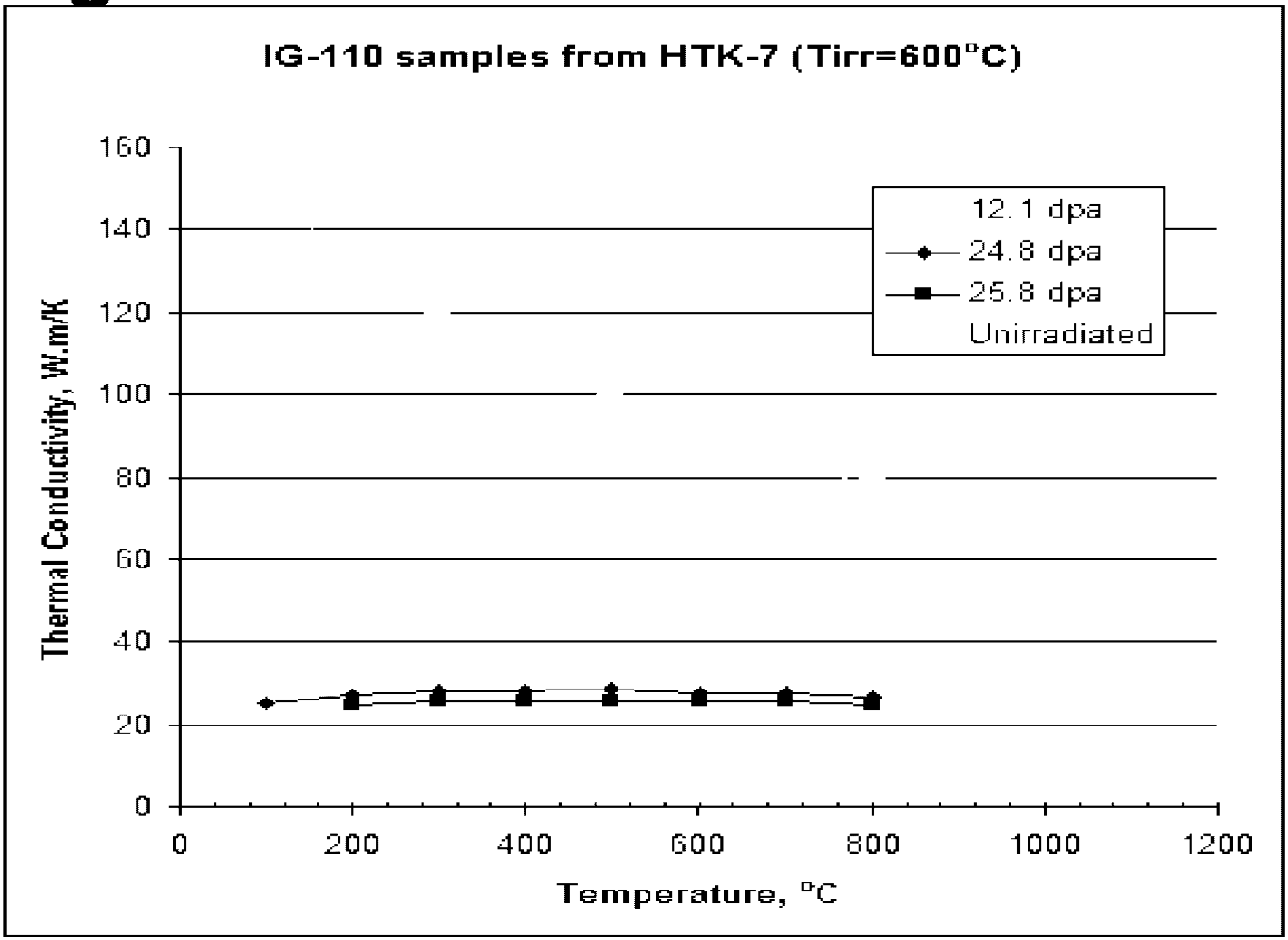
- Initial rise due to dislocation pinning
- Subsequent increase due to volume shrinkage (densification)
- Eventual turnover and reduction due to pore/crack generation and volume expansion
- $\sigma \propto (E)^{1/2}$

Graphite Thermal Conductivity is Temperature Dependent



Phonon-phonon scattering (Umklapp scattering) decreases Tc with increasing temperature

IG-110 Thermal Conductivity Changes



Irradiation Induced Dimensional Changes Result in Differential Strains

- Weaker graphites crack (pore generation)
- Stronger graphites resist pore generation and strains creep out (irradiation creep)
- Radiation creep is a two stage phenomena
- Primary (reversible) creep strain α $(1/E_0)$
- Secondary (irreversible) creep strain $f(\sigma,\gamma,E_0)$
- Mechanism of creep subject of disagreement
- Two effects must contribute
  - In-crystal deformation
  - Pore generation/pore re-orientation
- At high doses we must allow for structural changes
- Irradiation induced creep in graphite is the subject of a new IAEA Coordinated Research Project

International graphite irradiation programs

International Graphite Irradiation Programs

- European Framework (6^{th}, 7^{th}, 8^{th} ….)
  - Comprehensive irradiation program of available candidate graphites
- South Africa
  - MTR program (conducted at ORNL) for NBG-18 covers relevant dose and temperature range to PBMR (ON HOLD)
- China
  - Plans an MTR Program relevant to HTR-DM (IG-110)
- USA (DOE)
  - NGNP Graphite irradiation program for candidate graphites (See Technology Development Plan)
- International data will become available through the Gen IV International Forum

Graphite oxidation and other chemical reactions

Radiolytic Oxidation is Not a Problem In He Cooled HTRs

- $CO_2 + \gamma = CO_2^*$, an activated species that can gasify carbon at reactor temperatures
- Radiolytic weight loss can degrade physical properties
- Special measures include gaseous phase inhibitors
- Helium cooled reactors are immune from radiolytic oxidation
- Air/steam oxidation can occur in all graphite moderated reactors and will cause property degradation

Thermal oxidation (Air and Moisture)

- Air/steam oxidation can occur in all graphite moderated reactors and will cause property degradation
- Air ingress accident
  - $C+O_2 \rightarrow CO_2$
  - $CO_2+C \rightarrow 2CO$
- Moisture in Helium Coolant
  - $C + H_2O \rightarrow CO + H_2$
  - $C + 2H_2 \rightarrow CH_4$
- Oxidation = Loss of solid Carbon (Graphite)

Thermal oxidation (Air and Moisture)

- Properties degrade as a function of oxidative weight loss (burn-off)
- To predict burn-off we need to know:
  - Kinetics of oxidation reactions over the appropriate range of temperature and partial pressure (or concentration) of oxidizing species
  - Local partial pressure (or concentration) of oxidizing species within core/graphite block (Effective Diffusivity)
- Graphite purity also has an effect since some impurities act as oxidation catalysts

Erosion of graphite - tribology

Erosion of graphite - tribology

- Tribological data are needed to establish wear of components
- Friction Coefficients (in Helium, effect of pressure and temperature)
  - Graphite on graphite
  - Pebble on Pebble
  - Pebble on Graphite
- Wear rates need to be established
- Wear products (dust) are a fission product vector

Graphite performance modeling

Graphite Performance Modeling

Graphite Performance Modeling Requires:

- Whole core graphite behavioral model
  - How large are the stress?
- Fracture Model or Failure Theory
  - Do the stresses cause fracture?
- Assessment Criteria
  - What are the consequence of brick/block failure for core integrity?

Graphite Performance Modeling

- Whole core graphite behavioral model requires:
  - Stress analysis, constitutive equation
  - $\varepsilon_{Total} = \varepsilon_e + \varepsilon_t + \varepsilon_d + \varepsilon_c$
  - Core temperature (T) and dose distribution (γ)
  - Dimensional change data and model
  - Creep data and model, f (T, γ, σ)
  - Property change data and models, Tc, CTE, E, σ as a f (T, γ)

Graphite Performance Modeling

- Fracture Model or Failure Theory
  - Weibull model
  - Burchell model
  - CARES model
  - Fracture Mechanics
  - Maximum Deformation Energy Theory (ASME)
  - Maximum Strain Energy Theory
  - Maximum Principal Stress
  - Etc.

Graphite Performance Modeling

- Assessment Criteria
  - Consequence of brick/block failure for core integrity
  - Core structural redundancy
  - Fitness for purpose
  - In core monitoring to confirm predictions and increase confidence in core integrity
  - Replaceable components

Graphite Performance Modeling

- Need to determine the effect of weight loss on property
- Need to predict extent of property degradation
- Work in hand at INL and ORNL to determine oxidation kinetics and effect of oxidation on properties for candidate graphites
- Oxidation is a potential FP transport mechanism

Regulatory challenges

Regulatory Challenges

- For detailed analysis see:
  - NRC Graphite PIRT
  - NRC Graphite Experts Panel Report & Recommendations
- Acceptance/Endorsement of ASME GCC Code
- Assimilating unirradiated baseline characterization data from DOE programs
- Assimilating irradiated properties data from DOE and international (GIF) programs
- HT Stored Energy Release (being addressed)
- Graphite oxidation (effective diffusivity of species)
- Irradiation induced creep, a full understanding (IAEA Coordinated Research Project)

Summary

- > 60 years experience with graphite as a solid moderator
- Mechanism of radiation damage well understood
- A few grey areas remain
  - High temp stored energy release
  - Whole core models (and material models)
  - Irradiation creep
  - Tribology & wear
  - Effective diffusivity (oxidative weight loss)

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NRO Intellectual Curiosity Series

Basis for NRC Requirements on Pressurized Thermal Shock

6th November 2013
Mark Kirk, RES/DE/CIB

Agenda

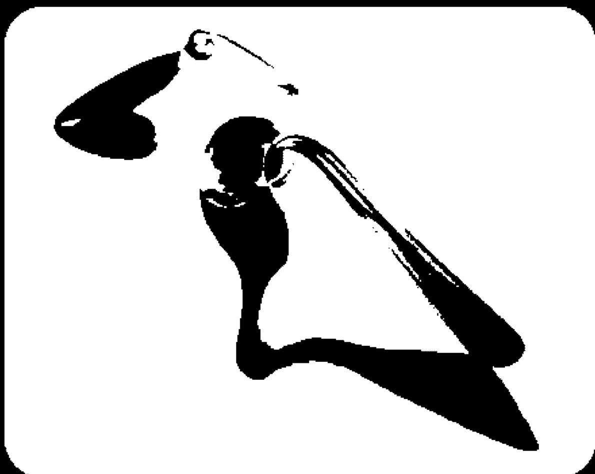
- Properties of metals
- Nuclear reactors: designed against embrittlement
- Pressurized thermal shock (PTS)
  - What is it?
  - How is safety ensured?
  - Specifics of Regulations
- Status at Palisades

Purpose

- Provide a perspective on vessel embrittlement and PTS, as regulated by the NRC for all PWRs
- Discuss embrittlement and PTS at the Palisades plant
- Answer questions from the public

Ductility & Embrittlement

Definitions



Ductile metals bend (absorb energy) when pulled upon

Embrittlement

Embrittlement (a loss of ductility) reduces how much a metal can bend before it breaks

NRC Regulations limit embrittlement to ensure safety

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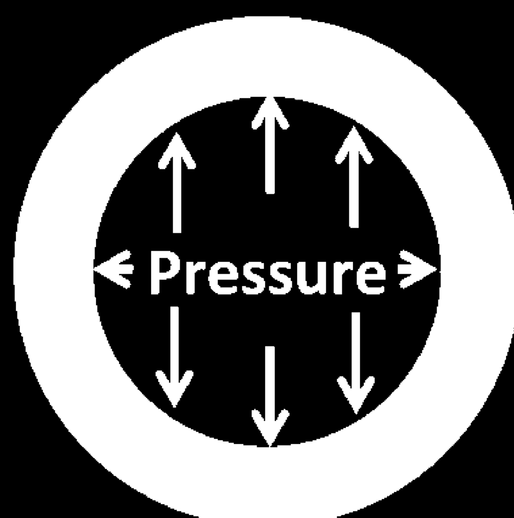
Reactor Pressure Vessels (RPV)
Designed Against Embrittlement

- The reactor coolant system produces forces on the RPV
  - Pressure
  - Thermal
- RPV designed to resist these forces, even after embrittlement
  - RPV steel has adequate toughness
  - Toughness measures ductility
- NRC screening criteria for embrittlement keeps the probability of fracture extremely low



<u>RPV</u>: Pressurized Cylinder

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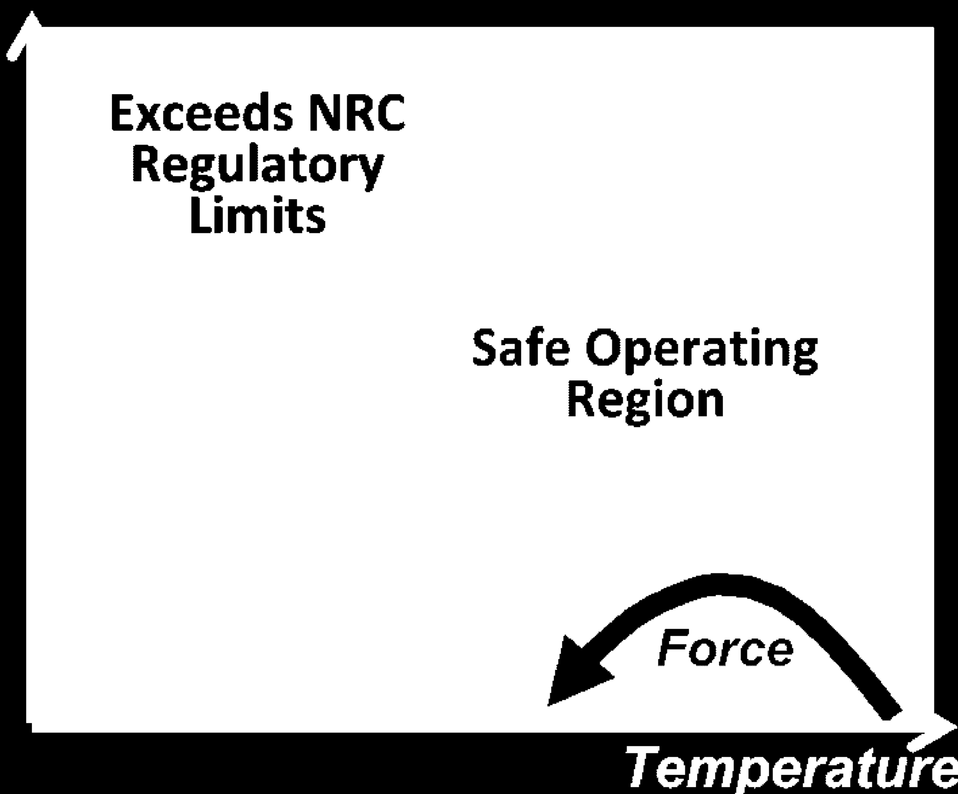
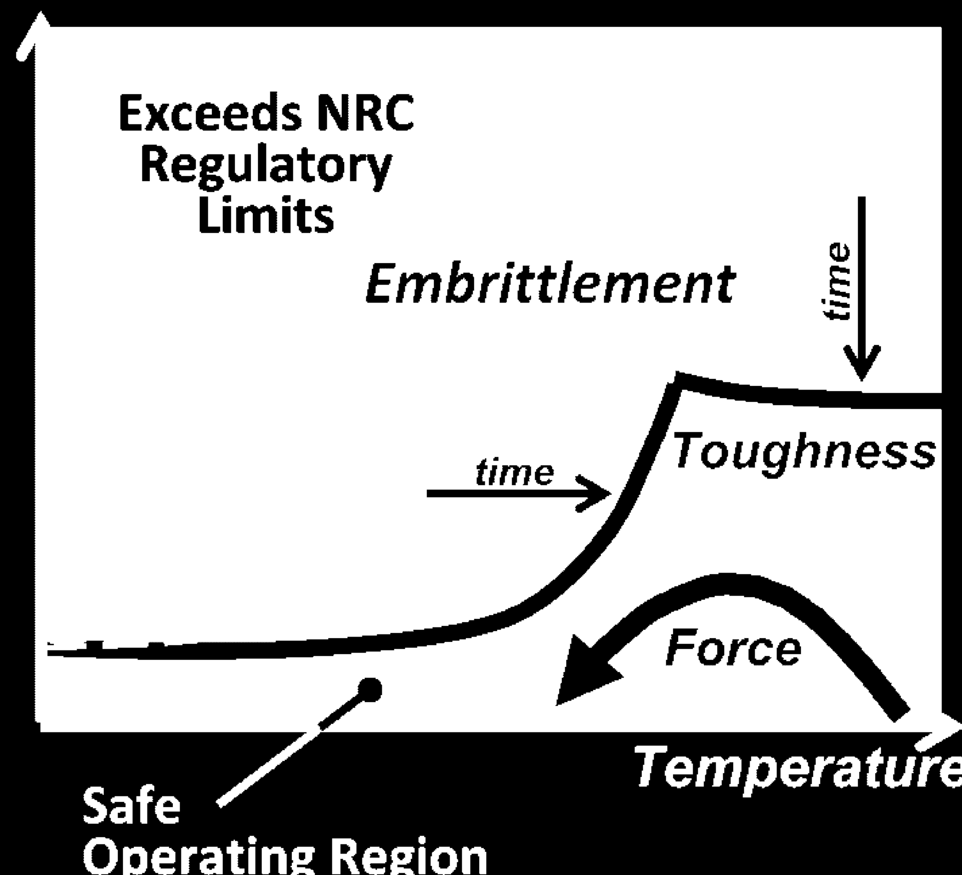
Toughness *vs.* Force

Toughness Always Greater

Before reactor operation
force < toughness



After operation (& embrittlement)
force (still) < toughness



Embrittlement is Measured

NRC Requires: 10 CFR 50 Appendix H

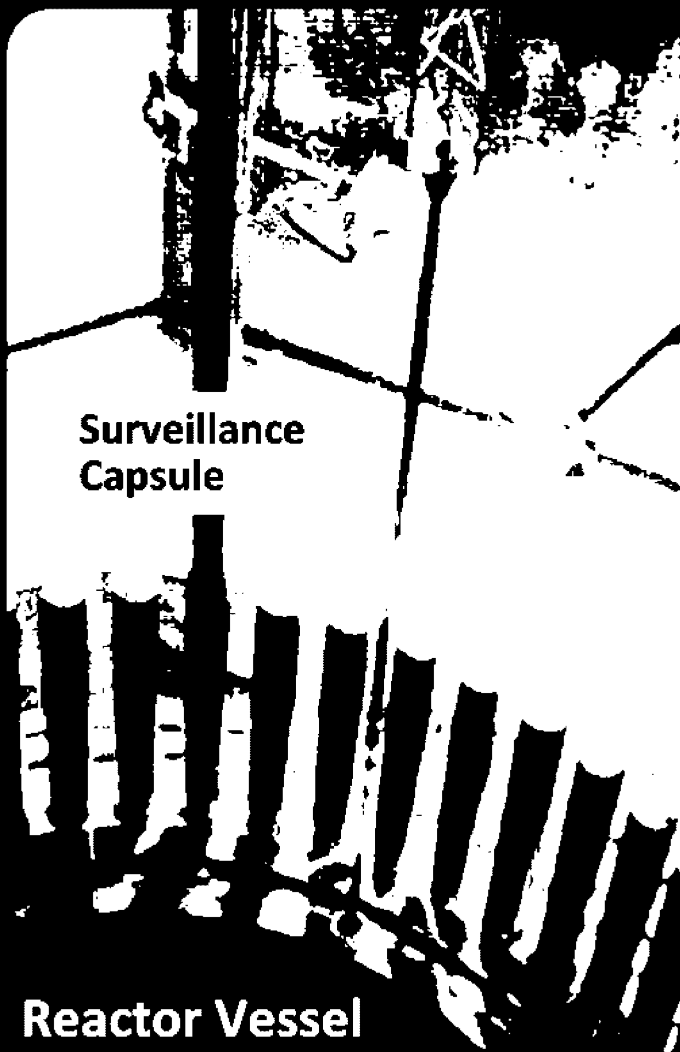
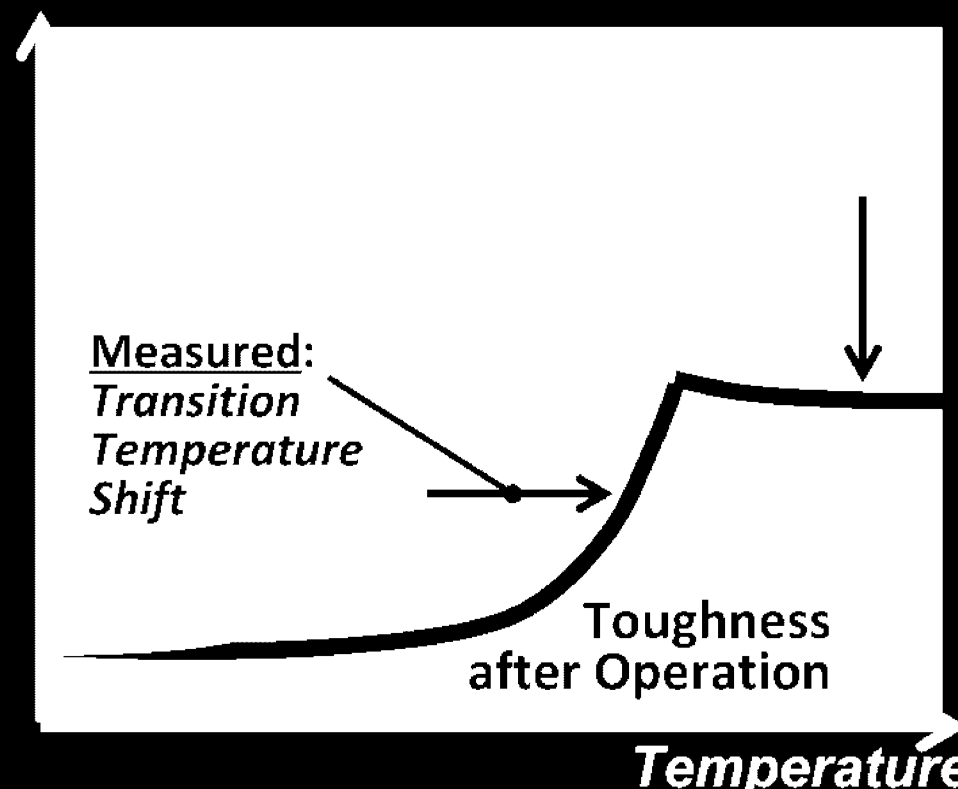


Photo courtesy of J. May, AREVA NP GmbH

Embrittlement monitored by surveillance programs & limited by regulations



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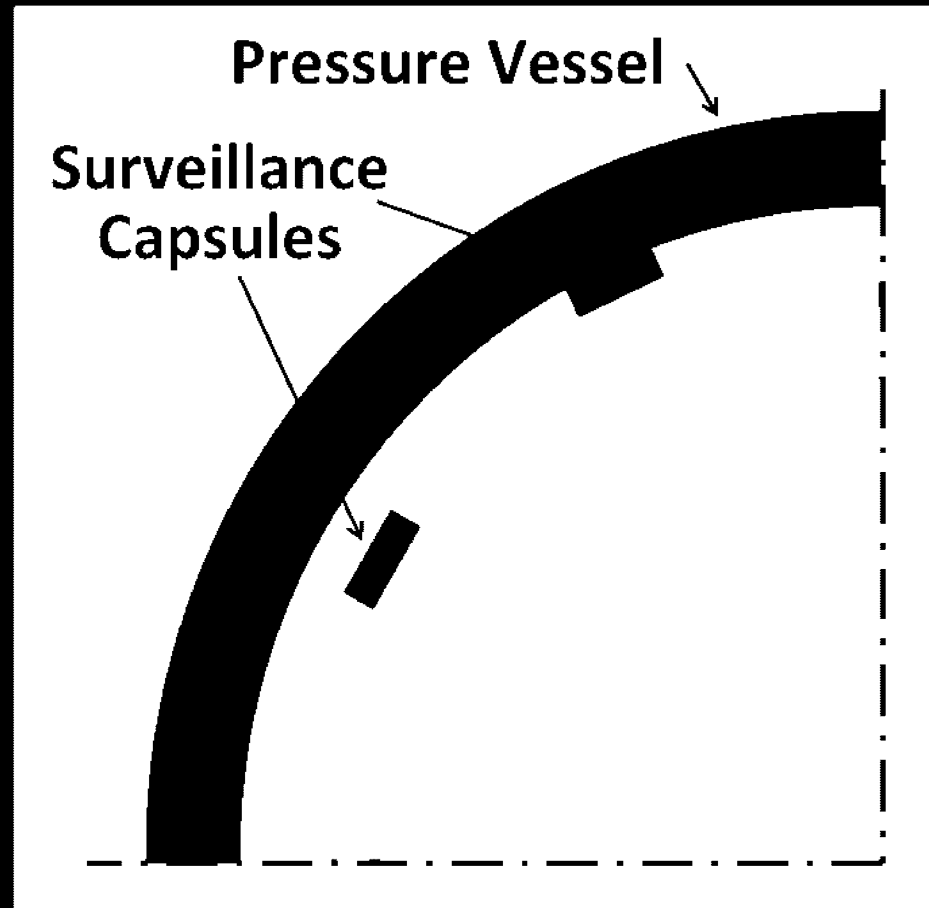
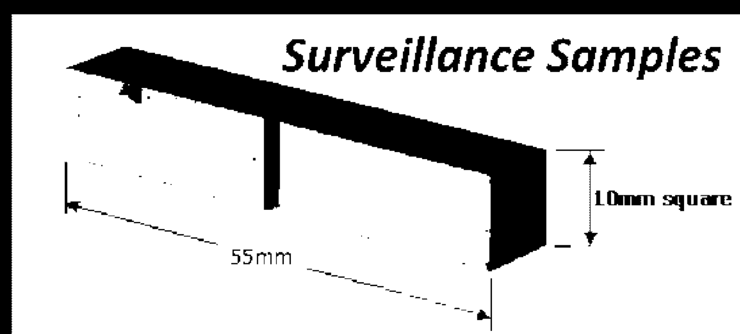
Palisades Surveillance Capsules

- **Purpose of surveillance**
  - Monitor embrittlement of specific steels in specific plants
  - Provides ***advance information*** on pressure vessel condition
  - Provides data for predictive models

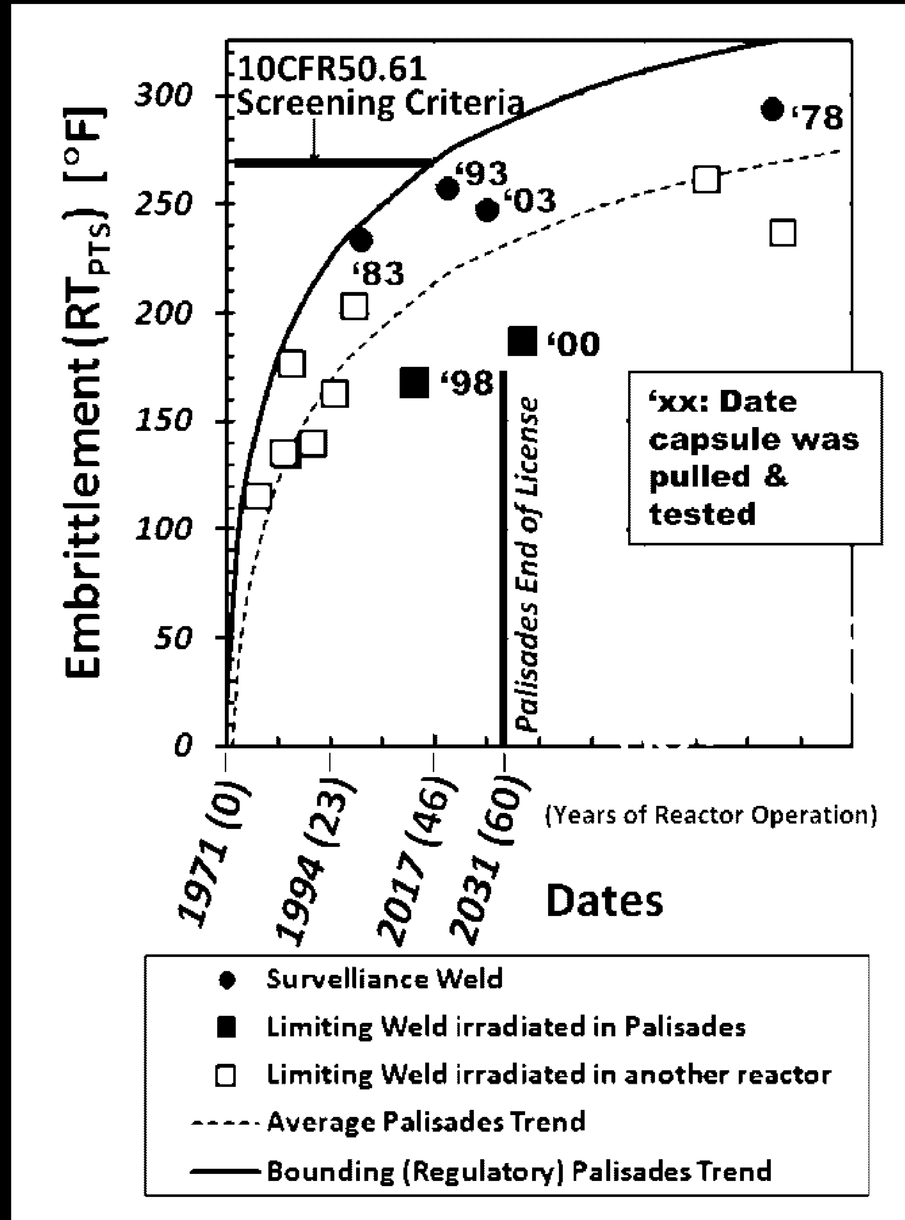
- **Data sources for Palisades**
  - Initial program
    - 8 irradiation
    - 2 thermal
  - Supplemental program
    - 2 irradiation
  - Other Plants
    - Indian Point 2&3
    - H.B. Robinson 2





Palisades Surveillance Data

- Palisades embrittlement known from 1st capsule pull in 1978
- Licensing dates lie well within data (no extrapolations)
- Data follows expected trends



Some Technical Details

- Program complies with ASTM-185(1966)
  - 10 CFR 50 Appendix H not in force when Palisades was licensed
  - Included high-Cu weld, but not exactly same as in Palisades pressure vessel
- Palisades supplemented surveillance program to get data on limiting weld
  - 2 capsules in Palisades
  - 8 capsules in other plants (HBR, IP)
- 4 capsules remain in Palisades. One more will be pulled before 2031.

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Brittle (or Embrittled) Materials
Used Safely

Nuclear RPVs

- Steel embrittles over time
- Embrittlement is
  - Understood
  - Measured
  - Limited
- NRC limits transition temperature shift so that toughness always exceeds force
  - Ensures safety

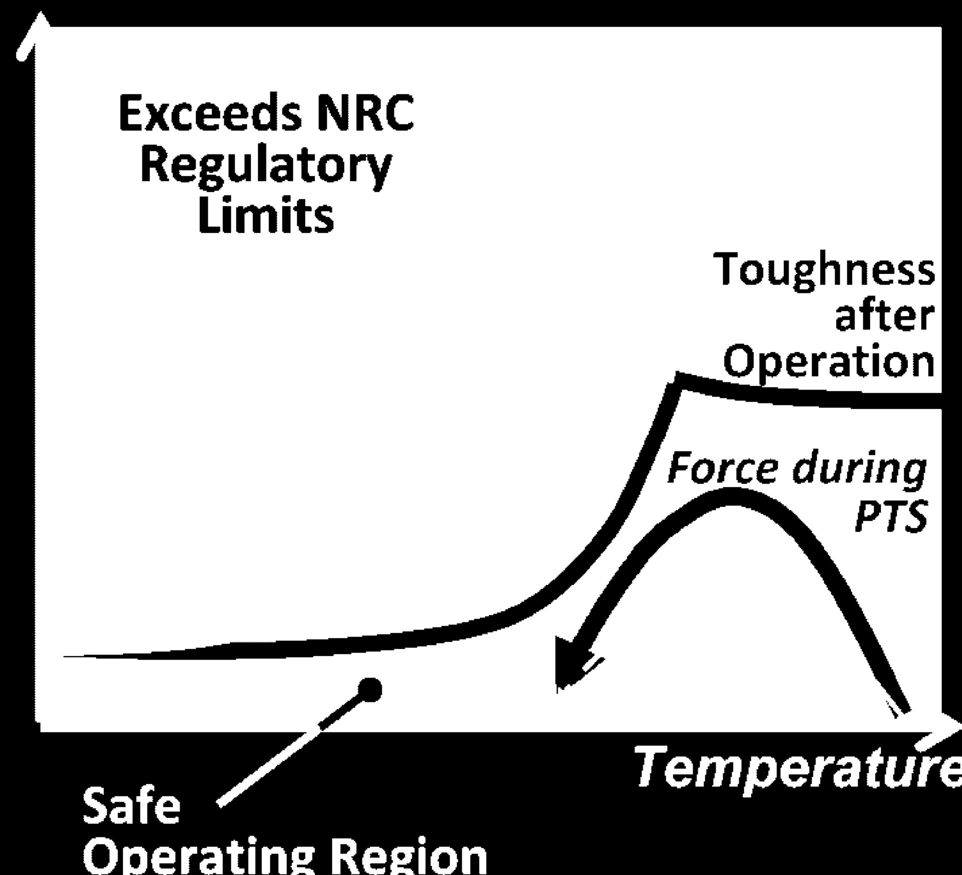
Non-Nuclear Example

- Aircraft landing gear
- Very high strength steels needed to resist landing forces
- High strength steels have lower toughness
  - Less ductile
  - More brittle
- Nuclear RPV toughness exceeds landing gear toughness (even after embrittlement)

Pressurized Thermal Shock
PTS

- **A rare event**
  - Designed against
  - Regulated for safety
- **More force applied to RPV during PTS**
  - Injection of cold water
  - Rapid cooling

Even during PTS
(& after embrittlement)
force (still) < toughness



NRC PTS Rules

10 CFR 50.61 (1984)

- Significant conservatisms restrict operations with no safety benefit
- Conservatisms include
  - Over-estimated force
  - Under-estimated toughness
- Conservatisms evidenced by
  - Additional toughness data
  - More realistic & thorough analyses
  - Scale model experiments that validate predictions

10 CFR 50.61a (2010)

- Considerations
  - Conservatisms in 50.61 limits will cause many plant -specific submittals
  - All submittals would address the same fundamental issues
- Alternative approaches
  - Many plant-specific assessments reviewed individually
  - Selected → **Comprehensive re-assessment of PTS risk performed proactively & with thorough review by technical experts**

Alternative PTS Rule

Rigorous Development Process

- Joint effort of NRC, national labs, universities, and industry (providing data & operating experience)
- Approximately 10 year project duration
- Many opportunities for public involvement
- Extensive expert technical reviews
  - Advisory Committee for Reactor Safeguards
  - Independent expert panel
- Full documentation available on NRC website

Alternative PTS Rule

Technical Approach & Insights

- Three analyses performed

Analysis		Purpose
PRA	Probabilistic Risk Assessment	Establish events that cause rapid cooling. Assess human factors.
TH	Thermal Hydraulics	Quantify force produced by rapid cooling
PFM	Probabilistic Fracture Mechanics	Quantify resistance to rapid cooling, accounting for embrittlement.

- Detailed assessments of three plants (Palisades, Beaver Valley, Oconee)
- Results generalized to all plants in USA
  - Only the most severe forces produce any risk
    - Similar across the fleet
  - Rapid cooldown to 200 °F below operating temperature needed to produce any risk
    - Operational controls limit the likelihood of such cooldowns occurring

Comparison of PTS Rules

10 CFR 50.61 & 10 CFR 50.61a

Aspect of Rule	10 CFR 50.61 REQUIRED	10 CFR 50.61a VOLUNTARY
Embrittlement Screening Criteria	More restrictive	Less restrictive
Plant-specific surveillance data check	Required: 1 test	Required: 3 tests
Plant specific inspection for flaws	Not required	Required

- 10 CFR 50.61a embrittlement limits are less restrictive than 10 CFR 50.61
  - Justification: More thorough, consistent, & realistic assessment
- 50.61a screening criteria can only be used when surveillance and inspection requirements are met
- Surveillance and inspection requirements ensure that key features of the 50.61a model apply to the plant being assessed
  a. The embrittlement of specific RPV materials
  b. The flaws in a specific RPV

Current Status at Palisades
Relative to PTS Limits

10 CFR 50.61

- One of the most embrittled plants in USA
- Palisades operates in compliance
- Embrittlement screening criteria will be exceeded in 2017
- Palisades must
  - Make safety case in 2014 (2017 minus 3 years) for continued operation, or
  - Shut down in 2017
- Options for operation beyond 2017
  - Annealing to reverse embrittlement,
  - Analysis and/or experiments to provide a plant specific safety justification, or
  - Use 10 CFR 50.61a

10 CFR 50.61a

- One option Palisades has to continue operation after 2017
- Would need to
  - Analyze data
  - Check embrittlement
  - Check flaws
- Generic analysis of US fleet in NUREG-1874 suggests this is a viable option for Palisades

Palisades Summary

- Palisades continues to operate safely
- The vessel at Palisades is one of the most embrittled in the USA
- Palisades continues to operate as long as it demonstrates compliance with NRC regulations
- There are several options by which continued safe operation of the Palisades vessel after 2017 could be demonstrated
  - The licensee decides what option to take

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What's New in Valves and Pumps in New Reactors?

Thomas G. Scarbrough
James M. Strnisha

Division of Engineering
Office of New Reactors

August 2013

Introduction

- Overview of pumps and valves for new reactors
- Lessons learned from operating experience
- ASME and industry activities
- New reactor pump and valve requirements
- Pump and valve qualification process
- Vendor inspection support
- NRC staff approach for evaluation of pumps and valves
- Regulatory Treatment of Non-Safety Systems (RTNSS)
- Small Modular Reactor (SMR) issues
- Close-out process for pump and valve functional qualification ITAAC
- Pump and valve inspections at Vogtle and Summer
- Future activities

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Pumps and Valves in New Reactors

- Centrifugal and positive displacement pumps
- Gate, globe, butterfly, and ball valves
- Swing check valves and nozzle check valves
- Power-operated valves including motor, pneumatic, hydraulic, solenoid, and pyrotechnic (squib) operators
- New design squib valves
- Safety and relief valves
- Manual valves in safety applications

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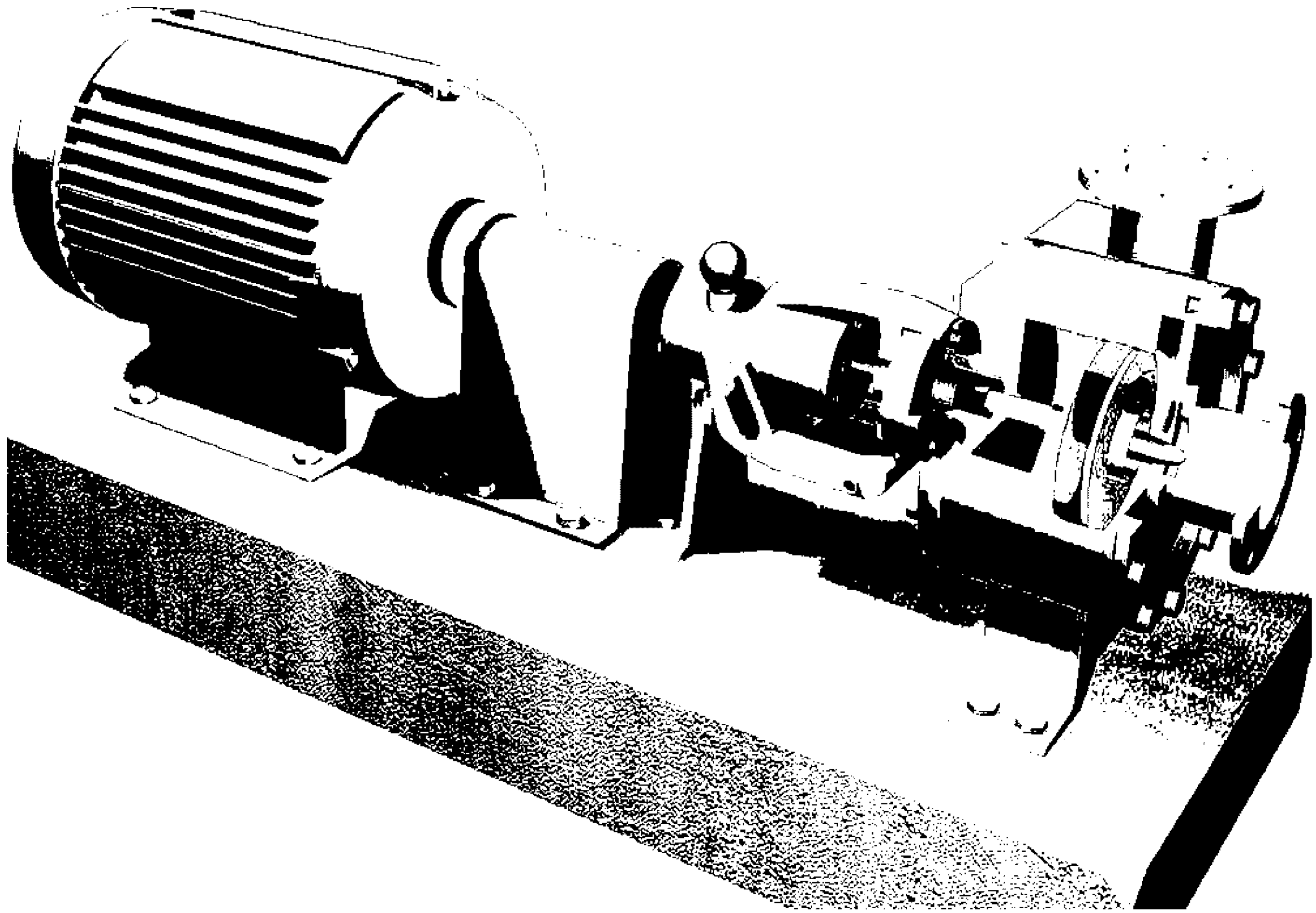
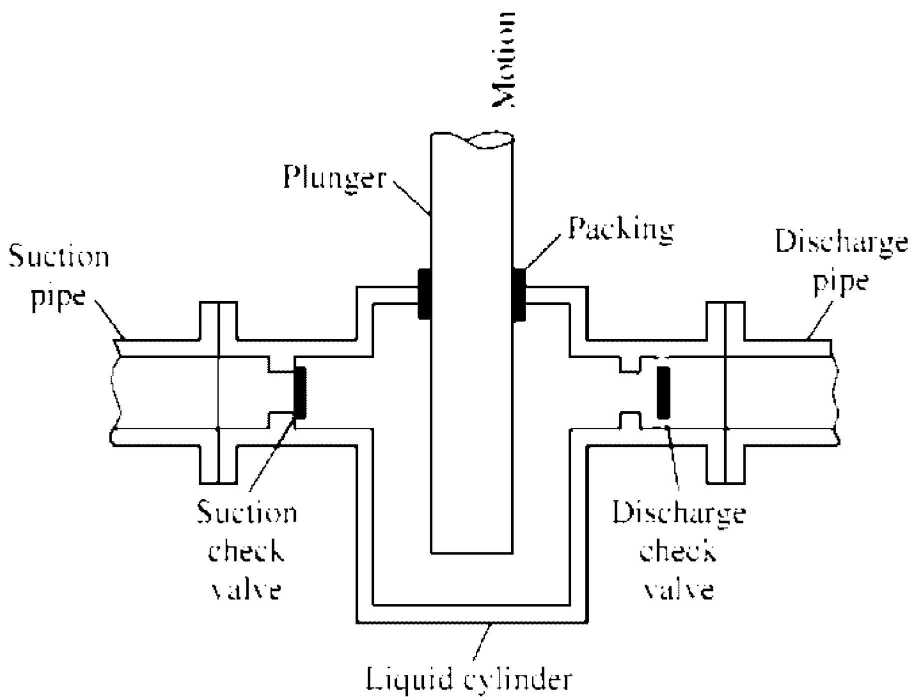
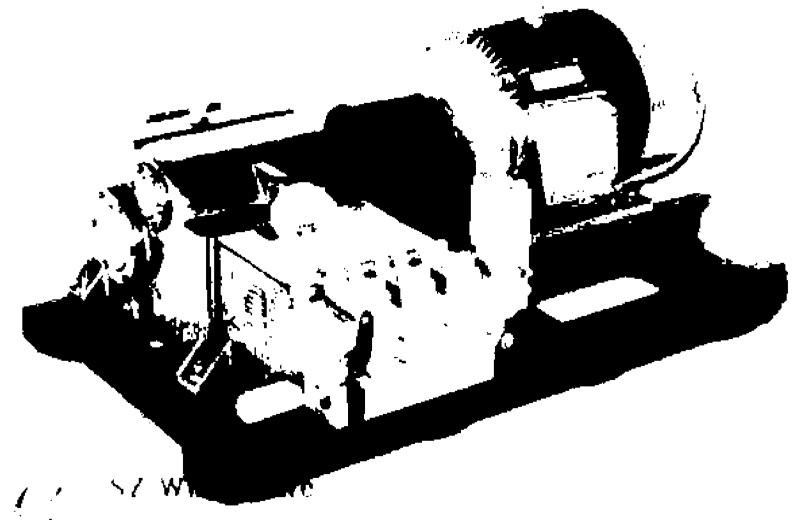


Centrifugal Pump (Wikipedia website)





Positive Displacement Pump
(http://www.bing.com/images/search)

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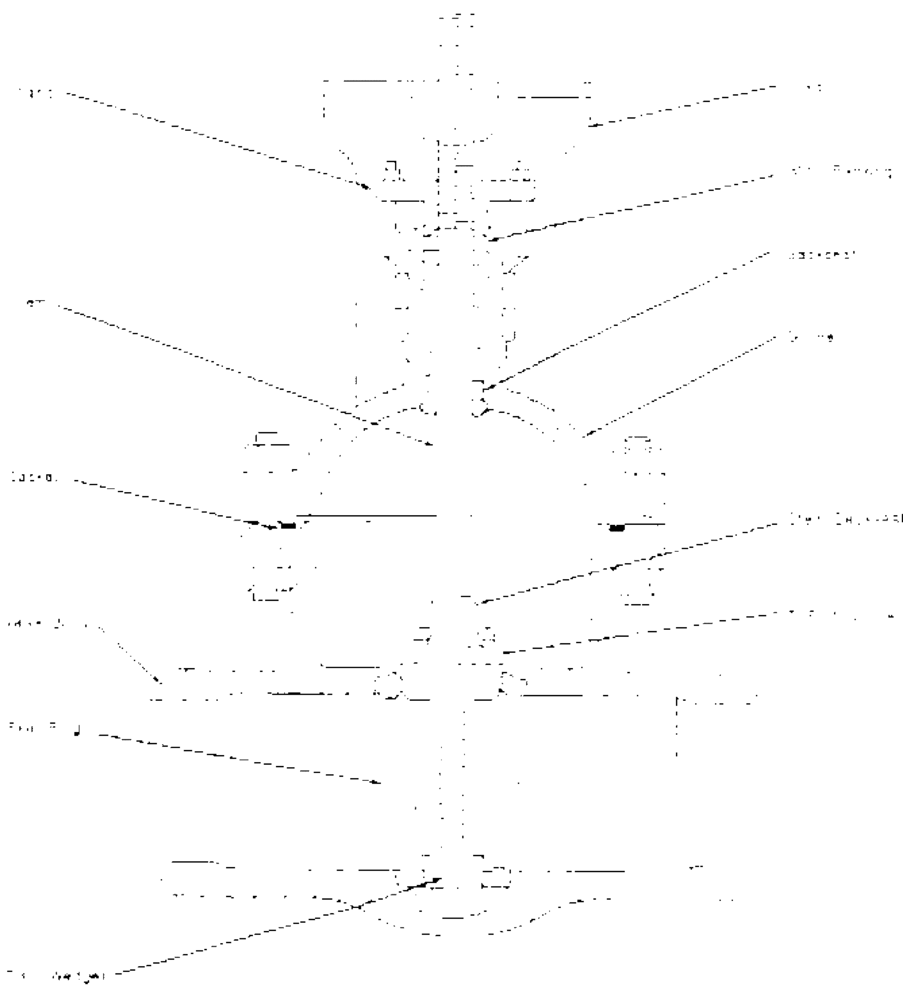
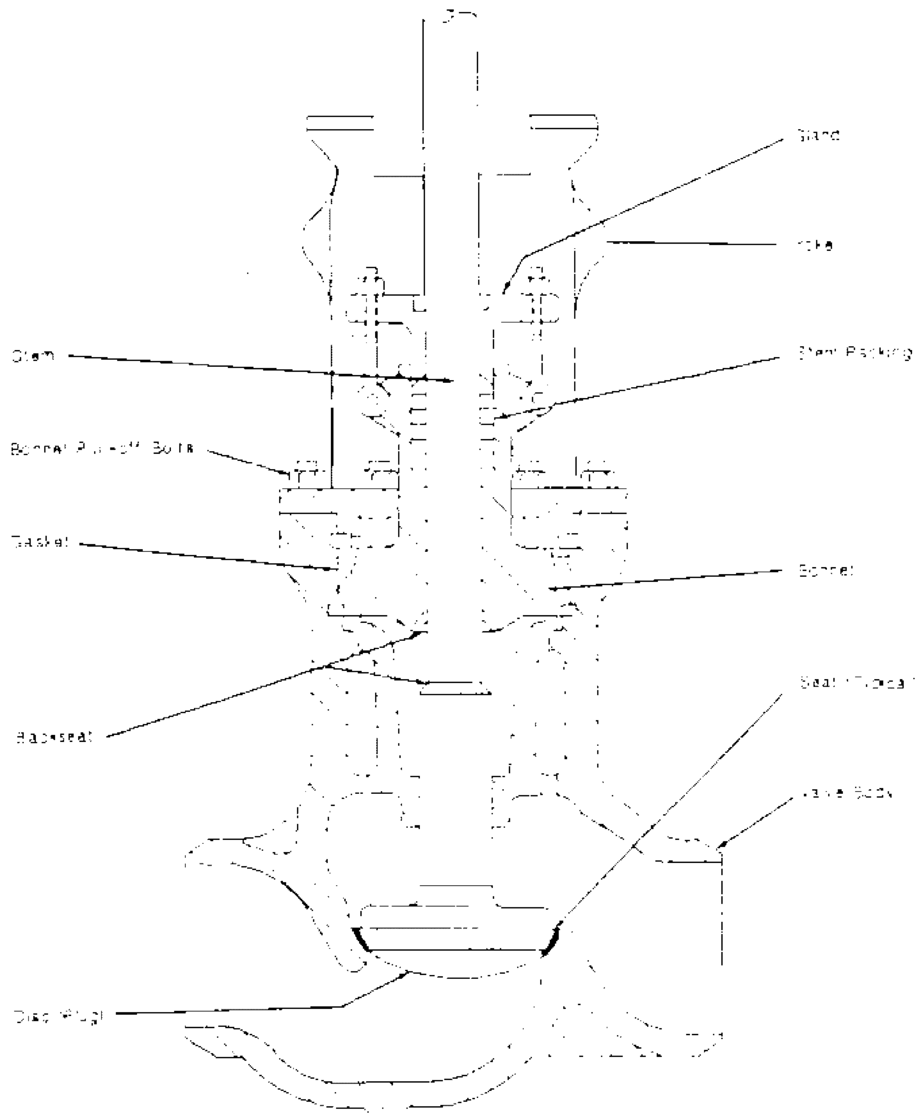
Gate Valve



Globe Valve


Gland
Yoke
Stem
Stem Packing
Bonnet Pull-off Bolts
Gasket
Bonnet
Seat (Typical)
Backseat
Valve Body
Disc (Plug)

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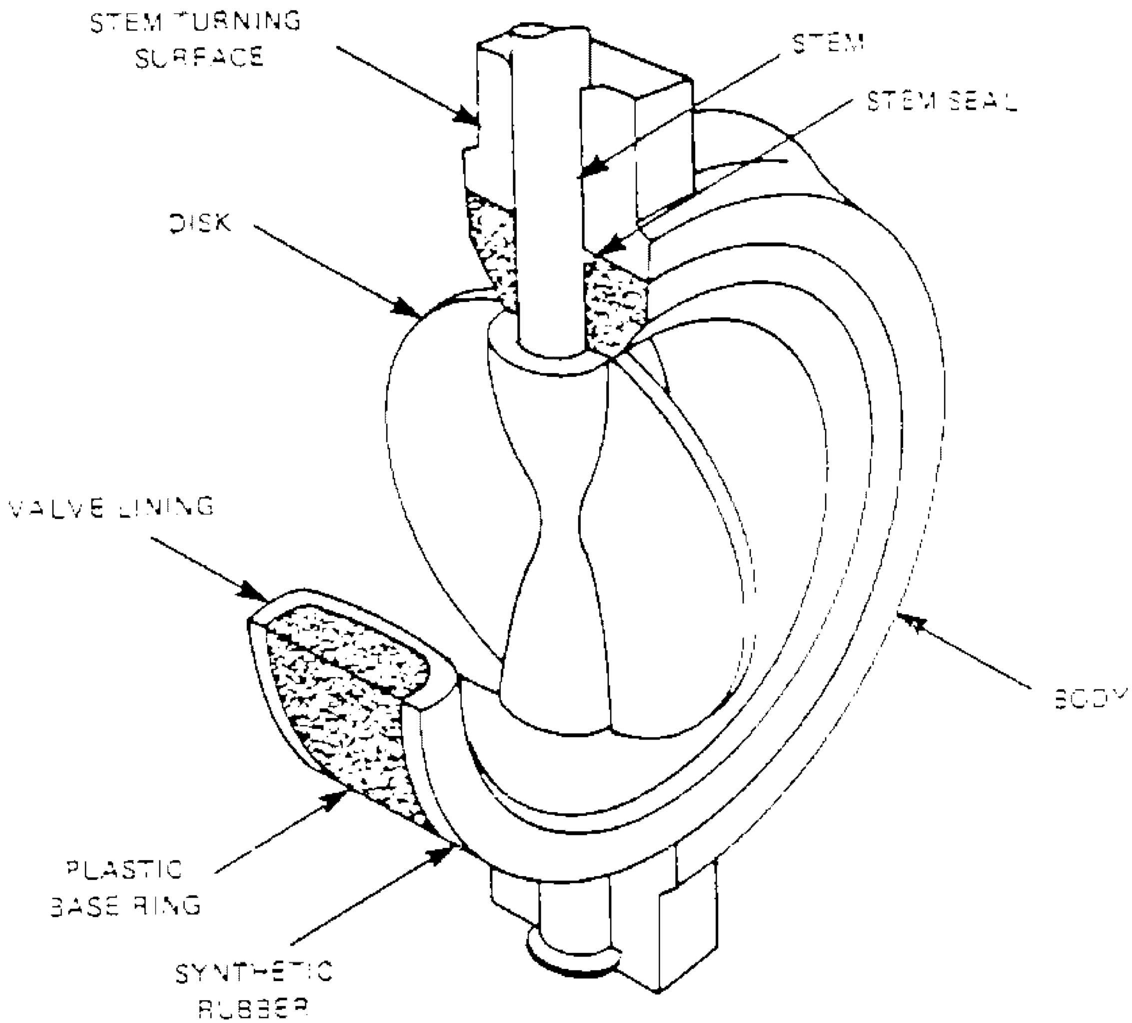
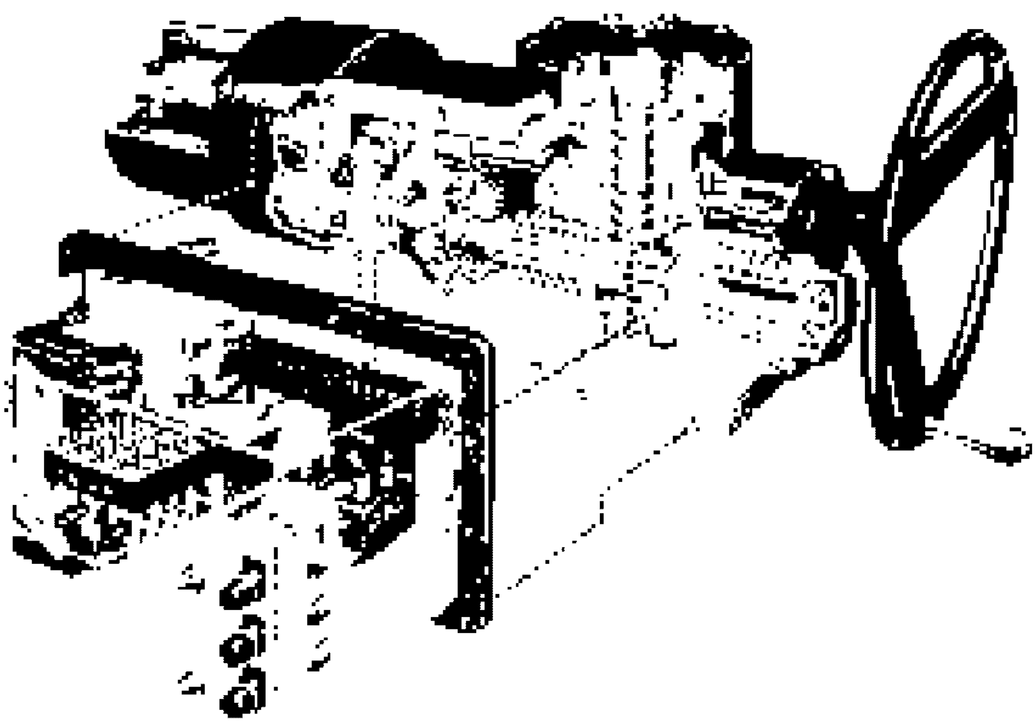
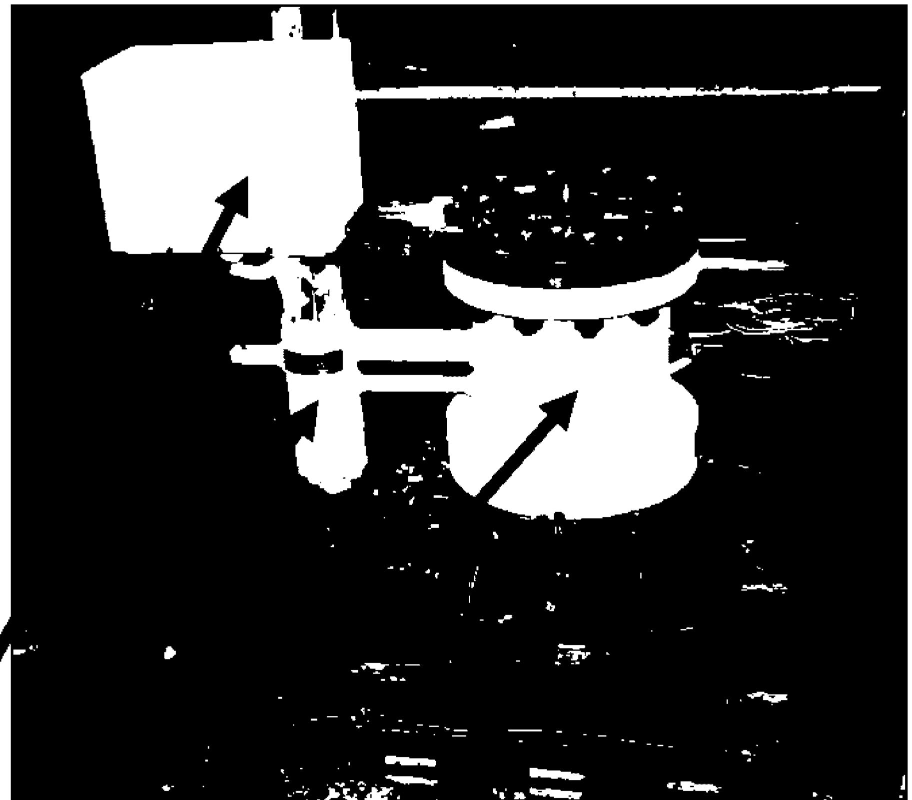
Symmetric Disc Butterfly Valve



Limitorque SMB-0





Actuator

Pilot Valve

ERV

Electromatic Relief Valve

AP1000 Passive Design

- AP1000 passive pressurized water reactor designed to provide reactor core cooling in response to LOCA without operator action or pump operation for 72 hours
- Passive Core Cooling System (PCCS) uses high pressure Core Makeup Tanks (CMTs) and Accumulators for initial core cooling, and dc-powered motor-operated valves and squib valves to reduce RCS pressure and allow gravity-driven cooling water flow
- After 72 hours, containment makeup water might be needed from external sources

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AP1000 PCCS

- PCCS has four passive injection sources:
  - 2 CMTs with borated water at RCS pressure
  - 2 Accumulators with borated water pressurized at 700 psig
  - In-Containment Refueling Water Storage Tank (IRWST) with borated water vented to atmosphere
  - Containment Sump allowing long-term recirculation
- PCCS uses 12 squib valves:
  - Four 14-inch Automatic Depressurization System (ADS) 4^{th} stage squib valves
  - Four 8-inch IRWST injection squib valves
  - Four 8-inch Containment Recirculation squib valves

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AP1000 Squib Valves

- SPX Copes-Vulcan is valve manufacturer
- AP1000 squib valves are larger and more complex than current BWR squib valves in standby liquid control system
- SPX Copes-Vulcan squib valve designs are proprietary

Squib Valve
Design and Qualification Issues

- Vendor inspections at Westinghouse, Copes Vulcan, and Wyle Laboratories for squib valve functional design and qualification process
- Several significant issues with squib valve explosive system
- Feb. 20, 2013, public meeting with Westinghouse to discuss need to implement systematic engineering design process sufficient to identify critical parameters of explosive system design and to establish acceptable tolerance ranges for each parameter

AP1000 Squib Valve Surveillance

- Vogtle/Summer FSARs specify that squib valve IST program will incorporate lessons learned from design and qualification process
- Vogtle/Summer COLs include license conditions for AP1000 squib valve surveillance
- License conditions specify preservice and inservice inspection and testing to verify external and internal component integrity, absence of degradation and foreign material, availability of electronic actuation circuitry, and explosive powder output capability

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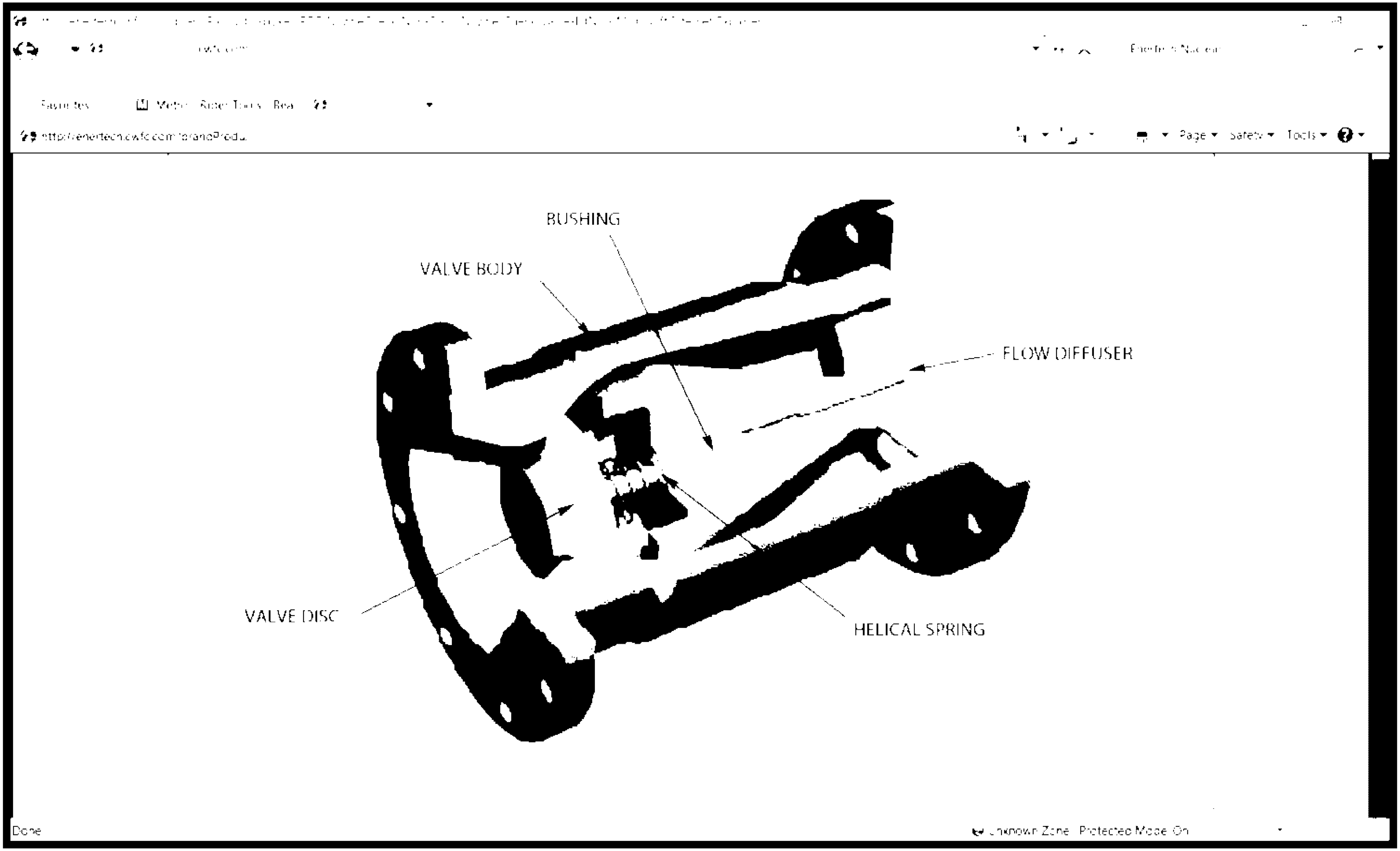
AP1000 Nozzle Check Valves

- AP1000 uses four 8-inch nozzle check valves in PCCS with open-close-open function in event of LOCA.
- AP1000 includes other nozzle check valves with standard open-close functions.
- Enertech is the valve manufacturer.
- Operating plants began using nozzle check valves in 1990s, but limited experience with open-close-open function.
- QME-1 qualification being performed at Enertech in CA and at Utah State University



Enertech website drawing

Pump Issues

- Mini-flow lines insufficient for pump testing
- Required Net Positive Suction Head (NPSHR) uncertainties
- GSI-191 LOCA Debris Issues
- AP1000 Reactor Coolant Pump impeller blade
- Pump Teflon Seal EQ

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Valve Issues

- ASME MOV IST stroke-time test inadequate
- Underprediction of thrust and torque requirements for original gate, globe and butterfly valves
- Unpredictable behavior of original gate valves under high flow conditions
- Overprediction of motor actuator output with loading, degraded voltage, temperature, and stem friction
- Valve stem and actuator lubrication issues
- Pressure locking and thermal binding of gate valves
- Valve stem/disc separation
- MOV dead band zone
- Flow induced vibration

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Regulatory Activities

- Extensive research program at Idaho National Laboratory on valve performance
- 10 CFR 50.55a revised to supplement ASME Code for MOV periodic design-basis capability
- Bulletin 85-03 and Generic Letters 89-10, 95-07, and 96-05
- Regulatory Issue Summaries 2000-03 and 2001-15
- Numerous Information Notices
- Reviews and inspections of MOV programs at current nuclear power plants
- SRP and inspection procedures updated

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ASME Activities

- ASME Standard QME-1-2007 incorporates valve lessons learned with pump improvements being considered
- Subsection ISTF in ASME OM Code (2011 Addenda) for new reactors specifies comprehensive pump testing
- Preservice and inservice testing provisions for squib valves in new reactors specified in Subsection ISTC of ASME OM Code (2012 Edition)
- ASME task group preparing guidance for treatment of RTNSS pumps and valves
- ASME OM Code evaluating SMR pump and valve surveillance issues

Industry Activities

- Electric Power Research Institute developed test-based valve performance methodology
- Joint Owners Group (JOG) developed MOV dynamic testing program in response to GL 96-05 (Guidance in RIS 2011-13 for JOG Class D valves)
- ComEd White Paper 125 (Rev. 3, 2/8/99) provides methodology for sizing motor actuators
- BWROG developed updated methodology for DC MOV output and stroke time

Design Certification Application Requirements

- 10 CFR 52.47(a)(9) requires design certification applications to evaluate design against NRC Standard Review Plan in effect 6 months before docket date
- 10 CFR 52.47(a)(22) requires design certification applications to address operating experience

COL Application Requirements

- 10 CFR 52.79(a)(11) requires COL applicant to provide description of programs and their implementation necessary to ensure that systems and components meet ASME BPV Code and OM Code per 10 CFR 50.55a
- 10 CFR 52.79(a)(37) requires COL applications to include information necessary to demonstrate how operating experience has been incorporated into plant design
- 10 CFR 50.55a(f)(4)(i) requires initial IST program to meet ASME Code incorporated in 10 CFR 50.55a 12 months before fuel loading
- Guidance in RIS 2012-08, Rev. 1, “Developing Inservice Testing And Inservice Inspection Programs Under 10 CFR Part 52”

Pump and Valve Qualification Process

- ASME Standard QME-1-2007 provides specific criteria for qualification process for valve assemblies, extrapolation of qualification to other valve assemblies, testing of production valve assemblies, and post-installation testing
- QME-1 specifies requirements for Qualification Plan, Functional Qualification Report, and Application Report
- NRC accepted QME-1-2007 in Revision 3 to RG 1.100 with staff positions
- New reactor vendors requiring use of QME-1-2007 in design specifications

Vendor Inspection Support

- CIB is providing support for vendor inspections of pumps and valves for new reactors
- Squib valve inspections at Westinghouse, Copes-Vulcan, and Wyle Laboratories
- Check valve inspections at Enertech and Utah State
- Relief valve inspections at Pentair Valves
- Motor-operated valve review during Wyle inspection
- Limitorque MOV actuator inspection
- AP1000 RHR Pumps at Flowserve (post-inspection)

NRC Staff Evaluation of Pump and Valve Design and Qualification

- CIB verifies Design Certification application specifies ASME QME-1-2007 as accepted in RG 1.100 (Rev. 3)
- CIB audits design/procurement specifications in support of design certification and COL application review to confirm use of QME-1-2007
- CIB supports vendor inspections to verify that QME-1-2007 design/procurement specification requirements applied to qualification and testing procedures
- CIB will support operational program and ITAAC inspections to address pump and valve qualification

Regulatory Treatment of Non-Safety Systems (RTNSS)

- SECY-95-132 specifies policy and technical issues associated with RTNSS in passive plant designs
- Passive plants rely on active systems to avoid use of passive systems and to provide backup for passive features
- SECY-95-132 states that RTNSS systems do not need to meet safety -related criteria, but staff will expect a high level of confidence that active systems are available
- SECY-95-132 states that specific positions on IST requirements for RTNSS components will be determined as part of staff's review of plant-specific implementation
- No safety-related pumps in AP1000, but specific pumps are within the scope of RTNSS program

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AP1000 RTNSS Systems

- Instrumentation Systems
  - DAS ATWS (Diverse Actuation System)
  - DAS ESF
- Plant Systems
  - RNS (Normal Residual Heat Removal System)
  - CCS (Component Cooling Water System)
  - SWS (Service Water System)
  - PCS Water Makeup (Passive Containment Cooling System)
  - MCR Cooling
  - I&C Room Cooling
  - Hydrogen Igniters
- Electrical Power Systems
  - AC Power Supplies
  - Non Class 1E DC and UPS (Uninterruptible Power Supply) System (EDS)

AP1000 DCD RTNSS Provisions

- DCD Tier 2, Section 3.2 provides general technical provisions for Class D SSCs (RTNSS), such as use of example industry standards
- Augmented QA provisions for RTNSS equipment described in DCD Tier 2, Section 17.3
- Each RTNSS component needs to be evaluated on a case-by-case basis for the technical standards and methods used to demonstrate its capability to perform the intended functions
- Specific ITAAC need to be satisfied for RTNSS equipment (such as RNS pumps)

Small Modular Reactor (SMR) Pump and Valve Issues

- Some passive SMR designs may include active systems as part of RTNSS program
- Some SMR designs have operating cycles longer than 2 years which affects relief valve testing frequency provisions and motor-operated valve lubrication basis
- Some SMR designs might use new valve combinations to minimize LOCA conditions
- Environmental qualification of valves and pumps might involve high temperature and radiation

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Close-out Process for Pump and Valve ITAAC

- AP1000 DCD includes ITAAC for seismic, environmental, hydrostatic, and functional qualification of specific valves
- Licensee must complete ASME Boiler & Pressure Vessel Code and ASME Standard QME-1-2007 to support qualification testing and analysis
- Licensee will notify NRC staff of ITAAC completion
- NRC staff has identified targeted ITAAC for detailed verification of completion
- NRO projects and technical staff will need to work together to ensure that ITAAC are adequately closed

Pump and Valve Inspections at Vogtle and Summer

- Inspection Procedure IP 73758 describes evaluation of the functional design and qualification, preservice testing, and inservice testing of pumps, valves, and dynamic restraints in new reactors
- CIB will assist Region in evaluating implementation of DCD and FSAR provisions for functional design and qualification, PST, and IST at Vogtle and Summer
- NRC staff will discuss plans for pump and valve inspections with Vogtle and Summer COL licensees when IST program developed

Future Activities

- Support vendor inspections of pumps and valves
- Interact with Westinghouse to address squib valve issues
- Participate in ASME OM Code activities (including RTNSS treatment guidance)
- Interact with Vogtle and Summer COL licensees on IST operational program development
- Assist Region II on operational program and ITAAC inspections
- Work with NRO Projects to evaluate ITAAC closure notifications for pumps and valves

PWR STARTUP AND POWER OPERATION

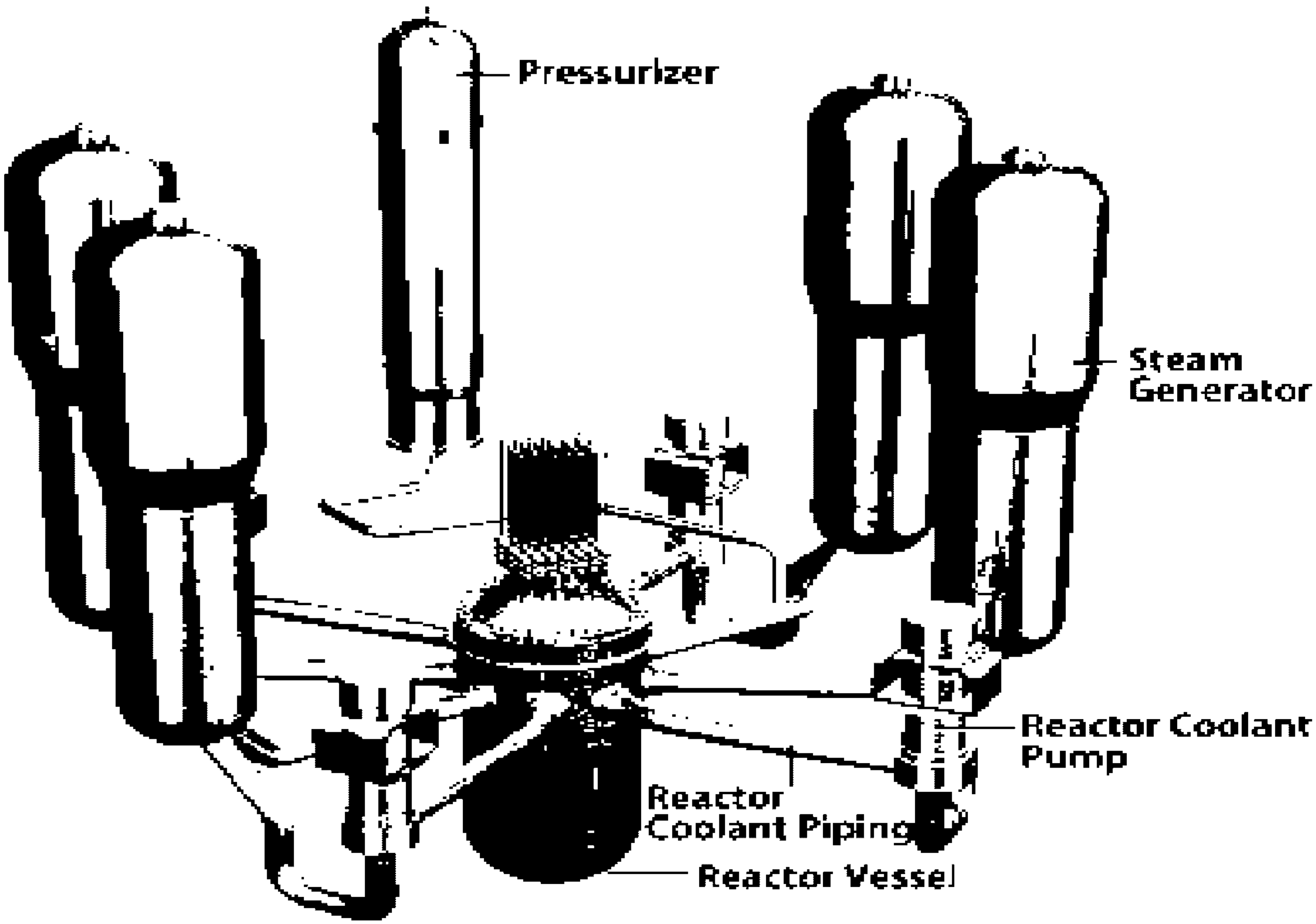
Background and Disclaimers

- Background and area of focus is predominately primary side operation
- My operational experience includes reactor startup and operation in support of licensed RO and SRO’s. Plant Certified at Palo Verde NPGS
- Will not provide detailed Technical Specifications LCOs as these can be plant specific
- Know little about secondary side chemistry and generator

Objective

Explain the basic steps of starting up a PWR reactor from refueling to full power

Typical 4-Loop PWR Primary Side



Typical PWR Modes of Operation

Mode	K_{eff}	Power	T_{cold}
1 - Power Operation	Greater than or equal to 0.99	Greater than 5% RTP	N/A
2 - Startup	Greater than or equal to 0.99	Less than or equal to 5% RTP	N/A
3 – Hot Standby	Less than 0.99	N/A	Greater or equal to 350 ^{o}F
4 – Hot Shutdown	Less than 0.99	N/A	200 ^{o}F to less than 350 ^{o}F
5 – Cold Shutdown	Less than 0.99	N/A	Less than or equal to 200 ^{o}F
6 - Refueling	N/A	N/A	N/A

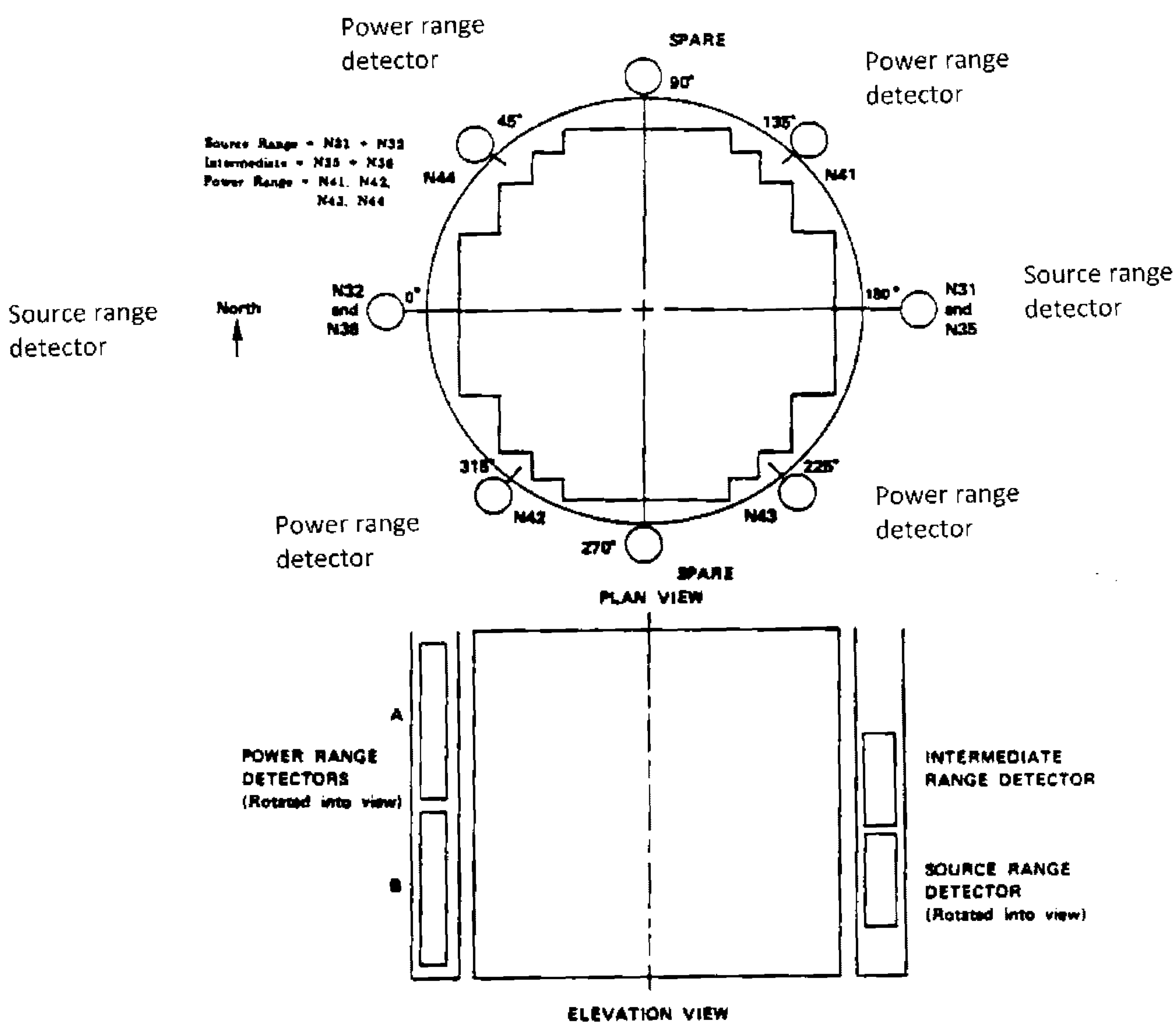
Mode 6 - Refueling

- Fuel is in the reactor vessel
- If all fuel is off loaded to the spent fuel pool Operational Mode is defined as N/A
- Reactor head either removed or not fully tensioned
- Residual Heat Removal (RHR) or Shutdown Cooling (SDC) are in service removing decay heat
- Reactor coolant is highly borated to provide TS minimum shutdown margin
- Source range detectors monitor neutron count rates

Mode 6 – Refueling (cont)

- Fuel is typically loaded nearest the source range detectors first and then work outward
- Refueling pattern is tightly controlled to ensure
  - Assemblies are placed in the correct locations as determined by the core designer
  - Two or more separate "cores" are not formed which may lead to a local criticality

Mode 6 - Refueling (cont)



Mode 6 – Refueling (cont)

- After fuel load core upper internals are reinstalled
- Control Rod drives attached but de-energized
- Reactor head is installed and Mode 5 is entered upon last head bolt being fully tensioned
- "Typical" refueling time 2-3 days

Mode 5 – Cold Shutdown

- T_{cold} is less than ~ 200 ^{o}F , Pressure ~300 psia
- RCS is filled and vented using highly borated water from the CVCS system
- Pressurizer level above heater cutoff setpoint but below nominal value (typically ~30%)
- Steam bubble in pressurizer is formed using pressurizer heaters
- Pressure controlled by heaters and auxiliary spray from CVCS system
- Dilution water sources secured or continuously monitored
- LTOPS in service to prevent over pressurization of the RCS

Mode 5 – Cold Shutdown (cont)

- Primary side heat up to Mode 4 – Hot Shutdown
  - Start one and then a second RCP to heatup the RCS. Typical RCP output is 4-7 MWs each
  - Control heatup rate using RHR (SDC) heat exchanger flow rate
  - No change in boron concentration and control rods still fully inserted
  - Enter Mode 4 (Hot Shutdown) on RHR or SDC

Mode 5 – Cold Shutdown (cont)

- Secondary side heat up to Mode 4 – Hot Shutdown
  - Done in parallel with Mode 5 primary side actions
  - Start a condensate pump to clean up secondary side. Sometimes called long path recirculation.
  - Condensate flow is heated using auxiliary heaters
  - Warm up main feedwater (MFW) pump by using condensate flow through the MFW pump
  - Condenser vacuum is drawn by starting air removal pumps. Normally takes ~ 2hrs.

Mode 4 – Hot Shutdown

- Primary purpose is to transition from RHR (SDC) to steam generator (S/G) heat removal
- Auxiliary feedwater pump and steam generator blowdown lines used to maintain S/G level (~30-50% normal level)
- Turbine or Steam bypass put in service to control RCS temperature and heat up rate
- RHR or SDC heat exchanger bypass reduced; RCS heats up and turbine bypass system controls RCS temperature

Mode 4 – Hot Shutdown (cont)

- Start third RCP to establish a heatup rate
- LTOPS valve are isolated at a given RCS temperature (~300 ^{o}F); pressurizer safeties provide RCS over pressure protection
- Depending on SDM requirements some plants may dilute RCS with CVCS system to reduce startup time

Mode 3 – Hot Standby

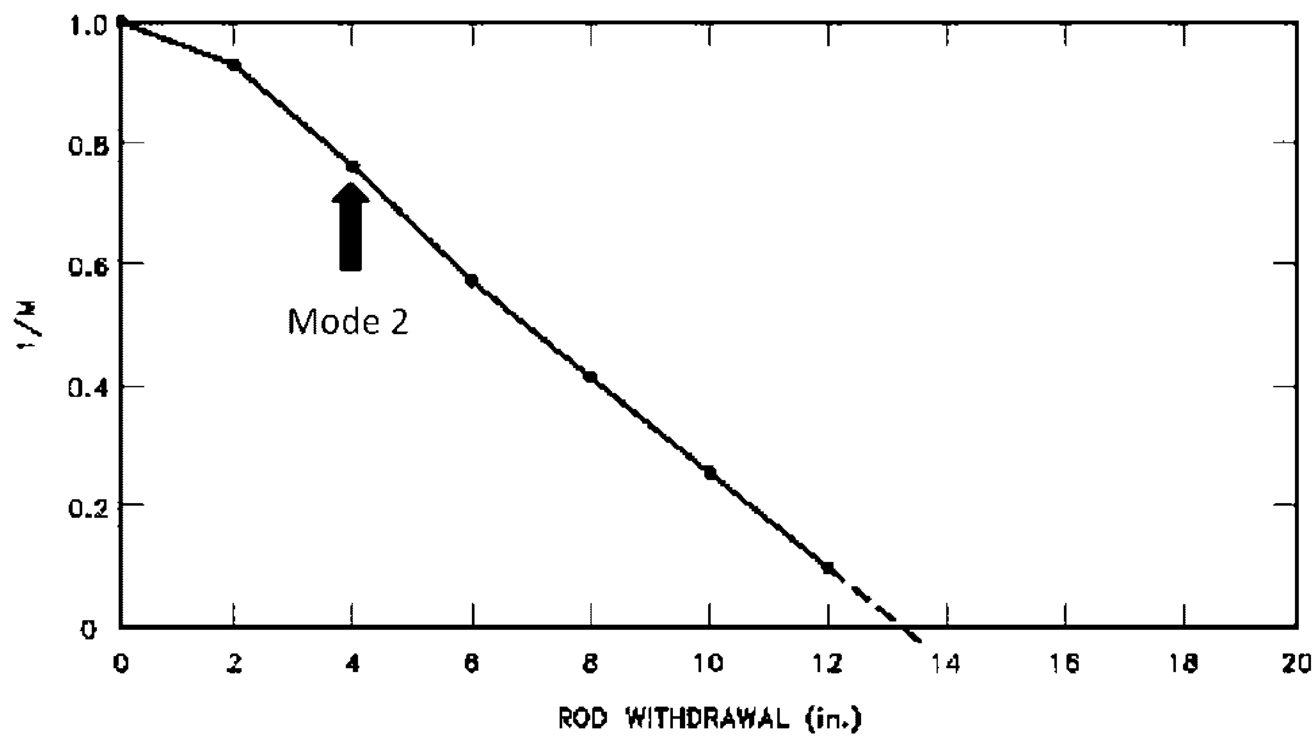
- Purpose to heat RCS up to normal operating pressure and temperature (~550 ^{o}F and 2250 psia)
- Fourth RCP is started to provide additional heat input
- Turbine (steam) bypass controls RCS temperature
- Pressurizer level controlled by CVCS letdown and charging rates. Now at nominal value typically ~50%
- Pressure controlled by RCS temperature and normal pressurizer spray
- Control rod drives energized and rod drop testing occurs
- Dilute to estimated critical boron concentration if starting up using control rods
- Usually most of the startup time is spent in Mode 3

Mode 2 – Startup

- Purpose to achieve a self sustaining chain reaction, k_{eff} =**1**, and startup physics tests
- K_{eff} = Number of neutrons in generation N/Number of neutrons in generation N-1, **including delayed neutrons**
- Two methods of going critical
  - Withdrawal almost all control rods and dilute RCS using CVCS system
  - Withdrawal control rods holding boron concentration constant
- All Shutdown rod banks withdrawn then regulating or control rod banks
- Core k_{eff} monitored using source range detectors by a 1/M plot and startup rate meter (SUR)
- Mode 2 is inferred based on data provided by core design

Mode 2 (cont)

M = Count Rate present/Count rate initial



Mode 2 (cont)

- Measured critical position compared to predictions
- Criticality shall not be achieved with control rods below power dependent insertion limits (PDILs)
- Criticality declared when a **sustained**, positive startup rate (i.e., count rates increasing) is observed with no reactivity insertion
  - In other words, criticality is declared after the reactor is slightly supercritical
  - Operators insert negative reactivity to achieve k_{eff} equal to one

Mode 2 (cont)

- HZP is $k_{eff} = 1$ and plant at standard operating temperature and pressure
  - HZP ~ $1X10^{-5}$% power, neutron flux ~$(10)^4$ n/cm^2-sec
- k_{eff} has **noting** to do with core power under **steady state** conditions; core power is proportional to neutron flux level
- Control rods are withdrawn to create a positive SUR
- SUR of less than one decade per minute

Mode 2 (cont)

- Point of adding heat (PAH) is found when reactivity feedbacks (negative ITC) limits power increase
- GDC 11 "…in the power operation range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity."
- New equilibrium power (neutron flux) is reached – additional positive reactivity must be added
- Additional control rod withdrawals add positive reactivity overcoming temperature defects and Xenon buildup

Mode 2 (cont)

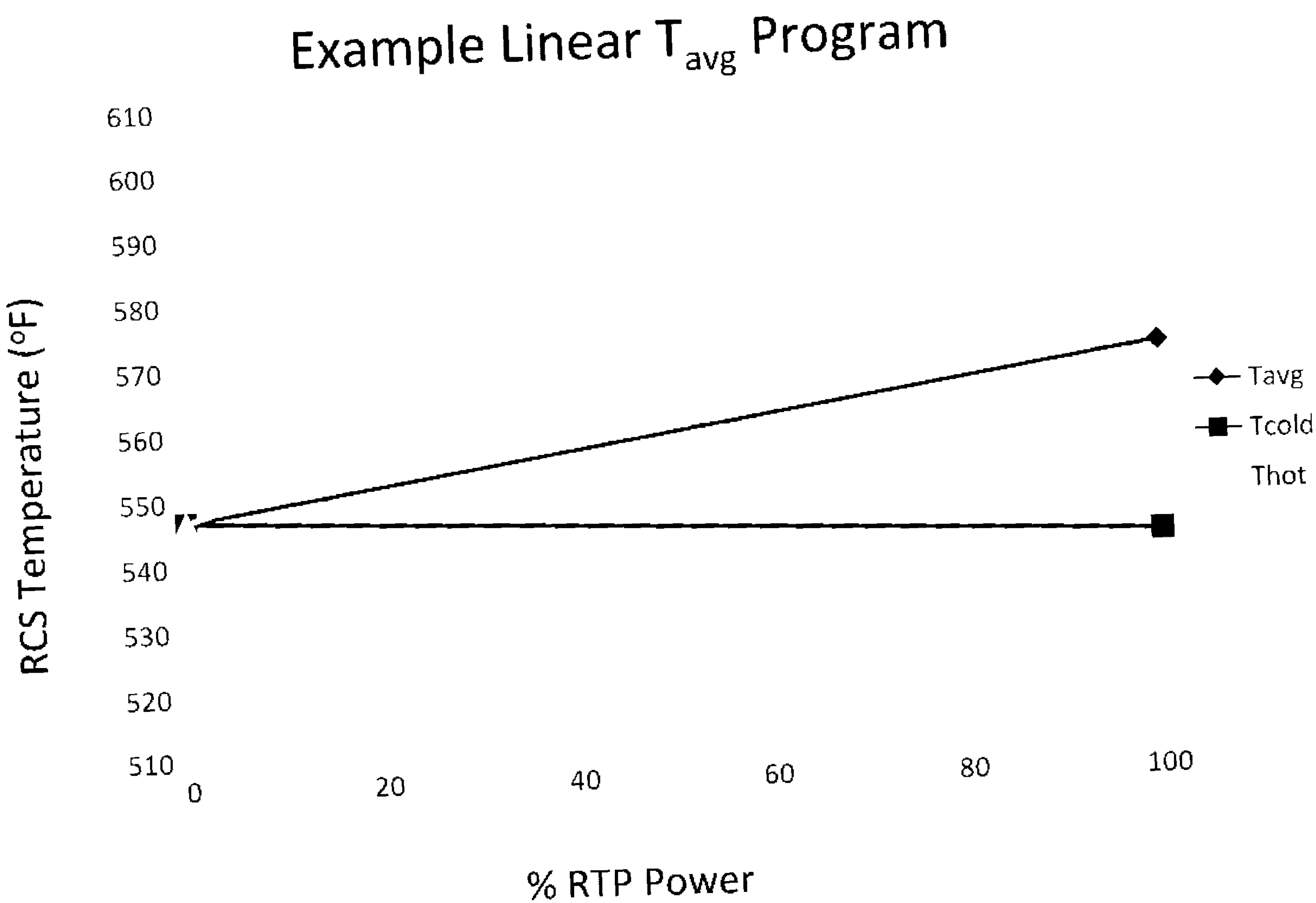
- Main feedwater pump brought online around 2% RTP (System 80 number)
- Mode 2 takes ~2-3 days including physics testing
- Mode 1 declared when core power is greater than 5% RTP

Mode 1 – Power > 5% RTP

Following is for typical System 80 plants

- Power increased to ~12% RTP using control rod withdrawals
- Turbine-Generator is put online ~12% RTP
  - Turbine bypass values close as turbine control valves open
  - Core power is now governed by steam demand
- Power is increased to ~20% RTP by increasing turbine load and withdrawing control rods to stay on the T_{avg} program
- Reactivity changes now affect reactor core temperature **not** reactor power (assuming quasi, steady-state operation)
- At 20% RTP incore flux maps are performed to check for mis-loaded or mis-manufactured fuel assemblies, or problems with reactor physics models
- Boron dilution, using the CVCS system, maintains the program average temperature as power increases

Mode 1 (cont)


Example Linear T_{avg} Program
610
600
590
580
570
560
550
540
530
520
510
RCS Temperature (°F)
0
20
40
60
80
100
% RTP Power
Tavg
Tcold
Thot

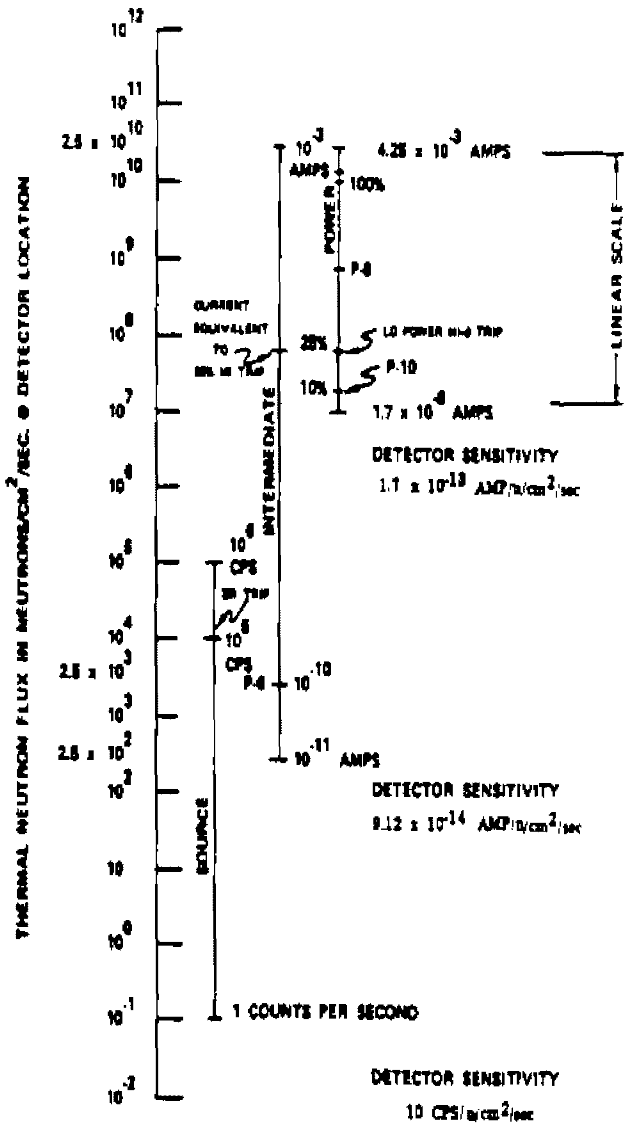
Mode 1 (cont)

- Normally measured T_{avg} is kept within 1 ^oF of programmed T_{avg} (T_{reff})
- Power is increased to 50-65% RTP and second feedwater pump is brought online (System 80 has two feed water pumps)
- Plant is stabilized at 70% RTP for additional incore flux maps and excore detector checks
- Power is increased to 90% RTP and held for final excore calibration to the secondary side calometric
- Power is increased to 100% RTP and final power ascension flux map is performed

Mode 1 (cont)

- At 70 and 100% RTP measured peaking factors checked against predicted values for TS surveillance
  - Ensures DNBR margin is available for possible AOOs
- At 100% RTP neutron flux is ~$(10)^{14}$ n/cm^2-sec or 10 decades above HZP
- Boron is added or removed for minor power changes, fuel and burnable poison depletion

Thanks for your attention



Typical Operating Range for Neutron
Detectors and Their Sensitivities

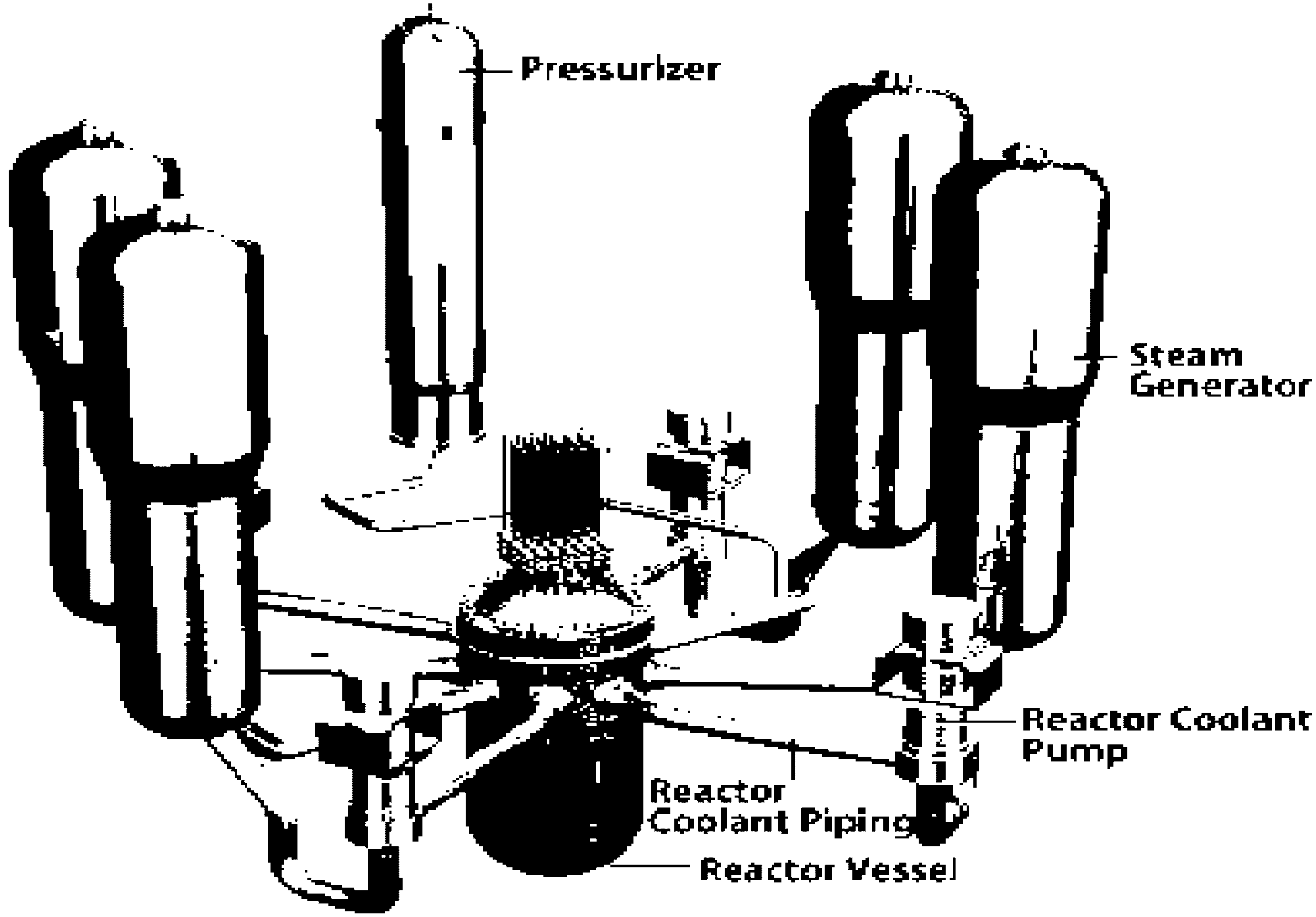
FIG. 2

Reactor Vessel and Internals: History, Issues & Resolution

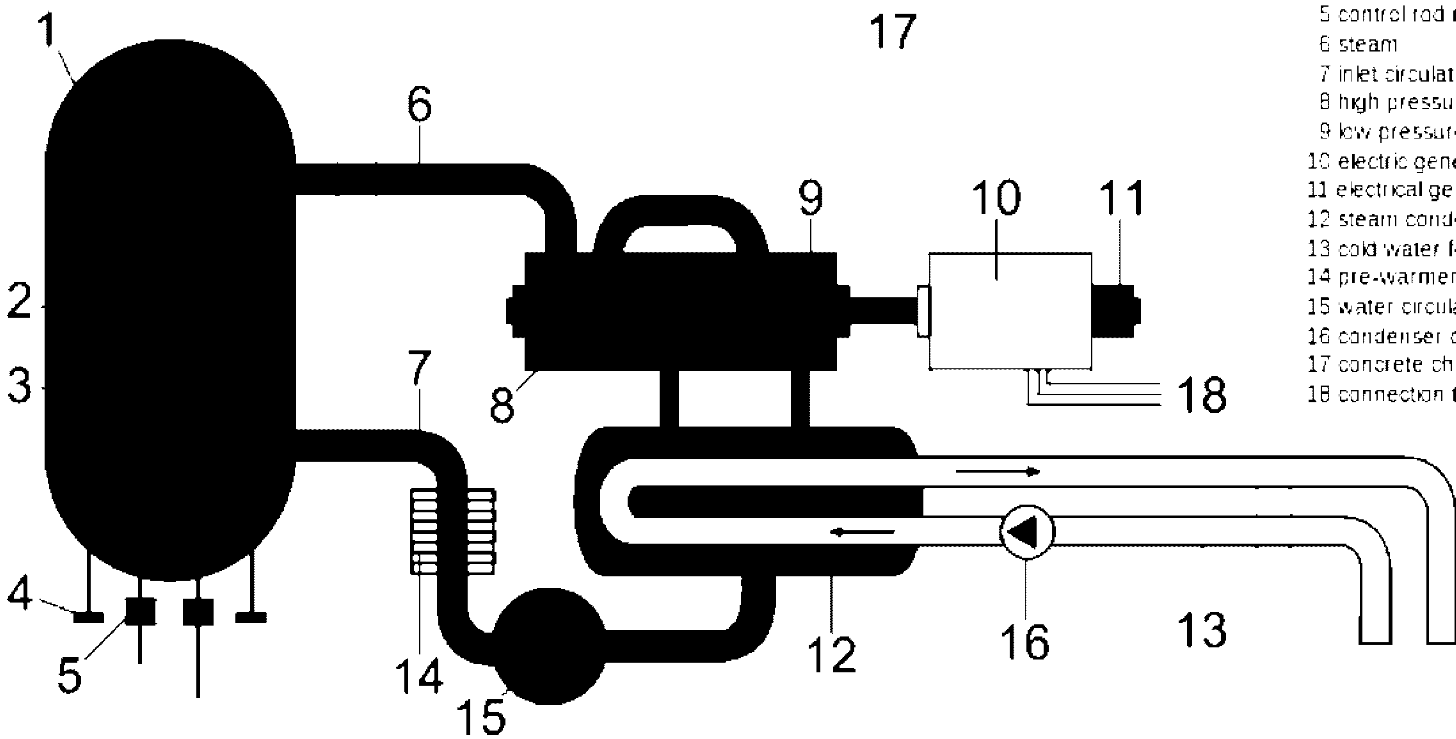
Neil Ray

May 2, 2013

Typica


Pressurizer
Steam Generator
Reactor Coolant Pump
Reactor Coolant Piping
Reactor Vessel

SCHEMATIC OF BWR PLANT



1 reactor vessel
2 fuel core element
3 control rod element
4 circulation pumps
5 control rod motors
6 steam
7 inlet circulation water
8 high pressure turbine
9 low pressure turbine
10 electric generator
11 electrical generator exciter
12 steam condenser
13 cold water for condenser
14 pre-warmer
15 water circulation pump
16 condenser cold water pump
17 concrete chamber
18 connection to electricity grid

Reactor Vessel: Facts and Fictions

Operating the plant for years, why does it take more time to startup/shutdown now—did not modify the RV?

Plant is operating so long—how vessel material properties changed?

RV is the only component for which the plant was forced to shut down (which one?)

RV never repaired or replaced –yet

RV never annealed in USA; however it was done several times in Russia

"Pressurized Thermal Shock (PTS) is like pouring cold water in a hot glass"

Reactor Vessel

Back Ground : Reactor Vessel

Reactor vessel construction

- Vessels designed and built per ASME Section III (prior to Section III, some vessels were built per Section I and VIII)
- Major types of construction in existing plants
  - Steel plates formed and welded to produce vessel structure
  - Ring forging, eliminated the longitudinal welds

Reactor Vessel (Cont'd)

U.S. Reactor vessel manufacturer

- B & W and CE manufactured their own vessels
- Westinghouse used other manufacturers:
  - B & W
  - CE
  - Chicago Bridge and Iron
  - Roterdam Dockyard Co
  - Creusot-Loire
  - MHI

Reactor Vessel (Cont'd)

BWR vessel materials and manufacturer

- Most of the vessels were manufactured by B&W, CE, and Chicago Bridge and Iron

Stainless steel cladding was used on the inside surface of the vessel (PWRs and BWRs) to minimize general corrosion

Reactor Vessel (Cont'd)

Typical current vessel thickness/radius (at beltline)

- Westinghouse
  - 4 loop – 8.625/86.5 in; 3 loop – 7.88/78.5 in; 2 loop – 6.5/66 in
  - Combustion Engineering: 8.5-8.62/86-86.8 ;System 80: 11.19/97.1
  - B & W: 8.44/85.5
  - GE : 90-140

AP1000: 8/78.6

EPR: 9.84/97.2

USAPWR: 8/79.5

ESBWR: 7.05/140

ABWR: 6.85/140

Issues Affecting Reactor Vessel Integrity

Fatigue

Underclad cracking

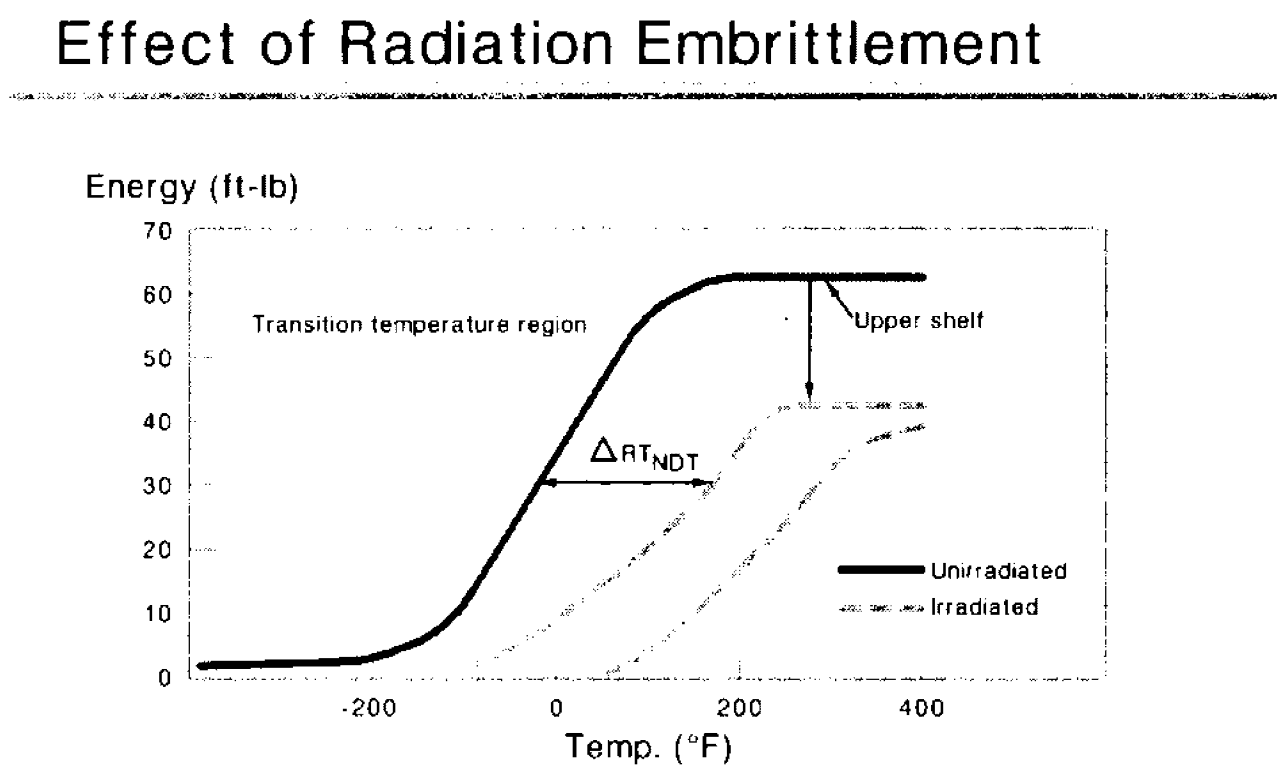
Radiation Embrittlement

- Pressurized Thermal Shock (PWRs)
- Heatup Cooldown Limits
- Hydrotest/Leak test
- Upper Shelf Energy

Radiation Embrittlement

- Displacements caused by high energy neutrons can change the metal microstructures
- Mechanical properties are affected by subtle changes in the microstructure
- Increase in yield strength can produce a shift in the ductile to brittle transition temperature (ΔTT)
- Embrittlement results in an increase in Charpy transition temperature and drop in upper-shelf energy (ΔUSE)

Radiation Embrittlement In Pressure Vessel Steels (cont'd)



Radiation Embrittlement In Pressure Vessel Steels (cont'd)

- Code of Federal Regulation adopted 10CFR50, Appendices G and H requirements for fracture toughness and materials surveillance (1972)
- NRC Regulatory Guide 1.99, Rev. 1 established embrittlement curve prediction method (1977)

Radiation Embrittlement In Pressure Vessel Steels (cont'd)

Reg. Guide 1.99, Rev. 2 updated trend curves to include copper and nickel predicting embrittlement in vessel materials (1988)

- Temperature dependant
- Material chemistry dependant
- High phosphorous RV cannot use Reg. 1.99, Rev 2

Reg. Guide 1.99, Rev. 3 currently under development

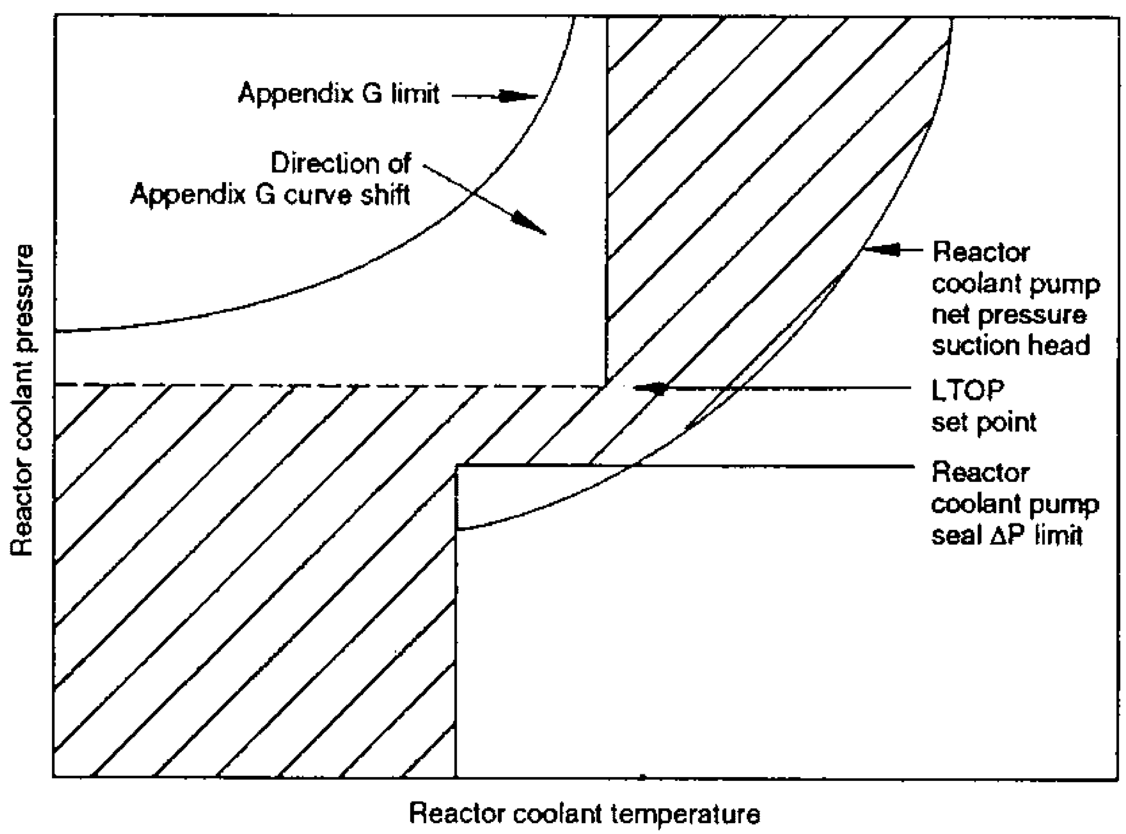
Pressure-Temperature Limits: Highlights

$2\ K_{IM} + 1.0\ K_{IT} < K_{Ic}$

Other factors affecting composite curves:

- Boltup temperature
- 10 CFR50 rule for closure flange regions: when pressure exceeds 20% of pre-service hydrostatic test pressure (621 psig for Westinghouse plants), temperature of closure flange regions must be >120°F plus initial reference temperature of material in those regions for normal operation and >90°F plus reference temperature for leak tests.
- Criticality limits: P-T limits for core operation are that the reactor vessel must be at a temperature equal to or higher than the minimum temperature required for the inservice hydrostatic test, and at least 40° F higher than the minimum permissible temperature in the corresponding P-T curve for heatup and cooldown.

Effects of Radiation Embrittlement on P-T limits



Pressurized Thermal Shock: Definition

PTS

- An event or transient in PWRs causing severe overcooling (thermal shock) concurrent with or followed by significant pressure in the reactor vessel

A PTS concern arises when

- A PTS transient occurs on the beltline region of a reactor vessel
- The beltline region has a reduced fracture resistance because of neutron radiation, and
- A flaw exists near the inner surface of the vessel wall

Current PTS Rule

- Established RT_{PTS} screening criteria (measure of fracture resistance)
- 270° F for plates, forgings, axial welds
- 300° F for circumferential weld materials
- All plants submitted RT_{PTS} values for end-of-license
- Plant-specific analysis within 3 years of reaching screening criteria

Basis for PTS Rule

The current PTS rule attempts to minimize risk of vessel failure by limiting the level of vessel embrittlement using a single index

Screening criteria is a function of

- Materials
- Fluence
- Transients

PTS Status for Current Reactors

NRC currently estimates that following 10 CFR 50.61, following plants will exceed PTS screening criteria during extended life:

- Point Beach 2 (2017)
- Palisades (2017)
- Diablo Canyon 1 (2033)
- Indian Point 3 (2025)
- Beaver Valley 1 (2033)

It is expected these plants may be able to qualify PTS using 10 CFR 50.61a

All other existing PWRs will not have any PTS issue during extended life

PTS Status: New Reactors

Based on the projected material properties and the radiation embrittlement

- New PWRs currently under NRC review will not have any PTS issues using the current criteria (10 CFR50.61)

Upper Shelf Fracture Toughness

RV beltline materials must have Charpy upper-shelf energy in the transverse direction for base material and along the weld for weld material of no less than 75 ft-lbs initially and must maintain no less than 50 ft-lbs unless it is demonstrated and accepted by the NRC (10 CFR Part 50, Appendix G)

For materials falling below 50 ft-lb, perform an equivalent margin fracture mechanics analysis

Addressing issues related to P-T, PTS, and USE: Current Reactors

P-T Limits

- Changed from $2K_{IM}+1.25K_{IT} < K_{IR}$ to $2K_{IM}+K_{IT}<K_{IC}$
- Reducing heatup/cooldown rates: longer time, cost money

P-T, PTS, USE

- Flux reduction: rearranging the burned fuel
- Neutron shield: stainless steel shielding around the core
- Annealing
- Revisiting PTS rule: 10 CFR 50.61a published

Reactor Vessel Integrity: PWR Vs. BWR (cont'd)

BWR does not use boron for controlling reactivity except in emergency

Bottom head penetrations

- PWR
  - Westinghouse: yes
  - B & W: Yes
  - C E: No, except Palo Verde
- BWR: Yes (includes CRDM and BMI)

PWR vessels: cladding the entire vessels

BWR vessels: No cladding in upper heads

Reactor Vessel Integrity: PWR Vs. BWR

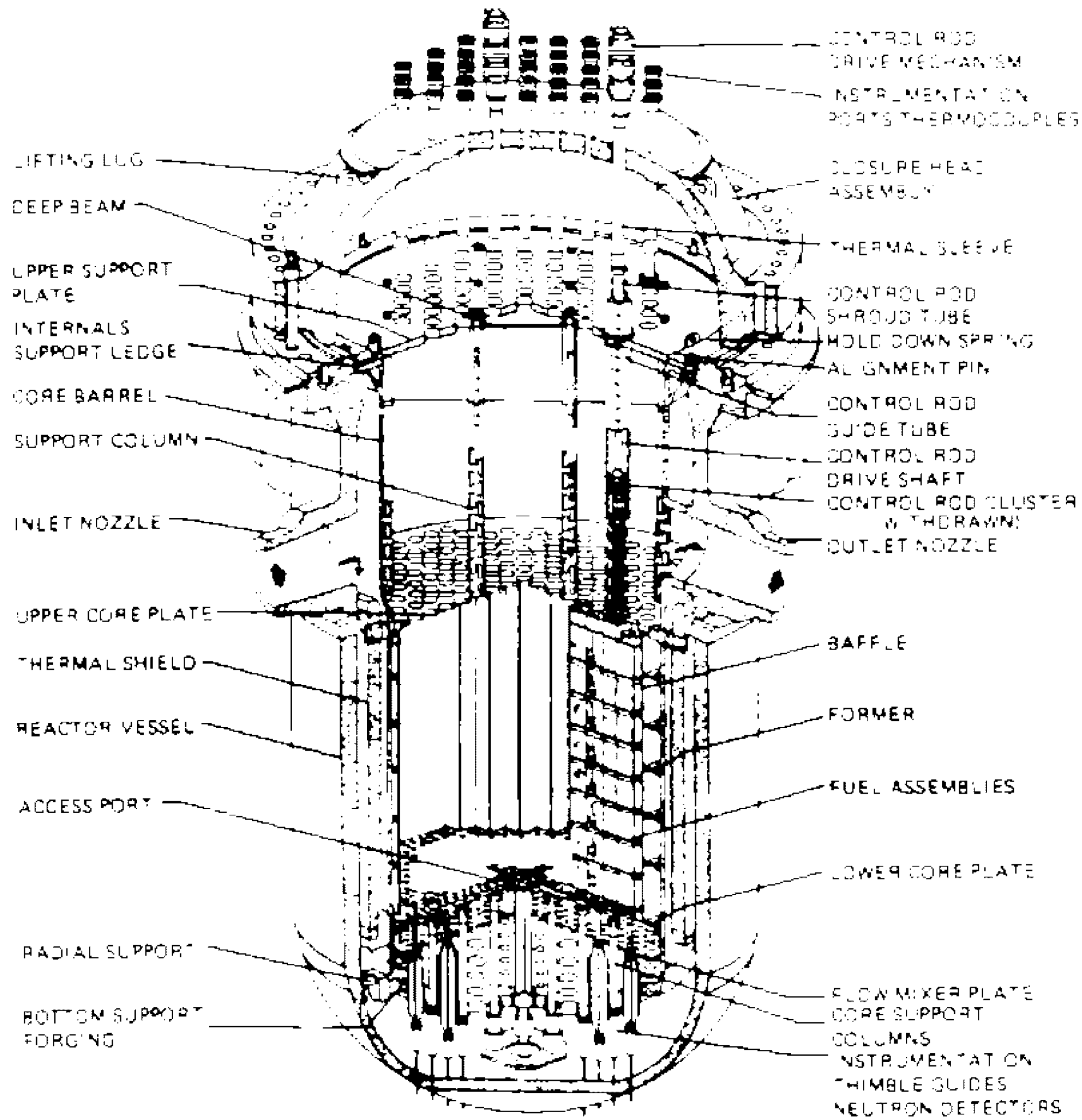
- Because of BWR's higher vessel diameter and fuel design, it accumulates less EOL fluence.
- PTS is not a concern for BWR
- BWR hydro problematic with higher embrittlement of vessel
- PWR: Can quickly heatup by using higher capacity reactor coolant pump, but limited by conflicting requirements (e.g., pump seal, NPSH, LTOP)

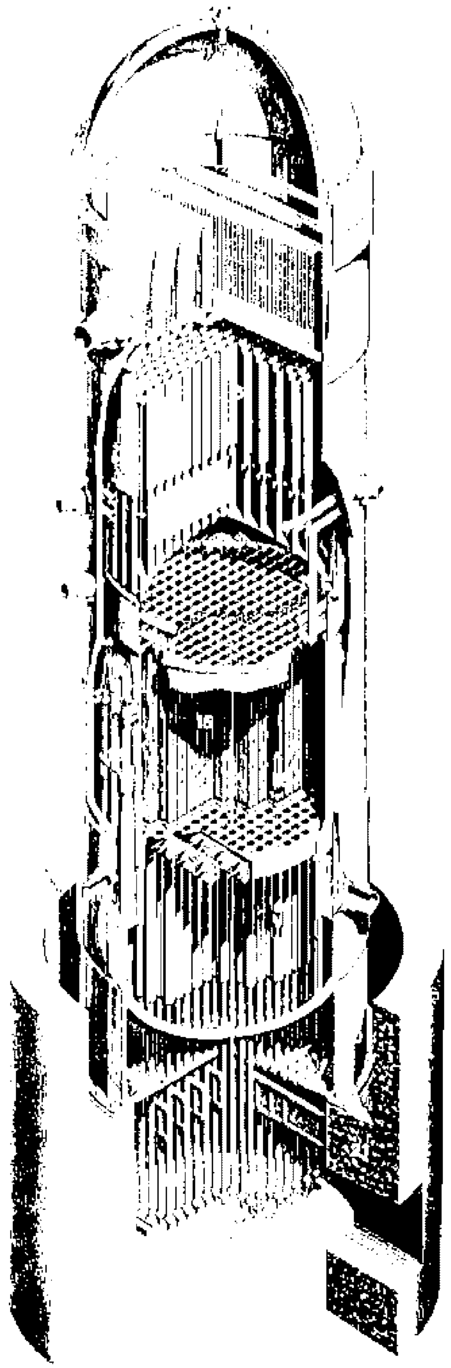
Addressing Issues: New Reactor Vessels

Radiation embrittlement

- Additional shielding to reduce cumulative fluence
- Reduced Cu/Ni content
- Eliminate beltline weld (AP1000 & ESBWR)
- Start with higher USE
- One P/T limit for 60 years
- PTS issues resolved (PWR only)

Reactor Vessel Internals


CONTROL ROD
DRIVE MECHANISM
INSTRUMENTATION
PORTS THERMOCOUPLES
CLOSURE HEAD
ASSEMBLY
THERMAL SLEEVE
CONTROL ROD
SHROUD TUBE
HOLD DOWN SPRING
ALIGNMENT PIN
CONTROL ROD
GUIDE TUBE
CONTROL ROD
DRIVE SHAFT
CONTROL ROD CLUSTER
WITHDRAWN
OUTLET NOZZLE
BAFFLE
FORMER
FUEL ASSEMBLIES
LOWER CORE PLATE
FLOW MIXER PLATE
CORE SUPPORT
COLUMNS
INSTRUMENTATION
THIMBLE GUIDES
NEUTRON DETECTORS
LIFTING LUG
DEEP BEAM
UPPER SUPPORT
PLATE
INTERNALS
SUPPORT LEDGE
CORE BARREL
SUPPORT COLUMN
INLET NOZZLE
UPPER CORE PLATE
THERMAL SHIELD
REACTOR VESSEL
ACCESS PORT
RADIAL SUPPORT
BOTTOM SUPPORT
FORGING



BWR/6

REACTOR ASSEMBLY

1. VENT AND HEAD SPRAY
2. STEAM DRYER LIFTING LUG
3. STEAM DRYER ASSEMBLY
4. STEAM OUTLET
5. CORE SPRAY INLET
6. STEAM SEPARATOR ASSEMBLY
7. FEEDWATER INLET
8. FEEDWATER SPARGER
9. LOW PRESSURE COOLANT INJECTION INLET
10. CORE SPRAY LINE
11. CORE SPRAY SPARGER
12. TOP GUIDE
13. JET PUMP ASSEMBLY
14. CORE SHROUD
15. FUEL ASSEMBLIES
16. CONTROL BLADE
17. CORE PLATE
18. JET PUMP / RECIRCULATION WATER INLET
19. RECIRCULATION WATER OUTLET
20. VESSEL SUPPORT SKIRT
21. SHIELD WALL
22. CONTROL ROD DRIVES
23. CONTROL ROD DRIVE HYDRAULIC LINES
24. IN CORE FLUX MONITOR

GENERAL ELECTRIC

PWR RVI Components/Susceptible Components

RVI components: (1) Plenum Assembly—top of the fuel assembly and supports the CRGT assembly and (2) Core Support Assembly which sits on top of core barrel assembly

Susceptible components—Baffle bolts, former bolts—Impacted by IASCC

Primary components

- core support shield (CSS) top flange, outlet nozzle, vent valve, upper and lower core barrel bolts—Impacted by SCC, wear, and thermal embrittleemnt in CASS valves
- core barrel assembly– baffle plates, baffle-to-former bolts, baffle-to-baffle bolts—Impacted by IASCC, neutron embrittlement, void swelling,irradiation-enhanced stress relaxation, fatigue and wear

Degradation Mechanisms: Internals

PWRs are exposed to higher neutron fluence than the BWRs

During the license renewal period, the PWR RVI components are susceptible to IASCC, neutron embrittlement, void swelling and irradiation-enhanced stress relaxation

In BWRs predominant aging degradation mechanism is IGSCC and it is due to higher (than the PWRs) oxygen content in RCS water.

- BWR normal water chemistry—200 ppb dissolved oxygen, 10 ppb dissolved hydrogen
- BWR hydrogen water chemistry (HWC) 1 ppb dissolved oxygen, 300 ppb dissolved hydrogen
- BWR HWC + Noble Metal Chemical Addition (NMCA) – 60 ppb dissolved oxygen, 30 ppb dissolved hydrogen-The exact water chemistries depend on the location inside the reactor vessel

Degradation Mechanisms: Internals

- Predominant aging mechanism—IGSCC which occurs due to the presence of tensile stress, oxygenated water [normal water chemistry (NWC)]-- 200 ppb, and sensitized stainless steel—Cold work can enhance IGSCC even in 316L material—steam dryers
- Addition of hydrogen ties up the oxygen and reduces IGSCC
- Protection is achieved when hydrogen water chemistry (HWC) is available for 80% of the total time at power operations or hot standby conditions
- Noble metal chemical addition (NMCA) + HWC will reduce the crack growth rates. HWC+NMCA will be effective only in water and they are least effective in steam phase or in duel phase(water and steam). HWC+NMCA should be present in RCS water at a minimum of 90% of the total time at power operations or hot standby conditions

Addressing Issues in New Reactor: Internals

Internals degradation

- Core barrel/core shroud
  - AP1000/EPR/USAPWR-reduced cumulative fluence
  - ESBWR- "L" grade austenitic stainless and Alloy 600
  - AP1000/EPR/USAPWR-"L" grade austenitic stainless steel
  - Significant change in design in all new reactors

Steam Dryer

- ESBWR/ABWR- avoid synchronized frequency
- Use of "L" grade austenitic stainless steel
- Support modified

Few words on SMRs

iPWRs: mPower, NuScale, Westinghouse

- NSSS is inside the RV
- NSSS is inside the RV: does it mean it is internal and no analysis needed?
- Neutron shield inside RV

HTGR: RV vessel is about twice the size of two AP1000 vessel

- Different environment
- High temperature issues

More Questions ?

Suggested Regulatory Modifications to Effectively Regulate I&C Diversity for Advanced Reactors

Ken Mott
NRO / DE / ICE1
August 5th, 2010

Agenda

Anticipated Transient Without Scram System

- Regulations, Policy and Guidance
- Purpose for Implementation

Diverse Actuation System

- Regulations, Policy and Guidance
- Purpose for Implementation

Comparison of ATWS and DAS Systems

Suggested Regulatory Modifications for Diversity Evaluations

- 10 CFR 50.62 and SRM to SECY-93-087
- NUREG/CR-6303 and NUREG/CR-7007

Anticipated Transient Without Scram
Regulations and Guidance

Regulation

10 CFR 50.62, "Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-water-cooled Nuclear Power Plants

Each pressurized water reactor must have equipment … , that is diverse from the reactor trip system, to automatically initiate the auxiliary (or emergency) feedwater system and initiate a turbine trip under conditions indicative of an ATWS.

Guidance

Standard Review Plan Section 7.8, Diverse Instrumentation and Control Systems

The objectives of this review are to assure that the ATWS mitigation systems and equipment are designed and installed in accordance with the requirements of 10 CFR 50.62.

Purpose for Implementation

Reduce the probability of unacceptable consequences following anticipated operational occurrences. [49FR26040]

Diverse Actuation System

Regulations and Guidance

Regulations

10CFR Part 50, Appendix A, General Design Criterion 22, Protection System Independence.

The protection system shall be designed to assure that the effects of … postulated accident conditions on redundant channels do not result in loss of the protection function…. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

Policy

SRM to SECY-93-087, Item II.Q, Defense Against Common-Mode failures in Digital Instrumentation and Control Systems

The vendor or applicant shall analyze **each** postulated common-mode failure for **each event** that is evaluated in the accident analysis section of the safety analysis report (SAR) using best-estimate methods. The vendor or applicant shall demonstrate adequate diversity within the design for each of these events

Purpose for Implementation

Defense Against Common-mode Failures in Digital Instrumentation and Control Systems

Comparison of ATWS and DAS Systems

Functionality

Protection System

Reactor Trip Portion

Comparison of ATWS and DAS Systems
continued

[covering PWRs only]

Initiate Turbine Trip

Initiate Auxiliary/Emergency Feedwater

Diverse Scram

[for CE or B&W only]

Initiate either the same protective function or a different function as failed protection system

Deterministic Analysis [10CFR100 limits]

Reactor Coolant System and/or Containment Integrity

Suggested Regulatory Modifications for Diversity Evaluations

10 CFR 50.62 and SRM to SECY-93-087

The DAS Policy, SRM to SECY-93-087, may supersedes the ATWS Rule, 10CFR50.62, for new generation plants

-The best estimate analysis covers postulated common-mode-failures for SAR events which includes ATWS events

The SRM to SECY-93-087 is probably more significant to safety than 10CFR50.62

-Far greater purpose than ATWS Rule (10CFR50.62)

-PWR certification applicants are currently designing 60 year DAS systems against SRM to SECY-93-087 policy

CONCLUSION:

-SRM to SECY-93-087, Item II.Q, should be a Rule.

Suggested Regulatory Modifications for Diversity Evaluations

NUREG/CR-6303 and NUREG/CR-7007

NUREG/CR-7007 Diversity Modifications to NUREG/CR-6303

NUREG/CR-6303 [1994]	NUREG/CR-7007 [2010]
Design diversity	Design diversity
Equipment diversity	**Equipment manufacturer**
	Logic processing equipment
Functional diversity	Functional diversity
Human diversity	**Life-Cycle**
Signal diversity	Signal diversity
Software diversity	**Logic diversity**

Suggested Regulatory Modifications for Diversity Evaluations

NUREG/CR-6303 and NUREG/CR-7007

NUREG/CR-6303, Method for Performing Diversity and Defense-in-Depth Analyses of Reactor Protection Systems [1994]

- -Does not provide the applicant with any certainty of having adequate and/or sufficient diversity within the proposed design
- -Design details for modern I&C component diversity evaluation are vague, lacking

NUREG/CR-7007, Diversity Strategies for Nuclear Power Plant Instrumentation and Control Systems [2010]

- -Provides Modified Diversity attributes for modern I&C designs
- -Provides Diversity Evaluation Tool (Excel spreadsheet based tool)
- -Provides design threshold for diversity evaluation

CONCLUSION:

NUREG/CR-7007 should replace NUREG/CR-6303 with all pertinent information being incorporated into NUREG/CR-7007.

Summary of Suggested Diversity Modifications

SRM to SECY-93-087, Item II.Q, should be a Rule

NUREG/CR-7007 should replace NUREG/CR-6303 with all pertinent information being incorporated into NUREG/CR-7007.

Questions

REFERENCES

10 CFR Part 50

[*http://www.nrc.gov/reading-rm/doc-collections/cfr/part050.html*]

SRM to SECY-93-087, Item II.Q, Defense Against Common-Mode failures in Digital Instrumentation and Control Systems [*ADAMS Accession No. ML003708056*]

NUREG/CR-6303, Method for Performing Diversity and Defense-in-Depth Analyses of Reactor Protection Systems [*ADAMS Accession No. ML071790509*]

NUREG/CR-7007, Diversity Strategies for Nuclear Power Plant Instrumentation and Control Systems [*ADAMS Accession No. ML100880143*]

Generic Letter 85-06, Quality Assurance Guidance for ATWS Equipment that is not Safety-Related [*ADAMS Accession No. ML031140390*]

Branch Technical Position 7-19, Guidance for Evaluation of Diversity and Defense-in-depth in Digital Computer-based Instrumentation and Control Systems

[ADAMS Accession No. ML070550072]

Station Blackout and Emergency Diesel Generator

Station Blackout: Amar Pal

Station Blackout and Fukushima Event: Roy Mathew

Emergency Diesel Generator Reliability and Testing: Om Chopra

STATION BLACKOUT

AMAR N. PAL
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STATION BLACKOUT

Definition

Station Blackout means the complete loss of alternating current (ac) electric power to the essential and nonessential switchgear buses in a nuclear power plant (i.e., loss of offsite electric power system concurrent with turbine trip and unavailability of the onsite emergency ac power system). Station blackout does not include the loss of available ac power to buses fed by station batteries through inverters or by alternate ac sources.

STATION BLACKOUT

Why concern for SBO

- Unacceptable consequences – Ultimate Core Melt/ Containment Failure
- Many total and partial LOOPS have occurred.
- Many EDG failures.
- Risk Analysis showed SBO important Risk Contributor (WASH 1400-1975, NUREG- 1032)

STATION BLACKOUT

Regulation

10 CFR 50.63, "Loss of All Alternating Current Power" (Station Blackout Rule) became effective July 1988. The regulation requires that each light-water-cooled nuclear power plant licensed to operate under 10 CFR Part 50, each light-water-cooled nuclear power plant licensed under subpart C of 10 CFR part 52 after the Commission makes the finding under § 52.103(g) of this chapter, and each design for a light-water-cooled nuclear power plant approved under a standard design approval, standard design certification, and manufacturing license under part 52 of this chapter must be able to withstand for a specified duration and recover from an SBO. Additionally, the reactor core and associated coolant, control, and protection systems, including station batteries and any other necessary support systems, must provide sufficient capacity and capability to ensure that the core is cooled and appropriate containment integrity is maintained in the event of an SBO for the specified duration.

The staff issued RG 1.155 to describe a method acceptable for complying with the Commission regulation.

The industry issued NUMARC 87-00 to provide guidance and methodologies for implementing SBO initiatives. The staff endorsed Rev. 0 of NUMARC 87-00.

STATION BLACKOUT

Coping duration

Emergency ac power configuration group, EDG Reliability, Offsite power design characteristic group

Operating plants – 4 hours to 16 hours

New Plants

- Passive Design (Coping with battery for 72 hours)
- Non-Passive Design (8 hours)

STATION BLACKOUT

Coping Method

Operating Plants

- Battery - 41 Plants (23 PWRs and 18 BWRs)
- Alternate ac (AAC) power source - 67 Plants (DG, CTG, GTG, Appendix R DG, Hydro Power, EDG Excess Capacity, EDG Excess Redundancy)
  - Single Unit Site - AAC must have capability for safe shutdown (Hot standby or hot shutdown) of the unit.
  - Multi Unit Site
    - At multi-unit sites, where emergency ac is not shared, AAC must have capability for safe shutdown of any of the units. One unit will be in SBO condition.
    - At multi-unit sites, where emergency ac is shared, AAC must have capability for safe shutdown of all units. All units will be in SBO condition.

New Plants

- Passive Design (battery)
- Non-Passive Design (AAC power with diverse design per unit) (SECY-90-16 and 91-078)

STATION BLACKOUT

Alternate ac power source (NUMARC 87-00, Appendix B)

AAC power source must meet NUMARC 87-00, Appendix B. These are:

Items B.1 and B.2 require that the AAC system need not be Class 1E or be protected against seismic events or failure or misoperation of other plant equipment.

Item B.2 requires that the AAC system be protected against likely weather related events.

Items B.4, B.5, B.6, and B.7 require that AAC source be physically and electrically separated (normally) from safety-related equipment or the preferred or onsite emergency ac power system so that it will not adversely impact this equipment or these systems.

Items B.8(a) through B.8(c) require that the AAC system have its own dc power source, air start system, fuel supply, and other support systems to minimize potential common cause failure of the AAC source and the onsite emergency ac power systems.

STATION BLACKOUT

Item B.8(d) requires that corrective action be taken for failures common to the AAC source and the onsite emergency ac power systems.

Item B.8(e) requires that a likely weather related event or single failure would not simultaneously fail the AAC source and the preferred and onsite emergency power systems.

Item B.8(g) requires that the AAC system be tested following maintenance activities.

Item B.9 requires that the AAC source be tested to demonstrate that it has the capacity and capability to maintain acceptable voltage and frequency while powering the SBO loads.

Item B.9 also states that the opposite unit of a multi-unit site may be used as AAC source provided it has sufficient excess capacity to simultaneously power the SBO loads of the blacked-out unit and the LOOP loads of the associated unit.

Items B.10 through B.12 require that the AAC source initially tested, periodically tested including timed start test, and periodically surveyed and maintained.

Item B.13 requires that the AAC source maintain a 95% reliability/availability

STATION BLACKOUT

AAC Time Classification

Ten minute AAC – AAC must be connectable to SBO load buses within 10 minutes. Does not require powering the loads within 10 minutes.

One hour AAC – Can power SBO loads within one hour. An AAC source that can power the loads in less than one hour (i.e., 30 minutes) is classified as one hour AAC.

STATION BLACKOUT

Coping Analysis

No coping analysis for 10 minute AAC plants

One hour coping analysis for AAC plants with greater than 10 minutes but less than one hour

Coping analysis for the coping duration for coping with battery.

SBO coping capability

- Condensate inventory for decay heat removal
- Class 1E battery capacity (load shedding after 30 minutes)
- Compressed air
- Effects of loss of ventilation
  - Control room and I&C cabinet room
  - Inverter room
- Containment isolation
- Reactor coolant inventory
- Communication and portable lighting

STATION BLACKOUT

Procedures and Training (NUMARC 87-00)

(1) SBO response guidelines

- Actions necessary to restore offsite power
- Actions to assure shutdown equipment is operable
- Actions to assure operability of AFWS/HPCIS/HPCS/RCICS
- Actions to prevent reactor inventory loss
- Actions to establish a flow path from the CST and to transfer to alternate sources
- Identification and actions to strip dc loads
- Actions to permit appropriate containment isolation
- Actions to permit safe shutdown valve operations
- Identification of portable lighting
- Identification of effects of ac power loss on area access

STATION BLACKOUT

- Actions to identify and mitigate effects of loss of ventilation:
  - Monitoring of room and cabinet temperatures
  - Actions to provide supplemental cooling
  - Actions to override HPCIS/RCICS on high temperature
  - Opening room and/or cabinet doors
  - Consideration of effects of high temperature on fire protection features
- Consideration of habitability requirements
- Actions to compensate for loss of heat tracing

STATION BLACKOUT

(2) <u>AC power restoration</u>

- Alternate methods of restoring power to nuclear units
- High priority to restoring at least one transmission line
- Priority for necessary manpower, equipment and materials
- Actions to obtain portable ac generator and associated equipment
- Upon restoration of ac power, actions to connect restored ac power to shutdown equipment

STATION BLACKOUT

(3) Severe weather guidelines

(a) Preparation for a hurricane

- Procedures should identify site-specific actions including following:
  - Identification and elimination of potential missiles
  - Assuring adequacy of site staff
  - Restoring out of service equipment to service
  - Warming and lubrication standby ac sources
  - Ensuring availability of AAC source (if available)
  - Increasing CST inventory
  - Placing battery charger in service
  - Testing EDGs

- Procedures should provide for identification of and method of contacting additional support staff
- Procedures should specify actions necessary to ensure equipment required for station blackout response is available.

STATION BLACKOUT

(b) Actions Prior to arrival of hurricane at site

Ensure that plant is in safe shutdown two hours prior to anticipated arrival of hurricane (wind speed › 73 mph)

Operator review of station blackout procedures

Operator review (if applicable) of procedures for switchyard spray down systems

(c) Actions for a Tornado

- Identification and elimination of potential missiles
- Restoring out of service equipment to service

STATION BLACKOUT

Challenges

10 minute clock

- NUMARC 87-00, Rev. 1, Appendix I, Response to Question 65 allow time to perform the immediate steps in the EOPs to verify scram, primary system parameters, etc., and attempt to restore offsite power and start EDGs from the control room. SBO clock starts after the failure of restoring offsite and onsite emergency ac power. Some plants took 15 minutes to perform above actions.
- 10 minute clock starts as soon as losses of offsite and onsite power occur.

SBO procedures for passive plants

- ESBWR DCD initially indicated that RG 1.155 is not applicable to passive design. In response to staff's RAI, the applicant revised DCD to include that RG 1.155 regarding SBO training and procedure is applicable to ESBWR design.
- AP 1000 COLA applicants provided SBO procedures in response to RAI.

Switchyard breakers dc power availability for offsite power restoration

- Switchyard breakers have one closing coil and two trip coils. Switchyard breakers are provided with redundant dc power supplies. ACRS members are concerned about availability of dc power with the failure of one dc power source for offsite power restoration.
- Additionally, switchyard battery duty cycle should be consistent with coping duration

STATION BLACKOUT

AAC power source capacity for cold shutdown

- SECY-91-016 recommends that new plants should have an AAC power source of diverse design and should have sufficient capacity to operate the systems necessary for coping with an SBO for time necessary to bring and maintain the plant in a safe-shutdown condition including cold shutdown.
  - ABWR design- AAC source (GTG) has enough capacity to bring the reactor to cold shutdown condition, can be made available within 10 minutes and has 7-day supply of fuel.
  - USAPWR design – AAC source (GTG) has enough capacity to bring the reactor to cold shutdown condition, can be made available within one hour and has 7 day supply of fuel.
  - EPR design – AAC source (DG) does not have enough capacity to bring the reactor to cold shutdown condition and has 24-hour supply of fuel but can be made available within 10 minutes. The applicant follows the safe shutdown definition for SBO which means bringing the plant to hot standby.
- New reactor with passive designs have ancillary diesel generators, in addition to standby diesel generators, designed to provide the post 72-hour power requirements following an extended loss of all power sources. ESBWR design provides 7 days fuel capacity. AP1000 design provides 4 days fuel capacity.

STATION BLACKOUT

References

- 10 CFR 50.63
- RG 1.155
- NUMARC 87-00
- SRP 8.4
- SECY-90-16
- SECY-91-078
- Temporary Instruction 2515/120
- WASH 1400-1975
- NUREG-1032
- NUREG- 1109

Station Blackout and Fukushima Event

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Division of Engineering

Station Blackout Background

WASH-1400, “Reactor Safety Study,” issued 1975, indicated that station blackout (SBO) could be an important contributor to the total risk from nuclear power plant accidents

In 1980, the Commission designated the issue of station SBO as Unresolved Safety Issue (USI) A-44, “Station Blackout”

NRC issued the final SBO Rule (10 CFR 50.63) on June 21, 1988

NRC issued Regulatory Guide (RG) 1.155, “Station Blackout,” on August 1988 and endorsed NUMARC 87-00 industry guidance to implement the SBO Rule

Station Blackout Requirements

U.S. Plants

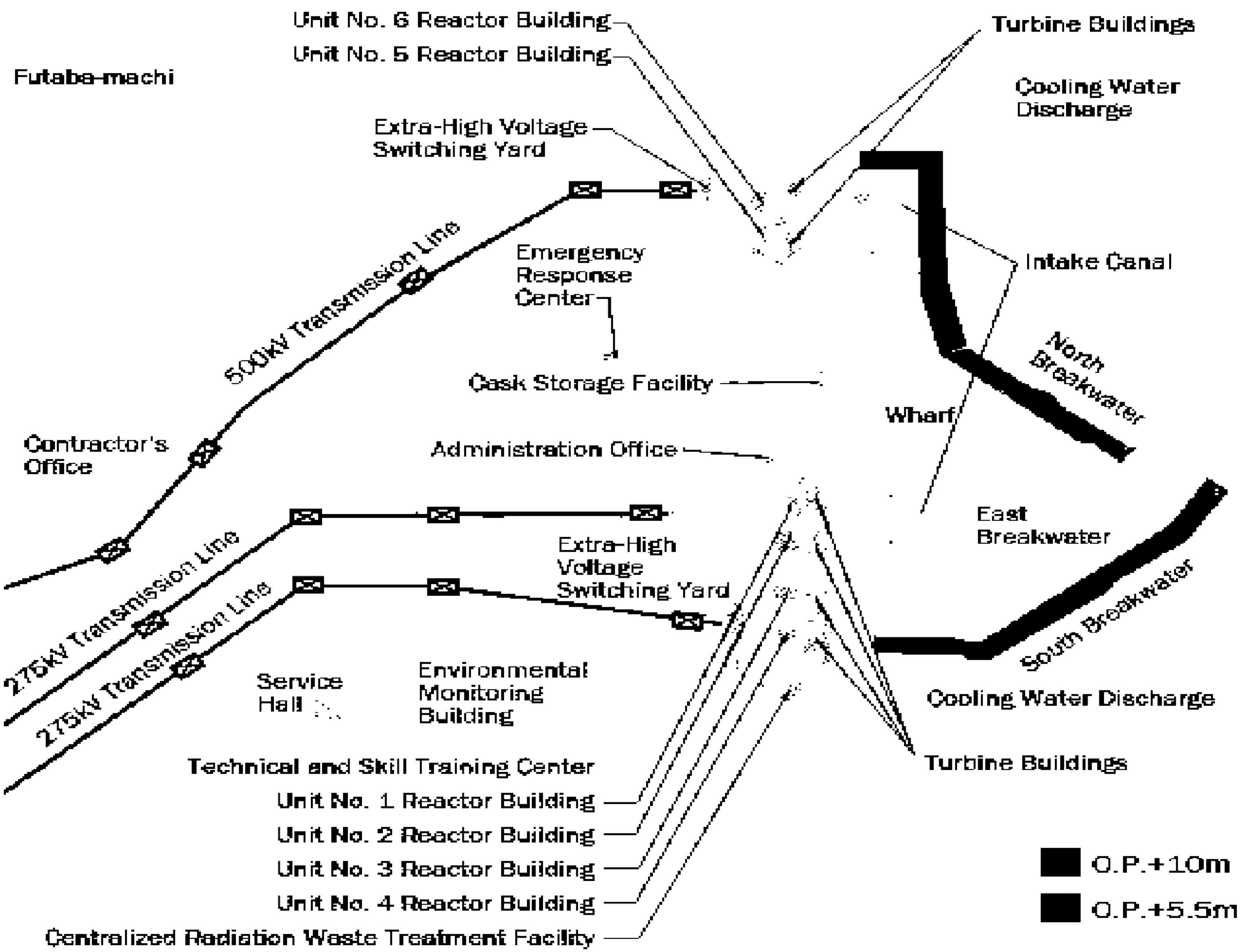
Each light-water-cooled nuclear power plant be able to withstand and recover from a station blackout (i.e., loss of the offsite electric power system concurrent with reactor trip and unavailability of the onsite emergency ac electric power system) of a specified duration (two to sixteen hours). (10 CFR 50.63)

Japanese Plants

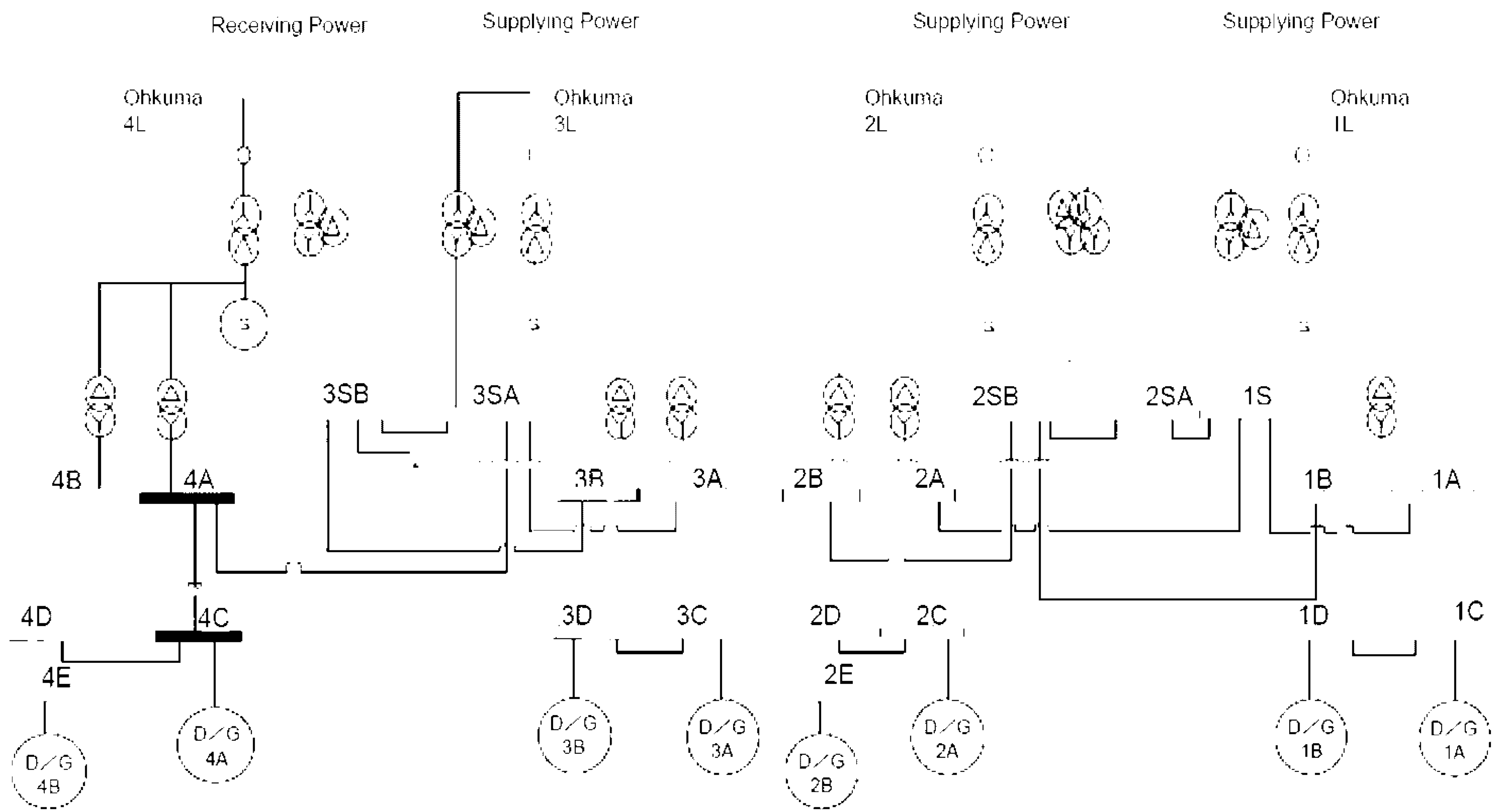
The nuclear power plant shall be provided with batteries that have capacity required to ensure that [they can] safely shutdown the reactor and cool it down after its shutdown even in the event of a loss of all alternating current (AC) power for a short period time. (Article 33)

- The nuclear reactor facilities shall be so designed that safe shutdown and proper cooling of the reactor after shutting down can be ensured in case of a short-term total AC power loss. (Regulatory Guide - Guideline 27)

Fukushima Site

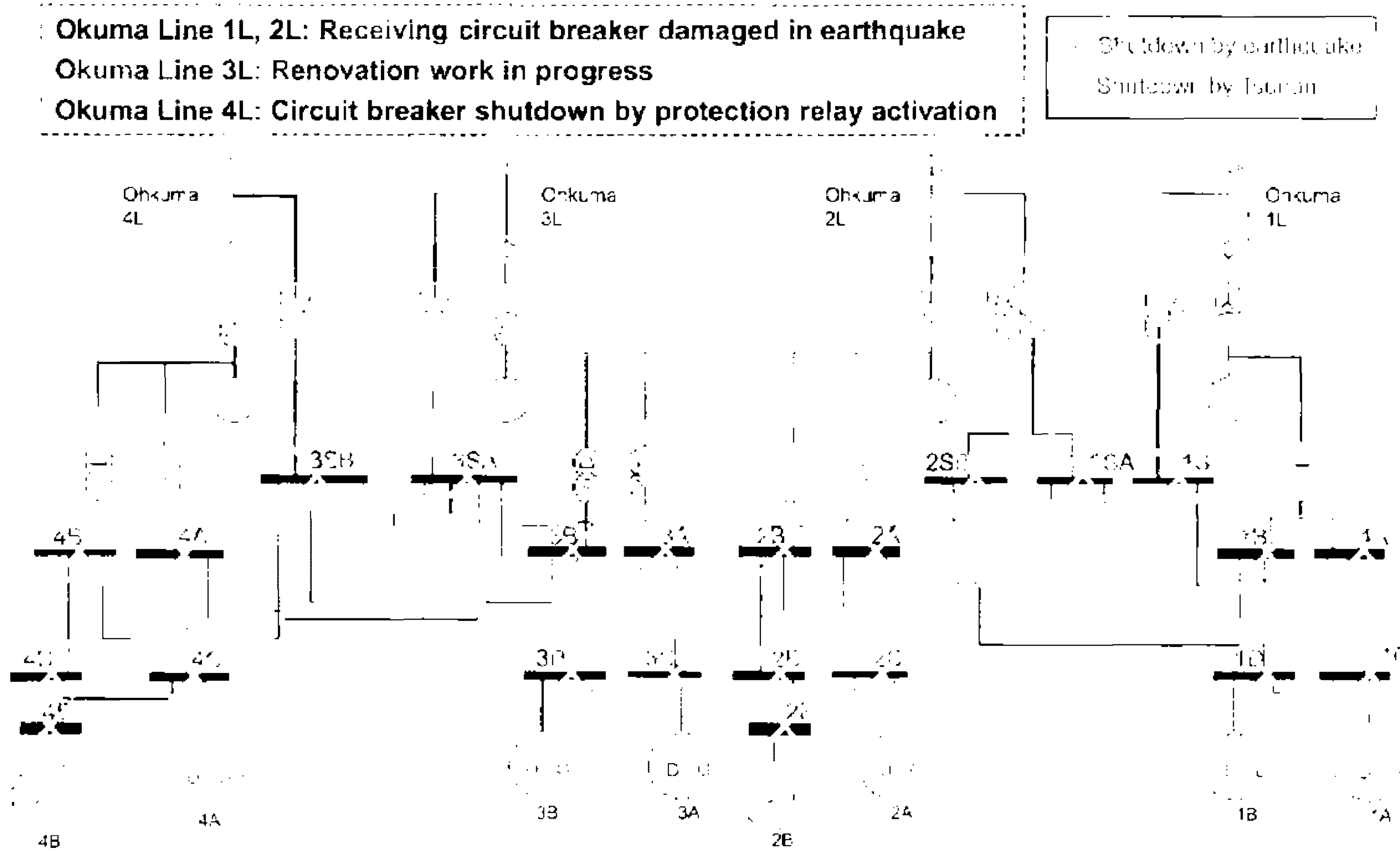


Fukushima Electrical System



Power supply of Unit 1-4 @ 1F

Okuma Line 1L, 2L: Receiving circuit breaker damaged in earthquake
Okuma Line 3L: Renovation work in progress
Okuma Line 4L: Circuit breaker shutdown by protection relay activation

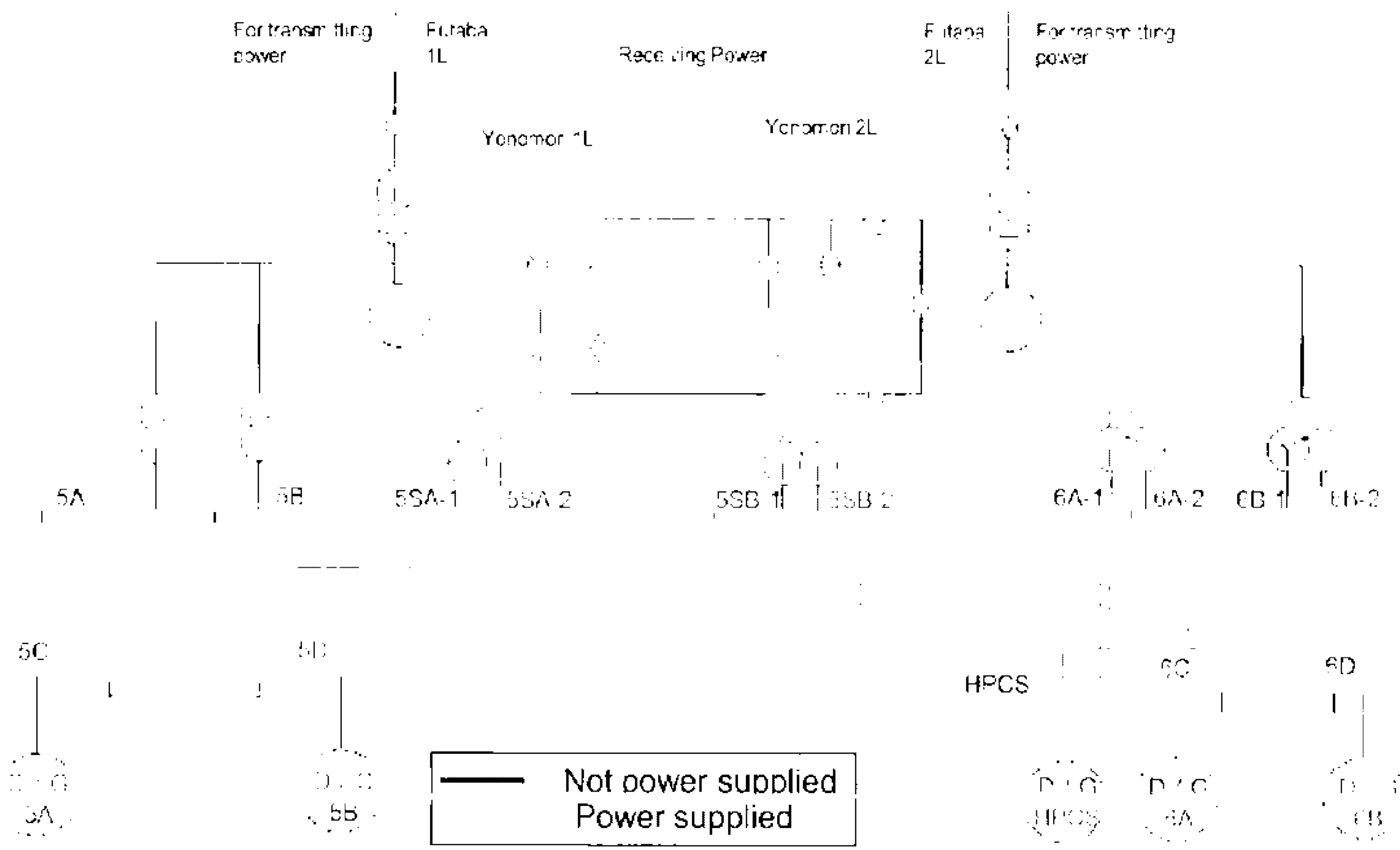


The DG lost the function due to either "M/C failure," "loss of sea water system," or "DG main unit failure."

24

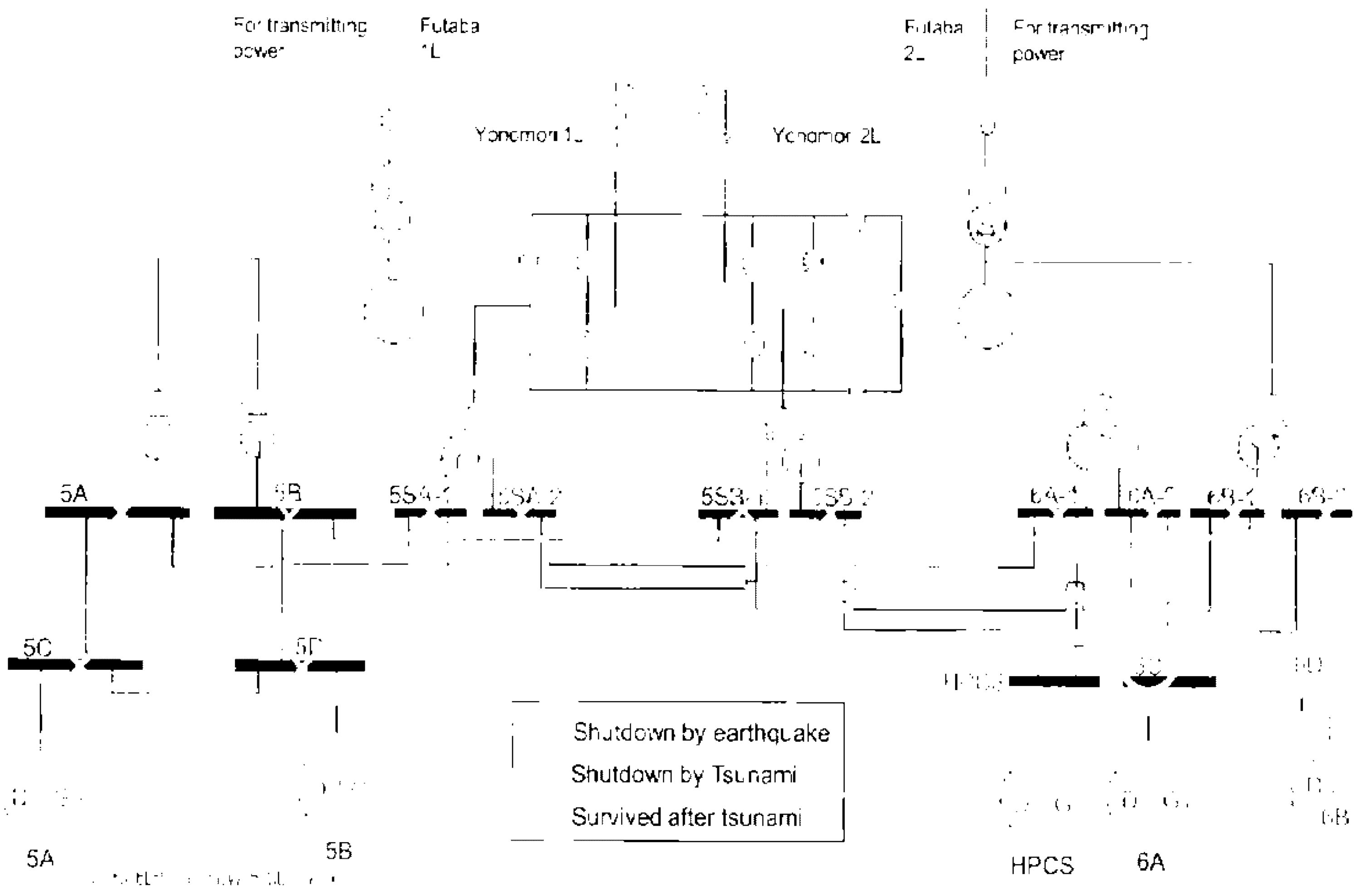
Power supply of Unit 5/6 @ 1F

Unit 5/6 during outage



25

Power supply of Unit 5/6 @ 1F



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		Fukushima Daiichi											
		Unit 1		Unit 2		Unit 3		Unit 4		Unit 5		Unit 6	
		Equipment Name	Status	Equipment Name	Status	Equipment Name	Status	Equipment Name	Status	Equipment Name	Status	Equipment Name	Status
EDG	(ac) = air cooled	EDG 1A	x	EDG 2A	x	EDG 3A	x	EDG 4A	x	EDG 5A	(2)	EDG 6A	(2)
		EDG 1B	x	EDG 2B (ac)	(1)	EDG 3B	x	EDG 4B (ac)	(1)	EDG 5B	(2)	EDG 6B (ac)	o
		--	--	--	--		--	--	--	--	--	HPCS EDG	(2)
6.9 kV Electrical Distribution	Vital	M/C 1C	x	M/C 2C	x	M/C 3C	x	M/C 4C	x	M/C 5C	x	M/C 6C	o
		M/C 1D	x	M/C 2D	x	M/C 3D	x	M/C 4D	x	M/C 5D	x	M/C 6D	o
		--	--	M/C 2E	x	--		M/C 4E	x	--	--	HPCS M/C	o
	Non Vital	M/C 1A	x	M/C 2A	x	M/C 3A	x	M/C 4A	x	M/C 5A	x	M/C 6A-1	x
												M/C 6A-2	x
		M/C 1B	x	M/C 2B	x	M/C 3B	x	M/C 4B	x	M/C 5B	x	M/C 6B-1	x
												M/C 6B-2	x
		M/C 1S	x	M/C 2SA	x	M/C 3SA	x	--		M/C 5SA-1	x	--	
										M/C 5SA-2	x		
				M/C 2SB	x	M/C 3SB	x			M/C 5SB-1	x		
										M/C 5SB-2	x		
480V Power Centers (P/C)	Vital	P/C 1C	x	P/C 2C		P/C 3C	x	P/C 4C	o	P/C 5C	x	P/C 6C	o
		P/C 1D	x	P/C 2D		P/C 3D	x	P/C 4D	o	P/C 5D	x	P/C 6D	o
		--	--	P/C 2E	x	--	--	P/C 4E	x	--	--	P/C 6E	o
	Non Vital	P/C 1A	x	P/C 2A		P/C 3A	x	P/C 4A	o	P/C 5A	x	P/C 6A-1	x
				P/C 2A-1	x	--	--	-	--	P/C 5A-1	o	P/C 6A-2	x
		P/C 1B	x	P/C 2B		P/C 3B	x	P/C 4B	o	P/C 5B	x	P/C 6B-1	x
		--	--	--	--	--	--		--	P/C 5B-1	o	P/C 6B-2	x
		P/C 1S	x	--	--	P/C 3SA	x	--	--	P/C 5SA	x	--	--
		--	--	--	--			--	--	P/C 5SA-1	x	--	--
			--	P/C 2SB	x	P/C 3SB	x	--	--	P/C 5SB	x	--	--
DC Power	[illegible]	DC 125V main bus A	x	DC 125V P/C 2A	x	DC 125V main bus 3A	o	DC 125V main bus 4A	x	DC 125V P/C 5A	o	DC 125V 6A	o
		DC 125V main bus B	x	DC 125V P/C 2B	x	DC 125V main bus 3B	o	DC 125V main bus 4B	x	DC 125V P/C 5B	o	DC 125V 6B	o
UHS		SW	x	RHR-S A	x	RHR-S A	x	RHR-S A	x	RHR-S A	x	RHR-S A	x
				RHR-S B	x	RHR-S B	x	RHR-S B	x	RHR-S B	x	RHR-S B	x

Status: x: damaged
o: available

Key: White background: Not damaged by the earthquake or tsunami
Blue background: Damaged or flooded by tsunami
Gray background: Support systems damaged or flooded by tsunami
(1): electrical distribution damaged or flooded
(2): ultimate hat sink damaged or flooded

Figure 7.4-7 Fukushima Daiichi Electrical Distribution Damage[12]

[12] "Overview of Accident at TEPCO Fukushima Nuclear Power Stations," July 22, 2011 - Tokyo Electric Power Company Co.

NRC ACTIONS - FUKUSHIMA EVENT

The Near Term Task Force (NTTF) issued Report on July 12, 2011

Recommended the Commission use orders to ensure that licensees take Near-Term Actions until requirements associated with future rulemakings can be implemented. Examples include:

- ✓ to reevaluate the seismic and flooding hazards at their sites
- ✓ to perform seismic and flood protection walkdowns to identify and address plant-specific vulnerabilities
- ✓ to provide reasonable protection for equipment currently provided pursuant to 10 CFR 50.54(hh)(2)
- ✓ to provide safety-related ac electrical power for the spent fuel pool makeup system.

Recommended the Commission strengthen SBO mitigation capability for design-basis and beyond design basis external events through amending the existing rule (10 CFR 50 63)

Status of SBO Rulemaking (Cont.)

Commission directed the staff in SRM SECY-11-0124 dated October 18, 2011, to implement the lessons learned from the Fukushima accident within five years - by 2016

- Initiate the SBO rulemaking as an Advanced Notice of Public Rulemaking (ANPR)
- Designate the SBO rulemaking as a high-priority rulemaking with a goal for completion within 24-30 months

  Monitor nuclear industry efforts to strengthen coping times and consider any interim controls required

Status of SBO Rulemaking (Cont.)

The staff is currently developing the ANPR package

Schedule

- Issue ANPR - By April 2012
- Conduct Public Meetings - 2012
- Address External Stakeholders Comments - 2012
- Proposed Rule to Commission - April 2013
- Final Rule to Commission - April 2014

Emergency Diesel Generator Reliability and Testing

Om Chopra
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Emergency Diesel Generator (EDG) Reliability and Testing

EDG Testing

- Qualification testing
- Preoperational testing
- Periodic testing

EDG Reliability

EDG Unavailability

EDG Qualification Testing

Branch Technical Position ICSB-2(PSB) 11/24/1975 NUREG - 75/087

Demonstrate start and load reliability of prototype EDG a 0.99 reliability by performing:

- Prior to fuel load two margin tests with some margin in excess of design requirement
- 300 valid start and load tests with no more than three failures allowed
    - At least 90% of the tests should be performed from design cold ambient conditions (design hot conditions if standby temperature control system is provided)
    - 10% from design hot equilibrium temperature conditions
    - Loading to at least 50% of the continuous rating

EDG Qualification Testing

This Branch Technical Position was subsequently superseded by IEEE-387 which requires a total of 100 valid start and load tests with no failures allowed. These tests will be conducted as follows:

- At least 90% of the tests should be performed from warm standby conditions
- 10% from design hot equilibrium temperature conditions
- Loading to at least 50% of the continuous rating
- Shall accept a single step load 50% of the continuous rating

EDG Pre-operational Testing

Preoperational testing per RG 1.108

- 69 consecutive valid start and load tests without any failures, with a minimum of 69/n test per EDG where n equals to the number of EDGs at a plant

The above requirements were subsequently superseded by RG 1.9 Rev. 3 which recommends a minimum of 25 valid start and-load demands without failure on each installed emergency diesel generator unit

EDG Periodic Testing

Periodic testing

- Monthly testing at least once in 31 days to ensure that EDG reliability is maintained at an acceptable level
- 6-monthly testing to demonstrate the capability of the EDG to start from standby and provide the necessary power to mitigate the loss-of-coolant accident coincident with loss of offsite power
- 24 months testing to demonstrate overall EDG unit design capability
- 10-year testing to demonstrate that the trains of standby electric power are independent

EDG Reliability

Reliability of EDG was identified as being one of the main factors affecting the risk from SBO. Thus, attaining and maintaining high reliability of EDGs was a necessary input to the resolution of USI A-44.

- In 1977 Generic Safety Issue B-56 (Diesel Reliability) was initiated based on examination of LERs which indicated EDG reliability of .94 as compared to goals of .99
- NRR rewarded a contract to University of Dayton Research Institute to identify more significant causes of EDG failures (reported in NUREG/CR-0660)
- In 1980 NRR recommended back fitting of RG 1.108 EDG testing frequency and associated failure reporting requirements to all operating plants as well as the implementation of the NUREG/CR-0660 recommended remedial actions at all operating plants as a final action to resolve GSI B-56

EDG Reliability

As a result, Tables 4.8.1.1.3, "Reports," and 4.8-1, "Diesel generator Test Schedule," were added to the TS. The test schedule was as follows:

- if the number of failures in the last 20 tests were one or less then the test frequency should be once per 31 days
- If the number of failures in the last 20 tests were two or more then the test frequency should be reduced to 7days
- this test frequency will be maintained until 7 consecutive failures in the last twenty tests have been reduced to one or less

In 1982, DST (Systems Technology) with the assistance of DSI, DL and IE prepared an interim diesel generator reliability program for operating plants which established a reliability of .95 as a minimum desired reliability and .9 as the minimum acceptable level of reliability and required additional actions based on number of failures in the last 20 test etc.

EDG Reliability

In 1984, as part of the technical evaluation of USI A-44,the staff issued Generic letter 84-15 and provided an example of EDG performance TS. The following items were requested from the licensees:

- To describe current program to avoid fast cold starts and reduce unnecessary testing
- Furnish current EDG reliability data
- Description of EDG reliability program

EDG Reliability

Until 1986 the reliability was calculated as a point estimate (number of failures/total number of starts) per RG 1.108

Per NSAC/108 the reliability was redefined as:

- EDG reliability=start reliability x load run reliability

Where Start reliability=number of successful starts/total number of valid demands and

Load run reliability=number of successful load runs/total number of load runs

EDG Reliability

The SBO rule was issued in 1988

In the implementation of SBO rule, the licensees were given the option to pick EDG reliability of .95 or .975. However, the SBO rule did not require the licensees to monitor and maintain these reliability values

The staff felt there was no realistic possibility of demonstrating that it had or it had not been met for any plant's EDG with the current failure rate data at that time

GSI B-56 was still not resolved

To resolve GSI B-56, the staff proposed generic letter 10 CFR 50.54(f) and revision to RG 1.9

The proposed revision to RG 1.9 would consolidate guidance on EDGs previously provided in RG.1.9, Rev. 2, and GL 84-15 in to a single guide. In addition, the guide added sections on EDG reliability monitoring including elements of EDG reliability program as well as the trigger values at which the action must be taken by the licensees.

EDG Reliability

In SECY-90-340 (1991) the commission disapproved the generic letter and the provisions of 10 CFR 50.54(f) as a vehicle for imposing requirements or securing enforceable commitments from licensees to address GSI B-56 and stated that this issue should be addressed thru rulemaking

The Commission endorsed a results oriented approach consistent with the MR and directed the staff to amend the SBO rule 50.63 and revise RG 1.9

The staff prepared the package which contained the revised rule that discusses the monitoring of EDG performance and related enforcement action

EDG Reliability

-- Monitoring Approach--

Early warning report (3 failures in the last 20 demands) – Notify the NRC in writing

Problem EDG (4 failures in the last 25 demands) – Notify NRC and subject the EDG to accelerated testing until 7 consecutive failure free tests are achieved

Double trigger (5/50 and 8/100 demands for .95 target, 4/50 and 5/100 for .975 target) occurrence is evidence of not meeting SBO selected EDG reliability target and the licensee is in noncompliance with 50.63. The Commission also asked the staff to describe the enforcement actions that the staff would take. The staff proposed the following.

Upon occurrence of the double trigger the licensee will:

- Implement appropriate corrective action
- Notify NRC operations center within 24 hours
- If restoration of EDG has not been restored within 30 days send a written report for this condition, the basis on which the EDG is operable and a description and schedule for corrective action to restore EDG reliability to assumed values

EDG Reliability

As a consequence of continuing ACRS concerns on the use of trigger values, in a letter to the Chairman, ACRS argued that the proposed rule amendment was unnecessary to ensure adequate EDG reliability and that the EDG reliability was generally good

Some members of ACRS disagreed and recommended that SBO rule should be issued for public comments

EDG Reliability

In 1990s the reliability of EDG significantly improved

Finally in SECY-93-044 (1993) the Commission stated that in view of the industry-wide average reliability EDG of .98, the Commission believed a rule was not necessary

As part of the resolution of GSI B-56, the Commission approved Option 4 as recommended by the staff

In Option 4, the staff recommended that licensee adopt the accelerated testing provisions of Improved Technical Specifications with an option to relocate accelerated testing requirements for EDGs from TS to the maintenance program after the maintenance rule (MR) goes in to effect in 1996

After further consideration, the staff decided that it was not necessary to await the effective date of the MR and to relocate the accelerated testing requirements to the maintenance program

EDG Reliability

The staff issued GL 94-01 with guidance for implementing a line – item TS improvement to remove accelerated testing and special reporting requirements for EDGs from the plant TSs or from docketed commitments. However, the licensees would continue to comply with the provisions of 10 CFR 50.72 and 50.73 to notify NRC and report EDG failures.

The staff's approval of this option was contingent upon a commitment to implement within 90 days of a license amendment a maintenance program for monitoring and maintaining EDG performance in accordance with the provisions of 50.65 and the guidance of RG 1.160, "Monitoring the Effectiveness of Maintenance at nuclear power Plants."

Subsequently, utilities docketed commitments to maintain their selected target reliability values (.95 or .975). Those values are being used as a goal or as a performance criterion for EDG reliability under the MR

EDG Reliability

EDGs are required to be handled under 10 CFR 50.65(a)(1) where they are subject to monitoring against licensee-established goals or under 10 CFR 50.65(a)(2) where they are subject to monitoring against licensee-established performance criteria

All EDGs under 10 CFR 50.65(a)(1) and (a)(2)) are subject to the requirements of 10 CFR 50.65(a)(3), including (1) periodic evaluation, (2) balancing reliability and unavailability, and (3) assessing the impact on plant safety of taking equipment out of service

EDG Unavailability

When SBO rule was developed in 1980s, EDG unavailability was estimated to be 0.007 which was significantly less than the EDG failure rates. Therefore, the SBO rule did not explicitly addressed maintenance unavailability, but emphasized the importance of reliable EDGs.

The operating data then showed an improvement in EDG reliability but an increase in unavailability due to maintenance, a significant portion of which due to routinely schedule maintenance

In 1991, the NRC staff reviewed EDG performance during actual demands. They found that in 5 of 128 demands the EDG did not function because it was out of service for maintenance. This value of 5/128 demands represents an unavailability due to time out-of-service for maintenance of .04 versus .007 previously used in developing the SBO rule.

EDG unavailability due to testing and maintenance was also estimated using out-of-service data over two years from June 1990 to May 1992, provided by NRC regional offices which reported EDG unavailability due to maintenance and testing of .017 during operation and 0.12 during shutdown

EDG Unavailability

EDG unavailability is also being monitored under the MR

Section (a)(3) of the MR rule requires that licensees make adjustments where necessary to ensure that the objective of preventing failures thru maintenance is appropriately balanced against the objective of minimizing unavailability due to monitoring or preventive maintenance i.e., licensees must periodically balance unavailability and reliability of the EDGs and assess the impact of removing EDGs from service on overall plant safety must also be performed.

Therefore, plant specific EDG unavailability should be monitored as goals under 10 CFR 50.65(a)(1) or established as performance criteria under the plant's preventive maintenance program under 10 CFR 50.639a)(2) taking in to the objective of 10 CFR 50.65(a)(3)

Emergency diesel generator unavailability values that were assumed in plant-specific individual plant examination (IPE) analyses should be compared to the plant-specific emergency diesel generator unavailability data regularly monitored and reported as industry-wide plant performance information. These values could also be used as the basis for a goal or performance criterion under the maintenance rule.

EDG ALLOWED OUTAGE TIME (AOT) EXTENSION

The NRC has been granting AOT extensions for EDGs to perform on-line preventive maintenance. This provides the licensees flexibility for performing various EDG maintenance and repair activities during power operation. it also reduces plant refueling outage duration. However, the AOT extensions for EDGs are granted to those licensees who have installed a qualified alternate ac (AAC) source credited for station blackout events which can be substituted for an inoperable EDG in the event of a loss of offsite power (LOOP).

The staff has also allowed BWR licensees to use the Division III diesel generator (high pressure core spray pump (HPCS) diesel generator) as an AAC power source to power safe shutdown loads if a cross-connect capability is provided so that the HPCS diesel generator can be cross-connected to either Division I or Division II ac buses to provide power in the event of a LOOP when one EDG is in the extended outage and the other EDG becomes unavailable. This cross-connection is generally accomplished within two hours.

The staff has required that in order for a HPCS diesel generator to be qualified as an AAC source, it must be free from other required safety functions (should not be relied upon as an station blackout mitigation system).

EDG ALLOWED OUTAGE TIME (AOT) EXTENSION

Some licensees have installed a commercial-grade diesel generator capable of supplying power to, as a minimum, the required safe-shutdown loads on the EDG train removed from service for the maintenance outage.

The staff evaluates each licensee's request for EDG AOT extension from a deterministic and probabilistic risk assessment (PRA) aspect. From a PRA perspective the licensee must demonstrate that the plant risk is low. From a deterministic perspective, the following compensatory measures are required to be implemented before entering the extended outage (after the august 14, 2003, grid event, the staff has required that all compensatory measures be included as regulatory commitments):

- The AAC power source or equivalent will be available as a backup to the inoperable EDG. After entering the extended AOT, the AAC source will be verified available every 8 hours and treated as protected equipment.
- The scheduling of EDG preplanned maintenance will be avoided during severe weather (tornado, thunderstorm, or ice storm conditions) or if grid stress conditions are high or forecasted to be high.
- The system load dispatcher will be contacted once per day to ensure no significant grid perturbations are expected during the extended. the system load dispatcher should inform the plant operator if conditions change during extended AOT such that unacceptable voltage would occur following a unit trip.

EDG ALLOWED OUTAGE TIME (AOT) EXTENSION

- Component testing or maintenance of safety systems and important non-safety equipment Including offsite power systems (auxiliary and startup transformers) that increases the likelihood of a plant transient or loop will be avoided. In addition, no discretionary switchyard maintenance will be allowed.

Technical specification requirements of verification that the required systems, subsystems, trains, components, and devices that depend on the remaining EDG(s) are operable and positive measures will be provided to preclude subsequent testing or maintenance activities on these systems, subsystems, trains, components, and devices.

Steam-driven emergency feed-water pump will be controlled as protected equipment.

More Questions!!

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ven National Laboratory

January 19, 2011

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Turbine Missiles – Explained

George Georgiev, Sr. Materials Engineer
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Office of New Reactors

Turbine Generator (TG) System Description

The TG does not perform or support any safety-related function, and thus, has no safety design basis.

The TG is, however, a potential source of high energy missiles that could damage safety-related equipment or structures.

Therefore, the turbine needs to be designed to minimize the possibility of failure of a turbine blade or rotor.

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Regulatory Basis

General Design Criterion (GDC) 4 states that structures, systems and components (SSCs) important to safety shall be appropriately protected against environmental and dynamic effects, including the effects of missiles, that may result from equipment failure.

Turbine rotors have large masses, rotate at relatively high speeds during operation and, therefore, failure of a rotor may result in the generation of high-energy missiles which may inflict damage on SSCs.

To satisfy GDC 4, turbine rotor integrity must be maintained to minimize the probability of turbine rotor failure.

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NRC Guidance and Review Documents

Regulatory Guide (RG) 1.115, “Protection Against Low-Trajectory Turbine Missiles,” and Standard Review Plan (SRP) Section 3.5.1.3, “Turbine Missiles,” guide the evaluation of the effect of turbine missiles on public health and safety.

SRP Section 10.2.3, Revision 2, “Turbine Rotor Integrity,” provides guidance to achieve integrity of the turbine rotor and ensure that the turbine rotor materials have acceptable fracture toughness and elevated temperature properties to minimize the potential for failure.

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Probability of Damage from Turbine Missiles

The probability of unacceptable damage from turbine missiles is expressed as the product of:

The probability of turbine missile generation resulting in the ejection of turbine blades (or internal structure) fragments through the turbine casing, (P_1)

The probability of ejected missiles perforating intervening barriers and striking safety-related SSCs, (P_2)

The probability of impacted SSCs failing to perform their safety functions, (P_3).

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Probability of Damage from Turbine Missiles (cont.)

Upon review of the operating experience of turbines and the NRC safety objectives in 1986, the NRC staff shifted its emphasis in the review of turbine missile issues from missile generation, strike, and damage probability, $P_1 x P_2 x P_3$, to the missile generation probability, P_1.

The minimum recommended reliability values of P_1 are less than 10^{-4} per reactor-year for favorably oriented turbines, and less than 10^{-5} per reactor-year for unfavorably oriented turbines.

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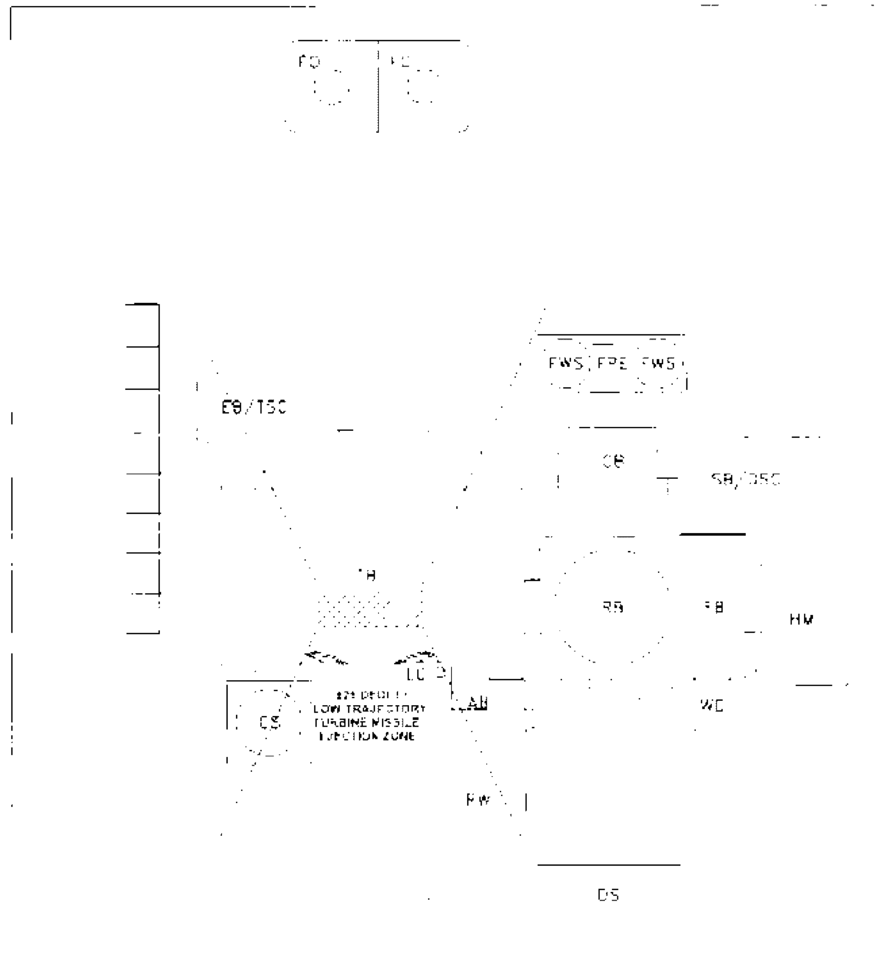

See Figure 1.1-1 for nomenclature

Figure 3.5-2. ESBWR Standard Plant Low-Trajectory Turbine Missile Strike Zone

TABLE 3.5.1.3-1

PROBABILITY OF TURBINE FAILURE RESULTING IN THE EJECTION OF TURBINE ROTOR (OR INTERNAL STRUCTURE) FRAGMENTS THROUGH THE TURBINE CASING (P_1) AND RECOMMENDED LICENSEE ACTIONS

Case	PROBABILITY PER YEAR FOR A FAVORABLY ORIENTED TURBINE	PROBABILITY PER YEAR FOR AN UNFAVORABLY ORIENTED TURBINE	RECOMMENDED LICENSEE ACTION
A	P_1 < 10[illegible]	P_1 < 10[illegible]	This condition represents the general minimum reliability requirement for loading the turbine and bringing the system on line.
B	10[illegible] < P_1 < 10[illegible]	10[illegible] < P_1 < 10[illegible]	If this condition is reached during operation, the turbine may be kept in service until the next scheduled outage, at which time the licensee must take action to reduce P_1 to meet the appropriate Case A criterion before returning the turbine to service.
C	10[illegible] < P_1 < 10[illegible]	10[illegible] < P_1 < 10[illegible]	If this condition is reached during operation, the turbine must be isolated from the steam supply within 60 days, at which time the licensee must take action to reduce P_1 to meet the appropriate Case A criterion before returning the turbine to service.
D	10[illegible] < P_1	10[illegible] < P_1	If this condition is reached during operation, the turbine must be isolated from the steam supply within 6 days, at which time the licensee must take action to reduce P_1 to meet the appropriate Case A criterion before returning the turbine to service.

Turbine Manufacturers in USA

- There are relatively few manufacturers that have supplied turbines to the nuclear power plant owners
- Westinghouse has the most turbines installed followed by General Electric
- Siemens and Alstom had refurbished low-pressure rotors in several nuclear power plants in USA

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Turbine Designs employed

In most of US operating plants the shrunk-on disk rotor were initially installed

For new plants all DCD applicants except ARIVA have proposed to use an integral forging rotor design

ARIVA has proposed to use an unique welded rotor design that Alstom employs in the fabrication of its turbines

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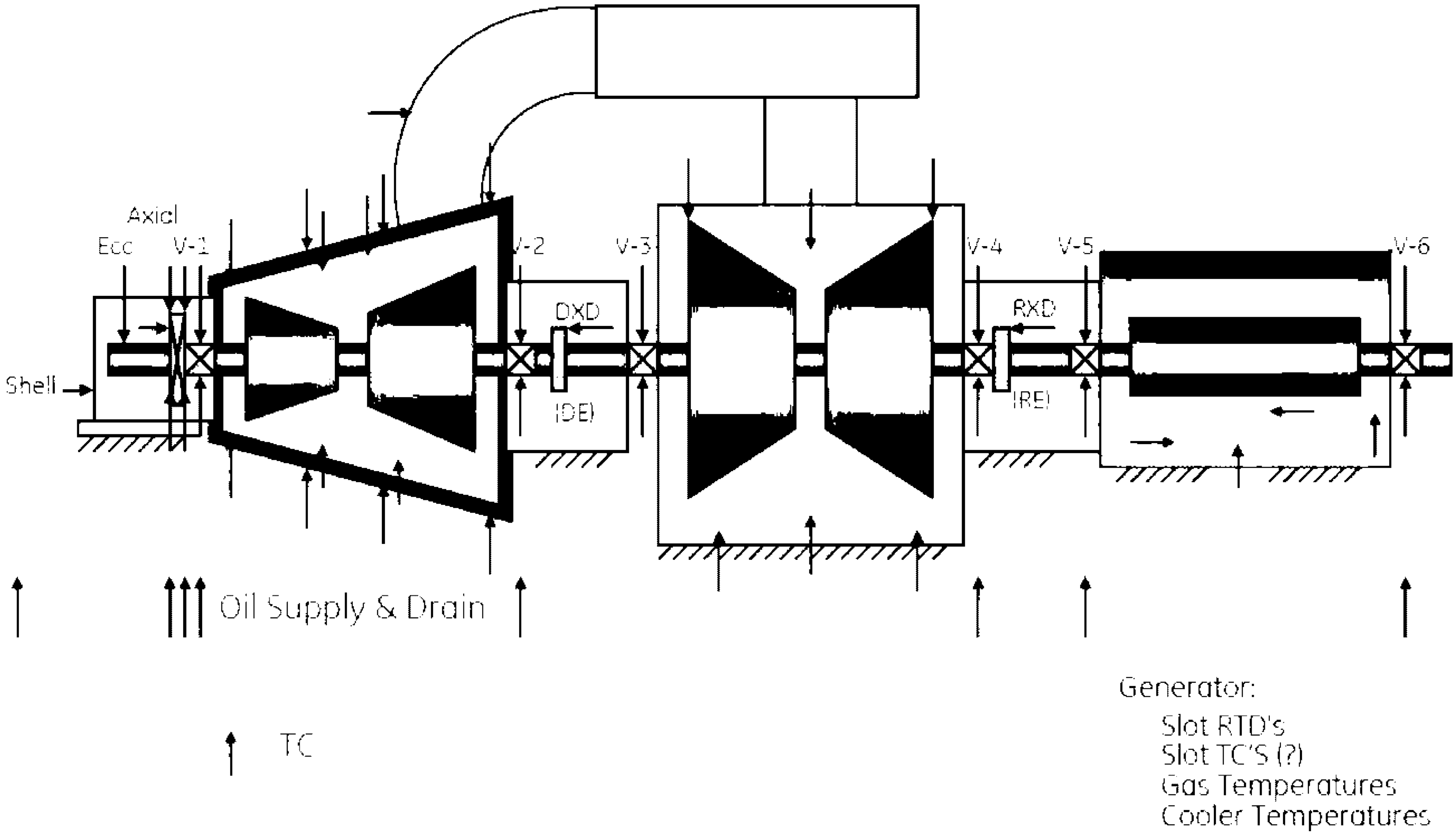
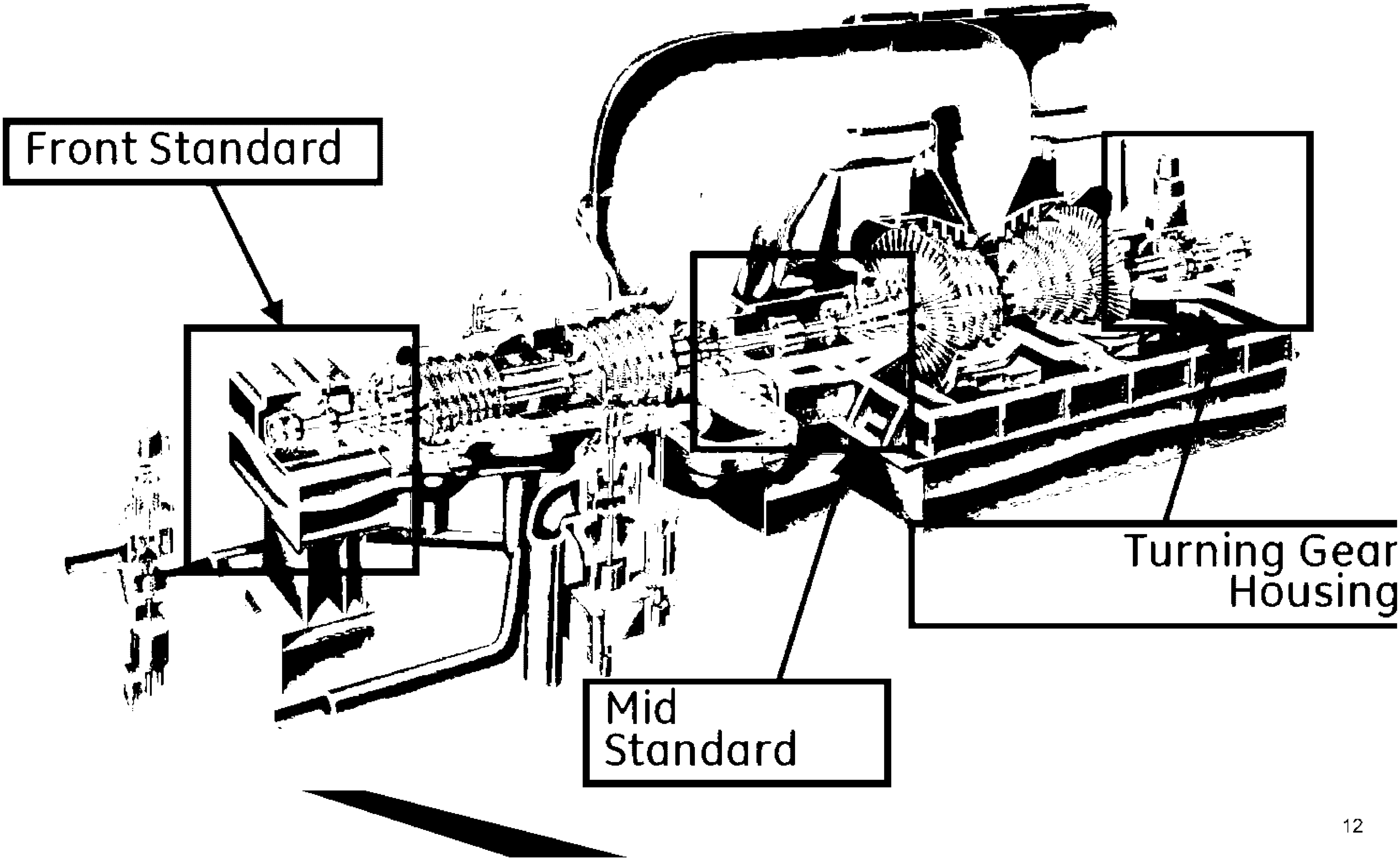


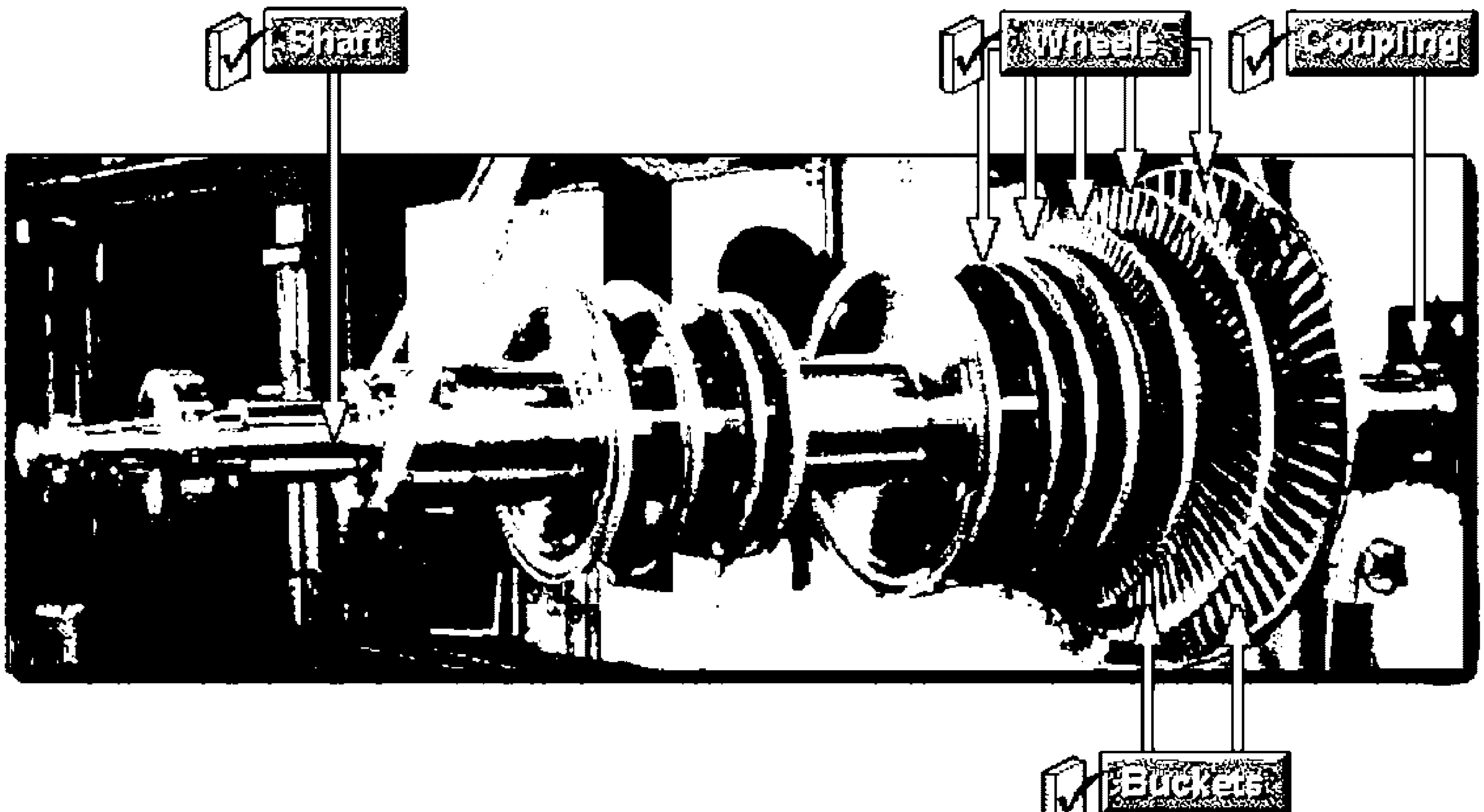
Steam Turbine Supervisory Instruments System

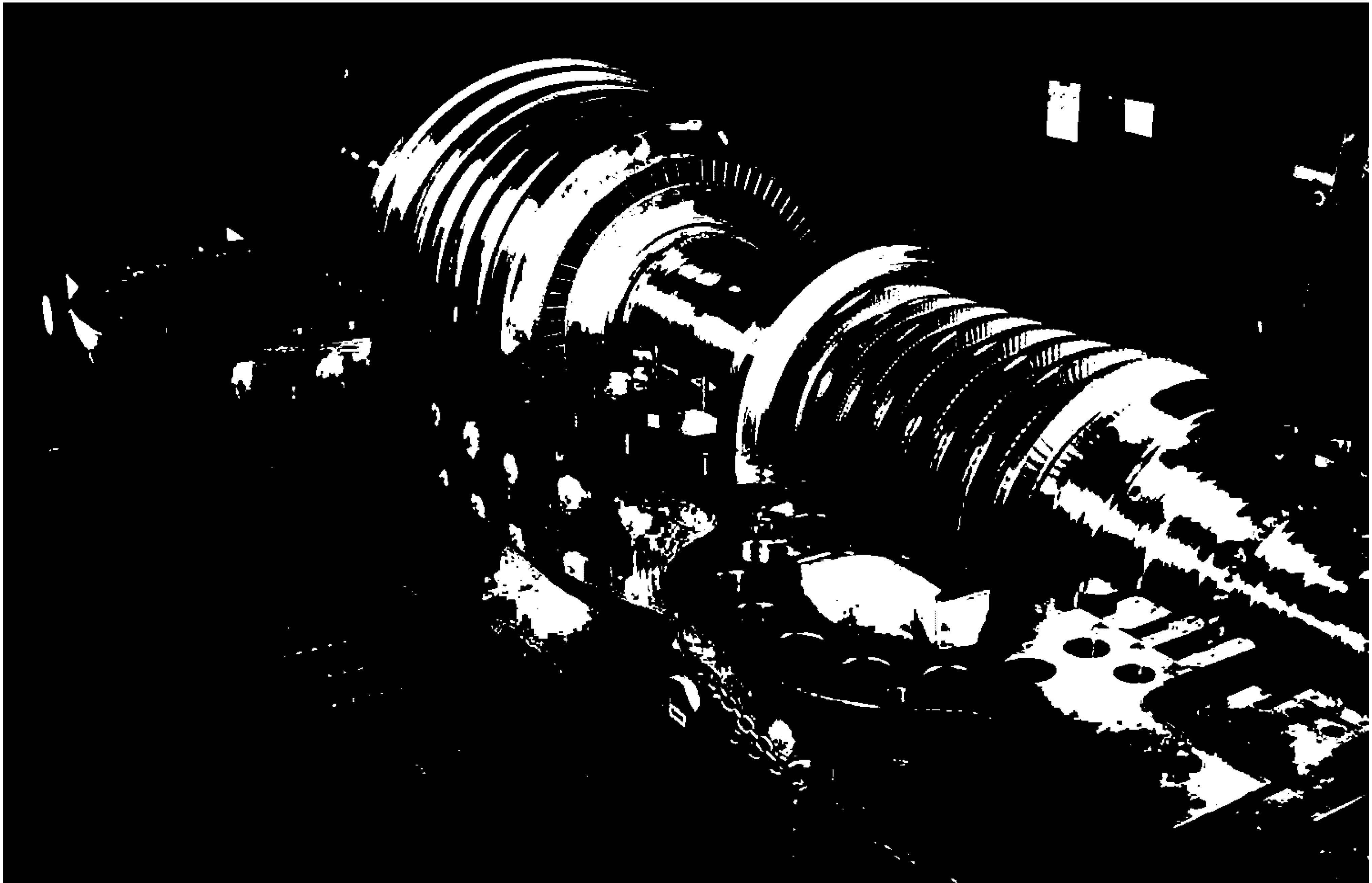
COMPONENTES

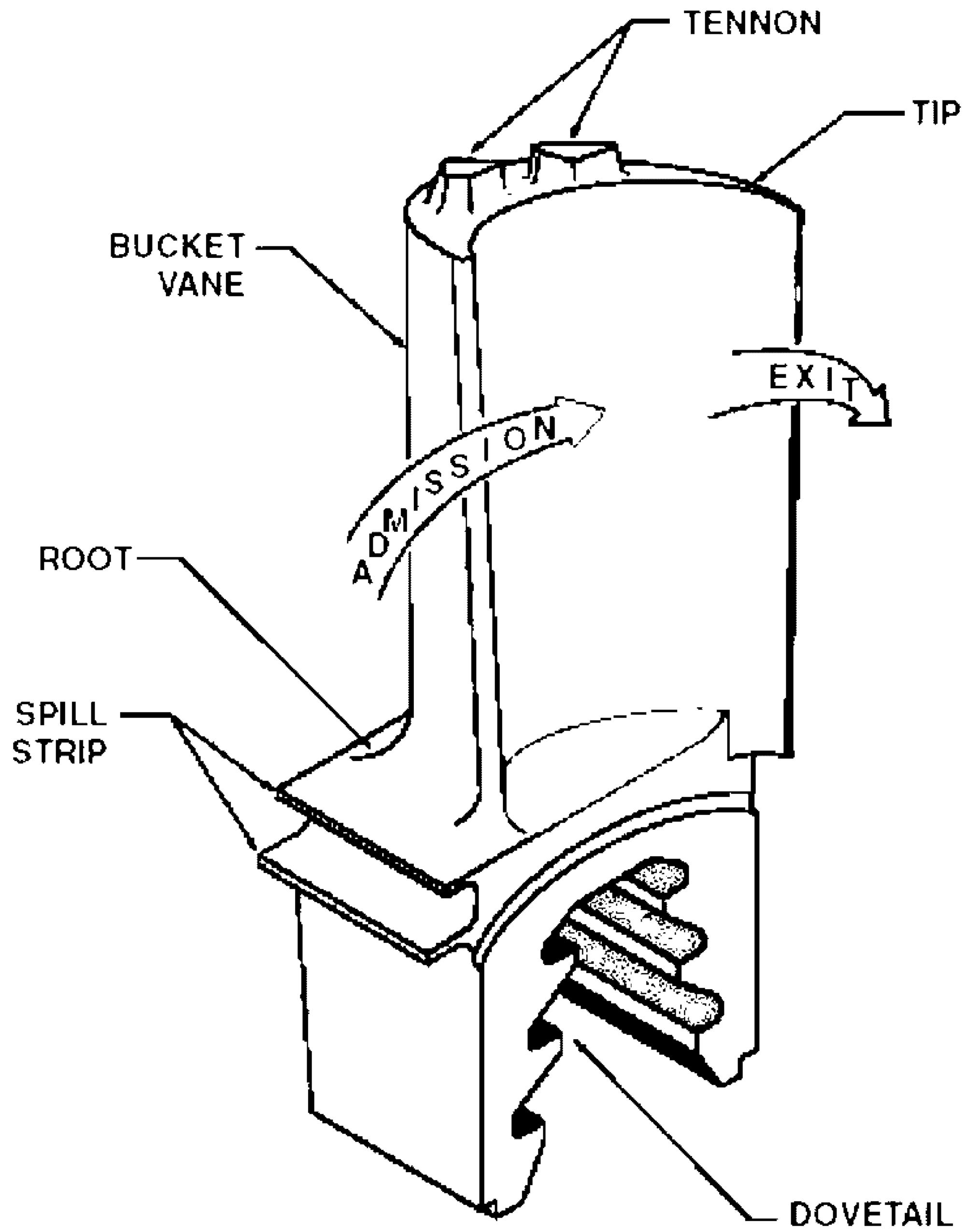


ROTORS

- A **typical rotor assembly** consists of a **shaft**, **wheels**, **buckets and couplings** which **transform the energy of the steam into rotating, mechanical energy.**





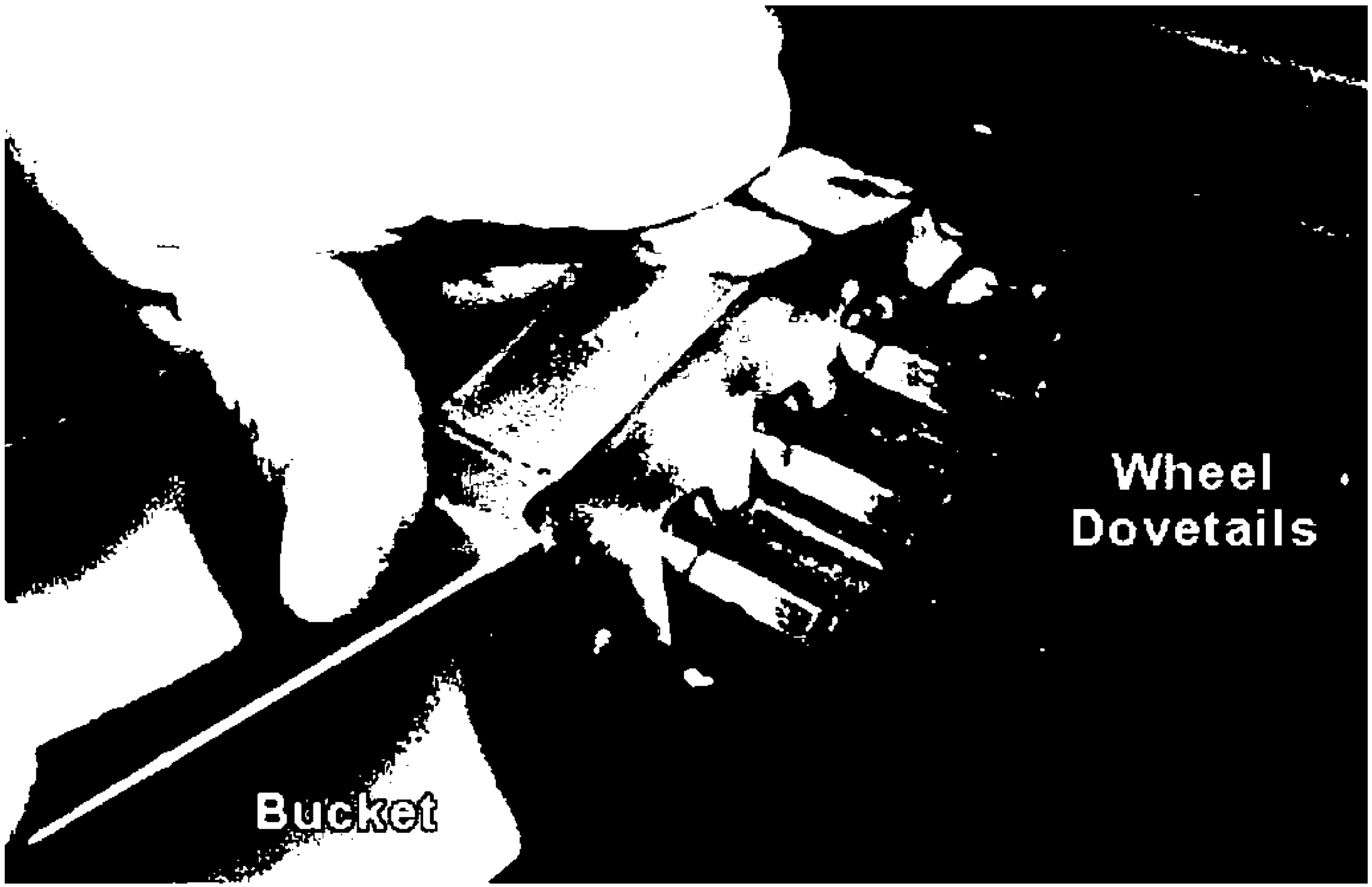


STEAM TURBINE
BUCKET DETAILS
(TYP. HP BUCKET)

WHEELS

Wheel dovetails

Machined surfaces of the outer circumference of a wheel to which buckets are securely fastened.



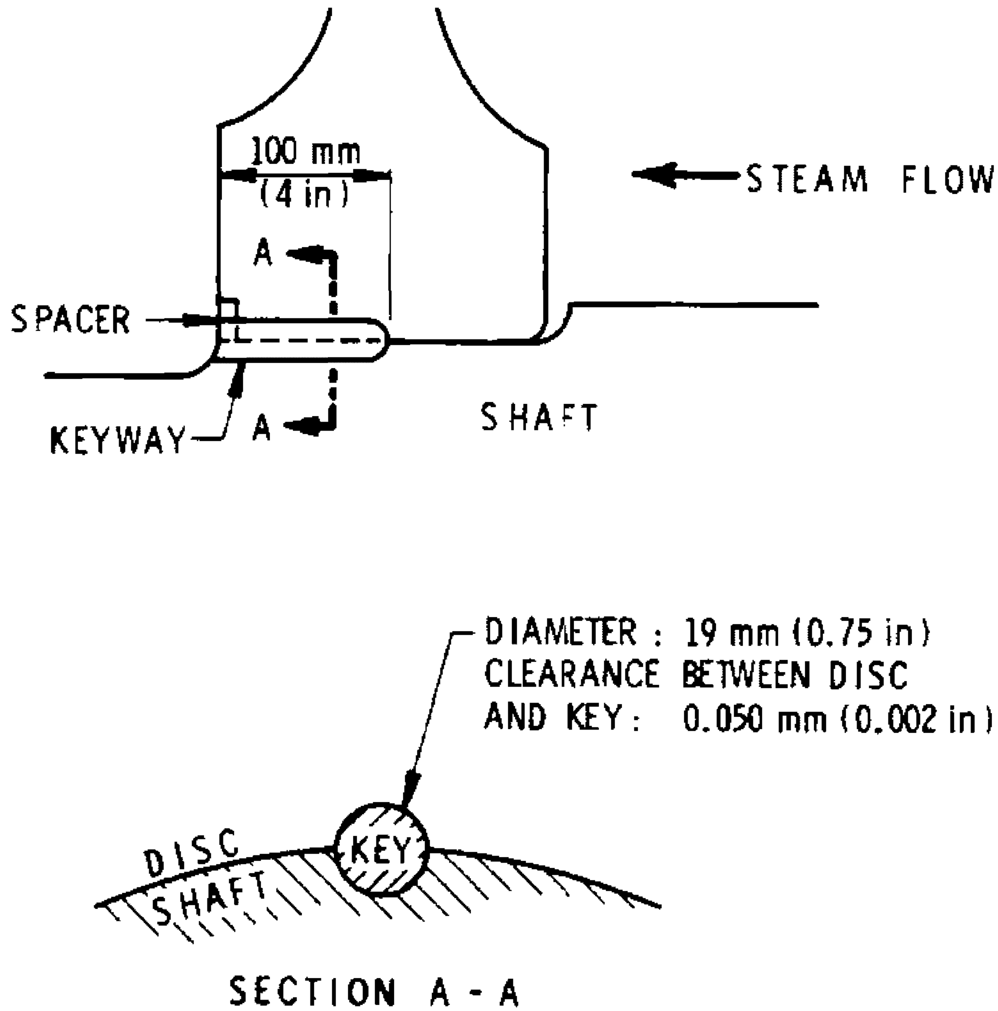


Figure 3-1. Schematic drawings of keyway design used by Westinghouse in shrunk-on disks of low-pressure rotors. (Source: EPRI NP-2429-LD, Vol. 6)

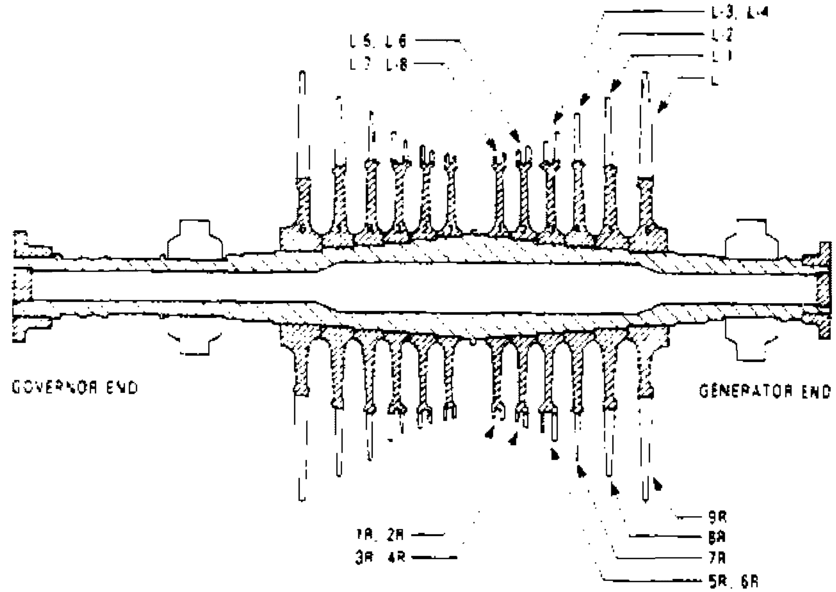


Figure 3-2. Schematic drawing of Westinghouse Model X-1 low-pressure rotor. All disks are shrunk on and the number of blade rows per disk is as indicated. Last-stage blades are 112 cm (44 in.) long. (Source: EPRI NP-2429-LD, Vol. 6)

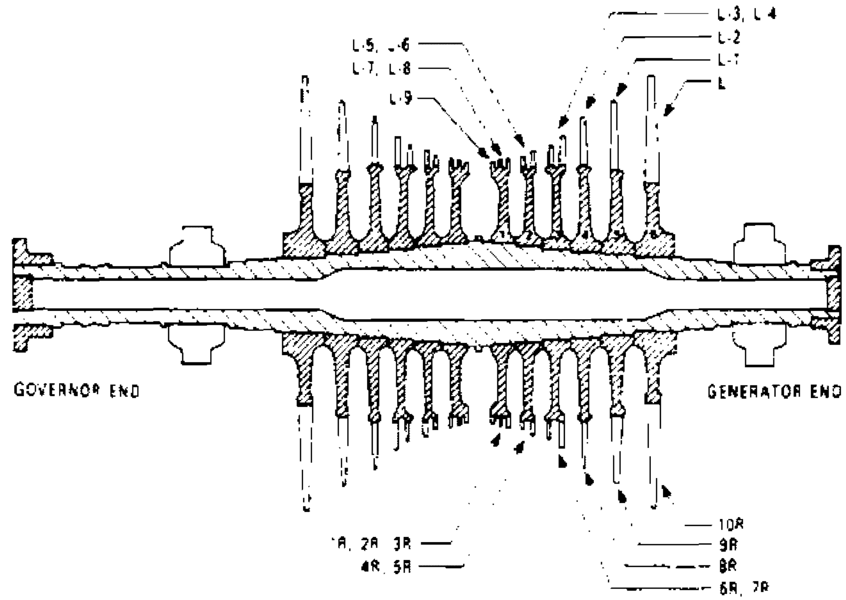
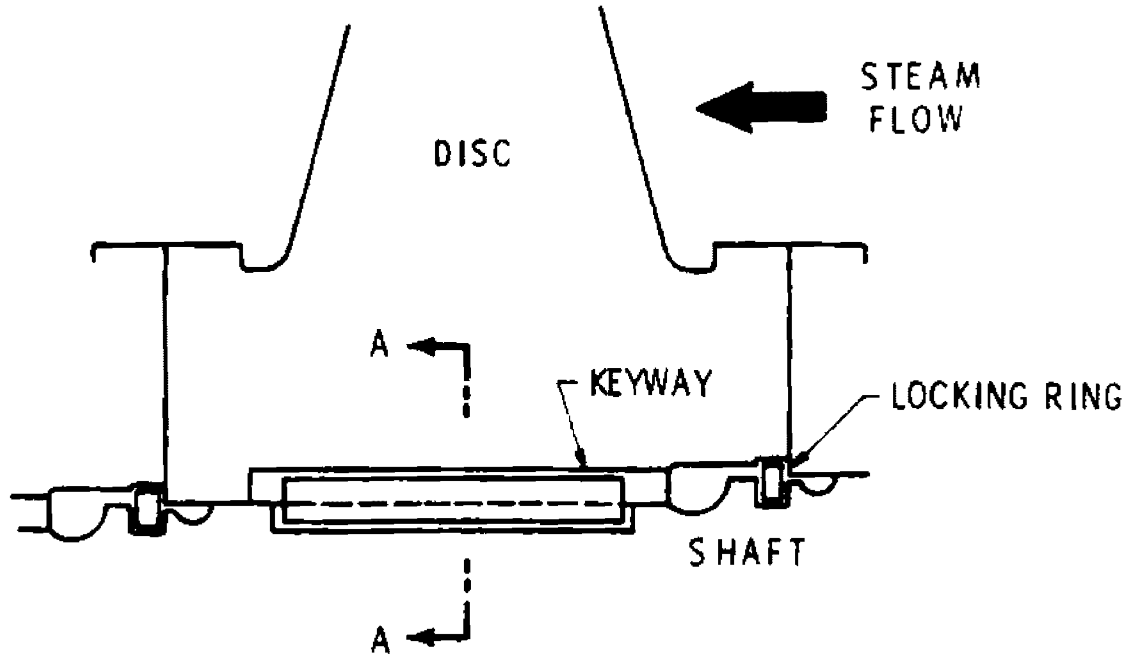


Figure 3-3. Schematic drawing of Westinghouse Model X-3 low-pressure rotor. All disks are shrunk on and the number of blade rows per disk is as indicated. Last-stage blades are 112 cm (44 in.) long. (Source: EPRI NP-2429-LD, Vol. 6)



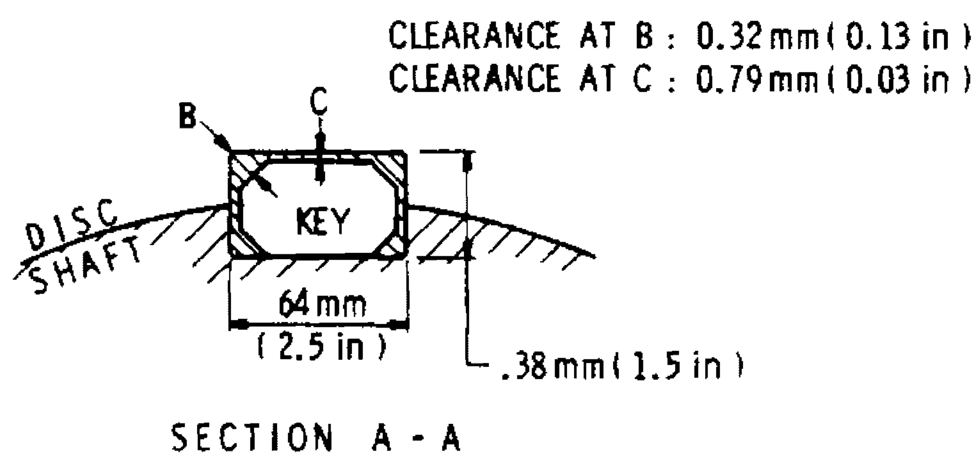


Figure 3-8. Schematic drawing of keyway design reportedly used by General Electric in shrunk on disks of low-pressure rotors.

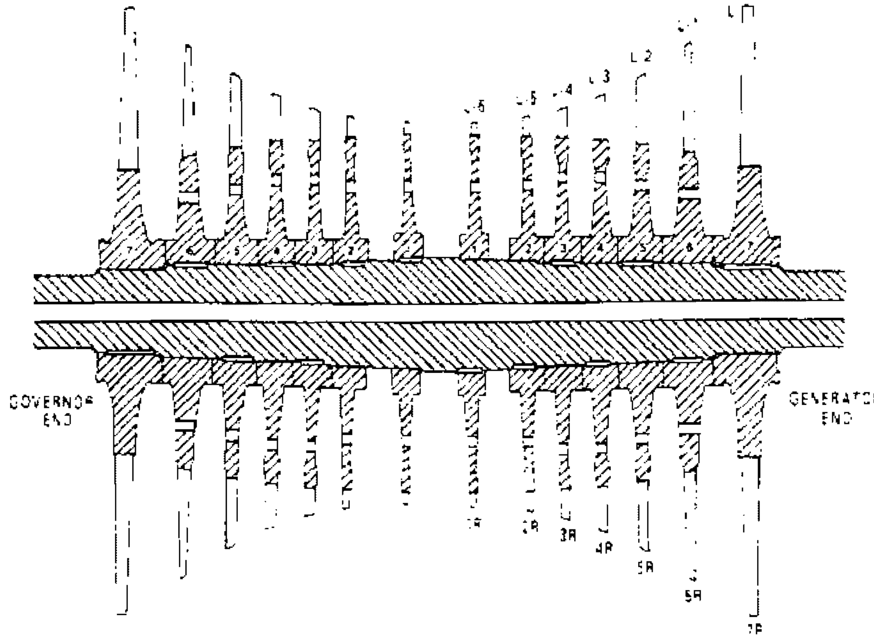


Figure 3-9. Schematic Drawing of General Electric Model Y-3 low-pressure rotor. All disks are shrunk on. Last-stage blades are 97 cm (38 in.) long. (Source: EPRI NP-2429-LD, Vol. 6)

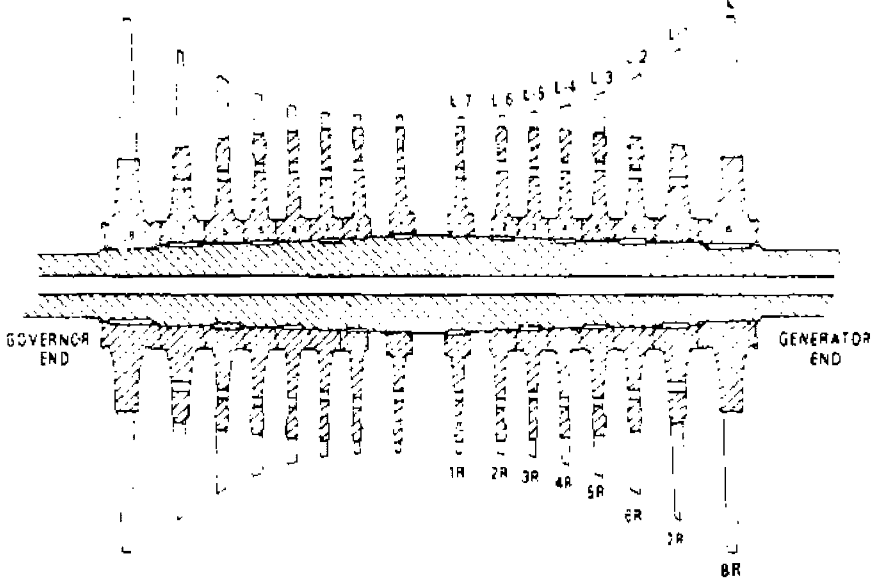


Figure 3-10. Schematic drawing of General Electric Model Y-1, Y-2, Y-4 and Y-5 low-pressure rotors. All disks are shrunk on. Model Y-1 last-stage blades are 97 cm (38 in.) long. Last-stage blades of Models Y-2, Y-4 and Y-5 are 109 cm (43 in.) long. (Source: EPRI NP-2429-LC, Vol. 6)

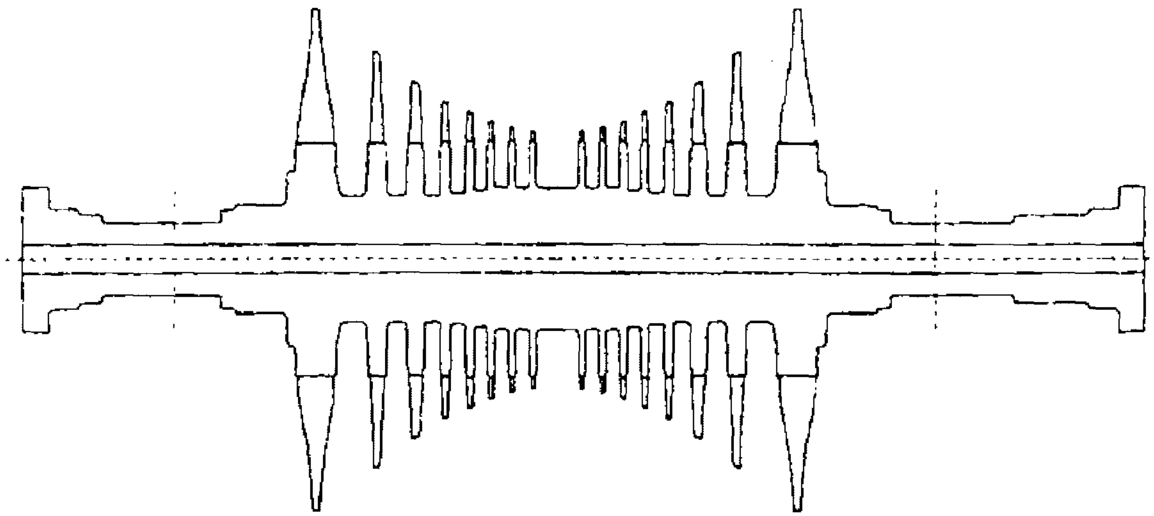


Figure 3-1 Typical Fully Integral Rotor Construction

How Ejection of Turbine Missiles Prevented

The ejection of turbine is prevented by following the turbine manufacturer's recommendations specified in the Turbine Maintenance Program

The NRC staff requires that the Turbine Maintenance program is submitted to the staff's review within three years after the plant is placed in operation

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How Ejection of Turbine Missiles Prevented (Cont.)

The turbine maintenance program specifies periodic in-service inspection of turbine subcomponents including rotors

The in-service inspection recommendations are based on actual as-build rotor material properties and actual preservice inspection results performed on the rotor prior to shipment to the site

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Information included in the Turbine Maintenance Program

- Probabilistic approach to evaluating P_1 that includes information on critical crack size, crack growth rate, rotor operating temperature and applied stress used in the evaluation model
- Numerical approach such as Monte-Carlo simulations and number of iterations required for reliable probability estimate

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Information included in the Turbine Maintenance Program (Cont.)

- Results of the preservice non-destructive examinations
- Rotor material mechanical test results including FATT and Charpy tests results
- Recommended turbine valve testing intervals
- Recommended in-service inspection intervals
- In general every 10 to 13 years in-service inspection is recommended for the rotor

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Status of Turbine Missile Generation Issue for New Reactors

The NRC staff had specified that a bounding analysis report exist assessing the probability of turbine missiles prior to approving COL application, if not the DCD applicant needs to provide an ITAAC requiring that the COL submit to the staff its Turbine Maintenance Program for review and approval prior to fuel load.

This process was communicated in two public meetings with the industry and other stake holders

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Status of Turbine Missile Generation Issue for New Reactors

All DCD applicants have provided the NRC staff with bounding analysis reports showing that they can meet the NRC recommended values for P_1

The bounding analysis reports also include assessment of various modes of failures such as ductile burst from destructive overspeed, high and low cycle fatigue and failure due to stress corrosion cracking

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Conclusion

Because all DCD applicants have provided bounding analysis reports showing that their turbines will meet the NRC requirements for P_1, it can be concluded that the turbine probability issue will not result in open items or otherwise impact the project licensing schedules

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Backup Slides

Illustrations for Cracks and Examples of evaluation methods

Monte Carlo Variables

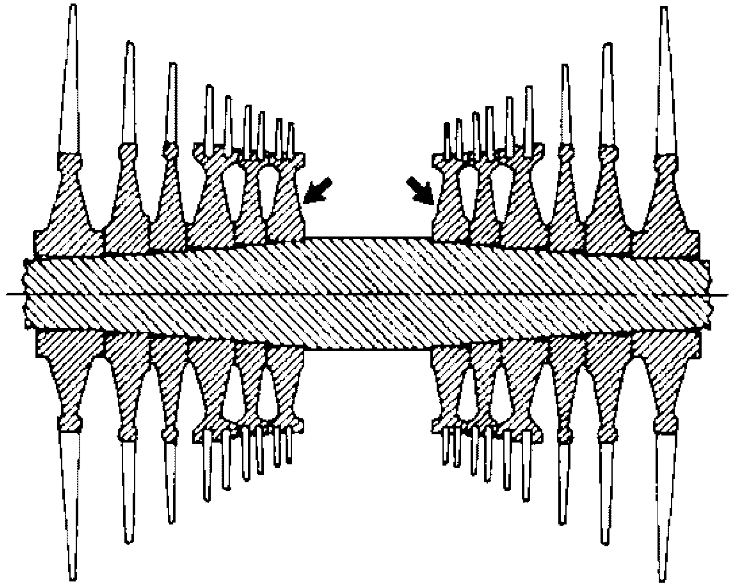


Figure 5-15 Schematic of Yankee Rowe LP rotor; arrows point out failed No.1 disks [8]

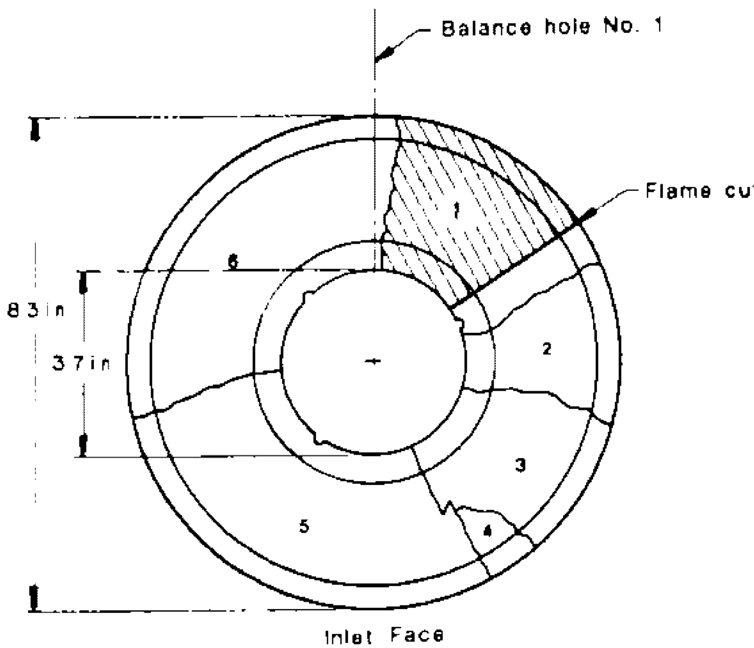
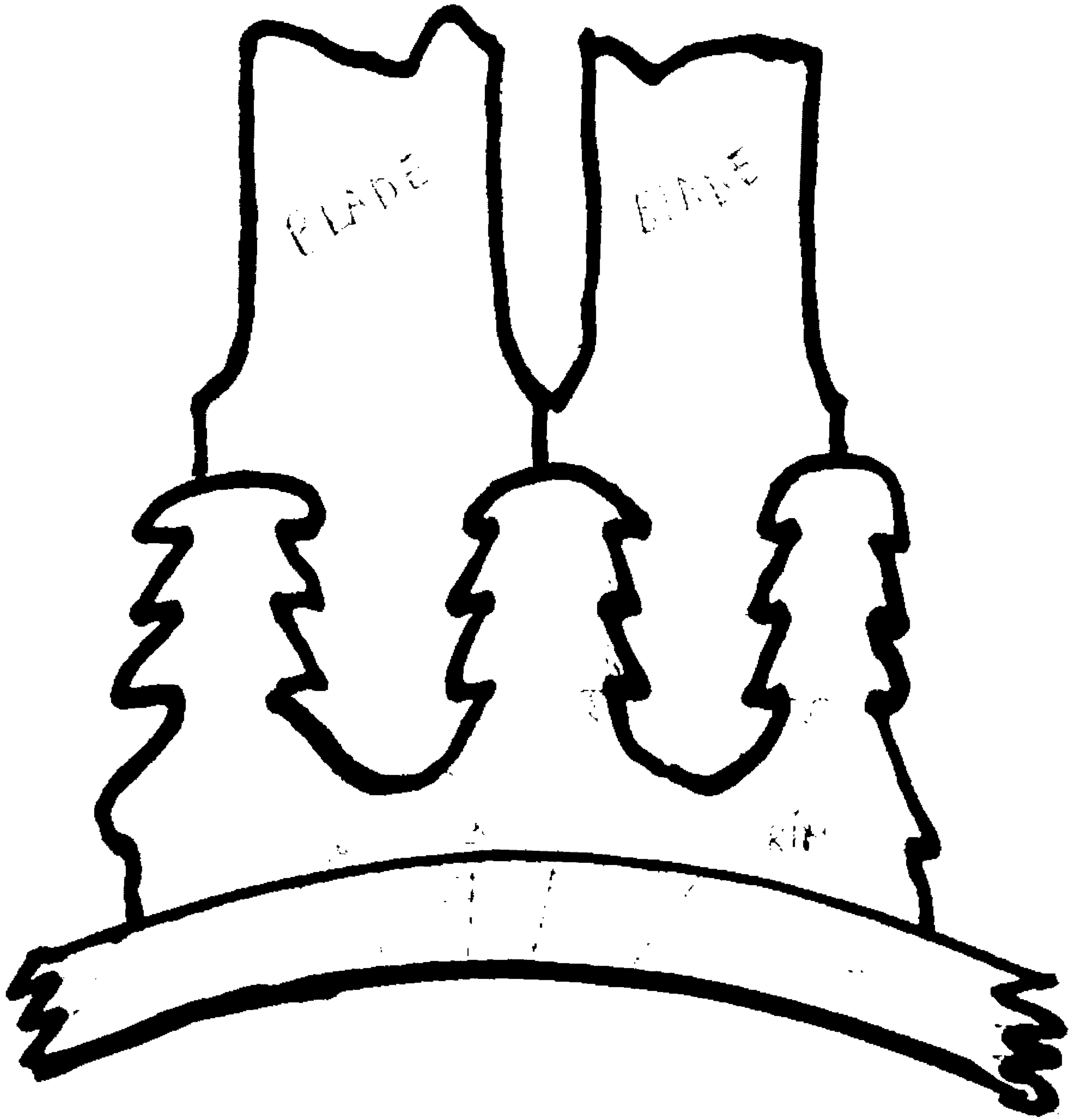


Figure 5-16 Diagram of Yankee Rowe failed generator-end no.1 disk (largest bore crack, 1.94" deep x 1.62" long, is at segments 5/6) [8]


BLADE
CRACKS

The basic edge-crack model is modified for use in the Westinghouse analysis to incorporate a flaw geometry factor, G, which accounts for both the gross shape of the flaw as well as the beneficial effects of crack branching and irregularities in the shape local to the crack tip. The resulting crack model is given in Equation 2-1 below.

$$a_{cr}^{*} = \frac{1}{1.21\pi}\left(\frac{K_{IC}}{\sigma}\right)^2$$

Equation 2-1

where:

a_{cr}^{*} is the nominal critical crack size accounting for shape parameter or branching factor
$a_{cr} = G \cdot a_{cr}^{*}$ is a larger, effective critical crack size that includes shape parameter and branching factor
K_{IC} is the material toughness
and, σ is the applied stress

Disks 8 and 9 These are Ni-Cr-Mo-V disks, with FATT values in the -135°F to 0°F range (Table 2). Therefore for normal operation these disks are in the upper shelf region. The following *Rolfe-Novak* upper-shelf relationship was used [8]:

$$\left(\frac{K_{Ic}}{\sigma_y}\right)^2 = 5\left(\frac{CVN}{\sigma_y} - 0.05\right) \qquad (7)$$

A lower bound K_{Ic} value of 181 ksi$\sqrt{\text{inch}}$ was estimated for the minimum reported CVN value of 70 ft-lbs and an average yield strength of 101 ksi (see Table 2).

Inputs for Monte Carlo variables

1. Scale factor for load/stresses (normal distribution)
2. Overspeed level (normal distribution)
3. Rotor startup temperature (normal distribution)
4. Rotor operating temperature (normal distribution)
5. Crack depth (normal distribution)
6. Crack ratio depth/length (normal distribution)
7. Yield strength (normal distribution)
8. Lower bound Fracture (normal distribution)
9. FATT (normal distribution)
10. Fracture Toughness (normal distribution)
11. Crack Initiation time (user-defined)
12. SCC Growth Rate Constant
13. SCC Growth Threshold (normal distribution)