"Materials Technology for Nuclear Power Plants : Research Needs and Opportunities"

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Materials Implications of the Upcoming Demand for Nuclear Power

	Near Term	Intermediate Term	Long Term
Plant Options	Relicence Existing Gen III Plants (40, 60 years)	Relicence Existing Gen III Plants (60, 80 years) Build New Gen III+ Plants	Relicence Existing Gen III+ Plants (80 years) Build New Gen IV Plants
Reactor Technologies	Existing BWR and PWR plants	Advanced BWR and PWR plants	Improved BWR and PWR plants, Supercritical Water Reactor, Gas Cooled Reactors, Lead and Sodium Cooled Reactors
Materials Technology Implications	Understand existing plant materials. Quantify time dependent materials properties	Extended understanding of traditional materials Validation of "improved materials" used to repair Gen III reactors	Material properties in "extended" conditions (temperature, environment, very long time etc.)

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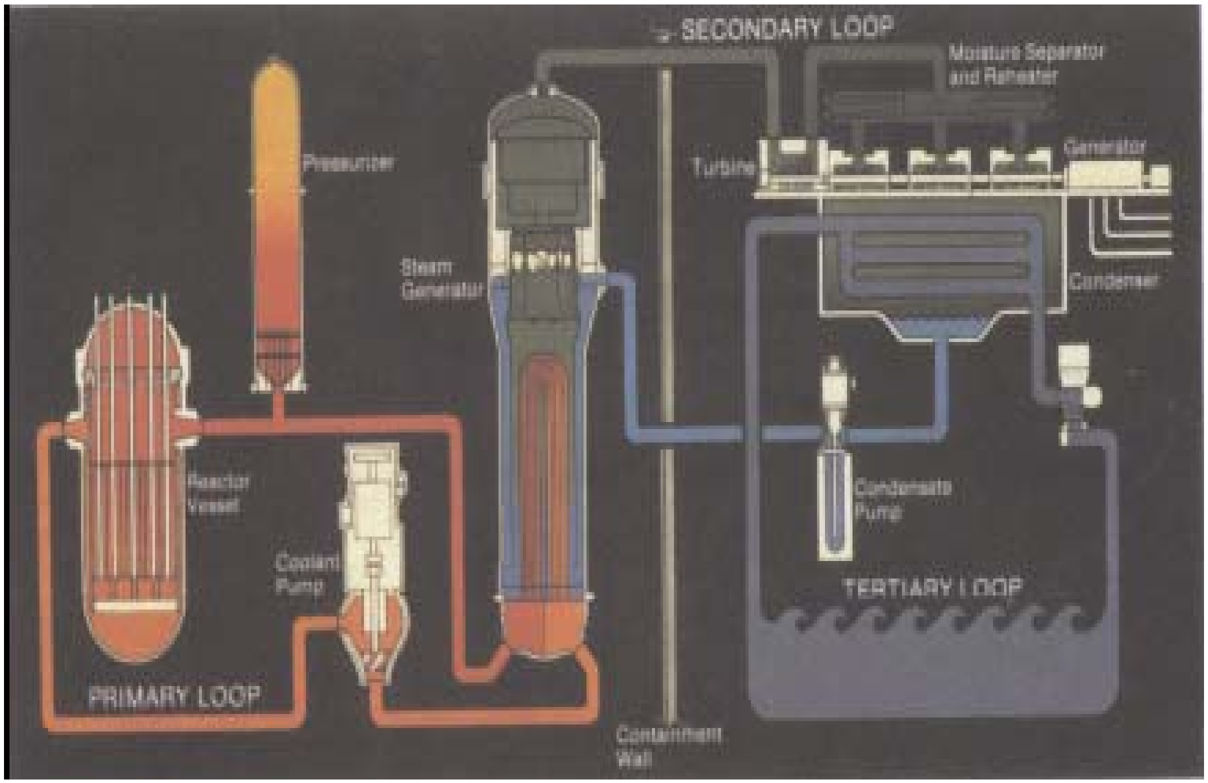
The Pressurized Water Reactor (PWR)



Two environments : Primary Side and Secondary Side

- **Primary Loop in Reactor**
- **Secondary Loop raises steam to drive turbine**

System is intentionally pressurized – gas pressure bubble in the electrically heated pressurizer.

Reactor Power is controlled by top inserted control rods and by B/Li chemistry additions to reactor coolant

Reactor Operating Conditions…
…..That are Imposed on Materials

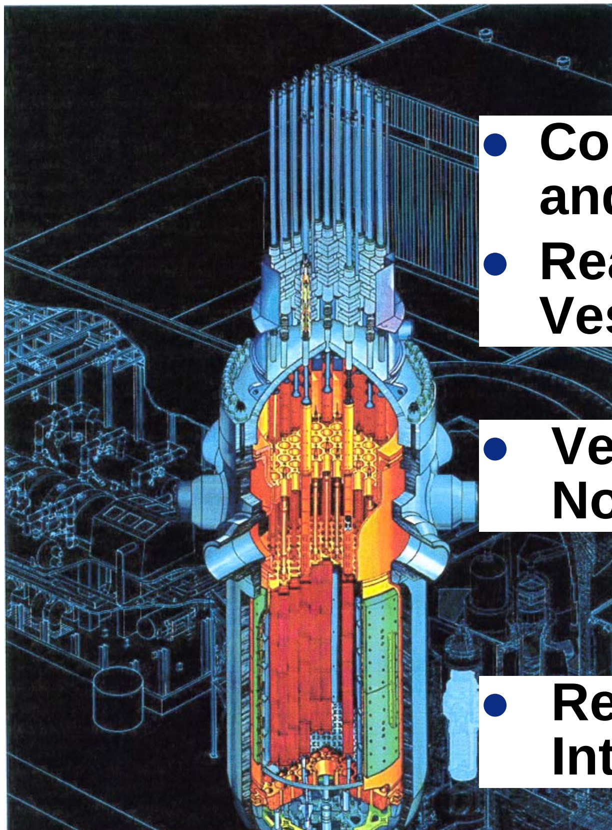
- **Coolant : water, with B:Li, and H additions, Ph~7.0,**
- **Coolant flow ~ 300,000 gals/minute, ~ 17 ft/sec**
- **Coolant operates at 150 atmospheres pressure**
- **Coolant T_{in} ~ 550F T_{out} ~ 620F**
- **Fast neutron flux ~ 10^{14} cm^2/sec, E >1 Mev**
- **Operating exposure: Fuel ~ 5 year, Plant ~ 60 years**

Reactor Materials

Ferritic Low Alloy steels	Reactor, Pressurizer, and Steam Generator Pressure Vessels, Steam Generator Tube Sheet, Large Bore Piping
Ni Base Alloys	RVH Penetrations, SG Tubing, Piping (D-M) Welds, Corrosion Resistant Cladding (by Weld Deposits) Fuel Assembly and Spring Applications
Stainless Steels	Reactor Internals (Barrel Plates, Formers, Bolts) Support Plates, Control Rods, Piping, Corrosion Resistant Cladding (by Weld Deposits)
Zr Alloys	Fuel Cladding, Fuel Grids, In Core Instrumentation Tubing

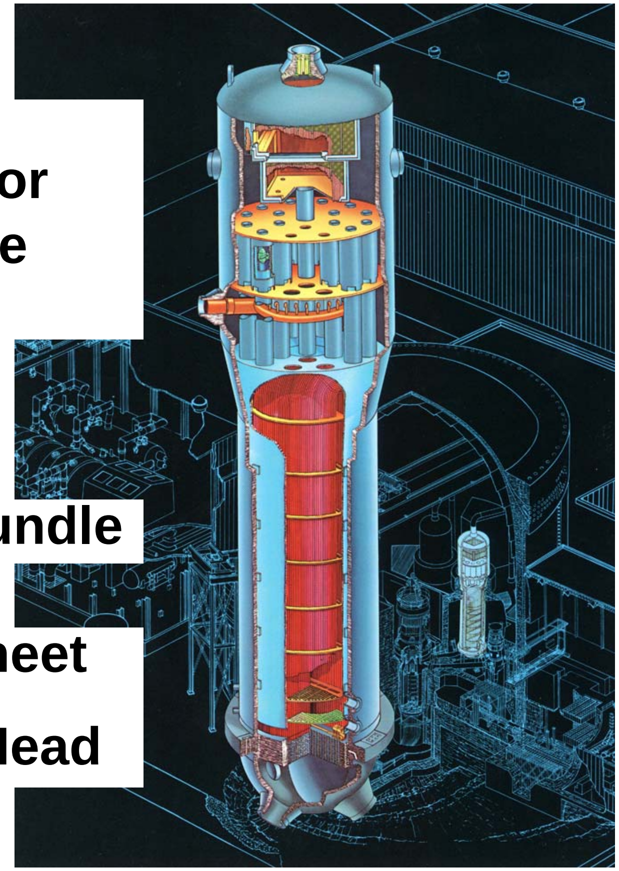
Assembly by Pinning, Welding and Bolting

Key Features of the PWR



- Control rods and Drives
- Reactor Vessel
- Vessel Nozzles
- Reactor Internals

PWR Reactor Vessel and Internals



- Steam Separator
- Pressure Vessel
- Tube Bundle
- Tube Sheet
- Lower Head

PWR Steam Generator

Materials Issues For Existing Plants

Ferritic Low Alloy steels	Irradiation Embrittlement →Updated PTS Rules, Resistance of Head Penetrations to SCC, Fatigue of Piping
Ni Base Alloys	SCC of Head Penetration Welds, SCC of Dissimilar Metal Welds in RV and Pressurizer Nozzles, SCC in SG Tubing, Protection of SG Tube Sheets,
Stainless Steels	Internals Hardening and Embrittlement, SCC of baffle Bolts, SCC of Welded Internals, Mitigation of Piping Welds, Thermal and Irradiation Embrittlement of CASS
Zr Alloys	Fuel Rod Leakage, Integrity of Welds, New Materials with Reduced Oxidation/Hydriding CRUD Formation Mitigation

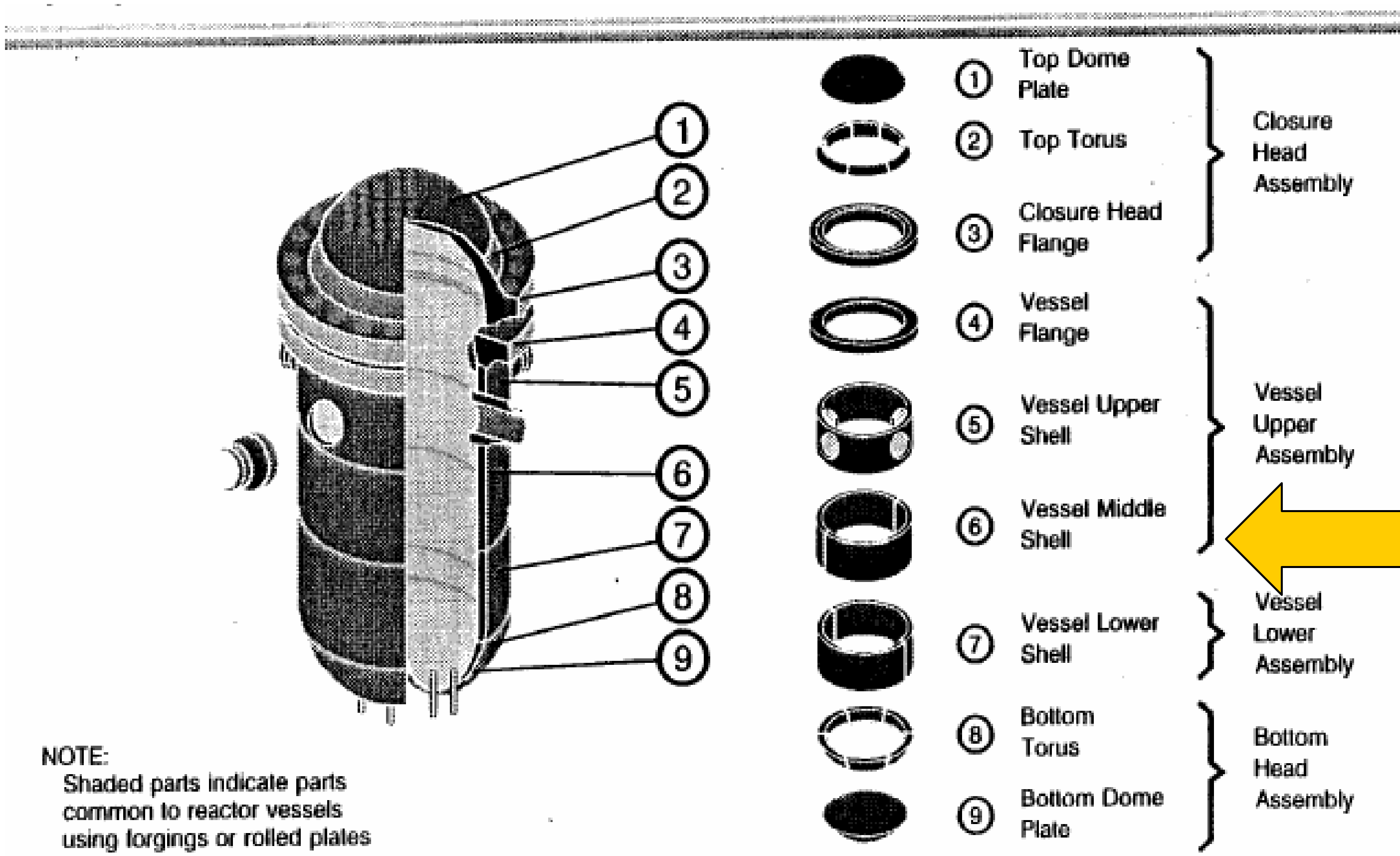
- ***The Nuclear Power Generation Industry Currently Manages All of These Issues to Keep Plants Operating at 90% of Capacity Factor***
- ***Technology Developments Must Be Able to Discriminate Between Materials Variants at These Levels for Long Time Behaviors***

Plant Relicensing

- Plants are Licensed to Operate by the NRC
- Licenses are held on a 20 year basis
- 2009-2013 Upcoming License Renewals Period for many US PWR Reactors
- Relicensing Application Must be Supported by Technical Data Demonstrating Safe Operating Capability
- Key Element of Relicensing plan is the Proposed Plant Specific, ***Inspection and Evaluation Program***

What to Inspect and Evaluate has been the subject of Concerted Industry activities via EPRI MRP program to Identify and Rank Most Probable Failures of Components

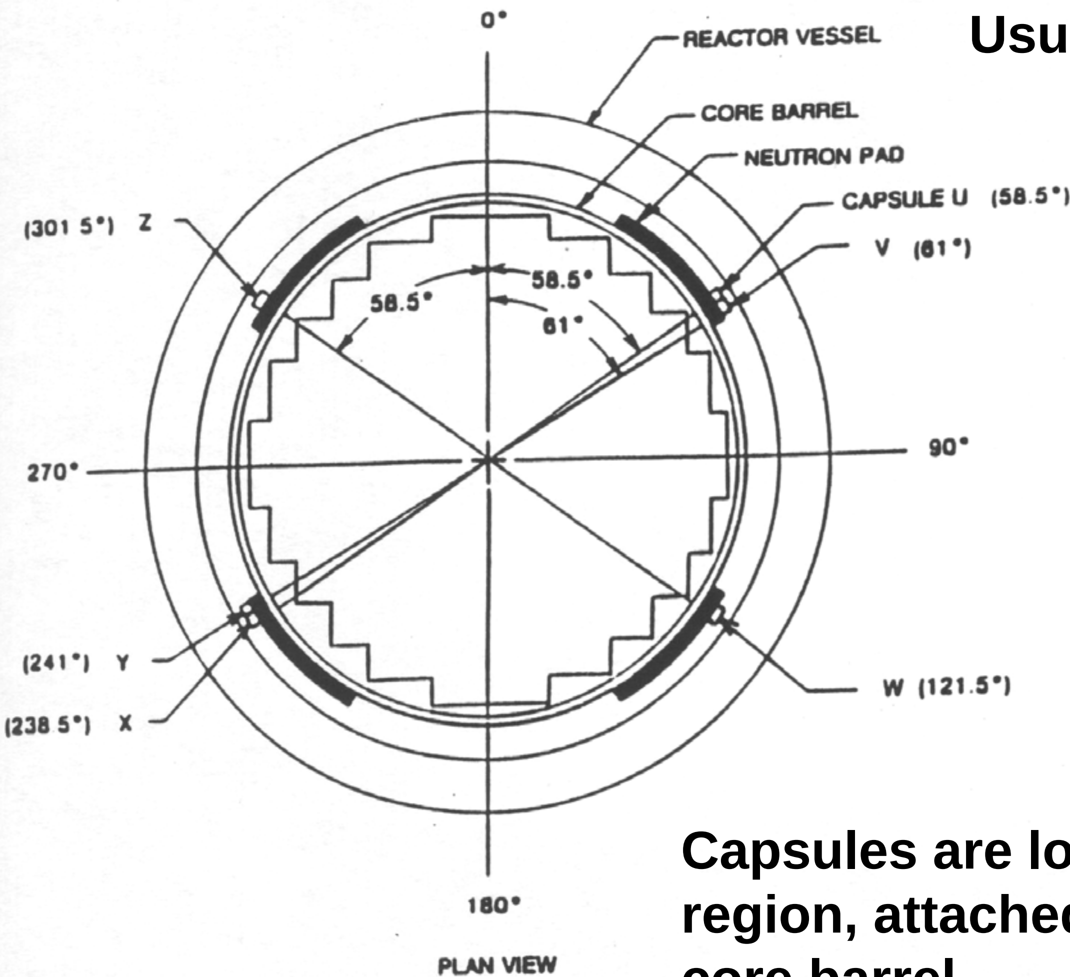
Reactor Containment Vessel Monitoring Against "Pressure Vessel Embrittlement"

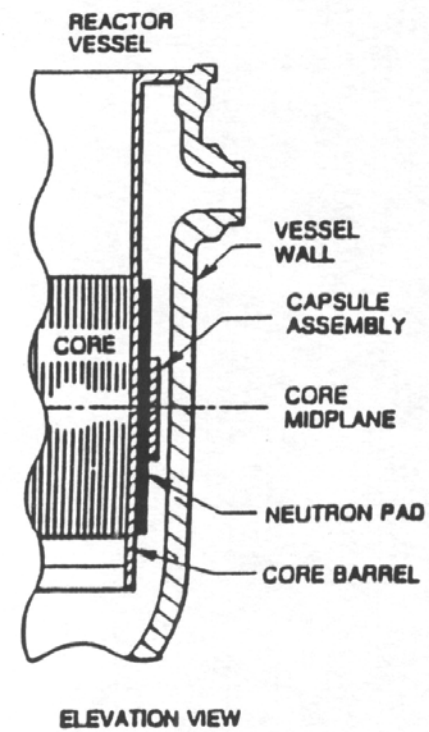


Key Region for Surveillance Programs so called "Beltline"

Typical Surveillance Capsule Locations

Usually 8 capsules per vessel





Capsules are locate in the highest intensity region, attached to the backside of the core barrel

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Direct Materials' Testing –
Ferritic RPV Steels Using Surveillance Capsules

- **Ferritic steels become embrittled in neutron and thermal environments**
- **These changes are manifested in :**
  - ➢ **Reduction in the toughness during ductile fracture**
  - ➢ **Tendency to brittle fracture at onsets at increasing temperatures**

Plant operations monitor vessel materials for embrittlement and feedback properties into analyses to support operations

- ➢**Surveillance capsule programs for vessels**
- ➢ **License requirement - Pressurized Thermal Shock (PTS)**
- ➢**Operational Limitations – heat up/cool down "curves"**

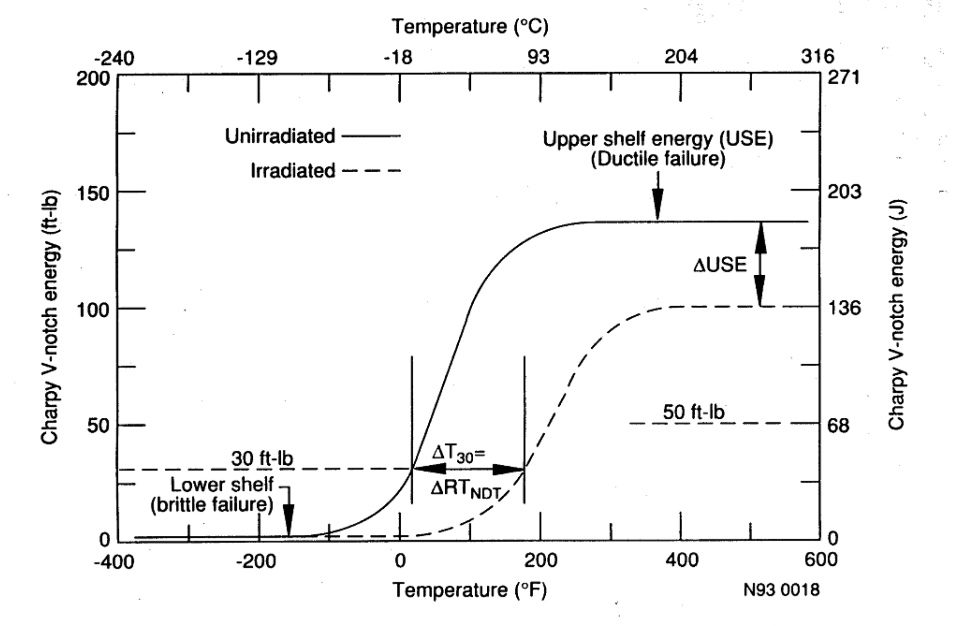

Figure 3-2. Charpy V-notch surveillance data, showing radiation embrittlement effects.

Acceleration Factor of Capsule Samples Leads Real Life Material

Fracture Behavior of Pressure Vessel Steels Continuing Issues and Research Needs

- Extrapolation of embrittlement curves to higher dpa for >80 year life
- Resolution of fast reactor data and extrapolation of surveillance capsule data to highest fluences
- Complete understanding of chemical effects and cross-interactions
- Full utilization of fracture toughness methodologies (c.f. Impact data)
- Projections to new steels etc.

Indirect Assessment of Plant Materials

- Recognize degradation has occurred but…
- …Extraction and testing of material is not feasible
- Need Indirect Methods
  - Generic Materials Degradation Rules
  - Systematic Analyses
  - Key Materials Properties
- Determine Most Likely Components Failure
- Build a Systematic Basis for Inspection and Subsequent Decision Making

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Relicensing Applications Based on "*Inspection and Evaluation" Approach*

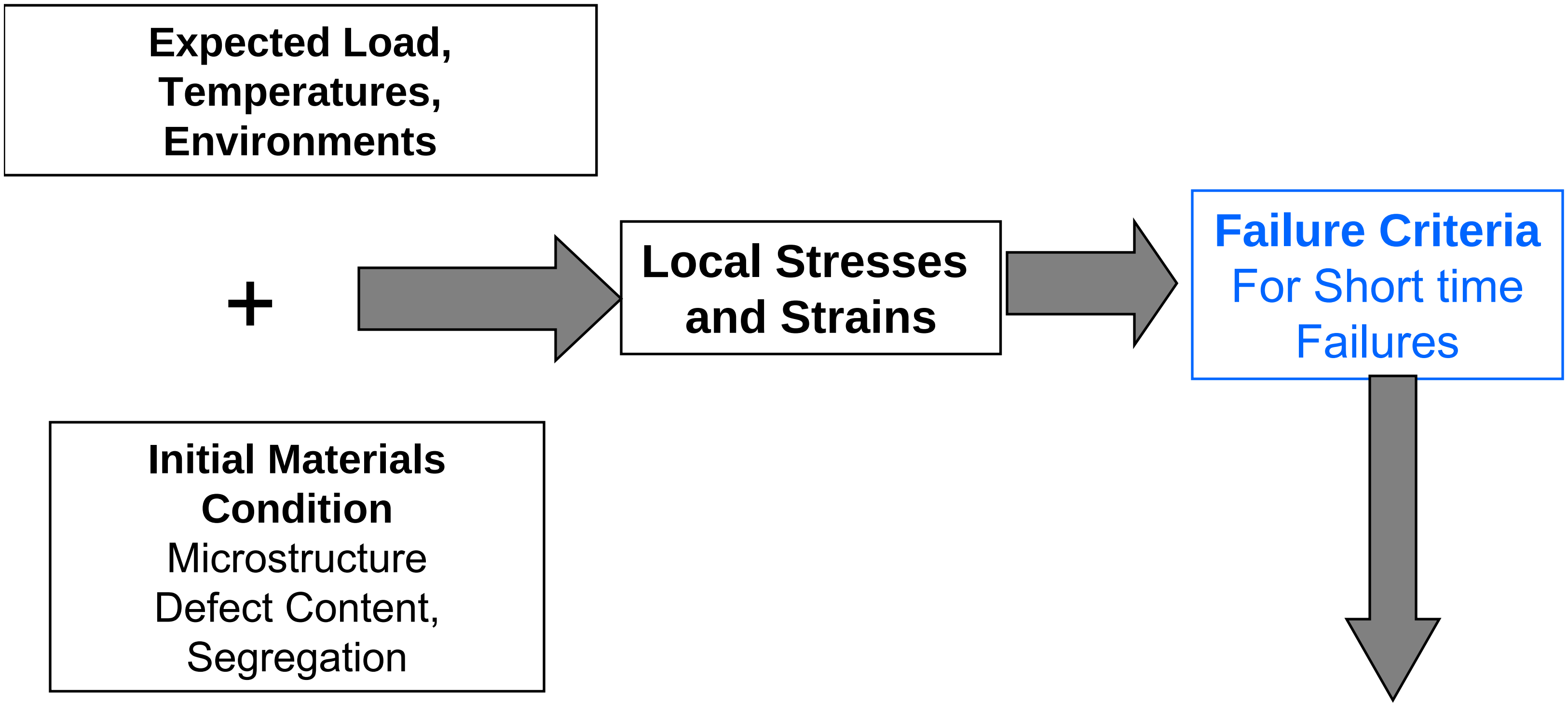
Propose Periodic Inspections and Potential for Failure Analyses Based on Findings of Inspections

- Systematic Analysis of All Components (in Plant Internals)
- Screen Probability of Failure under 8 Mechanism

Thermal Embrittlement	IASCC	Wear	Void Swelling
Irradiation Embrittlement	SCC	Fatigue	Irradiation Creep/Relaxation

- Classify Component/Mechanism Pairs and Determine Quantitative Rankings for Each Mechanism…
- … Propose this is the Order to Inspect and Evaluate Components ("Waterfall Charts")
- For Close Rankings and Key Components…
- … Needs Quantitative Understanding of Generic Materials Degradation

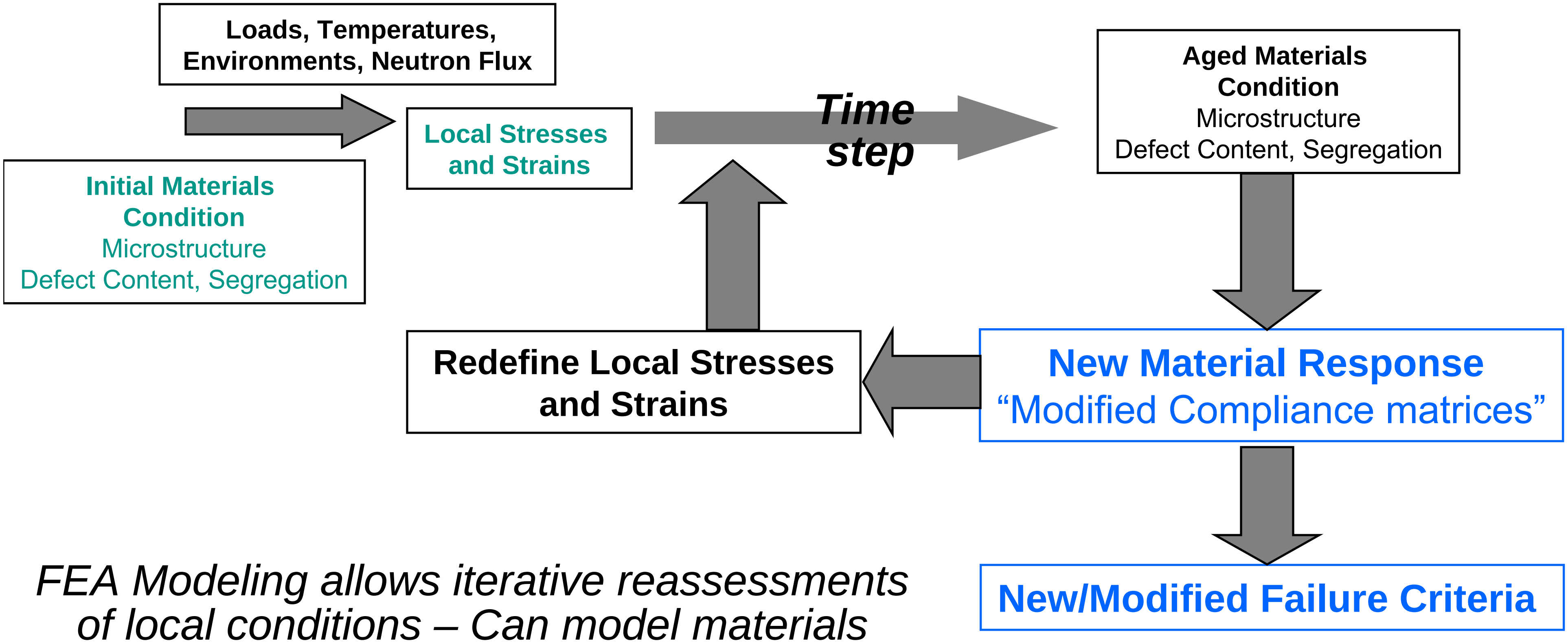
Conventional Plant Design and Analysis Process



Design Success !

But we have in-situ materials degradation …

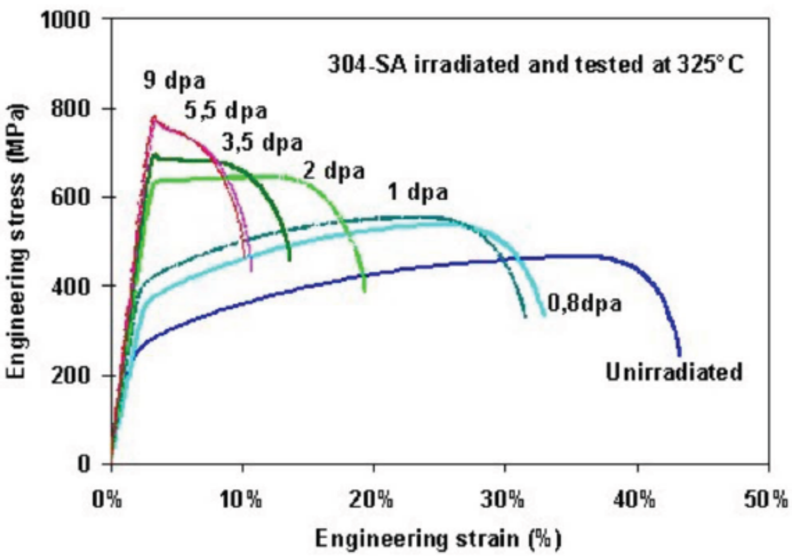
Materials' Constitutive Modeling for Quantitative Assessment of Reactor Internals Degradation



FEA Modeling allows iterative reassessments of local conditions – Can model materials degradation and assess potential for failure under local conditions of materials degradation

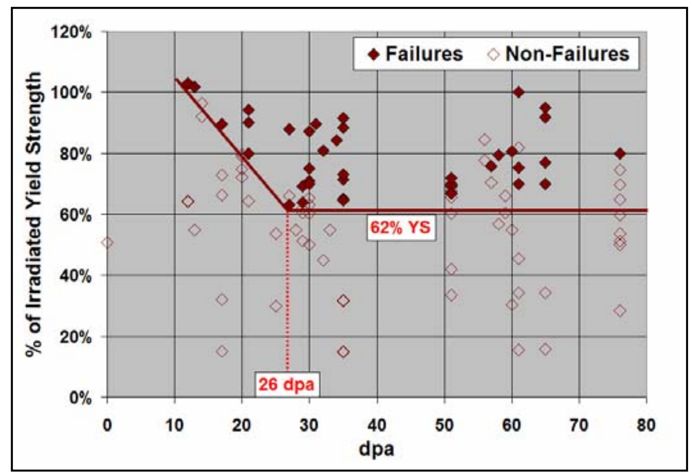
Inspection and Evaluation Basis for Internals
The Need for Quantitative Data

Reactor Structural Internals are made from Stainless Steels



Effects of Irradiation on Stainless Steels

- **Hardening & Loss of Ductility**
- **Stress Corrosion Cracking**
- **Irradiation Induced Creep**



Both Compliance and Failure Data Must be Developed to Support these Analyses

Irradiated RVI Stainless Steels Testing

Materials available after high dose - from decommisioned plants (or replaced internals ?)

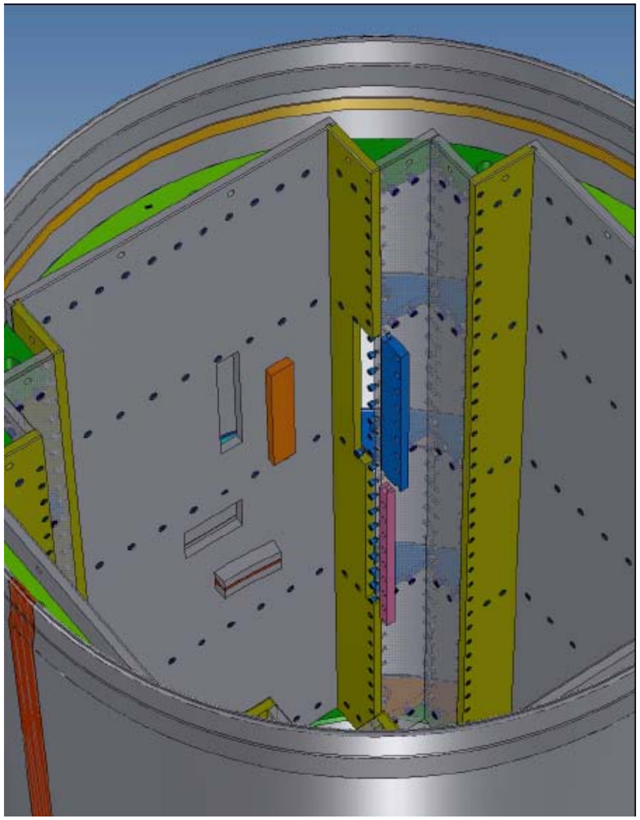
Sufficient fluence and local heating to predict behavior to long plant life ?

Properties :

- Stress/Strain Behavior
- Fracture Toughness
- Stress Corrosion Cracking

Test Environment (with or without neutrons)

- Hot Cell Testing
- Test Reactor Testing



Schematic Extraction Locations for RVI Test Material

Material Applications – Key Material Issues in a PWR Environment

	Degradation Mechanisms	*Performance Issues*
Low Alloy Steel	Embrittlement, Fatigue	PTS, Plant Operating Limits
Stainless Steel	Embrittlement, SCC, IASCC, Fatigue, Stress Relaxation, Wear, Errosion/corrosion	Baffle Bolts, Fatigue of Lower Core Plate,
Zr Alloys	Corrosion, Embrittlement, Wear, Irradiation Growth	Burnup Limits, Rod Reliability, LOCA Survivability, Control Rod Insertion …
UO_2	Pellet Densification/ Growth, Fission Gas Release	Rod Internal Pressures Pellet Clad Interactions
Alloy 600	PWSCC	Head Penetrations, BMI, SG Tubing, Nozzle Welds …
Specialty Materials	Wear, Corrosion, …….	Pump Seals, Boraflex/Boral, Instrumentation Insulation, Degradation of Concrete

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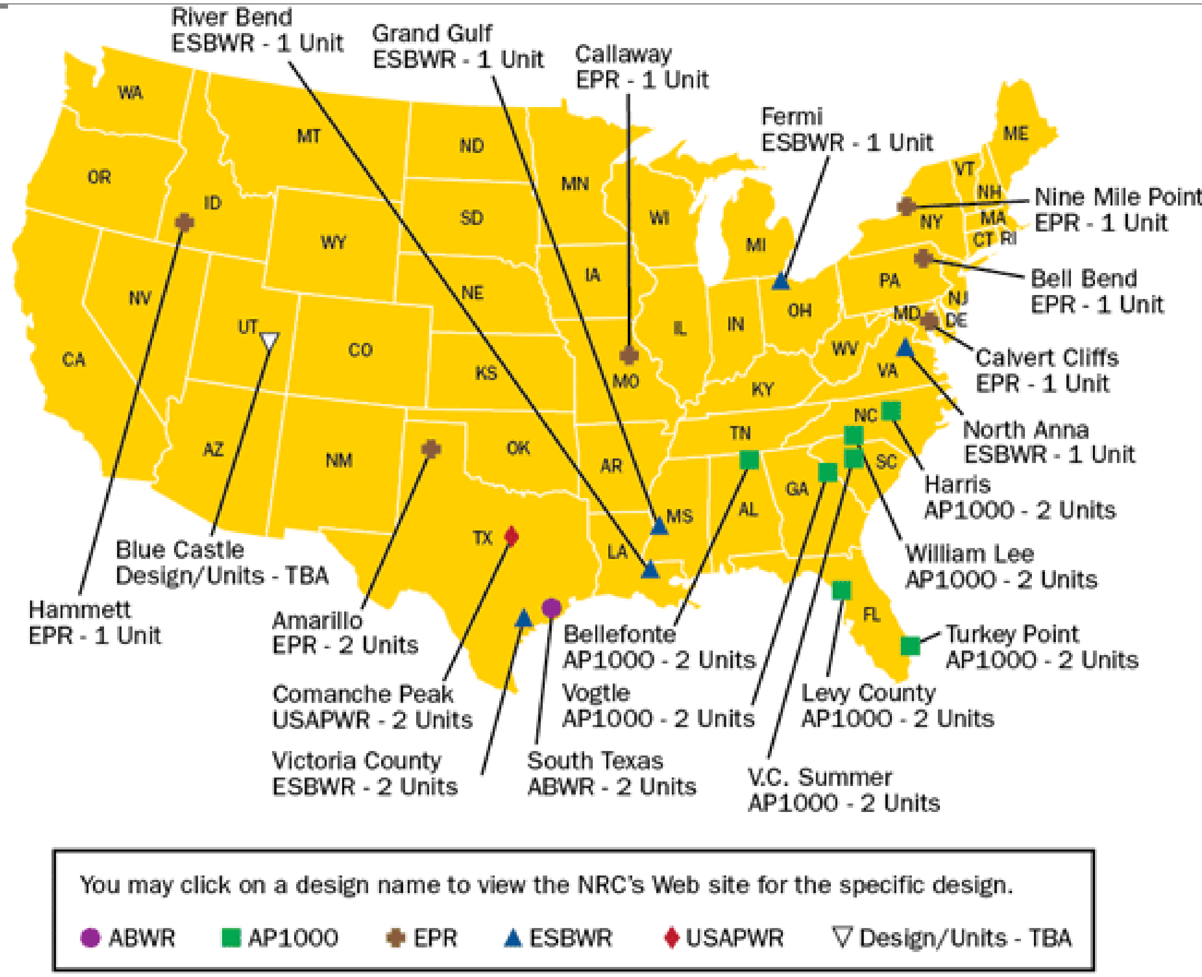
Planned New Build GENIII+ Plants in USA



Reactor Materials Options in New Plants

Building on Experience with Existing Plants …

Low Alloy steels	Availability in Large Sizes, Forgings rather than Welded Plates, Avoid Irradiation Embrittlement by Trace Element Control
Ni Base Alloys	Alloy 690 Class of Materials has replaced Alloy 600 Class of Materials
Stainless Steels	304 and 316 Materials have Proven Viable for Long Term Service. The Industry Understands How to Avoid Sensitization Issues
Zr Alloys	Improved Alloys Will be Used to Withstand Higher Burn-UP

The New Plant Designs Use Validated Materials Performance Currently Applied in Existing Plants to Continue to Meet Better than 90% Capacity Factor Operations

New Plant Materials Technology Issues

- Manufacturing Issues
- Product Validation
- Degradation Mechanism Understanding
- Operational Assessment Capability
- Mitigation Approaches
- Inspection & Evaluation Needs
- Repair / Replacement Options
- Regulatory Issues

Concern for New Plants : Component Supply

Reactor Pressure Vessel
Bottom Petal

- Dimension:(mm)
  7,636od X 5,290id X 1,631H
- Weight: 80ton

Reactor Pressure Vessel
Core Region Shell

- Dimension: (mm)
  7,478od X 7,122id X 3,962H
- Weight: 127ton.

Reactor Pressure Vessel
Integrated Type Closure Head

- Dimension: (mm)
  4,015od 1,705H
- Weight: 38ton

Requirements

•Manufacturing Infrastructure
•Testing and Validation
•Timely Supply

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Concern for New Plants : Validation of New Component Supply

High Performance Pressure Boundary Components Require :

- Uniform Chemistry and Structure
- Validated Mechanical Properties – Strength and Toughness
- Properties must be Exhibited in All Sections

Supplier as well as component development

High Performance Production Parts Require :
Materials Qualification Tests to Support Piece Acceptance

Integrally Forged Piping Segments During Processing

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Concern for New Plants : Component Fabrication Capability

- Large welded Structure need processes to minimize residual stresses etc.

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Materials Technology Needs for Nuclear Power Generation Applications

	Existing Plants	New Build
Plant Operations	Relicence Existing Gen III Plants	Build New Gen III+ Plants
Materials	Standard Materials – 30 year vintage	Standard Materials – 2010 Vintage
Materials Technology Required	Understand aging and degradation of properties. Quantify long time dependent behavior Inspection & Analysis tools Repair and Replace Options	Validation of new materials variants (e.g S level 0.002% spec as <0.030) Extension of property database Validate modern processing routes

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Materials Technology to Meet the Upcoming Demand for Nuclear Power

Calls for :

- ➢Well validated extensive database of material properties
- ➢Testing of existing materials into new regimes
- ➢Testing of new materials to the same level that existing materials have been validated
- ➢Testing under realistic conditions (Time temperature environment)
- ➢In depth understanding of the interactions of all potential effects on materials performance

	Near Term	Intermediate Term	Long Term
Plan			s
Reac Tech			n
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Materials Technology Development Where Universities Can Help

Corrosion and Stress Corrosion Cracking Data	Zirconium Alloys Behaviors
NDE Technology	Residual Stress
Metallurgy of Steels	Advanced Knowledge Management/ Databases
Structurals Materials Behavior (Creep, Fatigue : Piping Pressure Boundaries)	
Effect of Irradiation on Materials Properties – Values, Degradation, Why Values Degrade (Quantitative Mechanisms)	