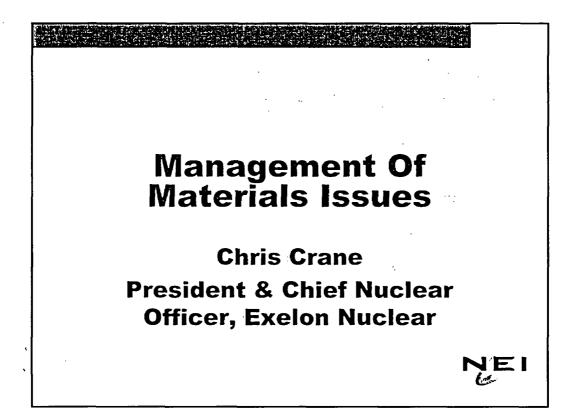
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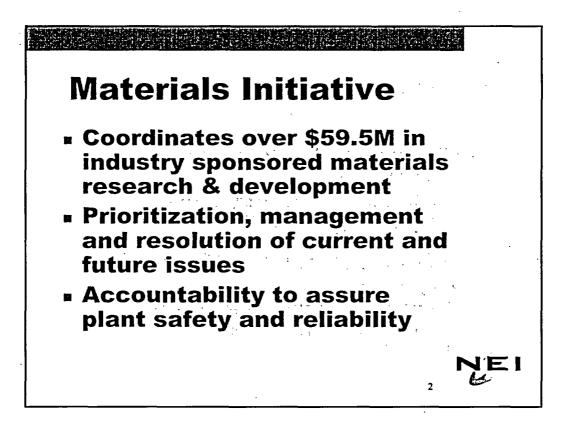
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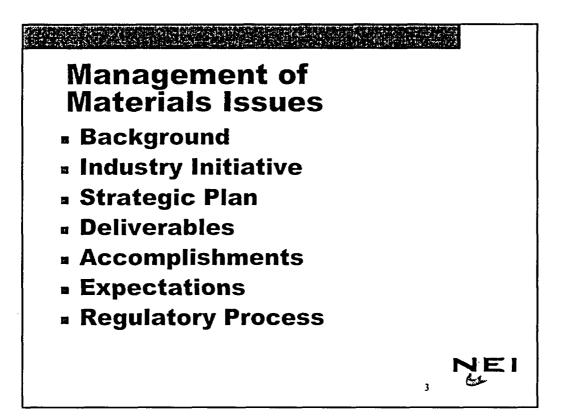
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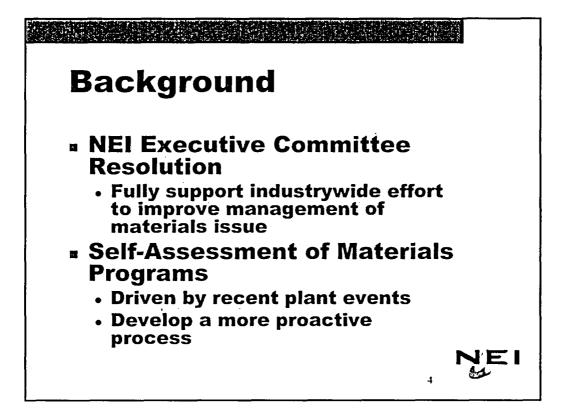
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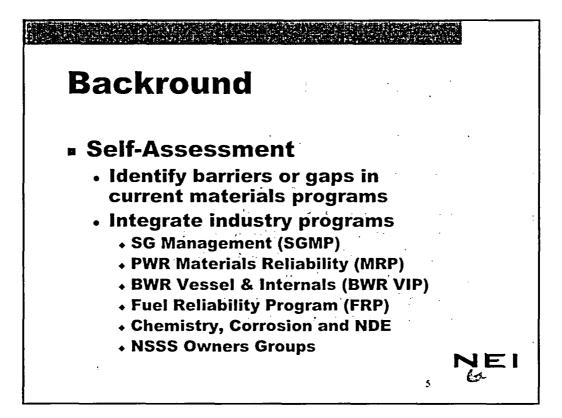
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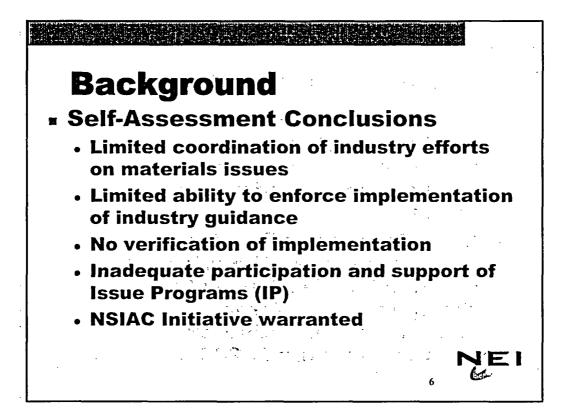


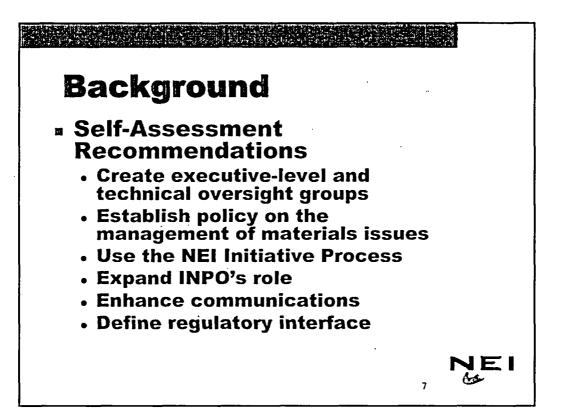


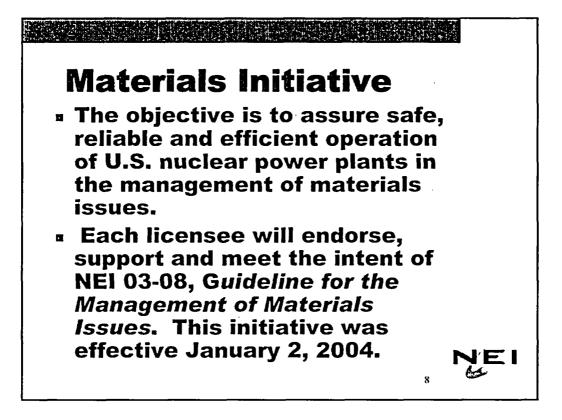


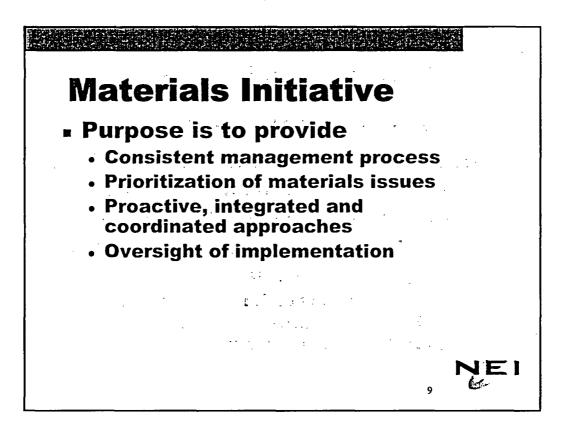


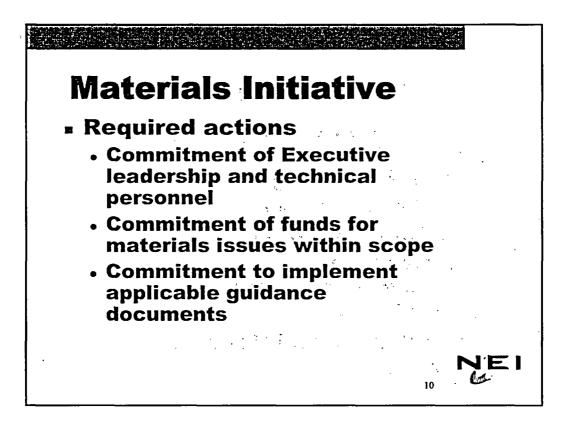


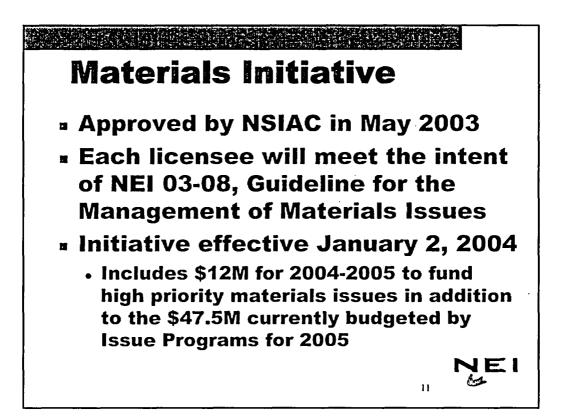


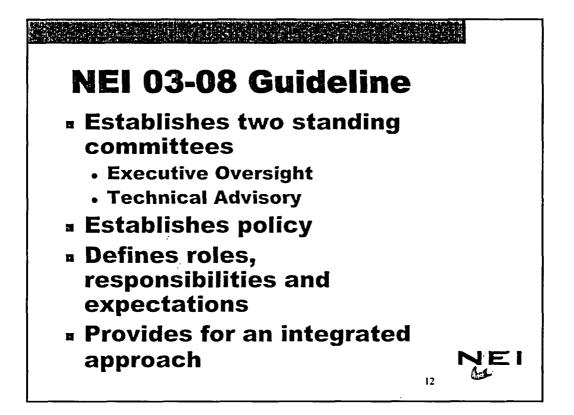


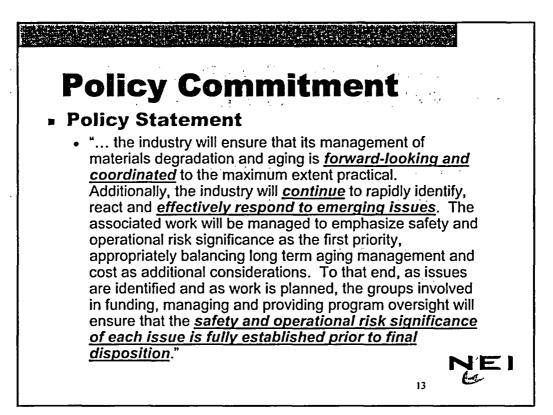


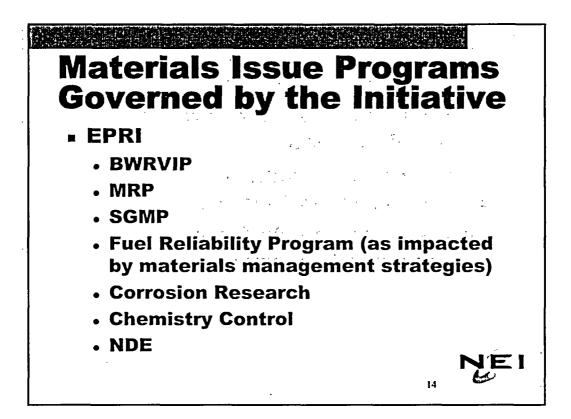


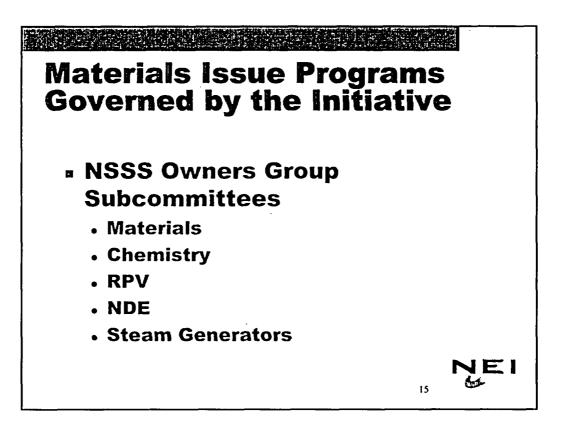


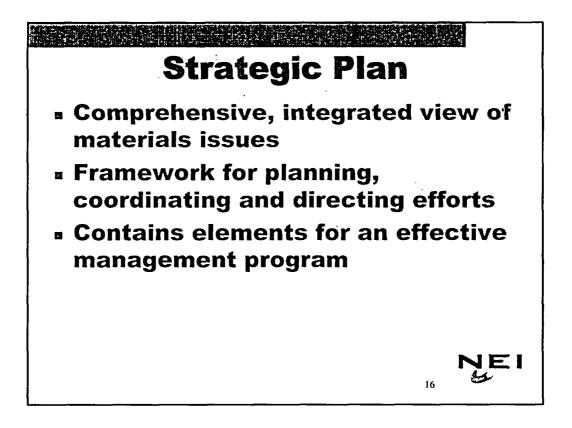


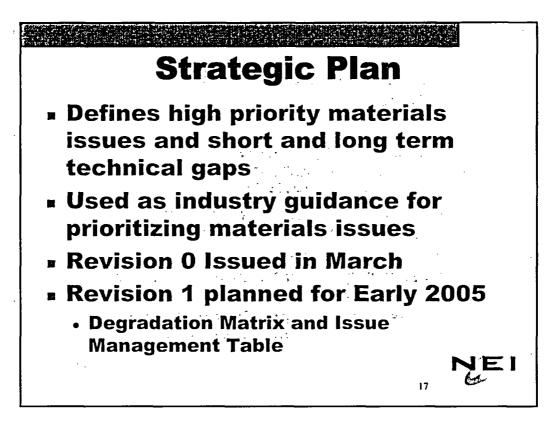


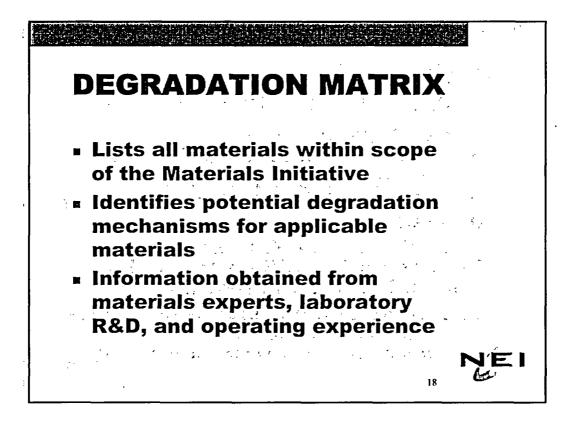


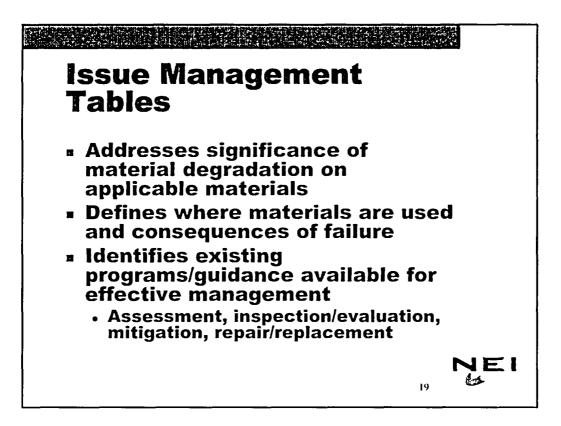


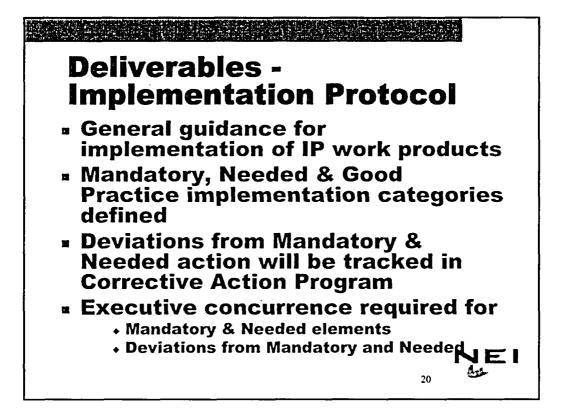


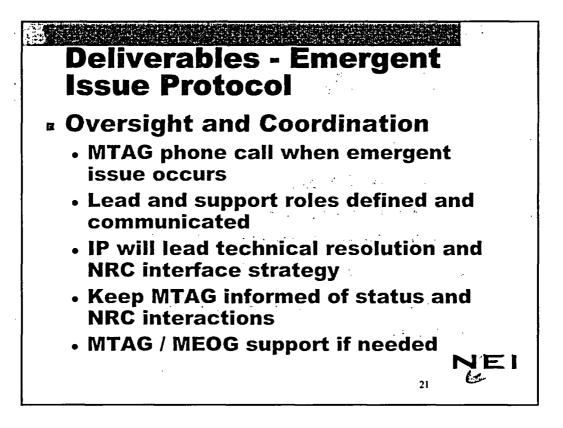


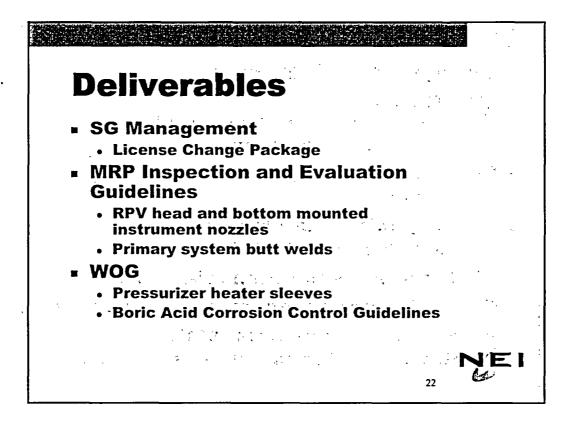


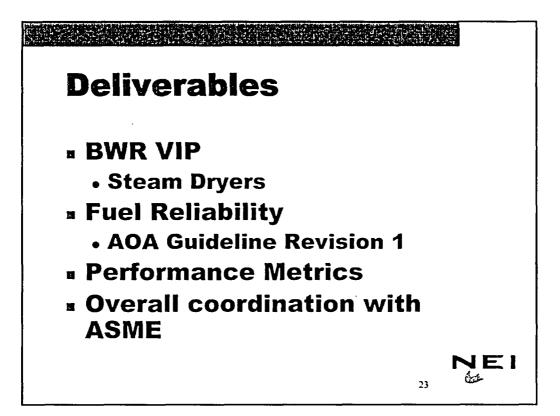


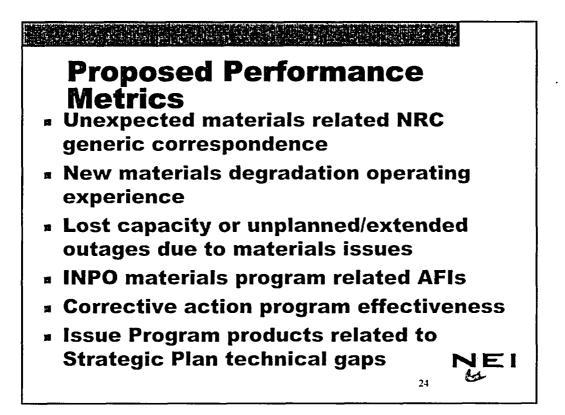


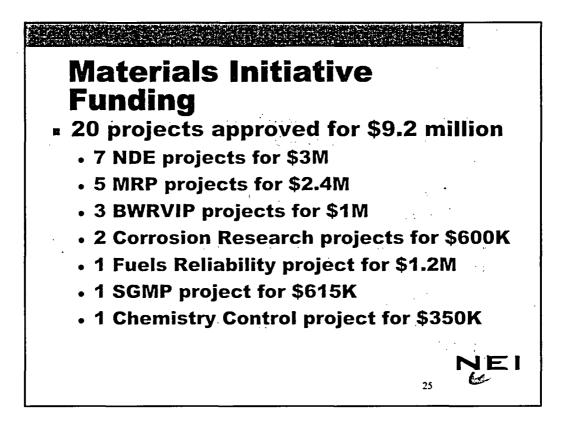


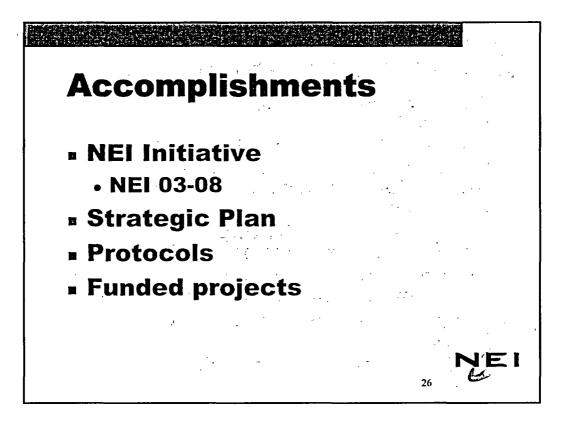


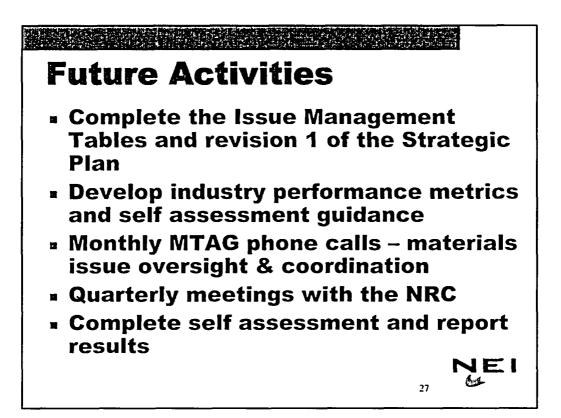




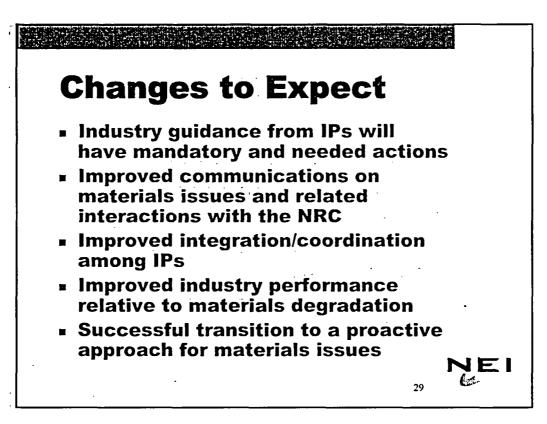


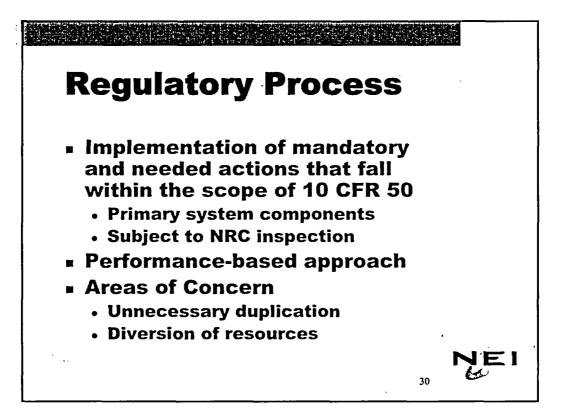


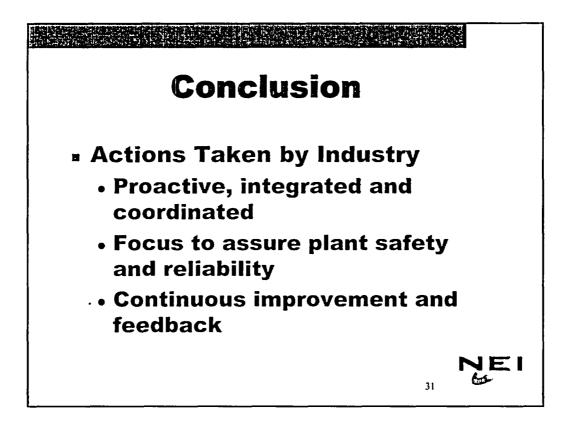


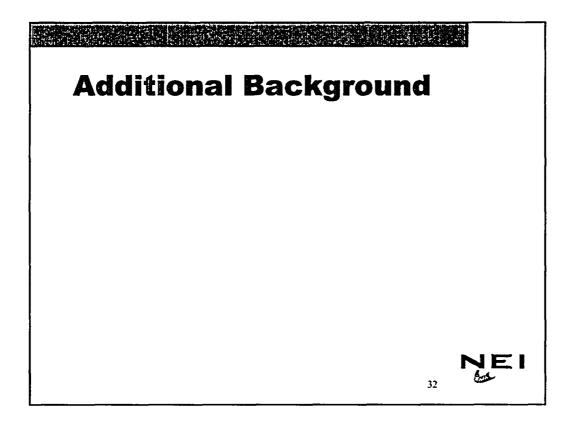


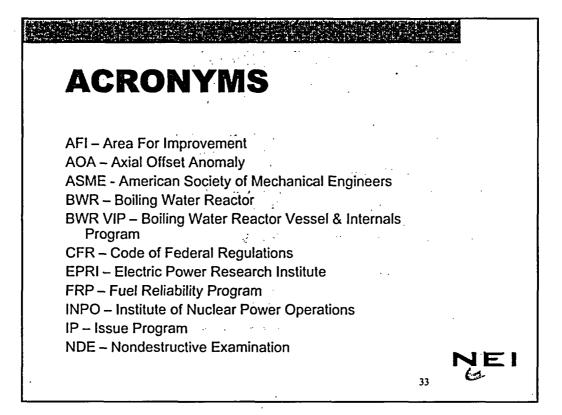


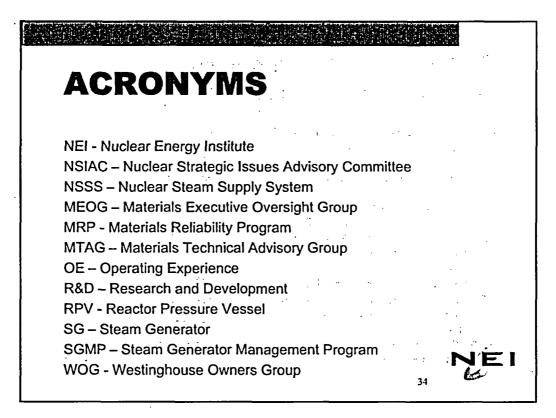






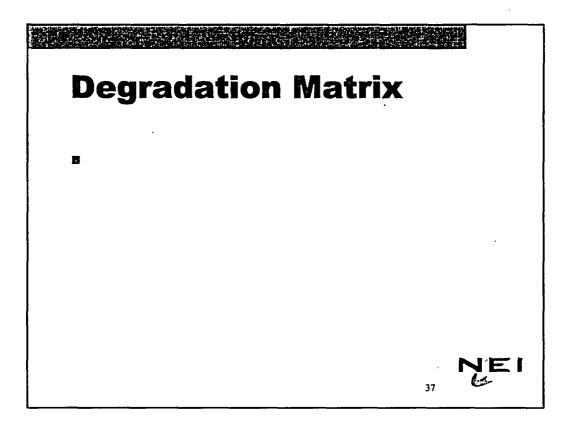




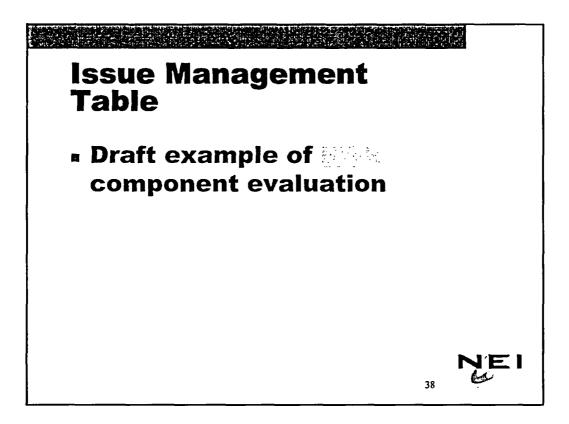








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#### **Materials Degradation Matrix**

Level 1

		BWR						
PWR Reactor Pressure Vessel	PWR Pressurizer	PWR SG Shell	PWR Reactor Internals	PWR Piping	PWR SG Tubes & Internals	BWR Pressure Vessel	BWR Reactor Internals	BWR Piping

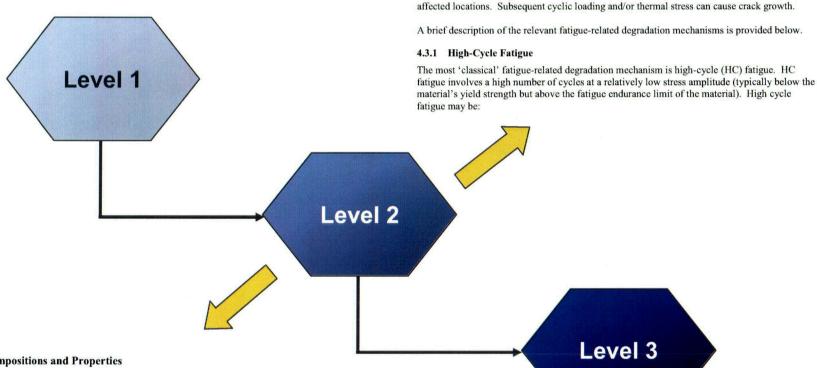
#### Level 2

PWR	Material	SCC					Corrosion/Wear <u>C &amp; W</u>			Fatigue <u>Fat.</u>		Reduction in Toughness <u>RiT</u>							
Component	Waterial	<u>SCC</u>				Aging						Irradiation							
	<sup>1</sup> Subdivision $\rightarrow$	IG	IA	TG	LTCP	PW	Wstg	Pit	Wear	FAC	HC	LC/Th	Env	Th	Emb	VS	SR	Th <sub>n</sub>	Fl
	<u>C&amp;LAS</u>	? e002	N	? <u>e002</u>	N	? e003	Y <u>e004</u>	N	N	Y <u>e005</u>	N	Y <u>e006</u>	Y <u>e007</u>	Y <u>e008</u>	N/A	N/A	N/A	N/A	N/A
PWR Pressurizer	<u>C&amp;LAS</u> Welds	? <u>e002</u>	N	? <u>e002</u>	N	? <u>e003</u>	Y <u>e004</u>	N	N	Y <u>e005</u>	N	Y <u>e006</u>	Y <u>e007</u>	Y <u>e008</u>	N/A	N/A	N/A	N/A	N/A
	Wrought SS	? <u>e012</u>	N	? <u>e012</u>	? <u>e013</u>	? <u>e012</u>	N	N	N	N	N	Y <u>e014</u>	Y <u>e015</u>	N	N/A	N/A	N/A	N/A	N/A
(Including Shell, Surge and Spray	SS Welds & Clad	Y <u>e016</u>	? <u>e017</u>	Y <u>e018</u>	? <u>e013</u>	? <u>e019</u>	N	N	? <u>e020</u>	N	N	? <u>e014</u>	Y <u>e015</u>	Y <u>e022</u>	N/A	N/A	N/A	N/A	N/A
Nozzles, Heater Sleeves and	Wrought Ni Alloys	N	N	N	? <u>e023</u>	Y <u>e023</u>	N	N	N	N	Y <u>e014</u>	Y <u>e014</u>	Y <u>e015</u>	N	N/A	N/A	N/A	N/A	N/A
Sheaths, Instrument Penetrations)	<u>Ni-base</u> Welds & Clad	N	? <u>e024</u>	N	Y <u>e023</u>	Y <u>e025</u>	N	N	N	N	N	Y <u>e014</u>	Y <u>e015</u>	N	N/A	N/A	N/A	N/A	N/A
Level 3	Ciau			L							L				L	LE	89 199	L	L

e030 Corrosion-assisted fatigue is a known phenomenon on secondary side (e.g., in the vicinity of girth welds in steam generator shells and in the region of feedwater nozzles) and is not like environmental fatigue described in other areas of this DM. Environmental fatigue research relevant to this specific phenomenon is not ongoing within MRP Fatigue ITG, and is a potential gap.

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#### **Materials Degradation Matrix**



4.3

**Fatigue Degradation Mechanisms and Mitigation Options** 

Fatigue is the structural deterioration that can occur as the result of repeated stress/strain cycles caused by fluctuating loads or temperatures. After repeated cyclic loading, if sufficient localized micro-structural damage has been accumulated, crack initiation can occur at the most highly

#### 3.2 Material Compositions and Properties

A large variety of welding materials and welding processes are used to join carbon and low-alloy steels, and it is not practical to show typical material compositions and material specifications. Section NB-2431.1 of Section III, Division I of the ASME Code requires that weld materials have tensile strength, ductility and impact properties that match those of either of the base materials being welded, as demonstrated by tests using the selected weld material and the same or similar base materials. Section NB-2432.2 of Section III, Division I of the ASME Code requires that the chemical composition of the welding material be in accordance with an appropriate ASME Code welding specification (in Section II.C of the Code), but leaves the choice of the specific material up to the manufacturer.

The most common weld processes used to join carbon steel and LAS parts include submerged arc welding, shielded metal arc welding (SMAW), and gas tungsten arc welding (GTAW). Postweld heat treatment is generally required per ASME Code rules after welding of the carbon and low-alloy steels used for reactor coolant system service.

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#### 4.3 Fatigue Degradation Mechanisms and Mitigation Options

Fatigue is the structural deterioration that can occur as the result of repeated stress/strain cycles caused by fluctuating loads or temperatures. After repeated cyclic loading, if sufficient localized micro-structural damage has been accumulated, crack initiation can occur at the most highly affected locations. Subsequent cyclic loading and/or thermal stress can cause crack growth.

A brief description of the relevant fatigue-related degradation mechanisms is provided below.

#### 4.3.1 High-Cycle Fatigue

The most 'classical' fatigue-related degradation mechanism is high-cycle (IIC) fatigue. HC fatigue involves a high number of cycles at a relatively low stress amplitude (typically below the material's yield strength but above the fatigue endurance limit of the material). High cycle fatigue may be:

#### 3.2 Material Compositions and Properties

A large variety of welding materials and welding processes are used to join carbon and low-alloy steels, and it is not practical to show typical material compositions and material specifications. Section NB-2431.1 of Section III, Division I of the ASME Code requires that weld materials have tensile strength, ductility and impact properties that match those of either of the base materials being welded, as demonstrated by tests using the selected weld material and the same or similar base materials. Section NB-2432.2 of Section III, Division I of the ASME Code requires that the chemical composition of the welding material be in accordance with an appropriate ASME Code welding specification (in Section II.C of the Code), but leaves the choice of the specific material up to the manufacturer.

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#### Draft BWR Issue Management Table

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Equipment	Material	<u>Failure</u> Mechanism	<u>Consequences</u> of Failure	Mitigation	<u>Repair /</u> Replace	<u>I &amp; E</u> Guidance	Gaps	Priority & Basis	Responsible Program(s)
BWR Recirculati on piping	SS (lc and hc), Inconel welds	SCC, fatigue	Leakage, forced outage	Yes, chemical and stress improvement	Yes, replace pipe or weld overlay	Yes, BWRVIP- 75		Low – solution available	BWRVIP, WCC
BWR Vessel	Cs/las, ss clad, welds	IGSCC, IASCC, TGSCC, FIV, Th & Env Fatigue, Emb, Th aging, Fluence	LOCA – loss of asset	Yes – HWC, NMCA	Yes – nozzle repair	Yes – covers embrittleme nt and weld degradation		Low – solution available	BWRVIP
BWR Internals	Ss, cass, cs, welds, Inc	IASCC, IGSCC, FIV, Wear, EF, Emb, Fluence – R&D needed	Core configuration	Yes – some, work needed	Yes – shroud and top-guide, costly – work needed	Yes (interim) – 13 BWRVIP I&E Guidelines – work needed		High – existing and potential unresolved issues	BWRVIP, WCC, FRP, Corrosion Research
Core Shroud	Stainless Steel	IGSCC, IASCC, Reduction in Toughness	This is a function of location. Vertical weld flaws have minor significance unless the intersecting circumferential weld is flawed through-wall. Very flaw	Hydrogen Water Chemistry at moderate levels can protect the shroud at lower levels and some benefit is available at higher levels in some plants. When augmented with Noble Metals, more shroud protect is	The repairs to date have been mechanical clamps that replace some or all of the circumferential welds. This is accomplished by developing high compressive loads on the shroud which	BWRVIP has developed a series of inspection guides that have been combined into one overall document (BWRVIP- 76). This	Fracture toughness decrease with increasing fluence and there is limited data to help the industry understan d long	High priority items for the shroud still exist. Initially was the number one priority. With programs in place for inspection, the needs for	BWRVIP with the Assessment Committee leading

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			tolerant for	possible due to the	negates the need	document	term	the shroud	
			circumferential	need for les	for the	includes "	issues.	are reduced.	
			flaws.	hydrogen due the	circumferential	inspection	Work is	· ]	
			Significance of	to the catalyst	welds. This	criteria for	underway	•	
			circumferential	effect of the noble	provides	circumferent	to develop		
			weld flaws is	metals. Plant-	significant	ial and	additional		
			based on	specific	redundancy since	vertical	data prior		
			location of the	assessments are	по	welds, and	to the 😳		
			weld.	needed to	circumferential	support ring	shrouds		
			Limiting	determine the	welds to date	welds, The	reaching		
			accident	exact level of	have come near a	criteria also	fluence		
•			scenario is	protection. Other	360 degree	address both	levels of		
			main steam	methods of	through-wall	repaired and	concern.		
			line break	protection are	flaw. Repair by	unrepaired	Similarly,		
			coupled with	being	welding is very	shrouds	work is		
	1		an earth quake.	investigated/consi	limited in	including	underway		
			Even then,	dered such as	prospect due to	repair	to		
			shroud	water jet peening	difficulty in	hardware.	properly	· ·	
			circumferential	etc.	welding on	The	understan	·	•
	}	· ·	welds were		highly irradiated	evaluation	d and	· · ·	
	.:		shown to	. 4	stainless steels.	criteria exist	characteri		
			perform -			for all	ze crack		
			adequately			locations	growth		
			with 90%			and is based	behavior		
			through wall			on long	in highly		
			flaw, 360° in			standing	irradiated		
			circumference.			ASME	stainless		
			It is worth			criteria	steels.		
			noting that			including			
		-	should a flaw			safety		1	
			develop for the	· .		factors.		1	
	· ·		full			Evaluation			
			circumference,			methods		.	•
			the shroud			also account	ł		
		· ·	would lift			for the			
			enough to			changes in			
			"burp" itself			crack			

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and provide operators with an indication of a problem and allow safe shutdown.	growth rate and fracture toughness a irradiation damage increases. The methods also allow for crack growth to be adjusted based on water chemistry,	

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### PLANT AGING AND MATERIALS DEGRADATION ISSUES

Jim Dyer, Director NRR Dr. Carl Paperiello, Director RES November 8, 2004

### Acronyms

- ASME: American Society of Mechanical Engineers
- BWR: Boiling-water reactor
- CCDF: Conditional core damage frequency
- IASCC: Irradiation-assisted stress-corrosion cracking
- IGSCC: Intergranular stress-corrosion cracking
- ISI: Inservice inspection
- NRR: Office of Nuclear Reactor Regulation
- PIRT: Phenomena identification and ranking table
- PWR: Pressurized-water reactor
- PWSCC: Primary water stress-corrosion cracking
- RCS: Reactor coolant system
- RES: Office of Nuclear Regulatory Research
- RPV: Reactor pressure vessel

# **Regulatory Overview**

- Regulatory framework
- Historical issues with materials degradation
- Current challenges
- Future goals

# **Regulatory Framework**

- Regulations
  - -ASME Code requirements for design and inspection
- Technical Specifications
  - Leakage monitoring
- Bulletins and Orders short term, emergent issues

# Regulatory Framework (continued)

- Integration of Research for degradation management programs
- Industry actions and initiatives

# Regulatory Framework (continued)

- Regulatory Oversight
  - Baseline ISI
  - **Temporary Instructions**
  - -Follow up interactions with licensees
  - Reactor Oversight Process indicators and findings

# Historical Issues With Materials Degradation

- Radiation embrittlement of RPVs
- IGSCC in BWR
  piping
- BWR internals cracking
- Steam generator tube degradation

- PWSCC in PWRs
- Boric acid corrosion
- IASCC in PWRs
- Vibratory fatigue
- Thermal fatigue
- Erosion/corrosion

### **Current Challenges**

- PWSCC of penetration nozzles in reactor and other vessels, nozzle welds
  - Addressed through Bulletins and Order
- PWSCC in dissimilar metal butt welds of RCS piping
  - Regulatory action being considered

# Current Challenges (continued)

- Extended power uprate adverse flow effects
  - Steam dryer cracking
  - Feedwater probe failure
  - Staff monitoring industry actions

# Current Challenges (continued)

- Steam generator degradation
  - -Active NRC oversight
  - Implementing new regulatory framework through technical specifications

#### **Future Goals**

- Return to more stable regulation
- Increase reliance on industry
- Proactive oversight
- Operating experience monitoring
- Independent NRC research

## **Proactive Materials Degradation Assessment**

- Research input into regulatory activities
- Approach
- Short-term program
- Longer term program
- Implementation schedule
- Utilization of results

# **Research Input Into Regulatory Activities**

- Environmentally assisted cracking
  - **BWR** pipe cracking
  - -Internals cracking
- Steam generator integrity
- Nondestructive Examination
- Pressure vessel integrity

## Approach

- Some of the materials degradation in nuclear power plant components experienced in the past has surprised both NRC and industry
- NRC & industry pursuing proactive approach to materials degradation

# Approach (Continued)

- Identify components of interest to ensure reliability
- Coordinate research for effective implementation of proactive materials degradation management

# Approach (Continued)

- Activities to identify components
  - Information on degradation that has been experienced – shortterm results
  - Phenomena identification and ranking table (PIRT) for potential future degradation – longer term results

### **Short-term Program**

- Identified components that have already experienced degradation
- Evaluating inspection and leak monitoring techniques and requirements
- Provide input to conditional core damage frequency (CCDF)
- Probabilities of failure for probabilistic risk assessments

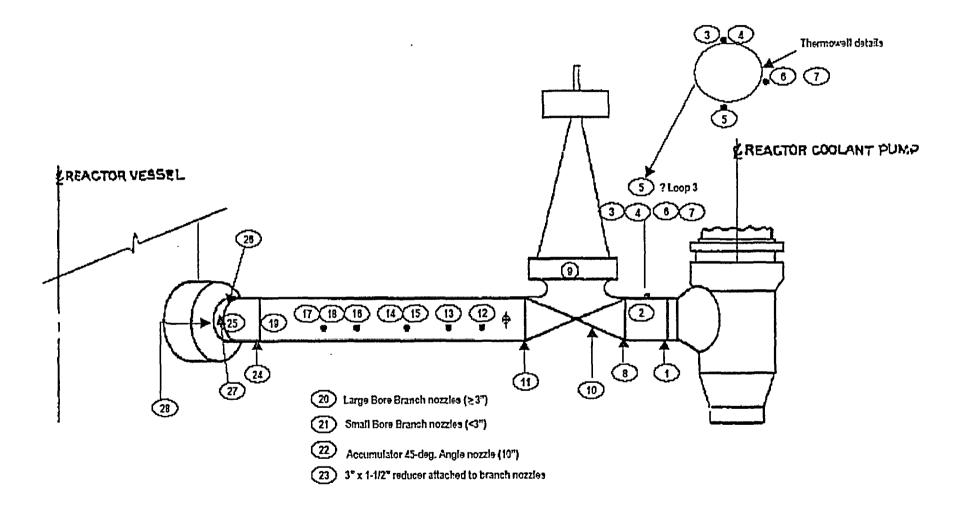
# Longer Term Program

- PIRT process
  - Expert elicitation
- International in scope
- Bounded important plant systems being addressed

# Longer Term Program (Continued)

- Background information provided to expert panel members
  - Component's environment and operational experience
- Experts identify and rate degradation mechanisms, assess knowledge levels, and provide bases for decisions

#### GROUP 1: RCS Cold Leg Piping (Covers worksheets RCS-CL – 1 thru 28)



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## **Implementation Schedule**

- First two PIRT meetings on PWR reactor coolant system and emergency core cooling system completed
  - Potential future degradation being identified
- Six more meetings to complete PWR and BWR evaluations
- PWR report June 2005
- BWR report December 2005

### **Utilization of Results**

- Lists of components susceptible to future degradation, bases for calls, and knowledge levels
- Input to cooperative integrated research programs
- Technical basis for effective proactive materials degradation management by NRC and industry