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Overview of NRC IASCC Research Program

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Final Meeting of CIR II Extension Program, EDF Les Renardieres, France

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NRC's Related Activities and Perspective

- Testing, evaluation & research on irradiated stainless steels
- Investigation of degradation of BWR internals
- Investigation of degradation of PWR internals
- Primary areas of interest to NRC
- Primary benefits & concerns

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Primary Interest to the NRC Program

- **Materials series with low dose exposure**
  - **Help understand thresholds for irradiation effects:**
    - ➔ **Fracture and tearing toughness,**
    - ➔ **Irradiation-Assisted Stress Corrosion Cracking**
- **Materials series with high dose exposure**
  - **Help understand if saturation of mechanical properties occurs:**
    - ➔ **Radiation-induced segregation**
    - ➔ **Void swelling (if any at a given temperature and exposure)**
    - ➔ **Fracture and mechanical properties**

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Technical Issues Addressed by NRC Research

- IGSCC – BWR/PWR
- IASCC-BWR/PWR
- Void Swelling
- Radiation Embrittlement
- Thermal and Radiation Embrittlement of Cast Austenitic Stainless Steel (CASS) Materials
- Radiation-Induced Stress Relaxation

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Degradation of BWR Internals

- Started years ago to address IGSCC of sensitized stainless steel.
- Conducted under test conditions corresponding to both normal water chemistry and hydrogen water chemistry
  - (fluence to ~3 dpa, high- and low-electrochemical potential test conditions)
- Conducted at ANL, additional technical analyses by NRC
  - ➔ 6 technical reports (NUREGs) since 2000, 1 more is planned as the program concludes

Important Data Established from IASCC in BWRs Research Program

- Increased resistance to IASCC in SSs (Type 304 and 316) if
  - ➔ sulfur (S) is below 0.002%
- Decreased resistance to IASCC in SSs if
  - ➔ *sulfur (S) is above 0.003%*
- Increased resistance to IASCC in SSs (Type 304 and 316) if
  - ➔ *carbon (C) in steels is above 0.03 %*
- CGR in HWC is more than one order of magnitude lower than that in NWC under similar cyclic loading conditions
- Irradiation of SSs (Type 304 and 316) exhibit strain softening; however, the effect of the strain softening on the plastic zone ahead of a crack tip is minimal.
- Degree of sensitization and 60% GBE has negligible effect on IASCC resistance.
- An increase in the total neutron dose significantly decreases both the resistance to IASCC and the fracture toughness of steels (Type 304 and 316).

IASCC in BWRs

Preliminary Review of Knowledge Gaps

- If there is a high dose limit for the beneficial effect of HWC – What is the dose or dpa limit ?
- Establish a relationship between the CGR results to the results from fractographic analysis of all CGR specimens.

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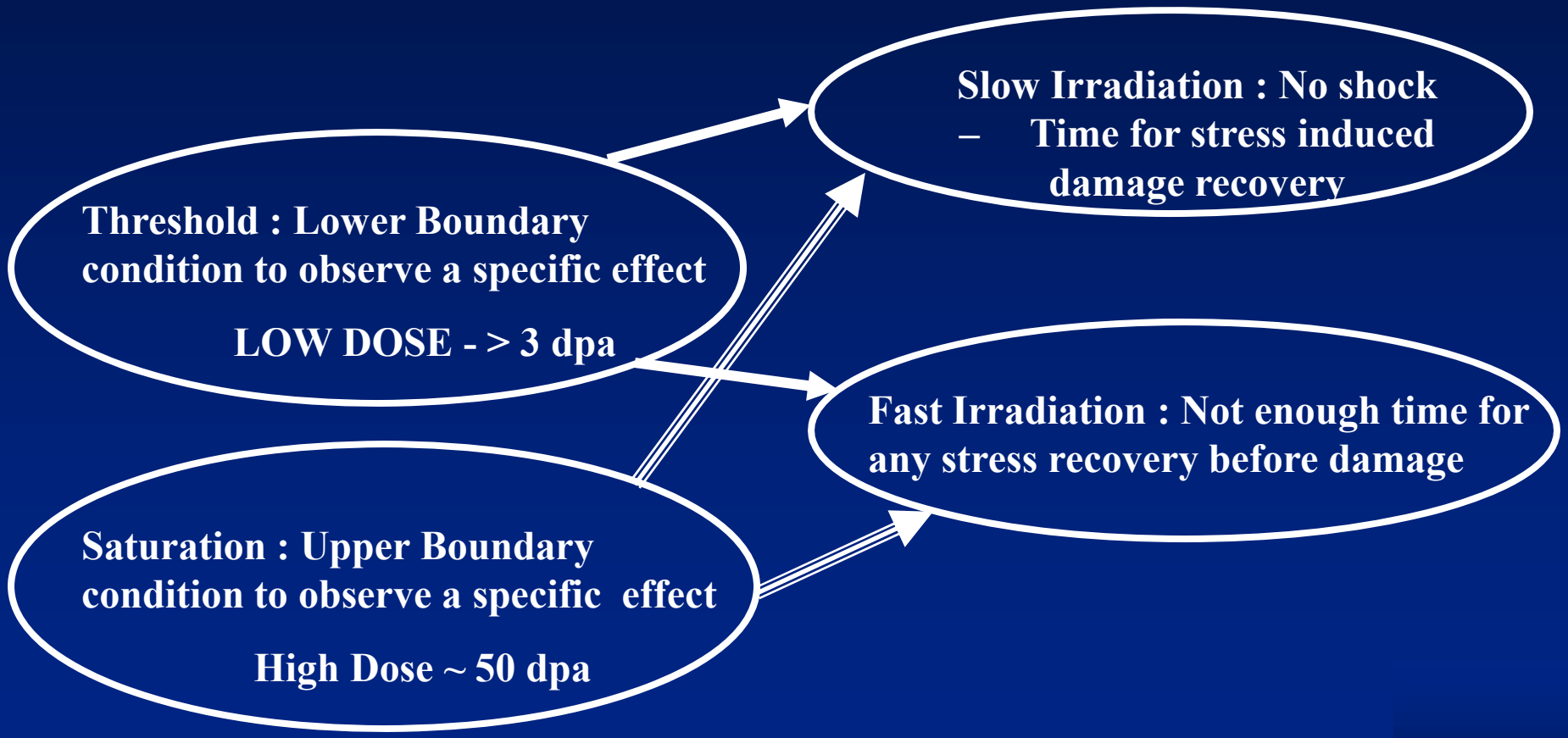
Degradation of PWR Internals

- Few resources expended to date, PWR effort increasing as BWR effort concludes
- Multifaceted: IASCC, embrittlement, creep, ever-changing compositions, radiation-induced segregation, void swelling, etc.
- Most test specimens from BOR-60 (fast reactor) augmented by irradiations using Halden reactor
- Track international results to capture all important information
- Current task should produce 3 PWR-related NUREGs in 2008-09; applications of data mostly in license renewal arena
  - Zorita core support materials would augment & corroborate fast reactor irradiations

IASCC in PWRs



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IASCC in PWRs

- **Materials at a series of low doses to understand thresholds for irradiation degradation:**
  - **Fracture and tearing toughness,**
  - **Irradiation-Assisted Stress Corrosion Cracking**
- **Materials at a series of high doses to understand if saturation of irradiation degradation of mechanical properties occurs:**
  - **Radiation-induced segregation**
  - **Void swelling, fracture and tensile properties**
  - **Characterization of deformation at high fluences**

IASCC in PWRs

Effectiveness of SCC Mitigation

- **Objectives:**
  - Evaluate IASCC susceptibility in PWR environment as a function of fluence, material chemistry, and processing condition (e.g., Solution Anneal or Cold Work)
- **Approach**
  - SSRT tests (5, 10, 40 dpa BOR-60 specimens)
    - To evaluate IASCC susceptibility (IG fraction) in PWR environment
    - To evaluate IASCC CGRs at PWR relevant condition
  - CGR tests
    - increase test temperature
    - Specimens
      - Halden Reactor Irradiation ➔ Started in May, 2009
        - ➔ ¼T-C(T) specimens Irradiated to Low Dose (5- 10 dpa)
          - Expect to arrive in May or June, 2010.
        - ➔ ¼T-C(T) specimens Irradiated to High Dose (~ 40 dpa)
          - Expect to arrive in May or June, 2013.

Void Swelling

- **Objectives:**
  - Assess void swelling behavior in austenitic SSs at <u>doses</u> and <u>temperatures</u> relevant to PWR conditions.

- **Approach**
  - Several 5, 10, 20 dpa specimens will be examined
  - TEM disks (< 40 dpa BOR-60 specimens) will be examined
  - Thermal calculations are planned on Zorita internals in order to determine the location for materials harvesting for void swelling test

Task 3 – Radiation Embrittlement

- Approach
  - CGR and Fracture toughness tests will be performed on specimens irradiated to 5-7 dpa.
  - Specimens ➤ Halden irradiation ➔ Started in May, 2009
    - ➢ Low Dose ¼T-C(T) specimens
      - ➢ Expect to arrive by June, 2010.
    - ➢ High Dose ¼T- C(T) specimens
      - ➢ Expect to arrive by June, 2012
    - ➢ Low dose specimens from BOR – 60 were examined
      - ➢ preliminary data shows no radiation embrittlement. But radiation induced stress relaxation.

Task 4 – Radiation and Thermal Embrittlement of CASS

- **Objective**
  - Evaluate synergistic effects of thermal and radiation embrittlement on CASS
- **Status**
  - A new irradiation has been initiated and post-irradiation CGR/JR tests will be carried out
- **Schedule**
  - Tentative (Low dose samples) Early 2011

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New Initiative - Zorita internals research program (ZIRP) Project Objectives

➢ Joint Industry and USNRC Collaborative effort

- **The current database for irradiated stainless steels is insufficient to assess core internals components performance over the full range of radiation service exposures (i.e., *extended operation beyond 40, and more importantly, 60 years*) anticipated for LWRs and in particular for PWRs.**
- **The objective of this cooperative project is to generate high quality data to fill existing gaps and improve the mechanistic understanding to avoid inadequate extrapolations in the evaluation of core internals in both BWRs and PWRs.**
- **Materials from Zorita NPP are considered adequate to generate a database representative of the behavior of internals components in LWRs.**
  - **High thickness of baffle plates**
  - **Doses up to 58 dpa.**
- **Results of this program will support**
  - **Licensing decisions, Inspection programs, Lifetime management and Lifetime extension**

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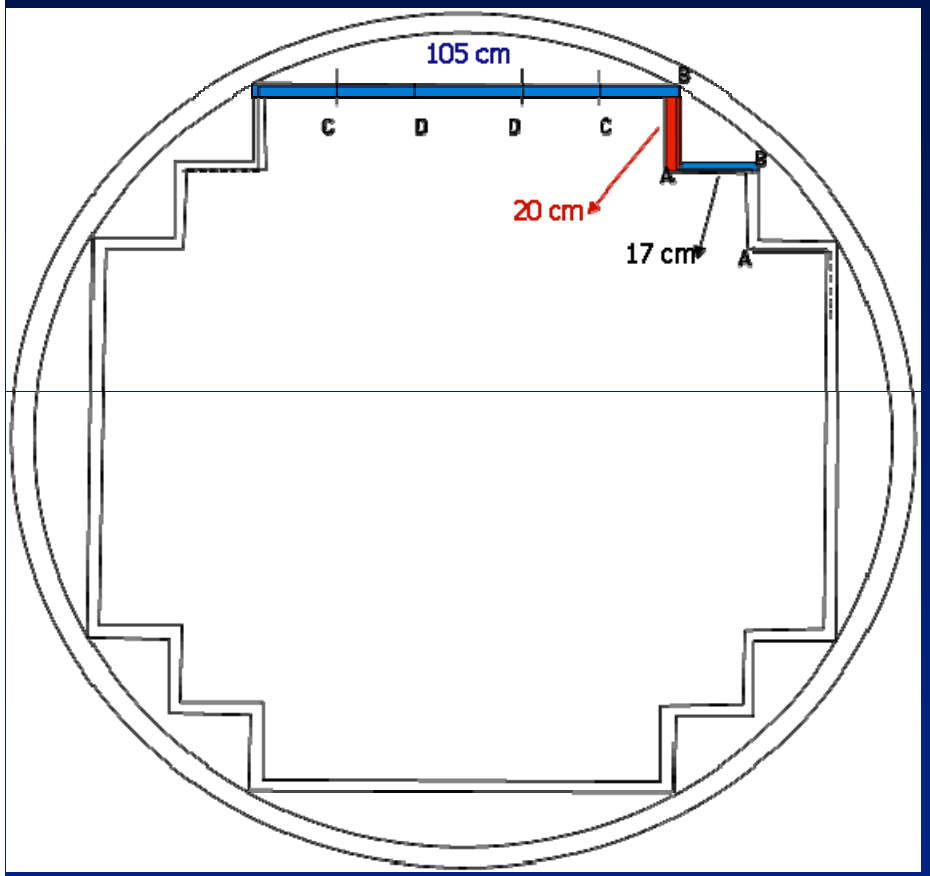
Baffle Configuration



ESTIMATION OF ACCUMULATED FLUENCES AND TEMPERATURES

Baffle dose (dpa), internal surface				
Position	A	B	C	D
Former 1	10.35	2.09	4.70	4.38
Former 2	45.11	9.13	20.47	17.70
Former 3	58.39	11.91	26.50	22.91
Former 4	54.59	11.06	24.82	21.45
Former 5	24.87	5.03	11.29	9.76

Baffle dose (dpa), external surface				
Position	A	B	C	D
Former 1	6.48	1.31	3.37	3.14
Former 2	28.22	5.71	14.69	12.70
Former 3	36.53	7.39	19.02	16.44
Former 4	34.22	6.92	17.82	15.40
Former 5	15.56	3.15	8.10	7.00

Estimated T at the Occluded Areas of the BFB is ~ 350-355°C

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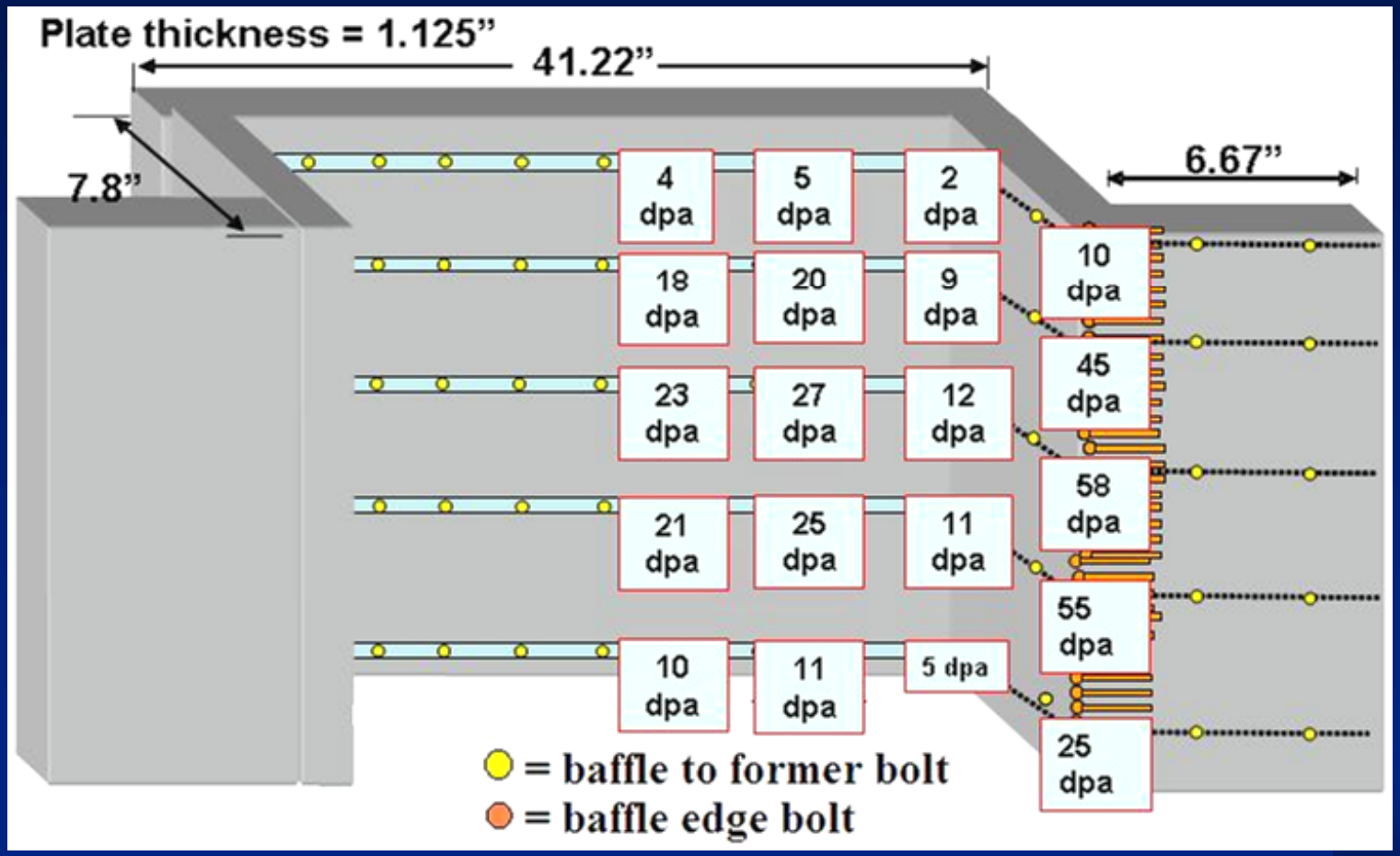
Data About Fluence Levels and Temperatures of the Internals

Irradiation Schematic Representation



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IASCC in PWRs

- Tests with high and low dpa materials.
- Proposed testing and evaluation
  - Task 1. Material Identification and Characterization, Including Cutting Plan, and Radiation and Temperature Analysis
  - Task 2. Extraction of Materials from the Zorita Plant, Shipping of Material from Zorita to the U. S., Machining of Test Specimens and Transfer to Test Facilities
  - Task 3. Fracture Toughness Measurements, Tensile Strength Measurements,
  - Task 4. Irradiation Assisted Stress Corrosion Cracking (IASCC) Initiation Testing
  - Task 5. IASCC Crack Growth Rate (CGR) Testing
  - Task 6. Mechanical, Micro-structural and Micro-chemical Characterization
  - Task 7. Data Analysis
  - Task 8. Waste Disposal

Conclusion

- Continue the Research Program to address IASCC in PWR environment

➔ It is Planned to

- determine crack growth rates;
- determine threshold limits (at low dose) and the
- saturation effects (at high dose) for the onset of IASCC due to the exposure to Neutron Fluences to high dose levels.