Materials Research Projects Presentation to the Advisory Committee on Reactor Safeguards Materials, Metallurgy and Reactor Fuels Subcommittee

Aladar Csontos
Timothy Lupold

July 7, 2009

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

Overview

- Materials research conducted by two branches:
  - RES/DE/CIB: Component Integrity Branch
    - Fracture Mechanics, NDE, and Safety Assessments
  - RES/DE/CMB: Corrosion & Metallurgy Branch
    - Corrosion, Metallurgy, and Advanced Reactors
- Research related to needs of other NRC offices
  - Generally, through a User Need Request (UNR)
  - ACRS letter for residual stress research program
  - Staff Requirements Memorandum from Commission

RES/DE/CIB Research Programs

RES/DE/CIB: Overview

- Materials research supporting NRR:
  - Evaluating short/long term regulatory issues:
    - UNRs and expedited needs
    - ‘Ready to serve’ efforts include research
- Topical areas:
  - Component integrity assessments:
    - Piping/CRDM/Reactor pressure vessel
    - Probabilistic/deterministic fracture mechanics
    - PWSCC mitigation and residual stress validation
    - High Density Polyethylene (HDPE) piping research
  - Non-Destructive Evaluation (NDE):
    - Dissimilar metal welds and advanced techniques
    - HDPE piping

Topical Areas: Projects

- Component integrity assessments:
  - Dissimilar metal welds (piping and CRDM)
    - PWSCC mitigations, residual stresses, & xLPR
    - N6319/6433/6637/6547/6774/6360/6687/6829/6438
  - HDPE piping: failure mechanisms
    - N6637/6433
  - Reactor pressure vessel: fracture mechanics
    - N6578/6438
- NDE: Metallic and HDPE piping
  - N6398/6319/6593

Component Integrity Assessment Programs

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

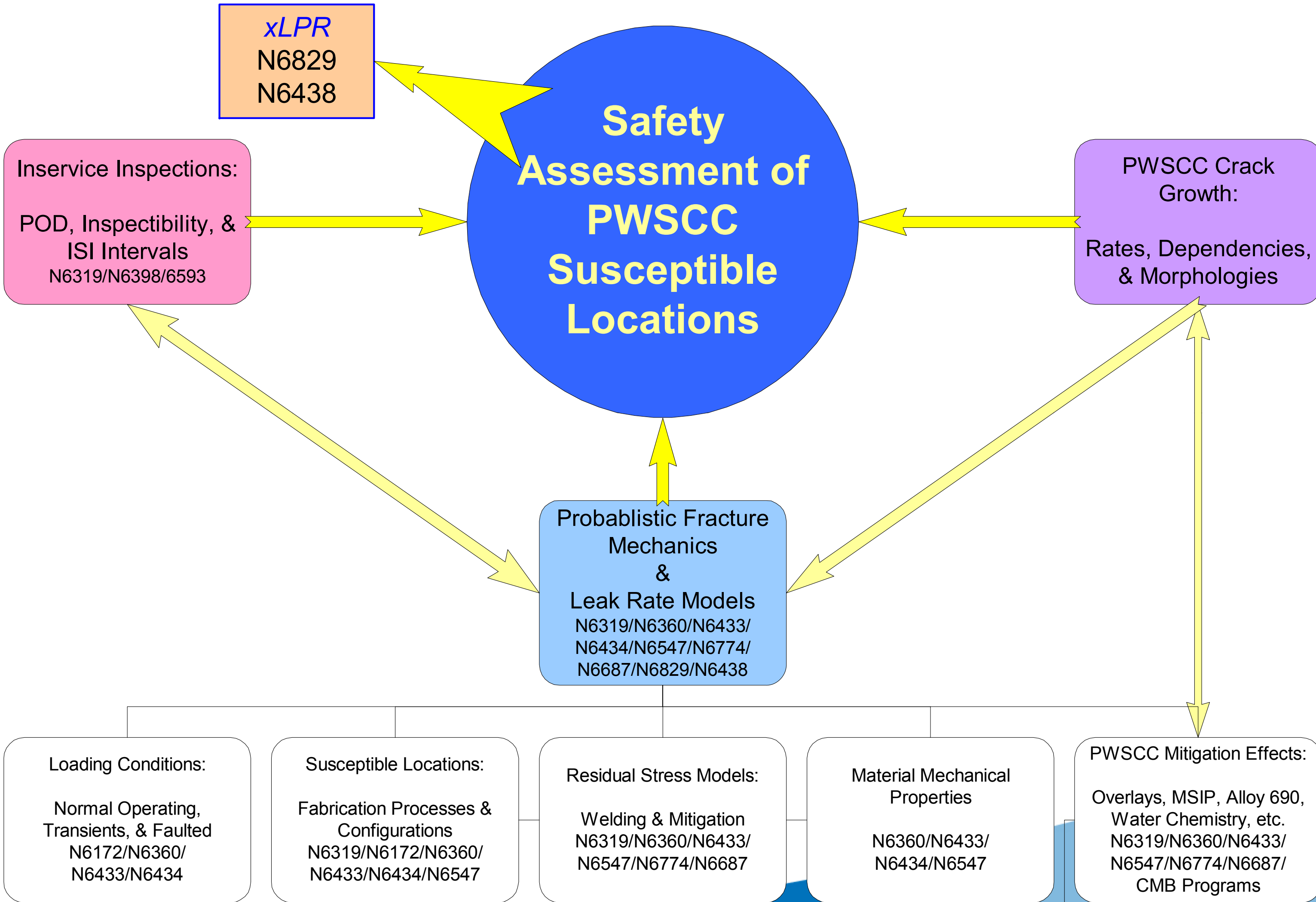
- **Component Integrity Assessments**
  - **Long Term Research Programs**
    - **Reactor Pressure Vessel:**
      - PTS 10CFR50.61a
      - Regulatory Guide 1.99
      - 10CFR50 App. G/H
    - **Piping:**
      - xLPR Project (LBB)
      - PWSCC in DM Welds
      - HDPE Code Support
    - **CRDM:**
      - Probabilistic Review
      - Flaw Evaluations
      - PWSCC Mitigations
  - **Short Term Regulatory Evaluations**
    - **Piping:**
      - PWSCC Mitigations
      - Flaw Evaluations
      - HDPE Code Support

PWSCC Piping Research

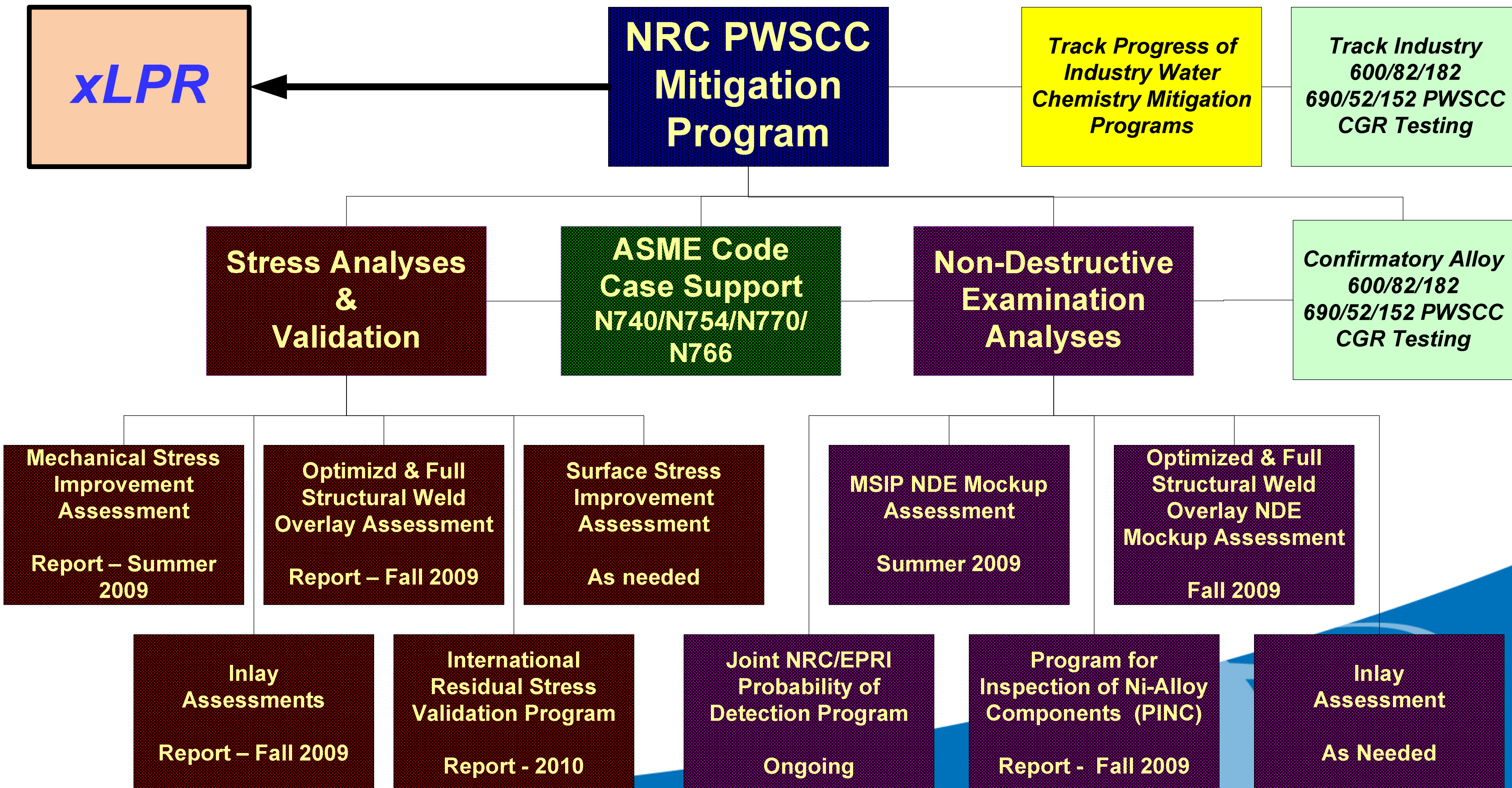
U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- Purpose:
  - Support NRR/NRO in considering appropriate regulatory requirements to address PWSCC in xLPR piping systems.
- Objectives:
  - Short Term (1-2 years):
    - Evaluate the near-term adequacy of industry's mitigation activities
    - Initial probabilistic fracture mechanics (PFM) pilot study
  - Long Term (3-5 years):
    - Complete and validate a regulatory PFM tool to assess xLPR in piping systems susceptible to active degradation mechanisms (PWSCC)
- Collaborations across key technical groups critical to developing the probabilistic xLPR tool
  - NDE, corrosion, fracture mechanics, fluid mechanics, metallurgy

PWSCC Piping Research



PWSCC Piping Research



Piping WRS Validation Research Program

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

WRS Validation Research Program

- Fabrication of DM Weld Mockups with Mitigations:
  - N6547/N6319: PNNL
- Residual Stress Finite Element Modeling:
  - N6433/N6637: Emc^2
  - N6360/N6687: Battelle
- WRS Measurements
  - N6774: ORNL (XRD/ND/IHD)
  - N6864: Veqter Ltd. (DHD)

Phase I-IV Piping WRS Validation Research Program

- Background:
  - Component integrity analyses for PWSCC in DM welds showed that the results were highly dependent upon WRS profiles
  - ACRS letter dated 10/19/07 supported further WRS research
- Purpose:
  - Refine WRS FEA model development for 82/182 DM welds through sequential development from Phase I to IV
  - Develop reasonable assurance that WRS FEA models are defensible through a blind validation using well controlled mockups to various WRS measurement testing techniques
- Expected Outcome:
  - Blind validation of WRS FEA models using well controlled mockups focusing on through-wall axial & hoop stresses
  - Develop uncertainty distributions in WRS modeling

Phase I-IV Piping WRS Validation Research Program

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

- Phase I: EPRI Simple Plates & Cylinders
  - EPRI Lead: Mockup fabrication, WRS measurements, & project aims
  - Purpose: Refine WRS FEA model development by varying welding parameters and validate models to ND and DHD techniques
- Phase II: NRC PZR Mockups (Intn'l. WRS & FSWOL)
  - NRC Lead: Mockup fabrication, WRS measurements, & project aims
  - Purpose: Blind validation of mockups to XRD, ICHD, DHD, and ND
- Phase III: EPRI WNP III Safety & Relief PZR Components
  - EPRI Lead: Mockup fabrication, WRS measurements, & project aims
  - Purpose: Blind validation of real components to XRD, ICHD, and DHD
- Phase IV: EPRI WNP III Cold Leg OWOL Validation
  - EPRI Lead: Mockup/OWOL design and fabrication, WRS measurements, FEA modeling, and project direction and planning
  - Purpose: Blind validation of OWOL process to XRD, ICHD, and DHD

Specific Research Programs

N6360: Evaluation of Leak-Before-Break Criteria

- Vision:
  - Evaluate industry proposed PWSCC mitigation strategies of full structural weld overlay and mechanical stress improvement (MSIP) for current LBB systems to ensure that the probability of fluid system piping rupture remains extremely low
  - Benchmark and validate finite element models of residual stress profiles of representative DM welds with MSIP and weld overlay configurations
  - Quantify the changes in operational risk that PWSCC susceptibility creates for acceptability of LBB criteria as found in Appendix A of GDC-4
- Basis:
  - PWSCC has been determined to be the cause of in-service failures of nickel-based alloy DM welds and the source of limiting indications found in reactor coolant pressure boundary components
  - Supports UNRs NRR-2002-018, NRR-2002-020, NRR-2005-011, NRR-2006-006
- Execution: Estimated completion - FY2009
- Deliverables:
  - Technical letter reports on the effectiveness of full structural weld overlays and MSIP to mitigate PWSCC growth in DM welds
- Coordination: Battelle Lead with PNNL, Emc^2, EPRI, MRP

N6433: Component Integrity Analytical Support

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

- Vision:
  - Develop more realistic flaw evaluation tools and fracture mechanics models to assess the risk of failure and leakage caused by PWSCC of nickel-base alloys
  - Conduct initial studies into integrity issues for polyethylene piping materials
- Basis:
  - PWSCC has been determined to be the cause of in-service failures of nickel-based alloy DM welds and the source of limiting indications found in reactor coolant pressure boundary components
  - Supports UNRs NRR-2002-018, NRR-2002-020, NRR-2005-011, NRR-2006-006
  - Relief requests have been submitted by licensees that proposed the use of polyethylene piping for safety-related applications
  - Supports NRR-2006-007 for polyethylene piping
- Execution: Estimated completion - FY2009
- Deliverables:
  - Improved tools for performing fracture mechanics evaluations of DM welds, assessments of the risk of failure and leakage of DM welds due to PWSCC and preliminary assessments of the integrity of polyethylene piping
- Coordination: Emc^2 Lead with Battelle, PNNL, EPRI, MRP

N6687: Reactor Coolant Pressure Boundary Analyses

- Vision:
  - Provide flexible technical analyses to NRR to develop and/or confirm the technical bases for future regulatory decisions related to reactor coolant pressure boundary and LBB system integrity with PWSCC mitigation assessments
  - Benchmark and validate finite element models of residual stress profiles of Phases I-IV of the NRC/EPRI WRS research program
- Basis:
  - PWSCC has been determined to be the cause of in-service failures of nickel-based alloy DM welds and the source of limiting indications found in reactor coolant pressure boundary components
  - Supports UNRs NRR-2002-018, NRR-2002-020, NRR-2005-011, NRR-2006-006
- Execution: Estimated completion - FY2011
- Deliverables:
  - Technical letter reports on the effectiveness of optimized weld overlays and other mitigation methods to mitigate PWSCC growth in DM welds
  - Improved tools for performing fracture mechanics evaluations of DM welds, assessments of the risk of failure and leakage of DM welds due to PWSCC
- Coordination: Battelle Lead with PNNL, Emc^2, EPRI, MRP

N6637: Pressure Boundary Integrity Analyses & Support

- Vision:
  - Support ASME code case review and confirmation for nickel-base alloy fabrication and inspection, polyethylene piping structural integrity, flaw tolerance, joining and inspection and other ASME Code-related activities determined to be necessary to support NRC regulatory considerations
  - Perform fracture mechanics based flaw tolerance evaluations of RCPB components including nickel base alloy welds and polyethylene piping base materials and joints
  - Confirmation of 50 year service life of high density polyethylene piping material, including fusion joints, considering the effects of flaws on slow crack growth rate at elevated service temperatures

N6637: Pressure Boundary Integrity Analyses & Support

- Basis:
  - Supports UNRs NRR-2002-018, NRR-2005-011, NRR-2006-006
  - New issues arose from prior research to support NRR-2002-018 that were out of scope of the existing contract, hence, N6637 developed
  - Relief requests have been submitted by licensees that proposed the use of polyethylene piping for safety-related applications
  - Supports NRR-2006-007 for polyethylene piping
- Execution:
  - Current research efforts conducted under job code N6637
  - Prior research conducted under job codes N6363 and N6433
- Deliverable:
  - Expedited and ongoing ASME code case reviews
  - Technical letter reports on component integrity analyses and model assessments used to predict the service life of polyethylene piping
- Coordination:
  - Industry/Licensees/EPRI/MRP/ASME

N6547: Weld Residual Stress Validation

- Vision:
  - Develop reasonable assurance that WRS FEA models are defensible through a blind validation using well controlled mockups to various WRS measurements
- Basis:
  - Component integrity analyses for PWSCC in DMWs showed that the results were highly dependent upon WRS profiles
  - PWSCC has been determined to be the cause of in-service failures of nickel-based alloy DM welds and the source of limiting indications found in reactor coolant pressure boundary components
  - Supports UNRs NRR-2002-018, NRR-2002-020, NRR-2005-011, NRR-2006-006
- Execution: Estimated completion - FY2009
- Deliverables:
  - Technical letter report on available WRS measurement techniques
  - Design and fabrication of two mockups for international round robin tests
- Coordination: EPRI and over 20 international WRS modeling and measurement groups for international round robin study

N6774: WRS Measurements and Assessments

- Vision:
  - Develop reasonable assurance that WRS FEA models are defensible through a blind validation using well controlled mockups to various WRS measurements
- Basis:
  - Component integrity analyses for PWSCC in DMWs showed that the results were highly dependent upon WRS profiles
  - PWSCC has been determined to be the cause of in-service failures of nickel-based alloy DM welds and the source of limiting indications found in reactor coolant pressure boundary components
  - Supports UNRs NRR-2002-018, NRR-2002-020, NRR-2005-011, NRR-2006-006
- Execution: Estimated completion - FY2010
- Deliverables:
  - WRS Measurements using the following techniques: X-ray diffraction, neutron diffraction, and incremental hole drilling for Phases II-IV of the WRS program
  - Provide website for international round robin uploads and downloads
  - Technical letter report on results of the international round robin study
- Coordination: EPRI and over 20 international WRS modeling and measurement groups for international round robin study

N6864: Deep Hole Drilling WRS Measurements

- Vision:
  - Develop reasonable assurance that WRS FEA models are defensible through a blind validation using well controlled mockups to various WRS measurements
- Basis:
  - Component integrity analyses for PWSCC in DMWs showed that the results were highly dependent upon WRS profiles
  - PWSCC has been determined to be the cause of in-service failures of nickel-based alloy DM welds and the source of limiting indications found in reactor coolant pressure boundary components
  - Supports UNRs NRR-2002-018, NRR-2002-020, NRR-2005-011, NRR-2006-006
- Execution: Estimated completion - FY2010
- Deliverables:
  - Through-wall WRS measurements using the proprietary Deep Hole Drilling technique for Phases II-IV of the NRC WRS program
  - Technical letter report on results of the measurements
- Coordination: EPRI and over 20 international WRS modeling and measurement groups for international round robin study

N6438: Probabilistic Pressure Boundary Safety Assessment

- Vision:
  - Develop more realistic flaw evaluation tools and fracture mechanics models to assess the risk of failure for reactor pressure vessels through validating and benchmarking physically-based material models and generic analysis methodologies to provide more robust prediction and assessment tools than current-day design-specific empirical and experimental approaches
- Basis:
  - Develop the technical bases for currently-identified RPV integrity needs:
    - Plant-specific PTS analysis guidance (Regulatory Guide 1.154)
    - Heat up and cool down limits (10CFR50 Appendix G)
    - Surveillance requirements (10CFR50 Appendix H)
    - Embrittlement trend prediction (Regulatory Guide 1.99)
    - MODULAR probabilistic code for structural integrity assessments
  - Supports UNR NRR-2007-001
- Execution: Estimated completion - FY2013

N6438: Probabilistic Pressure Boundary Safety Assessment

- Deliverables: Data, analyses, and reports for
  - Regulatory Guide 1.154 & 1.99
  - 10CFR50 Appendix G & H
  - Current embrittlement trends for 40+ & 60+ years
  - Modular code for component integrity assessment of structures
- Coordination:
  - EC Projects
    - PERFORM-60: Computational platform to project embrittlement & assess structural integrity to 60 years
    - STYLE: Assessment protocols for non-RPV components (e.g., piping)
    - PC-1: High fluence and flux effects on embrittlement
  - IAEA Projects
    - CRP-8: Master curve
    - CRP-9: PTS

N6578: Pressure Boundary Materials

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

- Vision:
  - Develop and validate predictive material property models aimed at the continued development, refinement, and generalization of structural integrity assessments
  - These models aim to ensure that NRC staff have tools to independently assess licensee submittals and to maintain the safety of the operating fleet by ensuring that all active embrittlement mechanisms and potential failure modes are appropriately accounted for
- Basis:
  - Accurate prediction of RPV structural integrity relies on data and models that describe the mechanical behavior of RPV materials across a spectrum of loading rate and temperature conditions, and how this behavior is influenced by the effects of neutron irradiation.
  - In recent years, trends across a wide variety of ferritic steels are now sufficiently well accepted that they are used in the probabilistic fracture mechanics computer codes as part of its risk-informed development of regulatory products. Nevertheless, certain datasets need to be developed to confirm the predictions made and the models that have been developed based on amalgam of smaller datasets, and to quantify and refine the scatter/uncertainty characterization adopted by current models. Because failure is usually predicted by probabilistic fracture mechanics models in the tails of toughness distributions, these models aid in reducing the uncertainty in failure predictions made by PFM computer codes
  - Supports UNR NRR-2007-001

N6578: Pressure Boundary Materials

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

- Execution: Estimated completion - FY2014
- Deliverable: Technical letter reports detailing the following items:
  - Heat treatment, metallurgical conditions, cleavage crack initiation and arrest toughness, ductile fracture toughness, and Charpy V-notch energy characterization of low and high transition temperature material
  - Design, analysis, and test results from an improved crack arrest specimen
  - Effect of prior hardening on the plastic flow and T_o properties of ferritic material
  - Cleavage crack initiation (K_{Jc}) characterization of the low and high transition temperature material using shallow crack specimens
  - Effect of elevated loading rate on fracture toughness characterization of the low and high transition temperature material
- Coordination: Carderock is the Lead with ORNL, Naval Air Systems Command, and various university and research organizations involved in fracture mechanics

N6319: PWSCC in Leak-Before-Break (LBB) Systems

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

- Vision:
  - Develop strategies for managing PWSCC in LBB systems to ensure that the probability of fluid system piping rupture remains extremely low (GDC-4)
  - Assess POD in DM welds through collaboration with EPRI PDI
  - Determine ability to detect existing cracks in post-MSIP and overlaid welds
- Basis:
  - PWSCC has been determined to be the cause of in-service failures of nickel-based alloy dissimilar metal (DM) welds and the source of limiting indications found in reactor coolant pressure boundary components
  - Supports UNRs NRR-2002-018, NRR-2002-020, NRR-2005-011, NRR-2006-006
- Execution: Estimated completion - FY2010
- Deliverables: Technical letter reports:
  - Overall PWSCC management strategy
  - NDE effectiveness of PWSCC mitigations
  - Reliability of NDE to detect PWSCC flaws in DM welds (POD curves)
  - Fabrication of Overlay and MSIP Mitigation NDE/WRS mockups
- Coordination: PNNL Lead with EPRI, MRP, Battelle, Emc^2

N6593: Assess Emerging NDE For DM Welds

- Vision:
  - Provide data and correlations necessary for NRC staff to independently evaluate licensee ISI programs for assessing integrity of DM welds
  - Evaluate current and emerging NDE techniques that licensees may be planning to apply for ISI of passive components.
  - Review and analyze results from the initial international Program for Inspection of Nickel Alloy Components (PINC) Round Robin Testing
  - Establish extended international collaborative partnerships (PINC II)
  - Expand upon the “Atlas” information tool from PINC
  - Use Results to Assess Current ISI Inspection and Acceptance Criteria Specified in ASME BPV Code Section XI
- Basis:
  - International PINC program with 7 participants was successful in addressing the effectiveness of some NDE techniques at finding PWSCC cracks in Alloy 600 and 82/182 components, i.e. particularly bottom mounted instrumentation penetration tubes and DM welds
  - PINC ended in FY09 with some issues unresolved
  - Supports UNR NRR-2006-006

N6593: Assess Emerging NDE For DM Welds

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

- Basis (continued):
  - PINC II or **P**rogram to **A**ssess **R**eliability of **E**merging **N**ondestructive **T**echniques for DM welds (PARENT) will focus on tight cracks, including PWSCC and hot cracks in welds in piping and in other components
  - PINC II or PARENT program (~10 participants) is designed to address some of the issues remaining from PINC and to look forward to new challenges for emerging NDE technologies
  - Supports UNR NRR-2006-006
- Execution: Estimated completion - FY2013
- Deliverables: Technical letter reports on the following:
  - Large-diameter inside-surface DM NDE techniques
  - Evaluate new techniques for rapid-growth degradation mechanisms
  - Atlas database tool with PWSCC crack morphology and corresponding NDE results, developed under the PINC program, will be reviewed, applied and extended to support NRC inspectors for ISI
  - NDE test results from mockups containing representative simulated and fabrication flaws
- Coordination: PNNL Lead with ~10 international participants

N6398: Reliability of NDE for NPP Inservice Inspection

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

- Vision:
  - Evaluate accuracy and reliability of NDE methods used for ISI
  - Provide info to assess adequacy of proposed industry changes to ISI programs
  - Evaluate effectiveness of ISI techniques for detecting service degradation, e.g.:
    - PWSCC in Alloy 600, 82, 182 DM welds and J-groove penetrations
    - IGSCC in austenitic welds
    - Potential degradation in cast stainless steel and weldments
    - High density polyethylene piping (HDPE)
    - Weld overlays/cladding/in-lays
    - Reactor internal examinations
    - Vessel Penetrations (CRDM and BMI nozzles)
  - Provide technical assistance to NRC program offices on as-needed basis
- Basis:
  - PWSCC has been determined to be the cause of in-service failures of nickel alloy DM welds and the source of limiting indications found in reactor coolant pressure boundary components
  - Relief requests submitted by licensees for use of HDPE piping for safety-related applications

N6398: Reliability of NDE for NPP Inservice Inspection

- Basis (continued):
  - Supports NRR UNRs:
    - Metallic Piping: NRR-2002-018, NRR-2002-020, NRR-2005-011, NRR-2006-006, NRR-2006-012
    - Polyethylene Piping: NRR-2006-007
    - Vessels: NRR-2006-012
- Execution: Estimated completion - FY2012
- Deliverables: NUREG/CRs and technical letter reports on the effectiveness of ISI techniques for detecting service degradation, e.g.:
    - PWSCC in Alloy 600, 82, 182 DM welds and J-groove penetrations
    - IGSCC in austenitic welds
    - Potential degradation in cast stainless steel and weldments
    - HDPE piping
    - Weld overlays/cladding/in-lays
    - Reactor internal examinations
    - Vessel Penetrations (CRDM and BMI nozzles)
- Coordination: PNNL Lead with EPRI, MRP, IRSN through MOU

RES/DE/CMB Research Programs

RES/DE/CMB: Overview

- Materials research supporting NRR:
  - Evaluating short/long term regulatory issues:
    - UNRs and expedited needs
    - ‘Ready to serve’ efforts include research
- Topical areas:
  - Pro-active Management of Materials Degradation
  - Corrosion/Metallurgy:
    - Environmentally Assisted Corrosion
    - Primary Water Stress Corrosion Cracking
    - Stress Corrosion Cracking of Stainless Steel in Marine Environments
  - Steam Generator Tube Integrity

Proactive Management of Materials Degradation

Pro-active Management of Material Degradation (PMMD) – N6029

- The objective of this program is to provide technical support to NRC staff in developing information regarding materials degradation mechanisms, inspection or monitoring, and behavior of materials. The goal is to proactively address potential future degradation in operating plants to avoid failures and to maintain integrity and safety. This work will become part of the activities of an international cooperative research group whose function will be to conduct research that is needed and share the results for implementation of programs to proactively manage materials degradation. The information developed will provide NRC a foundation to implement appropriate regulatory actions to keep materials degradation from adversely impacting safety and to evaluate licensee's programs for the proactive management of materials degradation.
- The research is to:
  - Develop a master program for the proactive management of materials degradation
  - Establish international collaborative partnerships
  - Expand upon the information tool that started under JCN N6019
  - Identify research needed to address/establish a level of understanding of the degradation processes to ensure the ability to proactively manage degradation

Pro-active Management of Material Degradation (PMMD) - Basis

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

- Degradation of materials in certain nuclear reactor components progressed to the point where the reactor pressure boundary and defense-in-depth features were compromised.
- Many plants are also applying for increases in power rating.
  - could increase the likelihood of materials degradation and underline the interest in proactive management.
- The majority of the U.S. reactor fleet is applying for license renewal to extend the operating life from the current 40 years to 60 years, and there is now active interest in extending the operating life to beyond 60 years.
- Material degradation processes from known and emerging mechanisms and those previously experienced probably will continue to affect susceptible plant components and may increase in occurrence as the operating fleet of reactors continues to age.
  - With aging nuclear power plants, degradation that was not an issue during the initial years of operation may become an important process during later operation.

Pro-active Management of Material Degradation (PMMD) - Approach

- Establish a refined plan
- Establish international collaboration
- Target items of low knowledge and high/medium susceptibility to degradation
- Create an information tool

Environmentally Assisted Cracking

SCC of Alloys 690/52/152 N6782

- The objective of this program is to obtain crack growth rate data for Ni-base alloys, with emphasis on those with higher Cr content, specifically Alloy 690 and its matching weld fillers Alloy 152, Alloy 52, and Alloy EN 52H. These alloys are likely replacements for Alloys 600/82 /182. Alloy 690 and its weld metals have been reported by industry to be resistant to SCC.
- In order to accomplish this objective, autoclave systems in suitable load frames and the associated water supply, conditioning and pressurization subsystems will be operated reliably.
  - These autoclave systems will allow testing under simulated and/or accelerated (e.g., increased temperature, more aggressive environments, increased load range or load interaction effects) PWR and BWR conditions.
  - Direct current electric potential drop (dcpd) methods will be used to acquire crack extension data, and effective reference electrodes will be used to acquire corrosion potential data.
  - The systems will be capable of both dynamic and static loading with load control to better than 1%.

SCC of Alloys 690/52/152 Basis

- Primary water stress corrosion cracking (PWSCC) in nickel-base alloy primary pressure boundary components is a significant safety concern due to the potential for reactor pressure boundary leaks and the associated potential of boric acid corrosion of low alloy steels and the development of flaws in piping or welds. Either condition, depending on the size and location of the flaws, could result in a significant loss of coolant accident. The use of Alloy 690 and associated weld metals, Alloy 52 and 152 have been reported by industry to be resistant to PWSCC. Although the issue of PWSCC susceptibility is being addressed by industry, user need request NRR-2006-006 specifically identifies the need to obtain PWSCC growth rates of these resistant alloys to determine the validity and acceptability of licensee flaw analyses, and to support regulatory inspection requirements.

SCC of Alloys 690/52/152 - Approach

- Stress-corrosion, CGR systems designed specifically for testing in high-temperature, simulated LWR coolant will be used.
- SCC behavior of Ni base alloys in PWR primary water will be evaluated.
- Each CGR test system will be able to test two samples (0.5T to 1T compact tension) simultaneously at temperatures up to 360°C.
- The autoclaves and the water make-up system will effectively simulate high-purity BWR and PWR water as well as control levels of oxygen, hydrogen and selected impurities.
- The systems will have active dcpd for crack-length measurement and load/K-control plus in-situ measurement capability for temperature and electrochemical potential (ECP).
- The hydrogen over pressure will be varied to evaluate the effect of ECP on crack growth rates.
- At least one CGR system will be capable of testing metallic alloys with low activity levels.

Properties of CRDM Welds – N6783

- The objective of this program is to conduct nondestructive testing, metallurgical evaluations, leak path assessment, mechanical tests, and crack growth rate tests on CRDM nozzles and nozzle welds using material that has been in service.
- Materials examined will include Nozzle 63 from North Anna Unit-2. Material from Davis Besse Nozzle 1 may also be examined.
  - Test specimens will be obtained with orientations and geometries that allow the characterization of the Alloy 600 CRDM nozzle materials and the Alloy 82/182 J-groove weld and butter.
  - Information from mechanical tests will be used to obtain yield and tensile strengths necessary to establish conditions for crack growth rate tests.
  - The results for the crack growth rate measurements will be compared to data for Alloys 600 and Alloys 82/182 obtained from a variety of specimens including the previously tested material from Davis-Besse and V.C. Summer.

Properties of CRDM Welds – Basis

- More than 30 head replacements have occurred at operating PWRs, however, only a limited number of materials that have actually been in service are available for characterization and testing. In the 2001 refueling outage, some of the North Anna Unit-2 nozzles were repaired using Alloy 52/152 including nozzles 63 and 51. In the 2002 refueling outage, 63 of 65 J-groove welds had indications and 42 of these welds would require repair. At that time, the utility decided to replace the reactor head. Previously, EPRI sponsored the removal and analysis of several nozzles. Nozzle 63 is available to the NRC to conduct independent tests. Some prior characterization of Nozzle 63 has been performed including visual examinations, a volumetric leak path assessment and surface examination of the J-groove welds that identified axial indications.

Properties of CRDM Welds – Approach

- Non destructive examination of the nozzles will be conducted to the requirements of 10 CFR 50.55a(g)(6)(ii)(D) to determine the as-left condition of the nozzle and welds to position and size indications, as well as perform a volumetric leak path assessment.
- The results of the non destructive evaluation should be compared to the previous examination results as well as identify regions where specimens will be extracted for additional analyses.
- After the nozzle has been removed from the low alloy steel head material, a visual inspection should be conducted of the low alloy steel head surface in the leak path area defined by information obtained from the current and previous volumetric leak path assessments.
- Remaining material will be used to obtain samples for metallurgical analyses, mechanical test specimens and crack growth rate specimens using specimens machined from the Alloy 600 CRDM nozzle material, and the Alloy 82/182 J-groove weld and butter.
- Crack growth rates will be compared to published data for previous laboratory tests as well as data obtained from the testing of the Davis-Besse and V.C. Summer materials.

Environmentally Assisted Cracking (EAC) – N6519

- The objective of this project are to:
  - evaluate the susceptibility of austenitic SS to irradiation-assisted stress-corrosion cracking (IASCC) in BWRs as a function of the fluence level, material chemistry, welding process, fabrication history, and water chemistry.
  - evaluate the susceptibility of austenitic SS core internals to IASCC in pressurized water reactors (PWRs) as a function of the fluence, water chemistry, material chemistry, and cold-work. At this time, the database and mechanistic understanding of IASCC under the PWR conditions of higher temperature and higher fluence are very limited.
  - provide the NRC with technical data and analytical methods on the cracking of nickel-alloy components and welds necessary to independently estimate CGRs in reactor components for regulatory determinations of residual life, inspection intervals, repair criteria, and effective countermeasures for reactor internal components.

Environmentally Assisted Cracking (EAC) – Basis

- Neutron radiation embrittlement of reactor core internal components constructed of cast SSs is considered significant if the neutron fluence is greater than 1 x 10^{17} n/cm2 (E >1 MeV). This conservative value for the threshold fluence has been proposed for cast SS internals because the possible synergistic effects of neutron radiation and thermal embrittlement are not known.
- For cast SSs with duplex austenite/ferrite structure, a loss of fracture toughness can occur due to three processes: (a) thermal embrittlement of ferrite, (b) radiation embrittlement of ferrite, and (c) radiation embrittlement of austenite.
- The kinetics of thermal embrittlement is well known and the kinetics of radiation embrittlement may be estimated based on vessel embrittlement data. However, concurrent exposure to high temperature and neutron fluence could result in a synergistic effect that leads to more rapid embrittlement than would be expected for either of the two processes individually.
- Nickel alloys, including Alloy 600, Alloy 690, and Alloy X-750 and welds using other nickel-base alloys (weld metals 82/182 and 52/152) appear to be susceptible to primary water stress-corrosion cracking (PWSCC) to varying degrees. Evaluations are needed of the time to form axial and circumferential cracks and the CGRs in such components and their welds under applicable service conditions.

Environmentally Assisted Cracking (EAC) – Approach

- Crack growth and fracture toughness J-R curve tests will be performed on SS base metal and weld heat affected zone (HAZ) material to further establish the effects of fluence level, material chemistry, thermal treatment, and welding process on IASCC.
- Models and codes developed under CIR-II and from industry sources will be benchmarked and used in conjunction with this work.
- Slow-strain-rate-tensile, CGR, and fracture toughness J-R curve tests will be conducted on austenitic SSs that have accumulated fluences typical of PWR components.
- CGR tests will be performed on a few compositions of thermally treated Alloy 690 and Alloy 152 weld, including the Alloy 690 HAZ material from Alloy 690/152 weld.
- Also, tensile property data will be obtained on thermally treated Alloy 690 and Alloy 152 weld metal at temperatures from room temperature up to 870°C.
- Furthermore, the possible deterioration of mechanical properties of low-alloy steel HAZ region will also be investigated.

Environmentally Assisted Cracking – Reactor Internals – N6818

- While the objective of the Environmentally Assisted Cracking (EAC) program at a global level is to address the regulatory concerns arising from irradiation induced materials issues and assure structural and functional integrity of reactor core internals, the objective of this supporting project is:
  - to assist in development of a research plan following a thorough review of the available literature based on the research performed by the NRC and the industry, and
  - to review the MRP 227/175 and the PWR Internals AMP and provide a detailed analysis projecting the gaps that must be addressed in both the MRP and AMP documents prepared by the Industry.

Environmentally Assisted Cracking – Reactor Internals – Basis

- Austenitic stainless steels (SSs) are used extensively as structural members in the internal components of light water reactor (LWR) pressure vessels because of their relatively high strength, ductility, and fracture toughness. However, exposure to neutron irradiation for extended periods changes the microstructure (radiation hardening) and microchemistry (radiation-induced segregation or RIS) of these. Irradiation leads to significant increase in yield strength and loss of ductility, degradation of fracture toughness, radiation embrittlement, susceptibility to irradiation assisted stress corrosion cracking (IASCC), void swelling, and radiation creep relaxation.
- The major concern regarding the structural and functional integrity of core internal components is IASCC of austenitic SSs. In addition, although radiation embrittlement has not been considered in the design of LWR core internal components constructed of austenitic SSs, it has become an important consideration in ensuring that adequate structural integrity exists over the license renewal period. Another issue related to high neutron exposures that are relevant for PWRs is void swelling, and its effect on fracture toughness.

Environmentally Assisted Cracking – Reactor Internals – Approach

- Document the important conclusions from earlier studies that identify (i) the materials and environmental conditions that lead to significant effect of neutron irradiation, (ii) establishment of the crack growth rates (CGR) for core internal materials (iii) the potential of radiation embrittlement under BWR and PWR operating condition including the synergetic effects of thermal and neutron embrittlement of cast SS, (iv) the effects of void swelling including its effect on fracture toughness and (v) the effectiveness of the methods proposed by industry to mitigate radiation effects and the deficiencies/ the knowledge gaps in the existing research.
- Propose research plans that addresses the issues found as research program gaps.
- Propose additional test plans that will aid in fulfilling the NRC objective to develop regulations for the license renewal for life beyond 60 years.
- Review and assess the industry's reactor internal aging management program.

Halden: Environmentally Assisted Cracking – Y6270

- The Halden Reactor Project has been in operation for 50 years and is the largest NEA joint project. It brings together an important international technical network in the areas of nuclear fuel reliability, integrity of reactor internals, plant control/monitoring and human factors. The program is primarily based on experiments, product developments and analyses carried out at the Halden establishment in Norway, and is supported by 130 organizations in 17 countries.
- The material work encompasses the embrittlement and cracking Behavior of internal reactor materials.
- Key program areas are:
  - plant lifetime assessments (reliability of internals).

Zorita Internals Research Project-K6202

- *OECD Nuclear Energy Agency*
  - Committee on the Safety of the Nuclear Installations (CSNI)
- Cooperative research project on ex plant materials from José Cabrera NPP (Zorita NPP)
- José Cabrera NPP (Zorita NPP) was shutdown on April 2006, and the owner of Zorita NPP, has offered materials of potential interest in R&D
- The research could be focused on properties of long time operating and in-plant irradiated materials
- The current proposal of this cooperative research project is limited to Core Internals
- Some important features of these internals are: 26,5 EFPY, high fluence and thick sections The Deliverables will be the results of the tests performed
- Potential applications are up to each participant, some potential applications: Licensing purposes, Inspection programs, Lifetime management and Lifetime extension

Steam Generator – Tube Integrity Program

Steam Generator Tube Integrity Project - N6582

- The overall objective of this program is to provide the experimental data and correlations to permit the NRC staff to independently assess licensees' programs for evaluating the integrity of steam generator (SG) tubes as plants age. The research program results will also support the office of Nuclear Reactor Regulation (NRR) in a variety of regulatory decisions and licensing actions. Currently, the program objectives of NRR envelop the needs of the Office of New Reactors.

Steam Generator Tube Integrity Project - Basis

- Steam generator tubes provide an integral part of the reactor coolant pressure boundary. They serve as a barrier to isolate the radiological fission products in the primary coolant from the secondary coolant and environment. Knowledge of SG degradation phenomena has evolved along with SG designs and the various SG chemistries. Degradation of SG tubes has resulted from corrosion and wastage, pitting, denting, stress-corrosion cracking, and intergranular attack. Both the primary and secondary sides of the SG tubes have experienced cracking. Axial cracks, as well as circumferential cracks, have occurred. Tubes with cracks, if not detected and either removed from service or repaired, may rupture and possibly release radiological products.
- Degradation of SG tubes is an issue that continues to pose a potential safety risk to the public. This degradation has occurred as (a) intergranular stress-corrosion cracking (IGSCC) in the free span of tubes, (b) intergranular attack and IGSCC (IGA/IGSCC) at the tube support plate and egg-crate location and in regions of sludge accumulation, and (c) axial and circumferential cracking at the top of the tubesheet.

Steam Generator Tube Integrity Project - Approach

- This program builds upon the findings and conclusions of previous research programs as well as recent licensee operating experiences. At the beginning of the program, the work will focus on developing plans of action to address issues that were identified during the previous SG tube integrity program. Then, the research will incorporate additional topics which are important to evaluating SG tube integrity. These new topics arose largely from licensee operating experience.
- Research tasks under this program include:
  - Assessment of inspection techniques and reliability
  - Tube integrity and predictions
  - Degradation modes

Spent Fuel Storage Casks

Spent Fuel Cask Corrosion in a Marine Environment – N6195

- The purpose of this experimental work is to investigate the susceptibility of austenitic stainless steels to chloride induced SCC in a representative atmospheric marine environment. The experiment is being conducted with representative materials and marine atmospheric conditions specific to the austenitic stainless steel surface (including weldments) on spent fuel storage casks. The experimental objective will be obtained through an accelerated testing method, in order to obtain results in the specified time of this contract.
- The research is to:
  - Investigate the susceptibility of various grades of austenitic stainless steels to chloride-induced SCC
  - Establish the effect of temperature on the susceptibility of austenitic stainless steels to chloride-induced SCC
  - Establish the relative susceptibility of austenitic stainless steels base metal, heat affected zone, and weld metal to chloride-induced SCC

Spent Fuel Cask Corrosion in a Marine Environment – Basis

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

- Some domestic spent fuel storage casks are to be located in areas where salt-laden air can come into contact with the surface of austenitic stainless steel spent fuel casks due to the proximity of the casks to either brackish water or sea water. The staff of the Spent Fuel Project Office (SFPO) has requested that the Office of Nuclear Regulatory Research (RES) assist in determining the susceptibility of austenitic stainless steel to chloride-induced stress corrosion cracking (SCC) when exposed to an atmospheric marine environment. The SFPO needs to know if a spent fuel storage cask would be susceptible to SCC when the cask is located at a coastal facility for 20 years or more. A foreign study has shown that austenitic stainless steels commonly used to construct spent fuel storage casks do fail by chloride induced SCC when synthetic sea water is applied by dripping on a heated U-Bend specimen.

Spent Fuel Cask Corrosion in a Marine Environment - Approach

- The test shall expose the test specimens to a representative atmospheric marine environment.
- use environments that best represent the environmental conditions contacting spent fuel storage casks located at domestic independent spent fuel storage installations in proximity to sea water.
- The specimens shall be continuously heated and continuously monitored with thermocouples (constant temperature) throughout the testing.
- specimen shall be photographed, and samples taken of any salt films present.
  – The salt films shall be chemically analyzed qualitatively and quantitatively.
- The specimens shall then be cleaned and the surface examined using a low powered microscope.
  – Any evidence of pitting or stress corrosion cracking shall be verified metallographically.