Joint CEOG/WOG Program
on
Reactor Vessel ISI Interval Extension

Presentation to NRC Staff

Extension of Reactor Vessel ISI Interval

- Meeting Purpose and Objectives
  - Update the Staff on progress since last meeting
    - January 27, 2000
  - Discuss technical approach and preliminary results to date
    - Obtain Staff concurrence on both the approach and use of the tools/codes
  - Present schedule for topical report submittal
    - Obtain preliminary Staff input on review time and schedule

Extension of Reactor Vessel ISI Interval

- **Presentation Overview**
  - Introduction/Background
  - Project Overview
    - Progress Status
  - Technical Basis Development
    - Sample cases
    - Approach to Non-Beltline regions
  - Schedule Projection
  - Summary
  - Discussion

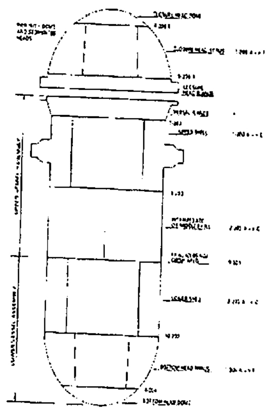
Extension of Reactor Vessel ISI Interval

- **Introduction/Background**
  - Continue discussions on an approach to justify the extension of the RPV In-Service Inspection(ISI) Interval from the current 10 year requirement to 20 years.
  - We are *not* here to eliminate the inspection content but to demonstrate that by continuing with the same breadth of inspections at a longer interval there is no significant increase in risk of component failure.
  - Sponsored by CEOG Section XI & WOG Material Subcommittees (~ 70 operating plants)

Extension of Reactor Vessel ISI Interval

- **Introduction/Background** *(Cont'd)*

FIGURE 1
REACTOR VESSEL WELDS



– Ultimate objective is to address all weld and HAZ regions to be examined during a RPV inspection
  - via an alternative to the ASME Code Rules

Extension of Reactor Vessel ISI Interval

- **Project Overview**
  - Execute a broad based Owners Group Program
    - Perform Sample Plant Analysis which demonstrates that objective can be achieved
    - Formalize approval via a Topical submittal and ASME Code case
      - Apply to specific plants to get relief
    - Communicate with regulators
      - Establish Technical level dialogue
  - Build on approach used on successful BWRVIP program

Extension of Reactor Vessel ISI Interval

- **Program Plan**
  - *Phase 1: Conceptual Feasibility Evaluation*
  - *Phase 2: Technical Feasibility / Sample Plant Application*
  - Phase 3: Technical Application / Topical report
    - Complete work on representative pilot plants
    - Submit topical for review
  - Phase 4: Licensing
    - Support topical review
  - Phase 5: Plant / Vessel Specific Evaluations
    - Using Topical, apply generic methodology to support individual requests for Exemptions

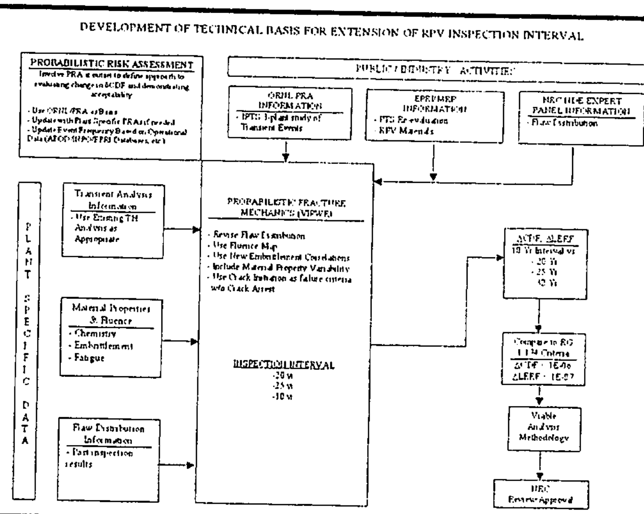
Extension of Reactor Vessel ISI Interval

- **Plant Survey**
  - Economic Survey
    - Man-Rem savings range 0.84 - 2.98 man-rem
    - Dollar savings range $2.3m - $7.1m

Risk Informed Basis for Change

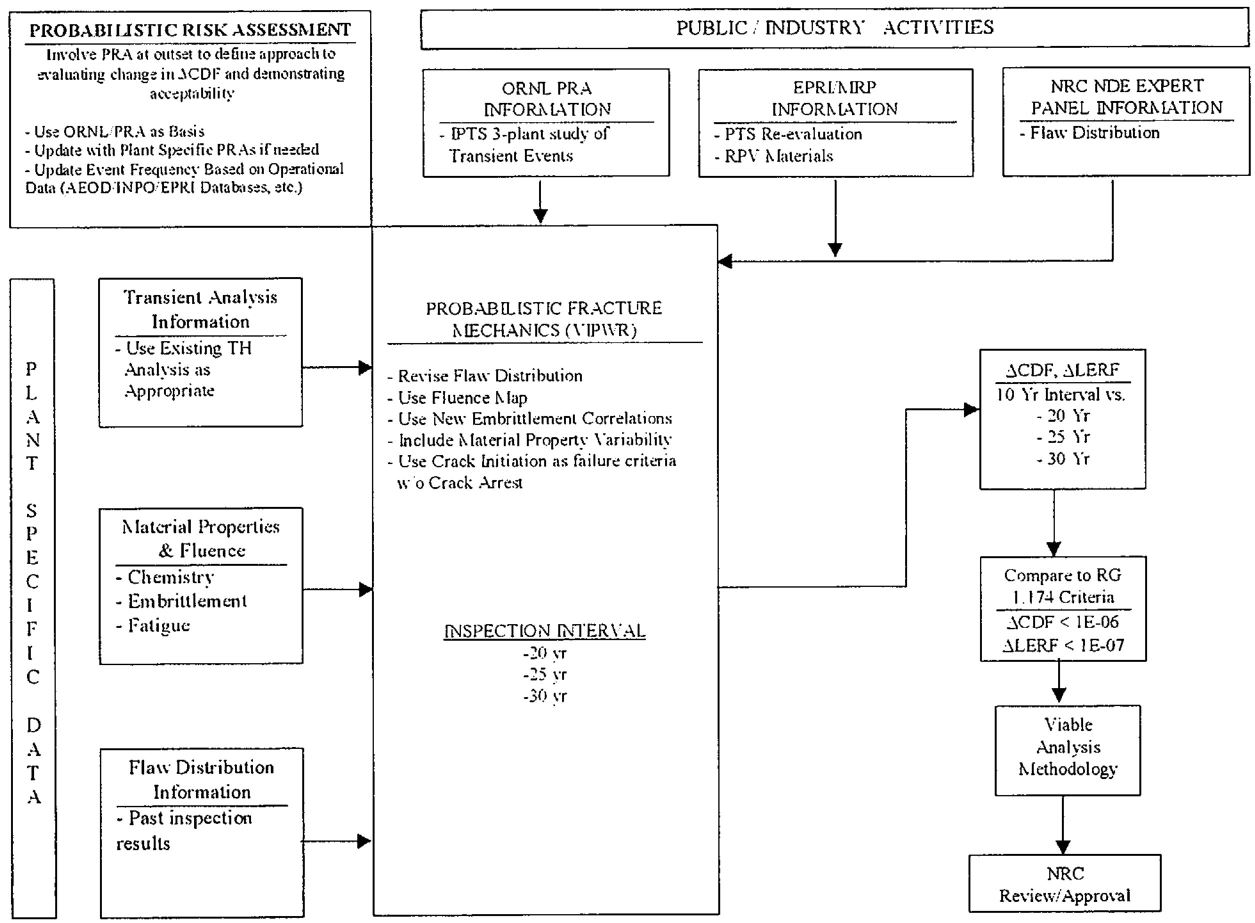
- Change to Inspection Interval will be demonstrated to meet RG 1.174 requirements.
- RG 1.174 - "An Approach for Using PRA In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis" requirements include:
  - Show Substantial Benefit from Change
  - Show small increase in Risk
    - Δ CDF < 1.0E-6
    - Δ LERF < 1.0E-7

Extension of Reactor Vessel ISI Interval



Extension of Reactor Vessel ISI Interval

DEVELOPMENT OF TECHNICAL BASIS FOR EXTENSION OF RPV INSPECTION INTERVAL



Extension of Reactor Vessel ISI Interval

- **Phase 2: Methodology Development**
  - **Input assumptions selected to test and evaluate methodology**
    - **ASME Lower Bound K_{IC} Assumption**
    - **Use of ORNL Transients subset in the Methodology Testing**
    - **Use of PVRUF flaw size Distribution**

Extension of Reactor Vessel ISI Interval

- **Technical Basis Development**
  - Material Properties / Fatigue
  - Comparison to NRC PTS Re-evaluation Effort
  - Probabilistic Fracture Mechanics
    - Preliminary Results
  - Non-Beltline Regions

RPV Beltline
Configuration, Properties & Crack Growth

Presented by Chris Hoffmann
Westinghouse CEET

RPV Beltline Configuration

- RPV Beltline Dimensions
  - As-Built Dimensions
  - Weld Locations & Weld Joint Dimensions
  - Cladding Thickness (Min. Specified)
- Definition of Beltline Subregions
- Stress Analysis Assumptions
  - Weld Residual Stress Distribution (HSST Data)
  - Cladding Stress-Free Temperature

RPV Beltline Material Properties

Beltline Plate and Weld Materials

- **Initial RT_{NDT} Values (Measured, Generic or MTEB 5.2)**
- **Copper & Nickel Content**
  - Chemistry Variability

Irradiated Beltline Material Properties

- **RPV Fluence Calculation**
  - Axial & Azimuthal Distributions
  - Reg. Guide 1.99, Rev. 2, Position 1.1
    - Adjusted Reference Temperatures (Initial + Shift + Margin)

Fatigue Crack Growth & Fracture Toughness

RPV Beltline Flaw Growth & Crack Extension

- **Fatigue Crack Growth**
  - Section XI Crack Growth Correlation
  - Best estimate crack growth with distribution

- **No Significant EAC or SCC Contribution to Flaw Growth for RPV Beltline**

- **Fracture Toughness Properties**
  - Section XI, Appendix A-4200 K_{Ic} Curve

Extension of Reactor Vessel ISI Interval

Discussion / Comments / Q&A

Comparison of CEOG/WOG Approach with NRC PTS Re-evaluation Effort

Presented by Stephen Byrne
Westinghouse CEET

Comparison of CEOG/WOG Task with NRC PTS Re-evaluation Effort

Objective:

- Compare CEOG/WOG and NRC approaches
- Demonstrate conservatisms in *VIPWR* Code relative to *FAVOR* Code

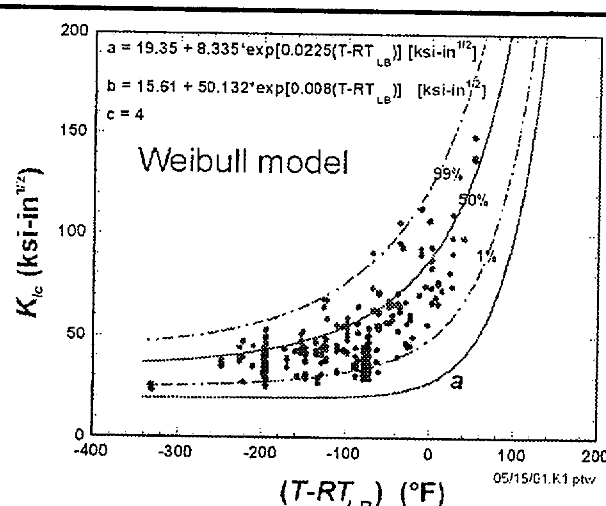
Comparison of CEOG/WOG Task with NRC PTS Re-evaluation Effort

- Crack Initiation Toughness
- Embrittlement Correlation
- Flaw Distribution
- Material Properties and Distributions
- Vessel Geometry
- Transients

Crack Initiation Toughness

- *FAVOR* Code uses mean K_{IC} curve indexed to RT_{LB} and assumes distribution of RT_{LB} using relation with fracture toughness measurements
- *FAVOR* Code also uses mean K_{Ia} curve indexed to RT_{LB}; failure = crack does not arrest

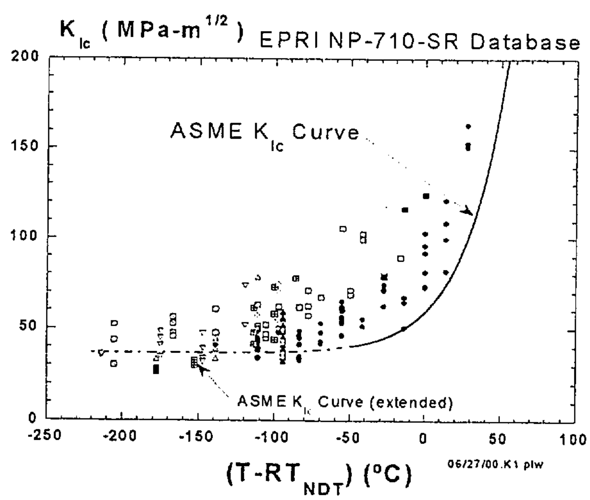
FAVOR Code K_{IC} Model



Crack Initiation Toughness

- *VIPWR* Code uses lower-bound K_{IC} curve indexed to RT_{NDT}
- *VIPWR* Code does not credit crack arrest; failure = crack initiates

VIPWR Code K_{IC} Model



Embrittlement Correlation

- *FAVOR* Code Embrittlement Prediction:
  - EASON Correlation = f(Cu, Ni, P, ϕt, T, t) and product form
  - Fluence attenuation per Regulatory Guide 1.99, Revision 02
  - New
- *VIPWR* Code Embrittlement Prediction:
  - Regulatory Guide 1.99, Revision 02 (Position 1.1)
  - Correlation = f(Cu, Ni, ϕt) and product form
  - Fluence attenuation per Regulatory Guide 1.99, Revision 02
  - Current regulatory requirement

Flaw Distribution

- *FAVOR* Code Flaw Distribution:
  - input provided by PNNL (see next slide)
  - flaws distributed through thickness
  - surface flaws approximated from measurements
  - weld repair flaws approximated from measurements
- *VIPWR* Code Flaw Distribution:
  - size distribution based on PVRUF data
  - considers portion of flaws near inner surface of limiting weld
  - *VIPWR* modifies size of sampled flaw by fatigue crack growth

Material Properties and Distributions

- Material chemistry (best estimate Cu, Ni) and initial RT_{NDT} based on RVID and NSSS designer information for both *FAVOR* and *VIPWR* inputs
- *FAVOR* uses local resampling for crack extension; *VIPWR* does not (crack initiation = failure)
- *FAVOR* uses generalized normal distribution for copper and nickel; *VIPWR* uses heat-specific normal distribution based on data in CE NPSD-1039, Revision 02

Vessel Region Sampling

- Vessel dimensions the same for both *FAVOR and VIPWR*
- *FAVOR* samples from each subregion
- *VIPWR* samples only from limiting location

[Note: *VIPWR* code can be modified to sample from different regions.]

Transients

- *VIPWR* Code sample analyses used transients and event frequencies from SECY-82-465 and plant-specific analysis
- *FAVOR* Code analyses using transients and event frequencies generated on plant-specific basis; available by mid-November 2001 (per SECY-01-0185)

Phase 3 *VIPWR* Code analyses intend to use transients and event frequencies generated for lead plant under PTS evaluation effort

Summary Comparison of *VIPWR* and *FAVOR*

- crack initiation toughness- *VIPWR* uses ASME K_{IC} lower bound curve; *FAVOR* uses mean K_{IC} indexed to RT_{LB}
- embrittlement correlation- *VIPWR* uses RG 1.99, R. 2; *FAVOR* uses Eason equation; both use same attenuation assumptions
- flaw size distribution- *FAVOR* uses PVRUF and other data specific to welds, repairs and base metal; *VIPWR* uses PVRUF distribution for welds and modifies size by fatigue crack growth

Summary Comparison of *VIPWR* and *FAVOR (Cont'd)*

- material properties- both use comparable inputs
- vessel region sampling- *FAVOR* samples from all regions; *VIPWR* samples from limiting locations
- transients- *FAVOR* using newly generated transients and event frequencies; *VIPWR* using transients and event frequencies generated for SECY-82-465

Extension of Reactor Vessel ISI Interval

Discussion / Comments / Q&A

Extension of PWR Reactor Vessel ISI Intervals - PFM Methodology

Peter C. Riccardella
S.S. (Stan) Tang

Structural Integrity Associates, Inc.

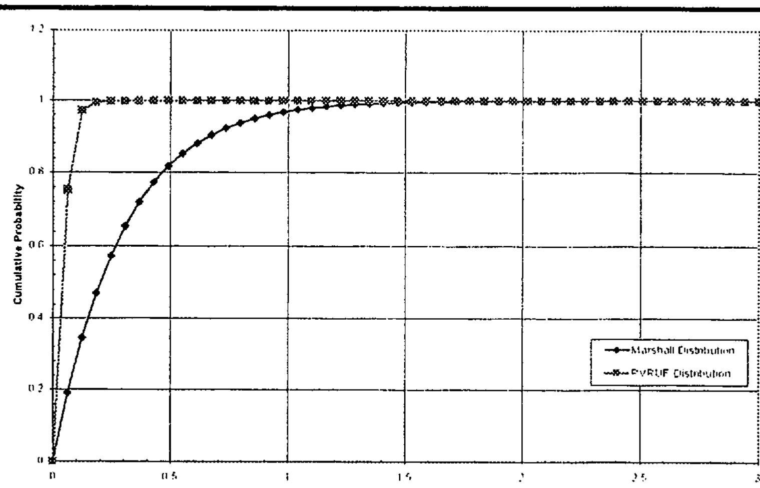
Overview

- General approach built upon methodology developed for BWR Vessel Inspection Evaluation (VIPER, BWRVIP-05)
  - **Retained agreed upon major assumptions**
  - **Adapted methodology, as appropriate, for applicability to PWR vessels (VIPWR)**
- Overall goal - to determine permissible increase in inspection intervals consistent with RG 1.174 Guidelines
- Analysis considers probabilities of vessel failure with current 10 year inspection intervals versus with various proposed alternatives (20 years, 25 years, . . .)

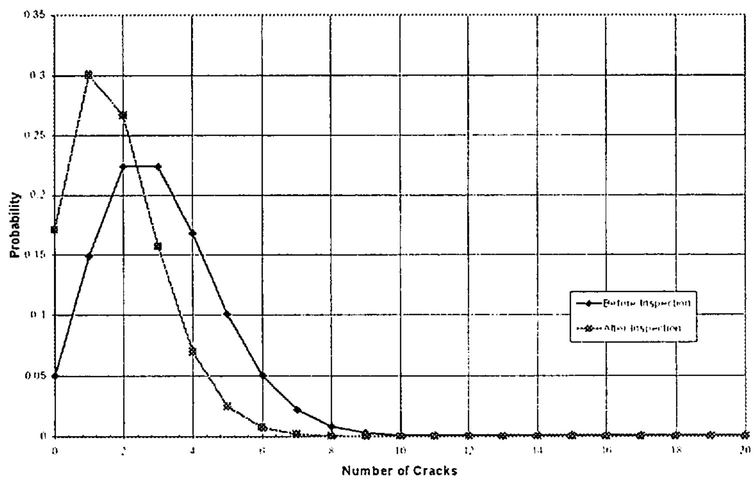
Key Inputs for VIPWR

- Flaw Distributions
- Vessel Fluence
- Material Fracture Toughness
- Limiting Operational Transients
- Crack Propagation
- Clad Residual Stress
- Effectiveness of Inspection (POD)

Flaw Size Distributions



Distribution of Number of Flaws



Vessel Fluence

- **Time Dependence:**

$$f_{surf}(t) = \phi t$$

- **Attenuation through-thickness:**

$$f(x) = f_{surf}(e^{-0.24x})$$

- **Axial and Circumferential Fluence Distributions may be addressed if available**

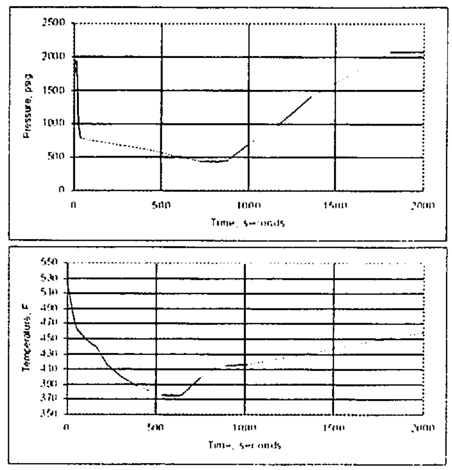
Fracture Toughness of Irradiated Weld Metal

- **Chemistry Factor based on Cu & Ni Content (Per RG 1.99 Rev. 2)** $CF = 360 * Cu * (1.0 + 1.38 * (erf(\frac{(0.3 * Ni - Cu)}{Cu}) + 1))$
- **Fluence Factor (Per RG 1.99 Rev. 2)**
  $FF = f^{(0.28-0.10*\log f)}$
- **Adjusted Reference Temperature**
  $ART = Initial\ RT_{ndt} + \Delta RT_{ndt} + Margin$
- **Fracture Toughness Curve**
  - ASME Section XI K_{ic}

Limiting Operational Transients

- Startup-Shutdown Transients used for Fatigue Crack Growth
- Various PTS Transients considered with Appropriate Event Frequencies
  - Small break LOCA with stagnated loop flow and stable pressure at ~900psi
  - Small break LOCA with stagnated loop flow and full repressurization late in the transient
  - Small break LOCA with stagnated loop flow and rapidly dropping pressure
  - Small steam line break at hot 0% power

Typical PTS Event (Steam Line Break)



Other Key Parameters

- Crack Propagation (FCG)
- Cladding Residual Stress
  - **Treated as a distributed variable**
  - **Indexed to clad temperature during transient**
- Effectiveness of Inspection (POD)

Extension of Reactor Vessel ISI Interval

Discussion / Comments / Q&A

Consideration of Non-Beltline Regions of the Reactor Vessel

Dave Ayres
John Ghergurovich

Consideration of Non-Beltline Regions of the Reactor Vessel

- **Previous discussions have focused on the beltline region of the vessel**
  - This is based on the presumption that the beltline is the governing region due to embrittlement.
- **In order to confirm that the beltline is the governing region, the fracture potential of other regions will be addressed**
- **Regions of high stress have previously been identified and evaluated to assess the impact on pressure – temperature (P-T) limits**

Consideration of Non-Beltline Regions of the Reactor Vessel

- Vessels have previously been evaluated according to ASME Section III Appendix G.
  - KIR Fracture toughness Limit
  - 1/4 thickness flaw
  - Level A&B Transients
- Regions typically considered:
  - Beltline
  - Vessel Wall Transition
  - Bottom Head Juncture
  - Core Stabilizer Lug Attachment
  - Nozzle Inner Radius
  - Flange Region

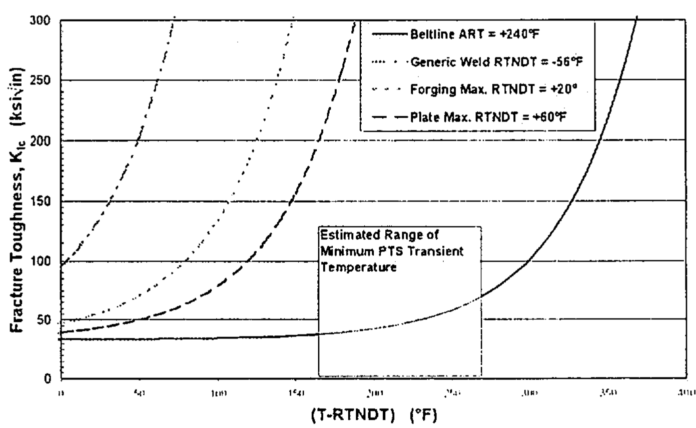
  $\Big\}$ *...In-Process of being re-evaluated*

Consideration of Non-Beltline Regions of the Reactor Vessel

- **Approach Overview**
  - All Vessel locations are evaluated for shutdown conditions of 70 °F or lower.
    - The lowest temperature of transients of concern is significantly higher than shut down temperatures.
  - Non-beltline locations have significantly lower fluence.
  - Therefore, non-beltline regions can be shown to be non-governing because of the relationship of the transient temperature and toughness properties.

Consideration of Non-Beltline Regions of the Reactor Vessel

Fracture Toughness of Various RV Regions



Consideration of Non-Beltline Regions of the Reactor Vessel

- **It is anticipated that the beltline will be shown to be the governing location for determination of the inspection interval.**
  - **Action:** Perform sensitivity study
    - Using Level C&D Transients
    - Bounding material values

Extension of Reactor Vessel ISI Interval

Discussion / Comments / Q&A

Extension of Reactor Vessel ISI Interval

- **Schedule**
  - **Phase 2: Technical Feasibility / Sample Plant Application**
    - Complete Major Modifications to the VIPWR Code and perform Initial Bounding Analyses and meet to discuss results *(Winter/Spring 2002)*
  - **Phase 3: Technical Application / Topical report**
    - Subsequent Topical Submittal to follow *(Spring/Summer 2002)*
  - **Phase 4: Licensing**
    - Review of Pilot Plant Submittal *(Summer/Fall 2002)*
  - **Phase 5: Plant / Vessel Specific Evaluations**
    - Per individual plant needs

Extension of Reactor Vessel ISI Interval

- **Recent Industry Activities**
  - PTS Re-evaluation ... Ongoing
  - Embrittlement Correlation ... Ongoing
  - Flaw Characterization and Distribution ... Ongoing
  - ASME
    - RV Inner Radius Inspection ... Pending
    - RV Flange PT Limit Elimination ... Pending
    - RV ISI Code Case Development ... Initiated

Extension of Reactor Vessel ISI Interval

- **In Summary**
  - Technical Discussions
    - Material Properties / Fatigue
    - Comparison to NRC PTS Re-evaluation Effort
    - Probabilistic Fracture Mechanics
      - Preliminary Results
    - Non-Beltline Regions
  - Industry Activities

... *Objective is Achievable*

Extension of Reactor Vessel ISI Interval

- **Final Discussion / Comments / Q&A**

- **Review Action Items / Open Issues**