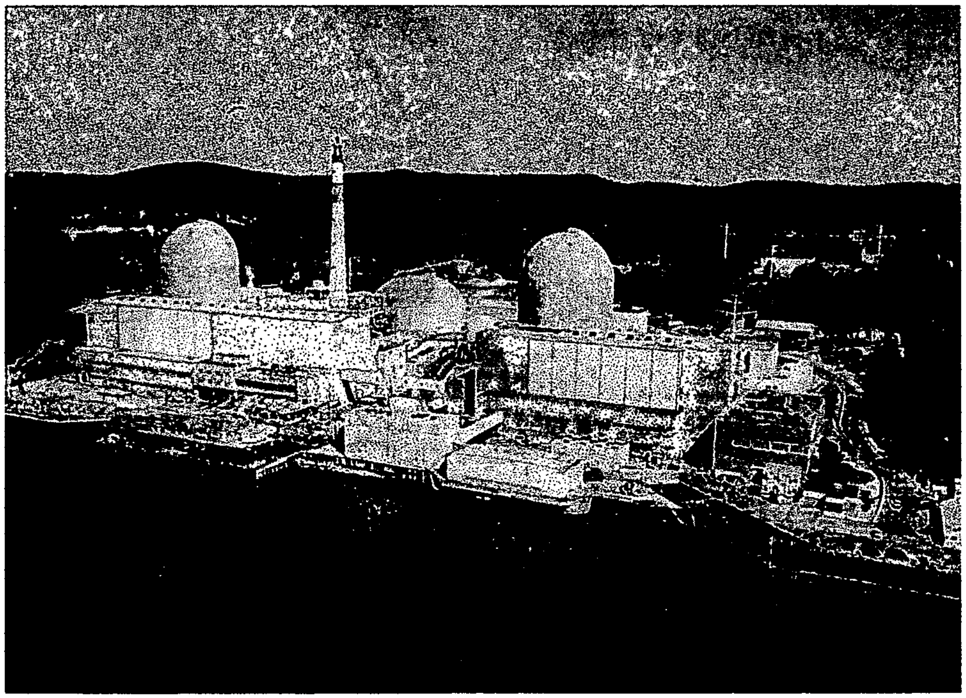
Indian Point Units License Renewal

Indian Point Unit 1, 2 and 3



Presentation by Karl Jacobs

Background

- Local Resident of Cortlandt Manor for 18 years
- Have never been in the employ of Entergy
- 20 years of experience with nuclear operations, maintenance, project management, installation of major multi million dollar safety related nuclear and non- nuclear equipment at IP3 that meets the required federal, state and industry accepted codes.
- 20 years of experience primarily on Indian Point Unit 3 in developing and implementing aging management programs for the Reactor Vessel, Reactor Internals, Pressurizer, Rector Coolant Piping and Steam Generators etc.
- Participated in the License Renewal rulemaking (10CFR50.54a) as IP3 Utility representative and as a Westinghouse Owners Group (PWR NSSS) Subcommittee Chairman, Nuclear Energy Institute (NEI), Electric Power Research Institute and the Nuclear Regulatory Commission
- Lead Technical Engineer for the technical and economical studies for Indian Point Unit 3 and James A. Fitzpatrick Nuclear Plant License Renewal evaluations. The IP3 studies were performed for the previous owner are identified.
  - License Renewal Comparison of IP3 design, operation and performance characteristics to the Industry Pilot Plant (Surry 1) .
  - Life Extension/ License Renewal Program Technical Summary Report
  - Cost/Benefit Analysis

Highlights of the 10CFR 50.54 and revised 10CFR51 Rule

Identification of the License Renewal Components for scoping and screening evaluations and if determined technically that a component does not meet the additional life extension requirements (an aging management programs would be identified for implementation (on –going current licensing basis programs, newly developed and required to be implemented during their license renewal period)

- **This scoping is also to include the identification and evaluation of time limited aging analysis (TLAA)**

Environmental Impact Studies – Opens the door for Cooling Towers to be evaluated and possibly installed in lieu of present Water Cooled Condenser System – The Cooling Towers would help address the zebra mussel issues which are an environmental issue that in the past has plagued the safety related service component and service water cooling systems for IP3 and IP2. (Reduction and possible removal of their chlorination injection program, will also benefit the Hudson River.)

Identify and /or develop aging management programs of the components that are identified through the screening process for managing aging effects and address TLAAs

Emergency Planning and Security is not part of the 10CFR50.54 and revised 10CFR50.51 rule and needs not to be addressed under License Renewal Application

Indian Point Unit 1 License Renewal Scoping Issues

The license renewal application (LRA) is for IP2, IP3 and shared systems with IP1

A review of the scoping of components in the LRA the does not identify Indian Point Unit 1 Containment structure and spent fuel systems and their support systems as being part of the License Renewal Application. See LRA Section 2.4.1 Describes only Unit 2 and Unit 3 Vapor Containment Structure. Unit 1containment structure is omitted.

Per the License Renewal Application for IP2 and IP3 under containment scoping and screening review in section 2.4.1 page 2.4.-2 state "the containment buildings have the following intended functions for 10CFR54.4(a)(1), (a)(2) and (a)(3)."

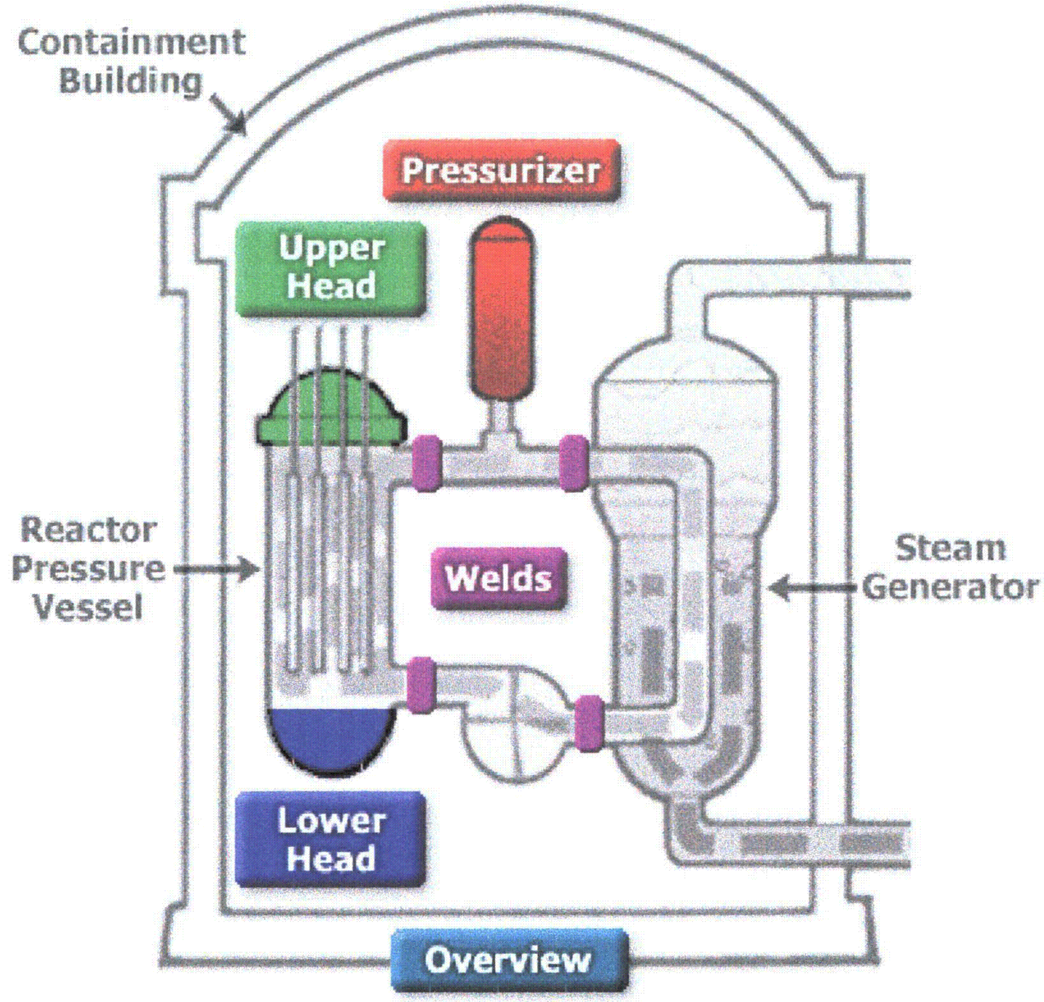
- **Provide support, shelter and protection for safety- related equipment**
- **Maintain essential leak tight barrier**
- **Maintain integrity such that safety –related equipment is not affected.**

Indian Point Unit 1 License Renewal Scoping Issues

Indian Point Unit 1 supports the spent fuel cooling system is located in the containment structure. IP Unit 1's containment performs intended functions as defined by the License Renewal rule function above. In addition other scoping of license renewal scoping and screening systems are inside the containment structure that have been excluded are spent fuel pools structures; HVAC filtration for radioactive airborne particulates, containment penetrations, spent fuel pool system cooling piping and their supports, spent fuel cooling pumps, instrumentation for monitoring the operations of the spent fuel system, electrical wiring, spent fuel bridge,cranes and radiation monitors etc.

With the Entergy IPEC LRA allowing for IP Unit 1 shared components to be included in their application has opened a doorway to allow for a full scoping and screening of IP Unit 1 systems and components to protect the health and safety of the public

IP2 and IP3 Typical RCS Integrity Boundary



Reactor Vessel and Reactor Internals Typical to IP2 and IP3

Westinghouse Nuclear Steam Supply System Designer and Fabricator of Reactor Internals

Combustion Engineering is the Reactor Vessel Fabricator

IP2 RPV Construction Code – ASME Section III 1965 Edition

IP3 RPV Construction Code – ASME Section III Edition Winter 1965 Addenda

Reactor Vessel (RPV)

- **Reactor Vessel Major Intended Functions**
  - Maintain the reactor pressure boundary
  - Support and contain the reactor core and core support structures
  - Support and guide reactor controls and instrumentation
  - Contain the reactor coolant around the reactor core and direct the coolant flow into the core and out into the reactor coolant piping and upper head
  - Interface with the RPV supports to provide a load path to the structural concrete
- **Subcomponents subject to an aging management review**
  - All of its subcomponents are passive, and only two of the subcomponents do not require an aging management.
  - There are only two subcomponents that do not require an aging management review. The RPV O-Rings, O-ring leak monitoring tubes and the refueling seal ledge do not support any RPV intended function

Reactor Vessel (RPV)

- **For RPV neutron embrittlement is a critical aging management failure mechanism issue that must be accurately evaluated for License Renewal for both IP2 and IP3 reactor vessels.**
- **This IP3 reactor vessel has a projected RTndt value that would have exceeded the 10CFR50 Appendix G criteria during life extension if the criteria was not revised by the NRC**
- **For IP3 the lower shell plate (B2803-3) is the limiting RPV plate material.**
- **The projected RTpts for this same lower shell plate is very close to the 10CFR50 Appendix G criteria for the end of license renewal. With augmented aging management programs being implemented which are low leakage fuel management for neutron flux reduction,significant expansion of the reactor vessel surveillance capsule monitoring program, implement research and development programs on material crack initiation and crack growth with similar low fracture toughness' properties, along with a higher frequency of volumetric examinations of the RPV beltline than the present frequency requirements of ASME Section XI and Regulatory Guide 1.150 the RTpts may be successfully managed to meet life extension.**
- **For the same plate, the projected upper shelf fracture toughness energy for 60 calendar years is less than 10CFR50, Appendix G minimum criteria of 50- ft-lbs. This is a critical issue, that Entergy will need the NRC's assistance in a 10CFR50 Appendix G rule change to revise the criteria to a lower threshold value. This plate was originally installed with an initial +74 RTndt value. This was a fabricator miscue to allow the original installation of a shell plate in the Reactor Vessel Beltline with a +74 RTndt material property value to be installed. The plates that are installed in reactor vessels should have minimum initial Rtndt value of zero or a minus value to support Reactor Vessel longevity.**

RPV

- The IP3 Reactor Vessel's lower upper shelf energy (a physical/mechanical properties of the RPV vessel wall) is a major concern for its lower shell plate B-2803-3. This plate material will not meet 10CFR50 Appendix G "Fracture Toughness Requirements" for license renewal. This plate is predicted to fall well below the 50 ft-lbs as measured by Reactor Vessel Surveillance Capsules charpy v –notch specimen testing.
- IP3 has two alternative approaches which are not even mentioned.
    1. An analysis is performed that conservatively demonstrates, making appropriate allowances for all uncertainties, the existence of equivalent margins of safety for continued operation. The margins against fracture must be equivalent to those required by the ASME Code, Section III, Appendix G
    2. Additional evidence of the fracture toughness of the beltline materials after exposure to neutron irradiation may be obtained from results of supplemental fracture toughness tests. The problem with this approach is the IP3 Reactor Surveillance Program remaining capsule specimens do not have the limiting plate material B2803-3 in any of this capsules. The statement made by Entergy in the license renewal application Section B.1.32,titled (Reactor Vessel Surveillance) page B-112 under the described enhancements that "The specimen capsule withdrawal schedules will be revised to draw and test a standby capsule to cover the peak reactor vessel fluence expected through the end of the period of extended operation."

RPV

- The IP Unit 3 RPV has an on going aging management program called reactor vessel surveillance capsule monitoring program that is in effect for its current licensing basis and does an extra capsules installed in the reactor vessel for life extension. The limiting plate material (B2803-3) as does not have any material in the remaining capsules to monitor the lower shell plate. This also will have a significant effect on their heat and cool down curves which are developed from the most limiting vessel plate material.
- The IP3 RPV materials has been volumetric examined (ultrasonic techniques) thoroughly every ten years from initial operation and no reportable indications were found. The volumetric equipment used for these inspections are very good, use a array of ultrasonic transducer probes, with high detection capabilities, sizing and locating any flaws are also very good. Please note that the RPV beltline is 100% inspected but access to allow for 100% of all RPV welds volume is not achievable do to interferences.
- Bottom line IP3 reactor vessel beltline plate material absolutely does not support license renewal unless the NRC revises 10CFR50 Appendix G requirement of maintaining a higher USE value of 50 ft –lbs for its belt line material

RPV

Present Industry Events and experiences has identified that the IP2 and IP3 Reactor Vessels' Heads must be replaced prior to life extension. This is a generic industry concern for the Westinghouse Reactor Vessel Heads' penetration tube welds that started in September 1991 @ the Bugey Unit 3 PWR nuclear plant in France. Then in May 1992 Ringhals Unit 2 , a Westinghouse- designed PWR –in Sweden found a 25 % around through wall crack in the CRDM penetration. Then it came to America. 1995 DC Cook Unit 2 (Westinghouse design) a crack measured as the deepest point of 6.88mm, 25% around the CRDM tube wall. VC Summer Plant was next, then Ringhals 3 and 4in June 2001, then Oconee and an Entergy Plant ANO-1. NRC Bulletins have been issued.

- **NRC Bulletins 2001 –01 Circumferential Cracking of Reactor Pressure Head Penetration Nozzles**
- **NRC Bulletin 2002-01 Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity**
- **NRC Bulletin 2002-02 Reactor Pressure Vessel Head and Vessel Head Penetration Nozzle Inspection Programs**

– **Entergy LRA response – Intend to use ASME Section XI, Sub Section IWB Inservice Inspection and Water Chemistry Control Programs. Detection of Cracking is accomplished through implementation of a combination of bare metal visual examination (external surface of head) and non-visual examination (underside of the head) techniques.**

– **Entergy has not realized as a company that safety and lowering the risk to public health comes first not economics This is real cracking issue that many same design plants are experiencing now! This cracking can lead to a control rod missile ejection followed with a small break loca. This failure would permanently shut IPEC down!**

– **Reactor Coolant Supports are located in a difficult to access area and limits inspection capabilities. Reactor Coolant Supports can corrode since the are serviced with cooling water. A inspection program to fully assess these reactor supports and cooling system requires a definitive aging management program.**

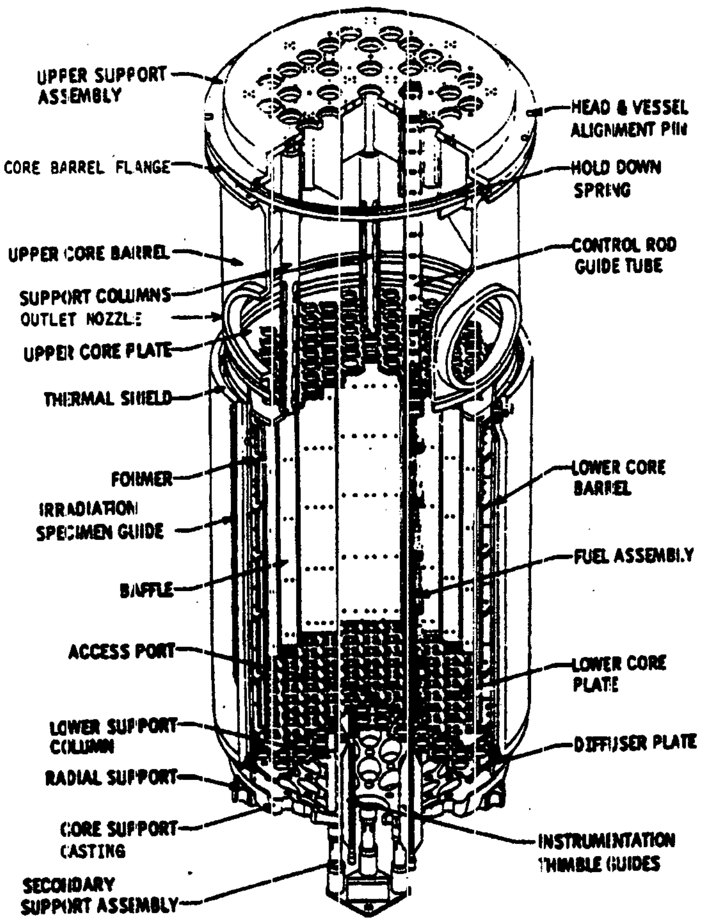
PZR

- Aging Degradation mechanisms for the IP 2 and IP3 Pressurizers are so significant that replacement is the only option for License Renewal. Some Highlights
- The pressure boundary materials of the Pressurizer are susceptible to Primary Water Stress Corrosion Cracking (PWSCC)
    - ❖ The pressurizer has Inconel 82/182 weld metal in pressurizer safety, relief, spray and surge nozzles which is susceptible cracking due to PWSCC
- The pressure boundary materials of the Pressurizer have significant end of life fatigue issues that will not meet life extension time frame
    - ❖ Fatigue of the upper portion of the pressurizer shell (44 years), the spray nozzle(49 years), the manway bolts (46 years), the seismic support lugs(41 years), lower head (due to insurge/outsurge transients), the heater wells (due to insurge and outsurge transients), the surge nozzle, the support skirt and flange (skirt –to-lower-head weld 54 years).
    - ❖ Then when you impose the NRC environmental effect to the fatigue calculations the list gets longer. Lower head (42 years), the safety and relief nozzles (53 years) and instrument nozzles (51 years)
    - ❖ This is back up by the NRC Final Safety Evaluation Report on the Acceptance for Referencing of a Generic License Renewal Program Topical Report by the Westinghouse NSSS Vendor "License Renewal Evaluation: Aging Management Evaluation For Pressurizers" dated October 26, 2000
    - ❖ Aging Management Program 2.3 needs to be imposed. This states that if the TLAA can not show acceptable usage for the license renewal period, the fatigue adequacy will be met by implementing a repair and replacement program in accordance with ASDE Section XI IWA- 004000 or IWA-7000
- NRC has issued a Final safety Evaluation Report for "Acceptance for referencing of Generic License Renewal Program Topical report entitle, "License Renewal Evaluation Aging Management Evaluation for Pressurizers" WCAP-14574 Revision 0, July 1996

Reactor Vessel Internals

- **Aging Management Evaluation for Reactor Internals – WCAP –14573**
- **WCAP –14573 was submitted to the NRC by the Westinghouse Owners Group for IP unit 2 and Unit 3 and received a NRC Safety Evaluation Report accepting this WCAP to support License Renewal**
- **Reactor Vessel Intended Functions**
  - **Ensuring the capability to shut down the reactor and maintain it in a safe shutdown condition**
  - **Providing (Non – Safety Related) intended Functions that support the function listed above**
  - **Ensuring the integrity of the reactor coolant pressure boundary (Bottom Mounted Instrumentation Flux Thimbles Only)**

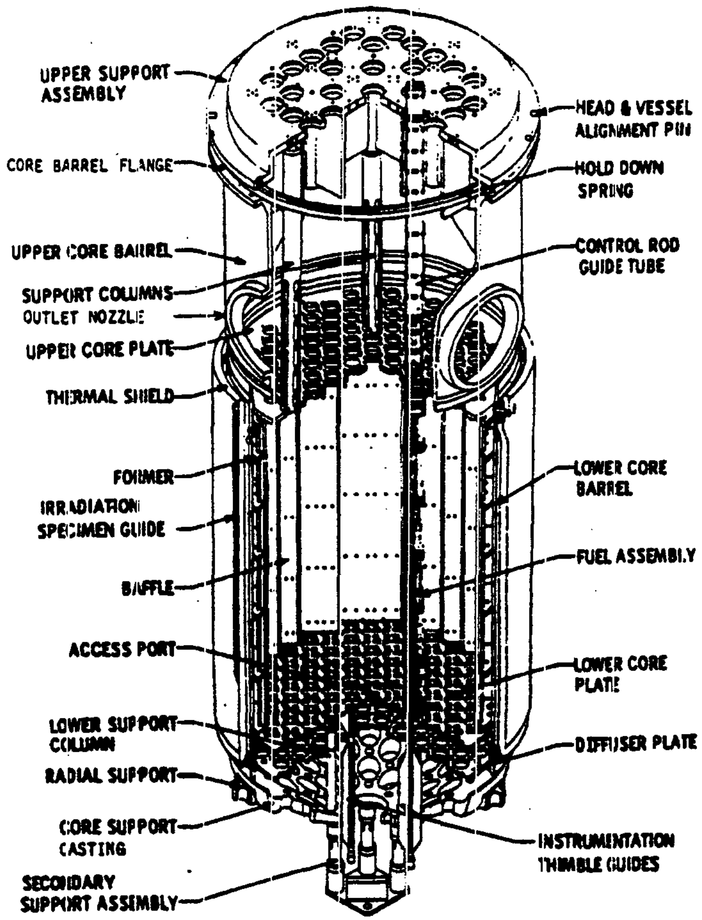
Reactor Vessel Internals



AGING MECHANISMS CONSIDERED

- IRRADIATION EMBRITTLEMENT
- STRESS CORROSION CRACKING
- IRRADIATION-ASSISTED STRESS CORROSION CRACKING
- EROSION and EROSION/CORROSION
- CREEP/IRRADIATION CREEP
- STRESS RELAXATION
- WEAR
- THERMAL AGING
- CORROSION
- FATIGUE
- SWELLING

Reactor Vessel Internals



SUMMARY OF REACTOR INTERNALS SUBCOMPONENTS REQUIRING AGING MANAGEMENT REVIEW

Part or Subcomponent	Aging Management Review Required?
Lower core plate and fuel alignment pins	YES
Lower support forging or casting	YES
Lower support columns	YES
Core barrel and core barrel flange	YES
Radial support keys and clevis inserts	YES
Baffle and former plates	YES
Core barrel outlet nozzle	YES
Secondary core support	YES
Diffuser plate	YES
Upper support plate assembly	YES
Upper core plate and fuel alignment pin	YES
Upper support column	YES
Guide tube and flow downcomers	YES
Upper core plate alignment pin	YES
Holddown spring	YES
Head and vessel alignment pins	YES
Control rod	NO
Drive rod	YES
Neutron panels/thermal shield	YES
Irradiation specimen guide	YES
BMI columns and flux thimbles	YES
Head cooling spray nozzles	YES
Upper instrumentation column, conduit, and supports	YES
Mixing device	YES
Bolts and locking mechanisms	YES
Specimen plugs	YES

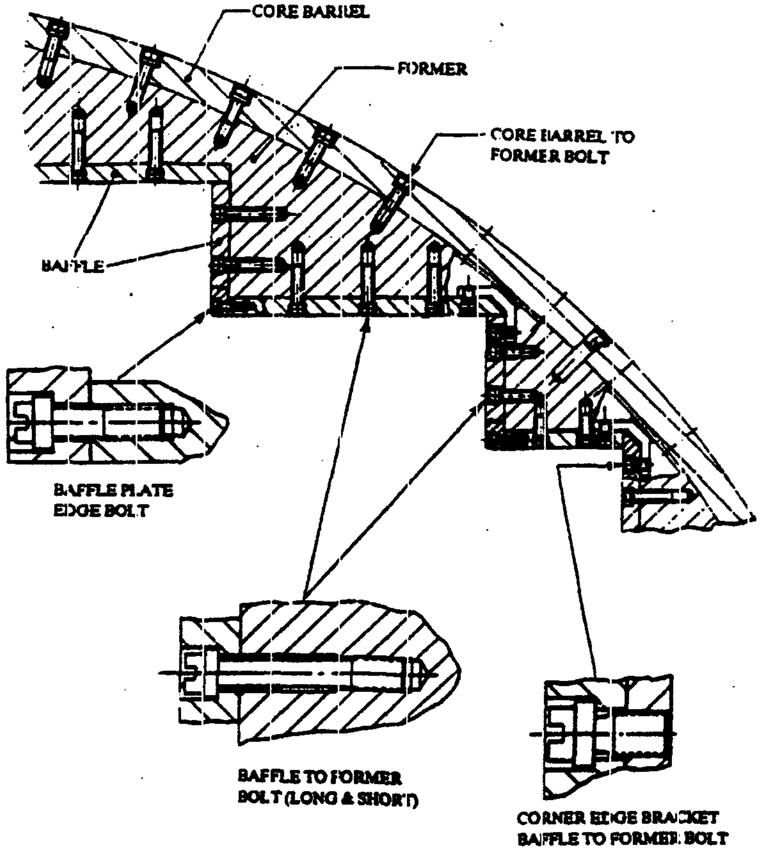
Reactor Vessel Internals

The following actions are needed for reactor vessel internals life extension as a minimum.

1. **Control Rods Replacement for both Units 2 & 3**
2. **Specific fatigue monitoring programs for numerous Reactor Vessel Internals parts that are fatigue sensitive.**
    1. **Baffle Former Bolts**
    2. **Barrel Former Bolt**
    3. **Lower Core Plate**
    4. **Lower Support Plate**
    5. **Radial Key Weld**
    6. **Core Barrel Nozzle Weld**
    7. **Guide Tube/flow downcomers**
    8. **Upper support plate assembly**

    Note these fatigue sensitive parts as calculated do not include the NRC request to include environmental effects.
3. **Replacement Program for Baffle Former Bolts as a Lead Indicator for the other plant and for managing Barrel Former Bolts aging degradation. Cracked Baffle Bolts have already been replaced at Point Beach Unit 2 and RC Ginna Nuclear Power Plant in upstate New York.**
4. **Wear Management program for BMI flux Thimbles; Upper core plate alignment pins; radial keys and clevis inserts Per Commitments to NRC I&E Bulletin 88-09**
5. **Split Pin Replacement for Unit 2 with flexure modification to flexure less insert . with split pin replacement results from Unit 2, the results could be a lead indicator for Unit 3 aging management for split pins. This is only to be considered for mitigating the consequences of loose parts in the Reactor Vessel, Reactor Internals, and protection of the Steam Generators' tube sheet.**

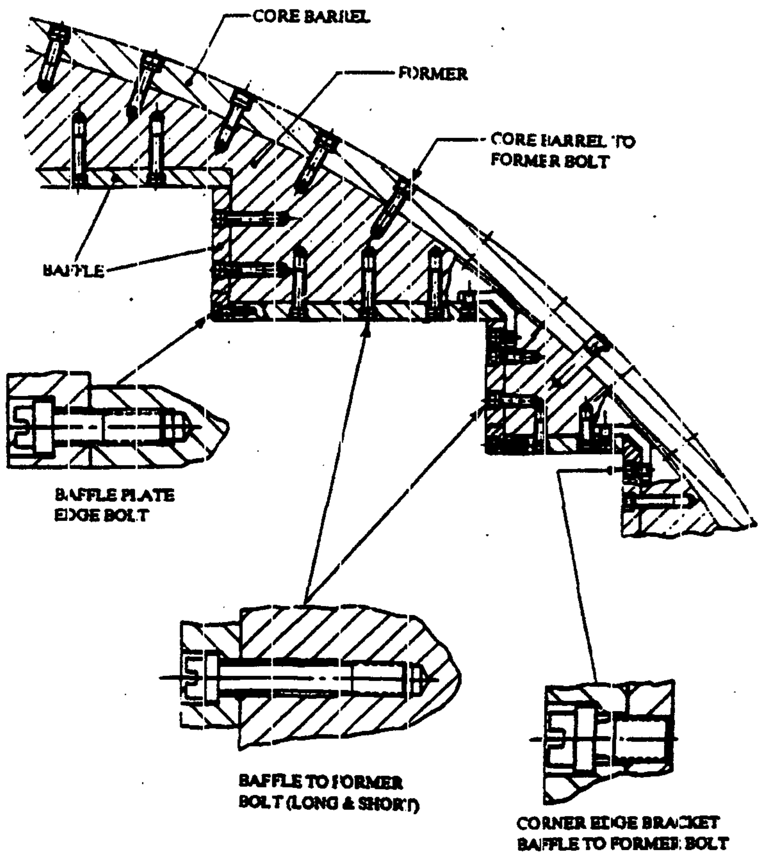
Reactor Vessel Internals



ADDITIONAL ACTIVITIES AND PROGRAM ATTRIBUTES FOR AGING MANAGEMENT OF BAFFLE/FORMER BOLTS (AMP-4.6)

Attribute	Description
Scope	Effects of cracking caused by fatigue, irradiation-induced changes in material properties, and irradiation-induced changes in stresses
Surveillance Techniques	• Visual inspection per Examination Category B-N-3 of ASME Section XI, Subsection IWB and Draft Subsection IWG • Loose parts detection monitoring system • Chemistry RC detection system • Augmented inspections (e.g., ultrasonic inspections)
Frequency	• Monitor with loose parts detection system • Monitor with RC chemistry detection system • ASME Section XI requirements, IWB-2410, -2411, -2412, -2420, -2430 and Draft IWG-2410, -2420, and -2430 • Perform sample baseline inspections prior to LR term with enhanced frequency in accordance with corrective actions
Acceptance Criteria	• Acceptable RC chemistry per technical specifications and • No loose parts from baffle/former bolt assembly and • Fatigue management program in Figure 4-1 and • Number of acceptable bolts and location ≥ the minimum number and location required to maintain core coolability and departure from nucleate boiling ratio (DNBR) within CLB limits, or if needed, for justification of continued operation (JCO), number of acceptable bolts and location ≥ JCO assumptions
Corrective Actions	The following courses of action depend on the bolt condition determined by the monitoring and inspection programs: • Supplemental examinations, analytical justifications or repair/replacement when relevant conditions are detected • Visual inspections, baffle gap measurements, augmented inspections (e.g., ultrasonic inspections), analytical justifications or repair/replacement when baffle/former bolt assembly loose parts are detected • Fuel inspections, visual baffle plate inspections, baffle gap measurements, augmented inspections (e.g., ultrasonic inspections), analytical justifications or repair/replacement when RC chemistry limits are violated • Adjustment of frequency of inspections and coverage • Analysis (e.g., fracture mechanics techniques, risk-based technology, advanced thermal/hydraulic methodologies) • Bolt replacement of a sample set so the existing bolts with indications may be analyzed (materials testing) and the new bolts monitored • Follow actions prescribed in fatigue management program
Confirmation	Acceptable performance per • Loose parts monitoring and RC chemistry programs • Augmented examinations (e.g., baffle gap inspections, ultrasonic examinations) • Analytical justification

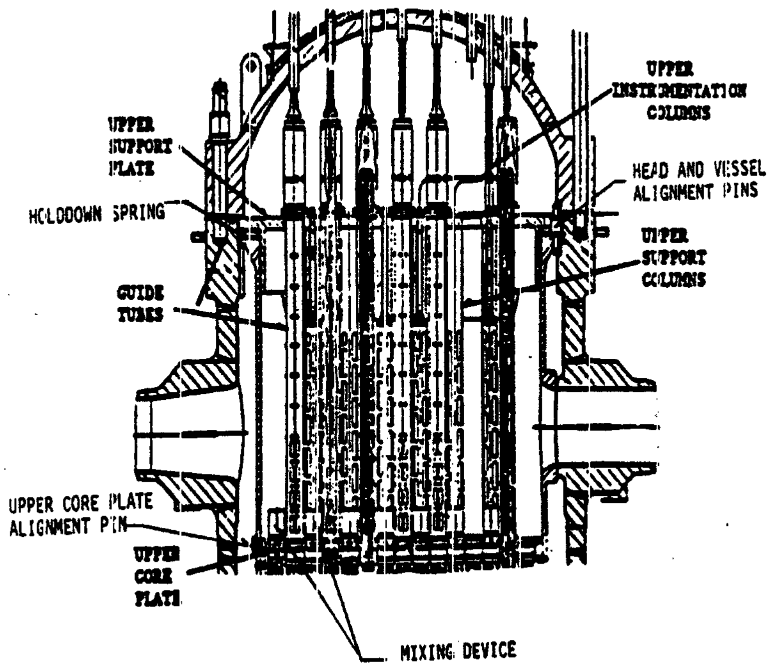
Reactor Vessel Internals



ADDITIONAL ACTIVITIES AND PROGRAM ATTRIBUTES FOR AGING MANAGEMENT OF CORE BARREL/FORMER BOLTS (AMP-4.7)

Attribute	Description
Scope	Effects of cracking caused by fatigue, irradiation-induced changes in material properties, and irradiation-induced changes in stresses
Surveillance Techniques	• Visual inspection per Examination Category B-N-3 of ASME Section XI, Subsection IWB and Draft Subsection IWG • Loose parts detection monitoring system • Augmented inspections
Frequency	• Monitor with loose parts detection system • ASME Section XI requirements, IWB-2410, -2411, -2412, -2420, -2430 and Draft IWG-2410, -2420, and -2430 • Perform sample baseline inspections prior to LR term with enhanced frequency in accordance with corrective actions
Acceptance Criteria	• No loose parts from barrel/former bolt assembly and • Fatigue management program in Figure 4-1 and • Number of acceptable bolts and location ≥ the minimum number and location required to maintain core coolability and DNBR within CLB limits, or, if needed, for JCO, number of acceptable bolts and location ≥ than JCO assumptions.
Corrective Actions	The following courses of action depend on the bolt condition determined by the monitoring and inspection programs: • Supplemental examinations, analytical justifications or repair/replacement when relevant conditions are detected • Visual inspections, augmented inspections (e.g., ultrasonic inspections), analytical justifications or repair/replacement when barrel/former bolt assembly loose parts are detected • Adjustment of frequency of inspections and coverage • Analysis (e.g., fracture mechanics techniques, risk-based technology, advanced thermal/hydraulic methodologies) • Bolt replacement of a sample set so the existing bolts with indications may be analyzed (materials testing) and the new bolts monitored • Follow actions prescribed in fatigue management program
Confirmation	Acceptable performance per • Loose parts monitoring program • Augmented examinations (e.g., ultrasonic examinations) • Analytical justification

Reactor Vessel Internals



Upper Head Aging Parts that require Aging Management Efforts

- **Guide Tubes (Guide Tubes) – Wear**
- **Control Rods – Wear, Cracking -**
- **GT Flexure Replacement for IP Unit 2- Original Flexures are susceptible to SCC**
- **Split Pins – Stress Corrosion Cracking**

Summary

- **Entergy references in the LRA their existing aging management program entitled IP3 and IP2 risk informed inservice inspection program for monitoring the welds and supports that is based on 40 operating service years not the life extension time frame 60 service years This aging management program will not adequate address life extension aging challenges to the plants pressure boundary materials and the materials in the reactor internals.**
- **The IP Unit 3 reactor vessel lower shell plate has limiting plate material does not meet the guidelines of NRC 10CFR50 Appendix G Regulations for life extension. The proposed aging management program does not assure that the reactor vessel beltline limiting plate will meet the additional 20 years of life extension.**
- **The IP3 and IP2 reactor vessel heads require a commitment to be replaced before License Renewal**
- **The pressurizer in numerous areas is fatigue sensitive and the Fatigue Cumulative Usage Factor of these numerous areas will exceed the value of 1. The aging management program proposed for monitoring this will not assure that the pressurizer's pressure boundary materials will meet the additional 20 years.**
- **The Reactor Vessel Internals has known current basis issues with fatigue and cracking of baffle bolts. Entergy has not identified in the LRA a specific aging management program that will address this situation.**
- **Scoping of Unit 1 to include Containment Structure, its spent fuel systems and support systems is justified by LRA defined intended functions scoping and screening criteria.**
- **This is just a very small sample of my evaluation (very tip of the ice berg) due to the time constraints of this meeting date. Many additional components need to be addresses e.g. Steam Generators, Reactor Coolant Pumps, Supports , RCS piping, Electrical Equipment, Containment Structures, Instrumentation, Control Rooms etc.**