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Constellation Energy
Nuclear Generation Group

February 27, 2009

U. S. Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: Document Control Desk

SUBJECT: **R.E. Ginna Nuclear Power Plant**
Docket No. 50-244

License Renewal Aging Management
Reactor Vessel Internals Program

Reference: (1) Letter from P.T. Kuo, NRC to R.C. Mecredy, R.E. Ginna, Subject: License Renewal Safety Evaluation Report for the R.E. Ginna Nuclear Power Plant, dated March 3, 2004.

(2) Letter from D. Holm, Ginna LLC, to NRC Document Control Desk, Subject: Change in Submittal Date for the Augmented Reactor Vessel Internals Inspection Program Document, dated July 31, 2007.

As documented in Reference (1), R.E. Ginna Nuclear Power Plant, LLC (Ginna LLC) committed to submit to the NRC staff a Reactor Vessel Internals Aging Management Program for their review and approval. A change in the commitment date was provided by Reference (2). Please find attached the R.E. Ginna Reactor Vessel Internals Program document.

There are no regulatory commitments contained in this letter. Should you have questions regarding this matter, please contact David Wilson (585) 771-5219, or david.f.wilson@constellation.com.

Very truly yours,

Joseph E. Pacher

Attachment: REACTOR VESSEL INTERNALS PROGRAM

cc: S. J. Collins, NRC
D. V. Pickett, NRC
Resident Inspector, NRC

A092
NRR

1002099

REACTOR VESSEL INTERNALS PROGRAM

LICENSE RENEWAL AGING MANAGEMENT PROGRAM BASIS DOCUMENT

GINNA STATION

REACTOR VESSEL INTERNALS PROGRAM

LR-RVI-PROGPLAN

Revision 2

Prepared By: 2-27-09
Aging Management Program Owner / Date

Reviewed By: 2/27/09
License Renewal Coordinator / Date

Approved By: 2/27/09
License Renewal Project Manager / Date

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1.0 PURPOSE

The purpose of this program basis document (PBD) is to document the Ginna Station Reactor Vessel Internals Aging Management Program (AMP), and identify those activities of the Reactor Internals AMP that are credited for license renewal. This document provides a description of the program as it relates to the management of aging effects identified in the aging management review process documented in LRAM-RVI (Ref. 9.35). This AMP also captures, the intent of the additional industry guidance for reactor internals augmented inspections through the programs sponsored by Utilities, through the EPRI-MRP and the Pressurized Water Reactor Owners Group (PWROG).

This AMP for the Ginna reactor internals demonstrates that the program adequately manages the effects of aging for reactor internals components and establishes the basis for providing reasonable assurance that the internals components will continue to perform their intended function through the Ginna license renewal period of extended operation.

Revision 2 of this program plan provides additional inspection requirements and acceptance criteria for reactor internals not previously available in the aging management outputs documented in the LRAM-RVI. As noted in the LRAM-RVI, additional requirements for inspections were to be developed by the industry on a generic applicability basis. These additional requirements are captured by this AMP and are based on the work of the EPRI-MRP and the PWROG.

Ginna also has a license renewal commitment to work with the industry to develop a reactor vessel internals aging management program. The development and implementation of this program meets this license renewal commitment.

2.0 BACKGROUND

2.1 Ginna License Renewal Background

By letter dated July 30, 2002, Rochester Gas & Electric Corporation (RG&E, the applicant at that time) submitted the License Renewal Application (LRA) for Ginna Station in accordance with Title 10, Part 54, of the Code of Federal Regulations (10 CFR Part 54 or the Rule). Through the LRA, RG&E requested that the U.S. Nuclear Regulatory Commission (NRC) renew the operating license for Ginna (license number DPR-18) for a period of 20 years beyond the current expiration of midnight, September 18, 2009. The safety evaluation report (SER) NUREG-1786 documented the technical review of the R.E. Ginna Nuclear Power Plant (Ginna) license renewal application (LRA) by the U.S. Nuclear Regulatory Commission Staff.

Section 6 of the SER concludes that based on the evaluation of the application as discussed in the SER, the staff determined that the requirements of 10 CFR 54.29(a) were met by the Ginna application.

The Renewed Facility Operating License No. DPR-18 for R.E. Ginna Nuclear Power Plant was granted as documented in the NRC letter of May 19, 2004 (Ref. 9.21). Reference 9.21 identifies the technical basis for issuing the renewed license as being set forth in the NUREG-1786, Safety Evaluation Report Related to the License Renewal of the R.E. Ginna Nuclear Power Plant, (Ref. 9.34).

2.2 Ginna Reactor Internals Aging Management Review /Industry Program Background

Per SER NUREG 1786, Ginna Station committed to establishing and providing an AMP for the Reactor Internals components based on the required ten elements of the GALL Report (Ref. 9.1).

The initial work performed to support the Ginna station application for license renewal included an aging management review LRAM-RVI (Ref 9.35) utilizing the methodology that was established by the WOG (now PWROG) published in WCAP-14577-1-A (Ref 9.2). The aging management approach utilized by the WCAP was found acceptable by the NRC as documented in their letter of Feb. 10, 2001(Ref. 9.66). Per the NRC letter, the WCAP approach was found acceptable, with NRC identified limitations of committing to the accepted aging management programs defined in the WCAP, and completing the renewal applicant actions items defined in section 4.1 of the WCAP final safety evaluation report. Therefore, application of the WCAP methodology provides reasonable assurance that the effects of aging will be adequately managed so that the intended functions of the reactor internals would be maintained consistent with the current licensing basis during the period of extended operation.

As presented in the AMR output table of LRAM-RVI (Ref. 9.35) provided herein as Attachment "C", a combination of existing programs, and additional work, to be identified by the "Reactor Vessel Internals Inspection Program" was credited for aging management of the Ginna Reactor Internals. The table in attachment "C", includes in the last column, those line items where additional Industry evaluations or inspection technique development were anticipated in order to complete this LR-RVI-PROGPLAN or "Reactor Vessel Internals Inspection Program" as referenced in the attachment "C" summary table.

The additional industry work on the Reactor Vessel Internals Inspection Program, referred to in the Ginna LRAM-RVI, culminated in the submittal of "PWR Internals Inspection and Evaluation Guidelines" MRP-227, Rev. 0 (Ref. 9.38) to the NRC for review and SER approval in January of 2009. The industry program is intended to provide a consistent approach to aging management of PWR reactor internals components across the PWR fleet.

2.3 Ginna Reactor Internals Aging Management Program Background

The Ginna Station Reactor Vessel Internals (RVI) AMP evaluates the Ginna AMP as compared to the 10 attributes of NUREG-1801, Generic Aging Lessons Learned (GALL) Report, Rev. 0, Section XI.M.16, "PWR Vessel Internals" (Ref. 9.1). The AMP credits inspections from existing Ginna station programs, industry programs issued under the guidance of NEI-03-08 (Ref. 9.54) as implemented by Ginna Procedures (Ref 9.47 and 9.51), and previous industry operating experience and incorporates recommendations for additional inspections provided by Industry Guidelines contained in MRP-227, Rev 0 (Ref 9.38). This AMP satisfies a Ginna NRC licensing commitment for life extension. The commitment is number #31, and is contained in the Ginna SER, NUREG -1786 (Ref. 9.37). Attachment "A" provides a list of action items required to implement this program.

It should be noted that Ginna station is among the first plants to submit an aging management program on reactor internals for regulatory review. As such, this program plan demonstrates, that the previously approved approach documented in the NRC letter of Feb. 10, 2001 (Ref. 9.66) is met through comparison of the Ginna program to the required GALL Report attributes while capturing the intent of the additional industry program (MRP-227) discussed above.

Existing Program aspects of this AMP such as the ASME Section XI program and primary system chemistry monitoring have been and continue to be on-going programs at Ginna Station. These programs are supplemented as needed to form the basis for the Ginna Station Reactor Vessel internals AMP.

An additional study, evaluating selected components of the Ginna internals, has also been performed and documented in WCAP 16881-P, "R.E. Ginna Reactor Vessel Internals Program Scheduling Evaluations", (Ref. 9.68), for the initial period of extended operation. This study indicates that no adverse conditions exist for the reactor internals "Primary Components" during the initial years of extended operation. An additional evaluation on split pins is documented in draft form by Westinghouse letter, MRP-227 "R.E. Ginna Aging Management Inspection Evaluations for Guide Tube Support Pins" (Ref. 9.69).

This AMP for reactor internals takes effect in 2009. Augmented inspections for aging degradation mechanisms have been scheduled to coincide with the ASME Section XI, 10 year In-service Inspections (ISI) program requirements during the 2011 RFO.

2.4 Ginna Station Aging Management Program Intent

The reactor internals AMP utilizes a combination of prevention, mitigation and condition monitoring. Where applicable, credit is taken for existing programs such as inspections prescribed by the ASME Section XI In-Service Inspection Program (Ref 9.45), other existing plant programs such as water chemistry (Refs. 9.10 9.11, 9.12), thimble tube inspections (Ref. 9.28) and past and future mitigation projects such as baffle bolt inspections and split pin replacement (Refs. 9.5, 9.32, and Section 4.2 of this document) combined with augmented inspections or evaluations as recommended by MRP-227.

Aging degradation mechanisms that impact internals have been identified in Ginna Aging Management review LRAM-RVI. The over-all outcome of both the LRAM-RVI and the additional work performed by the Industry summarized in MRP-227 is to provide appropriate augmented inspections for reactor internals components in order to provide early detection of degradation mechanisms identified below. Therefore, this AMP is consistent with both the WCAP 14577-1-A methodology and the additional industry work provided by MRP -227. Both documents address the concern for the following aging mechanisms:

- Stress Corrosion Cracking (SCC)
- Irradiation –Assisted Stress Corrosion Cracking (IASCC)
- Wear
- Fatigue
- Thermal Aging Embrittlement
- Irradiation Embrittlement
- Void Swelling and Irradiation Growth
- Thermal and Irradiation-Enhanced Stress Relaxation or Irradiation Enhanced Creep

Section 5, provides an evaluation of the ten elements of the GALL Report (Ref. 9.1) and incorporates all available programs and activities that are credited for managing the aging effects produced by the mechanisms listed above. All components of the Reactor Internals have been considered in the review. Tables indicating those aging effects requiring aging management based on a specific assembly, component, or sub-component of the reactor internals are included in the summary table in Attachment "C" as taken from the Ginna AMR, LRAM-RVI.

For those components listed in the Program/Activities column of attachment "C", where credit is taken for "Reactor Vessel Internals Inspections Program" the industry sponsored MRP-227 Reactor Internals Inspection Guidelines; (Ref 9.38) tables are consulted for appropriate direction on the inspection technique and frequency. The tables of MRP-227 are included as attachment "D". The MRP-227 tables were consulted so as to ensure the Ginna AMP is aligned with the anticipated industry inspections and frequencies of inspection for those components that credit this PROGPLAN and the anticipated industry work in the original Ginna LRAM-RVI. A brief justification is provided for inspection applicability to the Ginna LRAM-RVI required augmented inspections presented in section 5.3.3 where appropriate. The MRP-227 recommendations are based on additional Industry evaluations documented in references 9.40 through 9.43.

2.5 Ginna Reactor Internals Background

The Ginna reactor pressure vessel is a 2 Loop Westinghouse Design with internals similar to those shown in Figure1 (Attachment F). The internals, consisting of the upper and lower core support structure, are designed to support, align, and guide the core components, direct the coolant flow to and from the core components, and to support and guide the in-core instrumentation.

The components of the Ginna reactor internals per the Ginna Station UFSAR are divided into three parts consisting of the lower core support structure (including the entire core barrel and thermal shield), the upper core support structure, and the in-core instrumentation support structure.

It should be noted that industry work is typically divided into lower and upper internals only, where in the third Ginna component instrumentation such as the thimble tubes, is typically considered with the lower internals.

Additional discussions on the Ginna Reactor Internals structures is available in the Ginna Station UFSAR (Ref. 9.67) Section 3.9.5.

3.0 PROGRAM OWNER

The Nuclear Engineering Services Group is responsible for maintaining and implementing the Reactor Vessel Internals AMP.

4.0 DESCRIPTION OF THE GINNA AMP and INDUSTRY PROGRAMS

The Ginna Reactor Vessel Internals AMP is based on meeting the requirements of the ten elements of an aging management program as described by NUREG 1801, GALL Report section XI.M16 for PWR Vessel Internals. In the Ginna Station AMP, this is demonstrated through application of AMR methodology that credits inspections prescribed by the ASME Section XI In-Service Inspection Program, existing Ginna Station programs, and additional augmented inspections based on MRP-227 recommendations.

4.1 Existing Ginna Station Programs

Ginna Station has a number of existing programs that support aging management of reactor internals. Ginna Station has also performed modifications in response to Industry identified concerns that are described below.

4.1.1 ASME Section XI In-service Inspection (ISI) Requirements

The ASME Section XI ISI program is an existing program developed under ASME Section XI, Subsection IWB-2500, Examination Category B-N-1 (Ref. 9.27). Accessible areas of the reactor vessel interior, including internals, are examined using VT-3 visual techniques. This examination is performed on accessible components each period, or approximately every 3 years. At Ginna Station, this examination is performed under the direction of NDE procedure EP-VT-110, "Visual Examination of the Reactor Vessel and Removable Internal Structures" (Ref. 9.9).

ASME Section XI, IWB-2500, Examination Category B-N-2 (Ref. 9.27) applies to interior attachments beyond the beltline region. ASME Section XI, IWB-2500, Examination Category B-N-3, applies to removable core support structures. Examination of these interior attachments in the reactor vessel (Category B-N-2) and the reactor vessel internals (Category B-N-3) is required once every 10-year ISI interval in accordance with the Ginna Station ISI Program and Program Plan, Ref. 9.30. The examination is performed with the aid of remote visual examination tools in accordance with the EP-VT-110 procedure.

For bolting that is accessible for visual examination, the VT-3 examination can be relied upon to detect the effects of stress relaxation, which would cause the joints to loosen and ultimately crack due to high cycle fatigue. Ultrasonic examination may be performed to determine if there is any cracking evident in those areas where a direct visual examination cannot be performed.

Relevant conditions for these examination categories are found in ASME Section XI, IWB-3520, Standards for Examination Category B-N-1, Interior of Reactor vessel Examination

Category B-N-2, Welded Core Support Structures and Interior Attachments to Reactor Vessels, and examination Category B-N-3, removable core support structures.

The relevant conditions described include but are not limited to cracks, structural distortion, loose parts, missing parts, and foreign material inside the vessel. The VT-3 examination, combined with other nondestructive examination techniques, can be relied upon to detect wear damage on accessible surfaces and any other normal degradation found inside the reactor vessel. The Ginna Station Section XI, 4th Interval 10 year ISI examinations are currently being scheduled for the 2011 RFO.

4.1.2 Primary Water Chemistry Program

The Ginna Station Water Chemistry Control Program, (Ref. 9.10), as implemented by plant chemistry procedures CHA-PRI-STRATEGY, "Primary Water Chemistry Strategy" (Ref. 9. 11) and CH-PRI-SCHED "Primary System Analysis Schedule and Limits" (Ref. 9.12) can also be credited with mitigating IASCC. The Water Chemistry Control Program limits the concentration of oxygen, halogens, and sulfate species in the primary water, and therefore effectively prevents SCC and greatly reduces the probability of IASCC. The limits imposed by the chemistry monitoring program meet the intent of TR-1014986 -Rev 6, "EPRI PWR Primary Water Chemistry Guidelines", (Ref. 9.13).

4.1.3 LR-TTI-PROGPLAN (Thimble Tubes)

Flux thimble tubes are long slender stainless steel tubes seal welded at one end with flux thimble tube plugs which pass through the vessel penetration through, the lower internals assembly and finally extend to the top of the fuel assembly. The bottom mounted instruments (BMI) column assemblies provide a path for the flux thimbles into the core from the bottom of the vessel and protect the flux thimbles during operation of the reactor. The flux thimble provides a path for the neutron flux detector into the core and is subject to reactor coolant pressure on the outside and containment pressure on the inside.

The Ginna thimble tube inspection program is an existing plant specific program thatfollows Branch Technical position RLSB-1, Generic Aging management review which is included in appendix "A" of NUREG-1800 (Ref 9.52). The Ginna Thimble Tube inspection program manages cracking due to SCC of BMI guide tubes and cracking of BMI guide tube fillet welds. This program requires periodic visual inspection of fillet welds. The program also includes eddy current testing requirements for thimble tubes and includes criteria for determining sample size, inspection frequency flaw evaluation and corrective action in accordance with US NRC Bulletin 88-09 (Ref. 9.53), and NUREG 1801 (Ref 9.1). Specific Ginna actions are contained in the LR-TTI-PROGPLAN (Ref. 9.28).

4.2 Other Ginna Station Projects

4.2.1 Control Rod Guide Tube Split Pin Replacement Project

In general, SCC prevention is aided by adherence to strict primary water chemistry limits that effectively prevent SCC and greatly reduce the probability of IASCC. The limits imposed by the chemistry monitoring program are consistent with TR-1014986 "EPRI PWR Primary Water Chemistry Guidelines" Rev. 6, as described in the previous section.

Westinghouse Analysis indicated that failures of original Alloy X-750 Control Rod Guide Tube (CRGT) Split Pins was caused by SCC. The failed pins had been solution heat treated at less than 1800 Deg. F. after which they were age-hardened and then stressed during subsequent manufacture and installation processes.

In response to the industry concern, all 33 CRGT OEM split pins were preemptively replaced at Ginna Station during the 1986 RFO.

Replacement CRGT split pins for Ginna were supplied by Framatome. The replacement pins were supplied as "equivalent" replacements and manufactured from X-750 material that included special design considerations for installation torques, improved radius of curvature and heat treatments, which were intended to prolong the life of the split pin. Specific pin enhancements are discussed in the Ginna LRAM-RVI.

More recent inspections (2007/2008) have been performed at European facilities with similar split pins, known as Generation 4 split pins. Discussions with individuals familiar with the European experience resulted in knowledge of suspected signal anomalies in 5 Generation 4 split pins. Discussions indicate that additional review of the signal anomalies has resulted in a preliminary conclusion that SCC may have initiated in 1 of the 5 split pins investigated. While no replacement is currently planned at the European facility, additional inspections are anticipated in the Fall of 2009 at that facility.

Based on the preliminary information available from the 2007/2008 experience, the split pin issue has been entered into the Ginna Station Corrective Action Process. A contingency plan for split pin failure has been initiated for the Ginna Station CRGT split pins (Attachment "A", Table 1, CA-2009-003832 and Ref. 9.69). The evaluation provided by the Corrective Action Process based on reference 9.69, is expected to document that even if a split pin were to fail, the resulting condition is not an immediate safety concern.

The contingency includes a specific Metal Impact Monitoring System (MIMS) signal characteristic for control room recognition based on the potential failed split pin piece, size and weight, followed by prompt plant shut down in the case where the anticipated MIMS signal characteristic is received in the control room. Investigation of alternative split pin material is in process and split pin replacement or augmented inspections during the 2011 RFO are currently anticipated.

4.2.2 Baffle-Former Bolt Inspection/Replacement Project

Baffle-former bolts are potentially subject to IASCC, irradiation embrittlement, stress relaxation, and void swelling. Because cracking has typically occurred at the juncture of the bolt head and shank, and since this area is inaccessible for visual examination, VT-3 examination alone is not an effective method for managing aging effects that affect baffle-former bolts. VT-3 examination would detect missing bolt heads. Ultrasonic testing (UT) can be used to determine if cracks exist in bolts. Alternatively, selected bolts can be preemptively replaced with bolts fabricated from a more crack-resistant material, thus precluding or postponing future examinations. Because there is significant mechanical redundancy in the number of baffle-former bolts used in the baffle-former assembly, only a fraction of the total number of bolts would require replacement or an alternate pattern determined to guarantee integrity of the baffle-former joint until the end of extended plant life.

During the 1999 RFO, a preemptive baffle former bolt replacement was completed at Ginna Station as part of a WOG project known as "B" cubed (Refs. 9.32 & 9.33). Ginna Station Action Report (Ref. 9.6) summarizes that a total of 56 replacement bolts were installed during the 1999 refueling outage. These bolts were UT inspected and reported to have recordable defect-like indications. The bolts of concern were also part of a pre-qualified minimum bolt pattern for two-loop nuclear plants that was generated by the Westinghouse Owners Group and documented in WCAP-15036 (Ref. 9.7).

Additional hot cell testing of fourteen sample bolts was performed and results reported in "Ginna Hot Cell testing Report" (Ref 9.25). The destructive metallurgical analysis of the 14 whole bolts and 2 heads of the baffle former bolts removed from the Ginna RVI was performed to evaluate bolt microstructures. Voids were observed in the threaded end of one bolt but not in the head or the 304 SS locking device. The void volume, 0.004% maximum observed in the 347 SS bolt material, was small and preliminary extrapolation to the end of extended life using a simple square law suggested that void swelling should not be a concern.

Additional information on hot cell testing results from inspections performed on the Ginna baffle former bolts (BFB) removed in 1999 is available in reference 9.25.

The more recent work provided by MRP-227 calls for examination of Baffle former bolts based on additional evaluations and determinations that the 1999 inspections may have been performed early in plant life prior to the bolts reaching their maximum expected degradation peaks. As such, two potential courses of action, in accordance with Attachment "D", Table 1.1, are being evaluated for Ginna baffle former bolts going forward:

1) Inspect 100% of a revised minimum bolt pattern and include an option for inspection of alternative bolts predicted by a plant specific analysis. The option for alternative bolts is provided in the event that inaccessible bolts are discovered in the revised minimum required pattern.

   Only those bolts in the minimum pattern (or a suitable alternative bolt) would be inspected and the as-left pattern justified.

2) Perform a pre-emptive replacement campaign targeted at the revised replacement pattern bolts, with a 60 or 80 year life analysis option. Bolts removed would provide samples for destructive analysis (both original and perhaps those replaced in 1999, if replaced bolts are in the revised acceptable pattern) for hot cell testing and examination. The hot cell testing will allow for comparison to the previous examination results contained in Reference 9.25. Hot cell examinations and testing would then serve as the inspection basis for additional expansion examinations in future RFO's, if required, in order to meet the intent of MRP-227 expansion component inspections.

   Both options will require additional acceptable bolt pattern evaluations that could be based on more current leak before beak (LBB) Ginna-specific size applications and 60-80 year life options (Attachment "A", Table 1, Action Item NL-2009-000019-001). A reduction in the number of required bolts may be expected for at least the 60 year case, since the licensing basis LBB size for Ginna has been reduced to a smaller size then was previously used in the evaluations to support the 1999 inspections. The reduced LBB line break size combined with the fact that primary system conditions remained relatively constant following Uprate, as compared to the RCS conditions prior to Uprate, would indicate that the additional evaluation of the required number of BFB's would reduce the number of required bolts.

   Both options are currently being considered for Ginna going forward.

Barrel-former bolts are constructed from similar materials to baffle-former bolts, have a similar design to baffle-former bolts, and are similarly difficult to access and in some cases may be inaccessible. The barrel-former bolts are linked to the baffle-former bolts as an expansion component. However, they are located further from the core and thus receive less fluence, and are therefore considered less susceptible to the degradation mechanisms identified for the baffle-former bolts.

The susceptibility to cracking of these bolts will be evaluated considering baffle-former bolt examination results, and corrective actions for future outages determined as necessary.

The lower support column bolts of the lower support assembly are also made of stainless steel, are subject to the aging effects listed in Attachment "C" and are considered an expansion component item in Attachment "D", linked to the baffle-former bolts. Similar to the barrel-former bolts, additional hot lab evaluations of removed baffle former bolts will dictate the actions necessary to address the more distant lower support column bolts during future outages.

4.3 <u>Industry Programs</u>

4.3.1 WCAP-14577-1-A

As described previously in Section 2, the WOG topical report WCAP-14577-1-A, (Ref. 9.2) contains a technical evaluation of aging degradation mechanism and aging effects for Westinghouse RVI components. The WOG sent the report to the NRC staff to demonstrate that WOG member plant owners that subscribed to the WCAP could adequately manage effects of aging on Reactor Vessel Internals during the period of extended operation, using approved aging management methodologies of the WCAP to develop plant-specific aging management programs.

The aging management review for the Ginna Station internals, documented in LRAM-RVI, including required applicant action items, was completed in accordance with the requirements of the WCAP-14577-1A (Ref. 9.2). Attachment "C" tables present the results of the Ginna Station evaluation and the resulting programs; and additional inspections that are required for those components determined to be most susceptible to degradation mechanisms that are credited for license renewal.

Those components listed in attachment "C" "Program/Activity" column that credit "Reactor Vessel Internals Inspection Program", are addressed by the additional evaluations and analysis as described in the next section.

4.3.2 MRP-227

The MRP-227 Reactor Internals Inspections and Evaluation Guidelines were developed by a team of industry and NSSS vendors and International committee representatives who reviewed available data and industry experience on materials aging. The objective of the group was to develop a systematic approach for identifying and prioritizing inspection and evaluation requirements for reactor internals.

The key sequential steps in the process included the following:

- The development of screening criteria, with susceptibility levels for the eight postulated aging mechanisms relevant to reactor internals and their effects;
- An initial component screening and categorization, using the susceptibility levels and FMECA (failure modes, effects, and criticality assessment) to identify the relative ranking of the components;
- Functionality assessment of degradation for components and assemblies of components;
- Aging management strategy development combining the results of functionality assessment with component accessibility, operating experience, existing evaluations, and prior examination results to determine the appropriate aging management methodology, baseline examination timing, and the need for and the timing of subsequent inspections.

Through this process, the reactor internals for all three NSSS vendors operating PWR designs in the United States were evaluated in the MRP program and appropriate inspection, evaluation, and implementation requirements for reactor internals were defined.

The MRP-227 utilized the screening and ranking process to aid in the identification of required inspections for Primary and Expansion Components and credits Exiting component inspections when they were deemed adequate.

The basic description of each classification is as follows:

Primary:

Those PWR internals that are highly susceptible to the effects of at least one of the eight aging mechanisms were placed in the Primary group. The aging management requirements that are needed to ensure functionality of Primary components are described in these I&E guidelines. The Primary group also includes components which have shown a degree of tolerance to a specific aging degradation effect, but for which no highly susceptible component exists or for which no highly susceptible component is accessible.

Expansion:

Those PWR internals that are highly or moderately susceptible to the effects of at least one of the eight aging mechanisms, but for which functionality assessment has shown a degree of tolerance to those effects, were placed in the Expansion group. The schedule for implementation of aging management requirements for Expansion components will depend on the findings from the examinations of the Primary components at individual plants.

Existing Programs:

Those PWR internals that are susceptible to the effects of at least one of the eight aging mechanisms and for which generic and plant-specific existing AMP elements are capable of managing those effects, were placed in the Existing Programs group.

No Additional Measures Programs:

Those PWR internals for which the effects of all eight aging mechanisms are below the screening criteria were placed in the "No Additional Measures" group. Additional components were placed in the No Additional Measures group as a result of FMECA and the functionality assessment. No further action is required by these guidelines for managing the aging of the No Additional Measures components.

The categorization and analysis processes used by the MRP-227 approach is not intended to supersede any ASME B&PV Code Section XI (Ref .9.27) requirements. Any components that are classified as core support structures, as defined in ASME B&PV Code Section XI IWB 2500, Category B-N-3 have requirements that remain in effect and may only be altered as allowed by 10 CFR 50.55a.

The Industry program outputs of MRP-227 are classified in accordance with the requirements of the NEI-03-08 Guidelines (Ref. 9.54) and applied at Ginna Station through application of CNG-AM-4.01 (Ref 9.51) and IP-IIT-8 (Ref 9.47). For the MRP-227 guideline there are Mandatory, Needed and Good Practice elements as follows:

There is one Mandatory Element:

- **Mandatory:**

Each commercial U.S. PWR unit shall develop and document a PWR reactor internals aging management program (AMP) within thirty-six months following issuance of MRP-227-Rev. 0.

There are three "Needed" elements:

- **Needed:**

Each commercial U.S. PWR unit shall implement Tables 4-1 through 4-9 and Tables 5-1 through 5-3 for the applicable design within twenty-four months following issuance of MRP-227-A.

Initial Ginna Station Inspections have been scheduled to coincide with the 10 Year ISI program inspections scheduled for the 2011 RFO. The applicable Westinghouse tables contained in MRP Table 4.3 (Primary) 4.6 (Expansion) and Table 4.9 (Existing) are attached herein as Attachment "D" Tables 1.1, 1.2 and 1.3, respectively.

- **Needed:**

Examinations specified in these guidelines shall be conducted in accordance with the Inspection Standard MRP-228.

Inspection standards developed under MRP -228 (Ref 9.56) will be used for augmented inspection where required by MRP-227 outputs.

- **Needed:**

Examination results that do not meet the examination acceptance criteria defined in Section 5 of the MRP-227 guidelines shall be recorded and entered in the plant corrective action program and dispositioned.

Ginna station IP-CAP-1 Ref (9.18) will be applied as discussed in attribute section 5.7.

There is one Good Practice element:

Good Practice:

Each commercial U.S. PWR unit should provide a summary report of all inspections and monitoring, items requiring evaluation, and new repairs to the MRP Program Manager within 120 days of the completion of an outage during which PWR internals are examined. The MRP template should be used for the report.

As discussed in section 4.3.3 Ginna will participate in industry efforts going forward.

It should be noted that Appendix A of MRP-227 also includes a description of the attributes that comprise an acceptable AMP. These attributes are similar to the previously discussed attributes of the GALL Report and used for the Aging Management Review for Ginna station, LRAM-RVI as discussed previously in Sections 2.3 and 4.0. Evaluation of the Ginna License Renewal Reactor Vessel Internals Program Plan (LR-RVI-PROGPLAN) against GALL attribute elements is provided in Section 5 of this program plan.

As part of the Ginna License Renewal Project, Ginna Station agreed to participate in Industry activities associated with the development of the standard Industry Guideline for Inspection and Evaluation of Reactor Internals described above. These efforts have defined the required inspections and examination techniques for those components that credit this LR-RVI-PROGPLAN and the additional industry work. Those components correspond to the ones that are listed in the last column of the Attachment "C" tables that credit the "Reactor Internal Inspections Program" as being the basis for the recommended inspections. The results of the industry recommended inspections, published in the MRP-227 guidelines, serve as the basis for identifying any augmented inspections that are required to complete this PROGPLAN.

In that the MRP-227 guideline has been submitted to the NRC with the ultimate goal of producing a Safety Evaluation Review (SER) review and approval, discussions with the NRC have indicated that the Ginna program cannot be based solely on the MRP-227 work, at this time. Therefore, as discussed in previous section of this LR-RVI-PROGPLAN, the Ginna program has been rooted in meeting the GALL Report attributes and the previous approved work of the PWROG WCAP 14577-1-A. However, this LR-RVI-PROGPLAN also captures the results of additional evaluations, inspection recommendations, and forward looking concepts of the MRP-227 work summarized in the tables of Attachment "D".

4.3.2.1 MRP Applicability to Ginna Station

The applicability of MRP-227 to Ginna Station requires compliance with the following analysis assumptions:

- 30 years or less of operation with high leakage core loading patterns (fresh fuel assemblies loaded in peripheral locations) followed by implementation of a low-leakage fuel management strategy for the remaining 30 years of operation.

  Ginna Station fuel management program changed from a high to a low leakage core loading pattern prior to 30 years of operation.

- Base load operation, i.e., typically operates at fixed power levels and does not usually vary power on a calendar or load demand schedule.

  Ginna operates as a base load unit.

- No design changes beyond those identified in general industry guidance or recommended by the original vendors.

  MRP-227 states that the recommendations are applicable to all U.S. PWR operating plants as of May 2007 for the three designs considered. Ginna Station has not made any modifications to internals components in the time since May 2007.

  Conclusion:

  Based on the above review, the MRP-227 work is representative for Ginna Station.

4.3.3 Ongoing Industry Programs

4.3.3.1 Both the PWROG and the EPRI/MRP continue to sponsor activities related to Reactor Vessel Internals aging management. Ginna Station participation in current EPRI/MRP work will continue, through ongoing reporting of Ginna inspection results, and development of a standard submittal approach for other PWR's.

Ginna hosted a PEER team review of the initial draft of this PROGPLAN during January 2009. Substantial comments were received through the PEER team and were combined with the NRC telecom discussion comments noted previously in order to provide this revision of the PROGPLAN. The PEER team comments are expected to aid the MRP development of the generic submittal format for future plant submittals.

The PWROG is currently engaged in developing evaluation criteria methodology for Primary and Expansion components that require visual inspections as listed in MRP-227, attachment "D" primary and expansion tables.

In order to expedite the "primary component" evaluation criteria needed to support the Ginna 2011 RFO, Ginna station has funded and developed evaluation criteria for those "primary" components that require visual inspections (Ref 9.61). It is anticipated that the methodology used for the Ginna VT inspection "evaluation" criteria will serve as the basis for the PWROG approach to establish VT evaluation criteria going forward. It is anticipated that the Expansion component evaluation methodology will be based on a similar approach to that supplied for the Ginna station primary evaluation criteria, through the efforts of the PWROG (Ref. 9.62).

4.4 Summary of Section 4.0

It should be noted that both the WCAP and the MRP approach to aging management are based on the GALL approach to aging management strategies. That is, to determine which reactor internals passive components are most susceptible to the aging mechanisms of concern; and then determine the proper inspection or mitigating program that provides reasonable assurance that the component will continue to perform its intended function through the period of extended operation.

The WCAP based approach used at Ginna, for the initial basis of the license renewal application and subsequent NRC, SER. Applying the WCAP approach produced a Ginna specific AMR which produced the AMR summary table provided in attachment "C". The table credits the "Reactor Vessel Internals Inspection Program" for certain components as shown in the last column of the table. The Reactor Vessel Internals Inspection Program refers to this PROGPLAN and was intended to be based on additional Industry MRP requirements that are summarized by the Tables in attachment "D".

The additional evaluations and analysis completed by the MRP industry group, provided needed clarification to the level of inspection quality needed to determine the proper examination method and frequencies. The tables provided by the MRP-227 attachment "D", provide the level of examination required for each of the components evaluated.

Both approaches have yielded tables that credit existing programs or prescribe additional inspections that provide the input basis for revision of the reactor internals inspection implementation procedure EP-VT-110 procedure update.

This procedure is used for the In-service Inspection of the reactor internals in accordance with the ASME Section XI Code. Action item NL-2009-000021-001 documents the requirement for update of this procedure based on the augmented inspection required by this PROGPLAN.

It is the Ginna Station position that use of the AMR produced by the WCAP methodology combined with any additional augmented inspections required by the Industry tables provided in attachment "D", provides reasonable assurance that the reactor internals passive components will continue to perform their intended functions through the period of extended operation.

5.0 GINNA AGING MANAGEMENT PROGRAM ATTRIBUTE EVALUATIONS

The attributes of the Ginna Station Reactor Internals AMP and their compliance with NUREG-1801 (Generic Aging Lessons Learned (GALL) Report), Section XI.M.11, "PWR Vessel Internals" (Ref. 9.1) are described in this section.

5.1 SRP Element 1 - Scope of Program

Gall Report Element Definition:

The program is focused on managing the effects of crack initiation and growth due to stress corrosion cracking (SCC) or irradiation assisted stress corrosion cracking (IASCC), and loss of fracture toughness due to neutron irradiation embrittlement or void swelling. The program contains preventive measures to mitigate SCC or IASCC; ISI to monitor the effects of cracking on the intended function of the components; and repair and/or replacement as needed to maintain the ability to perform the intended function. Loss of fracture toughness is of consequence only if cracks exist. Cracking is expected to initiate at the surface and is detectable by augmented inspection.

The program provides guidelines to assure safety function integrity of the subject safety related reactor pressure vessel internal components, both non-bolted and bolted components. The program consists of the following elements: (a) identify the most susceptible or limiting items, (b) develop appropriate inspection techniques to permit detection and characterizing of the feature (cracks) of interest and demonstrate the effectiveness of the proposed technique, and (c) implement the inspection during the license renewal term. For example, appropriate inspection techniques may include enhancing visual VT-1 examinations for non-bolted components and demonstrated acceptable inspection methods for bolted components.

5.1.1 Ginna Program Scope

The Ginna Station RVIs consist of two basic assemblies:

- Upper internals assembly that is removed during each refueling operation to obtain access to the reactor core. The top of this assembly is clamped to a ledge below the vessel-head mating surface by the reactor vessel head. The core barrel fuel alignment pins of the lower internals assembly guide the bottom of the upper internals assembly.

- Lower internals assembly that can be removed, if desired following a complete core unload. This assembly is clamped at the same ledge below the vessel-head mating surface and closely guided at the bottom by radial/clevis assemblies.

Additional Reactor Vessel Internals details are provided in Section 3.9.5 and Section 4.2.1 of the UFSAR.

5.1.2 Subcomponents Subject to an Aging Management Review

The subcomponents of the Reactor Vessel Internals that require aging management review are indicated in the previously submitted, Table 2.3.1-3 of the R.E. Ginna Nuclear Power Plant Application for Renewal Operating License, (Ref 9.36). This table is included herein for convenience as Attachment "E". The table lists the subcomponents of the Reactor Vessel Internals that require aging management review along with each subcomponent's passive function(s) and reference(s) to the corresponding AMR Table(s) in Section 3 of the Ginna License Renewal Application.

The Ginna Station Reactor Internals Aging Management Review was conducted and documented in the Ginna Station Aging Management Review Report, LRAM-RVI (Ref. 9.35). The table, summarizing the results of that review, is reproduced herein as Attachment "C", "Ginna Station Aging Management Review Summary Table". The table identifies those components that are most susceptible or limiting for Ginna, based on the initial LRAM-RVI results. The last column in the table lists the Program/Activity that is credited to address the component and aging effect during the period of extended operation.

It should be noted that the program/activity credited in the last column referred to as "Reactor Vessel Internals Inspection Program", is this LR-RVI-PROGPLAN, which was intended to be based on additional industry MRP activities discussed in sections 4.3.2 and 4.4 above. The additional studies provided by the MRP-227 document, summarized herein in the tables of attachment "D", provides the basis for the required augmented inspections, inspection techniques to permit detection and characterizing of the feature (cracks) of interest, prescribed frequency of inspection, and examination acceptance criteria.

The Ginna program scope is based on previously established and approved GALL Report approaches through application of the WCAP 14577-1-A methodologies to determine those components that require aging management. Likewise, the additional evaluation summary results provided by the Industry MRP-227 results of attachment "D" is rooted in the GALL methodology and provides a basis for augmented inspections that were required to complete this Ginna Station Program by providing the inspection method, frequency of inspection and examination acceptance criteria.

5.1.3 Conclusion:

This element complies with the corresponding aging management attribute in NUREG-1801, Section XI.M16.

5.2 SRP Element 2 - Preventive Actions

GALL Report Element Definition:

The requirements of ASME Section XI, Subsection IWB, provide guidance on detection, but do not provide guidance on methods to mitigate cracking. Maintaining high water purity reduces susceptibility to cracking due to SCC. Reactor coolant water chemistry is monitored and maintained in accordance with the EPRI guidelines in TR-105714. The program description and evaluation and technical basis of monitoring and maintaining reactor water chemistry are presented in Chapter XI.M2, "Water Chemistry".

5.2.1 Ginna Preventative Action

The Ginna's reactor internals AMP includes the following existing program that complies with the requirements of this SRP Element. A description and applicability to the Ginna Station reactor internals AMP is provided in the following subsection.

5.2.2 Primary Water Chemistry Control Program

The Ginna Primary Water Chemistry Control Program (Ref. 9.10), as implemented by plant chemistry procedures CHA-PRI-STRATEGY, "Primary Water Chemistry Strategy" (Ref. 9. 11) and CH-PRI-SCHED "Primary System Analysis Schedule and Limits" (Ref. 9.12) reduces the susceptibility of RVI components to SCC and IASCC. The chemistry monitoring program places limits on oxygen, halogens, and sulfate species that effectively prevent SCC for austenitic stainless steel components and greatly reduce the probability of IASCC.
The limits imposed by the chemistry monitoring program are consistent with TR-1014986 "EPRI PWR Primary Water Chemistry Guidelines" Rev. 6 (Ref. 9.13).

5.2.3 Conclusion:

This element complies with the corresponding aging management attribute in NUREG-1801, Section XI.M16.

5.3 SRP Element 3 - Parameters Monitored, Inspected, and/or Tested

GALL Report Element Definition:

The program monitors the effects of cracking on the intended function of the component by detection and sizing of cracks by augmentation of the In Service Inspection requirements in accordance with the requirements of the ASME Code, Section XI, Table IWB 2500-1.

5.3.1 Ginna Parameters Monitored, Inspected, and/or Tested

The Ginna program monitors for the end effect of the identified degradation mechanisms of cracking, loss of bolted joint integrity and wear through application of existing ASME Code Section XI, ISI Program, augmented inspections, other existing plant and industry programs.

5.3.2 The Ginna Station In-service Inspection Program

This existing program (Ref. 9.45) is credited for inspection of those reactor internals assemblies requiring aging management as identified by the Ginna Station AMR and listed in Attachment "C". Industry programs also credited use of the ASME Code inspections where inspections are adequate to detect internals degradation mechanisms such as those listed in the Existing Programs of Attachment "D", Table 1.3.

The requirements of ASME Section XI, Subsection IWB, prescribe using VT-3 methods for examination of Categories B-N-1, B-N-2 and B-N-3. These examinations have been performed during each 10–year ISI interval, i.e.1979, 1989 and 1999 when the upper and lower internals assemblies were removed from the vessel. Inspections and examinations are documented in procedure EP-VT-110 (Ref. 9.9). These inspections are scheduled for the 2011 RFO.

The tables provided in attachments "C" and "D" identify which components credit the use of the ASME Section XI inspections for license renewal.

5.3.3 Augmented Examinations

Per the Ginna Station AMR, LRAM-RVI, the components requiring aging management are listed in summary table Attachment "C". In Attachment "C", credit was taken for the existing

Section XI ISI inspection Program, Water Chemistry Program, and this Reactor Vessel Internals Inspection Program, as noted in the last column identified as "Program/Activity".

Augmented inspections, frequencies and inspection techniques were to be determined based on the additional evaluations provided by the Industry in order to supplement the ASME Section XI Program inspections. These evaluations were to provide needed clarification as to the level of inspection quality needed; and to determine the proper examination method and frequencies.

The summary tables provided by the MRP-227, Attachment "D", provide the level of examination, method and frequency of inspection required for each of the components evaluated by the industry, which as previously discussed in Section 4.4.

Therefore, augmented inspections credited in Attachment "C" from the Ginna Reactor internals AMR, as specified in section 5.3.3.1, are taken from the additional industry evaluations summarized in Attachment "D".

In the case where Attachment "D" may list additional Primary Components that require inspections that do not appear in the Attachment "C" list, the additional components of attachment "D" are included for augmented inspection as listed in Section 5.3.3.2.

Therefore, the Ginna program will not only meet the original requirements of the LRAM-RVI methodology previously approved by the Ginna Station SER, but will also capture the intent of the industry MRP-227 (Rev. 0) guidelines.

5.3.3.1 Augmented Inspections required by LRAM-RVI Attachment 3.

These components are addressed in the order that they are found in the Table of attachment "C".

5.3.3.1.1 Lower Core Plate and Fuel Pins

The lower core plate (LCP) is inspected as part of the ISI program to VT-3 as shown in Figure 19 of procedure EP-VT-110. It should be noted that the evaluations of the lower core plate performed per MRP-227 have shown that no additional measures for the 12 foot core design is required for the LCP used at Ginna Station (Ref. Att. "D", Table 1.3, Existing Programs, Lower Internals Assembly). Therefore the existing ISI program is credited to detect aging in the lower core plate. No augmented inspections are required.

5.3.3.1.2 Lower Support Columns

The MRP tables divide this component in to two subassemblies, lower support column bolts and lower support column bodies.

5.3.3.1.2.1 Lower Support Column Bolts

Can be visually inspected to a level VT-3 following core removal. These bolts are identified as expansion components in the MRP tables of Attachment "D" Table 1.2. Therefore, future inspections would be based on the results of BFB hot cell testing results.

The augmented examination called for is a volumetric UT with initial and subsequent examinations dependent on results of baffle-former bolt examinations. Inspection coverage is to be 100% of accessible bolts or as specified by plant specific justification.

UT examination of bolts has been performed in other Internals locations such as the baffle former assembly. Therefore, it is reasonable that UT be specified in this location.

5.3.3.1.2.2 Lower Support Column Bodies

Lower support column bodies are identified as expansion components in the tables of Attachment "D" based on the industry guidelines. They are tied to examinations of the lower flange of the Control rod guide tube inspection results.

The augmented examination called for is an enhanced visual examination (EVT-1) with initial and subsequent examinations dependent on results of control rod guide tube lower flange inspection results.

Inspection coverage is to be 100% of accessible support columns.

Enhanced visual (EVT-1) examination has the same requirements as the ASME Code Section XI visual (VT-1) examination, with additional requirements as discussed in MRP-227. These enhancements are intended to improve the detection and characterization of discontinuities taking into account the remote visual aspect of reactor internals examinations. As a result, EVT-1 examinations are capable of detecting small surface breaking cracks and surface crack length sizing when used in conjunction with sizing aids (e.g., landmarks, ruler, and tape measure). EVT- 1 examination has been selected to be the appropriate NDE method for detection of cracking in plates or their welded joints. Thus the relevant condition applied for EVT-1 examination is the same as found for cracking in the ASME section XI code, which is crack-like, surface-breaking indications.

EVT-1 examinations have been applied to other similar type inspections for BWR internals. Therefore, it is reasonable to expect the EVT-1 examination to detect indications in PWR components.

5.3.3.1.3 Core Barrel and Flange

The core barrel and flange have been evaluated by the industry program to be a "Primary" component as shown in Attachment "D", Table 1.1. The additional evaluations show that the upper core barrel flange weld is the primary component of interest.

The required augmented inspection is periodic enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period with subsequent examinations at ten year intervals.

Inspection coverage is 100% of one side of the accessible surfaces of the weld and adjacent base metal.

The basis for use of an EVT-1 is provided in the previous section.

5.3.3.1.4 Thermal Shield

The thermal shields flexures (supports) have been evaluated by the industry program to be a "Primary" component as shown in Attachment "D", Table 1.1.

The required augmented inspection is a VT-3 examination performed no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations are on a 10 year interval.

The required examination coverage is 100% of the thermal shield flexures.

VT-3 examinations are used to detect cracking, loss of preload, and wear. The type of visual examination utilized will be, as a minimum, the VT-3 examination specified by ASME Section XI, Subsection IWB, Category B-N-3. Augmented examinations, if required for the flexures, may consist of VT-1, enhanced VT-1 examinations (defined as a visual examination with a finer resolution than the standard VT-1 requirements in ASME Section XI), or additional characterization when warranted by the size of the indication that must be detected to preserve intended functions.

5.3.3.1.5 Bolting

The last component group of Attachment "C" summary table, "Bolting" which includes the Lower support column bolts, baffle-former bolts, and barrel-former bolts.

5.3.3.1.5.1 Lower Support Column Bolts

The Lower support column bolts have been addressed in Section 5.3.3.1.2.1 above.

5.3.3.1.5.2 Baffle-Former Bolts

The baffle-former bolts have been evaluated by the industry program to be a "Primary" component as shown in Attachment "D", Table 1.1.

The required augmented inspection is a baseline volumetric (UT) examination between 25 and 35 EFPY, with subsequent examination after 10 to 15 additional EFPY to confirm stability of the bolting pattern. Re-examination for high-leakage core designs requires continuing examinations on a ten-year interval.

Additional discussion on the Ginna–specific options is contained in Section 4.4.2.

UT inspection of baffle-former bolts has been previously demonstrated during 1999 inspections performed at several PWR facilities.

5.3.3.1.5.1 Barrel-Former Bolts

The barrel-former bolts have been evaluated by the industry program to be an expansion component as shown in Attachment "D", Table1.2.

The required augmented inspection is a Volumetric UT examination with initial and subsequent examinations dependent on results of baffle-former bolt examinations and testing. The coverage is to be 100% of accessible bolts. Accessibility may be limited by presence of thermal shield.

UT inspection of baffle-former bolts has been previously demonstrated during 1999 inspections performed at several PWR facilities.

5.3.3.2 Augmented Inspections required by MRP-227

This section includes augmented inspections for those internals components that have been classified as "Primary Components" by the industry list provided in attachment "D".

This section adds additional industry augmented inspections required by the "Primary Component" list of Attachment "D", Table 1.1. By adding the augmented inspections for the Primary Components, Ginna station will not only meet the original Ginna AMR intent; but, will also meet the intent of the anticipated industry program described by MRP-227, Rev 0.

5.3.3.2.1 The following components will be inspected in accordance with the augmented requirements of Table 1.1 of Attachment "D":

1. Control Rod Guide Tube Assembly (Guide plates and cards)
2. Control Rod Guide Tube Assembly (Lower flange welds)
3. Baffle former Assembly, Baffle Edge Bolts
4. Alignment and interfacing components (Internals Hold Down Spring, N/A-Ginna is 410 Stainless Steel)
5. Baffle Former Assembly, Assembly

5.3.4 Industry Issues Program Outputs

5.3.4.1 Thimble Tubes

The Ginna Thimble Tube Inspection Program is an existing plant specific program that follows Branch Technical position RLSB-1, "Aging Management Review –Generic" which is included in Appendix "A" of NUREG-1800 (Ref 9.52). The Ginna Thimble Tube Inspection Program manages cracking due to SCC of BMI guide tubes and cracking of BMI guide tube fillet welds. This program requires periodic visual inspection of fillet welds.

The program also includes eddy current testing requirements for thimble tubes and includes criteria for determining sample size, inspection frequency flaw evaluation and corrective action in accordance with US NRC Bulletin 88-09 (Ref. 9.53), and NUREG 1801 (Ref 9.1). Specific Ginna actions are summarized in reference 9.28.

5.3.4.2 Industry Reactor Internals Guidelines-Discussion

Per the Ginna Station license renewal SER (Ref. 9.37), Ginna Station committed to participate in industry activities associated with developing a standard industry program to provide inspection and evaluation guidelines for reactor internals inspections. The industry program results are summarized as inspection and evaluation guidelines as documented in Ref 9.38. Augmented inspections as prescribed by the industry efforts are included in Attachment "D", Tables 1.1, 1.2 and 1.3.

The tables in Attachment "D" represent a consensus view of industry experts that call for additional inspections as noted. The tables are based on additional evaluations, analyses and expert opinion reviews that were performed by the industry in order to determine which components of the reactor internals require augmented inspections. This approach demonstrated that degradation mechanisms would not inhibit the intended functions of the internals components as they continue to operate through the period of extended operation. These evaluations form the basis for the completion of Ginna LRAM-RVI table Attachment "C" components that indicate additional evaluations by the industry were required to identify the necessary inspections, frequencies and methods.

The industry program compiled available research data; additional evaluation techniques, such as an IASCC predictive model; and expert opinion evaluations in order to determine what reactor internals components required additional inspection. The industry outputs are contained in References 9.40, 9.41, 9.42 and 9.43. Based on these outputs, a screening and ranking process was developed and applied to all removable reactor internals components in order to identify those components that required additional inspections above and beyond the inspections required by the ASME Section XI Code requirements.

Inspections, as identified in the tables of Attachment "D", are intended to provide reasonable assurance that the existing licensing basis functions would not be impacted by the degradation mechanisms identified in the tables. The augmented examinations are based on a hierarchy of components that leads to the tables provided in Attachment "D". The tables rank the components as Primary, Expansion and Exiting where in the "Primary" component list is predicted to contain the most susceptible component to the particular degradation mechanism. These components are expected to display the identified degradation mechanism at first on-set.

In the MRP process, each of the vessel internals subcomponents has been categorized using the screening and categorization process described in MRP-191 (Ref. 9.43).

Augmented inspections recommended by the MRP-227 work have been used in Section 5.3.3 to complete this LR-RVI-PROGPLAN which is credited in the LRAM-RVI Table Attachment "C". These augmented inspections have been taken from the tables of Attachment "D" and are scheduled for the 2011 RFO, in order to coincide with the Ginna Station 10 year ISI examinations. Existing Ginna Station NDE procedure EP-VT-110 (Ref. 9.46) will be modified following the 2009 RFO through action item NL-2009-000021-001 of Attachment "A", Table 1. Modifications to the procedure are being delayed until after the 2009 RFO as portions of the upper internals are scheduled to be examined during the up-coming Fall 2009 RFO. The 2009 components are part of the normal "period" examination required by ASME Section XI ISI Program (Ref 9.45).

5.3.4.3 Conclusion:

The Ginna Station AMP for reactor internals monitors for the effects of aging mechanisms by inspecting for cracking, wear and other relevant conditions in reactor internals components. These inspections performed in accordance with the requirements of the ASME Code, Section XI, Table IWB 2500-1, existing programs and the guidance of the MRP-227 program recommendations provide reasonable assurance that the intended functions of the components will continue to be met through the period of extended operation.

This element complies with the corresponding aging management attribute in NUREG-1801, Section XI.M16.

5.4 <u>SRP Element 4 - Detection of Aging Effects</u>

GALL Report Element Definition:

The extent and schedule of the inspection and test techniques prescribed by the aging management program are designed to maintain structural integrity and ensure that aging effects will be discovered and repaired before the loss of intended function. Inspections can reveal crack initiation and growth.

Vessel internal components are inspected in accordance with the requirements of ASME Section XI, Subsection IWB, examination category B-N-3 for all accessible surfaces of the reactor core support structure that can be removed from the vessel. The ASME Section XI inspection specifies visual VT-3 examination to determine the general mechanical and structural condition of the component supports by (a) verifying parameters, such as clearances, settings, and physical displacements, and (b) detecting discontinuities and imperfections, such as loss of integrity at bolted or welded connections, loose or missing parts, debris, corrosion, wear, or erosion.

5.4.1 Ginna Detection of Aging Effects

5.4.1.1 Visual examinations

The extent of the Ginna inspections that will be performed during the 2011 RFO in accordance with the requirements of ASME Section XI requirements and augmented inspections prescribed by the industry program are listed in EP-VT-110 and the Attachment "D" tables. Visual examinations are typically used to detect cracking and wear. The type of visual examination used will be, as a minimum, the VT-3 examination specified by ASME Section XI, Subsection IWB, Category B-N-3.

Augmented examinations for cracking may consist of VT-1, or Enhanced VT-1 (EVT-1) examination which is defined as a visual examination with a finer resolution than the standard VT-1 requirements of ASME Section XI, or volumetric examination when warranted.

Augmented inspections of Attachment “C” and Attachment “D”, Tables 1.1, 1.2, and 1.3 identify the effects of aging, the examination parameters to be monitored and the frequency and condition monitoring programs needed to maintain the continued functionality of the SSC.

Industry inspection standards for augmented inspections are contained in MRP-228 (Ref. 9.56).

Other examination methodologies selected for use in the industry guidelines are visual (VT-1 and EVT-1) examinations. These examinations were selected where a greater degree of detection capability than visual (VT-3) examination is needed to manage the aging effect. Unlike the detection of general degradation conditions by visual (VT-3) examination, visual (VT-1) and enhanced visual (EVT-1) examinations are conducted to detect discontinuities and imperfections on the surface of components, including conditions such as cracks, wear, corrosion, or erosion. Specifically, VT-1 is used for the detection of surface discontinuities such as gaps, while EVT-1 is used for the detection of surface breaking flaws.

When specified, a visual (VT-1) examination is conducted in accordance with the requirements of the ASME Code (Ref. 9.57). Enhanced visual (EVT-1) examination will be conducted in accordance with any additional requirements for (EVT-1) examination (such as camera scanning speed) if specified in the MRP 228 Inspection Standard (Ref 9.56).

The current ASME Code (Ref. 9.57) requirements for visual (VT-1) examination became more rigorous than the previous ASME Code versions. Many previous VT- 1 examinations were only required to discern a 1/32” black line on a gray background. These limitations led the NRC and industry to adopt modified visual examinations for use in detecting flaws discovered in boiling water reactor (BWR) internals. The most recent research conducted by the EPRI Non-Destructive Examination (NDE) Center established the VT-1 character heights specified in Reference 9.57 as equally or better able to detect the degradation effects than the modified visual examination requirements developed previously.

5.4.1.2 Volumetric Examinations

Ultrasonic examination was selected for volumetric examinations for cases where visual or surface examination is unable to detect the effect of the age related degradation mechanism such as for baffle-former bolts inspections. Ultrasonic testing (UT) or other equivalent NDE technique may be used to detect cracking. UT of the baffle-former bolts is used to detect cracking caused by IASCC under the baffle-former bolt head with go/no go criteria applied for detection.

Indications recorded during the volumetric examinations of baffle-former bolt will be evaluated using the procedures similar to those found in Ref. 9.7 which were developed for the 1999 inspection of baffle-former bolts, or additional plant or plant grouping specific evaluation /studies performed in conjunction with the original equipment manufacturer/or equivalent design supplier.

For the Ginna components requiring volumetric inspection, such as baffle-former bolts, it is anticipated that Ginna-specific or plant grouping evaluations, (such as a two loop design grouping) if feasible, will be performed in order to develop acceptable bolting patterns to support the previously discussed options on baffle-former bolts in Section 4.2.2, that will ensure structural integrity of the assembly for the remaining plant life (Action Item NL-2009-000019-001).

5.4.1.3 Conclusion:

Detection of indications that are required by the ASME Section XI, ISI Program are well established and field-proven through the application of the Section XI Program. Those augmented inspections that are taken from the MRP-227 recommendations will be applied through use of the MRP 228 Inspection Standard (Ref. 9.56).

This element is consistent with the corresponding aging management program attribute in NUREG-1801, Section XI.M16.

5.5 SRP Element 5 - Monitoring and Trending

GALL Report Element Definition:

Inspection schedules in accordance with IWB-2400, assessment of susceptible or limiting components or locations, and reliable examination methods provide timely detection of cracks. The scope of examination expansion and re-inspection beyond the baseline inspection are required if flaws are detected.

5.5.1 Ginna Program – Monitoring and Trending

Inspections credited in LRAM-RVI Attachment "C", are based on utilizing the Ginna 10 year ISI program, and the augmented inspections described in Section 5.3.3 derived from the industry program contained in Attachment "D". These inspections are scheduled to be conducted in conjunction with the 4th Interval, 10 Year ISI examinations.

Attachment "D" Tables 1.1, 1.2, and 1.3 identify the anticipated extent of the aging effects and timely inspection/mitigation actions for Primary components. As discussed in Section 2.3, and Reference 9.68 WCAP scheduling studies, an assessment of the MRP-227 "Primary" components provides reasonable assurance for demonstrating the "Primary" components current capacity to perform their intended functions.

Reporting requirements are included as part of the MRP-227 guidelines. Inspections results reporting will enable the industry to monitor reactor internals degradation on an on-going industry basis as the period of extended operation moves forward. Reporting of examination results will allow the industry to monitor and trend results and take appropriate pre-emptive action through update of the MRP guidelines.

5.5.2 Conclusion:

This element of the Ginna Station program complies with the corresponding aging management attribute in NUREG-1801, Section XI.M16.

5.6 SRP Element 6 - Acceptance Criteria

GALL Report Element Definition:

Any indication or relevant condition of degradation is evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400 and IWB-3500.

5.6.1 Ginna Program Acceptance Criteria

Those recordable indications that are the result of inspections required by the exiting Ginna Station ISI program scope are evaluated in accordance with the applicable requirements of the ASME Code through the existing IP-CAP-1 Corrective Action Process.

Inspection acceptance and expansion criteria have been provided in Attachment "D" Table 1.4. These criteria will be included in the update to EP-VT-110 tracked by Action Item NL-2009-000021-001 to enable the examiner to identify examination acceptance criteria.

Augmented inspections discussed in Section 5.3.3, that result in recordable relevant conditions, will be entered into the plant Corrective Action Program and addressed by appropriate actions that may include enhanced inspection, repair, replacement, mitigation actions or analytical evaluations using a "Flaw Evaluation" handbook-type approach such as those documented in WCAP 17016 NP (Ref. 9.61) or Reference 9.62 to support continued component or assembly functionality.

Additional analysis to establish acceptable bolting pattern evaluation criteria for the baffle-former bolt assembly is being tracked by Action Item NL-2009-000019-001.

Additional analysis to establish Attachment "D" expansion component evaluation criteria (Ref. 9.62) is being performed through the efforts of the PWROG and is tracked by Action Item NL-2009-000024-001.

5.6.2 Conclusion:

This element complies with the corresponding aging management attribute in NUREG-1801, Section XI.M16.

5.7 <u>SRP Element 7 - Corrective Actions</u>

GALL Report Element Definition:

Repair and replacement procedures are equivalent to those requirements in ASME Section XI. Repair is in conformance with IWB-4000 and replacement occurs according to IWB-7000. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing corrective actions.

5.7.1 Ginna Program

The NRC has addressed this element in NUREG-1801, Appendix, Quality Assurance for Aging Management Programs.

R.E. Ginna has an established 10 CFR Part 50, Appendix B, Program that addresses the elements of corrective actions, confirmation process, and administrative controls. The R.E. Ginna Program includes non-safety related structures, systems, and components. (IP-CAP-1, Ref. 9.18)

5.7.2 Conclusion:

This element complies with the corresponding aging management attribute in NUREG-1801, Section XI.M16.

5.8 <u>SRP Element 8 - Confirmation Process</u>

GALL Report Element Definition:

Site quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR 50, Appendix B. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing the confirmation process and administrative controls.

5.8.1 Ginna Program

See SRP Element 7- "Corrective Actions" in Section 5.7 above.

5.8.2 Conclusion:

This element complies with the corresponding aging management attribute in NUREG-1801, Section XI.M16.

5.9 SRP Element 9 - Administrative Controls

GALL Report Element Definition

See Item 8, above.

5.9.1 Ginna Program

See SRP Element 7- "Corrective Actions" above.

5.9.2 Conclusion:

This element complies with the corresponding aging management attribute in NUREG-1801, Section XI.M16.

5.10 SRP Element 10 - Operating Experience

Gall Report Element Definition:

Because the ASME Code is a consensus document that has been widely used over a long period of time, it has been shown to be generally effective in managing aging effects in Class 1, 2, or 3 components and their integral attachments in light-water cooled power plants.

In PWRs, stainless steel components have generally not been found to be affected by SCC because of low dissolved oxygen levels and control of primary water chemistry. However, the potential for SCC exists due to inadvertent introduction of contaminants into the primary coolant system from unacceptable levels of contaminants in the boric acid; introduction through the free surface of the spent fuel pool, which can be a natural collector of airborne contaminants (NRC IN 84-18); introduction of relatively high levels of oxygen during shutdown; or from aggressive chemistries that may develop in creviced regions. Cracking has occurred in Stainless Steel baffle-former bolts in a number of foreign plants (NRC IN 98-11) and has now been observed in plants in the United States.

5.10.1 Ginna Program

Extensive industry and Ginna Station operating experience has been reviewed during the development of the Reactor Vessel Internals Program. The experience reviewed includes NRC Information Notice 98-11, "Cracking Of Reactor Vessel Internal Baffle Former Bolts In Foreign Plants", and NRC Information Notice 84-18, "Stress Corrosion Cracking in PWR Systems". Most of the industry operating experience reviewed has involved cracking of austenitic stainless steel baffle-former bolts, or SCC of high-strength internals bolting. SCC of guide tube split pins has also been reported.

As described previously, a review of plant-specific experience with reactor vessel internals reveals that Ginna Station has responded proactively to industry issues with respect to reactor internals degradation. Two examples that demonstrate this proactive response are the preemptive replacement of control rod guide tube split pins in 1986, and augmented examination and preemptive replacement of baffle-former bolts in 1999. These are briefly described below.

The control rod guide tubes' (33) split pins were replaced at Ginna Station during the 1986 refueling outage. The new pins were of a new design supplied by Framatome, which had been installed in French nuclear reactors where SCC of split pins had been discovered. The new pin was compatible with the existing guide tube at Ginna Station, and utilized the same basic pin material as the original, i.e., Inconel X-750. Framatome had established, at that time, that the heat-treatment, machined configuration, and service conditions to which the material was subjected, and not the material itself, were responsible for the stress corrosion cracking.

During the 1999 refueling outage, all accessible baffle-former bolts were inspected by UT at Ginna Station (Ref. 9.6). A small number (approximately 9.2%) of these bolts were reported to contain defect-like indications. A sample of intact bolts with indications (as well as two bolt head segments) was sent to the Westinghouse hot cell for metallurgical analysis (Ref. 9.25). Destructive analysis of the bolts with flaw-like UT indications revealed no evidence of head/shank cracking. Fractographic analysis of one bolt head revealed fracture features typical of intergranular cracking which would be expected of IASCC. TEM studies showed evidence of voids near the threaded end of one bolt, but not at the head end. The void volume (0.004% maximum observed in the 347 SS bolt material) was small and preliminary extrapolation to the end of extended life using a simple square law suggested that void swelling should not be a concern. Detailed results of the hot cell tests and analyses are summarized in the AMR report for the reactor vessel internals and in Reference 9.25.

Early plant operating experience related to hot functional testing and reactor internals is documented in Ref. 9.65. Inspections performed as part of the 10 year ISI program would be expected to discover overall general internals structure degradation.

Ginna Station has also participated in the EPRI Reactor Vessel Internals Issue Task Group (EPRI RI-ITG, Focus Group, and Core Writers Group), and now participates in the MRP Assessment ITTG, which has now replaced the Core Writers Group as the group responsible for the Reactor Internals Guidelines.

The Assessment ITG continues to monitor and engage in ongoing research efforts on aging of reactor vessel internals as they represent themselves in the US and Worldwide research efforts and will continue to provide guidance to utilities on corrective actions for aging effects, as conditions warrant. The Pressurized Water Reactor Owners Group (PWROG) has established a materials sub-committee on rector internals primarily to establish expansion component evaluation criteria, as discussed above, and to provide an interface with the MRP working groups to monitor activities related to reactor vessel internals.

In addition, industry operating experience is routinely reviewed by Ginna Station System engineers using INPO OE and the barrier analysis concept for the determination of additional actions and lessons learned that can be incorporated into the plant systems quarterly health reports and for consideration for incorporation into plant programs.

A key element of the MRP-227 Guideline is the reporting of age related degradation of reactor vessel components. Ginna Station, through its participation in PWROG and EPRI-MRP activities, will continue to benefit from the reporting of inspection information and will share its own operating experience with the industry through those groups or INPO, as appropriate.

5.10.2 Conclusion:

This element complies with the corresponding aging management attribute in NUREG-1801, Section XI.M16.

5.11 Program Conclusion

The Reactor Vessel Internals Program is a new program that is consistent with NUREG-1801 Section XI.M16, "PWR Vessel Internals" (Ref. 9.1) with the following 2 noted exceptions:

1.) The first exception is that NUREG-1801, Section XI.M16 specifies examination schedules in accordance with IWB-2400, which requires core support structures to be inspected once during each 10-year interval. While this applies to the VT-3 examinations, some augmented examinations may be performed only once unless degradation is detected. The results of the initial augmented exams in 2011 and applicable industry operating experience will determine the frequency of subsequent examinations.

2.) The second exception is that NUREG-1801, Section XI.M16 specified an examination detection threshold method capable of resolving 0.0005 inch features of interest. Additional analysis and subsequent classifications by the industry as documented in the tables of Attachment "D", show that the inspection method required by the tables is adequate to detect the identified effect/mechanism.

6.0 DEMONSTRATION

The examinations required by ASME Section XI for the Ginna Station reactor vessel internals have been performed during each ten–year interval since plant operations commenced. The examination results are well documented and no unacceptable conditions have been reported.

Ginna Station has been proactive in responding to industry experience regarding reactor vessel internals degradation. Alloy X-750 guide tube split pins were replaced at Ginna Station in response to SCC failures of these pins in other Westinghouse plants. The replacement pins for Ginna were fabricated from the same nickel-based alloy, but with modified geometry and heat treatment to increase resistance to SCC.

All accessible Type 347 stainless steel baffle-former bolts in the Ginna Station reactor vessel internals were inspected by UT during the 1999 refueling outage (Ref. 9.6). The UT examination identified a number of bolts with flaw-like indications. A number of bolts sufficient to guarantee the structural adequacy of the baffle-former assembly were replaced. The replacement bolts were fabricated from Type 316 stainless steel, which is expected to be more resistant to IASCC than Type 347. Destructive metallurgical analysis of a sample of intact bolts containing flaw-like UT indications suggested false positive UT indications. Further details of the test results are documented in the AMR report (Ref. 9.31) for the Ginna reactor vessel internals and in the "Ginna Hot Cell Testing Report" (Ref. 9.25).

The Water Chemistry Control Program at Ginna Station has been effective in maintaining oxygen, halogens, and sulfate at levels sufficiently low to prevent SCC of the reactor vessel internals. LR-H2OC-PROGPLAN (Ref. 9.10), "Water Chemistry Control Program Basis Document", contains details on the effectiveness of primary water chemistry control.

Review of QA audit reports, NRC inspection reports, and INPO evaluations since 1999 indicate no unacceptable issues related to reactor vessel internals inspections.

As described in previous sections of this PROGPLAN, augmented inspections, derived from the evaluations that support the industry MRP-227 Guidelines, have been utilized in this PORGPLAN to build on existing plant programs. This approach is expected to encourage detection of a degradation mechanism at its first appearance. This approach provides reasonable assurance that the internals components will continue to perform their intended function through the period of extended operation.

This PROGPLAN also fulfills the approved license renewal methodology requirement to identify the most susceptible components and to inspect those components with an indication detection level commensurate with the expected degradation mechanism indication.

These examinations have been scheduled to be performed at Ginna Station during the 2011 RFO. The augmented inspections discussed in Section 5.3.3 will be integrated into the inspection procedures used to perform the ASME Section XI, 10 year ISI examinations procedure EP-VT-110 (Ref. 9.46). Integration of the required inspections will be tracked to completion by Action Item NL-2009-000021-001, as listed in Attachment "A", Table 1.

As discussed in Section 4.3.3, the industry MRP-227 Guidelines also provide for updates as experience is gained through inspection results. This feedback loop will enable updates based on actual inspection experience.

As documented in quarterly System and Program Health Reports, Operating Experience (OE) reports are continuously reviewed by system engineers and program engineers for applicable issues that indicate operating procedures or programs require updates based on new OE.

The augmented inspections described in Section 5.3.3, combined with the ASME Section XI, ISI program inspections, exiting Ginna Station programs, and use of OE reports provide reasonable assurance that the reactor internals will continue to perform their intended functions through the period of extended operation.

7.0 REQUIRED PROGRAM ENHANCEMENTS / IMPLEMENTATION SCHEDULE

The information that was included in this section is now consolidated in Attachment A, Table 1.

8.0 SUMMARY OF IMPLEMENTING DOCUMENTS / RESPONSIBLE DEPARTMENT

The information that was included in this section is now consolidated in Attachment A, Table 1.

9.0 REFERENCES

9.1 NUREG-1801 "Generic Aging Lessons Learned (GALL) Report," Dated July 2001

9.2 WCAP-14577 Rev. 1-A: "License Renewal Evaluation: Aging Management For Reactor Internals", Westinghouse Electric Company, March 2001

9.3 NRC Information Notice 84-18, "Stress Corrosion Cracking in PWR Systems", March 7, 1984

9.4 "Void Swelling of Austenitic Internals and its Possible Implications for PWRs", presentation by F.A. Garner at the EPRI RPV Internals/JOBB ITG Meeting, July 20-21, 1999, Milwaukee, Wisconsin

9.5 NRC Information Notice 98-11:"Cracking Of Reactor Vessel Internal Baffle Former Bolts In Foreign Plants", March 25, 1998

9.6 Action Report 99-0469, "Reactor Vessel Baffle Bolts with UT Indications"

9.7 WCAP-15036, Rev. 1, "Determination of Acceptable Baffle-Barrel Bolting Distribution Under Fault Conditions"

9.8 10 CFR 54 "Requirements for Renewal of Operating Licenses for Nuclear Power Plants"

9.9 EP-VT-110, "Visual Examination of the Reactor Vessel and Removable Internal Structures", Revision 0, dated December 5, 2007

9.10 LR-H2OC-PROGPLAN, "Water Chemistry Control Program Basis Document", Revision 1

9.11 CHA-PRI-STRATEGY "Primary water Chemistry Strategy, Revision 00801 12-7-2007 (In-process change to CH-101)

9.12 CH-PRI-SCHED "Primary System Analysis Schedule and Limits", Revision 02900, dated Jan 12, 2009

9.13 Pressurized Water Reactor Primary Water Chemistry Guidelines Rev 6, EPRI Report No1014986 December 2007.

9.14 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants"

9.15 ANSI N18.7-1976 "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants"

9.16 Updated Final Safety Analysis Report (UFSAR), Chapter 17, "Quality Assurance Program for Station Operation"

9.17 ND-QAP, "Quality Assurance Program", Revision 01100, dated August 13, 2008

9.18 IP-CAP-1, "Abnormal Condition Tracking Initiation or Notification (ACTION) Report", Revision 02800, dated Jan 14, 2008

9.19 A-1603.2, "Work Order Initiation", Revision 18, dated March 21, 2006

9.20 IP-OPS-4 , "Post-Maintenance Operability Testing", Revision 00100, dated August 13, 2008

9.21 NRC Letter dated May 19, 2004, Issuance of Renewed Facility Operating License No DPR-18 For R.E. Ginna Nuclear Power Plant

9.22 CNG-PR-101-1011, "Control of Station Specific Procedure Change Process " Revision 00300, dated November 4, 2008

9.23 IP-PRO-4, "Procedure Adherence Requirements", Revision 01500, dated 11-9 2007

9.24 CBG-PR-3.01-1000, "Records Management", Revision 00101, dated August 18, 2008

9.25 Conermann, J., Shogun, R. P., Junker, W., Wilson, I.L.W., and Spellward, P., Westinghouse Letter Report STD-MR-01-0012, Rev. 0, "Ginna Hot Cell Testing Report," Nov. 26, 2001

9.26 LR-PSPM-PROGPLAN, "Periodic Surveillance and Preventive Maintenance Program"

9.27 American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI, Rules for In-service Inspection of Nuclear Power Plant Components, 1995 Edition, 1996 Addenda, American Society of Mechanical Engineers, New York, NY (ISI program is currently being revised)

9.28 License Renewal Aging Management Program basis Document , "Thimble Tube Inspection Program", LR-TTI-PROGPLAN, Rev 2, Nov. 29, 2007

9.29 WCAP-15437, "Study of the Sensitivity of Acceptable Baffle-Barrel-Former Bolting Patterns to Fuel Type for Two-Loop Domestic Plants", September 2000

9.30 Ginna Station Fourth Interval ASME Section XI In-service Inspection Program, dated January 1, 2000

9.31 Ginna Station Aging Management Review Report, LRAM-RVI, "Reactor Vessel Internals"

9.32 WCAP-14831, "Baffle-Former Bolt Program for Westinghouse Owners Group Phase5; Operability Investigation ii", December 1997

9.33 WCAP-15029, "Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distribution Under Faulted Load Conditions", November 1998

9.34 NUREG 1786, Safety evaluation Report Related to the License Renewal of the R.E. Ginna Nuclear Power Plant, May 2004.

9.35 Ginna Nuclear Power Plant, License Renewal Project, Aging Management Review Report, LRAM-RVI, Revision 0, April 9, 2002.

9.36 Application for Renewed Operating License, R. E. Ginna Nuclear Power Plant, Section 2.3.1.3, Reactor Vessel Internals, August 2002.

9.37 Safety Evaluation Report Related to the License Renewal of R. E. Ginna Nuclear Power Plant (NUREG-1786), May 2004.

9.38 Materials Reliability Program: PWR Internals Inspection and Evaluation Guidelines (MRP-227), EPRI 1016596, Palo Alto, CA 2008 (Rev.0), Dec 2008.

9.39 Appendix D: Materials Guidelines: Implementation Protocol, in "Guidelines for the Management of Materials Issues," NEI 03-08, Nuclear Energy Institute, Washington, DC.

9.40 Materials Reliability Program: Framework and Strategies for Managing Aging Effects in Reactor Internals (MRP-134), EPRI, Palo Alto, CA: June 2005. 1008203.

9.41 Materials Reliability Program: Development of Material Constitutive Model for Irradiated Austenitic Stainless Steels (MRP-135), EPRI, Palo Alto, CA: 1011127.

9.42 Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values (MRP-175), EPRI, Palo Alto, CA: 1012081.

9.43 Materials Reliability Program: Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Design (MRP-191), EPRI, Palo Alto, CA: 2006. 1013234.

9.44 ASME Boiler and Pressure Vessel Code, Section XI, "Rules for In-service Inspection of Nuclear Plant Components," 2004 Edition.

9.45 LR-IWBD-PROGPLAN, Rev. 2, "ASME Section XI, Subsection IWB, IWC & IWD In-service Inspection Program," Ginna Station License Renewal Project, 5/2/07.

9.46 EP-VT-110, "Visual Examination of the Reactor Vessel and Removable Internal Structures," Ginna Station Program Document.

9.47 R. E. Ginna Nuclear Station Administrative Procedure, IP-IIT-8, "RCS Materials Degradation Management Program," Rev. 00100.

9.48 Materials Reliability Program: Pressurized Water Reactor Issue Management Tables (MRP-205), EPRI, Palo Alto, CA: 2006. 1014446.

9.49 Generic Aging Lessons Learned (GALL) Report, NUREG-1801, Revision 0, Volumes 1 & 2, U. S. Nuclear Regulatory Commission, July 2001.

9.50 Intentional Blank

9.51 RCS Materials Degradation Management Program, CNG-AM-4.01 Rev. 0100 dated, July 16 2007.

9.52 NUREG -1800, "Standard Review Plan for Review of License Renewal Application for Nuclear Power Plants," July 2001.

9.53 U.S. NRC Bulletin 88-09, "Thimble Tube Thinning in Westinghouse Reactors" March 28,1988.

9.54 NEI-03-08, Guideline for the Management of Materials Issues, Rev.1 April 2007.

9.55 RG&E Letter to NRC R.E Mecredy to Allen R. Johnson dated Ap. 8, 1993, Response to NRC Bulletin NO 88-09 Thimble Tube thinning in Westinghouse reactors" RGO-14088.

9.56 Materials Reliability Program: Inspection Standard for Reactor Internals (MRP-228). EPRI, Palo Alto, CA: 2009. 1016609.

9.57 ASME Boiler & Pressure Vessel Code, Section XI, Division 1, "Rules for In-service Inspection of Nuclear Power Plant Components," American Society of Mechanical Engineers, New York, NY, 2001 Edition, Plus 2003 Addenda, or later.

9.58 Evaluation of Remote Visual Examination Methods. EPRI, Palo Alto, CA: 2006. 1013537.

9.59 Letter to Reactor Internals Focus Group from MRP, Subject: Minutes of the Expert Panel Meetings on Expansion Criteria for Reactor Internals I&E Guidelines, MRP 2008-036(via email), June 12, 2008.

9.60 EP-3-S-0719, "Preparation of Aging Management Program Basis Document Guideline", Revision 0.

9.61 WCAP 17016-NP Rev 0 Draft "Handbook of Flaw acceptance Criteria for selected R.E. Ginna Reactor Vessel Internals MRP-227 Primary Components".

9.62 PWROG Document, WCAP XXYYY-NP Rev. (In-process), "Handbook of Flaw acceptance Criteria for selected Expansion Components (LATER)".

9.63 CENG Letter of May 10, 2008 "Fourth Ten Year Interval In-service Inspection Program, Submittal of Relief Request 18,19, 20 and 21."

9.64 WCAP -15036 Rev 1, Determination of Acceptable Baffle-Barrel Bolting For Two Loop Westinghouse Domestic Plants

9.65 WCAP – 7408, Robert E. Ginna Unit No.1, Internals Assurance Program

9.66 NRC Letter, "Acceptance for Referencing of Generic License renewal Program Topical report entitled "License renewal Evaluation: Aging Management for Reactor Internals" WCAP-14577, Revision 1, Oct. 2000" dated Feb 10, 2001

9.67 Ginna Station, Updated Final Safety Analysis Report (UFSAR)

9.68 WCAP 16881-P – "R.E. Ginna Reactor Vessel Internals Program Scheduling Evaluations". Nov. 2008.

9.69 Draft, Westinghouse Letter, LTR-RIDA-09-28, R.E. Ginna, Aging Management Inspection Evaluations for Guide Tube Split Pins

9.70 EP-3-S-0719, "Preparation of Aging Management Program Basis Document Guideline, Revision 0, dated February 28, 2007

ATTACHMENT A
Table 1
Summary Of Implementing Documents/ Enhancements/ Testing/ Initial Inspections/ Actions

1 Discussed in PBD Element No. and Title	2 AMR Activity Number	3 Implementing Document Number-Title	4 Document Revision #	5 Effective Step #(s)	6 Description of Activity / Enhancement	7 AIT Number
Documents Identified in this PBD - See AIT for Tracking of these Activities						
Section 4.4-Summary	N/A	EP-VT-110	00000	TBD	Revise EP-VT-110 for the 2011 RFO to include augmented inspections required by the LR-RVI-PROGPALN Section 5.3.3	NL-2009-000021-001
Section 5.2.2 Primary Water Chemistry Control Program	2	Pri-Sched (CH-120)	02900	Program	Basis Capture to reflect Pri-Sched Chemistry program is required to be in place by the RVI program. Meets intent of Pressurized Water Reactor Primary Water Chemistry Guidelines Rev. 6. 1014986 December 2007.	Complete
N/A	Action Item 2 &7 Section 7 LRAM-RVI	FATIGUE -PRO	N/A	N/A	Evaluate need for any additional FATIGUE pro monitoring that is required. i.e. cycle counting or tabulation of RX internals fatigue evaluations.	NL-2009-000020-001

1 Discussed in PBD Element No. and Title	2 AMR Activity Number	3 Implementing Document Number-Title	4 Document Revision #	5 Effective Step #(s)	6 Description of Activity / Enhancement	7 AIT Number
Inspections, Evaluations and/or Tests Identified in this PBD - See AIT for Tracking of these Activities						
Section 4.2.1 Control Rod Guide Tube Split pin Replacement Project	N/A	CR-2008-009906 CA-2008-003832 ODMI For Split Pins	0	N/A	Complete Westinghouse Letter LTR-RIDA-09-28. Aging Management Inspection Evaluations for Guide Tube Split Pins	NL-2009-000023-001 & 002.
5.6 Acceptance Criteria	N/A	Expansion Component Evaluation Criteria (PWROG)	0	N/A	Develop Expansion Component Evaluation Criteria with PWROG in order to disposition recordable indications.	NL-2009-000024-001
Summary of LR Actions / Other LRi Activities Identified in this PBD						
5.3.3.1.1 Lower Core Plate	N/A	Clarify Item 8.2.13 Of the EP-VT-110 checklist to include figure 19.	0000	8.2.13	Clarify checklist to include Fig-19. Implied in text discussion but not included in list of figures referenced	NL-2009-000020-001
4.2.2 Baffle Former Bolt Inspection Replacement Project	Applicant Action Item 10	Establish Acceptable Bolt Patterns for 2011 BFB Options.	0	N/A	Develop 60/80 year option for acceptable BFB replacement patterns.	NL-2009-000019-001
N/A	N/A	LRAM-RVI	0	N/A	Update AMR to be consistent with MRP-227 when approved.	NL-2009-000027-001
2.3 Ginna Reactor Internals AMP Background	N/A	LR-RVI-PROGPLAN	2	N/A	Submit Reactor Vessel Internals Program for review and approval.	AI-2003-011417-001

ATTACHMENT B

PBD - PEN&INK Non-Intent Change Form for Attachment A, Table 1

This process is only applicable to the information in Attachment A, Table 1, columns: "4- Document Revision #" and "5- Effective Step #(s)". See section 6.2.2, Records, Review and Approval.

Sequential Non-Impact Change #	Revised Document Number	Old Revision Number	New Revision Number	Old Implementing Step #	New Implementing Step #	LR Coordinator Review	
						Approve Yes/No	Signature/Date

ATTACHMENT C
Ginna Station Aging Management Review Summary Table LRAM-RVI

COMPONENT GROUP	INTENDED FUNCTION	MATERIAL	ENVIRONMENT	AGING EFFECTS REQUIRING MANAGEMENT[1]	PROGRAM/ACTIVITY
Lower Core Plate and Fuel Pins	Core Support Flow Distribution	Stainless Steel	Primary Water	Cracking due to IASCC Reduction in Fracture Toughness due to Irradiation Embrittlement	• ASME Section XI Inservice Inspection Program Examination Category B-N-3 • Reactor Vessel Internals Inspection Program
				Loss of Material due to Wear (Fuel Pins only)	• ASME Section XI Inservice Inspection Program Examination Category B-N-3
				Cracking due to SCC	• Water Chemistry Control Program
Lower Support Forging	Core Support Flow Distribution	Forged Stainless Steel	Primary Water	Cracking due to SCC	• Water Chemistry Control Program
Lower Support Columns	Core Support	Stainless Steel	Primary Water	Cracking due to IASCC Reduction in Fracture Toughness due to Irradiation Embrittlement	• ASME Section XI Inservice Inspection Program Examination Category B-N-3 • Reactor Vessel Internals Inspection Program
				Cracking due to SCC	• Water Chemistry Control Program

ATTACHMENT C
Ginna Station Aging Management Review Summary Table LRAM-RVI

COMPONENT GROUP	INTENDED FUNCTION	MATERIAL	ENVIRONMENT	AGING EFFECTS REQUIRING MANAGEMENT[1]	PROGRAM/ACTIVITY
Core Barrel and Flange	Core Support Flow Distribution	Stainless Steel	Primary Water	Cracking due to IASCC (Lower core barrel only) Reduction in Fracture Toughness due to Irradiation Embrittlement (Lower core barrel only)	• ASME Section XI Inservice Inspection Program Examination Category B-N-3 • Reactor Vessel Internals Inspection Program
				Loss of Material due to Wear (Flange only)	• ASME Section XI Inservice Inspection Program Examination Category B-N-3
				Cracking due to SCC	• Water Chemistry Control Program
Radial Keys and Clevis Inserts	Core Support	Stainless Steel (Radial Keys) Alloy 600 (Clevis Inserts)	Primary Water	Loss of Material due to Wear	• ASME Section XI Inservice Inspection Program Examination Category B-N-3
				Cracking due to SCC	• Water Chemistry Control Program

ATTACHMENT C
Ginna Station Aging Management Review Summary Table LRAM-RVI

COMPONENT GROUP	INTENDED FUNCTION	MATERIAL	ENVIRONMENT	AGING EFFECTS REQUIRING MANAGEMENT[1]	PROGRAM/ACTIVITY
Baffle and Former Assembly	Core Support Flow Distribution	Stainless Steel	Primary Water	Cracking due to IASCC Reduction in Fracture Toughness due to Irradiation Embrittlement	• ASME Section XI Inservice Inspection Program Examination Category B-N-3
				Cracking due to SCC	• Water Chemistry Control Program
Core Barrel Outlet Nozzle	Flow Distribution	Stainless Steel	Primary Water	Cracking due to SCC	• Water Chemistry Control Program
Secondary Core Support	Core Support Flow Distribution	Stainless Steel	Primary Water	Cracking due to SCC	• Water Chemistry Control Program
Diffuser Plates	Flow Distribution	Stainless Steel	Primary Water	Cracking due to SCC	• Water Chemistry Control Program

ATTACHMENT C
Ginna Station Aging Management Review Summary Table LRAM-RVI

COMPONENT GROUP	INTENDED FUNCTION	MATERIAL	ENVIRONMENT	AGING EFFECTS REQUIRING MANAGEMENT[1]	PROGRAM/ACTIVITY
Upper Support Plate Assembly	Guide and Support RCCAs	Stainless Steel	Primary Water	Cracking due to SCC	• Water Chemistry Control Program
Hold-down Spring	Hold-down Spring	Stainless Steel	Primary Water	Loss of Preload due to Stress Relaxation Loss of Material due to Wear	• ASME Section XI Inservice Inspection Program Examination Category B-N-3
				Cracking due to SCC	• Water Chemistry Control Program
Head/Vessel Alignment Pins	Core Support	Stainless Steel	Primary Water	Cracking due to SCC	• Water Chemistry Control Program
Thermal Shield and Neutron Panels (Ginna Neutron Panels - N/A)	Shield Vessel	Stainless Steel	Primary Water	Cracking due to IASCC Reduction in Fracture Toughness due to Irradiation Embrittlement	• ASME Section XI Inservice Inspection Program Examination Category B-N-3 • Reactor Vessel Internals Inspection Program
				Cracking due to SCC	• Water Chemistry Control Program

ATTACHMENT C
Ginna Station Aging Management Review Summary Table LRAM-RVI

COMPONENT GROUP	INTENDED FUNCTION	MATERIAL	ENVIRONMENT	AGING EFFECTS REQUIRING MANAGEMENT[1]	PROGRAM/ACTIVITY
BMI Columns and Flux Thimble Tubes	Guide and Support Instrumentation	Stainless Steel	Primary Water	Cracking due to SCC	• Water Chemistry Control Program
				Loss of Material due to Wear (Thimbles Only)	• Thimble Tube Inspection Program
Head Cooling Spray Nozzles	Flow Distribution	Stainless Steel	Primary Water	Cracking due to SCC	• Water Chemistry Control Program
Upper Instrumentation Column, Conduit and Supports	Guide and Support Thermocouples	Stainless Steel	Primary Water	Cracking due to SCC	• Water Chemistry Control Program
Bolting (Upper Support Column, Guide Tube, Clevis Insert)	Core Support	Stainless Steel Alloy X-750 (Clevis Insert Bolts)	Primary Water	Loss of Mechanical Closure Integrity due to Stress Relaxation	• ASME Section XI In-service Inspection Program Examination Category B-N-3
				Cracking due to SCC	• Water Chemistry Control Program

ATTACHMENT C
Ginna Station Aging Management Review Summary Table LRAM-RVI

COMPONENT GROUP	INTENDED FUNCTION	MATERIAL	ENVIRONMENT	AGING EFFECTS REQUIRING MANAGEMENT[1]	PROGRAM/ACTIVITY
Bolting (Lower Support Column, Baffle/Former, Barrel/Former)	Core Support	Stainless Steel	Primary Water	Cracking due to SCC, IASCC and Loss of Mechanical Closure Integrity due to IASCC, Reduction in Fracture Toughness, Irradiation Creep and Stress Relaxation	• ASME Section XI Inservice Inspection Program Examination Category B-N-3 • Reactor Vessel Internals Inspection Program • Water Chemistry Control Program

ATTACHMENT D

Table 1.1 (MRP-227) Westinghouse Plants Primary Components

Item	Applicability	Effect (Mechanism)	Expansion Link	Examination Method/Frequency	Examination Coverage
Control Rod Guide Tube Assembly- Guide plates (cards)	Fig. 3	Loss of Material (Wear)	None	Visual (VT-3) examination no later than 2 refueling outages from the beginning of the license renewal period, and no earlier than two refueling outages prior to the start of the license renewal period. Subsequent examinations are required on a ten-year interval.	20% examination of the number of CRGT assemblies, with all guide cards within each selected CRGT assembly examined.
Control Rod Guide Tube Assembly- Lower flange welds All	Fig. 3	Cracking (SCC, Fatigue)	Bottom-mounted instrumentation (BMI) column bodies, Lower support column bodies (cast)	Enhanced visual (EVT-1) examination to determine the presence of crack-like surface flaws in flange welds no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	100% of outer (accessible) CRGT lower flange weld surfaces and adjacent base metal.
Core Barrel Assembly- Upper core barrel flange weld	Fig. 4	Cracking (SCC)	Remaining core barrel welds, Lower support column bodies (non cast)	Periodic enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal.

ATTACHMENT D

Table 1.1 (MRP-227) Westinghouse Plants Primary Components

Item	Applicability	Effect (Mechanism)	Expansion Link	Examination Method/Frequency	Examination Coverage
Baffle-Former Assembly Baffle-edge bolts	Fig. 7	Cracking (IASCC, Fatigue) that results in lost or broken locking devices, failed or missing bolts or protrusion of bolt heads	None	Visual (VT-3 examination, with baseline examination between 20 and 40 EFPY and subsequent examinations on a ten-year interval.	Bolts and locking devices on high fluence seams. 100% of components accessible from core side.
Baffle-Former Assembly Baffle-former bolts.	Fig. 5	Cracking (IASCC, Fatigue)	Lower support column bolts, Barrel-Former bolts.	Baseline volumetric (UT) examination between 25 and 35 EFPY, with subsequent examination after 10 to 15 additional EFPY to confirm stability of bolting pattern. Re-examination for high-leakage core designs requires continuing examinations on a ten-year interval.	100% of accessible bolts or as supported by plant-specific justification. Heads accessible from the core side. UT accessibility may be affected by complexity of head and locking device designs.

ATTACHMENT D

Table 1.1 (MRP-227) Westinghouse Plants Primary Components

Item	Applicability	Effect (Mechanism)	Expansion Link	Examination Method/Frequency	Examination Coverage
Baffle-Former Assembly Assembly	Fig. 6	Distortion (Void Swelling), or Cracking (IASCC) that results in: •abnormal interaction with fuel assemblies •Gaps along high fluence baffle joint •Vertical displacement of baffle plates near high fluence joint •Broken or damaged edge bolt locking systems along high fluence baffle joint.	None	Visual (VT-3) examination to check for evidence of distortion, with baseline examination between 20 and 40 EFPY and subsequent examinations on a ten-year interval.	Core side surface as indicated.

ATTACHMENT D

Table 1.1 (MRP-227) Westinghouse Plants Primary Components

Item	Applicability	Effect (Mechanism)	Expansion Link	Examination Method/Frequency	Examination Coverage
Alignment and Interfacing Components Internals hold down spring	N/A Ginna Only plants with 304 stainless steel hold down springs Ginna is 410 SS.	Distortion (Loss of Load) Note: This mechanism was not strictly identified in the original list of age related degradation mechanisms [7].	None	Direct measurement of spring height within three cycles of the beginning of the license renewal period. If the first set of measurements is not sufficient to determine life, spring height measurements must be taken during the next two outages, in order to extrapolate the expected spring height to 60 years.	Measurements should be taken at several points around the circumference of the spring, with a statistically adequate number of measurements at each point to minimize uncertainty. Replacement of 304 springs by 403 springs is required when the spring stiffness is determined to relax beyond design tolerance.
Thermal Shield Assembly Thermal shield flexures	Fig. 9 & 10	Cracking (Fatigue) or Loss of Material (Wear) that results in thermal shield flexures excessive wear, fracture, or complete separation	None	Visual (VT-3) no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	100% of thermal shield flexures.

ATTACHMENT D

TABLE 1.2 - Expansion Components

Item	Applicability	Effect (Mechanism)	Expansion Link	Examination Method/Frequency	Examination Coverage
Core Barrel Assembly Barrel-former bolts	Fig . 4	Cracking (IASCC, Fatigue)	Baffle-former bolts	Volumetric (UT) examination, with initial and subsequent examinations dependent on results of baffle-former bolt examinations.	100% of accessible bolts. Accessibility may be limited by presence of thermal shields.
Lower Support Assembly Lower support column bolts	Fig. 1	Cracking (IASCC, Fatigue)	Baffle-former bolts	Volumetric (UT) examination, with initial and subsequent examinations dependent on results of baffle-former bolt examinations.	100% of accessible bolts or as supported by plant-specific justification.
Core Barrel Assembly Core barrel flange, core barrel outlet nozzles, Lower core barrel flange weld	Fig. 4	Cracking (IASCC, Fatigue)	Upper core barrel flange weld	Enhanced visual (EVT-1) examination, with initial examination and re-examination frequency dependent on the examination results for upper core barrel flange.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal.
Lower Support Assembly Lower support column bodies (non cast)	Ginna= A276 type 304 Fig. 1	Cracking (IASCC)	Upper core barrel flange weld	Enhanced visual (EVT-1) examination with initial examination and re-examination frequency dependent upon the examination results for upper core barrel flange weld.	100% of accessible support columns.
Lower Support Assembly Lower Support Column bodies (cast)	N/A Ginna	Cracking (IASCC) Including the detection of fractured support columns	Control rod guide tube (CRGT) lower flanges	Enhanced visual (EVT-1) examination.	100% of accessible support columns.
Bottom Mounted Instrumentation System Bottom-mounted instrumentation (BMI) column bodies	Fig .1	Cracking (Fatigue) including the detection of completely fractured column bodies	Control rod guide tube (CRGT) lower flanges	Visual (VT-3) examination of BMI column bodies as indicated by difficulty of insertion/withdrawal of flux thimbles. Flux thimble insertion/withdrawal to be monitored at each inspection interval.	100% of BMI column bodies for which difficulty is detected during flux thimble insertion/withdrawal.

ATTACHMENT D
TABLE 1.3 - Existing Program

Item	Applicability	Effect (Mechanism)	Expansion Link	Examination Method/Frequency	Examination Coverage
Core Barrel Assembly Core barrel flange	Figure 4	Loss of material (Wear)	ASME Code Section XI	Visual (VT-3) examination to determine general condition for excessive wear.	All accessible surfaces at specified frequency.
Upper Internals Assembly Upper support ring or skirt	Figure 1	Cracking (SCC, Fatigue)	ASME Code Section XI	Visual (VT-3) examination	All accessible surfaces at specified frequency.
Lower Internals Assembly Lower core plate XL lower core plate (XL= 14ft. Core)	N/A Ginna is ~12' Core	Cracking (SCC, Fatigue)	ASME Code Section XI	Visual (VT-3) examination of the lower core plates to detect evidence of distortion and/or loss of bolt integrity.	All accessible surfaces at specified frequency.
Lower Internals Assembly Lower core plate XL lower core plate (XL= 14ft. Core)	N/A Ginna is ~12' Core	Loss of material (Wear)	ASME Code Section XI	Visual (VT-3) examination	All accessible surfaces at specified frequency.
Bottom Mounted Instrumentation System Flux thimble tubes	Fig 1	Loss of material (Wear)	NUREG-1801 Rev. 1	Surface (ET) examination	Eddy current surface examination as defined in plant response to IEB 88-09.
Alignment and Interfacing Components Clevis insert bolts	Fig. 12	Loss of material (Wear)	ASME Code Section XI	Visual (VT-3) examination	All accessible surfaces at specified frequency.
Alignment and Interfacing Components Upper core plate alignment pins	Fig. 11	Loss of material (Wear)	ASME Code Section XI	Visual (VT-3) examination	All accessible surfaces at specified frequency.

ATTACHMENT D
TABLE 1.4 - Acceptance Criteria and Expansion Criteria

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Links(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Control Rod Guide Tube Assembly Guide plates (cards)	Fig 2 & 3.	Visual (VT-3) Examination The specific relevant condition is wear that could lead to loss of control rod alignment and impede control assembly insertion.	None	N/A	N/A

Note 1: The examination acceptance criterion for visual examination is the absence of the specified relevance condition(s).

ATTACHMENT D
TABLE 1.4 - Acceptance Criteria and Expansion Criteria

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Links(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Control Rod Guide Tube Assembly Lower Flange welds	Fig .3	Enhanced visual (EVT-1) examination The specific relevant condition is a detectable crack-like surface indication.	a. Bottom-mounted instrumentation (BMI) column bodies b. Lower support column bodies (CAST)	a. Confirmation of surface breaking indications in two or more CRGT lower flange welds, combined with flux thimble insertion/withdrawal difficulty, shall require visual (VT-3) examination of BMI column bodies by the completion of the next refueling outage. b. Confirmation of surface breaking indications in two or more CRGT lower flange welds shall require EVT-1 examination of cast lower support column bodies within three fuel cycles following the initial observation.	a. For BMI column bodies, the specific relevant condition for the VT-3 examination is completely fractured column bodies. b. For cast lower support column bodies, the specific relevant condition is a detectable crack-like surface indication.

ATTACHMENT D
TABLE 1.4 - Acceptance Criteria and Expansion Criteria

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Links(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Barrel Assembly Upper core barrel flange weld	Fig. 4	Periodic enhanced visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	a. Remaining core barrel welds b. lower support column bodies (non-cast)	The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the upper core barrel flange weld shall require that the EVT-1 examination, and any supplementary UT examination, be expanded to include the core barrel-to-support plate weld by the completion of the next refueling outage. If extensive confirmed indications in the core barrel-to-support plate weld are detected, further expansion of the EVT-1 examination shall include the remaining core barrel assembly welds. b. If extensive cracking in the remaining core barrel welds is detected, EVT-1 examination shall be expanded to include the upper six inches of the accessible surfaces of the non cast lower support column bodies within three fuel cycles following the initial observation.	a and b. the specific relevant condition is a detectable crack-like surface indication.

ATTACHMENT D
TABLE 1.4 - Acceptance Criteria and Expansion Criteria

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Links(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Baffle-Former Assembly Baffle-edge bolt	Fig 5& 6	Visual (VT-3) examination. The specific relevant conditions are missing or broken locking devices, failed or missing bolts, and protrusion of bolt heads.	None	N/A	N/A

ATTACHMENT D
TABLE 1.4 - Acceptance Criteria and Expansion Criteria

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Links(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Baffle-Former Assembly Baffle-former bolts	Fig 5&6	Volumetric (UT) examination. The examination acceptance criteria for the UT of the baffle-former bolts shall be established as part of the examination technical justification.	a. Lower support column bolts b. Barrel-former bolts	a. Confirmation that more than 5% of the baffle-former bolts actually examined on the four baffle plates at the largest distance from the core (presumed to be the lowest dose locations) contain unacceptable indications shall require UT examination of the lower support column bolts within the next three fuel cycles. b. confirmation that more than 5% of the lower support column bolts actually examined contain unacceptable indications shall require UT examination of the barrel-former bolts.	a and b. The examination acceptance criteria for the UT of the lower support column bolts and the barrel-former bolts shall be established as part of the examination technical justification.

ATTACHMENT D
TABLE 1.4 - Acceptance Criteria and Expansion Criteria

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Links(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Baffle-Former Assembly Assembly	Fig. 5 & 6	Visual (VT-3) examination. The specific relevant conditions are evidence of abnormal interaction with fuel assemblies, gaps along high fluence shroud plate joints, vertical displacement of shroud plates near high fluence joints, and broken or damaged edge bolt locking systems along high fluence baffle plate joints.	None	N/A	N/A

ATTACHMENT D
TABLE 1.4 - Acceptance Criteria and Expansion Criteria

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Links(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Alignment and Interfacing Components Internals hold down spring	All plants with 304 stainless steel hold down springs **NOTE:** N/A - Ginna Spring is 410 SS	Direct physical measurement or spring height. The examination acceptance criterion for this measurement is that the remaining compressible height of the spring shall provide hold-down forces within the plant-specific design tolerance. N/A Ginna	None	N/A	N/A
Thermal Shield Assembly Thermal shield flexures	Fig. 9	Visual (VT-3) examination. The specific relevant conditions for thermal shield flexures are excessive wear, fracture or complete separation.	None	N/A	N/A

ATTACHMENT E
Scope
LR-RVI-PROGPLAN

Subcomponent	Passive Function	Aging Management Reference
LOWER CORE PLATE AND FUEL PINS	CORE SUPPORT FLOW DISTRIBUTION	Table 3.2-1 Line Number (8) Table 3.2-1 Line Number (31) Table 3.2-1 Line Number (33) Table 3.2-2 Line Number (7) Table 3.2-2 Line Number (10) These apply to both passive functions.
LOWER SUPPORT FORGING	CORE SUPPORT FLOW DISTRIBUTION	Table 3.2-1 Line Number (8) Table 3.2-1 Line Number (31) Table 3.2-1 Line Number (33) Table 3.2-2 Line Number (7) Table 3.2-2 Line Number (8) Table 3.2-2 Line Number (9) These apply to both passive functions.
LOWER SUPPORT COLUMNS	CORE SUPPORT	Table 3.2-1 Line Number (8) Table 3.2-1 Line Number (31) Table 3.2-1 Line Number (33) Table 3.2-2 Line Number (7)
CORE BARREL AND FLANGE	CORE SUPPORT FLOW DISTRIBUTION	Table 3.2-1 Line Number (8) Table 3.2-1 Line Number (31) Table 3.2-1 Line Number (33) Table 3.2-2 Line Number (7) Table 3.2-2 Line Number (10) These apply to both passive

ATTACHMENT E
Scope
LR-RVI-PROGPLAN

Subcomponent	Passive Function	Aging Management Reference
		functions.
RADIAL KEYS AND CLEVIS INSERTS	CORE SUPPORT	Table 3.2-1 Line Number (8) Table 3.2-1 Line Number (28) Table 3.2-1 Line Number (33) Table 3.2-2 Line Number (7) Table 3.2-2 Line Number (9)
BAFFLE AND FORMER ASSEMBLY	CORE SUPPORT FLOW DISTRIBUTION	Table 3.2-1 Line Number (8) Table 3.2-1 Line Number (31) Table 3.2-1 Line Number (33) Table 3.2-2 Line Number (7) These apply to both passive functions.
CORE BARREL OUTLET NOZZLE	FLOW DISTRIBUTION	Table 3.2-1 Line Number (8) Table 3.2-1 Line Number (31) Table 3.2-1 Line Number (33) Table 3.2-2 Line Number (7) Table 3.2-2 Line Number (8) Table 3.2-2 Line Number (9)

ATTACHMENT E – Scope
LR-RVI-PROGPLAN

Subcomponent	Passive Function	Aging Management Reference
SECONDARY CORE SUPPORT	CORE SUPPORT FLOW DISTRIBUTION	Table 3.2-2 Line Number (11) This applies to both passive functions.
DIFFUSER PLATES	FLOW DISTRIBUTION	Table 3.2-2 Line Number (11) This applies to both passive functions.
UPPER SUPPORT PLATE ASSEMBLY	GUIDE AND SUPPORT RCCA'S	Table 3.2-1 Line Number (8) Table 3.2-1 Line Number (33) Table 3.2-2 Line Number (7) Table 3.2-2 Line Number (9)
UPPER CORE PLATE AND FUEL ALIGNMENT PINS	CORE SUPPORT FLOW DISTRIBUTION	Table 3.2-1 Line Number (8) Table 3.2-1 Line Number (33) Table 3.2-2 Line Number (7) Table 3.2-2 Line Number (9) Table 3.2-2 Line Number (10) These apply to both passive functions.
UPPER SUPPORT COLUMNS	GUIDE AND SUPPORT RCCA'S	Table 3.2-1 Line Number (8) Table 3.2-1 Line Number (33) Table 3.2-2 Line Number (7) Table 3.2-2 Line Number (9)
RCCA GUIDE TUBES AND FLOW DOWNCOMERS	GUIDE AND SUPPORT RCCAS	Table 3.2-1 Line Number (8) Table 3.2-1 Line Number (33) Table 3.2-2 Line Number (7) Table 3.2-2 Line Number (9)

ATTACHMENT E – Scope
LR-RVI-PROGPLAN

Subcomponent	Passive Function	Aging Management Reference
GUIDE TUBE SUPPORT PINS	GUIDE AND SUPPORT RCCAS	Table 3.2-1 Line Number (8) Table 3.2-2 Line Number (7) Table 3.2-2 Line Number (10) Table 3.2-2 Line Number (11)
UPPER CORE PLATE ALIGNMENT PINS	GUIDE AND SUPPORT RCCAS	Table 3.2-1 Line Number (8) Table 3.2-1 Line Number (28) Table 3.2-1 Line Number (33) Table 3.2-2 Line Number (7) Table 3.2-2 Line Number (9)
HOLD-DOWN SPRING	CORE SUPPORT	Table 3.2-1 Line Number (8) Table 3.2-1 Line Number (30) Table 3.2-1 Line Number (33) Table 3.2-2 Line Number (7) Table 3.2-2 Line Number (9) Table 3.2-2 Line Number (10) Table 3.2-2 Line Number (12)
HEAD/VESSEL ALIGNMENT PINS	CORE SUPPORT	Table 3.2-2 Line Number (11)

ATTACHMENT E – Scope
LR-RVI-PROGPLAN

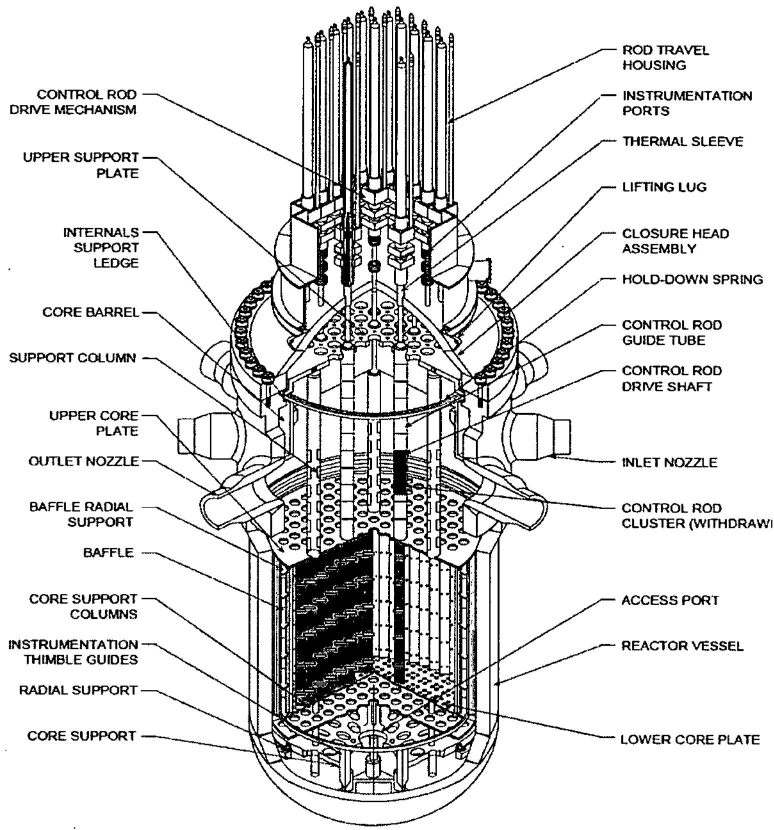
BMI COLUMNS AND FLUX THIMBLES	GUIDE AND SUPPORT INSTRUMENTATION	Table 3.2-1 Line Number (28) Table 3.2-2 Line Number (11)
HEAD COOLING SPRAY NOZZLES	FLOW DISTRIBUTION	Table 3.2-2 Line Number (11)
UPPER INSTRUMENTATION COLUMN, CONDUIT AND SUPPORTS	GUIDE AND SUPPORT THERMOCOUPLES	Table 3.2-2 Line Number (11)
BOLTING (UPPER SUPPORT COLUMN, GUIDE TUBE, CLEVIS INSERT)	CORE SUPPORT	Table 3.2-1 Line Number (8) Table 3.2-1 Line Number (30) Table 3.2-1 Line Number (31) Table 3.2-1 Line Number (33) Table 3.2-1 Line Number (36) Table 3.2-2 Line Number (7) Table 3.2-2 Line Number (9) Table 3.2-2 Line Number (12)

ATTACHMENT E – Scope LR-RVI-PROGPLAN		
Subcomponent	**Passive Function**	**Aging Management Reference**
BOLTING (LOWER SUPPORT COLUMN, BAFFLE/FORMER, BARREL/FORMER)	CORE SUPPORT	Table 3.2-1 Line Number (5) Table 3.2-1 Line Number (8) Table 3.2-1 Line Number (12) Table 3.2-1 Line Number (13) Table 3.2-1 Line Number (31) Table 3.2-1 Line Number (33) Table 3.2-1 Line Number (36) Table 3.2-2 Line Number (7) Table 3.2-2 Line Number (12)

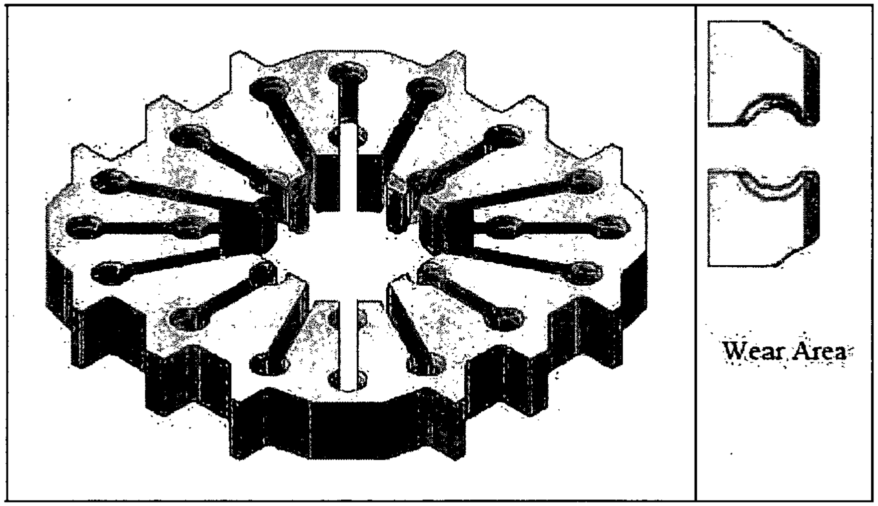
1 This information was extracted from R.E. Ginna Nuclear Power Plant Application for Renewed Operating License Technical and Administrative Information. Aging Management Reference Links are active in the Original Application.

ATTACHMENT F

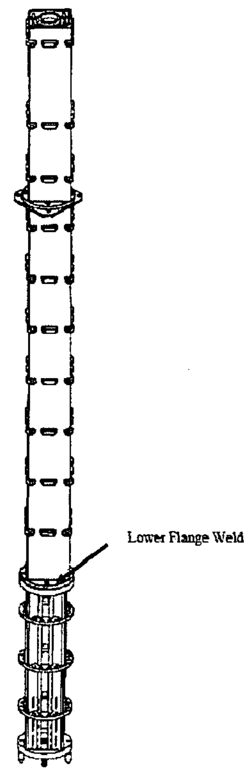
Fig. 1 Typical Westinghouse Internals



ATTACHMENT G
Figure 2 Typical Guide Card

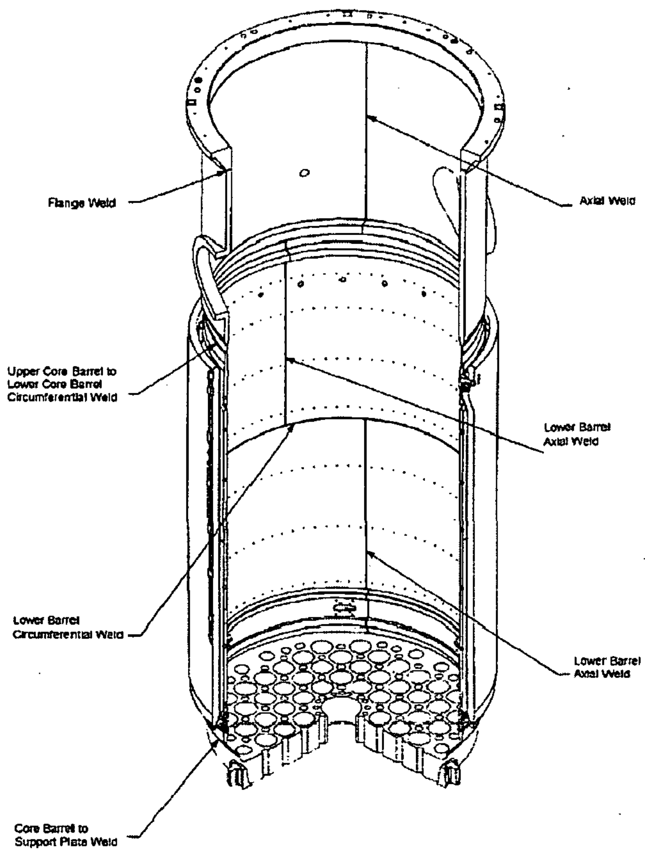


ATTACHMENT H
Figure 3 Typical Control Rod Guide Tube Assembly



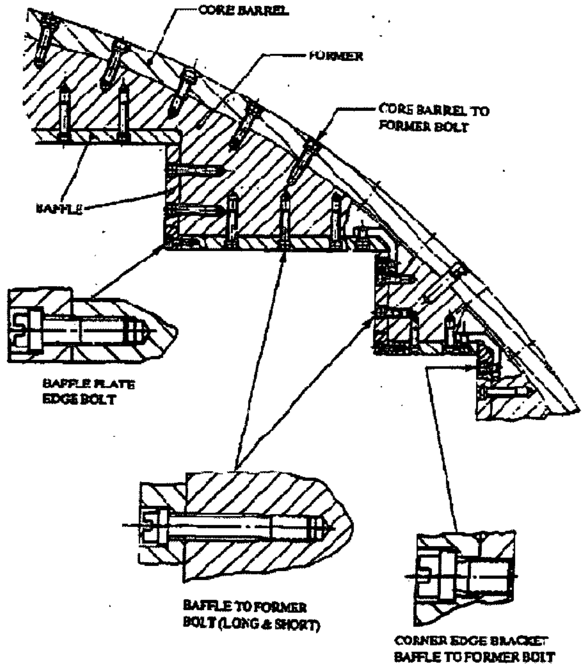
ATTACHMENT I

Figure 4 Typical Core Barrel

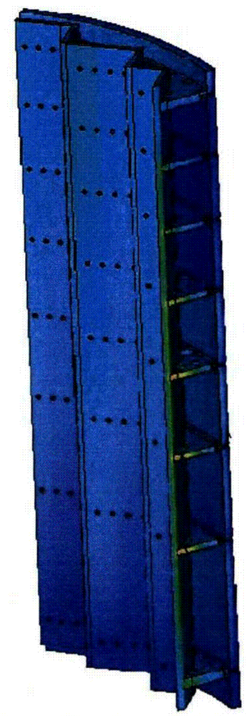


ATTACHMENT J

Figure 5 Typical Baffle-Former Bolt Locations

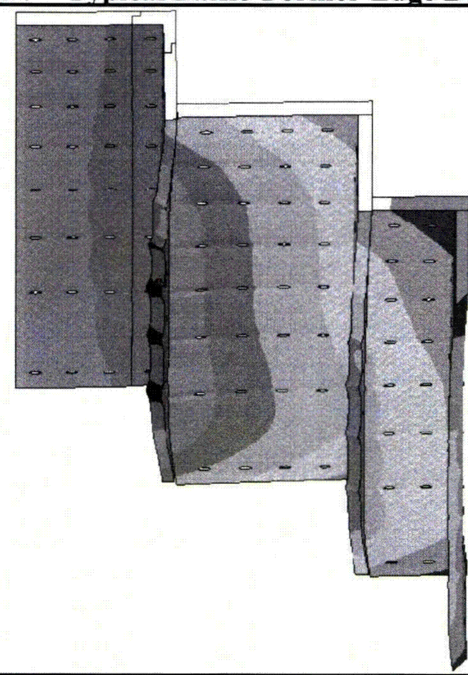


ATTACHMENT K
Figure 6 Typical Section of Baffle Former Assembly



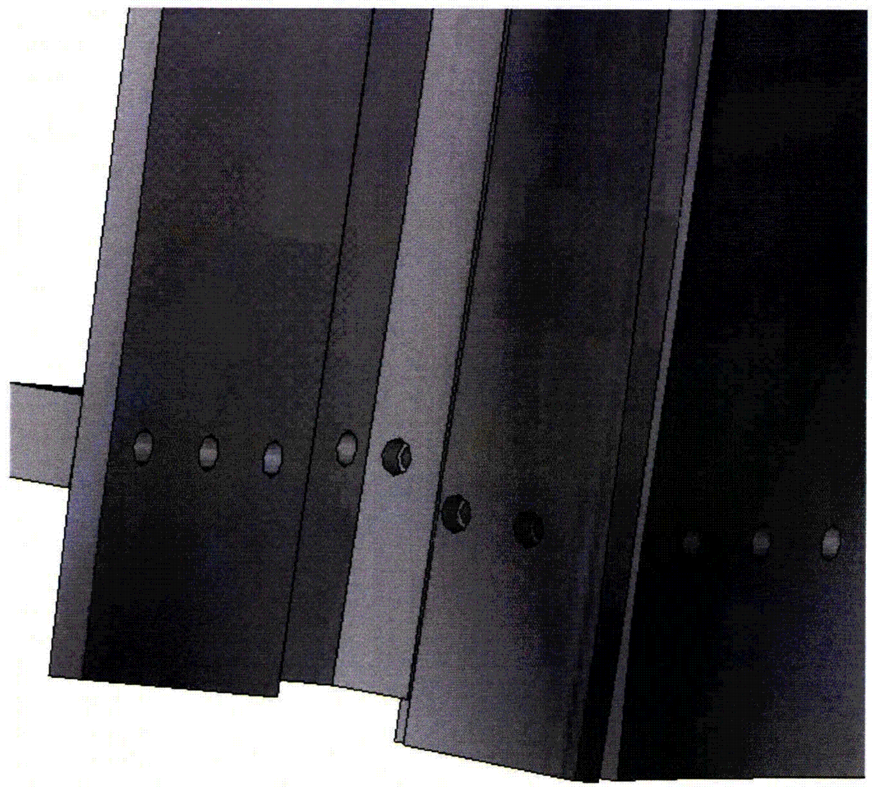
ATTACHMENT L

Figure 7 – Typical Baffle-Former Edge Bolts



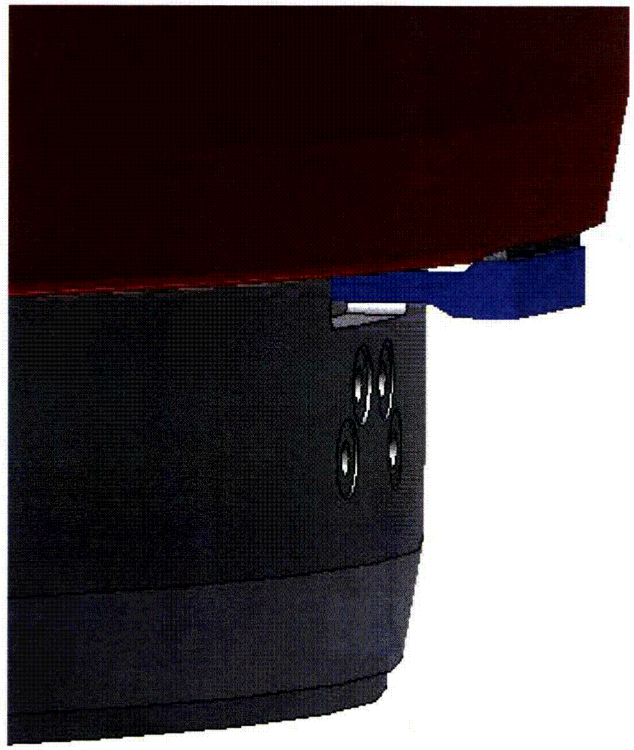
ATTACHMENT M

Figure 8 Typical Lower Baffle-Former Plates

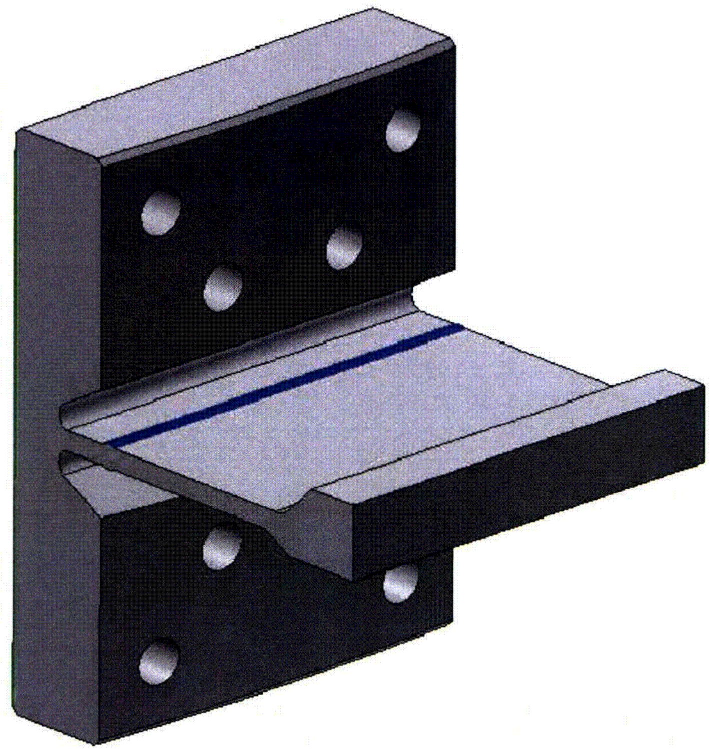


ATTACHMENT N

Figure 9 – Typical Thermal Shield Flexure



ATTACHMENT O
Figure 10 Typical Detail Thermal Shields Flexure



ATTACHMENT P
Figure 11 Typical Reactor Upper Internals

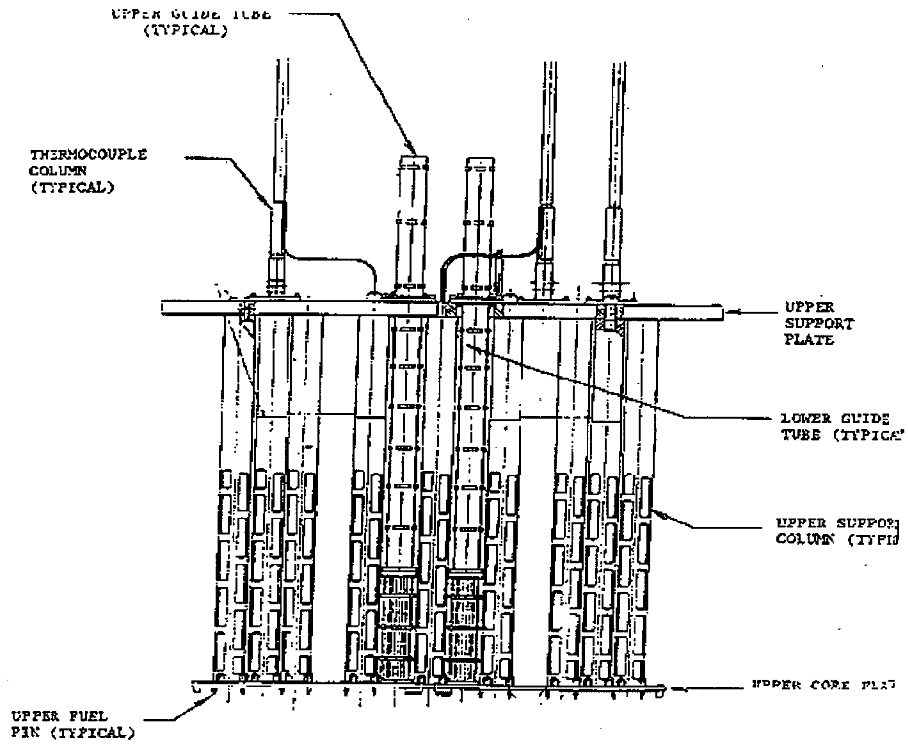


Figure 3. Reactor Upper Internals Assembly (Elevation)

figure3.bmp

ATTACHMENT Q
Figure 12 Typical Lower Radial Supports