ATTACHMENT TO

2CAN071603

PWR INTERNALS AGING MANAGEMENT PROGRAM PLAN FOR ARKANSAS NUCLEAR ONE, UNIT 2

Report No. 1500227.401
Revision 0
Project No. 1500227
July 2016

PWR Internals Aging Management Program Plan for Arkansas Nuclear One, Unit 2

Prepared for:

Entergy Nuclear Operations
Contract Order No: 10463207

Prepared by:

Structural Integrity Associates, Inc.
San Jose, California

Prepared by: Stephen M. Parker, P.E. — Date: 7/5/2016

Reviewed by: Timothy J. Griesbach — Date: 7/5/2016

Approved by: Christopher S. Lohse, P.E. — Date: 7/5/2016

Structural Integrity Associates, Inc.®

REVISION CONTROL SHEET

Document Number: **1500227.401**

Title: PWR Internals Aging Management Program Plan for Arkansas Nuclear One, Unit 2

Client: Entergy Nuclear Operations

SI Project Number: 1500227 Quality Program: ☒ Nuclear ☐ Commercial

Section	Pages	Revision	Date	Comments
1.0	1-1 – 1-16	0	7/5/16	Original Issue
2.0	2-1 – 2-24			
3.0	3-1 – 3-17			
4.0	4-1 – 4-5			
5.0	5-1 – 5-29			
6.0	6-1 – 6-4			

Table of Contents

List of Tables

List of Figures

LIST OF ACRONYMS

AMP	Aging Management Program
AMR	Aging Management Review
ANO-2	Arkansas Nuclear One, Unit 2
ASME	American Society of Mechanical Engineers
B&PV	Boiler and Pressure Vessel
B&W	Babcock & Wilcox
CASS	Cast austenitic stainless steel
CE	Combustion Engineering
CEA	Control Element Assembly
CEOG	Combustion Engineering Owners Group
CFR	Code of Federal Regulations
CLB	Current licensing basis
CSB	Core Support Barrel
EFPY	Effective full power years
EPRI	Electric Power Research Institute
ET	Eddy Current Testing
EVT	Enhanced visual testing (visual NDE method indicated as EVT-1)
FMECA	Failure modes, effects, and criticality analysis
GALL	Generic Aging Lessons Learned
I&E	Inspection and Evaluation
IASCC	Irradiation Assisted Stress Corrosion Cracking
ICI	In-Core Instrumentation
IE	Irradiation Embrittlement
IGSCC	Intergranular Stress Corrosion Cracking
INPO	Institute of Nuclear Power Operations
IP	Issue Programs
ISI	Inservice Inspection
ISR	Irradiation-Enhanced Stress Relaxation
LRA	License Renewal Application
LRAAI	License Renewal Application Action Item
MRP	Materials Reliability Program
MSC	Materials Subcommittee
NDE	Nondestructive Examination
NGF	Next Generation Fuel
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System

OE	Operating Experience
PDI	Performance Demonstration Initiative
PH	Precipitation-Hardenable
PWR	Pressurized Water Reactor
PWROG	Pressurized Water Reactor Owners Group
PWSCC	Primary Water Stress Corrosion Cracking
RCS	Reactor Coolant System
RIS	Regulatory Issue Summary
RV	Reactor Vessel
RVI	Reactor Vessel Internals
SCC	Stress Corrosion Cracking
SE	Safety Evaluation
SER	Safety Evaluation Report
SRP	Standard Review Plan
SS	Stainless Steel
TLAA	Time-limited Aging Analysis
TE	Thermal Embrittlement
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
UT	Ultrasonic Testing
UGS	Upper Guide Structure
VT	Visual Testing

1.0 INTRODUCTION

1.1 Objective

The purpose of this document is to describe the potential aging concerns in the reactor vessel internals (RVI) at Arkansas Nuclear One, Unit 2 (ANO-2). This document also describes the mandatory and recommended guidance for managing potential aging concerns at ANO-2 through the period of extended operation, which begins at midnight on July 17, 2018. This Aging Management Program (AMP) document satisfies the license renewal commitment as contained in the ANO-2 license renewal application (LRA) [1]. This program coordinates with the ANO-2 inservice inspection (ISI) program [2] and supplements that program with augmented examinations for managing the potential aging effects of the RVI. This program plan establishes appropriate monitoring and inspections to maintain the reactor vessel internals functionality; the strategy is to assure nuclear safety and plant reliability. This document provides assurance that operations at ANO-2 will continue to be conducted in accordance with the current licensing bases (CLB) for the RVI, and it will provide the technical basis for managing the time-limited aging concerns for the duration of the plant by fulfilling the license renewal commitments. This document identifies the internals components that must be considered for aging management review and identifies the augmented inspection plan for the ANO-2 reactor vessel internals. The program plan supports the NEI 03-08 Guideline for the Management of Materials Issues [3], the EPRI Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A [4]), and the Applicant/Licensee Action Items in the NRC Safety Evaluation (SE) [5].

The main objectives of the ANO-2 RVI AMP are:

- To demonstrate that the effects of aging on the RVI will be adequately managed for the period of extended operation in accordance with 10 CFR Part 54 [6].
- To summarize the role of existing ANO-2 AMPs related to aging management of RV internals.

- To define and implement the industry-defined (EPRI/MRP and PWROG) requirements and guidance for managing aging of RV internals.
- To provide inspection plans for the ANO-2 RV internals.

1.2 ANO-2 Reactor Vessel Internals Inspection Program Commitment

In order to meet license renewal Commitment 19 [1, 18], ANO-2 will submit this aging management program plan. This plan also addresses license renewal Commitment 15. MRP-227-A [4] and the response to A/LAI 7 of the NRC SE address the CASS internals components. Therefore, the CASS internals components are managed under the MRP-227-A AMP. The license renewal commitments listed below define the content for the program that ANO-2 has committed to implement for the RVI Components:

> *Commitment 15: The Reactor Vessel Internals CASS Program will manage aging effects of cast austenitic stainless steel reactor vessel internals components. This program will supplement the reactor vessel internals inspections required by the ASME Section XI Inservice Inspection Program. The program will manage cracking, reduction of fracture toughness, and dimensional changes using inspections of applicable components which will be determined based on the neutron fluence and thermal embrittlement susceptibility of the component. A description of the Reactor Vessel Internals CASS Program, which includes the inspection plan, will be submitted to the NRC for review and approval.*
>
> *Commitment 19: The Reactor Vessel Internals Stainless Steel Plates, Forgings, Welds, and Bolting Program will manage aging effects of reactor vessel internals plates, forgings, welds, and bolting. This program will supplement the reactor vessel internals inspections required by the ASME Section XI inservice inspection program. This program will manage the effects of crack initiation and growth due to stress corrosion cracking or irradiation assisted stress corrosion cracking, loss of fracture toughness due to neutron irradiation embrittlement, and distortion due to void swelling. This program will provide visual inspections and non-destructive examinations of reactor vessel internals.*
>
> *In the development of this program, Entergy will support reactor vessel internals aging effects research through EPRI, the Materials Reliability Program, and other applicable*

industry efforts to better characterize the internals aging effects and to provide material property data to generate acceptance standards for inspections. Appropriate examination techniques will be selected based on the results of these industry efforts.

A description of this program, which includes the inspection plan, will be submitted to the NRC for review and approval.

Augmented inspections, based on required program enhancements resulting from the industry programs referred to in these license renewal commitments, will become part of the ANO-2 ASME B&PV Code, Section XI ISI program [2]. Corrective actions for augmented inspections will either be developed using repair and replacement procedures equivalent to those required in ASME B&PV Code, Section XI [9], or more rigorous procedures will be determined by ANO-2 independently or in cooperation with the industry. ANO-2 is currently committed to the 2001 Edition through 2003 Addenda of the ASME B&PV Code [9], and the development of this AMP is based on that Edition of the Code and MRP-227-A [4]. However, later Editions and Addenda of the Code or additional Code Cases or Safety Evaluation Reports (SERs) may be incorporated or invoked as necessary or required by 10 CFR 50.55a. The use of MRP-227-A, as approved by the NRC, is consistent with current industry practice.

The Aging Management Program has been established so that the aging effects of the RVI components are adequately managed and to provide reasonable assurance that the internals components will continue to perform their intended function through the period of extended operation. Furthermore, this AMP will demonstrate the consistency of the program with the elements documented in NUREG-1801, Revision 2 [10], Chapter XI.M.16A, "PWR Vessel Internals." The operating experience provided by NUREG-1801, Revision 2 [10] will also be reviewed and incorporated into plant-specific programs.

1.3 ANO-2 Reactor Vessel Internals Aging Management Program Background

The managing of aging degradation effects in RVI is required for nuclear plants considering or entering license renewal, as specified in the NRC Standard Review Plan (SRP) for License Renewal Applications [11]. The U.S. nuclear industry has been actively engaged in supporting the industry goal of responding to these requirements. Various programs have been established within the industry over the past decade to develop guidelines for managing the aging effects of Pressurized Water Reactor (PWR) RV internals. In 2001, Combustion Engineering Owners Group (CEOG) issued CE NPSD-1216 "Generic Aging Management Review Report for the Reactor Vessel Internals" [12]. Later, in 2008, MRP-227, Revision 0 [8] was published by EPRI MRP to address the PWR vessel internals aging management issue for the three currently operating U.S. PWR designs, namely, Combustion Engineering (CE), Westinghouse, and Babcock & Wilcox (B&W).

The MRP first established a framework and strategy for the aging management of PWR internals components using proven and familiar methods for inspection, monitoring, surveillance, and communication. Based upon that framework and strategy, and on the accumulated industry research data, the following elements of an Aging Management Program were further developed [13 – 15]:

- Screening criteria were developed, considering chemical composition, neutron fluence exposure, temperature history, and representative stress levels, for determining the relative susceptibility of PWR internals components to each of eight postulated aging mechanisms.
- PWR internals components were categorized, based on the screening criteria, into categories that ranged from:
    - components for which the effects from the postulated aging mechanisms are insignificant,
    - components that are moderately susceptible to the aging effects, and
    - components that are significantly susceptible to the aging effects.
- Functionality assessments were performed to determine the effects of the degradation mechanisms on component functionality. These assessments were based on representative plant designs of PWR internals components and assemblies of components using irradiated and aged material properties.

Aging management strategies for implementing the appropriate aging management methodology, baseline examination timing, and the need and timing of subsequent inspections were developed. Development of these strategies was based on combining the results of functionality assessment with several contributing factors including component accessibility, operating experience, existing evaluations, and prior examination results.

The industry efforts, as coordinated by the EPRI MRP, has finalized the inspection and evaluation (I&E) guidelines for the RVI, and the NRC has endorsed this document by issuing a safety evaluation (SE). A supporting document addressing inspection requirements has also been completed. The industry guidance is contained in the following documents:

- Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A) [4] provides the industry background, listing of reactor vessel internal components requiring inspection, the type or types of nondestructive examination (NDE) required for each component, timing for initial inspections, and criteria for evaluating inspection results. MRP-227-A provides a standardized approach to PWR internals aging management for each unique reactor design (Westinghouse, B&W, and CE). The NRC has endorsed MRP-227-A by issuing a safety evaluation (SE) [5].
- MRP-228 [14], "Inspection Standard for PWR Internals," provides guidance on the qualification and demonstration of the NDE techniques and other criteria pertaining to the actual performance of the inspection.

The PWR Owners Group (PWROG) has developed and submitted for NRC review and approval WCAP-17096, "Reactor Internals Acceptance Criteria Methodology and Data Requirements" [16] for MRP-227-A inspections, where feasible. This document has been approved by the NRC through the issuance of a final safety evaluation [17]. Final reports are to be developed and be available for industry use in support of planned license renewal inspection commitments. Plant specific acceptance criteria can also be developed for some internals components if a generic approach is not practical.

The ANO-2 RVI are a part of the primary reactor coolant system (RCS), which is a two-loop CE designed nuclear steam supply system (NSSS).

A review of Section 2.3.1.2 of the ANO-2 LRA specifies that the RVI are comprised of the following component groups:

- Control Element Assembly (CEA) Shroud Assembly

- Core Shroud Assembly
- Core Support Barrel (CSB) Assembly
- Incore Instrumentation (ICI)
- Lower Internals Assembly (Lower Support Structure Assembly)
- Upper Internals Assembly (Upper Guide Structure Assembly)

The reactor coolant enters through the inlet nozzles of the reactor vessel, flows downward between the reactor vessel wall and the core barrel, passes through the flow skirt where the flow distribution is equalized, and then into the lower plenum. The coolant then flows upward through the core removing heat from the fuel rods. The heated coolant enters the outlet plenum where it flows around the outside of the CEA shrouds to the reactor vessel outlet nozzles. The CEA shrouds protect the CEAs from the effects of coolant crossflow in the outlet plenum.

The ANO-2 LRA lists the following functions for the reactor vessel internals [1]:

- Provide support and orientation for the reactor core
- Provide support, orientation, guidance, and protection of the CEAs
- Direct the reactor coolant flow from the core
- Provide a passageway for support, guidance, and protection for the incore instrumentation

The function of the RVI is described in Section 4.1 of the ANO-2 Updated Final Safety Analysis Report (UFSAR) [18]. The reactor internals support and orient the fuel assemblies, CEAs, and incore instrumentation, and guide the reactor coolant through the reactor vessel. They also absorb the static and dynamic loads and transmit the loads to the reactor vessel flange. They will safely perform their functions during normal operating, upset and emergency conditions. The internals are designed to safely withstand the forces due to deadweight, handling, pressure differentials, flow impingement, temperature differentials, vibration and seismic acceleration.

ANO-2 was granted a license for extended operation by the NRC through the issuance of an SER in NUREG-1828 [19]. In the SER, the NRC concluded that, "On the basis of its evaluation of the license renewal application, the NRC staff has determined that the requirements of 10 CFR 54.29(a) have been met." It was concluded that the ANO-2 LRA [1] adequately identified the RVI components that are subject to an Aging Management Review (AMR), and that the requirements of 10 CFR 54.21(a) [6] had been met. A listing of the ANO-2 RVI components and subcomponents subject to AMP requirements is summarized in Table 3.1.2-2 of the ANO-2 LRA [1].

1.4 ANO-2 Reactor Vessel Internals Aging Management Program Elements

The key elements of the ANO-2 Reactor Vessel Internals Aging Management Program, based on the GALL process contained in NUREG-1801 [10], are outlined in Table 1-1. The program attributes are described in detail in Section 2.3 of this document. Additionally, Entergy participates in PWR Owners Group Materials Subcommittee (PWROG MSC) and the MRP to focus on preventing material degradation, improve plant performance, sharing lessons learned from operating experience, and provide an effective interface with the NRC. As RVI examination experiences are shared amongst other utilities, MRP, and PWROG MSC, the RVI AMP key elements will be updated to include any relevant OE or lessons learned.

Table 1-1. Key Elements of the Reactor Vessel Internals Aging Management Program

	Plan Attribute	Attribute Description
1	Scope of Program	The scope of this AMP is MRP-227-A [4] and the SE for MRP-227, Rev. 0 [5]. Supplemental inspections of RV internals are described in MRP-227-A [4]. Additional actions and long range plans for aging management of internals are defined within this document. The scope of the program is described in more detail in Section 2.3.1 of this document.
2	Preventive Actions	Preventive measures are described in Section 2.3.2 of this document.
3	Parameters Monitored/Inspected	ANO-2 monitors, inspects, and/or tests for the effects of the eight aging degradation mechanisms on the intended function of the reactor vessel internals components as described in Section 2.3.3 of this document.
4	Detection of Aging Effects	The ANO-2 ASME Section XI ISI program [2] for B-N-3 internals components and the additional locations identified in MRP-227-A [4], form the inspection plan for detection and monitoring of aging effects in the RV internals as described in Section 2.3.4 of this document.
5	Inspection Program for Monitoring and Trending	This program, in combination with the ASME Section XI ISI program [2], provides direction for inspections required to support continued RV internals component reliability as described in Section 2.3.5 of this document.
6	Acceptance Criteria	Acceptance criteria used in the RV Internals Aging Management Program are based on the most appropriate ASME Section XI [9] and WCAP-17096 [16] criteria as described in Section 2.3.6 of this document.
7	Corrective Actions	Components with identified relevant conditions shall be dispositioned as described in Section 2.3.7 of this document. The disposition can include a supplementary examination to further characterize the relevant condition, an engineering evaluation to show that the component is capable of continued operation with a known relevant condition until the next planned inspection, or repair/replacement to remediate the relevant condition. Additional inspections of expansion category components may also be required as specified in MRP-227-A.
8	Confirmation Process	The confirmation process for the RV Internals Program is described in Section 2.3.8 of this document.
9	Administrative Controls	Administrative controls that apply to the RVI AMP, procedures, reviews and approval processes is described in Section 2.3.9 of this document
10	Operating Experience	Operating experience related to the ANO-2 RV internals is described in Section 2.3.10 of this document.

1.5 Responsibilities

The RVI Program Manager has overall responsibility for the development and implementation of the RVI aging management plan. The responsibilities for implementing the NEI 03-08, Materials Initiative Process, are described in Reference 21. The RVI program is implemented in accordance with EN-DC-133 [20]. Entergy actively participates in industry programs related to materials initiatives such as PWROG, EPRI MRP, and other programs related to aging management of reactor vessel internals.

The Reactor Vessel Internals Program Manager is responsible for:

- Overall development of the RVI aging management plan
- Administering and overseeing the implementation of the RVI aging management plan
- Ensuring that regulatory requirements related to inspection activities, if any, are met and incorporated into the plan
- Communicating with senior management on periodic updates to the plan
- Maintaining the RVI aging management plan to incorporate changes and updates based on new knowledge and experience gained
- Reviewing and approving industry and vendor programs related to RVI aging management activities
- Processing of any deviations taken from issue programs (IP) guidelines in accordance with NEI 03-08 [3] requirements
- Ensure prompt notification of RCS Materials Degradation Management Program Manager whenever an issue or indication of potential generic industry significance is identified
- Participate in the planning and implementation of inspections of the internals.

The ISI Engineer is responsible for:

- Planning and implementing inspections required by Section XI for B-N-3 components [2], the supplemental inspections identified in this program plan, and any other plant-specific commitments for inspection for managing aging of RVI.
- Participating in industry groups such as Performance Demonstration Initiative (PDI), EPRI MRP TAC Inspection Subcommittee, etc.

The ISI Engineer and Level III NDE Coordinator are responsible for:

- Providing NDE services
- Reviewing and approving vendor NDE procedures and personnel qualifications
- Providing direction and oversight of contracted NDE activities

1.6 Program Implementation

ANO-2's overall strategy for managing aging in reactor vessel internals components is supported by the following existing programs:

- ASME Section XI IWB, IWC, IWD, and IWF Inservice Inspection Program [2]
- Primary Chemistry Monitoring Program [22]

These are established programs that support the aging management of RCS components in addition to the RVI components.

1.6.1 ASME Section XI Inservice Inspection Program

The ANO-2 Inservice Inspection (ISI) Program [2] is a plant-specific program encompassing ASME Section XI Subsections IWB, IWC, IWD and IWF requirements. The ISI Program manages cracking, loss of mechanical closure integrity, and loss of material due to wear of reactor coolant system piping and components. The program implements the applicable requirements of ASME Section XI, Subsections IWB, IWC, IWD, and IWF, and other requirements specified in 10CFR50.55a with approved NRC alternatives and relief requests. The ISI program supports aging management of the ASME Category B-N-3 RVI components. The ISI Program is updated as required to the latest ASME Section XI code edition and addendum approved by the Nuclear Regulatory Commission in 10CFR50.55a.

1.6.2 Primary Chemistry Monitoring Program

The Primary Chemistry Monitoring Program at ANO-2 [22] is comparable to the program described in NUREG-1801, Section XI.M2, Water Chemistry [10].

The main objective of this program is to manage aging effects caused by corrosion and cracking mechanisms. The program relies on monitoring and control of water chemistry based on the EPRI guidelines for primary water chemistry [23].

1.7 Aging Management Review and Program Enhancements

1.7.1 Reactor Internals Aging Management Review Process

A comprehensive review of aging management of RVI was performed as part of the ANO-2 license renewal application [1]. The AMR performed for the LRA submittal documents the results of the aging management review for the ANO-2 RVI. The NRC indicated its approval of the ANO-2 LRA in NUREG-1828 [19]. The RVI components specifically noted as requiring aging management are identified in Table 3.1.2-2 of the LRA [1].

The assessments supporting the LRA performed the following:

1. Identified applicable aging effects requiring management
2. Associated aging management programs to manage those aging effects
3. Identified enhancements or modifications to existing programs, new aging management programs, or any other actions required to support the conclusions reached in the assessment

AMRs were performed for each ANO-2 system that contained long-lived, passive components requiring an aging management review, in accordance with the ANO-2 screening process. The results of these reviews have been incorporated into the ANO-2 RVI AMP.

1.8 Industry Programs

1.8.1 CE NPSD-1216, Aging Management of Reactor Internals

The Combustion Engineering Owner's Group (CEOG, now PWROG) topical report CE NPSD-1216 [12] contains a technical evaluation of aging degradation mechanisms and aging effects for CE RVI components. The CEOG report provided guidance for CEOG member plant owners to manage effects of aging on RVI during the period of extended operation, using approved aging management methodologies to develop plant-specific AMPs.

The AMR for the ANO-2 internals, documented in the ANO-2 license renewal application [1], was completed in a manner consistent with the approach of CE NPSD-1216 [12]. Both the ANO-2 specific AMR document [24] and the generic CE document were completed to facilitate plant license renewal in accordance with 10 CFR Part 54 [6].

1.8.2 MRP-227-A, Reactor Internals Inspection and Evaluation Guidelines

MRP-227-A was developed by a team of industry experts including utility representatives, NSSS vendors, independent consultants, and international representatives who reviewed available data and industry experience on materials aging. The objective of this project was to develop a consistent, systematic approach for identifying and prioritizing inspection requirements for reactor vessel internals. For details regarding this approach, refer to Section 2.2 of this document.

1.8.3 NEI 03-08 Guidance Within MRP-227-A

The industry program requirements of MRP-227-A are classified in accordance with the requirements of the NEI 03-08 [3] protocols. The MRP-227-A [4] guideline includes "mandatory," "needed," and "good practice" requirements defined as the following:

- <u>Mandatory</u>

  Each commercial U.S. PWR unit shall develop and document a PWR reactor internals aging management program within 36 months following issuance of MRP-227, Rev. 0 (that is, no later than December 31, 2011).

  ANO-2 Applicability: MRP-227, Revision 0 was officially issued by the industry in December 2008 [8]. An aging management program was to be developed by December 2011. In order to meet this "Mandatory" requirement, an aging management program plan for ANO-2 was completed in November 2011 [7]. This AMP replaces the previous AMP [7] to conform to the updated requirements of MRP-227-A.

  ANO-2 qualifies as a Category B plant according to the NRC Regulatory Issue Summary (RIS) [25] being a plant with a renewed license that has a commitment to submit an AMP/inspection plan based on MRP-227-A, but has not yet been required to do so by their commitment. This AMP fulfills the license renewal commitment to submit a description of this program, including the inspection plan, to the NRC for review and approval.

- <u>Needed</u>

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

  ANO-2 Applicability: MRP-227 augmented inspections will be incorporated in the ANO-2 ISI program for the license renewal period. The applicable CE tables contained in MRP-227-A for RVI components are Table 4-2 (Primary), Table 4-5 (Expansion), Table 4-8 (Existing Programs), and Table 5-2 (Acceptance and Expansion Criteria) and are attached herein as Table 5-1, Table 5-2, Table 5-3, and Table 5-4, respectively.

  This AMP has been developed in accordance with MRP-227-A [4].

2. *Examinations specified in the MRP-227-A guidelines shall be conducted in accordance with Inspection Standard [MRP-228].*

   ANO-2 Applicability: Inspections will be in accordance with the requirements of MRP-228 [14]. These inspection standards will be used for augmented inspections at ANO-2, as applicable, where required by MRP-227-A.

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

   ANO-2 Applicability: ANO-2 will comply with this requirement [26].

4. *Each commercial U.S. PWR unit shall 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 within the scope of MRP-227 are examined.*

   ANO-2 Applicability: ANO-2 will comply with this requirement.

5. *If an engineering evaluation is used to disposition an examination result that does not meet the examination acceptance criteria in Section 5 [of MRP-227-A], this engineering evaluation shall be conducted in accordance with a NRC-approved evaluation methodology.*

   ANO-2 Applicability: ANO-2 will comply with this requirement by using NRC-approved evaluation methodology (e.g. WCAP-17096 [16]).

1.8.4 MRP-227-A AMP Development Guidance

In addition to the implementation of the requirements of MRP-227-A in accordance with NEI

03-08, this RVI AMP addresses the 10 program elements as defined in the GALL Report Chapter XI.M16A (provided in Section 2.3 of this Report)

1.8.4.1 MRP-227-A Applicability to ANO-2

The applicability of MRP-227-A to ANO-2 requires compliance with the following MRP-227 assumptions:

- *30 years 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.*

  Applicability: ANO-2 historic core management practices meet the requirements of MRP-227-A [27].

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

  Applicability: ANO-2 operates as a base load unit [18, Section 10.2.1].

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

  Applicability: MRP-227-A states that the recommendations are applicable to all U.S. PWR operating plants as of May 2007 for the three designs considered. ANO-2 has not made any modifications of the RVI components beyond those identified in general industry guidance or recommended by the vendor (CE) since the May 2007 effective date of this statement, and therefore meets this requirement of MRP-227-A.

Hence, it is evident that operations at ANO-2 conform to the assumptions in Section 2.4 of MRP 227-A.

1.8.5 Ongoing Industry Programs

Entergy actively participates in the EPRI MRP, PWROG, and other activities related to PWR internals inspection and aging management and will address/implement industry guidance stemming from those activities, as appropriate under NEI 03-08 practices.

1.9 Summary

The GALL Report identifies which reactor internals passive components are most susceptible to the aging mechanisms of concern. Additionally, this report identifies the appropriate inspections or mitigation programs needed to manage the aging mechanisms of the reactor vessel internals to assure these components will maintain their functionality through the period of extended operation. The GALL Report was used at ANO-2 for the initial basis of their LRA. The NRC has reviewed ANO-2's LRA and their approval is documented in NUREG-1828 [19].

The ANO-2 RVI AMP has been created to address the reactor vessel internals aging concerns consistent with the information identified in the GALL Report, the guidance in MRP-227-A, and the SE of MRP-227, Revision 0 issued by the NRC. ANO-2 will manage their RVI inspections through their augmented ISI program and will complete any repairs and/or replacements in accordance with ASME Code requirements and any NRC approved methodologies. The ANO-2 AMP will be updated accordingly as operating experiences and new inspection requirements and technologies evolve associated with managing reactor vessel aging concerns.

2.0 AGING MANAGEMENT APPROACH

2.1 Mechanisms of Age-Related Degradation in PWR Internals

Aging degradation mechanisms that impact RVI have been identified and documented in ANO-2-specific AMRs in support of license renewal [1, Table 3.1.2-2]. The potential aging mechanisms that could affect the long term operation of PWR reactor vessel internals are discussed in this section. Initial screening performed as part of MRP-227-A was on the basis of susceptibility of PWR RVI to eight different age-related degradation mechanisms – stress corrosion cracking (SCC), irradiation-assisted stress corrosion cracking (IASCC), wear, fatigue, thermal aging embrittlement, irradiation embrittlement, void swelling, and the combination of thermal and irradiation-enhanced stress relaxation.

2.1.1 Stress Corrosion Cracking

Stress Corrosion Cracking (SCC) refers to local, non-ductile cracking of a material due to a combination of tensile stress, environment, and metallurgical properties. The actual mechanism that causes SCC involves a complex interaction of environmental and metallurgical factors. The aging effect is cracking.

2.1.2 Irradiation-Assisted Stress Corrosion Cracking

Irradiation-assisted stress corrosion cracking (IASCC) is a unique form of SCC that occurs only in highly-irradiated components. The aging effect is cracking.

2.1.3 Wear

Wear is caused by the relative motion between adjacent surfaces, with the extent determined by the relative properties of the adjacent materials and their surface condition. The aging effect is loss of material.

2.1.4 Fatigue

Fatigue is defined as the structural deterioration that can occur as a result of repeated stress/strain cycles caused by fluctuating loads and temperatures. After repeated cyclic loading of sufficient magnitude, microstructural damage can accumulate, leading eventually to macroscopic crack initiation at the most highly affected locations. Subsequent mechanical or thermal cyclic loading can lead to growth of the initiated crack. Corrosion fatigue is included in the degradation description.

Low-cycle fatigue is defined as cyclic loads that cause significant plastic strain in the highly stressed regions, where the number of applied cycles is increased to the point where the crack eventually initiates. When the cyclic loads are such that significant plastic deformation does not occur in the highly stressed regions, but the loads are of such increased frequency that a fatigue crack eventually initiates, the damage accumulated is said to have been caused by high-cycle fatigue. The aging effects of low-cycle fatigue and high-cycle fatigue are additive.

Fatigue crack initiation and growth resistance is governed by a number of material, structural and environmental factors, such as stress range, loading frequency, surface condition, and presence of deleterious chemical species. Cracks typically initiate at local geometric stress concentrations, such as notches, surface defects, and structural discontinuities. The aging effect is cracking.

2.1.5 Thermal Aging Embrittlement

Thermal aging embrittlement is the exposure of delta ferrite within cast austenitic stainless steel (CASS) and precipitation-hardenable (PH) stainless steel to high inservice temperatures, which can result in an increase in tensile strength, a decrease in ductility, and a loss of fracture toughness. Some degree of thermal aging embrittlement can also occur at normal operating temperatures for CASS and PH stainless steel internals. CASS components have a duplex microstructure and are particularly susceptible to this mechanism. While the initial aging effect is loss of ductility and toughness, unstable crack extension is the eventual aging effect if a crack is present and the local applied stress intensity exceeds the reduced fracture toughness.

2.1.6 *Irradiation Embrittlement*

Irradiation embrittlement is also referred to as neutron embrittlement. When exposed to high energy neutrons, the mechanical properties of stainless steel and nickel-base alloys can be changed. Such changes in mechanical properties include increasing yield strength, increasing ultimate strength, decreasing ductility, and a loss of fracture toughness. The irradiation embrittlement aging mechanism is a function of both temperature and neutron fluence. While the initial aging effect is loss of ductility and toughness, unstable crack extension is the eventual aging effect if a crack is present and the local applied stress intensity factor exceeds the reduced fracture toughness.

2.1.7 *Void Swelling and Irradiation Growth*

Void swelling is defined as a gradual increase in the volume of a component caused by formation of microscopic cavities in the material. These cavities result from the nucleation and growth of clusters of irradiation produced vacancies. Helium produced by nuclear transmutations can have a significant impact on the nucleation and growth of cavities in the material. Void swelling may produce dimensional changes that exceed the tolerances on a component. Strain gradients produced by differential swelling in the system may produce significant stresses. Severe swelling (>5% by volume) has been correlated with extremely low fracture toughness values. Also included in this mechanism is irradiation growth of anisotropic materials, which is known to cause significant dimensional changes in in-core instrumentation tubes fabricated from zirconium alloys. While the initial aging effect is dimensional change and distortion, severe void swelling may eventually result in cracking under stress.

2.1.8 *Thermal and Irradiation-Enhanced Stress Relaxation or Creep*

The loss of preload aging effect can be caused by the aging mechanisms of stress relaxation or creep. Thermal stress relaxation (or, primary creep) is defined as the unloading of preloaded components due to long-term exposure to elevated temperatures, such as seen in PWR internals. Stress relaxation occurs under conditions of constant strain where part of the elastic strain is replaced with plastic strain. Available data show that thermal stress relaxation appears to reach saturation in a short time (<1000 hours) at PWR internals temperatures.

Creep (or more precisely, secondary creep) is a slow, time and temperature dependent, plastic deformation of materials that can occur when subjected to stress levels below the yield strength (elastic limit). Creep occurs at elevated temperatures where continuous deformation takes place under constant strain. Secondary creep in austenitic stainless steels is associated with temperatures higher than those relevant to PWR internals even after taking into account gamma heating. However, irradiation-enhanced creep (or more simply, irradiation creep) or irradiation-enhanced stress relaxation (ISR) is a thermal process that depends on the neutron fluence and stress; and, it can also be affected by void swelling, should it occur. The aging effect is a loss of mechanical closure integrity (or, preload) that can lead to unanticipated loading which, in turn, may eventually cause subsequent degradation by fatigue or wear and result in cracking.

2.2 Aging Management Strategy

The guidelines provided in MRP-227-A [4] define a supplemental inspection program for managing aging effects and provide generic guidance to help develop this aging management program for ANO-2. The EPRI MRP Reactor Internals Focus Group developed these guidelines to support the continued functionality of RVI. The focus group also developed MRP-228, which addresses the inspection standard for the RVI. The aging management strategy used to develop the guidelines combined the results of functionality assessment with component accessibility, operating experience, existing evaluations, and prior examination results. The aging management strategy that was developed was used in the development of an appropriate aging management methodology, baseline examination timing, and the need and timing of subsequent inspections. Additionally, it was also used to identify the components and locations for supplemental examinations by categorization.

MRP-227-A used a screening and ranking process to aid the identification of required inspections for specific RVI components. The screening and categorization process also credited existing component inspections, when they were deemed adequate. Through the screening and categorization process, the RVI for all currently licensed and operating PWR designs in the U.S. were evaluated, and appropriate inspection, evaluation and implementation requirements for RVI were defined.

The RVI components are categorized in MRP-227-A as "Primary" components, "Expansion" components, "Existing Programs" components, or "No Additional Measures" components, described 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 examination 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:** 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.

A description of the categorization process used to develop the Guidelines is given below. The approach in these guidelines has been used to develop the ANO-2 AMP.

In accordance with the MRP-227-A I&E Guidelines [4], this inspection strategy consists of the following:

- Selection of items for aging management
- Selection of the type of examination or other methodologies appropriate for each degradation mechanism
- Specification of the required level of examination qualification
- Schedule of first and frequency of any subsequent examinations
- Sampling and coverage
- Expansion of scope if sufficient evidence of degradation is observed
- Examination acceptance criteria
- Methods for evaluating examination results not meeting the examination acceptance criteria
- Updating the program based on industry-wide results
- Contingency measures to repair, replace, or mitigate

The MRP first established a framework and strategy for the aging management of PWR internals components using proven and familiar methods for inspection, monitoring, surveillance, and communication. Based on the framework and strategy and on the accumulated industry research data, the following elements of an AMP were further developed [13, 15]:

- Screening criteria were developed, considering chemical composition, neutron fluence exposure, temperature history, and representative stress levels, for determining the relative susceptibility of PWR internals components to each of eight postulated aging mechanisms.
- PWR internals components were categorized, based on the screening criteria as follows:
  - Components for which the effects of the postulated aging mechanisms are insignificant

- Components that are moderately susceptible to the aging effects
- Components that are significantly susceptible to the aging effects

- Functionality assessments were performed based on representative plant designs of PWR internals components and assemblies of components, using irradiated and aged material properties, to determine the effects of the degradation mechanisms on component functionality.

Aging management strategies were developed combining the results of the functionality assessment with several contributing factors to determine the appropriate aging management methodology, baseline examination timing, and the need and timing of subsequent inspections. Factors considered included component accessibility, operating experience (OE), existing evaluations, and prior examination results.

2.3 ANO-2 Reactor Vessel Internals Aging Management Program Attributes

The attributes of the ANO-2 RVI AMP and their compliance with the ten elements of NUREG-1801 (GALL Report), Revision 2, Chapter XI.M16A, "PWR Vessel Internals" [10] are included in this section to ensure successful management of component aging.

This AMP is consistent with the GALL process and includes consideration of the augmented inspections identified in MRP-227-A [4]. Specific details of the ANO-2 RVI AMP are summarized in the following subsections.

2.3.1 NUREG-1801/AMP Program Element 1: Scope of Program

"The scope of the program includes all RVI components at the Arkansas Nuclear One, Unit 2, which is built to a CE NSSS design. The scope of the program applies to the methodology and guidance in the most recently NRC endorsed version of MRP-227, which provides augmented inspection and flaw evaluation methodology for assuring the functional integrity of safety-related internals in commercial operating U.S. PWR nuclear power plants designed by B&W, CE, and Westinghouse. The scope of components

considered for inspection under MRP-227 guidance includes core support structures (typically denoted as Examination Category B-N-3 by the ASME Code, Section XI), those RVI components that serve an intended license renewal safety function pursuant to criteria in 10 CFR 54.4(a)(1), and other RVI components whose failure could prevent satisfactory accomplishment of any of the functions identified in 10 CFR 54.4(a)(1)(i), (ii), or (iii). The scope of the program does not include consumable items, such as fuel assemblies, reactivity control assemblies, and nuclear instrumentation, because these components are not typically within the scope of the components that are required to be subject to an aging management review (AMR), as defined by the acceptance criteria set in 10 CFR 54.21(a)(1). The scope of the program also does not include welded attachments to the internal surface of the reactor vessel because these components are considered to be ASME Code Class 1 appurtenances to the reactor vessel and are adequately managed in accordance with an applicant's AMP that corresponds to GALL AMP XI.M1, "ASME Code, Section XI Inservice Inspections, Subsections IWB, IWC, and IWD."

"The scope of the program includes the response bases to applicable license renewal applicant action items (LRAAIs) on the MRP-227 methodology, and any additional programs, actions, or activities that are discussed in these LRAAI responses and credited for aging management of the applicant's RVI components. The LRAAIs are identified in the staff's safety evaluation on MRP-227 and include applicable action items on meeting those assumptions that formed the basis of the MRP's augmented inspection and flaw evaluation methodology (as discussed in Section 2.4 of MRP-227), and NSSS vendor-specific or plant-specific LRAAIs as well. The responses to the LRAAIs on MRP-227 are provided in Appendix C of the LRA."

"The guidance in MRP-227 specifies applicability limitations to base-loaded plants and the fuel loading management assumptions upon which the functionality analyses were based. These limitations and assumptions require a determination of applicability by the applicant for each reactor and are covered in Section 2.4 of MRP-227."

2.3.1.1 ANO-2 Program Scope

A description of the ANO-2 RVI design is provided in Section 3.0 of this program plan. Additional details regarding the RVI are provided in the ANO-2 UFSAR [18]. The ANO-2 RVI subcomponents that require aging management review are indicated in the ANO-2 LRA [1].

Table 3.1.2-2 of the ANO-2 LRA includes a summary of the results of the AMR. This table identifies the aging effects that require management. A column in the table lists the programs and activities at ANO-2 that are credited to address the aging effects for each management strategy presented in Table 3.1.2-2 of the ANO-2 LRA and Section 3.1.2.3.2 of the NRC's license renewal SER [19].

MRP-227-A provides the inspection and evaluation guidelines to develop plant specific programs to manage the effects of aging in PWR internals. MRP-227-A is also used as a guidance to develop an aging management program to satisfy license renewal commitments for the PWR fleet. A summary of the inspections required to be performed, the appropriate inspection techniques used to detect aging (i.e. cracking, loss of material, loss of preload, etc.), frequency of inspections, and the acceptance criteria for the inspections are provided in MRP-227-A (summarized in Table 5-1 through Table 5-4 of this AMP). Guidance provided in MRP-227-A in conjunction with the guidance provided in the NRC SE [5] for MRP-227, Revision 0 and the GALL Report were reviewed to establish the basis for the ANO-2 RVI AMP. The basic assumptions of MRP-227-A, Section 2.4 are met by ANO-2 and are addressed in Section 1.8.4.1 of this AMP. The Topical Report Conditions and Applicant/Licensee Action Items provided by the NRC in the SE on MRP-227, Revision 0 [5] are met by ANO-2 and demonstration of compliance is addressed in Section 5.0. In addition, plant specific existing programs such as the Section XI ISI program for ANO-2 will complement the augmented inspection requirements provided in MRP-227-A in successfully managing the effects of aging for ANO-2 during the period of extended operation.

2.3.1.2 Conclusion

This element is consistent with the corresponding aging management attribute in Revision 2 of NUREG-1801 [10], Chapter XI.M16A and Appendix A (Commitment 19) of the ANO-2 license renewal SER [19], and the ANO-2 UFSAR [18].

2.3.2 NUREG-1801/AMP Program Element 2: Preventive Actions

> *"The guidance in MRP-227 relies on PWR water chemistry control to prevent or mitigate aging effects that can be induced by corrosive aging mechanisms (e.g., loss of material induced by general, pitting corrosion, crevice corrosion, or stress corrosion cracking or any of its forms [SCC, PWSCC, or IASCC]). Reactor coolant water chemistry is monitored and maintained in accordance with the Water Chemistry Program. The program description, evaluation and technical bases of water chemistry are presented in GALL AMP XI.M2, "Water Chemistry."*

2.3.2.1 ANO-2 Preventive Action

The ANO-2 RVI AMP includes the Primary Chemistry Monitoring Program [22] as an existing program that complies with the requirement of this element. A description and applicability to the ANO-2 RVI AMP is provided in the following subsection.

2.3.2.2 Primary Chemistry Monitoring Program

The primary goal of this program is to mitigate loss of material due to general, pitting, and crevice corrosion, and cracking due to Stress Corrosion Cracking (SCC) by controlling the internal environment of systems and components. This program relies on monitoring and control of water chemistry to keep peak levels of various contaminants below the system-specification limits. The ANO-2 water chemistry program [22] is based on current, approved revisions of EPRI PWR Primary Water Chemistry Guidelines [23].

This program is consistent with the corresponding program described in Revision 2 for GALL Report [10]. The program description, evaluation, and technical basis of water chemistry are presented in GALL AMP XI.M2, "Water Chemistry."

The limits of known detrimental contaminants imposed by the water chemistry program are consistent with the EPRI PWR Primary Water Chemistry Guidelines [23].

2.3.2.3 Conclusion

This element is consistent with the corresponding aging management program attribute in Revision 2 of NUREG-1801 [10], Chapter XI.M16A and Appendix A (Commitment 19) of the ANO-2 license renewal SER [19].

2.3.3 NUREG-1801/AMP Program Element 3: Parameters Monitored/Inspected:

> *"The program monitors and manages the following age-related degradation effects and mechanisms that are applicable in general to RVI components at the facility: (a) cracking induced by SCC, PWSCC, IASCC, or fatigue/cyclical loading; (b) loss of material induced by wear; (c) loss of fracture toughness induced by either thermal aging or neutron irradiation embrittlement; (d) changes in dimension due to void swelling and irradiation growth, distortion, or deflection; and (e) loss of preload caused by thermal and irradiation-enhanced stress relaxation or creep. For the management of cracking, the program monitors for evidence of surface breaking linear discontinuities if a visual inspection technique is used as the non-destructive examination (NDE) method, or for relevant flaw presentation signals if a volumetric UT method is used as the NDE method. For the management of loss of material, the program monitors for gross or abnormal surface conditions that may be indicative of loss of material occurring in the components. For the management of loss of preload, the program monitors for gross surface conditions that may be indicative of loosening in applicable bolted, fastened, keyed, or pinned connections. The program does not directly monitor for loss of fracture*

toughness that is induced by thermal aging or neutron irradiation embrittlement, or by void swelling and irradiation growth; instead, the impact of loss of fracture toughness on component integrity is directly managed by using visual or volumetric examination techniques to monitor for cracking in the components and by applying applicable reduced fracture toughness properties in the flaw evaluations if cracking is detected in the components and is extensive enough to warrant a supplemental flaw growth or flaw tolerance evaluation under the MRP-227 guidance or ASME Code, Section XI requirements. The program uses physical measurements to monitor for any dimensional changes due to void swelling, irradiation growth, distortion, or deflection."

"Specifically, the program implements the parameters monitored/inspected criteria for CE designed Primary Components in Table 4-2 of MRP-227. Additionally, the program implements the parameters monitored/inspected criteria for CE designed Expansion Components in Table 4-5 of MRP-227. The parameters monitored/inspected for Existing Program Components follow the bases for referenced Existing Programs, such as the requirements of the ASME Code Class RVI components in ASME Code, Section XI, Table IWB-2500-1, Examination Categories B-N-3, as implemented through the applicant's ASME Code, Section XI program, or the recommended program for inspecting Westinghouse designed flux thimble tubes in GALL AMP XI.M37, "Flux Thimble Tube Inspection." No inspections, except for those specified in ASME Code, Section XI, are required for components that are identified as requiring "No Additional Measures," in accordance with the analyses reported in MRP-227."

2.3.3.1 ANO-2 Parameters Monitored/Inspected

ANO-2 monitors, inspects, and/or tests for the effects of the eight aging degradation mechanisms on the intended function of the reactor vessel internals components through inspection and condition monitoring activities in accordance with the augmented requirements defined under industry directives as contained in MRP-227-A [4] and ASME Section XI [9].

This AMP implements the requirements for the Primary Component inspections from Table 4-2 of MRP-227-A (Table 5-1 of this AMP), the Expansion Component inspections from Table 4-5 of MRP-227-A (Table 5-2 of this AMP), and the Existing Component inspections from Table 4-8 of MRP-227-A (Table 5-3 of this AMP). These tables contain requirements to monitor and inspect the RVI through the period of extended operation to address the effects of the eight aging degradation mechanisms.

For license renewal, the ASME Section XI Program [2] consists of periodic volumetric, surface, and/or visual examination of components for assessment, signs of degradation, and corrective actions. This program is consistent with the corresponding program described in the GALL Report [10].

2.3.3.2 Conclusion

This element is consistent with the corresponding aging management attribute in Revision 2 of NUREG-1801 [10], Chapter XI.M16A and Appendix A (Commitment 19) of the ANO-2 license renewal SER [19].

2.3.4 NUREG-1801/AMP Program Element 4: Detection of Aging Effects:

> *"The detection of aging effects is covered in two places: (a) the guidance in Section 4 of MRP-227 provides an introductory discussion and justification of the examination methods selected for detecting the aging effects of interest; and (b) standards for examination methods, procedures, and personnel are provided in a companion document, MRP-228. In all cases, well established methods were selected. These methods include volumetric UT examination methods for detecting flaws in bolting, physical measurements for detecting changes in dimensions, and various visual (VT-3, VT-1, and EVT-1) examinations for detecting effects ranging from general conditions to detection and sizing of surface-breaking discontinuities. Surface examinations may also be used as an alternative to visual examinations for detecting and sizing of surface breaking discontinuities."*

"Cracking caused by SCC, IASCC, and fatigue is monitored/inspected by either VT-1 or EVT-1 examination (for internals other than bolting) or by volumetric UT examination (bolting). The VT-3 visual methods may be applied for the detection of cracking only when the flaw tolerance of the component or affected assembly, as evaluated for reduced fracture toughness properties, is known and has been shown to be tolerant of easily detected large flaws, even under reduced fracture toughness conditions. In addition, VT-3 examinations are used to monitor/inspect for loss of material induced by wear and for general aging conditions, such as gross distortion caused by void swelling and irradiation growth or by gross effects of loss of preload caused by thermal and irradiation-enhanced stress relaxation and creep."

"In addition, the program adopts the recommended guidance in MRP-227 for defining the Expansion criteria that need to be applied to inspections of Primary Components and Existing Requirement Components and for expanding the examinations to include additional Expansion Components. As a result, inspections performed on the RVI components are performed consistent with the inspection frequency and sampling bases for Primary Components, Existing Requirement Components, and Expansion Components in MRP-227, which have been demonstrated to be in conformance with the inspection criteria, sampling basis criteria, and sample Expansion criteria in Section A.1.2.3.4 of NRC Branch Position RSLB-1."

"Specifically, the program implements the parameters monitored/inspected criteria and bases for inspecting the relevant parameter conditions for CE designed Primary Components in Table 4-2 of MRP-227 and for CE designed expansion components in Table 4-5 of MRP-227."

"The program is supplemented by the following plant-specific Primary Component and Expansion Component inspections for the program (as applicable): for ANO-2, no

additional Primary or Expansion components are relevant to the scope of aging management for the RVI."

Physical measurements: Per Revision 2 of NUREG-1801, this is not applicable for CE designed plants.

2.3.4.1 ANO-2 Detection of Aging Effects

Detection of indications required by the ASME Section XI ISI Program is well-established and field-proven through application of the Section XI ISI Program [2]. Those augmented inspections that are taken from the MRP-227-A recommendations will be applied through use of the MRP-228 [14] Inspection Standard.

Inspections can be used to detect physical effects of degradation including cracking, fracture, wear and distortion. The choice of an inspection technique depends on the nature and extent of the expected damage. The recommendations supporting aging management of the RVI, as contained in this program, are built around three basic inspection techniques: visual, ultrasonic, and physical measurement. The visual techniques include VT-3, VT-1, and EVT-1. Inspection standards developed by the industry for application of these techniques in augmented RVI inspections are documented in MRP-228 [14]. Continued functionality can be confirmed by physical measurements to detect degradation mechanisms such as wear, or loss of functionality as a result of loss of preload or material deformation. If components have been shown to be flaw tolerant, the scope of the inspections for detection of aging effects may be modified. Acceptance criteria for each inspection technique are provided in Section 4.1 of this AMP.

2.3.4.2 Conclusion

This element is consistent with the corresponding aging management attribute in Revision 2 of NUREG-1801 [10], Chapter XI.M16A and Appendix A (Commitment 19) of the ANO-2 license renewal SER [19].

2.3.5 NUREG-1801/AMP Program Element 5: Monitoring and Trending:

"The methods for monitoring, recording, evaluating, and trending the data that result from the program's inspections are given in Section 6 of MRP-227 and its subsections. The evaluation methods include recommendations for flaw depth sizing and for crack growth determinations as well for performing applicable limit load, linear elastic and elastic-plastic fracture analyses of relevant flaw indications. The examinations and re-examinations required by the MRP-227 guidance, together with the requirements specified in MRP-228 for inspection methodologies, inspection procedures, and inspection personnel, provide timely detection, reporting, and corrective actions with respect to the effects of the age-related degradation mechanisms within the scope of the program. The extent of the examinations, beginning with the sample of susceptible PWR internals component locations identified as Primary Component locations, with the potential for inclusion of Expansion Component locations if the effects are greater than anticipated, plus the continuation of the Existing Programs activities, such as the ASME Code, Section XI, Examination Category B-N-3 examinations for core support structures, provides a high degree of confidence in the total program."

2.3.5.1 ANO-2 Monitoring and Trending

Reporting operating experience with PWR internals has been generally proactive. Flux thimble wear and control rod, guide tube split pin cracking issues were identified by the industry and continue to be actively managed. The extremely low frequency of failure in reactor internals makes monitoring and trending based on operating experience somewhat impractical. The majority of materials aging degradation models and analyses used to develop the MRP-227-A guidelines are based on test data from RVI components removed from service. The data are used to identify trends in materials degradation and forecast potential component degradation. The industry continues to share both material test data and operating experience through the auspices of the MRP and PWROG. ANO-2 has in the past and will continue to maintain cognizance of industry activities and will continue to share operating experience information related to PWR internals inspection and aging management.

Inspections credited as part of the existing programs, where practical, are scheduled to be conducted in conjunction with typical 10-year ISI examinations, as documented in Section 5.0 of the ANO-2 ISI program [2].

Table 5-1 and Table 5-2 identify the inspection requirements for Primary and Expansion category components credited for aging management of RVI. As discussed in MRP-227-A [4], the sampling inspections of the "Primary" components, with the potential for expanding the sampling program if unexpected effects are found, provides reasonable assurance for demonstrating the ability of the reactor vessel internal components to perform the intended functions.

Reporting requirements are included as part of MRP-227-A guidelines. Consistent reporting of inspection results across all PWR designs will enable the industry to monitor RVI degradation on an ongoing basis as plants enter the period of extended operation. Reporting of examination results will allow the industry to monitor and trend results and take appropriate preemptive action through update of the MRP guidelines.

2.3.5.2 Conclusion

This element is consistent with the corresponding aging management attribute in Revision 2 of NUREG-1801 [10], Chapter XI.M16A and Appendix A (Commitment 19) of the ANO-2 license renewal SER [19].

2.3.6 NUREG-1801/AMP Program Element 6: Acceptance Criteria

> *"Section 5 of MRP-227 provides specific examination acceptance criteria for the Primary and Expansion Component examinations. For components addressed by*

examinations referenced to ASME Code, Section XI, the IWB-3500 acceptance criteria apply. For other components covered by Existing Programs, the examination acceptance criteria are described within the Existing Program reference document.

The guidance provided in MRP-227 contains three types of examination acceptance criteria:

- *For visual examination (and surface examination as an alternative to visual examination), the examination acceptance criterion is the absence of any of the specific, descriptive relevant conditions; in addition, there are requirements to record and disposition surface breaking indications that are detected and sized for length by VT-1/EVT-1 examinations;*
- *For volumetric examination, the examination acceptance criterion is the capability for reliable detection of indications in bolting, as demonstrated in the examination Technical Justification; in addition, there are requirements for system-level assessment for bolted or pinned assemblies with unacceptable volumetric (UT) examination indications that exceed specified limits; and*
- *Physical measurements: Per Revision 2 of NUREG-1801, this is not applicable for CE designed plants."*

2.3.6.1 ANO-2 Acceptance Criteria

Recordable indications that are the result of inspections required by ANO-2 existing ISI program scope [2] are evaluated in accordance with the requirements of the ASME Code and documented in the ANO-2 Corrective Action Process [26].

Inspection acceptance and expansion criteria are provided in Table 5-4 of this document. These criteria will be reviewed whenever new revisions of the NRC approved versions of MRP-227 and WCAP-17096 are published and as the industry continues to develop and refine the information. Changes applicable to the ANO-2 RVI will be included as part of updates to this AMP.

Recordable indications found during the MRP-227-A augmented inspections will be entered into the ANO-2 correction action program. These indications will be addressed by additional inspections, repair, replacement, mitigation, or analytical evaluations to further disposition these indications. Industry groups are working to develop a consistent set of tools compliant with approved methodologies to support this element. Additional analysis to establish evaluation acceptance criteria for "Expansion" category components has been developed by the PWROG in WCAP-17096-NP [16]. The status of these ongoing processes is monitored via Entergy participation in various industry programs related to aging management of PWR internals.

2.3.6.2 Conclusion

This element is consistent with the corresponding aging management attribute in Revision 2 of NUREG-1801 [10], Chapter XI.M16A and Appendix A (Commitment 19) of the ANO-2 license renewal SER [19].

2.3.7 NUREG-1801/AMP Program Element 7: Corrective Actions:

"Corrective actions following the detection of unacceptable conditions are fundamentally provided for in each plant's corrective action program. Any detected conditions that do not satisfy the examination acceptance criteria are required to be dispositioned through the plant corrective action program, which may require repair, replacement, or analytical evaluation for continued service until the next inspection. The disposition will ensure that design basis functions of the reactor internals components will continue to be fulfilled for all licensing basis loads and events. Examples of methodologies that can be used to analytically disposition unacceptable conditions are found in the ASME Code, Section XI or in Section 6 of MRP-227. Section 6 of MRP-227 describes the options that are available for disposition of detected conditions that exceed the examination acceptance criteria of Section 5 of the report. These include engineering evaluation methods, as well as supplementary examinations to further characterize the detected condition, or the alternative of component repair and replacement procedures. The latter are subject to the requirements of the ASME Code, Section XI. The implementation of the

guidance in MRP-227, plus the implementation of any ASME Code requirements, provides an acceptable level of aging management of safety-related components addressed in accordance with the corrective actions of 10 CFR Part 50, Appendix B or its equivalent, as applicable."

2.3.7.1 ANO-2 Corrective Actions

The existing ANO-2 procedure for the plant-specific Corrective Actions Program [26] is credited for this element. Repair and replacement activities will be performed in accordance with methodologies provided in Section 6 of MRP-227-A [4] and ASME Code Section XI [9]. The corrective actions for existing Section XI (B-N-3) examinations will include the identification of a repair plan and verification of acceptability of replacements. Any indications found during the Section XI examinations for the RVI will be documented in the corrective action program [26]. These indications will be addressed by additional inspections, repair, replacement, or analytical evaluations in accordance with or equivalent to the requirements of the ASME Code, Section XI. Actions to evaluate and monitor flaws or indications will be a part of the corrective action process. This evaluation guidance is included in MRP-227-A [4] and WCAP-17096-NP [16]. For example, the guidance provided in WCAP-17096-NP [16] may be used to evaluate component degradation that exceeds acceptance criteria in Section 5 of MRP-227-A [4] when it is observed during required inspections. Other methods may also be used if approved by the NRC.

2.3.7.2 Conclusion

This element is consistent with the corresponding aging management attribute in Revision 2 of NUREG-1801 [10], Chapter XI.M16A and Appendix A (Commitment 19) of the ANO-2 license renewal SER [19].

2.3.8 NUREG-1801/AMP Program Element 8: Confirmation Process

"Site quality assurance procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B, or equivalent, as applicable. It is expected that the implementation of the guidance in MRP-227 will provide an acceptable level of quality for inspection, flaw evaluation, and other elements of aging management of the PWR internals that are addressed in accordance with 10 CFR Part 50, Appendix B or their equivalent (as applicable), confirmation process, and administrative controls."

2.3.8.1 ANO-2 Confirmation Process

ANO-2 has an established 10 CFR Part 50, Appendix B Program [28] that addresses the elements of corrective actions, confirmation process, and administrative controls. The ANO-2 Section XI Inspection Program [2] and Corrective Action Process [26] meet the requirements for QA programs. In particular, all QA procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR 50, Appendix B [29].

2.3.8.2 Conclusion

This element is consistent with the corresponding aging management attribute in Revision 2 of NUREG-1801 [10], Chapter XI.M16A and Appendix A (Commitment 19) of the ANO-2 license renewal SER [19].

2.3.9 NUREG-1801/AMP Program Element 9: Administrative Controls:

"The administrative controls for such programs, including their implementing procedures and review and approval processes, are under the existing site 10 CFR 50 Appendix B Quality Assurance Programs, or their equivalent, as applicable. Such a program is thus expected to be established with a sufficient level of documentation and administrative controls to ensure effective long-term implementation."

2.3.9.1 ANO-2 Administrative Controls

ANO-2 QA procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR 50, Appendix B which are acceptable in addressing administrative controls.

2.3.9.2 Conclusion

This element is consistent with the corresponding aging management attribute in Revision 2 of NUREG-1801 [10], Chapter XI.M16A and Appendix A (Commitment 19) of the ANO-2 license renewal SER [19].

2.3.10 NUREG-1801/AMP Program Element 10: Operating Experience

> *"Relatively few incidents of PWR internals aging degradation have been reported in operating U.S. commercial PWR plants. A summary of observations to date is provided in Appendix A of MRP-227-A. The applicant is expected to review subsequent operating experience for impact on its program or to participate in industry initiatives that perform this function.*
>
> *The application of MRP-227 guidance will establish a considerable amount of operating experience over the next few years. Section 7 of MRP-227 describes the reporting requirements for these applications, and the plan for evaluating the accumulated additional operating experience."*

2.3.10.1 ANO-2 Operating Experience

Extensive industry and ANO-2 operating experience (OE) has been reviewed during the development of the ANO-2 RVI AMP.

Early plant operating experience related to hot functional testing and RVI is documented in plant historical records. Inspections performed as part of the 10-year ISI program have been conducted as designated by commitments and would be expected to discover general internals structure degradation. To date, little degradation has been observed industry-wide.

Industry and ANO-2 specific information relevant to aging has been compiled into the ANO-2 OE program [30]. Industry operating experience sources in this program include applicable NRC Generic Publications (including Information Notices, Circulars, Bulletins and Generic Letters), NRC Generic Aging Lessons Learned (GALL) Report, etc. Plant specific operating experience sources in the database include applicable maintenance work history, licensee event reports (LERs), corrective action process documents (CAPs, CRs, DRs, ERs), etc.

A review of industry and plant-specific experience with RVI reveals that the U.S. nuclear industry, including ANO-2, has responded proactively to issues relative to RVI degradation. Examples of ANO-2 proactivity are briefly described in the following paragraphs:

- ANO-2 Flux Thimble Tube Replacement

  In the CE-designed plants, zirconium-base alloy thimbles exhibited growth due to irradiation. This thimble growth was a major aging management issue, and the thimbles were subsequently replaced. ANO-2 has monitored the growth of the Zircaloy section of the thimble tube due to the high level of neutron radiation exposure and replaced ICI thimble tubes [32].
- Participation in PWROG OE Activities:

  Entergy participated in a PWROG project, which documented RVI aging degradation OE from the domestic and international PWR plants. The PWROG members were asked about prior inspections and results for the MRP-227-A RVI components. Entergy submitted survey responses detailing previous inspections, specific findings, and inspection timing. The results of this survey are documented in WCAP-17435-NP [33]. The OE in this report provides a benchmark from which to evaluate further RVI aging management events.

Entergy reported the following information for the MRP-227-A RVI components in the survey:

- Number of inspections per component
- Details of each inspection per component
- Record of indications for a given inspection
- Subsequent corrective actions (response) or indications found

- Cognizance of Industry OE:

  ANO-2 is committed to monitoring specific industry OE that could potentially affect the RVI during the period of extended operating at ANO-2 and at other domestic PWR facilities. For example, ANO-2 has monitored the emerging OE from the fuel leakage at a domestic CE designed power plant in fuel assemblies adjacent to the core shroud. Higher radiation levels may increase the susceptibility of stainless steel in the RVI to various material degradation mechanisms. These first burned peripheral fuel assemblies were detected at the specific plant and dispositioned. ANO-2 will incorporate related OE to ensure safe and reliable operation.

  Industry OE published by the Institute of Nuclear Power Operations (INPO) and other informational sources is routinely reviewed, as directed under the applicable procedure for the determination of additional actions and lessons learned. These insights, as applicable, can be incorporated into the plant system health reports and further evaluated for incorporation into the applicable plant programs.

  A key element of the MRP-227-A guideline is the reporting of age-related degradation of RVI components. Entergy, through its participation in EPRI MRP activities, will continue to benefit from the reporting of inspection information and will share its own OE with the industry through those groups or INPO, as appropriate.

2.3.10.2 Conclusion

This element is consistent with the corresponding aging management attribute in Revision 2 of NUREG-1801 [10], Chapter XI.M16A and Appendix A (Commitment 19) of the ANO-2 license renewal SER [19].

3.0 ANO-2 REACTOR VESSEL INTERNALS DESIGN AND OPERATING EXPERIENCE

The ANO-2 RVI are a part of the primary RCS, which is a two-loop CE designed NSSS. The ANO-2 reactor internals are designed to support and position the reactor core fuel assemblies and CEAs, provide hold-down for the fuel assemblies, absorb the dynamic loads and transmit these and other loads to the reactor vessel flange, provide flow paths for the reactor coolant and guide in-core instrumentation. The components of the reactor vessel internals are divided into sub-assemblies consisting of the core support structure, core shroud, flow skirt, upper guide structure assembly, and in-core instrumentation support system. The general arrangement of the ANO-2 reactor vessel internals is shown in Figure 3-1. Schematic representations of specific reactor vessel internals components and assemblies are provided in Figure 3-2 through Figure 3-10. Descriptions of the reactor vessel internals assemblies are obtained from the ANO-2 UFSAR [18, Section 4.2.2].

3.1 Core Support Structure

The major structural member of the reactor internals is the core support structure shown in Figure 3-2 and Figure 3-3. The core support structure consists of the core support barrel and the lower support structure. The material for the assembly is Type 304 stainless steel.

The core support structure is supported at its upper end by the upper flange of the core support barrel, which rests on a ledge in the reactor vessel. Alignment is accomplished by means of four equally spaced keys in the flange which fit into the keyways in the vessel ledge and closure head.

The lower flange of the core support barrel supports, secures and positions the lower support structure and is attached to the lower support structure by means of a welded flexural type connection. The lower support structure provides support for the core by means of a core support plate supported by columns mounted on support beams which transmit the load to the core support barrel lower flange. The core support plate provides support and orientation for the lower ends of the fuel assemblies. The core shroud, which provides a flow path for the coolant and lateral support for the fuel assemblies, is also supported and positioned by the core support plate. The lower end of the core support barrel is restricted from excessive radial and torsional movement by six snubbers which interface with the pressure vessel wall.

A. Core Support Barrel

The core support barrel is a right circular cylinder including a heavy ring flange at the top end and an internal ring flange at the lower end. The core barrel is supported from a ledge on the pressure vessel. The core support barrel, in turn, supports the lower support structure upon which the fuel assemblies rest. Press-fitted into the flange of the core support barrel are four alignment keys located 90 degrees apart. The reactor vessel, closure head and upper guide structure assembly flange are slotted in locations corresponding to the alignment key locations to provide proper alignment between these components in the vessel flange region.

The upper section of the barrel contains two outlet nozzles which interface with internal projections on the vessel nozzles to minimize leakage of coolant from inlet to outlet.

Since the weight of the core support barrel is supported at its upper end, it is possible that coolant flow could induce vibrations in the structure. Therefore, amplitude limiting devices, or snubbers, are installed on the outside of the core support barrel near the bottom end. The snubbers consist of six equally spaced lugs around the circumference of the barrel and act as a tongue-and-groove assembly with the mating lugs on the pressure vessel. Minimizing the clearance between the two mating pieces limits the amplitude of vibration. During assembly, as the internals are lowered into the pressure vessel, the pressure vessel lugs engage the core support barrel lugs in an axial direction. Radial and axial expansion of the core support barrel are accommodated, but lateral movement of the core support barrel is restricted. The pressure vessel lugs have bolted, captured, Inconel X shims and the core support barrel lug mating surfaces are hardfaced with Stellite to minimize wear. The shims are machined during initial installation to provide minimum clearance. The snubber assembly is shown in Figure 3-4.

B. Core Support Plate and Lower Support Structure

The core support plate is a Type 304 stainless steel plate into which the necessary flow distributor holes for the fuel assemblies have been machined. Fuel assembly locating pins are inserted into this plate.

The fuel assemblies and core shroud are positioned on the core support plate. This plate is welded to the top of a cylindrical structure at the base of which is welded a bottom plate. This structure seats on the lower flange of the core support barrel and transmits the lower support structure loads to the core support barrel. The core support plate is supported by an arrangement of columns welded at the base to support beams as shown in Figure 3-3. The bottoms of the beams are welded to a bottom plate which contains flow holes for primary coolant flow. The ends of the beams are welded to the lower cylinder. The cylinder guides the main coolant flow and provides core shroud bypass flow by means of holes in the cylinder.

3.2 Core Shroud Assembly

The core shroud provides an envelope for the core and limits the amounts of coolant bypass flow. The core shroud forms the perimeter of the core and acts as the transition structure between the rectilinear polygon core cross section and the cylindrical core support barrel. The shroud consists of two Type 304 stainless steel ring sections welded to each other and to the core support plate. The ANO-2 core shroud assembly is welded; there are no core shroud bolts.

A small gap is provided between the core shroud outer perimeter and the core support barrel in order to provide upward coolant flow between the core shroud and the core support barrel, thereby minimizing thermal stresses in the core shroud and eliminating stagnant pockets. The

ANO-2 core shroud assembly is shown in Figure 3-5. Examples of the ANO-2 welded core shroud configuration from MRP-227-A [4] are shown in Figure 3-6 and Figure 3-7. Four equally spaced lugs are furnished on the top of the core shroud to provide alignment of the shroud with the fuel alignment plate.

3.3 Flow Skirt

The Inconel flow skirt is a right circular cylinder perforated with flow holes. The flow skirt is used to reduce inequalities in core inlet flow distributions and to prevent formation of large vortices in the lower plenum. The skirt provides a nearly equalized pressure distribution across the bottom of the core support barrel. The skirt is supported by nine equally spaced, machined sections which are welded to the bottom head of the pressure vessel.

3.4 Upper Guide Structure Assembly

This assembly consists of the upper guide structure support plate assembly, CEA shrouds and a fuel assembly alignment plate (Figure 3-8). The upper guide structure assembly aligns and laterally supports the upper end of the fuel assemblies, maintains the CEA spacing, holds down the fuel assemblies during operation, prevents fuel assemblies from being lifted out of position during a severe accident condition, protects the CEAs (Figure 3-9) from the effect of coolant crossflow in the upper plenum and supports the in-core instrumentation plate assembly. The upper guide structure assembly is handled as one unit during installation and refueling.

The upper end of the assembly is a structure consisting of a support flange welded to the top of a cylinder. A support plate is welded to the inside of the cylinder approximately in the middle. The support plate is welded to a grid array of deep beams, the ends of which are welded to the cylinder. The support flange contains four accurately machined and located alignment keyways, equally spaced at 90 degree intervals, which engage the core barrel alignment keys. This system of keys and slots provides an accurate means of aligning the core with the closure head and thereby with the CEA drive mechanisms. The support plate aligns and supports the upper end of the CEA shrouds. The shrouds extend from the fuel assembly alignment plate to an elevation above the upper guide structure support plate. The CEA shroud consists of a cylindrical upper section welded to a base and a flow channel structure shaped to provide flow passage for the coolant through the alignment plate, while isolating the CEAs from crossflow. The shrouds are bolted and lockwelded to the fuel assembly alignment plate. At the upper guide structure support plate, the shrouds are connected to the plate by spanner nuts. The spanner nuts are tightened to proper torque to assure a rigid connection and lockwelded.

The fuel assembly alignment plate is designed to align the upper ends of the fuel assemblies and to support and align the lower ends of the CEA shrouds. Precision machined and located holes in the fuel assembly alignment plate engage machined posts on fuel assembly upper end fittings to provide accurate alignment. The fuel assembly alignment plate also has four equally spaced slots on its outer edge which engage with Stellite hardfaced pins protruding from the core shroud to limit lateral motion of the upper guide structure assembly during operation. The fuel alignment plate bears the upward force of the fuel assembly holddown devices. This force is transmitted from the alignment plate through the CEA shrouds to the upper guide structure support plate. The flange of the upper guide structure support plate is designed to resist axial upward movement of the upper guide structure assembly and to accommodate axial differential thermal expansion between the core barrel flange, upper guide structure and pressure vessel flange support ledge and head flange recess.

3.5 In-Core Instrumentation Support System

The complete in-core neutron flux monitoring system includes self-powered in-core detector assemblies, supporting structures and guide paths, and an amplifier system to process detector signals. The instrumentation supporting structures and guide paths are described in this section.

The support system begins outside the pressure vessel, penetrates the vessel boundary and terminates at the lower end of the fuel assembly. Each instrument is guided over its full length by the external guidance conduit, the instrument plate structure guide tubes and the thimbles that extend downward into selected fuel bundles. The in-core instrumentation guide tubes route the instruments so that the detectors are located and spaced throughout the core. The guide tubes and the in-core thimbles are attached to and supported by the instrument plate assembly shown in Figure 3-10.

The instrumentation plate assembly fits within the confines of the reactor vessel head and rests in the recessed section of the upper guide structure assembly. Its weight is supported by four bearing pins. The upper guide structure CEA shrouds extend through the instrumentation plate clearance holes. Above the instrumentation plate, the guide tubes bend and are gathered to form stalks which extend into the reactor vessel head instrumentation nozzles. The instrumentation plate assembly is raised and lowered during refueling to insert or withdraw all instruments and their thimbles simultaneously. The pressure boundaries for the individual instruments are at the instrumentation nozzle flange where the external electrical connections to the in-core instruments are also made.

The supporting structures for the in-core instruments are designed such that the temperature of the coolant surrounding the thermocouples in the in-core instruments is representative of fuel assembly outlet temperatures. The in-core instrument lengths and thimbles are designed to locate individual neutron detectors within a tolerance of ±2 inches.

The assemblies have an integral seal plug which forms a seal at the instrument flange and through which the signal cables pass. Carbon packing rings fitted in a recess in the instrument flange are used to seal against operating pressure.

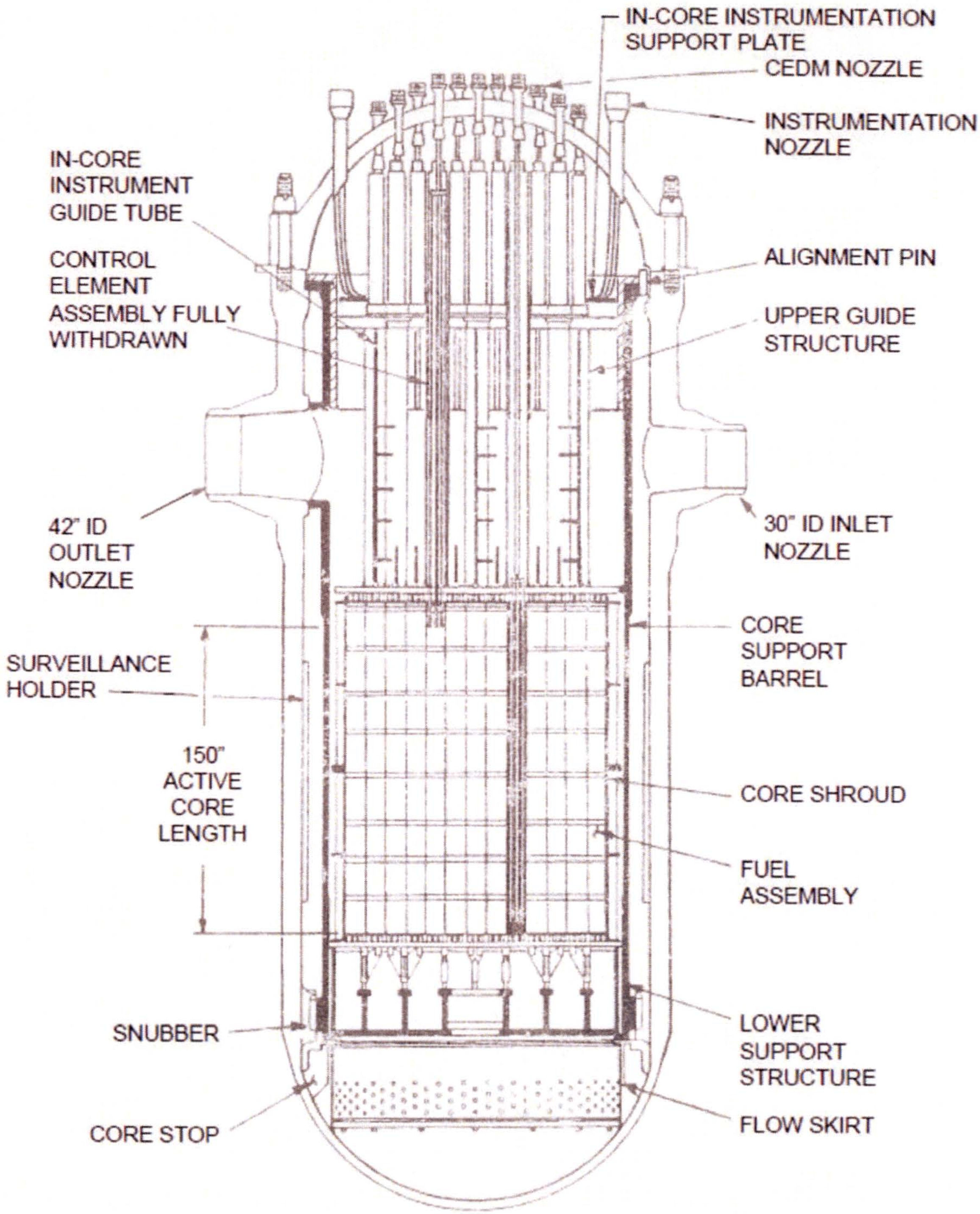

Figure 3-1. Illustration of the ANO-2 Vessel and Internals [18, Figure 4.1-1]

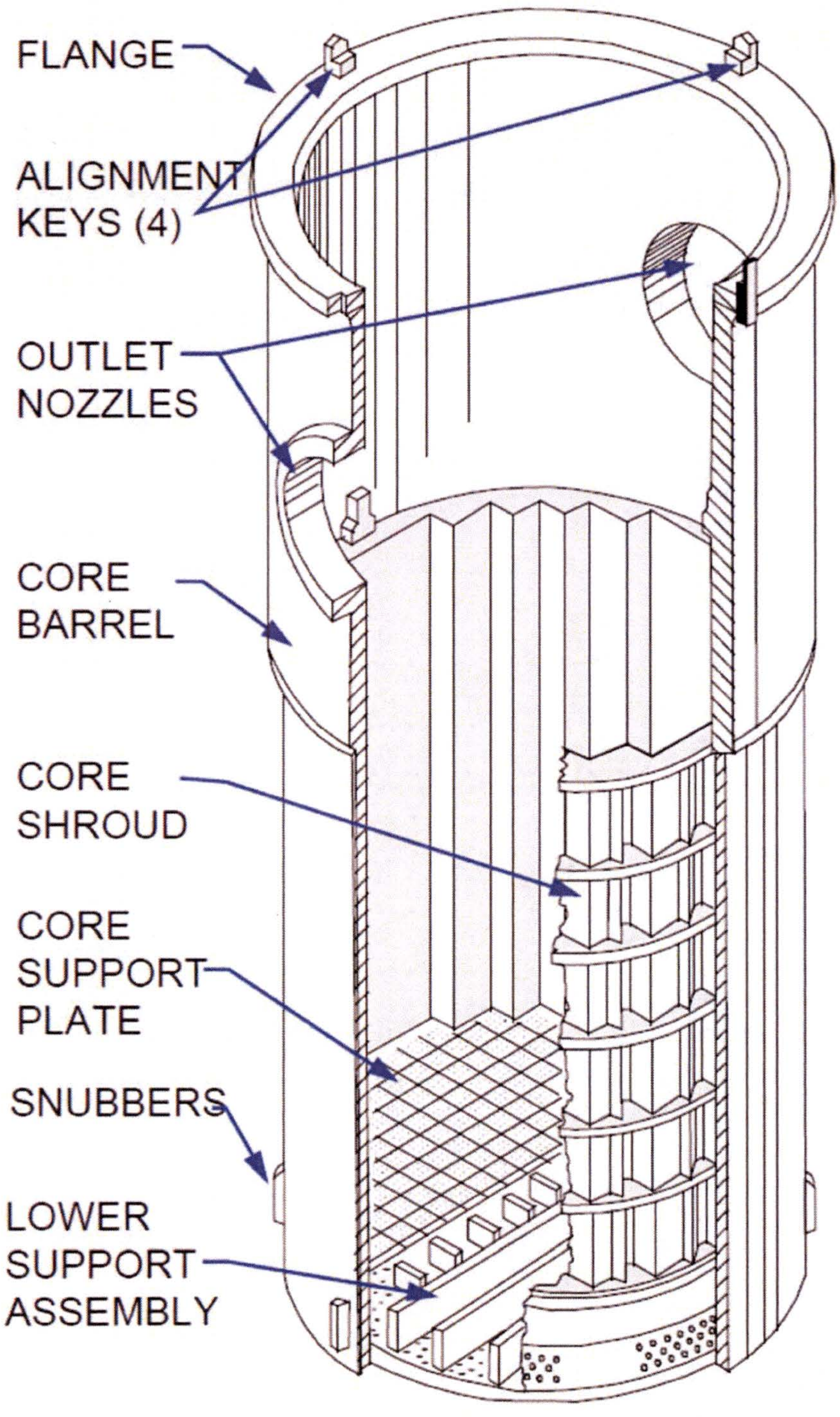

Figure 3-2. ANO-2 Core Support Barrel Assembly [12, Figure 4.1-3]

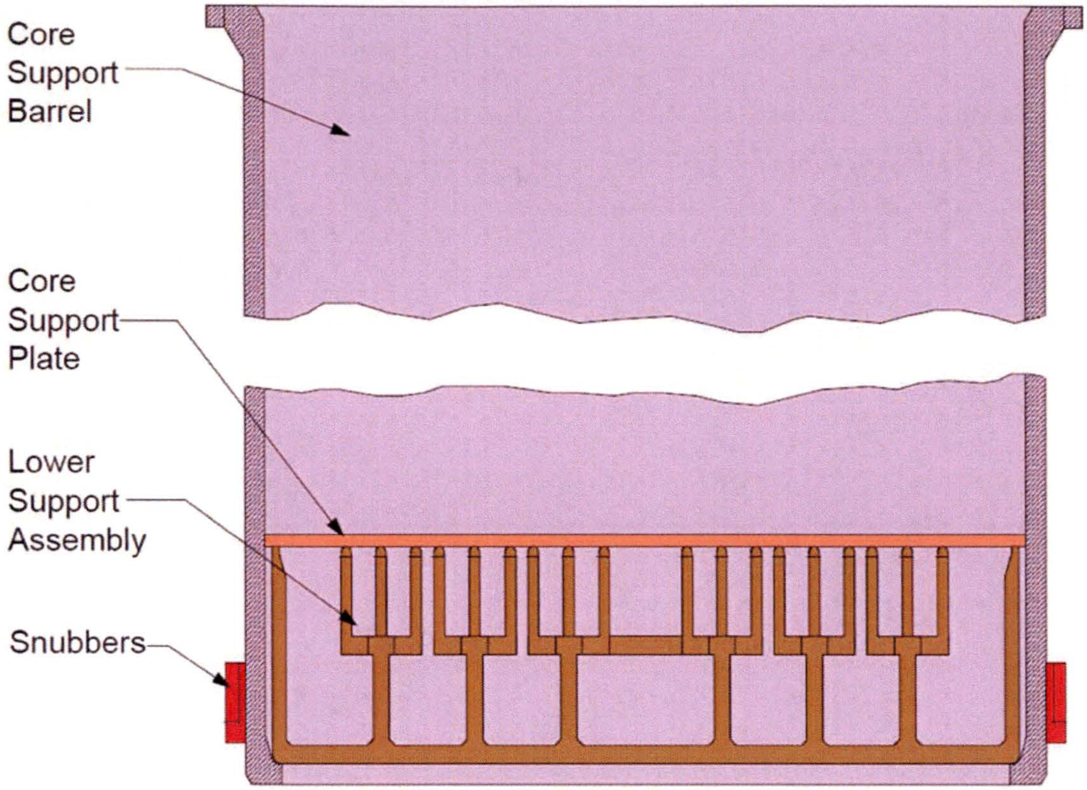

Figure 3-3. ANO-2 Lower Support Structure [12, Figure 4.1-5]

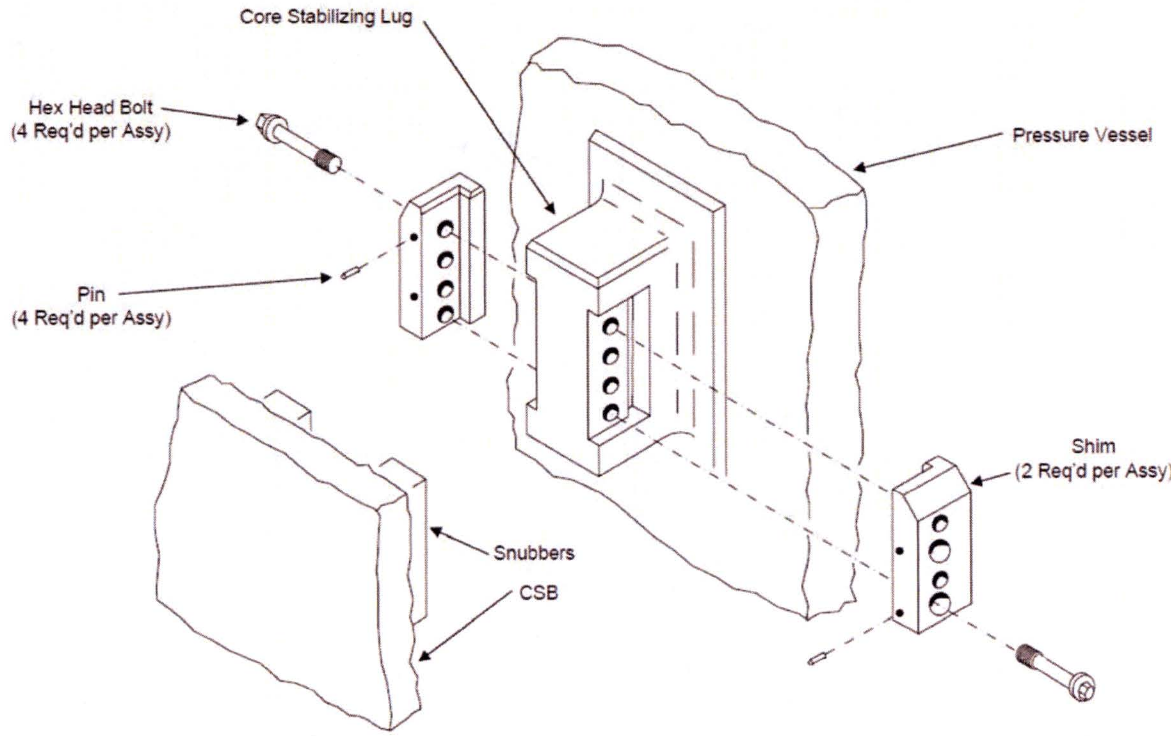

Figure 3-4. ANO-2 Snubber Assembly [18, 4.2-10]

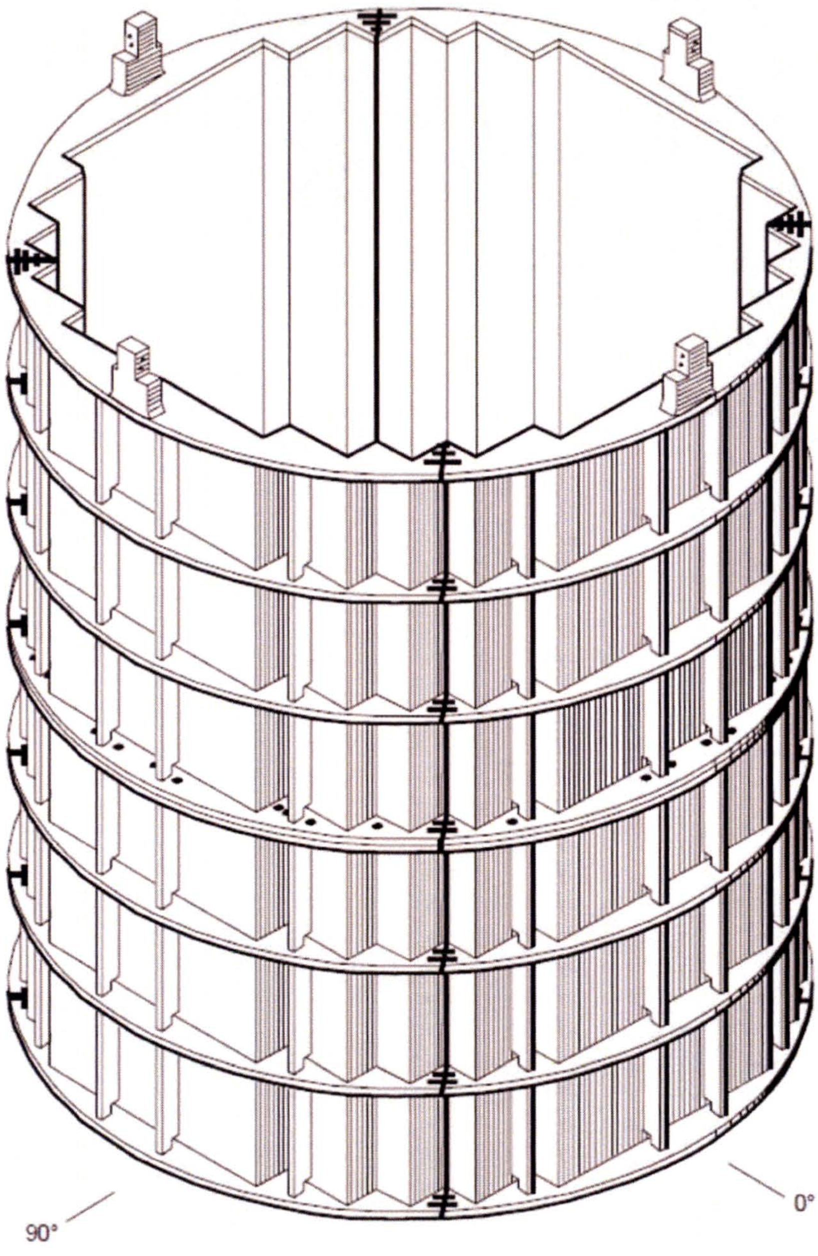

Figure 3-5. ANO-2 Core Shroud Assembly [18, Figure 4.2-11]

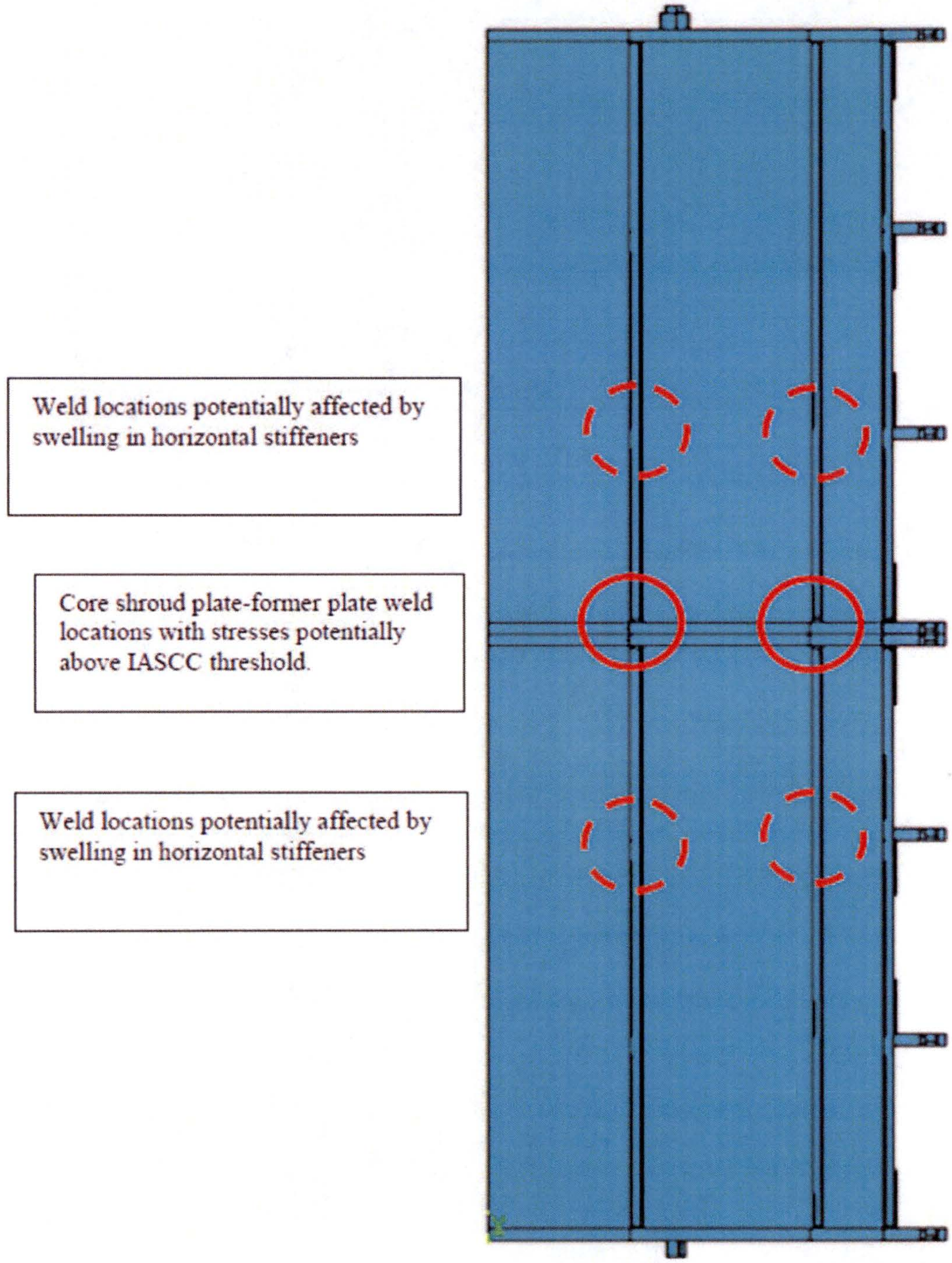

Figure 3-6. Potential Crack Locations for CE Welded Core Shroud Assembled in Stacked Sections [4, Figure 4-12]

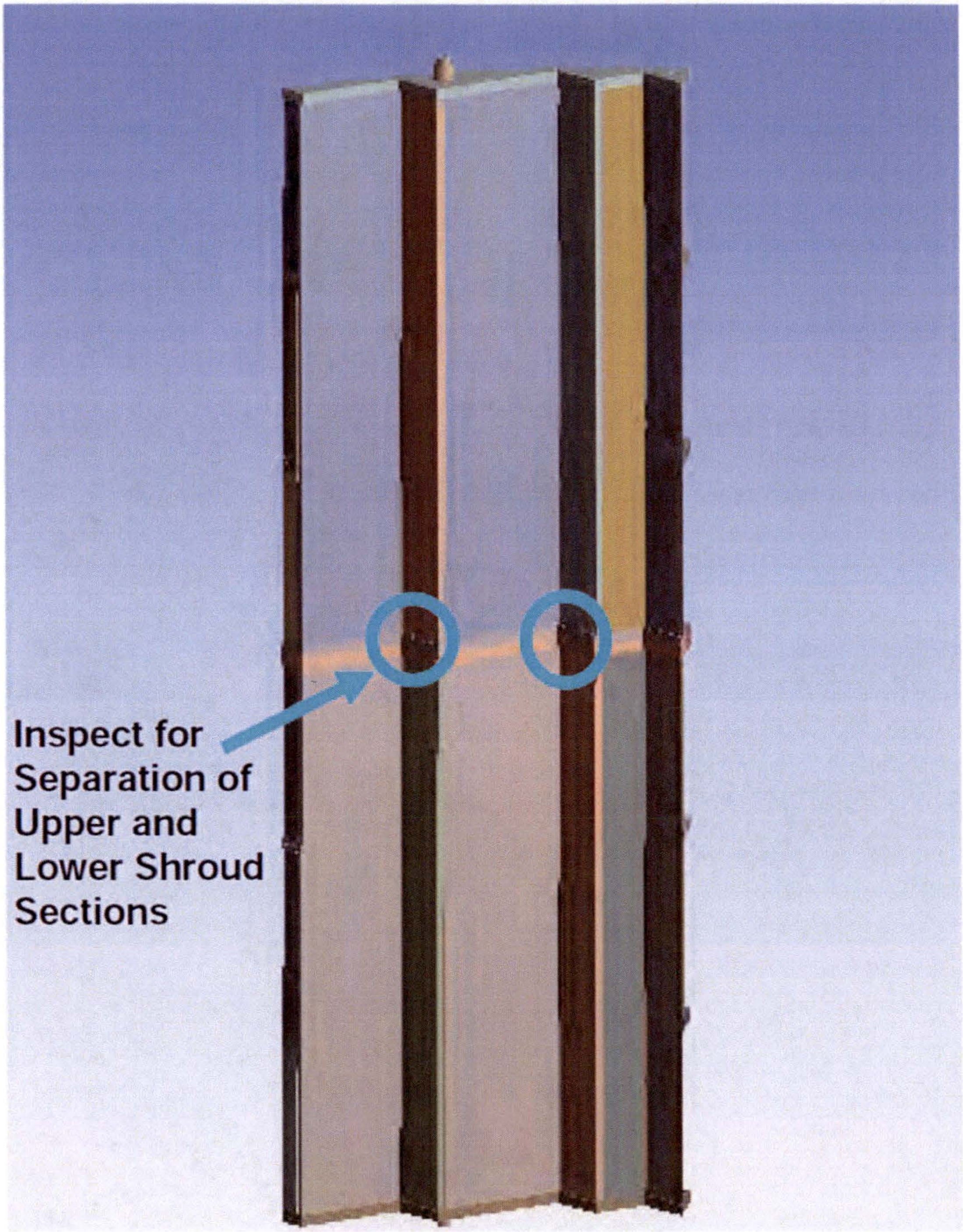

Figure 3-7. Locations of Potential Separation Between Core Shroud Sections Caused by Swelling Induced Warping of Thick Flange Plates in CE Welded Core Shroud Assembled in Stacked Sections [4, Figure 4-14]

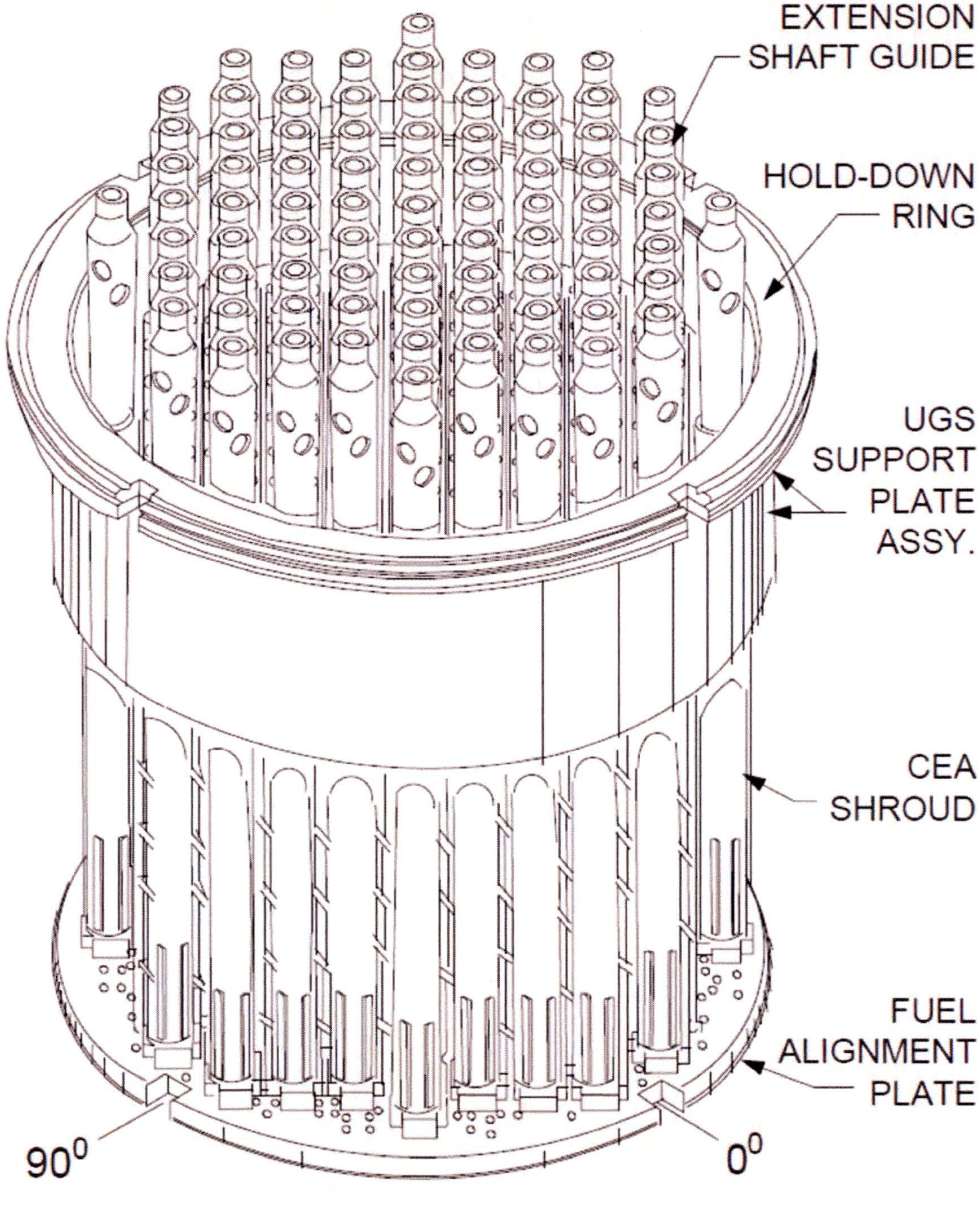

Figure 3-8. ANO-2 Upper Guide Structure Assembly [12, Figure 4.1-2]

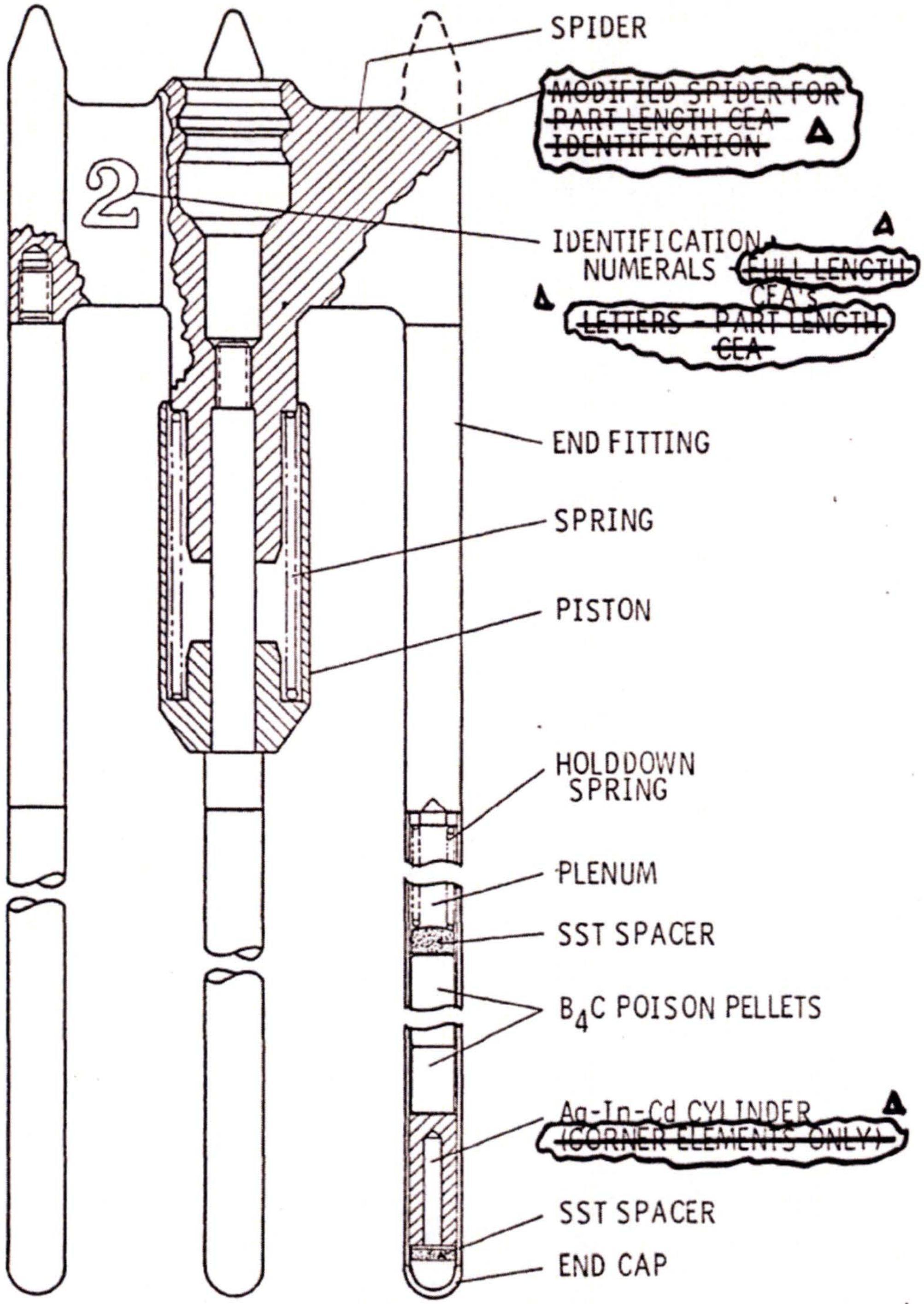

Figure 3-9. Control Element Assembly [31, Figure 42]

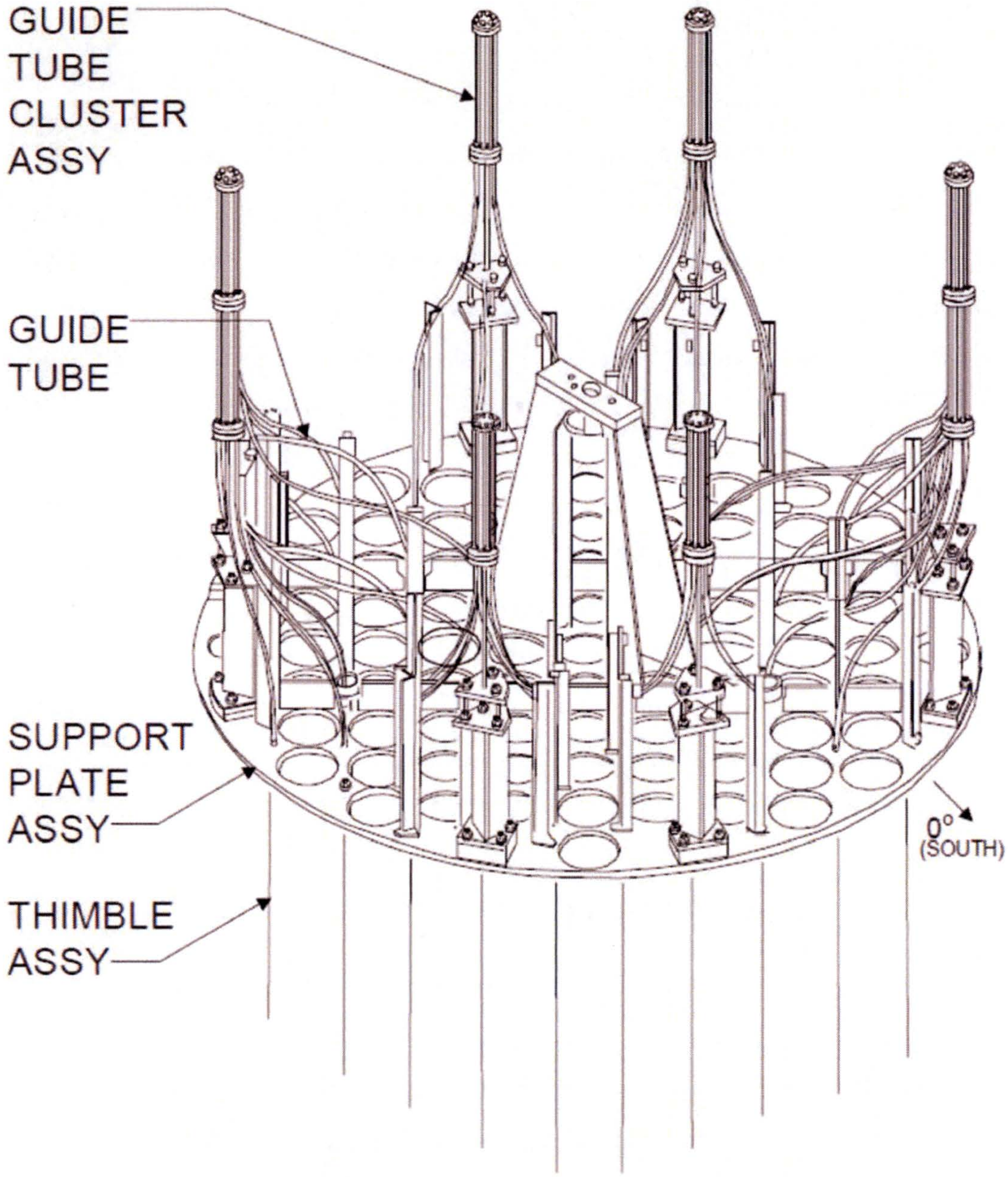

Figure 3-10. In-Core Support Assembly [12, Figure 4.1-8a]

3.6 ANO-2 Design Distinctions

Per Section 4.1 of the ANO-2 UFSAR [18]:

> *Arkansas Nuclear One - Unit 2 incorporates a Pressurized Water Reactor (PWR) with two reactor coolant loops. A vertical cross section of the reactor is shown in Figure 4.1-1. The reactor core is composed of 177 fuel assemblies and 81 Control Element Assemblies (CEAs). The fuel assemblies are arranged to approximate a right circular cylinder with an equivalent diameter of 123 inches and an active length of 150.0 inches for fuel Batches A through H and 149.610 inches for fuel Batches J through N. The active fuel length for Batch P (Cycle 12 reload batch) and subsequent reload batches is 150 inches. Each fuel rod shall contain a maximum total weight of 2114 grams of uranium. The fuel assembly, which provides for 236 fuel rod positions, consists of five guide tubes welded or bulged to spacer grids and is closed at the top and bottom by end fittings. The welded construction was used for fuel Batches A through Y. A bulged construction was introduced in Batch Z with the implementation of Next Generation Fuel (NGF). The guide tubes each displace four fuel rod positions and provide channels which guide the CEAs over their entire length of travel. In selected fuel assemblies, the central guide tube houses incore instrumentation.*

3.7 ANO-2 Unit Operating Experience

In the CE-designed plants, zirconium-base alloy thimbles exhibited growth due to irradiation. This thimble growth was a major aging management issue, and the thimbles were subsequently replaced. ANO-2 has monitored the growth of the Zircaloy section of the thimble tube due to the high level of neutron radiation exposure and replaced ICI thimble tubes [32].

4.0 EXAMINATION AND ACCEPTANCE AND EXPANSION CRITERIA

4.1 Examination Acceptance Criteria

4.1.1 Visual (VT-3) Examination

Visual (VT-3) examination has been determined to be an appropriate NDE method for the detection of general degradation conditions in many of the susceptible components. The ASME Code Section XI, Examination Category B-N-3 [9], provides a set of relevant conditions for the visual (VT-3) examination of removable core support structures in IWB-3520.2. These are:

1. Structural distortion or displacement of parts to the extent that component function may be impaired
2. Loose, missing, cracked, or fractured parts, bolting, or fasteners
3. Corrosion or erosion that reduces the nominal section thickness by more than 5%
4. Foreign materials or accumulation of corrosion products that could interfere with control rod motion or could result in blockage of coolant flow through fuel
5. Wear of mating surface that may lead to loss of functionality
6. Structural degradation of interior attachments such that the original cross-sectional area is reduced more than 5%

For components in the Existing Programs group, these general relevant conditions are sufficient. However, for components where visual (VT-3) is specified in the Primary or the Expansion group, more specific descriptions of the relevant conditions are provided in Table 5-1 through Table 5-4 of this document. Typical examples are "fractured material" and "completely separated material." One or more of these specific relevant condition descriptions may be applicable to the Primary and Expansion components listed in Table 5-1 and Table 5-2 of this document. The examination acceptance criteria for components requiring visual (VT-3) examinations is thus the absence of the relevant condition(s) specified in Table 5-4 of this document. The disposition can include a supplementary examination to further characterize the relevant condition, an engineering evaluation to show that the component is capable of continued operation with a known relevant condition, or repair/replacement to remediate the relevant condition.

Relevant conditions are defined in ASME Section XI, IWA-9000 [9]; they do not include fabrication marks, material roughness, and other conditions acceptable by material design, and manufacturing specifications of the component.

4.1.2 Visual (VT-1) Examination

Visual (VT-1) examination is defined in the ASME Code Section XI as an examination "conducted to detect discontinuities and imperfections on the surface of components, including such conditions as cracks, wear, corrosion, or erosion." For these guidelines VT-1 has only been selected to detect distortions as evidenced by small gaps between the upper-to-lower mating surfaces of CE welded core shroud assembled in two vertical sections. The examination acceptance criterion is thus the absence of the relevant condition of gaps that would be indicative of distortion from void swelling.

4.1.3 Enhanced Visual (EVT-1) Examination

Enhanced visual (EVT-1) examination has the same requirements as the ASME Code Section XI visual (VT-1) examination, with additional requirements given in MRP-228 [14]. 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 sizing surface crack length when used in conjunction with sizing aides (e.g. landmarks, ruler, and tape measure). EVT-1 examination is 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 for cracking in Section XI which is crack-like surface-breaking indications. The examination acceptance criterion for EVT-1 examination is the absence of any detectable surface-breaking indication.

4.1.4 Surface Examination

Surface ET (eddy current) examinations are specified as an alternative or as a supplement to visual examinations. No specific acceptance criteria for surface ET examination of PWR internals locations are provided in the ASME Code Section XI. Since surface ET is employed as a signal-based examination, a technical justification is documented in MRP-228 [14]. MRP-228 provides the basis for detection and length sizing of surface-breaking or near-surface cracks. The signal-based relevant indication for surface ET is thus the same as the relevant condition for enhanced visual (EVT-1) examination. The acceptance criteria for enhanced visual (EVT-1) examinations are therefore applied when this method is used as an alternative or supplement to visual examination.

4.1.5 Volumetric Examination

The intent of volumetric examinations specified for bolts and pins is to detect planar defects. No flaw sizing measurements are recorded or assumed in the acceptance or rejection of individual bolts or pins. Individual bolts or pins are accepted based on the lack of detection of any relevant indications established as part of the examination technical justification. When a relevant indication is detected in the cross-sectional area of the bolt or pin, that bolt or pin is assumed to be non-functional and the indication is recorded. A bolt or pin that passes the criterion of the examination is assumed to be functional.

Because there are no baffle-former bolts in the ANO-2 design, no volumetric examinations of the internals are needed to meet MRP-227-A requirements.

4.1.6 Physical Measurements Examination

Continued functionality can be confirmed by physical measurements where, for example, loss of material caused by wear, loss of pre-load of clamping force caused by various degradation mechanisms, or distortion/deflection caused by void swelling may occur. For CE designs, no physical measurements are specified; however, ANO-2 has a core barrel shroud assembled in two vertical sections. If gaps between the two vertical sections of the core barrel shroud are identified during the required VT-1 examination required by MRP-227-A (see Table 5-1), physical measurements must be performed for distortion in the gap between the top and bottom core shroud segments. See Section 5.5 for more details.

4.2 Expansion Criteria

The criteria for expanding the scope of examination from the Primary components to their linked Expansion components are contained in Table 5-4.

4.3 Evaluation, Repair, and Replacement Strategy

Any condition detected during examinations that do not satisfy the examination acceptance criteria of Section 4.1 shall be entered and dispositioned in the corrective action program.

The options listed below will be considered for disposition of such conditions. Selection of the most appropriate option(s) will be dependent on the nature and location of the indication detected.

1. Supplemental examinations will be used in order to further characterize and disposition a detected condition
2. Engineering evaluations that demonstrate the acceptability of detected conditions
3. Repair to restore a component with a detected condition to acceptable status
4. Replacement of a component

The methodology used to perform engineering evaluations to determine the acceptability of a detected condition (item 2 above) shall be conducted in accordance with an NRC approved evaluation methodology. WCAP-17096-NP [16] and other NRC approved methodologies will be used to provide acceptance criteria for Primary and Expansion category items.

4.3.1 Reporting

Reporting and documentation of relevant conditions and disposition of indications that do not meet the examination acceptance criteria will be performed consistent with MRP-227-A and the ANO-2 Corrective Action Program. Entergy shall provide a summary report to the EPRI MRP Program Manager of all inspections and monitoring, items requiring evaluation, and new repairs. This report shall be provided within 120 days of the completion of the outage during which the activities occur. This is part of the "Needed" requirement 7.6 under MRP-227-A. Inspection results having potential industry significance shall be expeditiously reported to the RCS Materials Degradation Program Manager for consideration of reporting under the NEI 03-08, Materials Initiative Protocol [3].

4.4 Implementation Schedule

The Program Enhancement and Implementation Schedule for ANO-2 is provided in Table 5-6.

4.5 Commitment Tracking

A summary of actions related to the Aging Management of Reactor Vessel Internals for ANO-2 is provided in Table 5-7.

5.0 RESPONSES TO NRC SAFETY EVALUATION APPLICANT/LICENSEE ACTION ITEMS

As part of the NRC Revision 1 of the Final Safety Evaluation of MRP-227 [5], a number of action items and conditions were specified by the staff. Table 5-5 documents ANO-2's conformance to the Topical Report Conditions and the Applicant/Licensee Action Items in the NRC Safety Evaluation of MRP-227 [5]. Wherever possible, these items have been addressed in the appropriate sections of this document. All NRC action items and conditions not addressed elsewhere in this document are discussed in this section.

5.1 SE Section 4.2.1, Applicant/Licensee Action Item 1 (Applicability of FMECA and Functionality Analysis Assumptions):

As addressed in Section 3.2.5.1 of this SE, each applicant/licensee is responsible for assessing its plant's design and operating history and demonstrating that the approved version of MRP-227 is applicable to the facility. Each applicant/licensee shall refer, in particular, to the assumptions regarding plant design and operating history made in the FMECA and functionality analyses for reactors of their design (i.e., Westinghouse, CE or B&W) which support MRP-227 and describe the process used for determining plant-specific differences in the design of their RVI components or plant operating conditions, which result in different component inspection categories. The applicant/licensee shall submit this evaluation for NRC review and approval as part of its application to implement the approved version of MRP-227.

The assumptions regarding plant design and operating history made in MRP-191 [13] are appropriate for ANO-2. The FMECA and functionality analyses were based on the assumption of 30 years of operation with high leakage core loading patterns followed by 30 years of low-leakage core fuel management strategy [27, Section 3.0]; therefore, ANO-2 is bounded by the assumption in MRP-191 [13].

As discussed in Section 1.8.4.1 of this document, operations at ANO-2 conform to the assumptions in Section 2.4 of MRP-227-A [4].

- ANO-2 historic core management practices [27] meet the requirements of MRP-227-A [4]
- ANO-2 operates as a base load unit [18, Section 10.2.1]
- No design changes were implemented beyond those identified in general industry guidance or recommended by the vendor (CE or Westinghouse)

ANO-2 is actively participating in a joint industry program under the PWROG aimed at addressing the 20% cold work issue for non-weld or bolting austenitic stainless steel components on a generic rather than plant-specific basis. A discussion of this ongoing program (PA) follows:

PA-MSC-1288, PWR Materials Assessment, was discussed with the NRC at the June 2-4, 2015 Annual Materials Programs Technical Information Exchange Public Meeting (Ref. ML15155B431). This PA utilizes a statistical approach for determining and assessing material or fabrication factors for PWR internals components. To date, plant-specific component manufacturing records have been gathered for over 50% of the domestic PWRs. A review of these records in accordance with the guidance provided in MRP 2013-025 (ML1322A454) has revealed the following:

- 20% cold work limitation was already recognized at the time of plant construction, i.e. from 1970's
- Plant fabricators quality programs were in place to adhere to limitations in cold work in austenitic stainless steels in these times
- Plant specific assessments conducted to date confirm that no non-fastener materials contain cold work greater than 20%
- Correlation of data based on searches to date demonstrates consistency across the PWR fleet - B&W, CE and, W show no cold worked non-fastener materials used in reactor vessel internals

A final report (PWROG-15105-NP [34]) for this PA was issued in April 2016. ANO-2 will continue to participate in and follow the progress of PA-MSC-1288, including interactions with the NRC. If necessary, plant specific information will be provided to the NRC to supplement this joint industry program, if requested.

5.2 SE Section 4.2.2, Applicant/Licensee Action Item 2 (PWR Vessel Internal Components Within the Scope of License Renewal):

As discussed in Section 3.2.5.2 of this SE, consistent with the requirements addressed in 10 CFR 54.4, each applicant/licensee is responsible for identifying which RVI components are within the scope of LR for its facility. Applicants/licensees shall review the information in Tables 4-1 and 4-2 in MRP-189, Revision 1, and Tables 4-4 and 4-5 in MRP-191 and identify whether these tables contain all of the RVI components that are within the scope of the LR for their facilities in accordance with 10 CFR 54.4. If the tables do not identify all the RVI components that are within the scope of LR for its facility, the applicant or licensee shall identify the missing component(s) and propose necessary modifications to the program defined in MRP-227, as modified by this SE, when submitting its plant-specific AMP. The AMP shall provide assurance that the effects of aging on the missing component(s) will be managed for the period of extended operation.

The information contained in Table 4-5 of MRP-191 [13] was reviewed and that review determined that this table contained all of the RVI components that are within the scope of license renewal for ANO-2. The aging management review performed as part of the ANO-2 LRA is described in Section 1.7.1 and summarized in Table 3.1.2-2 of the LRA.

5.3 SE Section 4.2.3, Applicant/Licensee Action Item 3 (Evaluation of the Adequacy of Plant-Specific Existing Programs):

As addressed in Section 3.2.5.3 in this SE, applicants/licensees of CE and Westinghouse are required to perform plant-specific analysis either to justify the acceptability of an applicant's/licensee's existing programs, or to identify changes to programs that should be implemented to manage the aging of these components for the period of extended operation. The results of this plant-specific analyses and a description of the plant-specific programs being relied on to manage aging of these components shall be submitted as part of the applicant's/licensee's AMP application. The CE and Westinghouse components identified for this type of plant-specific evaluation include: CE thermal shield positioning pins and CE in-core *instrumentation thimble tubes (Section 4.3.2 in MRP-227), and Westinghouse guide tube support pins (split pins) (Section 4.3.3 in MRP-227).*

The SE for MRP-227 [5] requires CE plants to evaluate whether existing plant-specific programs are adequate to manage the aging effects of thermal shield positioning pins and in-core instrument thimble tubes. ANO-2 complies with Applicant/Licensee Action 3 through management and replacement of in-core instrumentation thimble tubes as described in [32]. Thermal shields are not present in the ANO-2 reactor vessel internals.

5.4 SE Section 4.2.4, Applicant/Licensee Action Item 4 (B&W Core Support Structure Upper Flange Stress Relief):

As discussed in Section 3.2.5.4 of this SE, the B&W applicants/licensees shall confirm that the core support structure upper flange weld was stress relieved during the original fabrication of Reactor Pressure Vessel in order to confirm the applicability of MRP-227, as approved by the NRC, to their facility. If the upper flange weld has not been stress relieved, then this component shall be inspected as a "Primary" inspection category component. If necessary, the examination methods and frequency for non-stress relieved B&W core support structure upper flange welds shall be consistent with the recommendations in MRP-227, as approved by the NRC, for the Westinghouse and CE upper core support barrel welds. The examination coverage for this B&W flange weld shall conform to the staff's imposed criteria as described in Section 3.3.1 and 4.3.1 of this SE. The applicant's/licensee's resolution of this plant-specific action item shall be submitted to the NRC for review and approval.

This action does not apply to CE designed units.

5.5 SE Section 4.2.5, Applicant/Licensee Action Item 5 (Application of Physical Measurements as part of I&E Guidelines for B&W, CE, and Westinghouse RVI Components):

As addressed in Section 3.3.5 in this SE, applicants/licensees shall identify plant-specific acceptance criteria to be applied when performing the physical measurements required by the NRC-approved version of MRP-227 for loss of compressibility for Westinghouse hold down springs, and for distortion in the gap between the top and bottom core shroud segments in CE units with core barrel shrouds assembled in two vertical sections. The applicant/licensee shall include its proposed acceptance criteria and an explanation of how the proposed acceptance criteria are consistent with the plants' licensing basis and the need to maintain the functionality of the component being inspected under all licensing basis conditions of operation during the period of extended operation as part of their submittal to apply the approved version of MRP-227.

ANO-2 has a core barrel shroud assembled in two vertical sections. Per the examination coverage criteria defined in Table 4-2 of MRP-227-A (see Table 5-1), if a gap between the two core barrel shroud vertical sections are identified during the required VT-1 examination, three to five measurements of the gap opening from the core side at the core shroud re-entrant corners is required and an evaluation of the gap shall be performed to determine the frequency and method for additional examinations. Prior to performing the VT-1 examination ANO-2 will develop acceptance criteria that are consistent with the licensing basis to ensure that the core shroud remains capable of performing its required functions.

5.6 SE Section 4.2.6, Applicant/Licensee Action Item 6 (Evaluation of Inaccessible B&W Components):

As addressed in Section 3.3.6 in this SE, MRP-227 does not propose to inspect the following inaccessible components: the B&W core barrel cylinders (including vertical and circumferential seam welds), B&W former plates, B&W external baffle-to-baffle bolts and their locking devices, *B&W core barrel-to-former bolts and their locking devices, and B&W core barrel assembly internal baffle-to-baffle bolts. The MRP also identified that although the B&W core barrel assembly internal baffle-to-baffle bolts are accessible, the bolts are non-inspectable using currently available examination techniques.*

Applicants/licensees shall justify the acceptability of these components for continued operation through the period of extended operation by performing an evaluation, or by proposing a scheduled replacement of the components. As part of their application to implement the approved version of MRP-227, applicants/licensees shall provide their justification for the continued operability of each of the inaccessible components and, if necessary, provide their plan for the replacement of the components for NRC review and approval.

This action does not apply to CE designed units.

5.7 SE Section 4.2.7, Applicant/Licensee Action Item 7 (Plant-Specific Evaluation of CASS Materials):

As discussed in Section 3.3.7 of this SE, the applicants/licensees of B&W, CE, and Westinghouse reactors are required to develop plant-specific analyses to be applied for their facilities to demonstrate that B&W IMI guide tube assembly spiders and CRGT spacer castings, CE lower support columns, and Westinghouse lower support column bodies will maintain their functionality during the period of extended operation or for additional RVI components that may be fabricated from CASS, martensitic stainless steel or precipitation hardened stainless steel materials. These analyses shall also consider the possible loss of fracture toughness in these components due to thermal and irradiation embrittlement, and may also need to consider limitations on accessibility for inspection and the resolution/sensitivity of the inspection techniques. The requirement may not apply to components that were previously evaluated as not requiring aging management during development of MRP-227. That is, the requirement would apply to components fabricated from susceptible materials for which an individual licensee has determined aging management is required, for example during their review performed in accordance with Applicant/Licensee Action Item 2. The plant-specific analysis shall be *consistent with the plant's licensing basis and the need to maintain the functionality of the components being evaluated under all licensing basis conditions of operation. The applicant/licensee shall include the plant-specific analysis as part of their submittal to apply the approved version of MRP-227.*

The SE for MRP-227 [5] requires the applicants/licensees of CE reactors to develop plant-specific analyses to be applied for their facilities to demonstrate that the CE lower support columns will maintain their functionality during the period of extended operation. It also requires that the licensee provide technical justification that other RVI components that may be fabricated from CASS, martensitic stainless steel, or precipitation hardened stainless steel materials will maintain their functionality during the period of extended operation. ANO-2 does not have lower support columns fabricated from CASS, martensitic stainless steel, or precipitation hardened stainless steel as part of the reactor vessel internals. The ANO-2 lower support columns are 304 stainless steel.

The CEA shroud tube is the only component fabricated from CASS material listed in Table 3.1.2-2 of LRA [1], which is in Category A by MRP-191 [13]. This component item initially screened in for SCC (welds) and TE but the FMECA determined that these age-related degradation mechanisms have minimal likelihood to cause failure. Thus, this component was assigned to Category A per MRP-191. The CEA shroud tubes are considered a no additional measures component per MRP-227-A and the existing inservice inspection program [2] is adequate to manage this CASS RVI component during the period of extended operation. The CEA shroud tubes are not considered a primary or expansion component for CE plants in Table 4-2 and Table 4-5 of MRP-227-A [4].

5.8 SE Section 4.2.8, Applicant/Licensee Action Item 8 (Submittal of Information for Staff Review and Approval):

As addressed in Section 3.5.1 in this SE, applicants/licensee shall make a submittal for NRC review and approval to credit their implementation of MRP-227, as amended by this SE, as an

AMP for the RVI Components at their facility. This submittal shall include the information identified in Section 3.5.1 of this SE.

Section 3.5.1 of SE (Submittal of Information for Staff Review and Approval):

In addition to the implementation of MRP-227 in accordance with NEI 03-08, applicants/licensees whose licensing basis contains a commitment to submit a PWR RVI AMP and/or inspection program shall also make a submittal for NRC review and approval to credit their implementation of MRP-227, as amended by this SE. An applicant's/licensee's application to implement MRP-227, as amended by this SE shall include the following items (1) and (2). Applicants who submit applications for LR after issuance of this SE shall, in accordance with the NUREG-1801, Revision 2, submit the information provided in the following items (1) through (5) for staff review and approval.

1. *An AMP for the facility that addresses the 10 program elements as defined in NUREG-1801, Revision 2, AMP XI.M16A.*

The attributes of the ANO-2 RVI AMP and their compliance with the ten elements of NUREG-1801 (GALL Report), Revision 2, Chapter XI.M16A, "PWR Vessel Internals" [10] that are essential for successful management of component aging are described in Section 2.3 of this document.

2. *To ensure the MRP-227 program and plant-specific action items will be carried out by applicants/licensees, applicants/licensees are to submit an inspection plan which addresses the identified plant-specific action items for staff review and approval consistent with the licensing basis for the plant. If an applicant/licensee plans to implement an AMP which deviates from the guidance provided in MRP-227, as approved by the NRC, the applicant/licensee shall identify where their program deviates from the recommendations of MRP-227, as approved by the NRC, and shall provide a justification*

for any deviation which includes a consideration of how the deviation affects both "Primary" and "Expansion" inspection category components.

The aging management program plan for the ANO-2 will not deviate from the recommendations of MRP-227-A. Inspection of Primary, Expansion, and components credited as part of plant specific existing programs provided in Table 5-1 through Table 5-4 of this document will be performed in accordance with the requirements of MRP-227-A.

ANO-2 qualifies as a Category B plant according to the NRC Regulatory Issue Summary (RIS) [25]. This AMP fulfills the license renewal commitment to submit a description of this program, including the inspection plan, to the NRC for review and approval.

Table 5-1. CE Plants Primary Category Components from Table 4-2 of MRP-227-A [4]

Item	Applicability	Effect (Mechanism)	Expansion Link(1)	Examination Method/Frequency(1)	Examination Coverage	Comments
Core Shroud Assembly (Bolted) Core shroud bolts (Not applicable for ANO-2)	Bolted plant designs	Cracking (IASCC, Fatigue) Aging Management (IE and ISR) (2)	Core support column bolts, barrel-shroud bolts	Baseline volumetric (UT) examination between 25 and 35 EFPY, with subsequent examination on a ten-year interval.	100% of accessible bolts(3). Heads are accessible from the core side. UT accessibility may be affected by complexity of head and locking device designs. See Figure 4-24 of MRP-227-A	N/A
Core Shroud Assembly (Welded) Core shroud plate-former plate weld	Plant designs with core shrouds assembled in two vertical sections	Cracking (IASCC) Aging Management (IE) (2)	Remaining axial welds	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.	Axial and horizontal weld seams at the core shroud re-entrant corners as visible from the core side of the shroud, within six inches of central flange and horizontal stiffeners. See Figure 3-6 and Figure 3-7 (Figures 4-12 and 4-14 of MRP-227-A)	ANO-2 must perform the EVT-1 initial augmented inspection by refueling outage 27
Core Shroud Assembly (Welded) Shroud plates (Not applicable for ANO-2)	Plant designs with core shrouds assembled with full-height shroud plates.	Cracking (IASCC) Aging Management (IE) (2)	Remaining axial welds, ribs and rings	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.	Axial weld seams at the core shroud re-entrant corners, at the core mid-plane (± three feet in height) as visible from the core side of the shroud. See Figure 4-13 of MRP-227-A	N/A
Core Shroud Assembly (Bolted) Assembly (Not Applicable for ANO-2)	Bolted plant designs	Distortion (Void Swelling), including: • Abnormal interaction with fuel assemblies • Gaps along high fluence shroud plate joints • Vertical displacement of shroud plates near high fluence joint Aging Management (IE)	None	Visual (VT-3) examination no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	Core side surfaces as indicated. See Figures 4-25 and 4-26 of MRP-227-A	N/A

Table 5-1. CE Plants Primary Category Components from Table 4-2 of MRP-227-A [4] (continued)

Item	Applicability	Effect (Mechanism)	Expansion Link(1)	Examination Method/Frequency(1)	Examination Coverage	Comments
Core Shroud Assembly (Welded) Assembly	Plant designs with core shrouds assembled in two vertical sections	Distortion (Void Swelling), as evidenced by separation between the upper and lower core shroud segments Aging Management (IE)	None	Visual (VT-1) examination no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	If a gap exists, make three to five measurements of gap opening from the core side at the core shroud re-entrant corners. Then, evaluate the swelling on a plant-specific basis to determine frequency and method for additional examinations. See Figures 4-12 and 4-14 of MRP-227-A	ANO-2 must perform the VT-1 initial augmented inspection by refueling outage 27
Core Support Barrel Assembly Upper (core support barrel) flange weld	All plants	Cracking (SCC)	Lower core support beams Core support barrel assembly upper cylinder Upper core barrel flange	Enhanced visual (EVT-1) examination no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	100% of the accessible surfaces of the upper flange weld. (4) See Figure 4-15 of MRP-227-A	ANO-2 must perform the EVT-1 initial augmented inspection by refueling outage 27
Core Support Barrel Assembly Lower cylinder girth welds	All plants	Cracking (SCC, IASCC) Aging Management (IE)	Lower cylinder axial welds	Enhanced visual (EVT-1) examination no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	100% of the accessible surfaces of the lower cylinder welds (4) See Figure 4-15 of MRP-227-A	ANO-2 must perform the EVT-1 initial augmented inspection by refueling outage 27
Lower Support Structure Core support column welds	All plants	Cracking (SCC, IASCC, Fatigue including damaged or fractured material) Aging Management (IE, TE)	None	Visual (VT-3) examination no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	100% of the accessible surfaces of the core support column welds(5) See Figures 4-16 and 4-31 of MRP-227-A	ANO-2 must perform the VT-3 initial augmented inspection by refueling outage 27

Table 5-1. CE Plants Primary Category Components from Table 4-2 of MRP-227-A [4] (continued)

Item	Applicability	Effect (Mechanism)	Expansion Link[(1)]	Examination Method/Frequency(1)	Examination Coverage	Comments
Core Support Barrel Assembly Lower flange weld	All plants	Cracking (Fatigue)	None	If fatigue life cannot be demonstrated by time limited aging analysis (TLAA), enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examination on a ten-year interval.	Examination coverage to be defined by evaluation to determine the potential location and extent of fatigue cracking. See Figures 4-15 and 4-16 of MRP-227-A	ANO-2 must perform the EVT-1 initial augmented inspection by refueling outage 27
Lower Support Structure Core support plate	All plants with a core support plate	Cracking (Fatigue) Aging Management (IE)	None	If fatigue life cannot be demonstrated by time limited aging analysis (TLAA), enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examination on a ten-year interval.	Examination coverage to be defined by evaluation to determine the potential location and extent of fatigue cracking. See Figure 4-16 of MRP-227-A	ANO-2 must perform the EVT-1 initial augmented inspection by refueling outage 27
Upper Internals Assembly Fuel alignment plate (Not applicable for ANO-2)	All plants with core shrouds assembled with full-height shroud plates	Cracking (Fatigue)	None	If fatigue life cannot be demonstrated by time limited aging analysis (TLAA), enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examination on a ten-year interval.	Examination coverage to be defined by evaluation to determine the potential location and extent of fatigue cracking. See Figure 4-17 of MRP-227-A	N/A
Control Element Assembly Instrument guide tubes	All plants with instrument guide tubes in the CEA shroud assembly	Cracking (SCC, Fatigue) that results in missing supports or separation at the welded joint between the tubes and supports	Remaining instrument guide tubes within the CEA shroud assemblies.	Visual (VT-3) examination, no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examination on a ten-year interval. Plant-specific component integrity assessments may be required if degradation is detected and remedial action is needed.	100% of tubes in peripheral CEA shroud assemblies (i.e., those adjacent to the perimeter of the fuel alignment plate). See Figure 4-18 of MRP-227-A	ANO-2 must perform the VT-3 initial augmented inspection by refueling outage 27

Table 5-1. CE Plants Primary Category Components from Table 4-2 of MRP-227-A [4] (continued)

Item	Applicability	Effect (Mechanism)	Expansion Link(1)	Examination Method/Frequency(1)	Examination Coverage	Comments
Lower Support Structure Deep beams (Not applicable to ANO-2)	All plants with core shrouds assembled with full-height shroud plates.	Cracking (Fatigue) that results in a detectable surface-breaking indication in the welds or beams Aging Management (IE)	None	Enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examination on a ten-year interval, if adequacy of remaining fatigue life cannot be demonstrated.	Examine beam-to-beam welds, in the axial elevation from the beam top surface to four inches below. See Figure 4-19 of MRP-227-A	N/A

Notes:

1. Examination acceptance criteria and expansion criteria for the CE components are in Table 5-4 (MRP-227-A Table 5-2)
2. Void swelling effects on this component is managed through management of void swelling on the entire core shroud assembly.
3. A minimum of 75% of the total population (examined + unexamined), including coverage consistent with the Expansion criteria in Table 5-4, must be examined for inspection credit.
4. A minimum of 75% of the total weld length (examined + unexamined), including coverage consistent with the Expansion criteria in Table 5-4, must be examined from either the inner or outer diameter for inspection credit.
5. A minimum of 75% of the total population of core support column welds.

Table 5-2. CE Plants Expansion Category Components from Table 4-5 of MRP-227-A [4]

Item	Applicability	Effect (Mechanism)	Primary Link	Examination Method/Frequency (1)	Examination Coverage/Frequency (1)	Comments
Core Shroud Assembly (Bolted) Barrel-shroud bolts (Not applicable for ANO-2)	Bolted plant designs	Cracking (IASCC, Fatigue) Aging Management (IE and ISR)	Core shroud bolts	Volumetric (UT) examination. Re-inspection every 10 years following initial inspection.	100 % (or as supported by plant-specific justification) (2) of barrel-shroud and guide lug insert bolts with neutron fluence exposures > 3 displacements per atom (dpa). See Figure 4-23 of MRP-227-A	N/A
Core Support Barrel Assembly Lower core barrel flange	All plants	Cracking (SCC, Fatigue)	Upper (core support barrel) flange weld.	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of accessible welds and adjacent base metal (2). See Figure 4-15 of MRP-227-A	Contingency if indications are found in EVT-1 exam of Upper (core support barrel) flange weld
Core Support Barrel Assembly Upper cylinder (including welds)	All plants	Cracking (SCC) Aging Management (IE)	Upper (core support barrel) flange weld.	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of accessible surfaces of the welds and adjacent base metal (2). See Figure 4-15 of MRP-227-A	Contingency if indications are found in EVT-1 exam of Upper (core support barrel) flange weld
Core Support Barrel Assembly Upper Core Barrel Flange	All plants	Cracking (SCC)	Upper (core support barrel) flange weld	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection	100% of accessible bottom surface of the flange (2) See Figure 4-15 of MRP-227-A	Contingency if indications are found in EVT-1 exam of Upper (core support barrel) flange weld

Table 5-2. CE Plants Expansion Category Components from Table 4-5 of MRP-227-A [4] (continued)

Item	Applicability	Effect (Mechanism)	Primary Link	Examination Method/Frequency (1)	Examination Coverage/Frequency (1)	Comments
Core Support Barrel Assembly Core barrel assembly axial welds	All plants	Cracking (SCC)	Core barrel assembly girth welds	Enhanced visual (EVT-1) examination, with initial and subsequent examinations dependent on the results of core barrel assembly girth weld examinations.	100% of one side of the accessible weld and adjacent base metal surfaces for the weld with the highest calculated operating stress. See Figures 4-15 of MRP-227-A.	Contingency if indications are found in EVT-1 exam of Core barrel assembly girth welds
Lower Support Structure Lower core support beams	All plants except those with core shrouds assembled with full-height shroud plates	Cracking (SCC, Fatigue) including damaged or fractured material	Upper (core support barrel) flange weld	Visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of accessible surface.(2) See Figures 4-16 and 4-31 of MRP-227-A.	Contingency if indications are found in EVT-1 exam of Upper (core support barrel) flange weld
Core Shroud Assembly (Bolted) Core support column bolts (Not applicable for ANO-2)	Bolted plant designs	Cracking (IASCC, Fatigue) Aging Management (IE)	Core shroud bolts	Ultrasonic (UT) examination. Re-inspection every 10 years following initial inspection.	100 % (or as supported by plant-specific analysis) of core support column bolts with neutron fluence exposures > 3 dpa. (2) See Figures 4-16 and 4-33 of MRP-227-A	N/A
Core Shroud Assembly (Welded) Remaining axial welds, Ribs and rings (Not applicable for ANO-2)	Plant designs with core shrouds assembled with full-height shroud plates.	Cracking (IASCC) Aging Management (IE)	Shroud plates of welded core shroud assemblies	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	Axial weld seams other than the core shroud re-entrant corner welds at the core mid-plane, plus ribs and rings. See Figure 4-13 of MRP-227-A	N/A

Table 5-2. CE Plants Expansion Category Components from Table 4-5 of MRP-227-A [4] (continued)

Item	Applicability	Effect (Mechanism)	Primary Link	Examination Method/Frequency (1)	Examination Coverage/Frequency (1)	Comments
Control Element Assembly Remaining instrument guide tubes	All plants with instrument guide tubes in the CEA shroud assembly.	Cracking (SCC, Fatigue) that results in missing supports or separation at the welded joint between the tubes and supports.	Peripheral instrument guide tubes within the CEA shroud assemblies.	Visual (VT-3) examination. Re-inspection every 10 years following initial inspection.	100% of tubes in CEA shroud assemblies. (2) See Figure 4-18 of MRP-227-A	Contingency if indications are found in VT-3 exam of the peripheral instrument guide tubes within the CEA shroud assemblies

Notes:

1. Examination acceptance criteria and expansion criteria for the CE components are in Table 5-4 (MRP-227-A Table 5-2).
2. A minimum of 75% coverage of the entire examination area or volume, or a minimum sample size of 75% of the total population of like components of the examination is required (including both accessible and inaccessible portions).

Table 5-3. CE Plants Existing Program Components Credited in Table 4-8 of MRP-227-A [4]

Item	Applicability	Effect (Mechanism)	Primary Link	Examination Method	Examination Coverage and Schedule	Comments
Core Shroud Assembly Guide lugs Guide lug inserts and bolts	All plants	Loss of material (Wear) Aging Management (ISR)	ASME Code Section XI	Visual (VT-3) examination, general condition examination for detection of excessive or asymmetrical wear.	First 10-year ISI after 40 years of operation, and at each subsequent inspection interval. Accessible surfaces at specified frequency	To be inspected
Lower Support Structure Fuel alignment pins (Not applicable for ANO-2)	All plants with core shrouds assembled with full-height shroud plates	Cracking (SCC, IASCC, Fatigue) Aging Management (IE and ISR)	ASME Code Section XI	Visual (VT-3) examination to detect severed fuel alignment pins, missing locking tabs, or excessive wear on the fuel alignment pin nose or flange.	Accessible surfaces at specified frequency	N/A
Lower Support Structure Fuel alignment pins	All plants with core shrouds assembled in two vertical sections	Loss of Material (Wear) Aging Management (IE and ISR)	ASME Code Section XI	Visual (VT-3) examination	Accessible surfaces at specified frequency	To be inspected
Core Barrel Assembly Upper flange	All plants	Loss of Material (Wear)	ASME Code Section XI	Visual (VT-3) examination	Area of the upper flange potentially susceptible to wear	To be inspected

Table 5-4. CE Plants Examination Acceptance and Expansion Criteria from Table 5-2 of MRP-227-A [4]

Item	Applicability	Examination Acceptance Criteria (1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Shroud Assembly (Bolted) Core shroud bolts (Not applicable for ANO-2)	Bolted plant designs	Volumetric (UT) examination. The examination acceptance criteria for the UT of the core shroud bolts shall be established as part of the examination technical justification.	a. Core support column bolts b. Barrel-shroud bolts	a. Confirmation that >5% of the core shroud bolts in the four plates at the largest distance from the core contain unacceptable indications shall require UT examination of the lower support column bolts barrel within the next 3 refueling cycles. b. Confirmation that > 5% of the core support column bolts contain unacceptable indications shall require UT examination of the barrel-shroud bolts within the next 3 refueling cycles.	a and b. The examination acceptance criteria for the UT of the core support column bolts and barrel-shroud bolts shall be established as part of the examination technical justification.
Core Shroud Assembly (Welded) Core shroud plate-former plate weld	Plant designs with core shrouds assembled in two vertical sections	Visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication	Remaining axial welds	Confirmation that a surface-breaking indication > 2 inches in length has been detected and sized in the core shroud plate-former plate weld at the core shroud re-entrant corners (as visible from the core side of the shroud), within 6 inches of the central flange and horizontal stiffeners, shall require EVT-1 examination of all remaining axial welds by the completion of the next refueling outage.	The specific relevant condition is a detectable crack-like surface indication.

Table 5-4. CE Plants Examination Acceptance Criteria and Expansion Criteria from Table 5-2 of MRP-227-A [4] (continued)

Item	Applicability	Examination Acceptance Criteria (1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Shroud Assembly (Welded) Shroud plates (Not applicable for ANO-2)	Plant designs with core shrouds assembled with full-height shroud plates	Visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication	a. Remaining axial welds b. Ribs and rings	a. Confirmation that a surface breaking indication > 2 inches in length has been detected and sized in the axial weld seams at the core shroud re-entrant corners at the core mid-plane shall require EVT-1 or UT examination of all remaining axial welds by the completion of the next refueling outage. b. If extensive cracking is detected in the remaining axial welds, an EVT-1 examination shall be required of all accessible rib and ring welds by the completion of the next refueling outage.	The specific relevant condition is a detectable crack-like surface indication.
Core Shroud Assembly (Bolted) Assembly (Not applicable for ANO-2)	Bolted plant designs	Visual (VT-3) examination. The specific relevant conditions are evidence of abnormal interaction with fuel assemblies, gaps along high fluence shroud plate joints, and vertical displacement of shroud plates near high fluence joints.	None	N/A	N/A

Table 5-4. CE Plants Examination Acceptance Criteria and Expansion Criteria from Table 5-2 of MRP-227-A [4] (continued)

Item	Applicability	Examination Acceptance Criteria (1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Shroud Assembly (Welded) Assembly	Plant designs with core shrouds assembled in two vertical sections	Visual (VT-1) examination. The specific relevant condition is evidence of physical separation between the upper and lower core shroud sections.	None	N/A	N/A
Core Support Barrel Assembly Upper (core support barrel) flange weld	All plants	Visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	Lower core support beams Upper core barrel cylinder (including welds) Upper core barrel flange	Confirmation that a surface breaking indication >2 inches in length has been detected and sized in the upper flange weld shall require that an EVT-1 examination of the lower core support beams, upper core barrel cylinder and upper core barrel flange be performed by the completion of the next refueling outage	The specific relevant condition is a detectable crack-like surface indication.
Core Support Barrel Assembly Lower cylinder girth welds	All plants	Visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	Lower cylinder axial welds	Confirmation that a surface breaking indication >2 inches in the length has been detected and sized in the lower cylinder girth weld shall require an EVT-1 examination of all accessible lower cylinder axial welds by the completion of the next refueling outage.	The specific relevant condition for the expansion lower cylinder axial welds is a detectable crack-like surface indication
Lower Support Structure Core support column welds	All plants	Visual (VT-3) examination. The specific relevant condition is missing or separated welds.	None	None	N/A

Table 5-4. CE Plants Examination Acceptance Criteria and Expansion Criteria from Table 5-2 of MRP-227-A [4] (continued)

Item	Applicability	Examination Acceptance Criteria (1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Support Barrel Assembly Lower flange weld	All plants	Visual (EVT-1) examination. The specific relevant condition is a detectable crack-like indication.	None	N/A	N/A
Lower Support Structure Core support plate	All plants with a core support plate	Visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	None	N/A	N/A
Upper Internals Assembly Fuel alignment plate (Not applicable for ANO-2)	All plants with core shrouds assembled with full-height shroud plates	Visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	None	N/A	N/A
Control Element Assembly Instrument guide tubes	All plants with instrument tubes in the CEA shroud assembly	Visual (VT-3) examination. The specific relevant conditions are missing supports and separation at the welded joint between the tubes and the supports.	Remaining instrument tubes within CEA shroud assemblies	Confirmed evidence of missing supports or separation at the welded joint between the tubes and supports shall require the visual (VT-3) examination to be expanded to the remaining instrument tubes within the CEA shroud assemblies by completion of the next refueling outage.	The specific relevant conditions are missing supports and separation at the welded joint between the tubes and the supports.

Table 5-4. CE Plants Examination Acceptance Criteria and Expansion Criteria from Table 5-2 of MRP-227-A [4] (continued)

Item	Applicability	Examination Acceptance Criteria (1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Lower Support Structure Deep Beams (Not applicable for ANO-2)	All plants with core shrouds assembled with full-height shroud plates	Visual (EVT-1) examination. The specific relevant condition is a detectable crack-like indication	None	N/A	N/A

Note:

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

Table 5-5. ANO-2 Response to the NRC Final Safety Evaluation of MRP-227-A [5]

MRP-227 SE Item	ANO-2 Response
SE Section 4.1.1, Topical Report Condition 1: High consequence components in the "No Additional Measures" inspection category.	In accordance with SE Section 4.1.1, the Lower Core Support Beams, Core Support Barrel Assembly Upper Cylinder and Upper Core Barrel Flange have been added to the ANO-2 "Expansion" inspection category and are contained in Table 5-2. The components are linked to the "Primary" components Core Support Barrel Upper (core support barrel) flange weld.
SE Section 4.1.2, Topical Report Condition 2: Inspection of components subject to irradiation-assisted stress corrosion cracking.	In accordance with SE Section 4.1.2, the Core Support Barrel Assembly Lower Cylinder Girth Welds have been added to the ANO-2 "Primary" inspection category and are contained in Table 5-1. The examination method is consistent with the MRP recommendations for these components, the examination coverage conforms to the criteria described in Section 3.3.1 of the NRC SE, and the re-examination frequency is on a 10-year interval consistent with other "Primary" inspection category components.
SE Section 4.1.3, Topical Report Condition 3: Inspection of high consequence components subject to multiple degradation mechanisms	In accordance with SER Section 4.1.3, the Core Support Column (casting or wrought) welds in the lower support structure have been added to the ANO-2 "Primary" inspection category and are contained in Table 5-1. The examination method is consistent with MRP recommendations for these components. The coverage confirms to the criteria described in Section 3.3.1 of the NRC SE, and the re-examination frequency is on a 10-year interval consistent with the other "Primary" inspection category components.
SE Section 4.1.4, Topical Report Condition 4: Imposition of minimum examination coverage criteria for "expansion" inspection category components	In accordance with SE Section 4.1.4, ANO-2 will meet the minimum inspection coverage specified in the SE. The appropriate wording has been added to Table 5-2 (Note 2) examination coverage.
SE 4.1.5, Topical Report Condition 5: Examination frequencies for baffle former bolts and core shroud bolts	Not applicable for ANO-2.

Table 5-5. ANO-2 Response to NRC Final Safety Evaluation of MRP-227-A [5] (continued)

MRP-227 SE Item	ANO-2 Response
SE 4.1.6, Topical Report Condition 6: Periodicity of the re-examination of "Expansion" inspection category components	In accordance with SE Section 4.1.6, Table 5-2 requires a 10-year re-examination interval for all "Expansion" inspection category components once degradation is identified in the associated "Primary" inspection category component and examination of the expansion category component commences. "Re-inspection every 10 years following initial inspection" is added to every component under the Examination Method/Frequency column in Table 5-2.
SE Section 4.1.7, Topical Report Condition 7: Updating of MRP-227, Revision 0, Appendix A	This condition applies to update of the industry guidelines. No plant-specific actions are required.
SE Section 4.2.1, Applicant/Licensee Action Item 1: Applicability of FMECA and Functionality Analysis Assumptions	The evaluation of design and operating history demonstrating that MRP-227-A is applicable to ANO-2 is contained in Section 1.8.4.1 and Section 5.1 of this document.
SE Section 4.2.2, Applicant/Licensee Action Item 2: PWR Vessel Internals Components Within the Scope of License Renewal	The ANO-2 review of components within the scope of license renewal was compared against the information contained in Table 4-5 of MRP-191. The Aging Management Review performed as part of the ANO-2 LRA is described in Section 1.7.1 of this document and summarized as part of Applicant/Licensee Action Item 2 in Section 5.2.
SE Section 4.2.3, Applicant/Licensee Action Item 3: Evaluation of the Adequacy of Plant-Specific Existing Programs	ANO-2 complies through management and replacement of in-core instrumentation thimble tubes as described in [32]. Thermal shields are not present in the ANO-2 reactor vessel internals.
SE Section 4.2.4, Applicant/Licensee Action Item 4: B&W Core Support Structure Upper Flange Stress Relief	No action required. This action does not apply to CE designed units.

Table 5-5. ANO-2 Response to NRC Final Safety Evaluation of MRP-227-A [5] (continued)

MRP-227 SE Item	ANO-2 Response
SE Section 4.2.5, Applicant/Licensee Action Item 5: Application of Physical Measurements as part of I&E Guidelines for B&W, CE and Westinghouse RVI Components	ANO-2 has a core barrel shroud assembled in two vertical sections. Per the examination coverage criteria defined in Table 4-2 of MRP-227-A (see Table 5-1), if a gap between the two core barrel shroud vertical sections are identified during the required VT-1 examination, three to five measurements of the gap opening from the core side at the core shroud re-entrant corners is required. Prior to performing the VT-1 examination ANO-2 will develop acceptance criteria that are consistent with the licensing basis to ensure that the core shroud remains capable of performing its required functions.
SE Section 4.2.6, Applicant/Licensee Action Item 6: Evaluation of Inaccessible B&W Components	No action required. This action does not apply to CE designed units.
SE Section 4.2.7, Applicant/Licensee Action Item 7: Plant Specific Evaluation of CASS Materials	The CEA shroud tube is the only component fabricated from CASS material listed in Table 3.1.2-2 of LRA [1], which is in Category A by MRP-191 [13]. This component item initially screened in for SCC (welds) and TE but the FMECA determined that these age-related degradation mechanisms have minimal likelihood to cause failure. Thus, this component was assigned to Category A per MRP-191. The CEA shroud tubes are considered a no additional measures component per MRP-227-A and the existing inservice inspection program is adequate to manage this CASS RVI component during the period of extended operation. The CEA shroud tubes are not considered a primary or expansion component for CE plants in Table 4-2 and Table 4-5 of MRP-227-A [4].
SE Section 4.2.8, Applicant/Licensee Action Item 8	The responses to meet A/LAI No. 8 are contained in Section 5.8 of this document.

Table 5-6. ANO-2 Program Enhancement and Implementation Schedule

Refueling Outage	Cycle End Quarter/Year	AMP-Related Scope	Inspection Method and Criteria	Comments
2R25	Spring 2017	Not applicable	Not applicable	Extended Operation Period begins Midnight on July 17, 2018
2R26	Fall 2018	ASME Section XI 10 Year ISI inspections of core shroud assembly (guide lugs, guide lug inserts and bolts), core barrel assembly (upper flange), and lower support structure (fuel alignment pins)	Inspections in accordance with ANO-2 ISI Program	Not applicable
2R27	Spring 2020	Initial MRP-227-A augmented inspections for core shroud assembly (core shroud plate-former plate weld, assembly), core support barrel assembly (upper core support barrel flange weld, lower cylinder girth welds, lower flange weld), lower support structure (core support column welds, core support plate), and control element assembly (instrument guide tubes).	MRP-227-A inspections in accordance with MRP-228.	ANO-2 plans to begin extended operation during Cycle 26. ANO-2 has the option to perform these inspections until 2R27, which is no later than 2 refueling outages from the beginning of the license renewal period.
2R28	Fall 2021	Not applicable	Not applicable	Not applicable
2R29	Spring 2023	Not applicable	Not applicable	Not applicable
2R30	Fall 2024	Not applicable	Not applicable	Not applicable
2R31	Spring 2026	Not applicable	Not applicable	Not applicable
2R32	Fall 2027	Not applicable	Not applicable	Not applicable

Table 5-6. ANO-2 Program Enhancement and Implementation Schedule (continued)

Refueling Outage	Cycle End Quarter/Year	AMP-Related Scope	Inspection Method and Criteria	Comments
2R33	Spring 2029	ASME Section XI 10 Year ISI inspections of core shroud assembly (guide lugs, guide lug inserts and bolts), core barrel assembly (upper flange), and lower support structure (fuel alignment pins)	Inspections in accordance with ANO-2 ISI Program	Not applicable
2R34	Fall 2030	Subsequent MRP-227-A augmented inspections for core shroud assembly (core shroud plate-former plate weld, assembly), core support barrel assembly (upper core support barrel flange weld, lower cylinder girth welds, lower flange weld), lower support structure (core support column welds, core support plate), and control element assembly (instrument guide tubes).	MRP-227-A inspections in accordance with MRP-228.	Not applicable
2R35	Spring 2032	Not applicable	Not applicable	Not applicable
2R36	Fall 2033	Not applicable	Not applicable	Not applicable
2R37	Spring 2035	Not applicable	Not applicable	Not applicable
2R38	Fall 2036	Not applicable	Not applicable	Not applicable
N/A	July, 2038	Not applicable	Not applicable	Renewed Operating License expires Midnight on July 17, 2038.

Table 5-7. Summary of Actions Related to Aging Management of RVI for ANO-2

Item No.	ANO-2 Action	Program/Action Description
1	ANO-2 Update to the Reactor Vessel Internals Inspection Program Commitments	Submit the ANO-2 reactor vessel internals aging management program and inspection plans in accordance with MRP-227-A no later than two years prior to the period of extended operation (July 17, 2016).
2	ANO-2 will review plant specific and fleet operating experience based on updates to Appendix A of MRP-227-A and make updates to the RVI AMP as necessary.	Reactor Vessel Internals Aging Management Program
3	Participation in Industry Groups (e.g. PWR Owners Group Materials Subcommittee, EPRI MRP)	Reactor Vessel Internals Aging Management Program
4	Entergy Personnel Responsibilities: • Senior Management • Engineering	Ensure department specific actions are performed as it relates to the aging management of Reactor Vessel Internals
5	Plant Specific Programs: • ASME Section XI Inservice Inspection Program • Primary Chemistry Monitoring Program	The RVI AMP takes credit for plant specific programs. RVI AMP items credited from plant specific programs (e.g. ASME Section XI ISI program components credited as part of MRP-227-A inspections)
6	Examinations specified in the MRP-227-A guidelines shall be conducted in accordance with Inspection Standard MRP-228.	Reactor Vessel Internals Aging Management Program/ASME Section XI ISI Program
7	Examination results that do not meet the examination acceptance criteria defined in Section 5 of MRP-227-A guidelines shall be recorded and entered in the plant corrective action program and dispositioned.	ANO-2 Procedure, EN-LI-102, "Corrective Action Process."

Table 5-7. Summary of Actions Related to Aging Management of RVI for ANO-2 (continued)

Item No.	ANO-2 Action	Program/Action Description
8	Each commercial U.S. PWR unit shall 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 within the scope of MRP-227-A are examined	Reactor Vessel Internals Aging Management Program
9	If an engineering evaluation is used to disposition an examination result that does not meet the examination acceptance criteria in Section 5 of MRP-227-A, this engineering evaluation shall be conducted in accordance with an NRC-approved evaluation methodology.	ANO-2 will comply with this requirement by using NRC-approved evaluation methodology (e.g. WCAP-17096)
10	Inspection acceptance and expansion criteria will be reviewed whenever new versions of the NRC approved versions of MRP-227 and WCAP-17096 are published, as the industry continues to develop and refine the information. Relevant changes based on the review of these NRC approved documents will be included as updates to the RVI AMP.	Reactor Vessel Internals Aging Management Program

6.0 REFERENCES

1. "Arkansas Nuclear One - Unit 2 License Renewal Application," October 15, 2003.

2. Entergy Nuclear Engineering Program, SEP-ISI-ANO2-105, Revision 1, "Program Section for ASME Section XI, Division 1 ANO 2 Inservice Inspection Program," February 13, 2014 (SI File No. 1500227.204).

3. NEI 03-08, Revision 2, "Guideline for the Management of Materials Issues," dated January, 2010.

4. *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)*. EPRI, Palo Alto, CA: 2011. 1022863.

5. Letter from Robert A. Nelson (NRC) to Neil Wilmshurst (EPRI) dated December 16, 2011, "Revision 1 to the Final Safety Evaluation of EPRI Report, Materials Reliability Program Report 1016596 (MRP-227), Revision 0, Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines" (TAC No. ME0680)," NRC ADAMS Accession No. ML11308A770.

6. U.S. Code of Federal Regulations, Title 10, "Energy," Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants."

7. Westinghouse Report No., WCAP-17432-NP, Revision 0, "Reactor Vessel Internals Program Plan for Aging Management of Reactor Internals at Arkansas Nuclear One, Unit 2," November 2011 (SI File No. 1500227.202).

8. *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-Rev. 0)*. EPRI, Palo Alto, CA: 2008. 1016596.

9. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 2001 Edition through 2003 Addenda.

10. NUREG-1801, Revision 2, "Generic Aging Lessons Learned (GALL) Report," U. S. Nuclear Regulatory Commission, December 2010.

11. NUREG-1800, Revision 2, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, December 2010.

12. Combustion Engineering Owners Group Report No., CE NPSD-1216, Revision 0, "Generic Aging Management Review Report for the Reactor Vessel Internals," CEOG Task 1185, March 2001 (SI File No. 1500227.203).

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

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

15. *Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values (MRP-175).* EPRI, Palo Alto, CA: 2005. 1012081.

16. Westinghouse Report, WCAP-17096-NP, Revision 2, "Reactor Internals Acceptance Criteria Methodology and Data Requirements," December 2009.

17. U.S. Nuclear Regulatory Commission Letter, "Final Safety Evaluation of WCAP-17096-NP, Revision 2, 'Reactor Internals Acceptance Criteria Methodology and Data Requirements' (TAC No. ME4200)," May 3, 2016, NRC ADAMS Accession No. ML16061A243.

18. ANO-2 Updated Final Safety Analysis Report, SAR Amendment 26, Docket Number 50-368 (SI File No. 1500227.207).

19. NUREG-1828, "Safety Evaluation Report Related to the License Renewal of the Arkansas Nuclear One, Unit 2," June 2005.

20. EN-DC-133, Revision 0, "PWR Vessel Internals Program," Entergy Nuclear Management Manual, 3/31/14 (SI File No. 1200459.234).

21. EN-DC-202, Revision 6, "NEI 03-08 Materials Initiative Process," Entergy Nuclear Management Manual, November 21, 2013. (SI File No. 1200459.201).

22. Entergy Nuclear Engineering Program, 1000.106, Revision 011, "Primary Chemistry Monitoring Program" (SI File No. 1500227.210).

23. *Pressurized Water Reactor Primary Water Chemistry Guidelines, Volumes 1 and 2, Revision 7,* EPRI, Palo Alto, CA: 2014, 3002000505. EPRI Proprietary

24. Entergy Nuclear Engineering Report, 02-R-2005-01, Revision 0, "Arkansas Nuclear One Unit 2 License Renewal Project Aging Management Review of the Reactor Coolant System," September 25, 2003 (SI File No. 1500227.201).

25. U.S. Nuclear Regulatory Commission Document, RIS 2011-07, "NRC Regulatory Issue Summary 2011-07 License Renewal Submittal Information for Pressurized Water Reactor Internals Aging Management," July 21, 2011, NRC ADAMS Accession No. ML111990086.

26. Entergy Procedure, EN-LI-102, Revision 25, "Corrective Action Program," November 19, 2015 (SI File No. 1500227.214).

27. ANO-2 Memorandum, ANO-86-2-00710, "Arkansas Nuclear One-Unit 2 Review of Cycle 6 Reload Analysis Report (RAR)," August 1, 1986 (SI File No. 1500227.212).

28. ANO-2 Document, 1000.164, Rev. 001-00-0, "Quality Assurance," May 7, 2001 (SI File No. 1500227.215).

29. U.S. Code of Federal Regulations, Title 10, "Energy," Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

30. Entergy Procedure EN-OE-100, Rev. 24, "Operating Experience Program," August 11, 2015 (SI File No. 1500227.213).

31. Combustion Engineering Report, TD C490.0520, Revision 4, "Arkansas Power & Light Nuclear One Unit 2 Reactor Vessel Internals Instruction Manual," June 1977 (SI File No. 1500227.206).

32. ANO-2 Engineering Request, ER-ANO-2003-0399-003, Rev. 0, "NCP for ANO-2 ICI Thimble Tube Replacement," March, 9, 2006 (SI File No. 1500227.209).

33. Westinghouse Report, WCAP-17435-NP, Rev. 0, "Results of the Reactor Internals Operating Experience Survey Conducted under PWROG Project Authorization PA-MSC-0568," September 2011.

34. PWROG-15105-NP, Revision 0, "PA-MSC-1288 PWR RV Internals Cold-Work Assessment."