

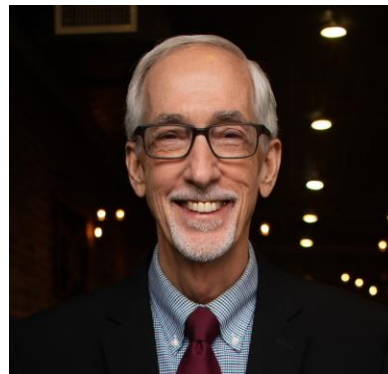
16-10 - Aging Management Approach and Existing AMPs

IAEA National Workshop on Ageing Management
during Early Stages of Operation

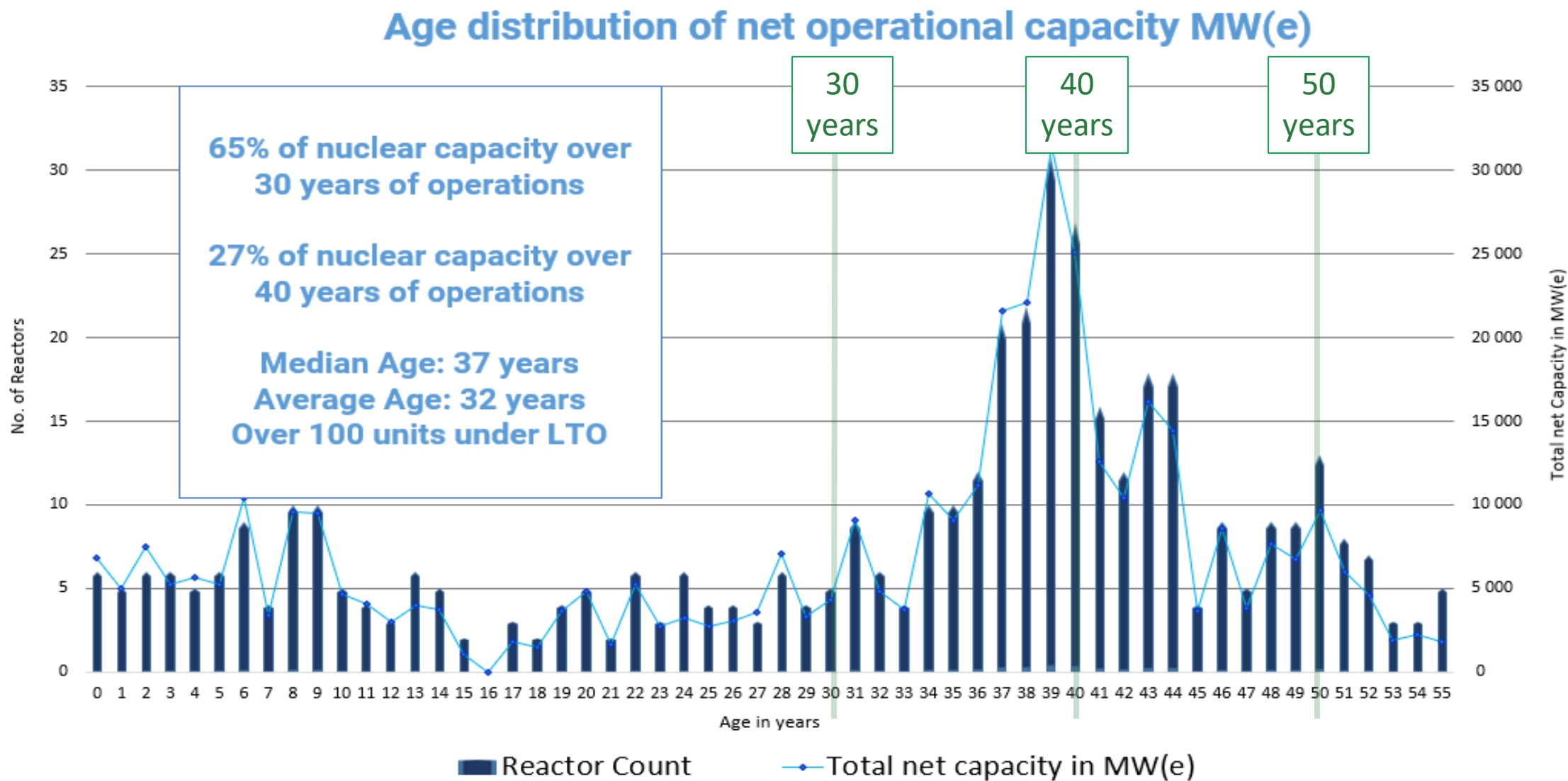


Garry G. Young
Technical Executive

Barakah NPP
Abu Dhabi, UAE
September 16, 2025



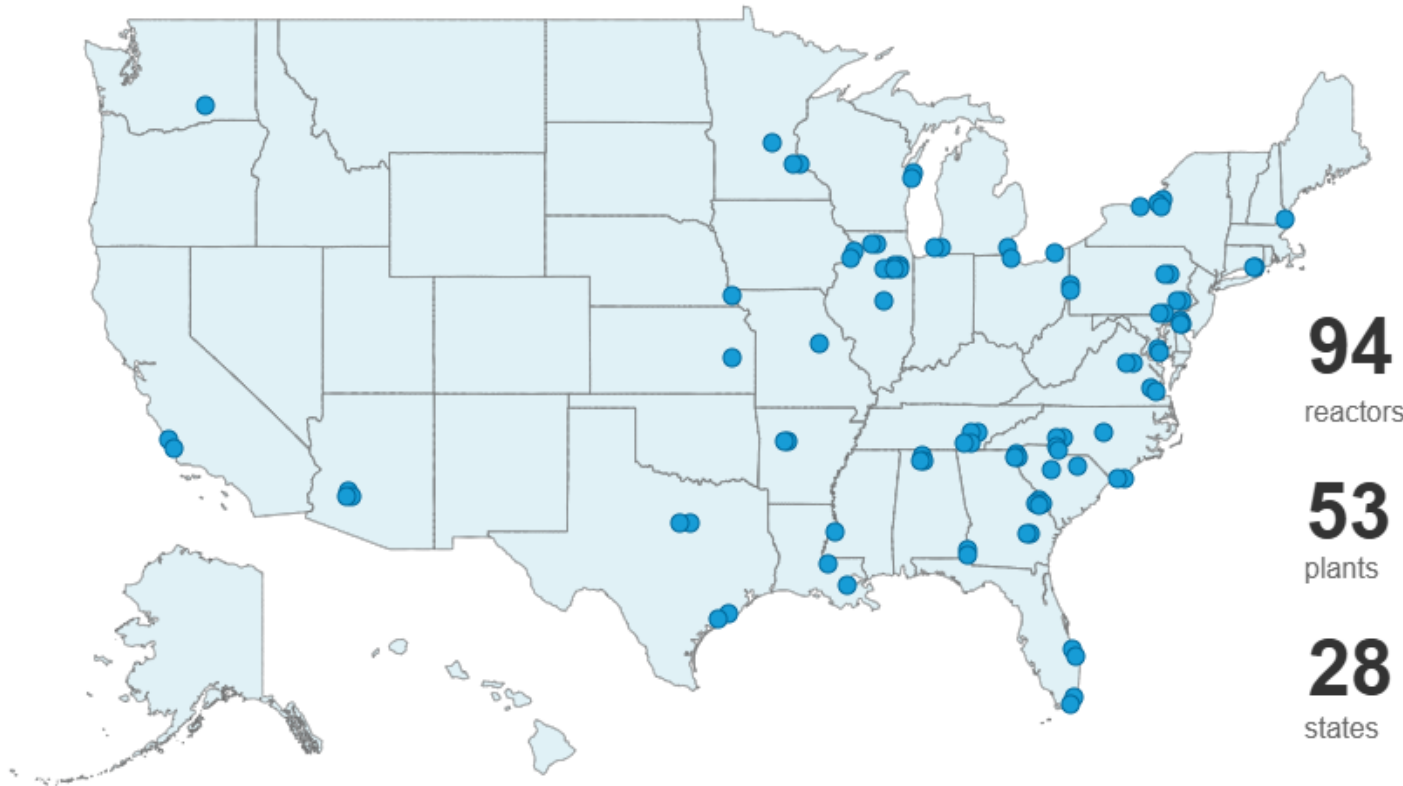
What is the Status of LTO Worldwide?



*IAEA PRIS database, as of February 2025

Can plants operate safely for more than 60 years?

NUCLEAR POWER ACROSS THE U.S.



- Recent NRC approvals
 - Oconee – March 2025 (SLR)
 - VC Summer – June 2025 (SLR)
 - Perry – July 2025 (LR)
- 7 reactors have 40-yr licenses, 3 are under review for renewal
- 74 reactors have 60-year licenses
- 13 reactors have 80-year licenses, 12 more are under review - SLR

Over 20 more letters of intent for SLR in the next 10 years
More than 90% of U.S. reactors are approved for LTO

What are the Key Technical Issues for 60+ Years of Operation?

- **Reactor pressure vessel – neutron embrittlement**

- Trends for high fluence levels
- Sufficiency of material surveillance program capsules

- **Reactor vessel internals - high fluence effects**

- Irradiation-assisted stress corrosion cracking
- Loss of fracture toughness; void swelling

- **Concrete and containment performance**

- Radiation and high temperature exposure
- Alkali-silica reaction

- **Electrical cables**

- Environmental qualification
- In-service testing
- Long-term submersion



Each issue was addressed in SLR guidance



Research continues to validate and update guidance, as necessary

Industry List of Major Issues for Long Term Operation

License Renewal Applications - Availability

- All Initial and Subsequent License Renewal Applications can be found on the NRC Website
- Initial LRA's:
<https://www.nrc.gov/reactors/operating/licensing/renewal/applications.html>
- Subsequent LRA's:
<https://www.nrc.gov/reactors/operating/licensing/renewal/subsequent-license-renewal.html>

SLR Applications

Applications Currently Under Review:

Plant Name and Unit(s)	Application Received
Point Beach Units 1 and 2	11/16/2020
St. Lucie Plant Units 1 and 2	08/03/2021
Virgil C. Summer Unit 1	08/17/2023
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 TOP

Completed Applications

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Peach Bottom Units 2 and 3	07/10/18	03/05/20	08/08/33 (Unit 2) 07/02/34 (Unit 3)
Surry Units 1 and 2	10/15/18	05/04/21	05/25/2032 (Unit 1) 01/29/2033 (Unit 2)
North Anna Units 1 and 2	08/24/20	08/28/24	04/01/2038 (Unit 1) 08/21/2040 (Unit 2)
Monticello Nuclear Generating Plant Unit 1	01/09/2023	12/30/2024	09/08/2030
Oconee Nuclear Station Units 1, 2, and 3	06/07/2021	03/31/2025	02/06/2033 (Unit 1) 10/06/2033 (Unit 2) 07/19/2034 (Unit 3)

Typical Contents of an LRA

- Section 1: Administrative Information
- Section 2: Scoping & Screening Methodology and Results
- Section 3: Aging Management Review Results
- Section 4: Time-Limited Aging Analyses
- Appendix A: UFSAR Supplement
- Appendix B: Aging Management Program Descriptions
- Environmental Report

Bookmarks

- ✓ DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3 - SUBSEQUENT LICENSE RENEWAL APPLICATION
 - COVER PAGE
 - > TABLE OF CONTENTS
 - > 1.0 ADMINISTRATIVE INFORMATION
 - > 2.0 SCOPING AND SCREENING METHODOLOGY FOR IDENTIFYING STRUCTURES AND COMPONENTS
 - > 3.0 AGING MANAGEMENT REVIEW RESULTS
 - > 4.0 TIME-LIMITED AGING ANALYSES
 - > APPENDIX A UPDATED FINAL SAFETY ANALYSIS REPORT SUPPLEMENT
 - > APPENDIX B AGING MANAGEMENT PROGRAMS
 - APPENDIX C RESPONSE TO BWRVIP LICENSE RENEWAL APPLICANT ACTION ITEMS
 - APPENDIX D TECHNICAL SPECIFICATION CHANGES

Dresden Nuclear Power Station, Units 2 and 3



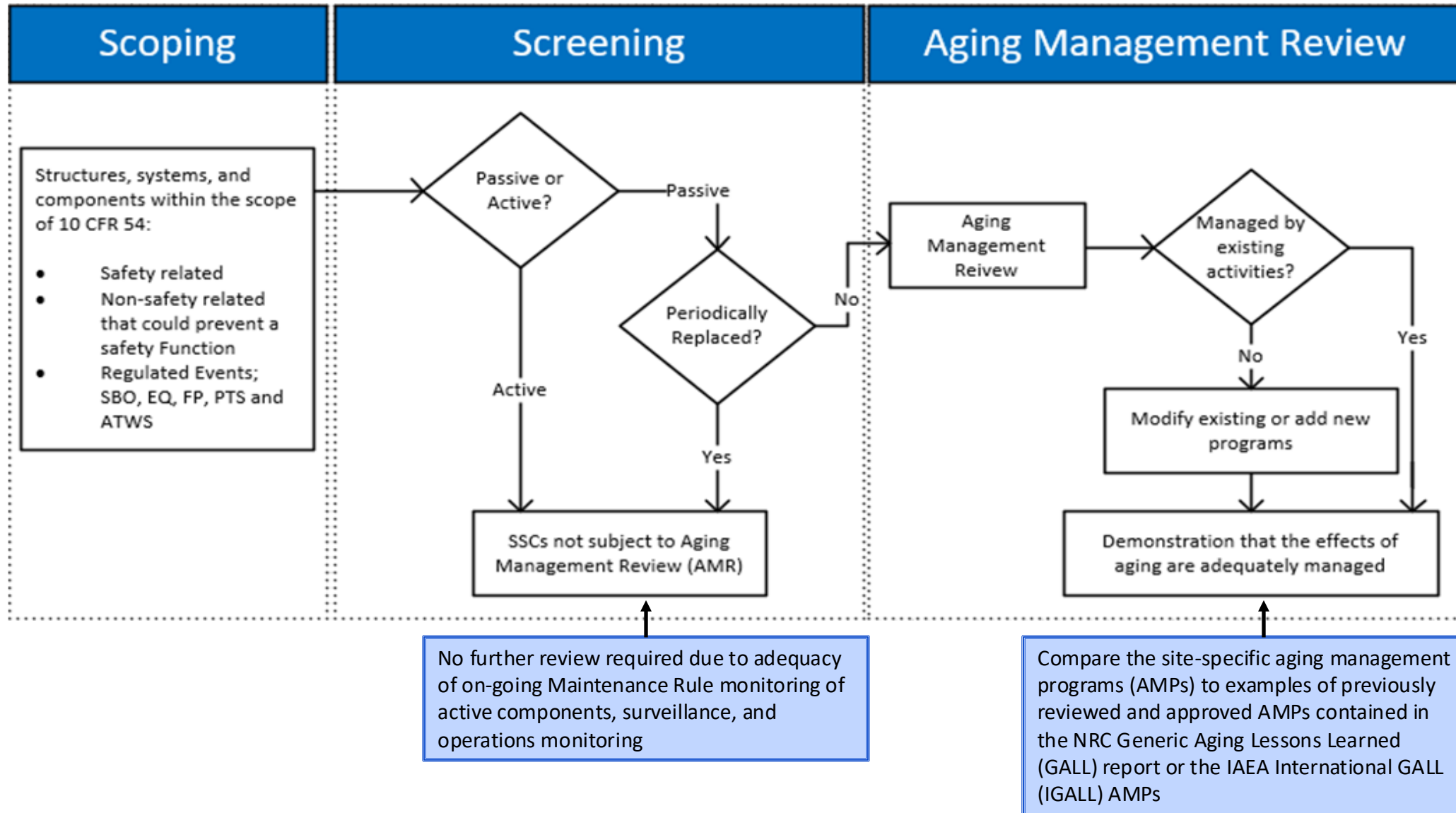
Subsequent License Renewal Application



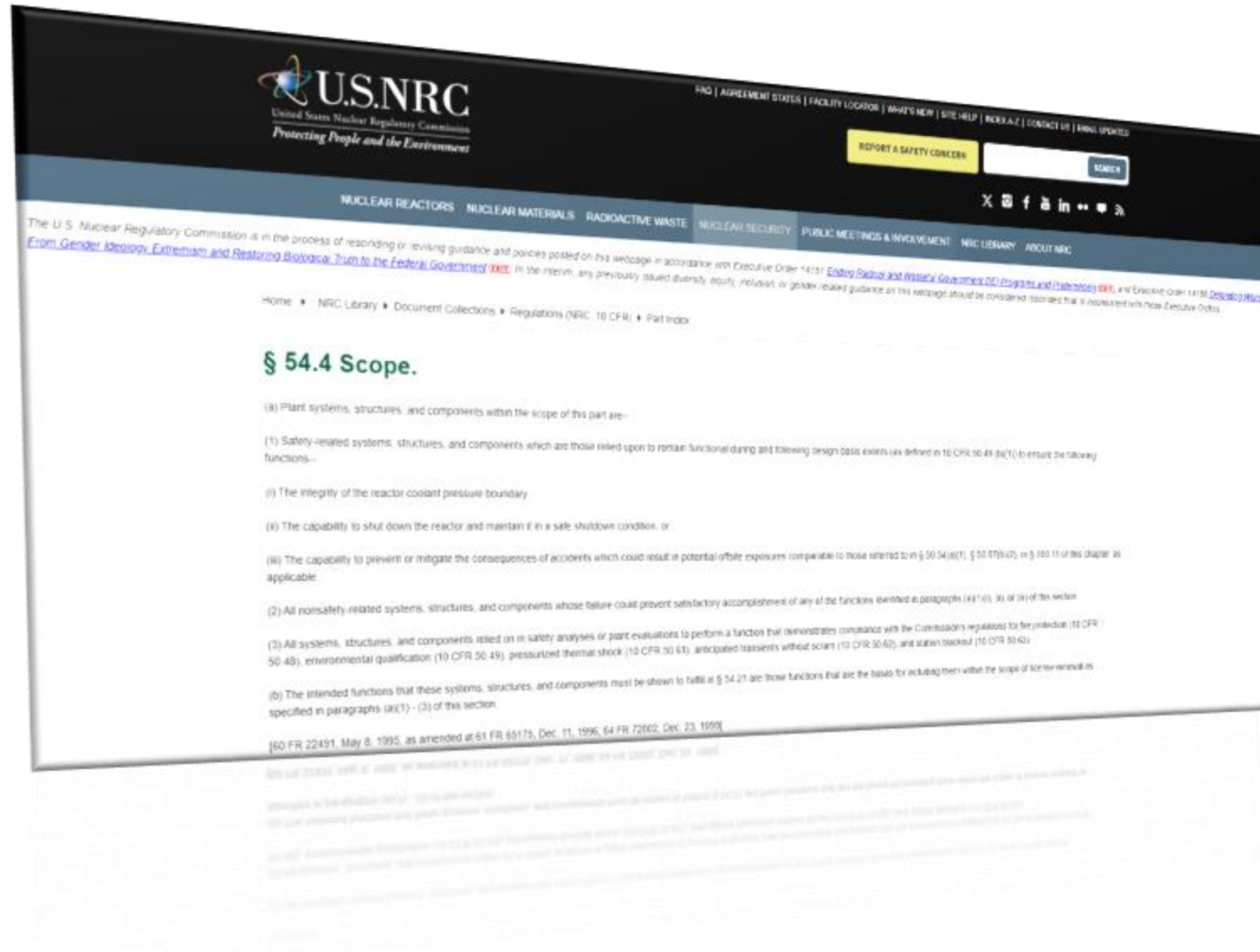


LRA Development Process: Integrated Plant Assessment

Integrated Plant Assessment (IPA) Flowchart



What Components are “in-scope”?



Scoping (10 CFR 54.4)

- (a)(1) – Safety-related systems, structures, and components (SSCs) which are relied upon to function during and following design-basis events to ensure the following functions:
 - Integrity of the reactor coolant pressure boundary
 - Capability to shut down the reactor and maintain it in a safe shutdown condition
 - Capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures (in accordance with other sections of the CFR)

Scoping (10 CFR 54.4)

- (a)(2) – all non-safety related SSC's whose failure could prevent satisfactory accomplishment of any function identified in 54.4(a)(1); examples:
 - Nonsafety-Related SSCs that [**functionally**] support Safety-Related functions
 - Nonsafety-Related Systems, Structures, and Components directly connected [**and provide structural support**] to Safety-Related SSC's
 - Nonsafety-Related Systems, Structures, and Components with potential for **spatial interaction** with Safety-Related SSCs

Scoping – (a)(2) *Continued*

[functional] support of Nonsafety-Related SSCs for a safety-related function.

Examples:

- A nonsafety-related deep well system credited by CLB for supporting SR Service Water system as back-up source of make-up water
- A nonsafety-related instrument air system that is relied upon to operate main steam relief valves.
- Nonsafety-related service air components that maintains air pressure on fuel pool gate seals

Scoping – (a)(2) *Continued*

Nonsafety-Related SSCs directly connected [**and provide structural support**] to Safety-Related SSCs:

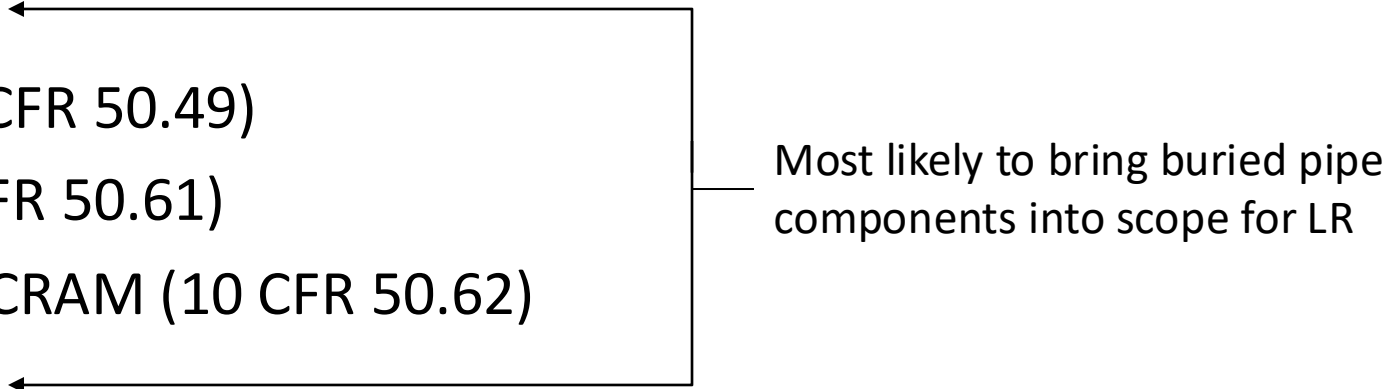
- Typically applies to Safety-Related / Nonsafety-Related interfaces
- For this condition, the Nonsafety-Related piping/components/supports up to and including the first seismic/equivalent anchor beyond the interface, would be in scope.

Scoping – (a)(2) *Continued*

Nonsafety-Related SSCs with potential for **spatial interaction** with Safety-Related SSCs:

- Where any Nonsafety-Related piping system (either not connected to Safety-Related piping, or beyond the equivalent anchor), could fail and affect a Safety-Related intended function.
- This is typically addressed on a ‘spaces’ approach.
 - Identify all the spaces that contain Safety-Related SSCs, and all the Nonsafety-Related SSCs in that same space would be in scope. Site-specific walkdowns are typically performed to confirm these conditions.
- Alternative: mitigative measures (pipe whip restraints, jet impingement shields, seismic supports, spray and drip shields, flood barriers) can be credited and managed for aging instead

Scoping (10 CFR 54.4)

- (a)(3) – all SSCs relied upon in safety analyses or plant evaluations to perform a function that demonstrates compliance with the Commissions regulations for:
 - Fire Protection (10 CFR 50.48)
 - Environmental Qualification (10 CFR 50.49)
 - Pressurized Thermal Shock (10 CFR 50.61)
 - Anticipated Transients without SCRAM (10 CFR 50.62)
 - Station Black Out (10 CFR 50.63)
- 
- Most likely to bring buried pipe components into scope for LR

Scoping Process

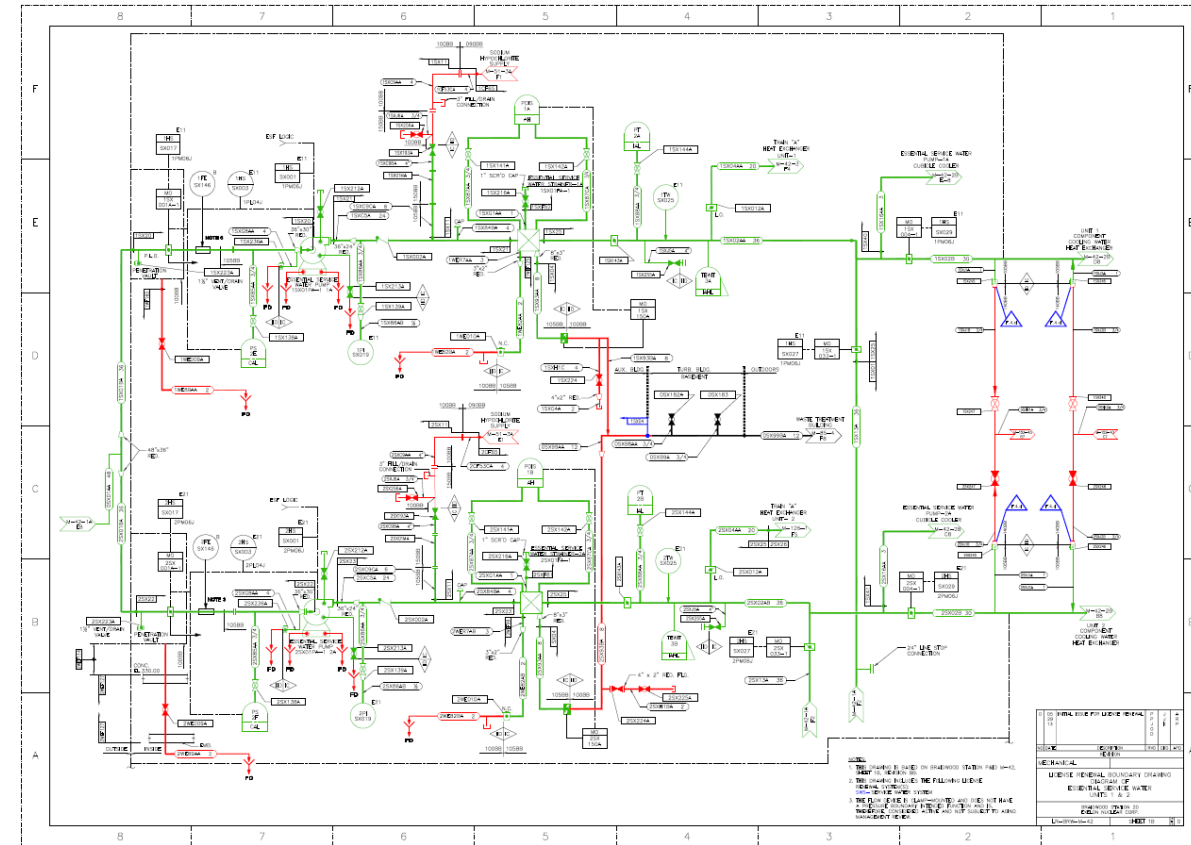
- Scoping is performed on a system and structure basis
- Every system and structure at the plant must be reviewed to determine if it is in scope of License Renewal
- Only those systems and structures that are in scope will be described in the License Renewal Application.
 - A Scoping Basis Document is prepared to document the LR intended functions of the system/structure
- The reasons for a system or structure not being in scope must also be documented in project basis documents

Intended Functions

1. Provide primary containment boundary. The Compressed Air System includes piping that penetrates the containment and includes equipment used for testing the personnel air lock seals. The containment penetrations, including containment isolation valves, are relied upon to ensure containment integrity. 10 CFR 54.4(a)(1)
2. Resist nonsafety-related SSC failure that could prevent satisfactory accomplishment of a safety-related function. Nonsafety-related service air components are required to provide functional support to maintain air pressure on the fuel pool gates inflatable seals. The Compressed Air System includes nonsafety-related piping that is directly attached and provides structural support to safety-related piping. The Compressed Air System also includes nonsafety-related water filled drain piping that has the potential for spatial interaction with safety-related SSCs. 10 CFR 54.4(a)(2)
3. Relied upon in safety analyses or plant evaluations to perform a function that demonstrates compliance with the Commission's regulations for Environmental Qualification (10 CFR 50.49). Solenoid valves and position switches associated with Compressed Air System air-operated containment isolation valves are included in the scope of the Environmental Qualification Program. 10 CFR 54.4(a)(3)

Scoping Process – Continued

- Boundary drawings are prepared for each mechanical system:
 - Drawings highlighted to show the system pictorially, typically using modified P&ID's
 - Boundary drawings show:
 - Systems that are within the scope of LR
 - Systems (& portions thereof) not within the scope of LR
 - Boundaries of system interfaces
 - Scoping function boundaries [(a)(1) vs (a)(2)]



Green: (a)(1) or (a)(3) function

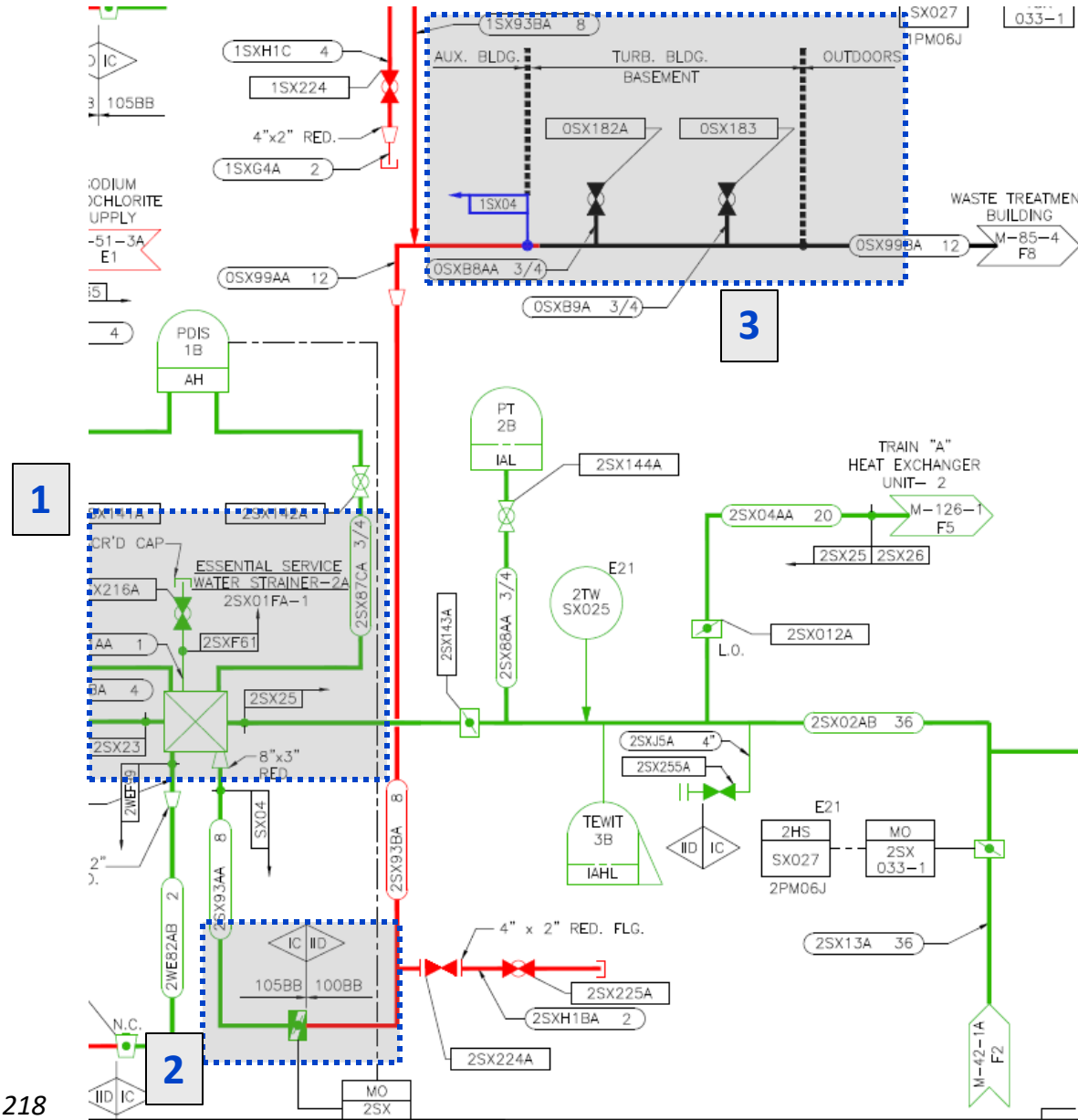
Source: ML13154A218

Red: (a)(2) function

Black: not within scope

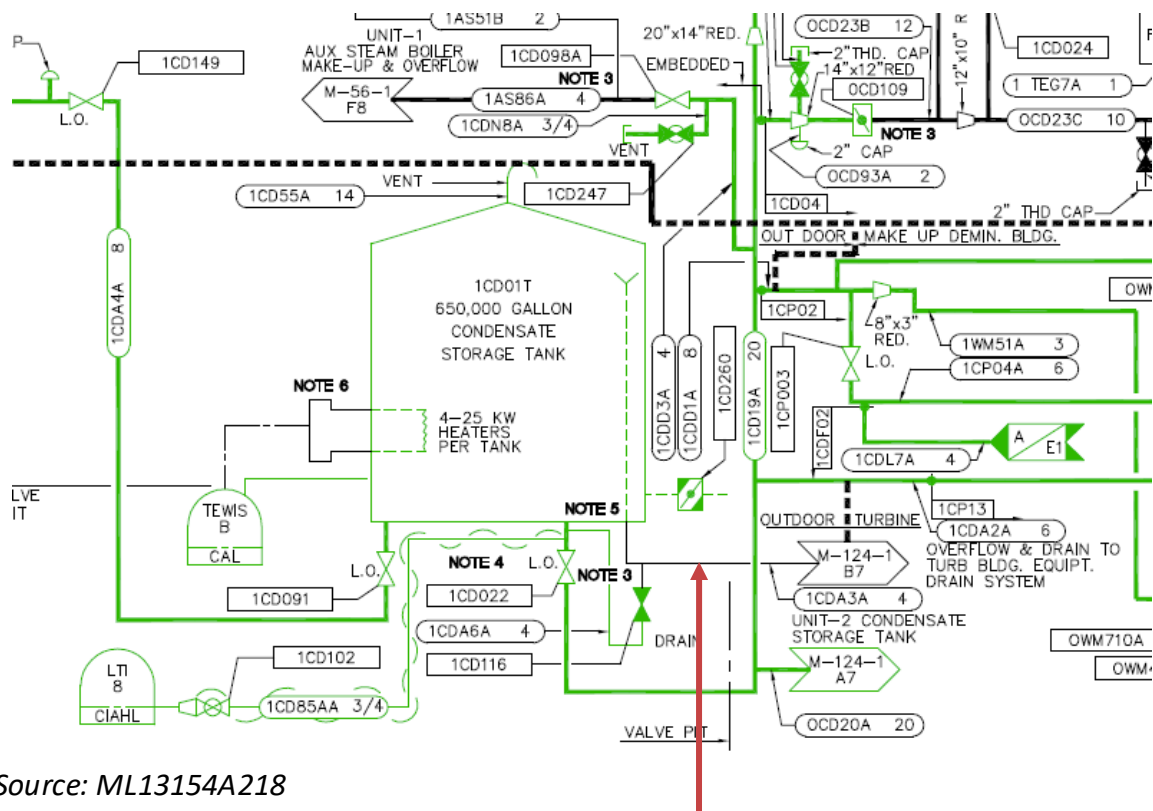
Scoping Example

- System:
 - Essential Service Water
-
- 1) ESW Strainer (Aux Bldg)
 - 2) Class break (a)(2)
 - 3) Building transition
 - In-scope for structural support beyond wall
 - Past anchor in Turb. Building, no longer in scope for spatial or structural support



Source: ML13154A218

Using Boundary Drawings



Source: ML13154A218

5. THE STANDPIPE SUPPORTS THE (A)(3) FUNCTION OF THE CONDENSATE STORAGE TANK BY MAINTAINING AN ADEQUATE WATER INVENTORY FOR THE AUXILIARY FEEDWATER SYSTEM. THE ATTACHED DRAIN PIPING IS NOT REQUIRED TO SUPPORT THIS FUNCTION. THE NONSAFETY-RELATED PIPING ATTACHED AT THIS BOUNDARY IS NOT REQUIRED FOR STRUCTURAL SUPPORT BECAUSE THE IN SCOPE PIPING IS NOT SAFETY-RELATED. THE PIPING BEYOND THIS BOUNDARY DOES NOT HAVE THE POTENTIAL FOR SPATIAL INTERACTION BECAUSE IT IS NOT LOCATED IN THE VICINITY OF SAFETY-RELATED COMPONENTS, AND IS, THEREFORE, NOT WITHIN THE SCOPE OF LICENSE RENEWAL.



Source: February BPIG 2018, Presentation C01

Screening (10 CFR 54.21)

- After Scoping is complete, next step in Integrated Plant Assessment (IPA) is Screening:
 - Determine which components are ACTIVE vs PASSIVE and LONG-LIVED
 - Passive: components that perform an intended function without moving parts or a change in configuration or properties
 - Long Lived: components that are not subject to replacement based on a qualified life or specified time period

Screening (10 CFR 54.21)

- Examples of Passive Intended Functions:

- Pressure boundaries
- Leakage boundaries
- Heat transfer

- Examples of Passive Components:

- Piping
- Heat exchangers
- Valves (valve body = passive, valve actuator = active)
- Pump casing
- Restricting orifice

- Examples of Short-Lived Components:

- Oil, grease
- Fire extinguishers
- Any component planned and managed to be periodically replaced
- Fuel assemblies



Aging Management Review (AMR) Process

Aging Management Review

For each system:

- 1) Group components by generic component type
- 2) Use scoping results to identify component intended functions
 - Pressure boundary, leakage boundary, structural support, etc.
- 3) Determine materials of construction for each component type/function
 - Group materials by generic type, based on similarities in aging effect applicability
- 4) Determine different internal and external surface environments
- 5) Assess aging effects / mechanism
- 6) Determine which existing or new Aging Management Program currently or will manage the effects of aging

Aging Management Review – Outcome

- AMR documented 9-column tables for each system
- The AMR includes a comparison of the aging effect, and the utility identified program to the same information in the GALL Report

Table 3.4.2-3 Main Condensate and Feedwater System (Continued)

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-1801 Item	Table 1 Item	Notes
Piping, piping components, and piping elements	Leakage Boundary	Stainless Steel	Treated Water (Internal)	Loss of Material	Water Chemistry (B.2.1.2)	VIII.D1.SP-87	3.4.1-16	A
			Treated Water > 140 F (Internal)	Cracking	One-Time Inspection (B.2.1.20)	VIII.D1.SP-88	3.4.1-11	A
					Water Chemistry (B.2.1.2)	VIII.D1.SP-88	3.4.1-11	A
				Cumulative Fatigue Damage	TLAA	VII.E3.A-62	3.3.1-2	A, 4
				Loss of Material	One-Time Inspection (B.2.1.20)	VIII.D1.SP-87	3.4.1-16	A
					Water Chemistry (B.2.1.2)	VIII.D1.SP-87	3.4.1-16	A
	Pressure Boundary	Aluminum Alloy	Air - Outdoor (External)	Loss of Material	External Surfaces Monitoring of Mechanical Components (B.2.1.23)	VIII.H.SP-147	3.4.1-35	A
					One-Time Inspection (B.2.1.20)	VIII.D1.SP-90	3.4.1-16	A
			Treated Water (External)	Loss of Material	Water Chemistry (B.2.1.2)	VIII.D1.SP-90	3.4.1-16	A
					One-Time Inspection (B.2.1.20)	VIII.D1.SP-90	3.4.1-16	A
		Carbon Steel	Treated Water (Internal)	Loss of Material	One-Time Inspection (B.2.1.20)	VIII.D1.SP-90	3.4.1-16	A
					Water Chemistry (B.2.1.2)	VIII.D1.SP-90	3.4.1-16	A
			Air - Indoor Uncontrolled (External)	Loss of Material	External Surfaces Monitoring of Mechanical Components (B.2.1.23)	VIII.H.S-29	3.4.1-34	A
					External Surfaces Monitoring of Mechanical Components (B.2.1.23)	VIII.H.S-41	3.4.1-34	A
			Air with Borated Water Leakage (External)	Loss of Material	Boric Acid Corrosion (B.2.1.4)	VIII.H.S-30	3.4.1-4	A
					External Surfaces Monitoring of Mechanical Components (B.2.1.23)	VIII.H.S-29	3.4.1-34	A
			Soil (External)	Loss of Material	Buried and Underground Piping (B.2.1.28)	VIII.E.SP-145	3.4.1-47	B

Utility AMR

Comparison to
GALL / GALL-SLR

Aging Management Programs

Credited AMPs include both New and Existing Program

Existing Plant Programs

- ASME Section XI, Subsection IWB, IWC, IWD
- Flow-Accelerated Corrosion Program
- Buried Pipe Program
- Steam Generators
- Open-Cycle Cooling Water
- Water Chemistry
- Boric Acid Corrosion

New Programs

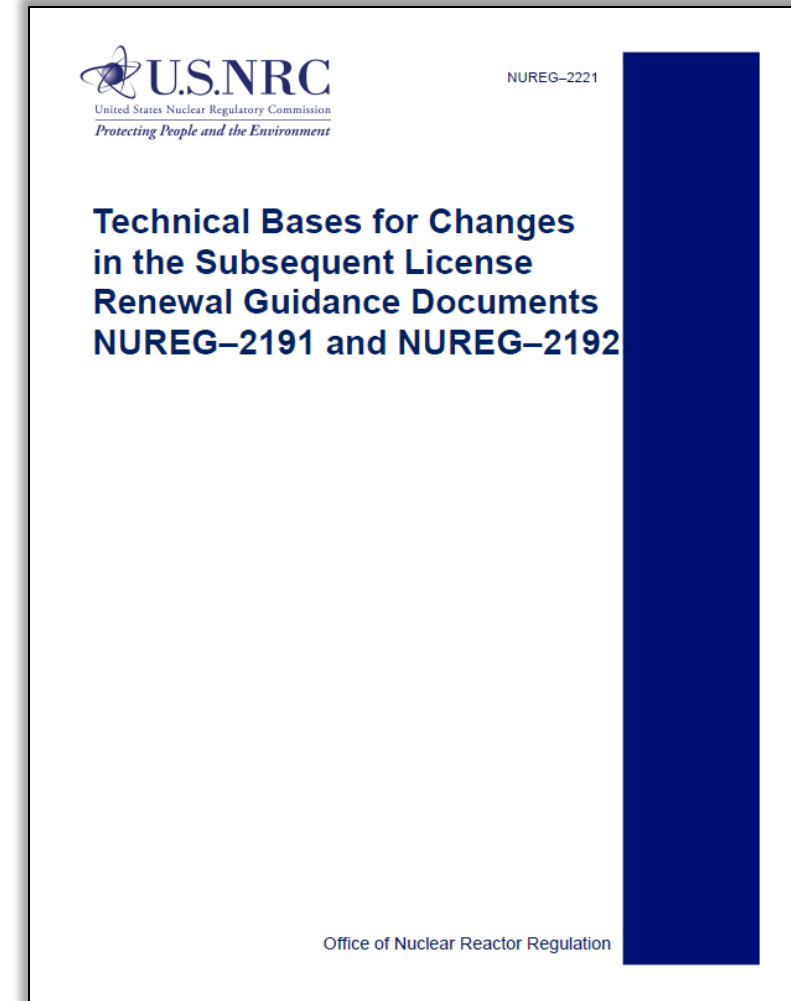
- Selective Leaching
- One-Time Inspection
- External Surfaces Monitoring of Mechanical Components



Items of Interest & Awareness

GALL / GALL-SLR Revision

- GALL / GALL-SLR revisions undergo public comment periods
 - Anyone can submit comments for recommended changes to the programs
 - NRC disposition of comments are documented in NUREGs
 - NUREG-1832: Analysis of Public Comments on GALL R1
 - NUREG-1950: Disposition of Public Comments on GALL R2
 - NUREG-2222: Disposition of Public Comments on GALL-SLR
 - NUREG-2191, Appendix C: Disposition of Public Comments on Draft GALL-SLR R1
- Each GALL / GALL-SLR revision is accompanied by a Technical Basis for change NUREG
 - NUREG-1833: Technical Basis for Revision to GALL R1
 - NUREG-1950: Technical Basis for Changes to GALL R2
 - NUREG-2221: Technical Basis for GALL-SLR Changes
 - NUREG-2221, Supplement 1: Technical Basis for GALL-SLR R1 Changes



Long-Term Operations Wiki – [LTO.EPRI.COM](https://lto.epri.com)

- Designed as an overview of the process to apply for LTO both in the US and Internationally and the steps that are involved both prior to PEO and after PEO to maintain the aging management programs.
- Publicly available with links to specific information that may require a login to access.



[LTO Beta](#)
[LTO Home Page](#)
[LTO Acronyms and Descriptions](#)
[LTO Regulatory and Guidance Documents](#)
[LR Application Process](#)
[AMPs](#)
[TLAA](#)
[Implementation](#)
[Review of LRA \(USNRC and International\)](#)
[Pre-PEO](#)
[Post-PEO](#)
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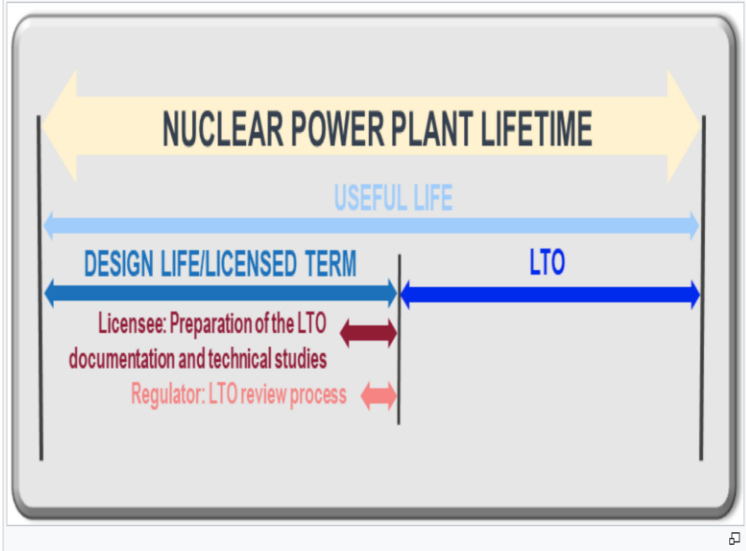
Nuclear Power Plant Long-Term Operations (LTO)

LTO Acronyms and Descriptions

This wiki is intended to provide a structured framework for presenting information relevant to extending the life of nuclear plants beyond their initial authorized or designed period both in the United States (US) and internationally.

Long- United States of America can be defined as operation, justified by a comprehensive safety assessment, that goes beyond a previously established time frame corresponding to initial design assumptions (typically referred to as "original design lifetime" and usually 30-40 years depending on the design). These initial assumptions, however, do not represent a technical constraint inhibiting longer operating time periods and should not be confounded with the remaining, useful life of the facility, which is periodically re-evaluated, taking into account the actual plant conditions and the latest available knowledge.

The International Atomic Energy Agency (IAEA) defined the term long-term operation as "the operation beyond an established period defined by the license term, the original plant design, relevant standards, or national regulations" ([Specific Safety Guide SSG-48, "Ageing Management and Development of a Programme for Long Term Operation of Nuclear Power Plants \(2\)"](#)).



The diagram illustrates the timeline of a nuclear power plant's operation. A large yellow arrow at the top represents the 'NUCLEAR POWER PLANT LIFETIME'. Below it, a blue arrow represents the 'USEFUL LIFE'. Within the Useful Life, a blue arrow represents the 'DESIGN LIFE/LICENSED TERM'. Following the Design Life, a blue arrow represents the 'LTO' (Long-Term Operations) period. Below the Design Life/Licensed Term arrow, two red double-headed arrows indicate the 'Licensee: Preparation of the LTO documentation and technical studies' and the 'Regulator: LTO review process'.

EPRI LTO Contact Information

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Long-Term Operations Program:
Assessment of Research and Development Supporting Aging Management Programs for Long-Term Operation. EPRI, Palo Alto, CA: 2013. 3002000576.



Visit EPRI LTO Wiki at LTO.EPRI.COM



TOGETHER...SHAPING THE FUTURE OF ENERGY®

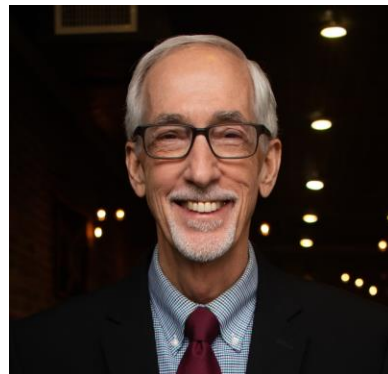
17-01 – AMPs to develop & Implement within 5 years

IAEA National Workshop on Ageing Management during Early Stages of Operation



Garry G. Young
Technical Executive

Barakah NPP
Abu Dhabi, UAE
September 17, 2025



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North Anna Units 1 and 2	08/24/20	08/28/24	04/01/2038 (Unit 1) 08/21/2040 (Unit 2)
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Oconee Nuclear Station Units 1, 2, and 3	06/07/2021	03/31/2025	02/06/2033 (Unit 1) 10/06/2033 (Unit 2) 07/19/2034 (Unit 3)

License Renewal Applications – Existing Programs

Palo Verde, Units 1, 2, 3, License Renewal Application – Appendix B

- XI.M1 ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD
- XI.M2 Water Chemistry
- XI.M3 Reactor Head Closure Studs
- XI.M10 Boric Acid Corrosion
- XI.M11A Nickel-Alloy Penetration Nozzles Welded To The Upper Reactor Vessel Closure Heads of Pressurized Water Reactors
- XI.M17 Flow-Accelerated Corrosion
- XI.M18 Bolting Integrity
- XI.M19 Steam Generator Tube Integrity
- XI.M20 Open-Cycle Cooling Water System
- XI.M21 Closed-Cycle Cooling Water System
- XI.M23 Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems
- XI.M26 Fire Protection
- XI.M27 Fire Water System
- XI.M30 Fuel Oil Chemistry
- XI.M31 Reactor Vessel Surveillance
- XI.M35 One-Time Inspection of ASME Code Class 1 Small-Bore Piping
- XI.M39 Lubricating Oil Analysis
- XI.E2 Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits
- XI.S1 ASME Section XI, Subsection IWE
- XI.S2 ASME Section XI, Subsection IWL
- XI.S3 ASME Section XI, Subsection IWF
- XI.S4 10 CFR 50, Appendix J
- XI.S5 Masonry Wall Program Masonry Wall Program
- XI.S6 Structures Monitoring Program
- XI.S7 RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants
- X.M1 Metal Fatigue of Reactor Coolant Pressure Boundary
- X.E1 Environmental Qualification (EQ) of Electrical Components
- X.S1 Concrete Containment Tendon Prestress

License Renewal Applications – Existing Programs

Vogtle Electric Generating Plant (VEGP) Units 1 & 2, License Renewal Application – Appendix B

VEGP Aging Management Programs	New or Existing	App. B Section
VEGP Mechanical Aging Management Programs		
ACCW System Carbon Steel Components Program	New	B.3.1
Bolting Integrity Program	New	B.3.2
Boric Acid Corrosion Control Program	Existing	B.3.3
Buried Piping and Tanks Inspection Program	New	B.3.4
CASS RCS Fitting Evaluation Program	New	B.3.5
Closed Cooling Water Program	Existing	B.3.6
Diesel Fuel Oil Program	Existing	B.3.7
External Surfaces Monitoring Program	New	B.3.8
Fire Protection Program	Existing	B.3.9
Flow-Accelerated Corrosion Program	Existing	B.3.10
Flux Thimble Tube Inspection Program	Existing	B.3.11
Generic Letter 89-13 Program	Existing	B.3.12
Inservice Inspection Program	Existing	B.3.13
Nickel Alloy Management Program for Non-Reactor Vessel Closure Head Penetration Locations	New	B.3.14
Nickel Alloy Management Program for Reactor Vessel Closure Head Penetrations	Existing	B.3.15
Oil Analysis Program	Existing	B.3.16
One-Time Inspection Program	New	B.3.17
One-Time Inspection Program for ASME Class 1 Small Bore Piping	New	B.3.18
One-Time Inspection Program for Selective Leaching	New	B.3.19
Overhead and Refueling Crane Inspection Program	Existing	B.3.20
Periodic Surveillance and Preventive Maintenance Activities	Existing	B.3.21
Piping and Duct Internal Inspection Program	New	B.3.22

VEGP Aging Management Programs	New or Existing	App. B Section
Reactor Vessel Closure Head Stud Program	Existing	B.3.23
Reactor Vessel Internals Program	New	B.3.24
Reactor Vessel Surveillance Program	Existing	B.3.25
Steam Generator Tubing Integrity Program	Existing	B.3.26
Steam Generator Program for Upper Internals	Existing	B.3.27
Water Chemistry Control Program	Existing	B.3.28
VEGP Civil/Structural Aging Management Programs		
10 CFR 50 Appendix J Program	Existing	B.3.29
Inservice Inspection Program – IWE	Existing	B.3.30
Inservice Inspection Program – IWL	Existing	B.3.31
Structural Monitoring Program	Existing	B.3.32
Structural Monitoring Program – Masonry Walls	Existing	B.3.33
VEGP Electrical Aging Management Programs		
Non-EQ Cables and Connections Program	New	B.3.34
Non-EQ Inaccessible Medium-Voltage Cables Program	New	B.3.35
Non-EQ Cable Connections One-Time Inspection Program	New	B.3.36
VEGP TLA Aging Management Programs		
Environmental Qualification Program	Existing	B.3.37
Fatigue Monitoring Program	Existing	B.3.38

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Long-Term Operations Program:
*Assessment of Research and Development
Supporting Aging Management Programs
for Long-Term Operation.* EPRI, Palo Alto,
CA: 2013. 3002000576.



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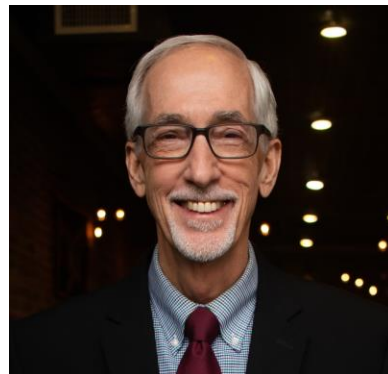
17-06 – Buried Piping Aging Management Program

IAEA National Workshop on Ageing Management
during Early Stages of Operation



Garry G. Young
Technical Executive

Barakah NPP
Abu Dhabi, UAE
September 17, 2025



Vogtle 3 & 4 Status

- Scoping has been done
- Mostly plastic pipe
- Few carbon steel pipes are coated
- Not expecting any exterior corrosion concerns for many, many years

Contact:

Ryan Linebarger

Fire Protection Engineer

Southern Nuclear Co - Vogtle 3&4

RSLINEBA@SOUTHERNCO.COM

Actions Completed:

- Construction photos of installation of major pipelines saved to program file folder (does not include smaller lines like service air lines)
- List of underground pipes with basic data (system code, pipe material, pipe contents, program scope applicability)
- PMs generated for monitoring leak detection indication (float switches, capacitance leak alarms)
- Underground surveys capture as-built location
- Pipe isometric drawings for underground pipe in site document database.

Planned Actions:

- Input detailed pipe data into BPWorks using supplemental staff

Not Planned:

- Any excavation inspections (opportunistic inspections only)

Vogtle 3 & 4 Example Buried Piping Data List

System	Material	Expected risk ranking in BP Works	Surveillance / PMs	Asset Management Plan	Notes
CAS-Compressed and Instrument Air System	Instrument Air is Stainless Steel	Low	Opportunistic inspection only	No action required.	Air supply to SWS cold basin level bubbler instrument.
	Service Air is Carbon Steel, corrosion resistant coating	Low			Expected to be encased in concrete or flowable fill.
CFS – Chemical Feed System	Stainless steel pipe inside Stainless steel pipe with leak detection	Low	Leak detection check – 5Y, start 1 year after startup	No action required.	Environmental hazard.
	Plastic pipe (PVC) inside Plastic pipe (PVC) with leak detection	Zero			
	Alloy 20 pipe inside Plastic pipe (PVC) with leak detection	Zero			
CWS-Circulating Water System	Large diameter Prestressed Concrete pipe	Low	Opportunistic inspection only	No action required.	CWS pump discharge to condenser inlet, and Condenser outlet to cooling tower
	Some Plastic HDPE	Zero			CWS blowdown control valve to WWRB
	Wrapped carbon steel	Low			24” supply and return to CMS & TCS heat exchangers
CWS - Circulating Water System - Injection Sub-System	Plastic pipe (HDPE) inside Plastic pipe (HDPE) with leak detection	Zero	Leak detection check – 5Y, start 1 year after startup	No action required.	Not currently in standards due to dilution flow but included as potential environmental hazard.
DOS-Standby Diesel and Auxiliary Boiler Fuel Oil System	Carbon steel pipe inside HDPE with leak detection	Low	Leak detection check – weekly Operator Rounds	No action required.	Environmental hazard. Fuel Oil transfer package to Day Storage Tanks
DRS-Storm Drain System	Plastic HDPE pipe	Zero	Opportunistic inspection only	No action required.	(Drawings under SV0-0000-XD-)
DWS – Demineralized Water Transfer and Storage System	Stainless Steel pipe	Low	Opportunistic inspection only	No action required.	DWST to Annex Building. Should be incased in concrete or flowable fill.
FPS – Fire Protection System	Plastic HDPE pipe	Zero	Opportunistic inspection only	No action required.	

Vogtle 3 & 4 Example Buried Piping Data List

System	Material	Expected risk ranking in BP Works	Surveillance / PMs	Asset Management Plan	Notes
PGS – Plant Gas System	Carbon steel underground pipe incased in reinforced concrete. (CO2, LP-N2, HP-N2)	Low	Opportunistic inspection only	No action required.	
	Stainless steel underground pipe incased in reinforced concrete. (LP-H2)	Low			
PWS – Potable Water System	Plastic HDPE pipe. Copper Type K to DG building	Low	Opportunistic inspection only	No action required.	
RWS – Raw Water System	Plastic HDPE pipe	Zero	Opportunistic inspection only	No action required.	RWS to Circ Water & WWS Dilution
SDS – Sanitary Drainage System	Mostly Plastic HDPE pipe	Zero	Opportunistic inspection only	No action required.	Rest rooms to sewage treatment
	Ferrous Cast Iron from TB/Annex to first manhole. Corrosion resistant coating	Low			
SWS – Service Water System	Carbon steel inside a concrete trench	Low	Opportunistic inspection only	No action required.	Trench is 7 feet high and 20 feet across to allow walking around pipe. SW pump suction line and return line to cooling tower
WLS – Liquid Radwaste System	Stainless Steel pipe inside plastic HDPE pipe with periodic leak detection points	Low	Leak detection check – 1 Month (System inspection)	No action required.	Regulatory material. Annex Building to connection with WWS
WWS – Waste Water System	Carbon steel pipe, corrosion resistant coating	Low	Opportunistic inspection only	No action required.	TB to Oil Separator Diesel Bldg to Oil Separator Diesel Fuel Oil Area Sump to Oil Separator Transformer Area Sump to Oil Separator DOS area drains to Oil Separator
	Plastic HDPE pipe	Zero	Opportunistic inspection only	No action required.	Oil Separator to WWRB WWRB to discharge point
YFS – Yard Fire Water System	Plastic HDPE pipe	Zero	Opportunistic inspection only	No action required.	

Vogtle 3 & 4 Example Buried Piping Data List

System	Material	Expected risk ranking in BP Works	Surveillance / PMs	Asset Management Plan	Notes
ZRS – Offsite Retail Power System - Diesel Fuel Transfer Piping near Bldg 010	Carbon steel pipe inside HDPE with leak detection	Low	No direct inspection. Weekly operator rounds inspect tank levels to check for unexpected change	No action required.	Environmental hazard.

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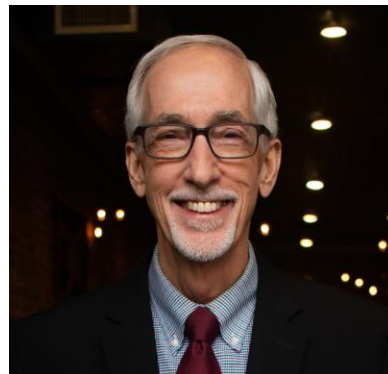
17-08 – Reactor Vessel Internals Aging Management Program

IAEA National Workshop on Ageing Management
during Early Stages of Operation



Garry G. Young
Technical Executive

Barakah NPP
Abu Dhabi, UAE
September 17, 2025



What are the Key Technical Issues for 60+ Years of Operation?

- **Reactor pressure vessel – neutron embrittlement**

- Trends for high fluence levels
- Sufficiency of material surveillance program capsules

- **Reactor vessel internals - high fluence effects**

- Irradiation-assisted stress corrosion cracking
- Loss of fracture toughness; void swelling

- **Concrete and containment performance**

- Radiation and high temperature exposure
- Alkali-silica reaction

- **Electrical cables**

- Environmental qualification
- In-service testing
- Long-term submersion



Each issue was addressed in SLR guidance



Research continues to validate and update guidance, as necessary

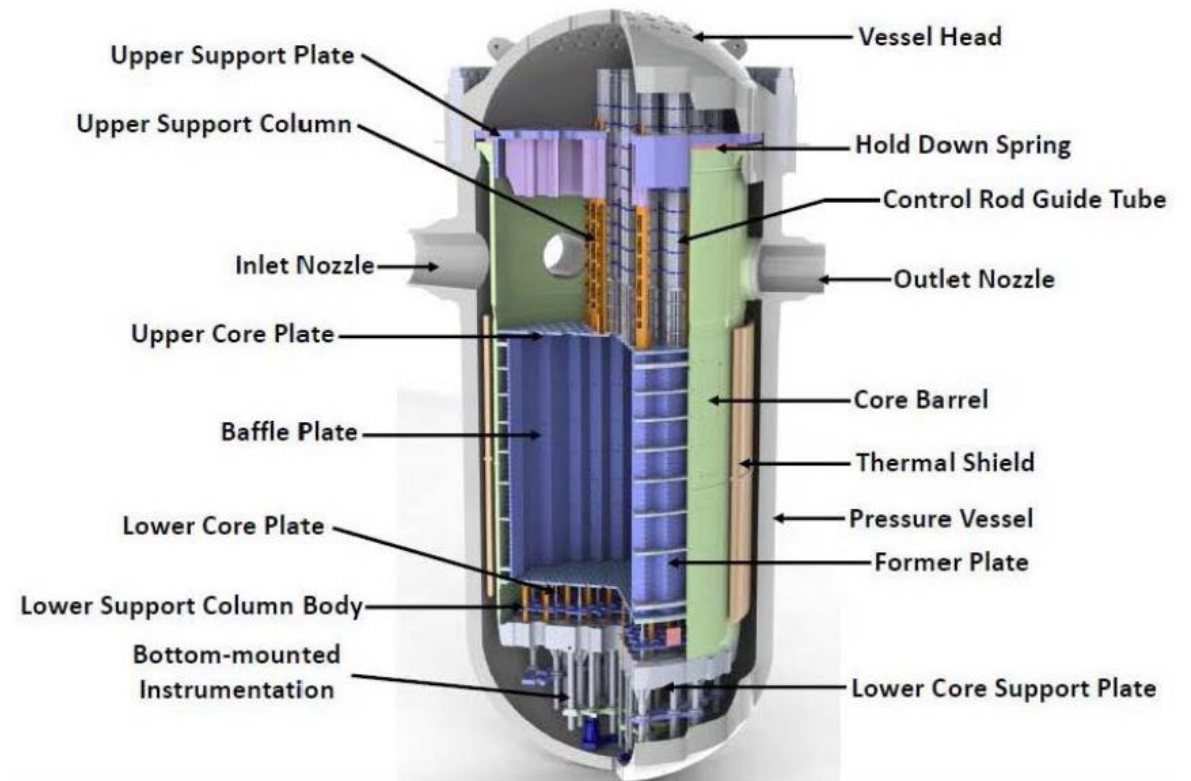
Industry List of Major Issues for Long Term Operation

Update on MRP-227 Revision 2 (NRC approval)

EPRI Activities on Reactor Internals Aging Management

PWR Internals Inspection & Evaluation Guide Rev.0 issued in 2008

- NRC-approved version of PWR Internals Inspection and Evaluation (I&E) Guidelines, MRP-227-A (Product ID 1022863) was issued in Dec.2011
 - Rigorous exams done in year-40, 50, 60 and year-70 ISI RFOs
 - Most RVI components have not been inspected since original fabrication/construction
- Updated guidelines:
 - MRP-227, Rev.1 was issued in October 2015 (Product ID 3002005349)
 - NRC approved Safety Evaluation received in May 2019
 - MRP-227, Rev.1-A was issued in December 2019 (Product ID 3002017168)
 - MRP-227, Rev.2 was issued in September 2021 (ID 3002020105) and submitted to US NRC
 - *Incorporated "SLR gap analysis" in MRP 2018-022*
 - NRC safety approval received in Jan. 2025
 - Revision 2-A was published in May 2025



Overview of Typical Westinghouse Internals

NRC Safety Evaluation Update for MRP-227, Rev. 2

Summary/Conclusion of NRC's Safety Evaluation:

Ref. NRC ADAMS ML24305A007

4.0 LIMITATIONS AND CONDITIONS

The staff did not identify the need to include or issue any limitations, conditions, or A/LAIs in relation to the contents of the MRP-227, Revision 2, as supplemented.

5.0 CONCLUSION

To meet 10 CFR 54.21(a)(3), an application must, for each structure and component identified as subject to an aging management review under 10 CFR 54.21(a)(1), demonstrate that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation. Based on its review, the NRC staff finds that MRP-227, Revision 2, provides an acceptable means of demonstrating that the impacts of the effects of aging on the structural integrity of PWR RVI components will be adequately managed during the 40 to 80 -year license renewal period of extended operation.

The NRC staff verified that the time-dependent assessments of aging RVI components adequately address a minimum 80-year cumulative period of licensed operations for the PWR facilities.

For WEC-design PWR RVI AMPs, the NRC staff also finds that the updated I&E criteria for BFBs appropriately incorporates the tier-based I&E criteria for the WEC-design BFBs established in the EPRI letter MRP 2017-009, "Transmittal of NEI-03-08 'Needed' Interim Guidance Regarding Baffle-Former Bolt Inspections for PWR Plants as Defined in Westinghouse NSAL 16-01, Revision 1," dated March 15, 2017 (EPRI 2017c). The NRC staff finds that MRP-227, Revision 2, provides an acceptable basis for the NRC staff's closure of the prior A/LAI #1 topic and action that was previously issued in Section 3.1.3.7 of the NRC staff's SE for MRP-227, Revision 1-A.

ACRS Materials SC Meeting was held on 11/21/24 to discuss SE - see ML24310A204
- Meeting Transcript was published in ML24337A186

Example Primary Components in MRP-227 Revision 2-A

Westinghouse-design PWRs

Westinghouse Primary Table for MRP-227, Rev. 2-A

Table 4-3
Westinghouse plants Primary components

Primary Item	Applicability (Note 12)	Effect (Mechanism)	Expansion Link (Note 2)	Examination Method/Frequency (Note 2)	Examination Coverage
W2. Control Rod Guide Tube Assembly Lower flange welds	All plants	Cracking (SCC, IASCC, fatigue) Aging management (IE and TE)	W2.1. Remaining CRGT assembly lower flange welds W2.2. BMI column bodies	Enhanced visual (EVT-1) examination to determine the presence of cracklike surface flaws in flange welds no later than two refueling outages from the beginning of the license renewal period and subsequent examination at a 10- year interval.	100% of outer (accessible) CRGT lower flange weld surfaces and 0.25 in. (0.64 cm) of the adjacent base metal on the individual periphery CRGT assemblies. (Note 3) See Figure 4-31.
W3. Core Barrel Assembly Upper flange weld (UFW)	All plants	Cracking (SCC)	W3.3. Lower flange weld (LFW) W3.2. Upper axial welds (UAW) W3.4. Lower support forging or casting (secondary expansion of W3.3. LFW)	Enhanced visual (EVT-1), full volume/full thickness volumetric (UT), or surface (ET) examination, no later than two refueling outages from the beginning of the license renewal period and subsequent examination at a 10- year interval. (Note 4)	100% of the accessible weld length of both surfaces (ID surface and OD surface) of the UFW and 0.75 in. (1.91 cm) of adjacent base metal shall be examined. If UT is performed, it need only be completed from one surface, either ID or OD. (Note 5) See Figure 4-32.
W3a. Core Barrel Assembly Upper girth weld (UGW)	All plants	Cracking (SCC)	W3.2. Upper axial weld(s) (UAW) W3.3. Lower flange weld (LFW) W3.4. Lower support forging or casting (secondary expansion of W3.3 LFW)	Enhanced visual (EVT-1), full volume/full thickness volumetric (UT), or surface (ET) examination, no later than two refueling outages from the beginning of the license renewal period and subsequent examination at a 10- year interval. (Note 4)	100% of the accessible weld length of both surfaces (ID surface and OD surface) of the UGW and 0.75 in. (1.91 cm) of adjacent base metal shall be examined. If UT is performed, it need only be completed from one surface, either ID or OD. (Note 5) See Figure 4-32.

*
MRP 2023-005

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Westinghouse Primary Table for MRP-227, Rev. 2-A

Table 4-3 (continued)
Westinghouse plants Primary components

Primary Item	Applicability (Note 12)	Effect (Mechanism)	Expansion Link (Note 2)	Examination Method/Frequency (Note 2)	Examination Coverage
W4. Core Barrel Assembly Lower girth weld (LGW)	All plants	Cracking (SCC, IASCC) Aging management (IE)	W4.1. Upper core plate W4.4. Lower support column bodies (cast, non-cast) W4.2. Middle axial weld(s) (MAW) W4.3. Lower axial weld(s) (LAW)	Enhanced visual (EVT-1), full volume/full thickness volumetric (UT), or surface (ET) examination, no later than two refueling outages from the beginning of the license renewal period and subsequent examination at a 10- year interval. (Note 4)	100% of the accessible weld length of the OD surface of the LGW and 0.75 in. (1.91 cm) of adjacent base metal shall be examined (ID surface is inaccessible for visual/ET-based surface exams due to baffle-former assembly). If UT is performed, it shall be from the OD surface. Accessibility may be limited by the thermal shield or neutron pads, but removal of the thermal shield or neutron pads is not required. (Note 6) See Figure 4-32.
W5. Baffle-Former Assembly Baffle-edge bolts	All plants with baffle-edge bolts	Cracking (IASCC, fatigue) that results in: • Lost or broken locking devices • Failed or missing bolts • Protrusion of bolt heads Aging management (IE and ISR) (Note 7)	None	Visual (VT-3) examination, with baseline examination no later than 40 effective full power years (EFPY) and subsequent examinations at a 10-year interval.	Bolts and locking devices on high fluence seams. 100% of components accessible from core side. See Figure 4-33.
W6a. Baffle-Former Assembly Baffle-former bolts (Tier 1) (Note 8)	Tier 1 plants (4-loop downflow) [47]	Cracking (IASCC, fatigue) Aging management (IE and ISR) (Note 7)	W6.2. Lower support column bolts W6.1. Barrel-former bolts (secondary expansion of W6.2. Lower support column bolts if large clusters of degraded baffle-former bolts are not detected; see Table 4-6)	Baseline volumetric (UT) examinations have been completed at all these plants. Subsequent examination is dependent on the plant design and the results of the baseline inspection. (Note 9)	100% of accessible bolts. (Note 10) See Figure 4-34.

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Westinghouse Primary Table for MRP-227, Rev. 2-A

Table 4-3 (continued)
Westinghouse plants Primary components

Primary Item	Applicability (Note 12)	Effect (Mechanism)	Expansion Link (Note 2)	Examination Method/Frequency (Note 2)	Examination Coverage
W6b. Baffle-Former Assembly Baffle-former bolts (Tier 2) (Note 8)	Tier 2 plants (2-loop and 3-loop downflow plants) [47]	Cracking (IASCC, fatigue) Aging management (IE and ISR) (Note 7)	W6.2. Lower support column bolts W6.1. Barrel-former bolts (secondary expansion of W6.2. Lower support column bolts if large clusters of degraded baffle-former bolts are not detected; see Table 4-6)	Baseline volumetric (UT) examination no later than 30 EFPY. Subsequent examination is dependent on the plant design and the results of the baseline inspection. (Note 9)	100% of accessible bolts. (Note 10) See Figure 4-34.
W6c. Baffle-Former Assembly Baffle-former bolts (Tier 3 and 4) (Note 8)	Tier 3 plants (converted upflow plants) (Note 11) Tier 4 plants (design upflow plants) [47]	Cracking (IASCC, fatigue) Aging management (IE and ISR) (Note 7)	W6.2. Lower support column bolts W6.1. Barrel-former bolts (secondary expansion of W6.2. Lower support column bolts if large clusters of degraded baffle-former bolts are not detected; see Table 4-6)	Baseline volumetric (UT) examination no later than 35 EFPY. Subsequent examination is dependent on the plant design and the results of the baseline inspection. (Note 9)	100% of accessible bolts. (Note 10) See Figure 4-34.
W7. Baffle-Former Assembly Assembly (includes baffle plates, bracket bolts (if applicable), corner bolts (if applicable), and indirect effects of void swelling in former plates)	All plants	Distortion (void swelling), or Cracking (IASCC) that results in: - Abnormal interaction with fuel assemblies - Gaps between plates - Vertical displacement of baffle plates - Broken or damaged bolts	None	Visual (VT-3) examination to check for evidence of distortion, with baseline examination no later than 40 EFPY and subsequent examinations at a 10-year interval.	Core side surface: - High fluence baffle joints - Top and bottom edge of baffle plates - Bolts and locking devices See Figure 4-35.

OE

OE
(baffle jetting)

Westinghouse Primary Table for MRP-227, Rev. 2-A

Table 4-3 (continued)
Westinghouse plants Primary components

Primary Item	Applicability (Note 12)	Effect (Mechanism)	Expansion Link (Note 2)	Examination Method/Frequency (Note 2)	Examination Coverage
W8. Alignment and Interfacing Components Internals hold-down spring	All plants with Type 304 stainless steel hold-down springs	Distortion (loss of load due to stress relaxation)	None	Direct measurement of spring height within three cycles of the beginning of (before or after) the license renewal period. If the first set of measurements is not sufficient to assess remaining life to the end of licensed plant life (whether to the end of the first period of extended operation or the end of SLR), additional spring height measurements will be required.	Measurements should be taken at several points around the circumference of the spring, with a statistically adequate number of measurements at each point to minimize uncertainty. See Figure 4-36.
W9. Thermal Shield Assembly Thermal shield flexures	All plants with thermal shields (see WEC TB-19-5 [50])	Cracking (SCC, fatigue) that results in thermal shield flexure fracture, or complete separation	None	Visual (VT-3) no later than two refueling outages from the beginning of the license renewal period. Subsequent examinations at a 10-year interval.	100% of accessible surfaces of 100% of thermal shield flexures. Minimum coverage shall include the top and bottom accessible surfaces of the inner flexure weld (closest to the attachment to the core barrel) and the outer surface of the thermal shield to flexure weld, including the transition radii of this weld. See Figure 4-37.

OE
(replaced)

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Example “Existing” Components in MRP-227 Revision 2-A

(Mostly ASME B&PV Code Section XI ISI Exams)

Westinghouse “Existing” Table for MRP-227, Rev. 2-A

(Mostly ASME B&PV Code Section XI ISI Exams)

Table 4-9 (continued)
Westinghouse plants Existing Programs components

Item	Applicability	Effect (Mechanism)	Reference	Examination Method	Examination Coverage
W16. Upper Internals Assembly Upper fuel alignment pins	All plants with Malcomized fuel alignment pins (see TB-16-4 [59])	Loss of material (wear)	ASME Code Section XI as supplemented by TB-16-4 [59]	Visual (VT-3) examination to determine general condition for excessive wear	All accessible surfaces at specified frequency. See Figure 4-47.
W17. Lower Internals Assembly Lower fuel alignment pins	All plants with Malcomized fuel alignment pins (see TB-16-4 [59])	Loss of material (wear)	ASME Code Section XI as supplemented by TB-16-4 [59]	Visual (VT-3) examination to determine general condition for excessive wear	All accessible surfaces at specified frequency. See Figure 4-47.
W18. Lower Internals Assembly XL Lower fuel alignment pins	All XL design plants	Cracking (IASCC, fatigue), Aging management (IE)	ASME Code Section XI	Visual (VT-3) examination to determine general condition	All accessible surfaces at specified frequency. See Figure 4-47.
W19. Alignment and Interfacing Components Upper core plate inserts	All plants with upper core plate inserts	Loss of material (wear)	ASME Code Section XI	Visual (VT-3) examination	All accessible surfaces at specified frequency. See Figure 4-48.
W20. Radial Support Keys Radial support keys	All plants	Loss of material (wear)	ASME Code Section XI	Visual (VT-3) examination	All accessible surfaces at specified frequency See Figure 4-49.
W21a. Alignment and Interfacing Components Thermal sleeves	All plants with thermal sleeves (see NSAL-18-1 [52])	Loss of material (wear)	MRP 2018-027 [54] IN 2018-10 [53] OG-19-101 [77]	Per the requirements of PWROG- 16003-P [76]	All thermal sleeves at locations with control rods See Figure 4-50.

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NEI 03-08 “Good Practice” for MRP-227 PWR Internals Aging Management and In-Service Inspections

(MRP 2021-007, Revision 1, dated 5/29/2025)

NEI 03-08 “Good Practice” for MRP-227 PWR Internals

MRP 2021-007, Revision 1, dated 5/29/2025

Subject: Transmit PWR Internals Asset Preservation/Asset Management Considerations for Westinghouse and CE Designs to PWR Utility Members for Use and Consideration when Implementing MRP-227 Guidance

Dear PMMP and MRP RIC members,

This letter and information in the Enclosure supports optional utility use in implementing the following NEI 03-08 Good Practice: *“Each commercial U.S. PWR unit should develop and maintain an asset management strategic plan for PWR reactor internals.”*

Long-term asset preservation and management planning for PWR reactor internals can substantially mitigate severely adverse impacts of otherwise high-economic-risk inspection findings (e.g., refueling outage extensions and high-repair-cost components). The utility should (in accordance with this NEI 03-08 Good Practice guidance) consider developing an asset management strategic plan (e.g., LTAM) for PWR reactor internals materials issues. This is consistent with NEI 03-08 Section 6 roles and responsibilities, specifically utilities are to *“evaluate current business and strategic plans for appropriate focus on materials issues”*. Additional background information and material supporting this strategic plan may be found in MRP-191 Revision 2, MRP-232 Revision 2, PWR internals risk “heat maps”

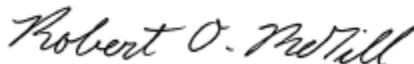
documented in letter report MRP 2018-018, MRP 2018-022, and the Enclosure to this letter. EPRI notes that this letter may be revised at some point either by OE, MRP research or PWROG project report. Salient updates were recently made to PWR Owners Group reports PWROG-19028-NP¹ and PWROG-19024-P².

Utility owners of B&W-designed plants should consider engaging with their NSSS vendor of choice for these considerations at those stations.

Sincerely,



Rex Meeden, Vice President, Nuclear Engineering
Arizona Public Service Nuclear - PMMP EC Chair



Robert O. McGill, P.E., Program Manager
EPRI Materials Reliability Program

Does owner have Strategic Plan for PWR Asset Management and Preservation?

NEI 03-08 “Good Practice” for MRP-227 PWR Internals

Continuing MRP partnership with PWR Owners Group ...

documented in PWROG-19028-NP, which includes both the basis for the selection of those twelve components and the basis for not selecting the remaining Economic Category B and C components. Note that asset management considerations associated with B&W designed plants are addressed outside of this guidance on a unit-specific basis.



The Westinghouse-designed components suggested for asset management were:

- Conduit seal assembly: body, tubesheets, tubesheet welds, and tubes
- Barrel-former bolts
- Thermal shield support block bolts
- Upper Core Plate (UCP)
- UCP insert and UCP alignment pins
- Thermal shield flexures
- Lower support column bolts, lower core plate (LCP) bolts, manway bolts

The CE-designed components recommended for asset management were:

- Fuel alignment plate
- System 80 Core support deep beams
- Control Element Assembly (CEA) shroud bolts
- Core shroud assembly - Shroud plates
- Core shroud assembly – Former plates (specifically welds)

Refer to PWROG-19028-NP Revision 1 and PWROG-19024-P

How does a PWR utility make use of MRP-227?

Case study example: Farley AMP

US Utility Aging Management Program Plan for MRP-227

- Farley RVI AMP is implemented by a plant-specific program document developed by the utility owner
- Submitted to US NRC in 2015
 - ADAMS ML15226A229
- Document serves as the Farley RVI Inspection Plan which summarizes the overall Farley RVI Aging Management Program (AMP)

Enclosure 1 to NL-15-1507
FNP-1 RVI Aging Management Program

Westinghouse Non-Proprietary Class 3

WCAP-18011-NP
Revision 0

**PWR Vessel Internals Program Plan for Aging Management
of Reactor Internals at J.M. Farley Nuclear Plant Unit 1**

Bradley T. Carpenter*
Reactor Internals Aging Management

Mary Ann T. Walsh*
Reactor Internals Aging Management

Karli N. Szweda*
Reactor Internals Aging Management

July 2015

Approved: Patricia C. Paesano*, Manager
Reactor Internals Aging Management

*Electronically approved records are authenticated in the electronic document management system.

Westinghouse Electric Company LLC
1000 Westinghouse Drive
Cranberry Township, PA 16066, USA

US Utility Aging Management Program Plan for MRP-227

- Farley RVI AMP was based upon EPRI report MRP-227-A, "*Pressurized Water Reactor Internals Inspection and Evaluation Guidelines*" and updated to incorporate Revision 1-A in 2020
 - AMP Program plan contains a description summary of the degradation mechanisms of concern, the categorization of components, and the inspection requirements
- Farley is a Westinghouse-design pressurized water reactor (PWR), 3-loop reactor coolant system (RCS) configuration, startup in 12-1977
 - Very similar to many international PWR units
- Farley RVI components were screened for susceptibility to eight degradation mechanisms considering their chemical compositions, neutron fluence exposures, operating temperatures, stress levels
- Farley RVI AMP was submitted to US NRC for approval on 8/12/2015 (ADAMS # ML15226A227)
 - US NRC approved this L.R. AMP in a Safety Evaluation in 2017 (ADAMS # ML17171A351)

US Utility Aging Management Program Plan for MRP-227

- Inspections detailed in SNC's AMP tables are required to manage aging effects in Farley's Primary and Existing Program Components
 - SNC will transition to MRP-227, Revision 2-A before entering SLR PEO
- Proven inspection methods (VT-3/EVT-1) in accordance with MRP-228 are used to detect evidence of the relevant aging mechanism(s)
- Specified inspection frequencies are considered adequate to manage aging effects during LTO; however more frequent inspections may be warranted based upon an internal and external OE
- All detected relevant conditions must be addressed in the CAP prior to plant restart upon completion of refueling outage
 - Possible disposition options include: 1) supplemental exams to further characterize a detected condition; 2) engineering evaluation for continued service until the next inspection; 3) repair; or 4) replacement.

US Utility Aging Management Program Plan for MRP-227

Example: Farley RVI AMP inspection plan

APPENDIX C MRP-227-A AUGMENTED INSPECTIONS

Table C-1. MRP-227-A Primary Component Inspection and Monitoring Recommendations for Westinghouse-Designed Internals

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

US Utility Aging Management Program Plan for MRP-227

Example: Farley RVI AMP inspection plan

Table C-1. MRP-227-A Primary Component Inspection and Monitoring Recommendations for Westinghouse-Designed Internals (cont.)

Item	Applicability	Effect (Mechanism)	Expansion Link ⁽¹⁾	Examination Method/Frequency ⁽¹⁾	Examination Coverage
Core Barrel Assembly Upper core barrel flange weld	All plants	Cracking (SCC)	Lower support column bodies (non-cast) Core barrel outlet nozzle welds	Periodic enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal ⁽⁴⁾ . See Figure A-4
Core Barrel Assembly Upper and lower core barrel cylinder girth welds	All plants	Cracking (SCC, IASCC, Fatigue)	Upper and lower core barrel cylinder axial welds	Periodic enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal ⁽⁴⁾ . See Figure A-4
Core Barrel Assembly Lower core barrel flange weld ⁽⁵⁾	All plants	Cracking (SCC, Fatigue)	None	Periodic enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examinations on a ten-year interval.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal ⁽⁴⁾ .

PUBLIC Ref. <https://www.nrc.gov/docs/ML1522/ML15226A229.pdf>

US Utility Aging Management Program Plan for MRP-227

Example: Farley RVI AMP inspection plan

Table C-1. MRP-227-A Primary Component Inspection and Monitoring Recommendations for Westinghouse-Designed Internals (cont.)

Item	Applicability	Effect (Mechanism)	Expansion Link ⁽¹⁾	Examination Method/Frequency ⁽¹⁾	Examination Coverage
Baffle-Former Assembly Baffle-edge bolts	All plants with baffle-edge bolts Note: FNP Unit 1 has baffle edge bolts	Cracking (IASCC, Fatigue) that results in • Lost or broken locking devices • Failed or missing bolts • Protrusion of bolt heads Aging Management (IE and ISR) ⁽⁶⁾	None	Visual (VT-3) examination, with baseline examination between 20 and 40 EFPY and subsequent examinations on a ten-year interval.	Bolts and locking devices on high-fluence seams. 100% of components accessible from core side ⁽³⁾ . See Figures A-5, A-6, and A-7
Baffle-Former Assembly Baffle-former bolts	All plants	Cracking (IASCC, Fatigue) Aging Management (IE and ISR) ⁽⁶⁾	Lower support column bolts, Barrel-former bolts	Baseline volumetric (UT) examination between 25 and 35 EFPY, with subsequent examination on a ten-year interval. Note: Farley Unit 1 will perform a baseline examination of the replacement baffle-former bolts within this EFPY range.	100% of accessible bolts (Note 3). Heads accessible from the core side. UT accessibility may be affected by complexity of head and locking device designs. See Figures A-5 and A-6

US Utility Aging Management Program Plan for MRP-227

Example: Farley RVI AMP inspection plan

Table C-1. MRP-227-A Primary Component Inspection and Monitoring Recommendations for Westinghouse-Designed Internals (cont.)

Item	Applicability	Effect (Mechanism)	Expansion Link ⁽¹⁾	Examination Method/Frequency ⁽¹⁾	Examination Coverage
Alignment and Interfacing Components Internals hold down spring	All plants with 304 stainless steel hold down springs Note: FNP Unit 1 hold down spring is 304 SS	Distortion (Loss of Load) Note: This mechanism was not strictly identified in the original list of age-related degradation mechanisms.	None	Direct measurement of spring height within three cycles of the beginning of the license renewal period. If the first set of measurements is not sufficient to determine life, spring height measurements must be taken during the next two outages, in order to extrapolate the expected spring height to 60 years.	Measurements should be taken at several points around the circumference of the spring, with a statistically adequate number of measurements at each point to minimize uncertainty. See Figure A-9
Thermal Shield Assembly Thermal shield flexures	All plants with thermal shields Note: FNP Unit 1 does not have a thermal shield.	Cracking (Fatigue) or Loss of Material (Wear) that results in thermal shield flexures excessive wear, fracture, or complete separation	None	Visual (VT-3) no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	100% of thermal shield flexures. See Figures A-6 and A-10

Plant-Unit-Specific Applicability Item

Other Examples of RVI AMPs for PWR Utility Owners

■ NextERA's Point Beach units 1 and 2 dated 12/19/2011

- Two 2-loop (~600 MWe) Westinghouse-design PWRs
- Operations startup for unit 1 in mid-1970 and for unit 2 in mid-1972

[Refer to NRC ADAMS # ML113540301]

- On 3/30/2015, NRC approved NextERA's RVI AMP in ML15079A087
- Year-40 MRP-227 inspections were performed in RFOs in 2013 and 2015
- Year-50 MRP-227 inspections will be performed in upcoming RFOs in 2022-2024
- Point Beach site will transition to MRP-227, Revision 2-A before entering SLR PEO in 2030

■ Xcel Energy's Prairie Island units 1 and 2 dated 10/1/2012

- Two 2-loop (~600 MWe) Westinghouse-design PWRs
- Operations startup for unit 1 in late-1973 and for unit 2 in late-1974

[Refer to NRC ADAMS # ML12276A041]

- On 3/6/2015, NRC approved Xcel's RVI AMP in ML15037A458
- Year-40 MRP-227 inspections were performed in RFOs in 2013 and 2014
- Year-50 MRP-227 inspections were performed in recent RFOs in 2022-2024
- Prairie Island site will transition to MRP-227, Revision 2-A before entering SLR PEO in 2033

EPRI MRP-227 Contact Information

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Long-Term Operations Program:
Assessment of Research and Development Supporting Aging Management Programs for Long-Term Operation. EPRI, Palo Alto, CA: 2013. 3002000576.



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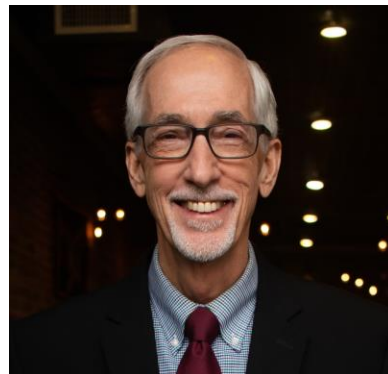
17-11 – General Corrosion & SCC Aging Management Program

IAEA National Workshop on Ageing Management
during Early Stages of Operation



Garry G. Young
Technical Executive

Barakah NPP
Abu Dhabi, UAE
September 17, 2025



EPRI Report 3002000459

Nondestructive Evaluation: Update on License Renewal One-Time Inspection and Best NDE Practices

As nuclear plants in the United States apply for renewed licenses to extend the lifetime of their plants beyond 40 years, nondestructive evaluation (NDE) has become increasingly important. A common commitment made by U.S. plants seeking license renewal is implementation of a one-time inspection (OTI) program using suitable and applicable NDE techniques to confirm the insignificance of various potential aging effects in treated water, lubricating oil, and fuel oil environments. The U.S. Nuclear Regulatory Commission's (NRC's) *Generic Aging Lessons Learned (GALL) Report*, Section XI.M32, One-Time Inspection, defines the purpose of the program as follows:

A one-time inspection of selected components is used to verify the system-wide effectiveness of an aging management program (AMP) that is designed to prevent or minimize aging to the extent that it will not cause the loss of intended function during the period of extended operation.

Plants have been performing numerous inspections to verify the effectiveness of their water chemistry, lubricating oil analysis, and fuel oil chemistry aging management programs.

IAEA equivalent: IGALL AMP119, One-time Inspection

General Corrosion & SCC Aging Management Program

EPRI Report 3002000459

Nondestructive Evaluation: Update on License Renewal One-Time Inspection and Best NDE Practices

Research Overview

EPRI has collected OTI data from a number of plants in an effort to create a repository of OTI data. The data include OTI data from both BWRs and PWRs across the four NRC regions of the United States. Results from more than 1100 OTIs were reported for components in these environments across nine nuclear plants, including 3 BWRs and 6 PWRs. This report describes some of the important findings from the OTI data collected across the nuclear industry.

Conclusions

Metallic components in lubricating oil and fuel oil environments appear to be minimally—if at all—susceptible to the anticipated aging effects covered by the OTI program. These results appear to confirm the effectiveness of current programs in minimizing or eliminating aging effects in these environments.

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EPRI Report 3002000459

Nondestructive Evaluation: Update on License Renewal One-Time Inspection and Best NDE Practices

The inspection technique selected for implementation of the OTI must be capable of detecting the following aging mechanisms for the selected components:

- Crevice corrosion
- Galvanic corrosion
- General corrosion
- Microbiologically influenced corrosion (MIC)
- Pitting corrosion
- Erosion
- Fouling
- Stress corrosion cracking (SCC) or cyclic loading

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Nondestructive Evaluation: Update on License Renewal One-Time Inspection and Best NDE Practices

Important elements of the OTI program:

- Sample size determination based on an assessment of materials of fabrication, environment, plausible aging effects, and OE
- Identification of the inspection locations
- Determination of the examination technique, including acceptance criteria
- Evaluation of the need for follow-up examinations if age-related degradation is found

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Nondestructive Evaluation: Update on License Renewal One-Time Inspection and Best NDE Practices

OTI programs can be conducted by considering the following outlined methods, but others may be used as appropriate:

- **Ultrasonic testing (UT)** is used as a volumetric inspection method for detecting numerous forms of aging degradation. Ultrasonic examination can be used for detecting localized corrosion, wall thinning, cracks, and so on.
- **VT-1 visual examination** is used to detect discontinuities and imperfections - such as cracks and various forms of corrosion - located on the surface of the components under inspection.
- **VT-3 visual examination** is used to determine the general mechanical and structural condition of components under inspection, including a condition assessment for loose, deformed, broken, or missing components.
- **Penetrant testing (PT)** is used to examine the surface of a material for cracking, pitting, and other surface discontinuities.
- **Radiographic testing (RT)** is used to examine the full thickness of a material for discontinuities such as cracks, pits, voids, wall thinning, and so on.
- **Destructive testing (DT)** is used to examine components removed from service to quantitatively determine the presence and locations of defects throughout a material.

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Nondestructive Evaluation: Update on License Renewal One-Time Inspection and Best NDE Practices

OTI programs can be conducted by considering the following outlined methods, but others may be used as appropriate (continued):

- Eddy current testing (ET) is used to examine the surface and near-surface regions of a material for cracking, pitting, and other near-surface discontinuities.
- Hardness testing (HT) is used to directly measure the surface properties of a material to verify that the properties have not fluctuated outside a specified range.

OTI programs have most commonly used visual, ultrasonic, and destructive examinations for performing their OTIs, although all of these NDE methods have also been used.

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Nondestructive Evaluation: Update on License Renewal One-Time Inspection and Best NDE Practices

The following best practices were identified:

- **Use and follow appropriate procedures.** In all cases, the NDE procedure should be followed. If questions or concerns arise with the implementation of the procedure, these should be documented during the course of the examination and presented to the Level III for acceptance. If issues or concerns arise that call into question the validity of an inspection, the Level III should be consulted as soon as possible.
- **Plan ahead.** Make license renewal efforts part of day-to-day operations and at the forefront of the minds of plant personnel. Better planning results in fewer lost opportunities, helps to minimize schedule impacts, and results in fewer last minute rushes. No plan will be perfect, as circumstances may arise in some cases that cannot be foreseen. In such cases, plan ahead, but allow plans to be flexible so that they may change as needed based on the current circumstances. Improved planning will not avoid all problems, but it will help to minimize those that arise.
- **Perform opportunistic inspections.** When practical, take advantage of maintenance activities occurring around the plant to maximize the efficiency of performing OTIs (for example, if a component in a particular M/E that fits the predetermined sampling criteria will already be open, inspect the component before maintenance closes it back up). Using this methodology, many OTIs can be performed while the plant is online and running, minimizing impacts of OTIs on outages; however, if significant degradation is discovered, it could impact the return to service of the component being inspected, and this risk should be taken into account.

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Nondestructive Evaluation: Update on License Renewal One-Time Inspection and Best NDE Practices

The following best practices were identified (continued):

- **Review past inspections and take credit for previously inspected components.** If previous inspections have been performed on the component of interest, it may be possible to take credit for these inspections as a part of the OTI program. It is important to verify that these inspections were adequately documented, were performed according to an acceptable procedure, occurred within the allowable timeframe for OTI, and purposely inspected for the aging effect of concern for the M/E condition addressed by the OTI program. If the inspection meets the criteria for OTI, the inspection can be used as the OTI verification that the component has not experienced the aging effect of concern and will be able to perform its intended function during the period of extended operation.
- **Standardize reporting criteria.** Standardized reporting criteria are important to accurately evaluate the results of OTIs. As a large number of inspections are reported, it is important to properly group these inspections by material and environment combinations (including operating temperatures, water chemistry, and so on).

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Nondestructive Evaluation: Update on License Renewal One-Time Inspection and Best NDE Practices

The following best practices were identified (continued):

- **Start early.** Many different components must be inspected in an OTI program; therefore, it is important to start the process early. This gives the most time for planning inspections, using opportunistic inspections, and reviewing past inspections for applicability to be used in the OTI program. Also, starting early helps to avoid issues related to rushing to perform inspections before a set deadline, which can cause numerous issues related to human performance. Starting the process early helps to minimize or eliminate these issues.
- **Be aware that inspections can be rejected.** The regulator has refused to accept certain inspections; for example, if an inspection was reported as a Code examination but was not performed in accordance with the ASME Code requirements for the particular inspection or if an inspection was not performed as directed in the procedure.
- **Create detailed inspection notes and reports.** It is critical to record pertinent data from each inspection performed in accordance with the procedure requirements. Instruments must be periodically calibrated, essential system parameters must be controlled and recorded, and so on, for an inspection to be properly executed and recorded. If proper records are not created and retained, inspections may be found to be unacceptable and may need to be repeated. Such activities waste time, cost, and often dose. Retaining records is also a method to record best practices to aid in future NDE inspections.

General Corrosion & SCC Aging Management Program

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Nondestructive Evaluation: Update on License Renewal One-Time Inspection and Best NDE Practices

The following best practices were identified (continued):

- **Be aware of original component condition.** Many components installed at a nuclear plant are procured under NQA-1 program guidelines. Although following NQA-1 guidelines significantly reduces the occurrence of installing components with original manufacturing defects, it does not eliminate the risk. Plants should not assume that new or replacement components procured under any quality assurance program will arrive 100% defect free. Pre-service inspections are completed for many components that are placed in service at a nuclear plant, which can help to reduce the potential for a manufacturing defect to be placed into service.
- **Be aware of training and experience.** Both training and experience are important for performing NDE inspections properly. An operator's training can help to ensure proper use of equipment and implementation of the procedure, and an operator's experience can help to avoid problems or pitfalls that may occur either while performing an inspection or while preparing an inspection procedure for field use. Personnel qualification and certification procedures have been developed to increase the quality and reliability of NDE inspections. The knowledge and skill obtained through this training and experience can be especially helpful during an outage for identifying potential problems with an inspection early, avoiding the need to restart or repeat inspections or having the results from inspections rejected after a significant data collection time, resulting in increased cost and dose.

General Corrosion & SCC Aging Management Program

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Nondestructive Evaluation: Update on License Renewal One-Time Inspection and Best NDE Practices

The following best practices were identified (continued):

- **Perform proper calibration.** For an inspection to be accepted, proper calibration, as outlined in the applicable procedure, is an essential step in the inspection process. Calibration may include instrument calibration (often done by the original equipment manufacturer) and calibration of the NDE equipment used for performing the inspection. For example, an ultrasonic inspection procedure may outline the process for calibrating the equipment on a calibration reflector within a calibration block to verify the wedge delay, sound velocity in the wedge, angle of incidence of the sound wave, and so on. Instruments must be calibrated periodically to ensure that problems, such as electronic drift, have not occurred over time and that the entire system is functioning properly. In addition, the probes and sensors used must be evaluated to ensure that they are functioning properly and that no components have been damaged from storage, transport, or previous use. Finally, the procedure must be followed to ensure that accurate, repeatable calibration is performed. Thus, it is important to verify proper equipment and sensor functioning through calibration before use.

General Corrosion & SCC Aging Management Program

EPRI Report 3002000459

Nondestructive Evaluation: Update on License Renewal One-Time Inspection and Best NDE Practices

The following best practices were identified (continued):

- **Use an appropriate scan rate.** Performing an inspection at the proper scan rate is important to ensure high-quality data. Generally, the acceptable scan rates are outlined in the procedure. For visual testing, high-speed scanning, especially with remote camera systems, can cause blurring and may miss the types of defects that the system is otherwise able and qualified to detect. For ultrasound or eddy current systems, the electronics may be capable of high-speed sampling, but the physical scan speed limitation is generally the limiting factor. In other words, due to friction combined with the surface roughness of the component being scanned, all contact scanning methods will suffer from vibration and/or effects from the transducer leaving contact with the surface (bouncing) if the scan speed becomes too high. For obvious schedule reasons, excessively low scan speeds should also be avoided. Thus, it is important to maintain a reasonable scan speed, as outlined in the applicable procedure, that is neither too high nor too low.

General Corrosion & SCC Aging Management Program

EPRI Courses on Aging Management

General

- *Fundamentals of Aging Management Programs*
- *Fundamentals of Aging Degradation & Management*
- *Nondestructive Evaluation for Engineers*
- *Visual Identification of Aging Degradation*
- *Obsolescence Program Management*
- *EPRI Cyber Security Procurement Methodology*

Concrete and Civil Structures

- *Concrete Aging Degradation Mechanisms*
- *Radiation of Concrete*
- *Alkali Silica Reaction Training – Modules 1, 2, & 3*
- *Visual Examination of Concrete Containments*
- *Criteria for Selecting a Corrosion Mitigation Technique in Reinforced Concrete*
- *Aging Management Activities for Corrosion of Concrete Reinforcement*

EPRI Training - <https://www.epri.com/training/courses>

EPRI Courses on Aging Management

Electrical

- *Electrical Equipment Aging Degradation Mechanisms*
- *Electronic and I&C Equipment Aging Degradation Mechanisms*
- *Cable Aging – Adverse Environment & Visual/Tactile Assessment*

Materials

- *Metals Aging Degradation Mechanisms*
- *Materials Handbook for Nuclear Plant Pressure Boundary Applications*
- *Nuclear Power Plant Materials Eng. Fundamentals*
- *Intro to EPRI Materials Degradation Matrix*

EPRI Training - <https://www.epri.com/training/courses>

EPRI Courses on Aging Management

Mechanical

- *Polymers Aging Degradation Mechanisms*
- *Protective Coatings Aging Degradation Mechanisms*
- *Nuclear Coatings Training*
- *Selective Leaching*
- *Buried Pipe Program Owner Training*
- *Cathodic Protection System Owner*
- *BPWORKSTM*
- *Buried Pipe Condition Assessment and Repair*
- *Flow Accelerated Corrosion Program Owner*
- *Flow Accelerated Corrosion: Mechanics and Control*
- *Flow Accelerated Corrosion for non-FAC Personnel*
- *Erosion In Piping Systems*
- *Microbiologically Induced Corrosion*
- *Boric Acid Corrosion Control Limited*

EPRI Training - <https://www.epri.com/training/courses>

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Assessment of Research and Development Supporting Aging Management Programs for Long-Term Operation. EPRI, Palo Alto, CA: 2013. 3002000576.



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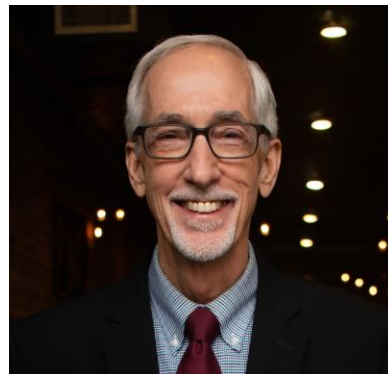
18-03 – Lessons Learned from Aging Management

IAEA National Workshop on Ageing Management during Early Stages of Operation



Garry G. Young
Technical Executive

Barakah NPP
Abu Dhabi, UAE
September 18, 2025



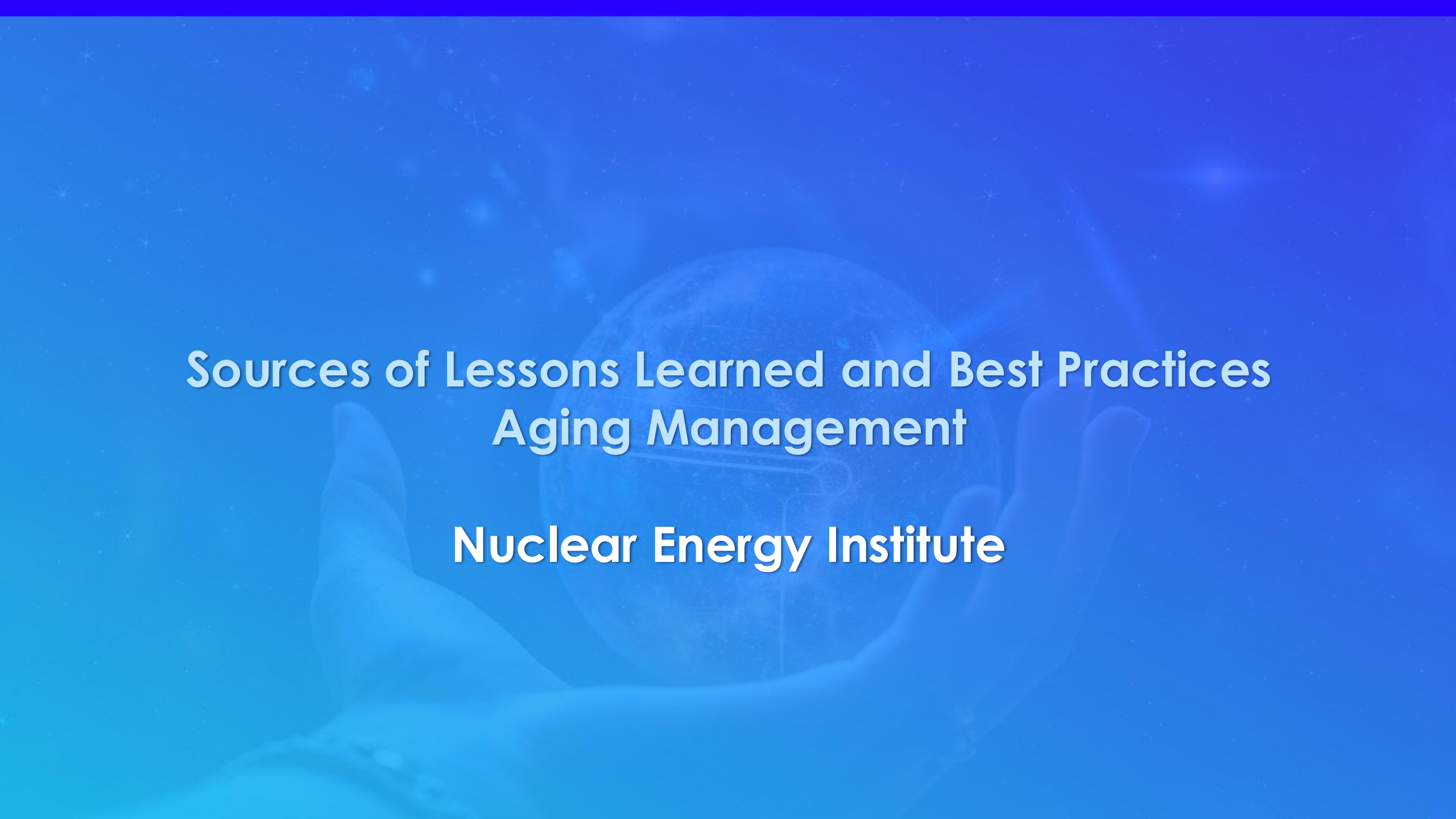
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Sources of Lessons Learned and Best Practices Aging Management

EPRI Users Groups

Example EPRI Users Groups



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Sources of Lessons Learned and Best Practices Aging Management

Nuclear Energy Institute

The Nuclear Energy Institute (NEI)

Who We Are

- The Nuclear Energy Institute is the policy organization of the nuclear technologies industry, based in Washington, D.C.

Membership and Mission

- Industry led and member driven policy organization – representing 100% of U.S. Operating Reactors.
- A unified industry voice to Congress, federal and state legislatures, federal regulators and international organizations on key policy issues.
- Promote the use and growth of nuclear energy through efficient operations and effective policy.

The Nuclear Energy Institute (NEI)



Mission Statement – License Renewal Task Force (LRTF)

To maintain a high level of involvement necessary to ensure regulatory consistency and predictability and to function as an industry clearinghouse for issues that may arise during the license renewal process and during the new license implementation period.

The License Renewal Task Force meets at least quarterly with the NRC to continually monitor and improve the license renewal process by providing an industry focal point to engage the NRC on application information and reviews and implementation issues. Some of the typical engagement areas are the Audit Process, Aging Management Programs (AMPs), Generic Aging Lessons Learned (GALL), Concurrent Submittals, and License Renewal inspections.

The task force consists of the task force and 5 working groups: Mechanical, Civil, Electrical, Implementation and Life Beyond 60. The working groups are complete and separate from the task force. They each meet periodically to exchange information from recent audits and submittals and to share site experiences and issues.

The Nuclear Energy Institute (NEI)

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License Renewal Task Force (LRTF) Contacts

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Ameren – RDavis3@ameren.com
- Sarah Mongrieg, Vice Chair
Constellation – sarah.mongreig@constellation.com

Monthly LRTF “Open Forum” to discuss issues, benchmark, and provide updates

The background of the slide features a blue gradient with a faint image of two hands holding a globe. The globe is centered and has a grid of latitude and longitude lines. The sky around the globe is filled with small, white, star-like specks, giving it a cosmic or global feel.

Sources of Lessons Learned and Best Practices Aging Management

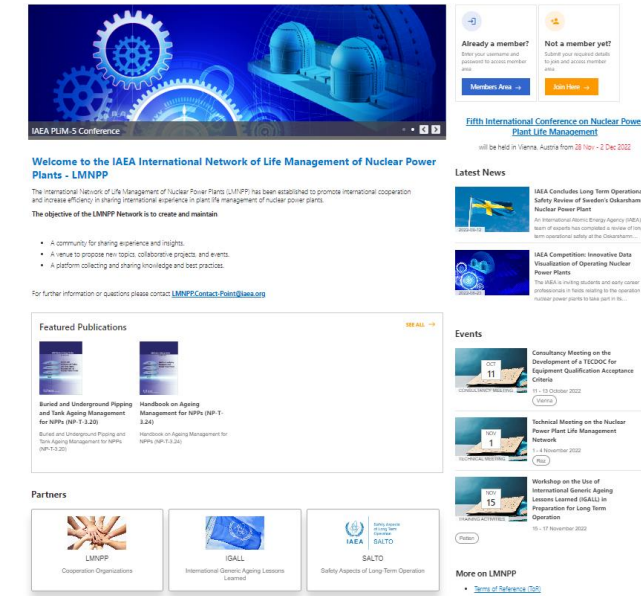
IAEA LMNPP Network and Working Groups

IAEA International Network on Life Management of Nuclear Power Plants (LMNPP)

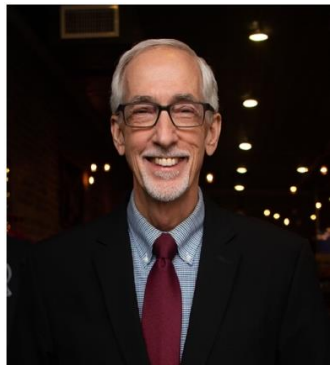
■ Objective:

- Promote international cooperation,
- Increase efficiency in sharing international experience in life management of nuclear power plants, and
- Provide for various project-based Working Groups to further support Member State long term operation efforts and knowledge transfer.

- **Scope:** All aspects of plant engineering, operation and maintenance including degradation, monitoring, testing, inspection, materials, design, fabrication, as well as strategic and administrative aspects of life management at all operating NPPs.
- Officially launched the LMNPP Network in **Nov. 2022**.



1st Steering Committee Meeting of LMNPP



Garry G Young,
Chair of the Steering
Committee



26 Member States and 8 International Organizations

- Argentina
- Armenia
- Bangladesh
- Belarus
- Brazil
- Bulgaria
- Canada
- China
- Czech Republic
- Finland
- France
- Hungary
- Iran, Islamic Republic of
- Japan
- Korea, Republic Of
- Mexico
- Netherlands
- Pakistan
- Romania
- Russian Federation
- Slovakia
- Spain
- Sweden
- Switzerland
- Türkiye
- United States of America

LMNPP Cooperation Organizations



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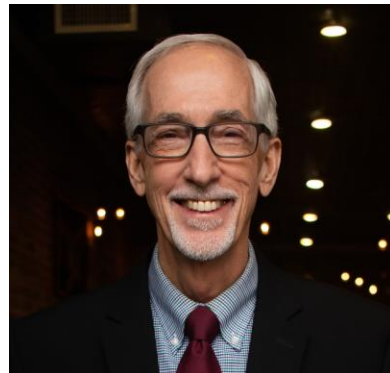
BNPP Aging Management Matrix (BAMM)

IAEA National Workshop on Ageing Management
during Early Stages of Operation



Garry G. Young
Technical Executive

Barakah NPP
Abu Dhabi, UAE
September 16-18, 2025



EPRI 1024568 - EPRI Materials Management Matrix Project: KHNP Advanced Pressurized Water Reactor (APR1400) Materials Management Tables -- Revision 0

- The aim of EPRI 1024568 was to present a pre-construction review of a new plant design and identify potential opportunities to improve materials performance. Once the plant(s) are constructed and begin operation, the intent was to transition the management of knowledge gaps to an appropriate EPRI Issue Program for long-term maintenance. EPRI does not have plans to revise EPRI 1024568; however, we currently plan to develop Issue Management Tables (IMTs) specific to the APR-1400 design that reflect significant improvements in design, based on industry operating experience. We hope to have KHNP and other support in this effort. EPRI is in the final stages of revising the PWR IMTs, with Rev 5 to be published this year. As this would be a key basis document for the development of **IMTs for the APR-1400**, our aim is to **produce the APR-1400 document in 2026**.
- The EPRI MDM and (PWR) IMT would be valuable input documents to the BNPP Aging Management Matrix. The MDM identifies the areas of material aging susceptibility and contains valuable information pertaining to the industry state of knowledge in the various areas. The PWR IMTs (MRP-205) can inform on the prioritized component-level knowledge gaps that are important to the broader PWR fleet. Please note that the latest revision (MRP-205 Rev 5) is not yet published.
- Although the APR1400 MMM was developed to address pre-construction issues, the component risk ranking in the report represents an FMEA of sorts that could be used as a starting point for developing an aging management plan for the reactor internals. Current MRP guidance in MRP-227R2 should also be used. There is some overlap between APR1400 internals and current generation PWR internals.

BNPP Aging Management Matrix (BAMM)

EPRI 1021088 - EPRI Materials Management Matrix Project: Advanced Light-Water Reactor - Pressurized Water Reactor Degradation Matrix - Revision 1

Abstract

The Advanced Light Water Reactor - Pressurized Water Reactor Degradation Matrix (ALWR PWR DM) is an integral piece of the EPRI's Materials Management Matrix (MMM) initiative for ALWR designs. The MMM provides a tool to assist the industry in proactive identification and consideration of materials issues and mitigation/management opportunities from the design phase through component fabrication and plant construction to operations and maintenance.

Background

The ALWR PWR DM is based upon an expert elicitation process. The DM panel conclusions provide a comprehensive assessment of degradation mechanisms that may potentially affect advanced PWR nuclear steam supply system components. The results of this expert elicitation are used to improve design-specific ALWR risk assessments and as inputs to materials management tables, including associated gaps and opportunities. The EPRI Advanced Nuclear Technology (ANT) DM framework is similar to the EPRI Materials Degradation Matrix (MDM), a tool for implementing the materials management initiative for operating plants. The results of the most recent MDM version available (2010 version) formed a starting point for the panel's evaluation.

This revision 1 of the ALWR PWR DM represents an update to the version published in 2008 (EPRI report 1018153). Specifically for this revision of the ALWR PWR DM, materials of construction for the APR1400 and APR+ designs were added to the DM.

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