U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

RIC 2017



Research Activities in Support of Subsequent License Renewal

Highlights

- The key technical issues for research are identified in the staff requirements memorandum (SRM) on SECY-14-0016 (August 29, 2014; ADAMS Accession No. ML14241A578):
  - Reactor pressure vessel embrittlement at high fluence
  - Irradiation-assisted stress-corrosion cracking of reactor vessel internals
  - Concrete degradation
  - Electrical cable qualification and condition assessment
- In response to the SRM, there has been significant progress in addressing the key technical issues:
  - Accomplished through increased leverage with DOE and EPRI through "deep-dive" meetings (cables aging, concrete degradation, vessel internals, nondestructive examination (NDE) of buried pipes)
- Progress resulted in enhanced aging management programs (AMPs) addressed in the draft SLR guidance documents
- Research objectives focused on FY 2018/2019 (expected time period for initial SLR applications):
  - Near term to support review of initial SLR applications
  - Longer term to augment the technical basis for further updates to SLR guidance

NRR-RES Team Effort Informed the Development of Draft Guidance Documents: GALL-SLR (NUREG-2191) and SRP-SLR (NUREG–2192)

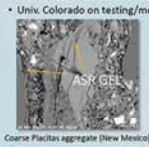
- RES support to NRR (2008 – 2016):
  - Expanded materials degradation assessment (EMDA) (NUREG/CR-7153, Vols. 1-5)—technical issues
  - AMP effectiveness pilot audits (ML13122A009)—implementation/lessons learned
  - Assessment of international periodic safety reviews (PSR)—lessons learned
  - Participation in codes and standards committees (ASME, ASTM, ACI, IEEE) to review/revise applicable documents
- Insights/results from previous and ongoing research activities:
  - Irradiation-assisted degradation of stainless steel plate and weld materials in RPV
  - Thermal and neutron embrittlement of cast austenitic stainless steels (CASS)
  - Environmentally assisted fatigue of stainless steels
  - RPV embrittlement: enhancement of surveillance database, enhancement of ΔT models, ASME BPVC work on master curve fracture toughness
  - Containment liner corrosion
- Operational experience (alkali-silica reaction (ASR), cable condition monitoring, selective leaching of buried pipes, coatings)
- 97 specialized expert panels (EPs) comprising staff from NRR, RES, and the NRC regions for the 52 AMPs, the seven chapters containing tables of AMR line items in NUREG-1801 (Generic Aging Lessons Learned (GALL)), and corresponding sections in NUREG-1800):
  - 37 EPs for mechanical AMPs
  - 9 EPs for structural AMPs
  - 6 EPs for electrical AMPs
  - 10 EPs for time-limited aging analyses (TLAAs)
  - 14 EPs for other SRP-LR sections
  - 18 EPs for other GALL sections and chapters (including 3,000 AMR line items)
- Nine public meetings to discuss and disposition public comments on draft guidance documents: GALL-SLR (NUREG-2191) and SRP-SLR (NUREG-2192)

Domestic and International Collaboration on Aging Management Research

- Objectives:
  - Ensure the timely exchange of information on planned and ongoing research
  - Ensure the timely sharing of technical data generated from these research programs
  - Share operating experience
  - Assess the capabilities of current and future aging management technology, methods, and tools
- The NRC to independently evaluate the data and information to arrive at conclusions or criteria to inform its regulatory decision-making
- Domestic:
  - NRC-DOE MOU:
    - Cooperative Nuclear Safety Research Related to Long-Term Operations (ML14107A144)
  - NRC-EPRI MOU addenda:
    - Long-Term Operations Beyond 60 Years (ML16223A504)
    - Aging, Qualification, and Condition Monitoring of Electrical Cables (ML16223A501)
- International:
  - CSNI Working Group on Integrity and Aging of Components and Structures—regulatory organizations from Americas, Europe, and Asia
  - International Forum on Reactor Aging Management (IFRAM)—participating countries from North America and Asia
  - Bilateral agreements between the United States and all major countries with commercial LWRs

Concrete Degradation

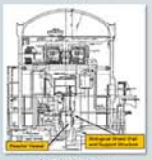
- Research Objectives:
  - Develop the technical basis for guidance to evaluate degradation of nuclear power plant concrete structures:
    - Evaluate structural performance and capability to perform intended safety functions under design-basis loads and accidents
    - Assess AMPs to monitor and manage aging and degradation
- Alkali-Silica Reaction:
  - Chemical reaction that can occur over time in concrete between the highly alkaline cement paste and reactive noncrystalline silica, commonly found in many aggregates
  - Leads to expansion and cracking of concrete that may affect structural performance
- Effects of Irradiation on Concrete Structures:
  - Confirmatory review of DOE work on characterization of concrete irradiation damage and of EPRI research on susceptible plant configurations and their structural integrity (EC: 2018)
  - Assessment of neutron fluence and gamma dose on the bio-shield concrete (EC: 2018)
  - Evaluation of benefits and opportunities to harvest irradiated concrete from decommissioned plants for confirmatory testing (EC: 2020)
- Post-tensioned Concrete Containments:
  - Review of operating experience with post-tensioned containments (loss of prestress, trend analysis of prestress forces, corrosion of prestressing systems, and cracking of anchor heads, for example)
  - Confirmatory research and review of creep effects on prestress losses and of potential for creep and fracture interactions (EPRI)
  - Participation on NEA/CSNI's benchmarking studies
- NRC-Sponsored Research:
  - NIST on effects of ASR on concrete structural performance (EC: early 2019)
  - Northwestern Univ. on service life degradation (EC: 2018)
  - Univ. Colorado on testing/modeling of ASR beams (EC: 2018)



Coarse Placitas aggregate (New Mexico)

ASR


Cracked concrete, corrosion, and grease leakage at tendon anchor heads

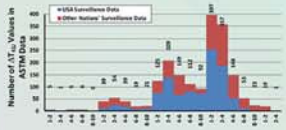

Containment areas susceptible to irradiation

Reactor Pressure Vessel Embrittlement

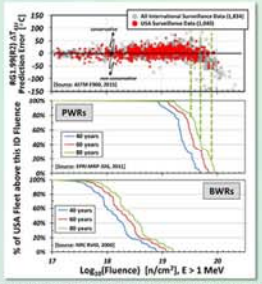
A well-established framework of documents provides formulae to predict the evolution of the RPV's mechanical properties into SLR.

	Documents	Last Revised by NRC	Last Reviewed by NRC
ΔT_{30}	RG 1.99	1988	2014
	10 CFR 50.61	1996	Mid 2000s
	10 CFR 50.61a	2010	2010
ΔUSE	RG 1.99	1988	2014
J-R	RG 1.161	1995	2014
Surveillance	10 CFR Part 50 Appendix H	1983	Underway

Industry programs are working to collect more data at high SLR fluence in advance of their occurrence in service.



Advance evidence from surveillance programs shows that some of these formulae may need updating as irradiation continues, but this is not yet an issue for the operating fleet.



Top: Variation of RG 1.99 prediction error with fluence. Each symbol represents an individual surveillance measurement.

Middle: Percentage of PWRs operating in USA that will reach the specified fluence after the specified operating time.

Bottom: Percentage of BWRs operating in USA that will reach the specified fluence after the specified operating time.

Cable Qualification and Condition Assessment

- Research Objectives:
  - Confirmation or validation (through testing) that electrical cables environmental qualification will be maintained through the SLR period
  - Development and validation of criteria for cable condition monitoring methods and techniques for the SLR period to perform service life prediction
- Technical Issues:
  - Synergistic effects between thermal and radiation aging
  - Diffusion limited oxidation
  - Inverse temperature
  - Activation energy
  - Dose rate effect
  - Cable degradation due to submergence
- Evaluation of Condition Monitoring Techniques:
  - NRC-sponsored research at NIST and SNL to assess cable aging and evaluate monitoring techniques, such as Tan Delta, under combined gamma radiation and temperature exposure (EC: early 2019)
- Submergence Issues:
  - Currently reviewing EPRI report on medium-voltage (MV) Kerite submergence qualification (EC: 2017)


Thermal aging of jacketed cables

Vessel Internals

Irradiation-assisted degradation of stainless steel plate and weld materials

- The NRC initiated collaborative programs with domestic and international partners:
  - International Zorita internals research project (ZIRP): testing of ex-plant 304 SS plates (EC: 2017).
  - NRC-EPRI collaborative program: testing of weld materials harvested from Zorita plant (up to 2 dpa) (EC: 2017)
  - Halden Research Program: further irradiation/testing of Zorita weld materials (8 dpa) (EC: 2022)

Cracking in a PWR baffle bolt

Fluence (dpa)	Plate	Weld	Heat-Affected Zone
1	Previous research	Ongoing	
3			
5		Ongoing	
8			
10	Ongoing	Beyond expected fluence at 80 years	
25			
50			
65	Planning		
80			

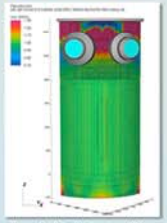
Testing and characterization includes crack growth rate (CGR), fracture toughness (JT), tensile properties, and microstructure (void swelling).

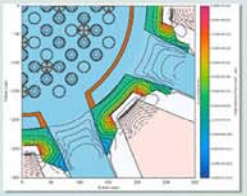
Cast austenitic stainless steel (CASS):

- Further testing of CASS components (3 dpa) (EC: 2017)

Neutron Fluence Calculations

- RG 1.190 describes acceptable methods for computing neutron flux in the RPV active core height (beltline) region
- During extended period of operations, components located outside of the beltline, such as nozzles and vessel internals, experience higher levels of neutron exposure
- Research is being conducted to analyze fluence at vessel locations above and below the reactor core (EC: 2018)
- Develop technical basis for either revision to RG 1.190 or new RG


Ratio of neutron flux calculated using two different methods shows similar predictions within the beltline region but significantly different predictions near the nozzles due to the streaming effect


Neutron flux near the centerline of the PWR inlet and outlet nozzles

Acronyms

ACI – American Concrete Institute
ADAMS – Agencywide Documents Access and Management System
AMP – Aging management program
AMR – Aging management review
ASME – American Society of Mechanical Engineers
ASR – Alkali-silica reaction
ASTM – American Society for Testing and Materials
BPVC – Boiler and Pressure Vessel Code
BWR – Boiling water reactor
CASS – Cast austenitic stainless steel
CFR – Code of Federal Regulations
CSNI – Committee on the Safety of Nuclear Installations
DOE – U.S. Department of Energy
DPA – Displacement Per Atom
EP – Expert Panel
EPRI – Electric Power Research Institute
EC – Expected completion
EMDA – Expanded materials degradation assessment
FY – Fiscal year
GALL – Generic Aging Lessons Learned
IEEE – Institute of Electrical and Electronics Engineers
IFRAM – International Forum on Reactor Aging Management
LWR – Light-water reactor
MOU – Memorandum of understanding
MV – Medium voltage
NDE – Nondestructive examination
NEA – Nuclear Energy Agency
NIST – National Institute of Standards and Technology
NRC – U.S. Nuclear Regulatory Commission
NRR – Office of Nuclear Reactor Regulation
PSR – Periodic safety reviews
PWR – Pressurized water reactor
RES – Office of Nuclear Regulatory Research
RG – Regulatory Guide
RVID – Reactor Vessel Integrity Database
RPV – Reactor pressure vessel
SECY – Office of the Secretary
SLR – Subsequent license renewal
SNL – Sandia National Laboratories
SRM – Staff requirements memorandum
SRP – Standard review plan
SS – Stainless Steel
TLAA – Time-limited aging analysis
USA – United States of America
ZIRP – Zorita Internals research project