UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 25, 2026

Laura Basta
Site Vice President
H.B. Robinson Steam Electric Plant
Duke Energy Progress, LLC
3581 West Entrance Road, RNPA11
Hartsville, SC 29550

SUBJECT: H.B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2 – REPORT FOR THE AGING MANAGEMENT AUDIT REGARDING THE SUBSEQUENT LICENSE RENEWAL APPLICATION REVIEW SUPPLEMENT

Dear Laura Basta:

By letter dated April 1, 2025 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML25091A290), as supplemented by letters dated August 28, 2025 (ML25240B655), and October 1, 2025 (ML25274A131), Duke Energy Progress, LLC (Duke or the applicant) submitted an application for subsequent license renewal of Facility Operating License No. DPR-23 for H.B. Robinson Steam Electric Plant, Unit 2 (Robinson), to the U.S. Nuclear Regulatory Commission (NRC or staff), pursuant to Title 10 of the *Code of Federal Regulations* Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants."

The NRC staff completed its aging management audit from April 28, 2025 – August 22, 2025, in accordance with the audit plan (ML25113A162), and published the audit report (ML25262A143) on November 14, 2025. A supplement to the audit report is enclosed.

If you have any questions regarding this matter, I may be reached at 301-415-8583, or by email at Mark.Yoo@nrc.gov.

Sincerely,

Signed by Yoo, Mark
on 02/25/26

Mark L. Yoo, Senior Project Manager
License Renewal Project Branch
Division of Materials and License Renewal
Office of Nuclear Reactor Regulation

Docket Nos. 50-261

Enclosure:
Audit Report Supplement

cc w/encl: Listserv

SUBJECT: H.B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2 – REPORT FOR THE AGING MANAGEMENT AUDIT REGARDING THE SUBSEQUENT LICENSE RENEWAL APPLICATION REVIEW SUPPLEMENT
DATE: FEBRUARY 25, 2026

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ADAMS Accession No.: ML26054A008 ***via e-concurrence** **NRR-106**

OFFICE	NLRP/DNRL/PM	DEX/ESEB/BC	NLRP/DNRL/BC (A)
NAME	MYoo	ITseng	CHobbs
DATE	2/24/26	9/11/25	2/24/26
OFFICE	NLRP/DNRL/PM		
NAME	MYoo		
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AMR Items Not Associated with an AMP

SLRA AMR 3.5.2.2.2.6 Reduction of Strength and Mechanical Properties of Concrete due to Irradiation

Summary of Information in the Application. During the audit, the staff reviewed plant documentation associated with SLRA Section 3.5.2.2.2.6, which describes the applicant's "Further Evaluation or FE" of effects of aging on the Primary Shield Wall (PSW) and Secondary Shield Wall (SSW) concretes, grout, the reactor vessel (RV) steel supports, structural/anchor bolting and the embedded wide-flange (WF) steel sections and welded plates (PL) within the PSW exposed to neutron and gamma radiation and radiation induced heating in air–indoor uncontrolled environment and the following Table 2 AMR items:

Table 3.5.2-1 Containments, Structures, and Component Supports - Reactor Containment Building - Aging Management Evaluation, which cites Concrete Elements for Shelter, Protection; Structural Support exposed to air-indoor uncontrolled (external) environment for reduction of strength, loss of mechanical properties associated with Table 1 item 3.5.1-097

Table 3.5.2-21 Containments, Structures, and Component Supports - NSSS Supports - Aging Management Evaluation, which cites Grout for Structural Support exposed to air-indoor uncontrolled environment (external) for reduction in concrete anchor capacity associated with Table 1 item 3.5.1-055

Table 3.5.2-21 Containments, Structures, and Component Supports - NSSS Supports - Aging Management Evaluation, which sites Bolting (Structural) for Structural Support exposed to air-indoor uncontrolled environment (external) for loss of material, loss of preload associated with Table 1 items 3.5.1-081 and associated 3.5.1-087, respectively

Table 3.5.2-21 Containments, Structures, and Component Supports - NSSS Supports - Aging Management Evaluation, which sites Bolting (Structural) for Structural Support exposed to air with borated water leakage (external) for loss of material associated with Table 1 item 3.5.1-089

Table 3.5.2-21 Containments, Structures, and Component Supports - NSSS Supports - Aging Management Evaluation, which sites High-Strength Bolting for Structural Support exposed to air (external) for cracking, loss of preload associated with Table 1 items 3.5.1-068 and associated 3.5.1-087, respectively

Table 3.5.2-21 Containments, Structures, and Component Supports - NSSS Supports - Aging Management Evaluation, which sites High-Strength Bolting for Structural Support exposed to air with borated water leakage (external) for loss of material associated with Table 1 item 3.5.1-089

Table 3.5.2-21 Containments, Structures, and Component Supports - NSSS Supports - Aging Management Evaluation, which sites Steel Elements (to include support members, bearing plates, base plates and connections) for Shelter Protection; Structural Support exposed to air-indoor uncontrolled environment (external) for loss of material associated with Table 1 item 3.5.1-091

Table 3.5.2-21 Containments, Structures, and Component Supports - NSSS Supports - Aging Management Evaluation, which sites Steel Elements (to include support members, bearing plates, base plates and connections) for Shelter Protection; Structural Support exposed to air with borated water leakage (external) for loss of material associated with Table 1 item 3.5.1-089

Table 3.5.2-21 Containments, Structures, and Component Supports - NSSS Supports - Aging Management Evaluation, which sites sliding surfaces (Lubrite; Fluorogold; Lubrofluor) for Structural Support exposed to air-indoor uncontrolled environment (external) for loss of mechanical function associated with Table 1 item 3.5.1-075

Audit Activities. The table below lists documents that were reviewed by the staff and were found relevant to the FE 3.5.2.2.2.6. The staff will document its review of this information in the SE.

Document	Title	Revision / Date
Duke		
SLR-RNP-FERR- 0500	Containments, Structures, and Component Supports	Revision 2
RNP-PMech-1020	Thermal Study of the Shield (concrete) - Carolina Power and Light Company Project (Ref. 5.20)	Revision 0
RNP-PMech-1023	Thermal Analysis of Expansion Joints and Reactor Supporting Structure - Carolina Power and Light Company Project (Ref. 5.31)	Revision 0
RNP Specification 14-65	Concrete - Medium and Small Work (Ref. 5.41)	Revision 1
Attachment C - BAC eval of RV nozzle supports, AR 2359059, NCR 2359059	Equipment Tag RPV-A, B, and the nozzle supports (boric acid corrosion evaluation template) bye	12/01/2020
EC 418725	Reactor Vessel Support Evaluation (R2R32) – Includes Corrosion Rates - Expires 11/23/2022	12/01/2020
Attachment A – EC 418725	Structural Engineering Review and Comparison of RX Vessel Support Material Condition	11/25/2020
Attachment B – EC 418725	Reactor Cavity Leakage	12/01/2020
Attachment C – EC 418725	Reactor Cavity Leakage	12/01/2020
RNP-M/HVAC-1076	Reactor Support Temperature Following Loss of HVH-6A and B	Revision 2
American Forge & Mfg. Co. Test Report (from Duke records)	Certified material test report for the three ASTM A 508 Class 2 support shoes of the RV steel support assembly (references Ebasco Dwg G-190561)	6/27/68
WO 20436048-01	RNP Unit 2; Vessel, clean boric acid off supports; Clean and inspect by use of robotics (Deficient Non-Critical)	04/30/2023
WO 20436048-02	RNP Unit 2; Vessel, clean boric acid off supports; Clean and inspect by use of robotics (Deficient Non-Critical)	04/30/2023
WO20436048-20	RNP Unit 2; Vessel, clean boric acid off supports; Clean and inspect by use of robotics (Corrective Actions, Inspections, Generic Issues)	04/30/2023
WO20436048-21	RNP Unit 2; Vessel, Perform ISI (F 1.40 VT-3) of Support 101/A – Photos included (Corrective Actions, Inspections, Generic Issues)	04/30/2023
WO20436048-22	RNP Unit 2; Vessel, Perform ISI (F 1.40 VT-3) of Support 101/B – Photos included (Corrective Actions, Inspections, Generic Issues)	04/30/2023
WO20436048-23	RNP Unit 2; Vessel, Perform ISI (F 1.40 VT-3) of Support 101/C – Photos included (Corrective Actions, Inspections, Generic Issues)	04/30/2023
NCR (AR) 02359059	ISI Visual Examination of RPV A, B and C Nozzle Supports	11/21/2020

NCR (AR) 02452522	Clean Boric Acid Residue in Pump Bay from Rx Cavity Leakage	12/10/2022
NCR (AR) 00173392	Boric Acid on Underside of 275' Floor Slab	10/19/2005
NCR (AR) 00300705	Unknown White Residue on Wall in "A" Pump Bay	10/10/2008
NCR (AR) 02556901	Dry White Boric Acid on Seal Table	06/04/2025
NCR (AR) 02454068	Seal Table Room has 20 DPM Leak on Thimble	12/26/2022
NCR (AR) 02474816	BACC Program – Seal Table Leakage	06/01/2023
NCR (AR) 02534414	30 DPM Active Boric Acid Leak (Seal Table)	11/05/2024
NCR (AR) 02450859	CV Cavity Leakage	11/28/2022
NCR (AR) 02454238	QC Identified Leakage During Mode 3 NOP/NOT VT-2 Walk down	12/28/2022
NCR (AR) 00289438	Information on Drawing G-190561 Inappropriately Voided	07/302008
NCR (AR) 00398950	Water in Reactor Support Cooling Ducts (includes cavity leakage support/refute matrix)	05/12/2010
NCR (AR) 00399274	R226; Apparent Material Wastage on RX Vessel Support Leg "A"	05/13/2010
NCR (AR) 00399274	Degraded/Non-conforming condition resolution (content as noted immediately above)	Undated
AR 00399274-19, Attachment 13, sheet 1/4, FORM CAP NGGC-0200-13-28	Adverse Condition Investigation - Equipment Report: Robotic Examination of the Reactor Vessel Supports.	Undated
NCR 00399274 (EGR-NGGC-0207, Rev. 3)	Boric Acid Corrosion Control Program Evaluation. Component Tag: Reactor Vessel "A" Nozzle Supports	Undated
Westinghouse		
CPL-CA120-TM-SA-000001	H. B. Robinson Unit 2 Subsequent License Renewal: Primary Shield Wall Concrete Assessment (Ref.5.36)	Revision 1
CPL-CA120-TM-SA-000001	H. B. Robinson Unit 2 Subsequent License Renewal: Primary Shield Wall Concrete Assessment	Revision 2
CPL-CA120-TM-SA-000001	H. B. Robinson Unit 2 Subsequent License Renewal: Primary Shield Wall Concrete Assessment	Revision 3
CPL-REAC-TM-AA-000001	Reactor Vessel, Vessel Support, and Bioshield Concrete Exposure Data (Ref. 5.43)	Revision 1
CPL-REAC-TM-AA-000001	Reactor Vessel, Vessel Support, and Bioshield Concrete Exposure Data	Revision 2
CPL-RV000-TR-CF-000003-NP	H. B. Robinson Unit 2 Subsequent License Renewal: RPV Supports Fracture Mechanics Response to NRC Supplement TRP 76	Revision 0
WCAP-18751-NP	Analysis of Ex-Vessel Neutron Dosimetry from H.B. Robinson Unit 2 – Cycle 32 (Ref. 5.44)	Revision 0 (March 2022)
EPRI		

TR-3002013084	Long-Term Operations: Subsequent License Renewal Aging Effects for Structures and Structural Components (Structural Tools) (Ref 5.6)	November 2018
TR-3002011710	Irradiation Damage of the Concrete Biological Shield (Ref 5.8)	May 2018
TR-3002018400	2020 Update to Irradiation of Concrete Guidance	Revision 1
EBASCO		
EBASCO Calculation Book 3	CPL Reactor Vessel Support (Ref G 190188) (Ref 5.10)	11/16/1967
EBASCO Calculation Book 3	Reactor Building Interior (pg. 869)	03/23/1967
	H.B. Robinson Unit No. 2 Specifications	
EBASCO DWG G-190312	HVAC Reactor Building Plans – Sheet 1	Revision 6 (10/17/2005)
EBASCO DWG G-190352	Reactor Building, Containment Vessel Cyl Sects & Dets - Reinf	Revision 2 (06/13/1968)
EBASCO DWG G-190379	Reactor Building Primary Shield-SECTS-MAS1970 (Ref 5.17)	Revision 4 (09/10/1968)
EBASCO DWG G-190383	Reactor Building, Primary Shield - Plans - Reinf	Revision 2 (06/11/1968)
EBASCO DWG G-190384	Reactor Building, Primary Shield-Sects - Reinf	Revision 2 (06/11/1968)
EBASCO DWG G-190386	Reactor Building, Internal Conc Masonry Placing Sequence	Revision 2 (07/24/1968)
EBASCO DWG G-190389	Reactor Cavity - Plan & Sects - Reinf	Revision 1 (09/09/1968)
EBASCO DWG G-190394	Reactor Building, Operating Floor-Plan & Sects-MAS	Revision 3 (10/17/2005)
EBASCO DWG G-190370	Reactor Building. Internal Concrete Plans and Sections (Ref. 5.19) – Sh 1	Revision 7
EBASCO DWG G-190551	Reactor Building Equipment Supports, SH 1 (RV structural steel support material and fabrication notes) (Ref. 5.27)	Revision 9 (06/25/2015)
EBASCO DWG G-190561	Reactor Building Equipment Supports, SH 2 (Ref. 5.9)	Revision 4 (06/25/2015)
EBASCO DWG G-190563	Canal Liner Details, SH 2	Revision 7 (05/18/2007)
EBASCO DWG G-190564	Canal Liner Details, SH 1 (Ref. 5.9)	Revision 5 (11/05/2013)
	RNP Unit 2 HVAC Drawings	
EBASCO DWG G-190388	Reactor Building - Reactor Cavity - Plans Sects - MAS	Revision 3 (07/01/1968)
EBASCO DWG G-190385	Reactor Building – Primary Shield – Details - Reinf	Revision 2 (06/11/1968)
EBASCO DWG G-190187	General Arrangement Reactor Building Equipment Supports, SH 2 (Ref. 5.22)	Revision 9 (06/25/2015)
EBASCO DWG G-190391	Reactor Building – Refueling Canal - Plan & Sects - MAS	Revision 3
EBASCO DWG G-190378	Reactor Building – Primary Shield - Plans – MAS Equipment Supports, SH 2 (Ref. 5.22)	Revision 4 (09/10/1968)

EBASCO DWG G-190188	Reactor Building - General Arrangement - Sections	Revision 16 (Redrawn- 11/12/1983)
Other Construction/Retrofitting Drawings		
TRANSCO, INC DWG 3638-1	Elevation of Reactor Vessel and Typical Panel Sections (Ref. 5.21) – Note 3: L-Panel Metal reflective Insulation Replaced	Revision 4
TRANSCO, INC DWG 3638-2	Panel Layouts for Reactor Vessel	02/26/1968
TRANSCO, INC DWG 3638-3	Sections Thru (?) Insulation Panels	02/26/1968
TRANSCO, INC DWG 3638-4	Sections and Details of Insulation Panels	02/26/1968
TRANSCO, INC DWG 3638-6	Sections and Hemispheric Bottom	02/23/1968
TRANSCO, INC DWG 3638-7	Sections and Hemispheric Bottom	02/27/1968
TRANSCO, INC DWG RM-50761-THL/SH 1	Reactor Top Head Insulation Layout THL	12/05/2012
TRANSCO, INC DWG RM-50761-THL/SH 2	Reactor Top Head Insulation Layout THL	12/05/2012
PB&IW/EBASCO DWG 3-30/5379-1420	Reactor Supports, Pittsburgh Bridge & Iron Works/ Erection Drawing, EBASCO (Ref. 5.24)	Revision 0
PB&IW/EBASCO DWG 3-30/5379-1421	Reactor Supports, Pittsburgh Bridge & Iron Works/ Erection Drawing, EBASCO (Ref. 5.25)	Revision 0
PB&IW/EBASCO DWG 3-30/5379-1422	Reactor Supports, Pittsburgh Bridge & Iron Works/ Erection Drawing, EBASCO (Ref. 5.26)	Revision 0
AREVA/PROGRESS DWG HBR2-13007 (Sheets 1-8)	Robinson Nuclear Plant- Permanent Canal Seal Plate Assembly	Revision 3 (08/06/2012)

During the audit, the staff made the following observations:

For Fluence, the applicant clarified that the:

- Difference in the reactor vessel and PSW fluence models between the referenced WCAP-18751-NP model and that used to calculate fluence in the SLRA was due to different modeling symmetries.
- Twenty percent extended beltline fluence uncertainty analysis is lower than that used in previous SLRA reviews, because the RNP calculated peak fluence is closer to the RV beltline and therefore subject to less error in the analysis methodology used. The uncertainty was not included in the SLRA reported concrete gamma dosage, and thus the need for a revised concrete gamma dosage estimate. However, the applicant clarified that the neutron fluence and 3.74 in depth of exceedance were calculated with a 20% uncertainty. Usage of partial-length stainless steel shielding rods are augmented the peak fluence being above the core midplane

For PSW/pedestal concrete (Radiation on Concrete, Grout, and Steel Embedded in Concrete), the staff noted that:

- Section 3.5.2.2.2.6 of the SLRA will be supplemented to describe concrete areas and location where gamma dose exceeds the threshold.
- SLRA Table 3.5.2-21, AMR item 869 would be revised with plant specific note(s) (PSN) to state that "loss of preload includes loosening," and referenced SLRA Table 3.5.1-055 will be modified to state that, inspection of the RV support grout pad is age managed by the ASME Section XI, IWF program.
- Current condition of the RV supports and PSW concrete to be in good condition and any monitoring and mitigating measures taken would be to ensure that the RV supports and RV cavity concrete remains in good health to the end of SPEO.
- There is adequate volume of air flow through each RV steel support assembly and that the grout and concrete below the RV steel support bottom plate is below 150°F.
- SLRA Table 3.5.2-21, AMR item 778 would be revised to reference the IWF AMP for aging management of the grout and explain in a PSN that the aging effect of this line item is reduction in structural support. Additionally, SLRA Table 3.5.2-1 AMR item 5502 PSN 1 would be modified to disassociate the AMR item 5502 from RV grout pads.
- Table 5-26 through Table 5-29 of Attachment 1 of CPL-REAC-TM-AA-000001, Revision 2, indicates that the region of concrete receiving gamma dose exceeding the 1E10+8 Gy six inches deep is between elevations -38cm to +91cm from the mid-core.
- The Applicant would confirm that the grout strength is conservatively assumed at 3,000psi based on industry standard, and even with the conservative assumption of using the lower bound curve from Figure 3-1 of EPRI Report 3002018400 for a 20% reduction in grout strength, there is still adequate margin for the design of the RPV support base plate.
- There are iron deposits on back side of PSW (pump room) that may need to be age managed by the SMP.

For Ventilation/Gamma Heating on PSW Concrete, the applicant noted that:

- The UFSAR states that the air flow through 2.5-inch RV cavity annulus space at a peak temperature of 140.3°F was estimated to be 12,000 cfm, and the design flow to the reactor cavity is 16,800 cfm. A supplement will be provided to discuss the impacts to the reactor cavity air flow due to RIVE if quartz aggregates were used during construction.

For the RV steel supports, the staff noted that:

- Because of the installation of a permanent canal seal plate (as discussed in SLRA page 3-920) in refueling outage 28 to minimize or permanently eliminate leakage, the supports are expected to be dry during operation, and a baseline corrosion rate for boric acid crystals in humid air at 70°F from EPRI Boric Acid Corrosion Guidebook is used for the evaluation of metal loss.
- Regarding the graphitic steel shims that were coated with a dry film lubricant mentioned in SLRA page 3-915, the applicant would be adding a PSN to SLRA Table 3.5.2-21, AMR item 779 to include steel shims and dry-film lubricants.
- Regarding the embedded (cast in concrete) anchor plates discussed in SLRA page 3-316, the applicant would be adding a PSN to SLRA Table 3.5.2-21, AMR item 868 explaining that

the ASME Section XI IWF inspection of the high strength anchor bolts to indirectly manage the embedded anchor plates.

- o Regarding the RV support structural steel elements discussed in SLRA page 3-920, the applicant would be providing information on the initial fabrication of the RV support structural steel elements, devoid of any crack growth in these steel elements.
- o Regarding the discussion of fracture toughness for the reactor vessel steel supports on SLRA page 3-921, the applicant explained that with the use of the K_{IR} fracture toughness curve, the fracture toughness values for the top plates, diaphragm plates, and I-beams were slightly reduced compared to those in SLRA Table 3.5.2.2.2.6-3, but the margins are still above 1.0. The applicant plans to supplement the SLRA to reflect this information. This was a result of the NRC staff's observation in one of the proprietary enclosures included with the SLRA (WCAP-18939-P, Revision 1) regarding the first step in determination of fracture toughness values for the plates and I-beams.
- o Regarding the loading conditions stated in SLRA page 3-921, the applicant would be clarifying the SLRA to state that "normal" loads include earthquake.

The staff will consider issuing RAIs and/or use a voluntary SLRA supplement offered by the applicant to obtain the necessary information to address issues identified and communicated during the audit. The staff will document its review of this information and resolution of the issues in the SE.