Tennessee Valley Authority, 1101 Market Street, Chattanooga, Tennessee 37402

CNL-15-098 Revision 1

July 10, 2015

10 CFR 50.4
10 CFR Part 54

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Sequoyah Nuclear Plant, Units 1 and 2
Facility Operating License Nos. DPR-77 and DPR-79
NRC Docket Nos. 50-327 and 50-328

Subject: **Sequoyah Nuclear Plant – Revision to Commitment No. 28 and Review of Impacts to the SQN Reactor Vessel Internals Aging Management Program Due to Dislodged Reactor Vessel Surveillance Capsules in Unit 1 Reactor**

References:

1. TVA Letter to NRC, "Sequoyah Nuclear Plant, Units 1 and 2 License Renewal", dated January 7, 2013
2. TVA letter to NRC, "Sequoyah Reactor Pressure Vessel Surveillance Capsule Withdrawal Schedule Revision Due to License Renewal Amendment," dated January 10, 2013
3. NRC letter to TVA, "Sequoyah Nuclear Plant, Units 1 and 2 – Revise the Reactor Pressure Vessel Material Surveillance Capsule Withdrawal Schedule due to License Renewal Amendment (TAC NOS. MF0631 and MF0632)," dated September 27, 2013
4. TVA letter to NRC, "Sequoyah Nuclear Plant - Revision to Reactor Pressure Vessel Surveillance Capsule Withdrawal Schedule for License Renewal," dated May 14, 2015
5. NRC letter to TVA, "Safety Evaluation Report Related to the Sequoyah Nuclear Plant, Units 1 and 2, License Renewal Application (TAC NOS. MF0481 and MF0482)," dated January 29, 2015
6. Westinghouse Report , "Sequoyah Unit 1 U1R20 10-YR Reactor Vessel IVVI & Loose Part Inspections Field Service Report," dated May 2015
7. Westinghouse Procedure WDI-STD-088, Rev. 12 for Sequoyah Unit 1

During the Sequoyah Nuclear Plant (SQN) Unit 1 end of cycle (EOC) R20 outage, inspections of the Unit 1 reactor vessel internals revealed that the reactor pressure vessel (RPV) material surveillance capsules S and W had become dislodged from their intact designated baskets and were no longer available to provide fluence data. These capsules had been relocated in Unit 1 EOC R19. ASME Section XI inservice inspections of the Sequoyah (SQN) Unit 1 reactor vessel internals identified that specimen pieces of the capsules had come in contact with some of the reactor vessel internals components.

On April 30, 2015, TVA conducted a phone call with Nuclear Regulatory Commission (NRC) staff to inform the staff of the damage to the Unit 1 Capsules S and W. The NRC requested that SQN evaluate the impacts to the SQN License Renewal Application (LRA) (Reference 1). TVA has evaluated the following potential impacts: (1) the reactor vessel surveillance capsule withdrawal schedule, (2) LRA commitment changes, (3) potential impacts to MRP-227-A assumptions that support the SQN Reactor Vessel Internals Aging Management Program defined in the LRA, and (4) Potential Aging Management mechanisms involved in the damaged S and W Capsules. Results of the impact reviews are provided herein. Planned corrective actions to prevent recurrence are described in this letter.

TVA's evaluation of the impact to the SQN LRA are addressed below.

1. **Reactor Vessel Surveillance Capsule Withdrawal Schedule**

In Reference 2, TVA provided the NRC with a revised Reactor Vessel Surveillance Capsule Withdrawal Schedule for review and approval. As discussed in Reference 2, relocation of SQN Unit 1 Capsule S to a location of higher fluence was required to ensure that a Unit 1 capsule withdrawal schedule consistent with the expectations of American Society for Testing and Materials (ASTM) E-185-82 and compliance with 10 CFR 50, Appendix H could be attained. The Unit 1 withdrawal schedule required Capsule S to be removed during U1 EOC R28 outage. The capsule withdrawal schedule revision was approved by the NRC in Reference 3. SQN relocated the Unit 1 Capsule S to the 40-degree higher fluence location during the EOC R19 outage. Capsule W was also relocated during EOC R19 outage to a location of higher fluence (220-degree) for future removal to support a potential plant life extension beyond 60 years. The Capsule W withdrawal schedule (Standby) is unchanged from the current Final Safety Analysis Report.

Because the Capsule S is no longer available, the revised capsule withdrawal schedule provided to the NRC in Reference 2 and approved by the NRC in Reference 3 cannot be completed.

In Reference 4, TVA submitted a second revised Reactor Vessel Surveillance Capsule Withdrawal Schedule for SQN Unit 1 to the NRC for review. The second revised schedule complies with the expectations of ASTM E-185-82, NUREG-1801, Revision 2, "Generic Aging Lessons Learned (GALL) Report," and the requirements of 10 CFR 50, Appendix H.

2. LRA Commitment Changes

In letter dated January 7, 2013 (Reference 1), TVA submitted an application to the NRC to renew the operating licenses for SQN Units 1 and 2. The request would extend the licenses for an additional 20 years beyond the current expiration dates.

NRC developed its final Safety Evaluation Report (SER) (ML15021A356) to document findings associated with the safety review of TVA's LRA and supporting documentation for SQN. In Reference 5, NRC determined that TVA's LRA was complete and closed all open items.

Appendix A of the NRC SER provides a list of TVA commitments related to SQN's aging management programs. The list included Commitment No. 28 that addresses SQN's Reactor Vessel Surveillance Program. Commitment No. 28 (Item B) provides a schedule for withdrawal of SQN's capsules to meet the ASTM E-185-82 requirements, including the possibility of operation beyond 60 years.

<u>Commitment No. 28 (Item B)</u>

B. Revise Reactor Vessel Surveillance Program procedures to incorporate an NRC-approved schedule for capsule withdrawals to meet ASTM E 185-82 requirements, including the possibility of operation beyond 60 years (refer to TVA Letter to NRC, "Sequoyah Reactor Pressure Vessel Surveillance Capsule Withdrawal Schedule Revision Due to License Renewal Amendment," dated 01/10/13, ML 13032A251; NRC final safety evaluation report approved on 09/27/13, ML 13240A320)

As noted above, SQN Unit 1 cannot comply with the specimen withdrawal schedule for Capsule S as defined by the TVA letter to the NRC referenced in this commitment. As noted previously, TVA has submitted a revised capsule withdrawal schedule (Reference 5) that complies with (ASTM) E 185-82, NUREG-1801, Revision 2, "Generic Aging Lessons Learned (GALL) Report," and the requirements of 10 CFR 50, Appendix H. Accordingly, TVA is proposing a revision to Commitment No. 28 to divide Item B into two parts (B.1 and B.2) as follows:

(Note that deletions are struck-through and additions are underlined.

<u>Revised Commitment No. 28 (Item B)</u>
<u>B.1</u> Revise <u>Unit 2</u> Reactor Vessel Surveillance Program procedures to incorporate an NRC-approved schedule for capsule withdrawals to meet ASTM-E185-82 requirements, including the possibility of operation beyond 60 years (refer to the TVA Letter to NRC, "Sequoyah Reactor Pressure Vessel Surveillance Capsule Withdrawal Schedule Revision Due to License Renewal Amendment," dated 01/10/13, ML13032A251; NRC final safety evaluation report approved on 09/27/13, ML13240A320).

<u>B.2</u> Revise <u>Unit 1</u> Reactor Vessel Surveillance Program procedures to incorporate an NRC-approved schedule for capsule withdrawals to meet ASTM E 185-82 requirements, including the possibility of operation beyond 60 years (refer to TVA Letter to NRC, ~~"Sequoyah Reactor Pressure Vessel Surveillance Capsule Withdrawal Schedule~~

~~Revision Due to License Renewal Amendment," dated 01/10/13, ML 13032A251; NRC FSER approved on 09/27/13, ML 13240A320~~ *Sequoyah Nuclear Plant - Revision to Reactor Pressure Vessel Surveillance Capsule Withdrawal Schedule for License Renewal), dated May 14, 2015.*

Enclosure 1 provides the revised Commitment No. 28.

3. Potential Impact to MRP-227-A Assumptions

The Westinghouse Reactor Vessel Internals group reviewed the results of the SQN Unit 1 Reactor Vessel Internals Foreign Object Search and Retrieval (FOSAR) effort and ASME Section XI VT-3 Inservice Inspections to determine the possibility of internals damage associated with the dislodged RPV surveillance capsule loose parts. Completed structural evaluation and disposition reports as well as the chemistry evaluation and disposition reports were also reviewed. The results are reported in Enclosure 2.

References 6 and 7 support the Westinghouse conclusion that the observed Reactor Vessel Internals components were within existing design considerations and qualification of the components. This conclusion was based on: (1) there was no measurable impact to original design data, (2) there was no risk to structural integrity for any loading condition, (3) there was no impact on corrosion potential as a result of dissolution of materials in the primary water, and (4) there was no risk for increased crack initiation as a result of localized stress, noting specifically the lower support casting and instrumentation guide extension evaluation. References 6 and 7 are available at the site (TVA offices) for review.

To determine if the inspection findings necessitated a change to the MRP-227-A requirements and inspection protocols for the SQN Reactor Vessel Internals (RVI) aging management program (AMP), the following were considered:

a. Did aging related degradation contribute to the failure of the specimen capsule?
b. Did the surveillance capsule loose parts cause damage to the reactor vessel internals components that would impact their design function or expected life?
c. Do the indications cause a change to the SQN Unit 1 RVI AMP MRP-227-A requirements and protocols (e.g., expansion to primary)?

Factored into this review is the following SQN Unit 1 Operating Experience (OE) relative to this event.

a. No evidence of increases in fluence on components other than what would be expected as a result of operation.
b. No evidence of increases in temperature.
c. No material changes due to chemicals in the primary water.
d. No evidence of cracking.
e. No indication of structural frequency shifts.
f. Multiple indications of superficial wear that are not isolated to being caused by this event.

The Westinghouse OE evaluation for the dislodged SQN Unit 1 capsule specimens concluded that without a driver for degradation initiation or propagation, there is no basis to identify or technically justify a cause for concern that this event would cause a noncompliance or need for change in the existing MRP-227-A.

As a result, Westinghouse concluded that the assessment for MRP-227-A applicability, including consideration of the specific requirements of NRC SER Applicant Licensee Action Items (A/LAI) 1 and 2, remain valid as is. There is no impact to the existing SQN Reactor Vessel Internals aging management programs that was previously reviewed and approved by the NRC in Reference 5.

4. Potential Aging Management mechanisms involved in the damaged capsules.

The TVA and Westinghouse root cause analyses (RCAs) for the damaged specimens in the SQN Unit 1 Reactor Vessel are complete. The direct cause of this event is installation errors and installation procedural inadequacies. Specimen and specimen basket failure due to unanticipated potential aging effects was not identified as a direct or contributing cause.

Enclosure 1 provides the proposed revision to TVA's Commitment No.28 (Item B). Enclosure 2 provides an Evaluation of Failed Specimen Capsules on the Reactor Internals Aging Management Program Plan at Sequoyah Unit 1.

Please address any questions regarding this submittal to Erin Henderson, SQN Site Licensing Manager, at (423) 843-7170.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 10th day of July 2015.

Respectfully,

J. W. Shea
Vice President Nuclear Licensing

Enclosures:

1. Proposed Revisions to Commitment No. 28 and LRA Sections A.1.35 and B.1.35

2 Westinghouse Letter LTR-RIAM-15-40-NP, R0, Evaluation of Failed Specimen Capsules on the Reactor Internals Aging Management Program Plan at Sequoyah Unit 1, (Non-Proprietary)

cc:

Regional Administrator, RII
NRC Project Manager, Sequoyah License Renewal Project
NRC Senior Resident Inspector, Sequoyah Nuclear Plant

ENCLOSURE 1

Tennessee Valley Authority

Sequoyah Nuclear Plant Units 1 and 2,

Proposed Revisions to Commitment No. 28 and LRA Sections A.1.35 and B.1.35

Current Commitment No.28

28	A. Revise **Reactor Vessel Surveillance Program** procedures to consider the area outside the beltline such as nozzles, penetrations and discontinuities to determine if more restrictive P-T limits are required than would be determined by just considering the reactor vessel beltline materials. B. Revise Reactor Vessel Surveillance Program procedures to incorporate an NRC-approved schedule for capsule withdrawals to meet ASTM-E185-82 requirements, including the possibility of operation beyond 60 years (refer to the TVA Letter to NRC, "Sequoyah Reactor Pressure Vessel Surveillance Capsule Withdrawal Schedule Revision Due to License Renewal Amendment," dated 01/10/13, ML13032A251; NRC final safety evaluation report approved on 09/27/13, ML13240A320). C. Revise Reactor Vessel Surveillance Program procedures to withdraw and test a standby capsule to cover the peak fluence expected at the end of the period of extended operation.	B.1.35	SQN1: Prior to 03/17/2020 SQN2: Prior to 03/15/2021	Letter ML13190A276 (7/1/13)

Revised Commitment No. 28

28	A. Revise **Reactor Vessel Surveillance Program** procedures to consider the area outside the beltline such as nozzles, penetrations and discontinuities to determine if more restrictive P-T limits are required than would be determined by just considering the reactor vessel beltline materials. <u>B.1</u> Revise <u>Unit 2</u> Reactor Vessel Surveillance Program procedures to incorporate an NRC-approved schedule for capsule withdrawals to meet ASTM-E185-82 requirements, including the possibility of operation beyond 60 years (refer to the TVA Letter to NRC, "Sequoyah Reactor Pressure Vessel Surveillance Capsule Withdrawal Schedule Revision Due to License Renewal Amendment," dated 01/10/13, ML13032A251; NRC final safety evaluation report approved on 09/27/13, ML13240A320). <u>B.2 Revise Unit 1 Reactor Vessel Surveillance Program procedures to incorporate an NRC-approved schedule for capsule withdrawals to meet ASTM-E185-82 requirements, including the possibility of operation beyond 60 years (refer to the TVA Letter to NRC, Sequoyah Nuclear Plant, Revision to Reactor Pressure Vessel Surveillance Capsule Withdrawal Schedule for License Renewal dated May 14, 2015).</u> C. Revise Reactor Vessel Surveillance Program procedures to withdraw and test a standby capsule to cover the peak fluence expected at the end of the period of extended operation.	B.1.35	SQN1: Prior to 03/17/2020 SQN2: Prior to 03/15/2021	Letter ML13190A276 (7/1/13)

Changes to **LRA Section A.1.35**, Reactor Vessel Surveillance Program, follow with additions underlined and deletions line through.

The Reactor Vessel Surveillance Program will be enhanced as follows.

- Revise Reactor Vessel Surveillance Program procedures to consider the area outside the beltline such as nozzles, penetrations and discontinuities to determine if more restrictive pressure-temperature limits are required than would be determined by just considering the reactor vessel beltline materials.

- Revise Unit 2 Reactor Vessel Surveillance Program procedures to incorporate an NRC-approved schedule for capsule withdrawals to meet ASTM E 185-82 requirements, including the possibility of operation beyond 60 years (refer to the TVA Letter to NRC, Sequoyah Reactor Pressure Vessel Surveillance Capsule Withdrawal Schedule Revision Due to License Renewal Amendment," dated January 10, 2013 (ML13032A251)).

- Revise Unit 1 Reactor Vessel Surveillance Program procedures to incorporate an NRC-approved schedule for capsule withdrawals to meet ASTM E 185-82 requirements, including the possibility of operation beyond 60 years (refer to the TVA Letter to NRC, Sequoyah Nuclear Plant, Revision to Reactor Pressure Vessel Surveillance Capsule Withdrawal Schedule for License Renewal dated May 14, 2015).

Changes to **LRA Section B.1.35**, Reactor Vessel Surveillance Program, follow with additions underlined and deletions line through.

The following enhancements will be implemented prior to the period of extended operation.

Element Affected	Enhancement
4. Detection of Aging Effects	Revise Unit 2 Reactor Vessel Surveillance Program procedures to incorporate an NRC-approved schedule for capsule withdrawals to meet ASTM-E185-82 requirements, including the possibility of operation beyond 60 years (refer to the TVA Letter to NRC, Sequoyah Reactor Pressure Vessel Surveillance Capsule Withdrawal Schedule Revision Due to License Renewal Amendment, dated January 10, 2013 (ML13032A251)). Revise Unit 1 Reactor Vessel Surveillance Program procedures to incorporate an NRC-approved schedule for capsule withdrawals to meet ASTM E 185-82 requirements, including the possibility of operation beyond 60 years (refer to the TVA Letter to NRC, Sequoyah Nuclear Plant, Revision to Reactor Pressure Vessel Surveillance Capsule Withdrawal Schedule for License Renewal dated May 14, 2015).

ENCLOSURE 2

Tennessee Valley Authority

Sequoyah Nuclear Plant, Unit 1

Westinghouse Letter LTR-RIAM-15-40-NP, R0,
Evaluation of Failed Specimen
Capsules on the Reactor Internals Aging
Management Program Plan
at Sequoyah Unit 1
(Non-Proprietary)

Westinghouse

To:	Ronald Kucharski, Linda Evans	Date:	June 25, 2015
cc:	Cheryl L. Boggess		
From:	Reactor Internals Aging Management	Your ref:	N/A
Ext:	(412) 374-3692	Our ref:	LTR-RIAM-15-40-NP, Rev 0
Fax:	(724) 940-8559		

Subject: **Evaluation of Failed Specimen Capsules on the Reactor Internals Aging Management Program Plan at Sequoyah Unit 1**

References:

1. Westinghouse Document, WCAP-17690-NP, Rev. 0, "PWR Vessel Internals Program Plan for Aging Management of Reactor Internals at Sequoyah Nuclear Plant Unit 1," August 14, 2013.
2. TVA Letter, CNL-14-221, "Response to NRC Request for Additional Information Regarding the Review of the Sequoyah Nuclear Plant, Units 1 and 2, License Renewal Application, Set 23, B.1.34-9.d (Follow up) (TAC Nos. MF0481 and MF0482), December 11, 2014. (ADAMS ML14350A683).
3. Westinghouse Letter, LTR-AMER-MKG-15-738, Rev. 0, "Westinghouse Offer to TVA to Revise Capsule Pull Schedule with Existing Fluence Projections at Sequoyah Units 1 and 2 and Evaluation of Outage Specimen Basket Operating Experience on the Reactor Internals Aging Management Program Plan at Sequoyah Unit 1," May 4, 2015. (Proprietary)
4. TVA Letter, N10698, 05-14-2015.
5. TVA Letter, N10694, 05-11-2015.

Attachments:

1. Summary of Resolutions to TVA Letter N10698 Review Comments
2. Evaluation of Failed Specimen Capsules on the Reactor Internals Aging Management Program Plan at Sequoyah Unit 1

The purpose of this letter is to request transmittal of the non-proprietary version of the Westinghouse evaluation of the failed specimen capsules operating experience effect on the reactor internals aging management program plan (AMP) for Sequoyah Unit 1 (U1), WCAP-17690-NP [1, 2]. Attachment 2 contains the results of the evaluation, including a summary of the conclusions of the evaluation supported by component level justifications as applicable. Comments as received from the Tennessee Valley Authority (TVA) review of the draft of the evaluation were provided in [4]. This letter incorporates the Westinghouse resolutions to the comments from [4] and reclassifies that document as non-proprietary for general release. The evaluation for impacts on the TVA U1 AMP as provided in Attachment 2 is suitable for submittal to the U. S. Nuclear Regulatory Commission (NRC).

The conclusion that there is no impact to the existing reactor internals aging management program plan contained in WCAP-17690-NP [1] or updates to the contents as a result of NRC requests for additional information (RAI) [2] remains unchanged as a result of the resolutions to the TVA review. This evaluation and the conclusions herein are heavily dependent on the TVA communication in [5], which stated that specimen capsule failures due to unanticipated aging effects are not probable and are not being considered by the TVA root cause analysis (RCA) team. Visual inspection of the accessible area of the intact capsules and their respective baskets indicated

no obvious signs of degradation and the communication stated that an aging effect associate with surveillance capsule materials and environment is unlikely.

Please transmit this letter to Dennis Lundy and David LaFever. Please include all Westinghouse addressees on copy in the project letter distribution and any subsequent communications on the content.

If there are any questions or need for additional information, please contact Josh McKinley at (412) 374-3692 or Charlie Meyer at (412) 374-2688.

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Attachment 1: Summary of Resolutions to TVA Letter N10698 Review Comments

TVA Letter N10698, 05-14-2015, Enclosure 1. "Doc. Rev. 00(Draft B) Rev. Date: 05-12-2015."

Comment(s) By: Gay Halliburton Date: 05/14/2015

Resolution(s) By: LTR-RIAM-15-40 Rev. 0

No.	TVA Comment/Reason	Resolution
1	Attachments 2, Page 9 – For consistency with TVA Letter N10694, please revise the first bullet in the second paragraph of the "Conclusion" section to read, "Specimen capsule failure due to an unanticipated aging effect is sufficiently remote that it is not being considered by the TVA RCA".	Updated as requested.
2	Attachment 2, Page 10 – Please update Reference 2 to reflect issue of Revision 00 of the report on May 13, 2015.	Updated as requested.
3	Attachment 2, Page 10 – Please update Reference 13 to reflect submittal of TVA Letter N10696 on May 13, 2015.	Updated as requested

Attachment 2:
Evaluation of Failed Specimen Capsules on the
Reactor Internals Aging Management Program Plan at Sequoyah Unit 1

Introduction and Purpose

Recent operating experience (OE) at Tennessee Valley Authority (TVA) Sequoyah Unit 1 (U1) confirmed failed specimen capsules in the reactor vessel. The utility focus is now on evaluating impacts of the OE on the plant operating requirements and protocols for the current licensing basis (CLB) and the period of extended operation (PEO). Included in the effort is an evaluation of the impact of the OE on requirements for implementation and execution of the MRP-227-A reactor internals materials aging management program plan (AMP).

The purpose of this evaluation is to assess the impact of the failed surveillance capsule OE on the implementation of the TVA U1 MRP-227-A program to manage aging of the reactor internals. The assessment is intended to either affirm the continued applicability of the existing AMP requirements or identify changes needed as a result of the OE. The primary considerations in the assessment are:

- A summary of the OE, including efforts to monitor the event using the Loose Parts Monitoring System (LPMS) and neutron noise monitoring.
- Damage Assessments
  - ASME Section XI (SXI) In-Service Inspections (ISI) observations
  - Foreign Object Search and Retrieval (FOSAR) observations
  - Available structural evaluation and disposition reports
  - Chemistry evaluation and disposition reports
- Evaluation of factors that impact MRP-227-A degradation mechanisms
  - Review of factors that contribute to MRP-227-A aging degradation mechanisms
  - Review of OE to identify conditions that would potentially impact MRP-227-A protocols

This reactor vessel internals (RVI) impact evaluation did not address the root cause of the capsule failure. The conclusions drawn here are based on the premise that surveillance capsule failures were not triggered by an aging-related mechanism in either the capsules or the baskets that hold the capsules. This starting assumption is based on the results to-date from the root cause analysis (RCA) being conducted by TVA [1]. If an aging-related root cause is subsequently identified, the impact of MRP-227-A would need to be evaluated separately.

This evaluation references results and conclusions of the SXI ISI and the FOSAR programs. This report is based on the SXI ISI report [2]. The impact of the observations recorded in the SXI ISI report on aging degradation in the internals components is considered; however, final disposition of the inspection findings are the responsibility of plant engineering and are beyond the scope of the MRP-227-A aging management impact evaluation.

As a result of the evaluation summarized in this report, it was concluded that there is no impact to the existing TVA U1 reactor internals aging management program plan contained in WCAP-17690-NP [3] and subsequent updates to the WCAP contents as a result of U. S. Nuclear Regulatory Commission (NRC) requests for additional information (RAI) [4].

Background

As noted in the NRC Safety Evaluation Report (SER) [5], TVA U1 is scheduled to enter the PEO on September 17, 2020. Implementation of MRP-227-A [6], the industry inspection and

evaluation (I&E) guidelines for reactor vessel internals (RVI) aging management, includes regular assessments of OE and evaluation of information on the plant-specific AMP. TVA U1's AMP is defined in WCAP-17690-NP [3] and PEO commitments [4]. Although TVA U1 has not yet entered the PEO, this capsule failure event has initiated an effort to review the currently defined PEO RVI AMP requirements and assess any impacts on the requirements and protocols.

Sequoyah Unit 1 completed their 1R19 (Fall 2013) refueling outage in November 2013. After returning to service, unusual activity was noted on the reactor vessel bottom head channels of their LPMS [7]. Following detection of the loose part, the bottom vessel channel was monitored three times a week for the duration of the cycle by listening to the audio produced by the LPMS. A loose parts assessment, to bound the extent of the potential effects, was performed. In addition, periodic and historic neutron noise monitoring data was analyzed to confirm normal reactor internals dynamic behavior and that there was no wedged part concern [8]. It was confirmed that the dynamic behavior of the internals was unchanged from baseline data, and it was concluded that there was no wedged loose part. The assessment recommended the loose parts be retrieved in the subsequent outage.

Extensive efforts to identify and assess impacts of the event have now been completed. FOSAR was performed during the 1R20 (Spring 2015) outage. The scheduled 10-year ISI activities were also performed [2]. Two TVA U1 Reactor Vessel (RV) surveillance capsules, repositioned in the U1C19 Refueling Outage (RFO), were confirmed to not be in the location of record. The specimen capsules fell into the bottom of the vessel and were destroyed over time. A portion of the contents of the capsules were confirmed as being released into the reactor vessel. There were also intact sections of the capsule recovered which contain additional specimens. Damage was observed on some of the RVIs, including dings, wear, and other anomalies identified on RVI components below the bottom of the fuel. The four-face inspection of the fuel during the core-off loading process revealed small objects resting beneath or attached to the bottom fuel nozzles.

The loose parts were found to have caused no significant damage to the vessel or its internals [7]. In the unlikely event that small loose parts are in the vessel, it was determined that none of the potential parts pose a significant safety risk to the form, fit or function of the reactor internals components for future plant operation [2, 7, 8, and 9]. Key items considered in the assessments completed were loads as a result of a part hitting another component in its path or wedging of a loose part that would impede component fit-up.

RVI Component Damage Assessment

A general condition SXI VT-3 ISI was performed on reactor internals components [2]. In addition to the reactor vessel interior, the scope of the SXI RVI inspections included the accessible areas of the Upper Internals, lower internals exterior, and lower internals interior. A complete list of components included in the SXI ISI scope is contained in [2]. Of note are the following components:

1. Upper core plate
2. Guide tubes
3. Upper core plate alignment pins
4. Hold down spring
5. Upper core barrel flange
6. Upper core barrel to flange weld

7. Core barrel outlet nozzles
8. Upper to lower core barrel girth weld
9. Thermal shield flexures
10. Lower core support (clevis insert and radial keys)
11. Attachments to the bottom of the accessible areas of the lower support plate (casting) dome of the lower internals
12. Lower core plate (top surface)
13. Baffle-former assembly plates and former bolts

The ISI Quality Assurance (QA) compliant inspection results report [2] concluded that there were "No Recordable Indications" (NRI) [10] for a majority of the components that may interact with the reactor vessel internals (typically ASME B-N-3 categorization), such as at the radial support keyways, and for the SXI internals components. Exceptions with the identification of "Recordable Indications" (RI) [2, 10] were noted at the following locations. These locations were either on RVI components or could affect internals components:

1. Three gouges (previously reported) on one core barrel outlet nozzle
2. Debris near the upper support plate to drive shaft connection
3. Typical corrosion products foreign material on the top former plate top ledge of the baffle to former plate
4. Deep rub at an instrument guide extension (G-5 location) and light rub on instrument guide extension at H-4
5. Flow nozzle, light dents
6. Thermal shield flexure rub marks at 56^{o} and 124^{o} locations
7. Energy absorber deep rub
8. Typical corrosion products and foreign material on the top former plate
9. Baffle plate lower grid strap rub marks
10. Lower core plate (LCP) corrosion products and staining

Item 1 was previously evaluated and dispositioned in [11, 12], where it was concluded that there was no concern for continued operation. This past information was coupled with the current ISI data [2] and re-evaluated for this OE impact report from a MRP-227-A impact perspective. The "use as-is" previous disposition and conclusion is confirmed. There is no need to perform monitoring inspections more frequently than the existing 10-year ISI interval, and it is concluded that the MRP-227-A aging requirements as defined in the TVA U1 AMP [3, 4] remain valid and applicable.

Items 2 and 3 were accepted as is by Reactor Engineering [2] and no further assessment is needed to assess impacts on the MRP-227-A based AMP.

The bottom-mounted instrumentation (BMI) guide extension wear (Item 4) was not explicitly evaluated in [7]; however a similar, but less significant, abrasion of a BMI column was identified during FOSAR and was dispositioned in [7]. In the evaluation of the BMI column abrasion [7], it was determined that because the BMI column is not a core support structure or pressure boundary component, and the stresses in this component are low, that the light wear (abrasion) would have no adverse effect on plant operations such that the design functions of any systems, structures or components would be adversely affected during subsequent fuel cycles. By comparison, the wear identified in Item 4 appears to have a measureable depth; however, the cross-section in this region of the BMI extension is significantly greater than the BMI column evaluated in [7], and loads in this region of the BMI extension are less than those applied to the BMI column considered for

abrasion in [7]. Therefore, the same conclusions (i.e., no adverse effect on plant operations such that the design functions of any systems, structures or components would be adversely affected during subsequent fuel cycles) are expected to apply. This is consistent with the conclusions of the disposition by TVA [13]. Furthermore, as summarized in [7], stresses in this region are low, and there is no concern for intensified stress as a result of discontinuities caused by wear. These components are in a low temperature and low fluence area of the vessel and remote from the fuel. Based on these factors, it is concluded that there is no impact on the TVA U1 aging scenario for Item 4.

Items 5 to 10 were dispositioned by TVA [13] similar to items 1 to 4. A review of these items reveals nothing that would impact the structural integrity of these components or the RVI. As such, it is concluded there is no impact on the TVA U1 existing aging scenario for items 5 to 10.

To support evaluations for impacts of the OE and locate loose parts, additional inspections outside of the existing SXI ISI requirements were conducted on the internals components. FOSAR video inspections of the lower internals lower dome area that were not of SXI VT-3 quality were also completed [7]. A general condition video was generated for accessible areas of the following components:

- Lower core plate
- Diffuser plate
- Lower support columns
- Lower support casting
- BMI columns

Components were found to be in good general condition without any obvious damage or flaws. Items to note included:

- A scratch on a baffle plate
- An abrasion on an instrumentation column
- Material loss on the lower support casting
- A dent on the edge of a lower support casting flow hole

These types of minor surface defects were attributed to the movement of debris within the reactor during operation. There is no detrimental impact to the function of these components as a result of the identified damage, and none of these surface defects present a risk to the structural integrity of the components listed [7]. Assessments evaluated did not include depth confirmation of any items as a result of the damage, but the general assessments considered postulated worst-case scenarios and postulated localized stress impacts as a result of the observed damage and potentially unidentified loose parts [7]. The loose parts impacting assessments concluded that stress and fatigue for both short and long term postulated bounding effects were low and would not impart loads that would compromise the existing plant design limiting loads, and no impacts to form/fit/function of any components as a result of unidentified loose parts.

Assessments also included investigation of primary water chemistry [9]. The chemistry assessment concluded that there was no impact on corrosion of pressure boundary materials or reactor internals.

The TVA RCA for the lost specimen capsule event has (to date) identified no potential causes of the event that are associated with aging effects of RVI components within the scope of MRP-227-A [1]. There is no indication to date that the failure is associated with an aging-related basket failure [1].

After coupling the RCA observation with the conclusions made in the structural and chemical assessment, it is concluded that the damage is within existing design considerations and qualification. However, to ensure safety and manage impacts from unknowns, an inventory of the intact recovered capsule sections was recommended [7]. Continued monitoring and confirmation of no additional loose parts at the next RFO were also recommended. It is concluded that following these actions could be credited with providing a level of margin to mitigate any postulated potential concerns and unknown contributing factors, such as expansion of the observed damage or additional postulated damage scenarios due to unrecovered loose parts.

MRP-227-A Applicability Evaluation as a Result of the U1C20 OE

TVA U1's AMP for reactor internals, as defined in WCAP-17690-NP [3] and PEO commitments [4], follows MRP-227-A [6]. The PEO commitment was added to the generic industry requirements in response to an NRC RAI, and augments the MRP-227-A requirements for the Primary inspection of the upper and lower core barrel cylinder girth welds to include expansion to the bottom surface of the upper core plate. This augmented requirement addresses the potential for irradiation embrittlement as a result of higher fluence levels on the core side of the upper core plate [4]. All other requirements for TVA U1 are identical to the MRP-227-A requirements. Demonstration of MRP-227-A applicability and implementation timing of the AMP were satisfied. This is consistent with the CLB requirements for the PEO, as summarized in the NRC PEO SER [5]. TVA U1 is scheduled to enter the PEO on September 17, 2020. The expected date for completion of inspection and testing activities for TVA U1 to support entering the PEO is prior to March 17, 2020 or the end of the last refueling outage prior to entering the PEO, whichever occurs later.

The TVA U1 CLB specifies an ASME Section XI ISI sampling-based program as an element of its strategy for management of RVIs [3, 5]. The CLB program consists of periodic volumetric, surface and/or visual examination of components for assessment, signs of degradation and corrective actions. Review of industry and plant-specific OE is included in these periodic activities. The evaluation of the OE undertaken here is compliant with these CLB items.

TVA U1 is positioned to enter the PEO and has initiated an evaluation to assess the failed specimen capsule OE for impacts on the reactor internals AMP applicability and requirements. Key questions to consider in evaluating the OE for impacts in the PEO include:

1. Did aging related degradation contribute to the failure of the specimen capsule?

   A root cause evaluation by TVA is currently in progress [1]. The evaluation is proceeding with the premise that there is no indication that the event is associated with an aging-related failure [1]. The RCA has identified no potential causes of the event that are associated with aging effects of RVI components within the scope of MRP-227-A [6]. Because there is no evidence of aging degradation, there is no driver for revising the existing component susceptibility ranking and inspection requirements contained in the TVA U1 AMP to address the specimen capsule failure [3, 4].

2. Did the OE cause damage to the RVIs that would impact their design function or expected life?

   The ISI report [2], the completed loose parts evaluation [7] and the completed chemistry report [9] contain a summary of observations and evaluations as a result of the SXI and FOSAR inspections. The discussion on impacts from OE damage concluded that there were no impacts to form/fit/function of any components, considering:

   - There was no measurable impact to original design data.
   - There was no risk to structural integrity for any loading condition.
   - There was no impact on corrosion potential as a result of dissolution of materials in the primary water.
   - There was no risk for increased crack initiation as a result of localized stress, noting specifically the lower support casting and instrumentation guide extension evaluation.

3. Do the indications cause a change to the TVA U1 AMP MRP-227-A requirements and protocols? This would include considerations such as:

   - Adding a component to be inspected or monitored.
   - Adding an aging mechanism or mechanisms for any of the impacted components.
   - Promoting the component to a different inspection category (e.g., expansion to primary [6]).
   - Modifying or enhancing the specified inspection protocols (initial and subsequent timing, coverage, or inspection rigor [EVT-1 over a VT-3]).

   Assessments for aging over the licensed life of the plant must consider a range of scenarios. The TVA U1 AMP and MRP-227-A are based on a broad set of assumptions about plant operation, which encompass the range of plant conditions. Generally, the material susceptibility and ranking assessments of the individual reactor internals components were dependent on the population of components, component materials, stress, fluence, fabrication and chemistry considerations. The MRP-227-A material degradation mechanisms are dependent on specific drivers, and would result in the following physical effects:

   I. Cracking
      - SCC (stress and cold work)
      - IASCC (stress and dose)
      - Fatigue (transient loading)

   II. Embrittlement: Changes in material properties such as strength (increase), ductility (decrease) and toughness (decrease)
      - IE (dose and temperature)
      - TE (temperature and composition)

   III. Dimensional stability: Component distortion that could modify the stress-strain distribution in the structure
      - Void swelling (temperature and dose)
      - Thermal and irradiation enhanced stress relaxation or irradiation-enhanced creep (dose and stress)

   IV. Wear: Loss of material

Information available to date as a result of the OE contains:

- No evidence of increases in fluence on components other than what would be expected as a result of operation
- No evidence of increases in temperature
- No material changes due to chemicals in the primary water [9]
- No evidence of cracking as result of the OE [2 , 7]
- No indication of structural frequency shifts [8]
- Multiple indications of superficial wear that are not isolated to being caused by the OE

Additionally, the evaluation of the OE determined that:

- OE resulting from the capsule failure does not change the design-based inputs used in the original MRP-227-A susceptibility and ranking, but the mechanisms are dependent on contributing factors to different degrees.
- There is no known generic aging degradation that caused the specimen capsule failure. However, aging management protocols for monitoring and managing aging must be reviewed considering the condition of the components is not as originally installed or as considered for the basis of the original MRP-227-A aging assessments.

Without a driver for degradation initiation or propagation, there is no basis to identify or technically justify a cause for concern that the results of this OE would cause a noncompliance or need for a change in existing MRP-227-A based PEO component aging management requirements. The evaluations completed to date have not indicated any driver; therefore, it would be logical to conclude that there is no impact on the existing AMP requirements for the PEO.

It should be noted that the currently gathered information is based on several factors that could influence this evaluation and any conclusions, noting the following:

- The inspections completed to date were completed using VT-3 or less rigorous inspection techniques.
- There was no depth determined for the observed damage.
- Lost parts remain unrecovered.

However, the recommendations contained in the completed evaluations [7, 8] are intended to provide oversight and reasonable assurance that any postulated risk was managed when reviewing the observed damage.

The damage observed was attributed to normal operation, which was already considered in the original MRP-227-A assessments, or artificially introduced as a result of the failed specimen capsule OE. Given the evidence presented and considering the factors that would drive a need to change the AMP MRP-227-A based requirements and protocols, it is concluded that:

- No additional components need to be added for active management since there is no evidence of an active mechanism other than what was already considered, and no changes that would increase the factors that drive the degradations of concern.

- There is no need to add any mechanisms for aging management of any component since damage was attributed to known and expected drivers or to the artificially-induced effects of the failed specimen capsule loose parts, and because continual monitoring is in place. The continued monitoring was previously recommended to assist in managing potential downstream effects of damage or unknowns.
- No changes need to be made to existing inspection timing, coverage or rigor since damage was artificially introduced. No impact that increases driving parameters, such as temperature, fluence or stress, was generated, and the damage was within tolerances.

Conclusion

This evaluation reviews the recent OE against the MRP-227-A program parameters and requirements in determining any impact on the TVA U1 AMP [3, 4]. It utilizes engineering evaluation and disposition of ASME SXI and FOSAR ISI observations completed for TVA U1 to assess impacts on the PEO aging management requirements.

This evaluation to determine impacts and confirm applicability or provide recommendations on the reactor internals PEO AMP as a result of the recent lost specimen OE at TVA U1 has uncovered the following:

- Specimen capsule failure due to an unanticipated aging effect is sufficiently remote that it is not being considered by the TVA RCA.
- There is no adverse effect on plant operations or components such that the design functions of any systems, structures or components would be adversely affected during subsequent fuel cycles.
- There is no change to the RVI component population considered and evaluated in the TVA U1 AMP.
- Appropriate actions and programs are active at TVA U1 and will provide reasonable assurance of continued safe operation through activities such as monitoring and trending, to address any unknowns remaining in the evaluation completed to date.
- The evaluations completed to date have not indicated any new, enhanced, or accelerated driver for degradation.

Without a driver for degradation initiation or propagation, there is no basis to identify or technically justify a cause for concern that the results of this OE would cause a noncompliance or need for a change in existing MRP-227-A based PEO component aging management requirements.

As a result, the assessment for MRP-227-A applicability, including consideration of the specific requirements of NRC SER Applicant Licensee Action Items (A/LAI) 1 and 2 (as summarized in the TVA U1 AMP), remain valid as is. It is concluded that there is no impact to the existing reactor internals aging management program plan contained in WCAP-17690-NP [3] or updates to the contents because of NRC RAIs [4].

References:

1. TVA Letter, N10694, May 11, 2015.

2. WesDyne Report, WDI-PJF-1314382-FSR-001, Rev. 0, "10-Yr Reactor Vessel IVVI & Loose Part Inspections," May 13, 2015.

3. Westinghouse Document, WCAP-17690-NP, Rev. 0, "PWR Vessel Internals Program Plan for Aging Management of Reactor Internals at Sequoyah Nuclear Plant Unit 1," August 2013.

4. TVA Letter, CNL-14-221, "Response to NRC Request for Additional Information Regarding the Review of the Sequoyah Nuclear Plant, Units 1 and 2, License Renewal Application, Set 23, B.1.34-9.d (Follow up) (TAC Nos. MF0481 and MF0482)," December 11, 2014. (ADAMS ML14350A683).

5. NRC Safety Evaluation Report Related to the License Renewal of Sequoyah Nuclear Plant Units 1 and 2 Docket Nos. 50-327 and 50-328, January 2015 (ADAMS ML15021A356).

6. *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A).* EPRI, Palo Alto, CA: 2011. 1022863. (ADAMS ML12339A705).

7. Westinghouse Letter, LTR-RIDA-15-89, Rev. 1, "Sequoyah Unit 1 Failed Irradiation Specimen Capsule Safety Evaluation," May 11, 2015. (Proprietary)

8. Westinghouse Letter, LTR-RIDA-15-9, Rev. 0, "Evaluation of Core Barrel Vibration Monitoring System (CBVMS) Neutron Noise Data for Sequoyah Unit 1 Reactor Vessel Internals From January 9, 2015," February 12, 2015. (Proprietary)

9. Westinghouse Letter, LTR-CCOE-15-30, Rev. 0, "Impact on RCS Chemistry of Non-Steel Debris from Sequoyah Unit 1 Capsules W and S," April 28, 2015. (Proprietary)

10. WesDyne Document, WDI-STD-088, Rev. 12, "Underwater Remote Visual Examination of Reactor Vessel Internals," November 25, 2014. (Proprietary)

11. Westinghouse Letter, LTR-RIDA-06-29, Rev. 0, "Disposition of the Notices of Indications for the Visual Indications Found on the Sequoyah Unit 1 Loop 4 Reactor Vessel Outlet Nozzle Projection and Interfacing Core Barrel Nozzle During the 10-Year ISI," April 26, 2006. (Proprietary)

12. Westinghouse letter, LTR-RIDA-15-82, Rev. 0, "Transmittal of Sequoyah Unit 1 Outlet nozzle Indication Force Estimate Results," May 8, 2015. (Proprietary)

13. TVA Letter, N10696, May 13, 2015.