ATTACHMENT 2

Framatome ANP-3658 Revision 0

Evaluation of the McGuire Units 1 and 2 Upflow Modification for 60 Years

Technical Report

framatome

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Revision 0

Technical Report

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Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
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Nomenclature

(If applicable)

Acronym	Definition
AMP	Aging Management Program
BOL	Beginning-of-Life
CFR	Code of Federal Regulations
CUF	Cumulative Usage Factor
EOL	End-of-Life
EPRI	Electric Power Research Institute
FMECA	Failure Modes, Effects, and Criticality Analysis
I&E	Inspection and Evaluation
ISR/IC	Irradiation-Enhanced Stress Relaxation and Creep
MRP	Materials Reliability Program
NRC	Nuclear Regulatory Commission
PWR	Pressurized Water Reactor
RV	Reactor Vessel
SER	Safety Evaluation Report
SRP	Standard Review Plan
SR	Stress Relaxation
TLAA	Time-Limited Aging Analysis

ABSTRACT

In the 1990s, a bypass flow reversal was performed by Framatome at the McGuire Nuclear Station Units 1 and 2 to reduce the potential for baffle gap jet flow-induced fuel damage. The modification consists of reversing the flow in the region of the reactor vessel internals formed by the core barrel and the baffle plates to significantly reduce the pressure differential, and thus the jet flow potential across the baffle plate gaps.

The purpose of this document is threefold, which fulfills a regulatory commitment for Duke Energy and addresses time-limited aging analyses associated with the upflow modification under 10 CFR 54.21(c)(1)(i) and (ii):

1. Report the results of the screening, categorization, and inspection strategy steps for the plugs in the core barrel and the lower former plate.

2. Report the results of the evaluation of whether or not placing holes in the top former plate changes the screening parameters for the unmodified former plates in MRP-191.

3. Report the results of the five time-limited aging analyses for the core barrel plugs, lower former plate plugs, and the modified top former plate (with holes).

The results of the evaluations discussed above are as follows:

1. The same process used in MRP-191, consisting of data compilation and review; screening of components; a failure, modes, effects, and criticality analysis; and finally categorization and ranking, was used for the core barrel and lower former plate plugs. This resulted in the expert panel determining that Category A is appropriate, which screens these components out of further consideration. Therefore, no additional inspections are required per MRP-227-A for these plugs.

2. The process used to evaluate the modified top former plate included reviewing screening parameters associated with the new holes in the top former plate and determining if a modification of the screening parameters listed in Table A-1 in MRP-191 for the modified former plates is needed. The conclusion from this effort was that no changes due to the modification of the top former plates are needed to the screening parameters in Table A-1 of MRP-191.

3. Five time-limited aging analyses were considered for the period of extended operation; three time-limited aging analyses related to fatigue usage factors (one each for the modified top former plate (with holes), the core barrel plugs, and the lower former plate plugs) and two time-limited aging analyses related to irradiation-enhanced stress relaxation (one each for the core barrel plugs and the lower former plate plugs.)

   The fatigue usage calculation of the modified top former plate (with holes) was based on a stress range due to the thermal stress from gamma heating at full power (and pressure). It is assumed this is applicable for the current 60-year license for the modified top former plate (with holes).

   The transients shown below were considered in the fatigue analysis for the original 40-year license for the core barrel plugs and lower former plate plugs. These are enveloping transients which were used to bound the actual plant design transients. It is assumed that these transients are applicable for the current 60-year license for the core barrel plugs and lower former plate plugs for a fatigue analysis:

   - Normal conditions

     A thermal ramp up to steady state operating temperature, followed by a ramp back down.

- Upset condition number 1

  A large thermal step change up and down.

- Upset condition number 2

  A small thermal step change up and down.

The minimum required stress ratio for the core barrel and lower former plate plugs, based on loads and calculations of end-of-interface life pressure (core barrel plugs), minimum required test plug spring back (lower former plate plugs), and required contact pressure (core barrel and lower former plate plugs), was calculated for 60 years. This minimum required stress ratio value was compared to and determined to be less than the respective estimated stress ratio for both the core barrel and lower former plate plugs. Therefore, the core barrel and lower former plate plugs meet the requirements to remain in place for 60 years of plant life.

1.0 INTRODUCTION AND PURPOSE

In the 1990s, a bypass flow reversal was performed by Framatome at the McGuire Nuclear Station Units 1 and 2 to reduce the potential for baffle gap jet flow-induced fuel damage. The modification consists of reversing the flow in the region of the reactor vessel (RV) internals formed by the core barrel and the baffle plates to significantly reduce the pressure differential, and thus the jet flow potential across the baffle plate gaps.

In the original configuration (Figure 1-1(a)), reactor coolant flows from the downcomer through the core barrel holes, into the baffle region and downward between the core barrel and baffle plates. This configuration is deemed a "downflow" configuration. Due to the large pressure losses in the core, a significant pressure differential exists between the core barrel/baffle region and the core, especially at the top of the core. During reactor operation, the pressure differential across the baffle plate provides the driving force for flow jetting through any gaps that may exist in the region. The modified configuration (Figure 1-1(b)) includes plugging the existing holes in the core barrel, opening holes in the top former plate and plugging a percentage of the flow holes in the lower former plate. This configuration is deemed an "upflow" configuration. This modification effectively changes the direction of flow in the core barrel/baffle region from down to up and significantly reduces the baffle gap driving force that creates the change in pressure.

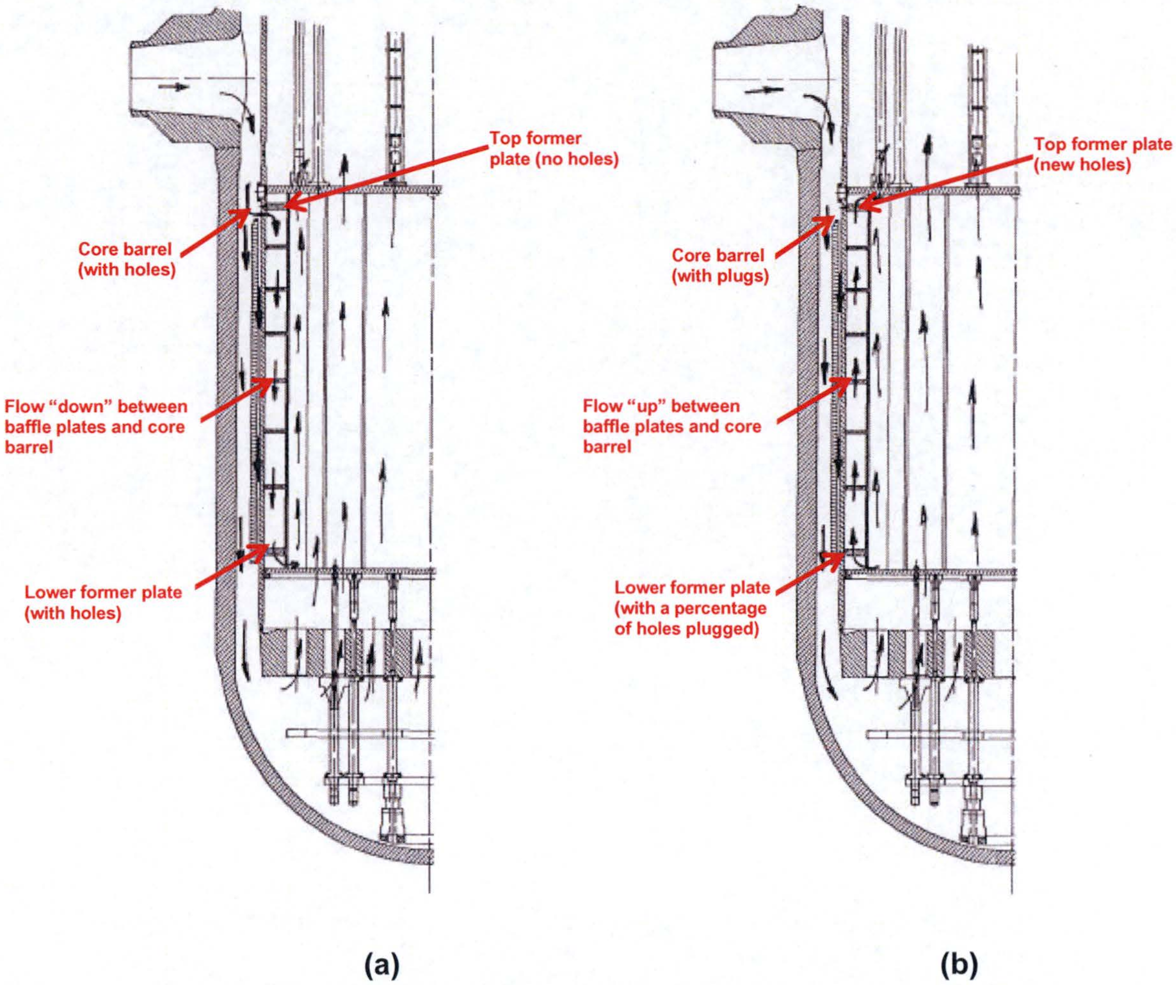

Figure 1-1. Schematic Representations of a Typical (a) "Downflow" Core Barrel-Baffle Plate Region Design (b) "Upflow" Core Barrel-Baffle Plate Region Design

Management of aging effects in RV internals is required for nuclear plants applying for renewed operating licenses, as specified in the Nuclear Regulatory Commission (NRC) Standard Review Plan (SRP) for License Renewal Applications [1]. The U.S. nuclear industry has been actively engaged in supporting the industry goal of responding to these requirements. Various programs have been established within the industry over the past 10-15 years to develop guidelines for managing the aging effects of pressurized water reactor (PWR) RV internals.

In December 2008, the Electric Power Research Institute (EPRI) issued Revision 0 of MRP-227 inspection and evaluation (I&E) guidelines for managing long-term aging of PWR RV internals and submitted the report to the NRC for review and approval in January 2009. In June 2011, a safety evaluation report (SER) for topical report MRP-227, Revision 0 was issued by the NRC and the SER was then subsequently revised to Revision 1 in December 2011 to ensure that all NRC required changes were included in the approved version of the topical report. Thus, EPRI re-issued the MRP-227 I&E guidelines, containing the final NRC SER, as MRP-227-A in December 2011 [2]. MRP-227-A provides generic augmented inspection requirements for the currently operating fleet of US PWRs.

Duke Energy submitted an Aging Management Program (AMP) and Inspection Plan for the McGuire Nuclear Station Units 1 and 2 Reactor Vessel Internals in December 2017, consistent with MRP-227-A [3, 4]. Section 4.1.7 of Reference 4 discusses background information related to the upflow conversion and the need for five time-limited aging analyses (TLAAs), three related to fatigue usage factors and two related to irradiation-enhanced stress relaxation (ISR), to be performed for this upflow conversion. Sections 6.2.1 and 6.2.2 of Reference 4 discuss the need to evaluate the core barrel and lower former plate plugs and the modified top former plate per the requirements of MRP-227-A and Applicant/Licensee Action Items 1 and 2. Section 8 of Reference 4 provides open items for the AMP and inspection plan, including the evaluation of the core barrel and lower former plate plugs and the modified top former plate. Regulatory commitment 4 is stated in Attachment 6 of Reference 3 as "Submittal to address Upflow Conversion Modification in accordance with Applicant/Licensee Action Items 1 and 2."

The purpose of this document is threefold, which will fulfill regulatory commitment 4 (as stated in Attachment 6 of Reference 3) and address TLAAs associated with the upflow modification under 10 CFR 54.21(c)(1) (i) and (ii) [5]:

1. Report the results of the screening, categorization, and inspection strategy steps for the plugs in the core barrel and lower former plate.

2. Report the results of the evaluation of whether or not placing holes in the top former plate changes the screening parameters for the unmodified former plates in MRP-191 [6].

3. Report the results of the five TLAAs for the core barrel plugs, lower former plate plugs, and the modified top former plate (with holes).

2.0 METHODOLOGY

The methodology for evaluation of each of the items of interest to this document will be discussed in this section.

2.1 *Core Barrel and Lower Former Plate Plugs (MRP-227-A Evaluation)*

The same process used in MRP-191 was used for evaluation of the core barrel and lower former plate plugs. Each description below reflects the specific actions used for the evaluation.

1. Data compilation and review

The conditions under which the plugs in the core barrel and in the lower former plate operate were characterized and documented.

2. Screening of Components

The age-related degradation mechanisms and the screening criteria used for this task were taken from MRP-175 [7]. The screening criteria were applied to the plugs for susceptibility to each of the age-related degradation mechanisms and the results were documented.

3. Failure Modes, Effects and Criticality Analysis (FMECA)

An expert panel consisting of knowledgeable Framatome individuals was consulted. Members with reactor internals design experience were included as appropriate in the FMECA process. The panel contributed expertise in the following areas:

- Component design
- Structural modeling and analysis
- Neutron fluence and radiation analysis
- Materials degradation and failure experience

- Component inspection experience

- Risk assessment

- System function and operating experience

Appropriate members of the expert panel reviewed the results of the susceptibility screening and endorsed or modified the results. The final results of this assessment were used in a "nine-box assessment" (3x3 matrix with low, medium or high probabilities of degradation vs. low, medium or high probabilities of damage/plant reliability) in order to arrive at a grouping of components for the categorization and ranking task. Irradiation-enhanced stress relaxation and irradiation creep (ISR/IC), fatigue, and wear were not considered as these age-related degradation mechanisms are being addressed in TLAAs under 10 CFR 54.21(c)(1) (i) and (ii) directly or indirectly (see Section 2.3) and are therefore not required to have an AMP.

4. Categorization and Ranking

Appropriate members of the expert panel used the initial ranking of component/effects to assign each screened-in component to one of the initial categorizations (Category A, B, or C). Factors that were considered in this activity included: the probability of occurrence of a degradation mechanism, the probability of failure, and the severity of any consequences with respect to operation, safety, financial impact, and plant reliability.

2.2 *Modified Top Former Plate with Holes (MRP-227-A Evaluation)*

The unmodified top former plate (with no holes) was screened in MRP-191. The process used to evaluate the modified top former plate (with new holes) includes reviewing screening parameters associated with the new holes in the top former plate and determining if a modification of the screening parameters listed in Table A-1 in MRP-191 for the former plates is needed.

2.3 *Time-Limited Aging Analyses*

Five TLAAs are described in Section 4.1.7 of Reference 4 for the upflow conversion; three TLAAs related to fatigue usage factors and two TLAAs related to ISR. The following subsections describe the methodology used for both types of TLAAs.

2.3.1 TLAAs Related to Fatigue Usage Factors

The transient design cycles for 60 years are assumed to be equal to the design cycles for 40 years. Therefore, a change in fatigue of the reactor vessel internals need not be addressed.

2.3.2 TLAAs Related to Irradiation-Enhanced Stress Relaxation

There are two TLAAs related to ISR for the upflow conversion; one each for the core barrel plugs and the lower former plate plugs. The end-of-life (EOL) interface pressure between the plug and its respective item (i.e., core barrel or lower former plate) is reduced from the beginning-of-life (BOL) results due to short-term and long-term thermal stress relaxation (SR) and ISR. The stress ratio of EOL to BOL ($\sigma_{EOL}/\sigma_{BOL}$) was estimated using updated fluence values for 60 years, and compared to a calculated minimum required stress ratio. This methodology was executed for both the core barrel and lower former plate plugs.

3.0 EVALUATION RESULTS

The following sections describe the results of the evaluations, which were performed using the methodologies described in Section 2.0.

3.1 *Core Barrel and Lower Former Plate Plugs (MRP-227-A Evaluation)*

Screening parameters were selected and documented for the Type 316L core barrel plugs and lower former plate plugs that were used in the upflow modification at McGuire Units 1 and 2 in the 1990s. The screening parameters selected were consistent with MRP-191 Table A-1: material type, grade, or class, effective stress threshold, structural weld presence, wear potential, cumulative usage factor, required preload, and neutron fluence. Additionally, pertinent information regarding the material's cold-work and temperature was collected. Each screening parameter was identified, justified, and compared against the screening criteria contained in Reference [7]. The following age-related degradation mechanisms were considered applicable to both types of plugs, after consideration of the screening criteria: irradiation embrittlement, ISR/IC, fatigue (due to irradiation concerns and cumulative usage factor [CUF] value), and wear (solely for irradiation concerns). Next, a FMECA was performed; ISR/IC, fatigue, and wear were not considered as these age-related degradation mechanisms were addressed in TLAAs under 10 CFR 54.21(c)(1) (i) and (ii) directly or indirectly (see Section 3.3) and are therefore not required to have an AMP. Component failure likelihood and conditional damage likelihood were assessed and a category of "low" was selected for both likelihoods. Finally, justification was made for a preliminary category; the expert panel determined that Category A was appropriate, which screens these components out of further consideration. Therefore no additional inspections are required per MRP-227-A for these plugs.

3.2 *Modified Top Former Plate with Holes (MRP-227-A Evaluation)*

The screening parameters identified in Table A-1 of MRP-191 for the former plates are as follows:

- Material type/grade and material category
- The presence of a structural weld
- Wear potential
- If preload is required for the component item
- Fluence
- Effective stress threshold
- Whether cumulative usage factor exceeds 0.1

In addition, the condition of temperature being greater than or equal to 608°F (320°C) was also researched as this is a screening criterion for void swelling. All seven screening parameters plus temperature were determined to be unaffected by the modification due to the upflow conversion. As none of the screening results were determined to change due to the modification of the top former plates, the results obtained in Table A-1 of MRP-191 remain valid for the modified top former plates, with holes.

3.3 *Time-Limited Aging Analyses*

The results of the three TLAAs related to fatigue usage factors and the two TLAAs related to ISR are discussed in this section.

3.3.1 TLAAs Related to Fatigue Usage Factors

The transient design cycles for 60 years are assumed to be equal to the design cycles for 40 years. Therefore, a change in fatigue of the reactor vessel internals need not be addressed.

3.3.2 TLAAs Related to Irradiation-Enhanced Stress Relaxation

The minimum required stress ratio for the core barrel plugs, based on loads and calculations of required EOL interface pressure and contact pressure, was calculated for 60 years. This minimum required stress ratio value was compared to and determined to be less than the estimated stress ratio for the core barrel plugs, which was determined by considering data from laboratory studies in several technical peer-reviewed papers. Therefore, the core barrel plugs meet the requirements to remain in place for 60 years of plant life.

The minimum required stress ratio for the lower former plate plugs, based on loads and calculations of minimum required test plug spring back and required contact pressure, was calculated for 60 years. This minimum required stress ratio value was compared to and determined to be less than the estimated stress ratio for the lower former plate plugs, which was determined by considering data from laboratory studies in several technical peer-reviewed papers. Therefore, the lower former plate plugs meet the requirements to remain in place for 60 years of plant life.

4.0 CONCLUSIONS

A modification to the McGuire Nuclear Station was performed by Framatome in the 1990s to reduce the potential for baffle gap jet flow-induced fuel damage. The modification consisted of roll expanding plugs into the core barrel holes and a percentage of the lower former plate holes and modifying the top former plate with new holes, which reversed the flow in this region of the RV internals. Since these modifications are plant-specific modifications to the McGuire Nuclear Station, evaluation of the modifications in accordance with the MRP-227 process is needed. Additionally, TLAAs performed for the original license need to be completed for 60 years.

The same process used in MRP-191, consisting of data compilation and review, screening of components, a FMECA, and finally categorization and ranking, was used for the core barrel and lower former plate plugs. This resulted in the expert panel determining that Category A was appropriate for the core barrel and lower former plate plugs, which screens these components out of further consideration. Therefore, no additional inspections are required per MRP-227-A for these plugs.

The process used to evaluate the modified top former plate included reviewing screening parameters associated with the new holes in the top former plate and determining if those changes would result in a modification of the screening parameters listed in Table A-1 in MRP-191 for the modified former plates. The conclusion from this effort was that no changes due to the modification of the top former plates are needed to the screening parameters in Table A-1 of MRP-191.

Five TLAAs were considered for the period of extended operation; three TLAAs related to fatigue usage factors (one each for the modified top former plate (with holes), the core barrel plugs, and the lower former plate plugs) and two TLAAs related to ISR (one each for the core barrel plugs and the lower former plate plugs.)

It is concluded that a change in fatigue of the reactor vessel internals need not be addressed as the transient design cycles for 60 years are assumed to be equal to the design cycles for 40 years.

The minimum required stress ratio for the core barrel and lower former plate plugs was calculated for 60 years. This minimum required stress ratio was based on loads and calculations of EOL interface pressure (core barrel plugs), minimum required test plug spring back pressure (lower former plate plugs), and required contact pressure (core barrel and lower former plate plugs). The minimum required stress ratio values were compared to and determined to be less than the respective estimated stress ratio for both the core barrel and lower former plate plugs. Therefore, the core barrel and lower former plate plugs meet the requirements to remain in place for 60 years of plant life.

The results of these evaluations fulfill regulatory commitment 4, as stated in Attachment 6 of Reference 3, and address the five TLAAs, under 10 CFR 54.21(c)(1) (i) and (ii), associated with the upflow modification.

5.0 REFERENCES

1. NRC Document NUREG-1800, Revision 2, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants – Final Report," December 2010 (ADAMS Accession Number ML103490036).

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

3. Letter MNS-17-050 from Thomas D. Ray, Duke Energy to U. S. Nuclear Regulatory Commission, "Duke Energy Carolinas, LLC (Duke Energy), McGuire Nuclear Station, Units 1 and 2, Docket Nos. 50-369 and 50-370, Review Request for the Aging Management Program and Inspection Plan for the McGuire Nuclear Station Units 1 and 2 Reactor Vessel Internals to Implement MRP-227-A," December 13, 2017, NRC Accession Number ML17356A177.

4. WCAP-18265-NP, Revision 0, Attachment 1 to Letter MNS-17-050, "Aging Management Program and Inspection Plan for the McGuire Nuclear Station Units 1 and 2 Reactor Vessel Internals (Application to Implement MRP-227-A)," December 2017, NRC Accession Number ML17356A178.

5. Code of Federal Regulations, Title 10, Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," Section 21, "Contents of application – technical information."

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

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