Response to Request for Additional Information – ANP-10353P

ANP-10353,
Revision 0
Q1NP, Revision 0

Topical Report

January 2022

(c) 2022 Framatome Inc.

Copyright © 2022

Framatome Inc.
All Rights Reserved

FRAMATOME TRADEMARKS

ARCADIA, ARTEMIS, GALILEO, and S-RELAP5 are trademarks or registered trademarks of Framatome or its affiliates, in the US or other countries.

Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1	All	Initial Issue

Contents

Page

1.0 RAI 1 1-1
2.0 RAI 2 2-1
3.0 RAI 3 3-1
4.0 RAI 4 4-1
5.0 RAI 5 5-1
6.0 RAI 6 6-1
7.0 RAI 7 7-1
8.0 RAI 8 8-1
9.0 REFERENCES 9-1

List of Tables

Table 7-1 Fuel Performance Code Initialized Parameter Comparison for W3 7-3
Table 7-2 Fuel Performance Code Initialized Parameter Comparison for CE 7-4

List of Figures

Figure 1-1 Differences Between Calculated and Measured Keff for All Critical Experiments 1-3

Figure 5-1 [ ] vs Material Depletion for Different Enrichment Values 5-4

Figure 6-1 Detector Sensitivity over Age in Material Depletion [ ] 6-4

Figure 6-2 Influence on the [ ] due to Wide Enrichment Spread 6-5

Nomenclature

Acronym	Definition
ANS	American Nuclear Society
ANSI	American National Standards Institute
BEPU	Best Estimate Plus Uncertainty
CE	Combustion Engineering
EM	Evaluation Model
FPC	Fuel Performance Code
LOCA	Loss of Coolant Accident
NRC	Nuclear Regulatory Commission
PCT	Peak Cladding Temperature
PWR	Pressurized Water Reactor
RAI	Request for Additional Information
RIP	Rod Internal Pressure
(R)LBLOCA	(Realistic) Large Break LOCA
SBLOCA	Small Break LOCA
SER	Safety Evaluation Report
TR	Topical Report
W3	Westinghouse 3-loop

INTRODUCTION

A Nuclear Regulatory Commission (NRC) Request for Additional Information (RAI) related to the Increased Enrichment for PWRs Topical Report (TR) ANP-10353P (Reference 2) was provided in Reference 1. Responses to all the questions in Reference 1 are provided herein. No markups to the ANP-10353P TR are expected due to the RAI responses.

1.0 RAI 1

Question:

The critical benchmark experiments used to justify use at higher enrichments of the ARCADIA code package are nearly all enriched to [ ] U-235. Justify that the selected critical experiments provide sufficient coverage and that the effects and uncertainties of increasing enrichment are well understood between 5 and [ ] U-235.

Response:

Critical experiments are used to support validation of APOLLO2-A for various parameters. [ ] sets of experiments, containing a total of [ ] experiments were modeled in Reference 3. The U-235 weight percent used in the experiments ranged from [ ] Within each set of experiments, additional materials were present and included [ ] An estimated uncertainty of [ ] for the reactivity of [ ] fueled critical experiments was cited in Reference 3, Section 4.1.1. The range of differences between the calculated and measured keff values included in Reference 3 was [ ]

Several additional sets of experiments were included in the Increased Enrichment TR (Reference 2) which extended the maximum U-235 enrichment to [ ] Additional materials within these sets included [ ] The range of differences between the calculated and measured keff values included in Reference 2 was [ ]

[ ] the range of the differences seen in the Reference 3 results for less than 5 wt% U-235 experiments.

Figure 1-1 provides a plot of the difference between the calculated and measured keff values for all experiment included in References 2 and 3. As can be seen in Figure 1-1, [

]

Figure 1-1
Differences Between Calculated and Measured Keff for All Critical Experiments

[

]

2.0 RAI 2

Question:

ANP-10353P, Revision (Rev.) 0, Section 3.1.7, "Depletion," states that the depletion calculations do not change for higher enrichments. However, this section does not discuss any related changes to uncertainties in the depletion calculations above or below 5 wt% U-235. Provide justification that the uncertainties in the depletion calculations above 5 wt% U-235 remain the same, or are bounded by, uncertainties below 5 wt% U-235. If the uncertainties are not equal or less than those below 5 wt% U-235, please provide an estimation of the increase in uncertainty for enrichments between 5 and [ ] U-235.

Response:

Several data sources contribute to the justification of the uncertainty associated with depletion:

[ ]

Additional justification for acceptability of pin power uncertainties with depletion, based on a code-to-code comparison, is discussed in Reference 4 (SER, Section 2.5.2). The NRC concluded that cold criticals were acceptable for depletion uncertainties.

Experimental data for fission rate and depleted fuel isotopic comparisons are not available for greater than 5 wt% U-235. However, the depletion equations used in ARCADIA are not a function of enrichment and the critical experiments presented in Reference 2 provide evidence that the cross section data is adequate for enrichments up to [ ] U-235. [ ]

[ ] The critical experiments included in Reference 2 are presented to show the adequacy of the code and no reduction in uncertainty is requested for enrichments greater than 5 wt% U-235.

Colorset analyses estimates the error of the ARTEMIS dehomogenization with and without burnup compared to APOLLO2-A. Several multi-assembly problems were included in the analysis of local peaking uncertainty:

- 20 colorsets originally included in Reference 4 (see Table 12.2.2-1 of Reference 4),
- 4 colorsets added in Reference 3 (see Table 8-3 of Reference 3), and
- 7 colorsets added in Reference 2 using [ ] U-235 (see Table 3-2 of Reference 2 for a description of the additional colorsets for greater than 5 wt%).

References 2 through 4 presented multi-assembly statistics for the group of colorsets included in the respective reference. Table 3-4 of Reference 2 provided a comparison of these multi-assembly statistics from References 2 through 4. The results developed using the higher enriched colorsets for Reference 2 [ ] the previous results reported in References 3 and 4. Therefore, the colorset results developed in Reference 2 support the conclusion that the uncertainty due to depletion for enrichments up to [ ] U-235 would be bounded by the results from References 3 and 4.

3.0 RAI 3

Question:

In ANP-10353P, Rev. 0, Section 3.3.1, "Local Peaking Uncertainty," it is stated that there would be a minimal effect on enthalpy rise hot channel factors ($F_{\Delta H}$ and F_R) and heat flux hot channel factor, F_Q, as a result of increasing uncertainties. Please quantify "minimal" or provide additional justification that quantification is unnecessary or that any changes to the local peaking factors remain bounded by previously acceptable values.

Response:

Local peaking uncertainty is composed of comparisons between measured and predicted pin fission rates and from the multi-assembly comparisons between APOLLO2-A and ARTEMIS. As stated in Reference 2, Section 3.3.1, fission rate comparisons for pins with greater than 5 wt% U-235 are not available; however, the method and equations used to calculate the pin powers are not a function of enrichment and the uncertainty related to this component would be the same as for fuel with less than 5 wt% U-235.

The colorset analysis discussion from the response to RAI 2 states that the results developed using the higher enriched colorsets for Reference 2 [ ] the previous results reported in References 3 and 4. Uncertainty values for F_Q and $F_{\Delta H}$ were generated in both References 3 and 4 using the results from the colorset calculations. In both references the final calculated uncertainty values were less than the licensing values approved in the topical reports. F_Q and $F_{\Delta H}$ uncertainty values were not calculated using the Reference 2 updated colorset values. Because the colorset values are bounded by the References 3 and 4 colorset values, the F_Q and $F_{\Delta H}$ uncertainty values calculated with the updated colorset results would [ ] the calculated uncertainties reported in References 3 and 4, thus remaining below the approved licensing values used for the criteria on F_Q and $F_{\Delta H}$.

4.0 RAI 4

Question:

ANP-10353P, Rev. 0, Section 3.3.1, "Local Peaking Uncertainty," provides an estimation of changes in the local error standard deviation, but Section 3.3.2, "Inferred Power Distribution Uncertainty," and Section 3.3.3, "Calculated Power Distribution Uncertainty," do not contain a similar analysis. Please provide a discussion that explains how uncertainties related to the inferred power distribution and calculated power distribution are affected by increasing enrichment.

Response:

Inferred power distribution uncertainty is the ability of ARCADIA to accurately predict power in uninstrumented locations using both measured and predicted values. This component of the final uncertainty is dependent on the ability of ARTEMIS to calculate relative power distributions in assembly locations that do not contain a detector, using data from the measured locations. The methodology and equations used by ARTEMIS to calculate these "measured" values in unistrumented locations are not dependent on enrichment. Furthermore, as discussed previously, colorset results presented in Reference 2 indicate that the dehomogenization capability in ARTEMIS is consistent between fuel with less than or equal to 5 wt% U-235 and fuel with up to [ ] U-235.

In addition, the measured data used for reconstructing the power distribution in uninstrumented locations is dependent on the behavior of the detectors. Detector behavior when used with greater than 5 wt% U-235 is discussed in Reference 2, Section 3.3.4, and further discussed as responses to RAI 5 and RAI 6. These discussions show that detectors remain capable of providing [ ] of measurement results for fuel with greater than 5 wt% U-235 fuel as they do with fuel with less than or equal to 5 wt% U-235.

Because the ARTEMIS capability and detector behavior for fuel up to [ ] U-235 are both consistent with current application, the uncertainty for inferred power distribution uncertainty remains applicable as determined in References 3 and 4.

As with the inferred uncertainty, assessment of the calculated power distribution uncertainty requires core measurement values. Since there are no cores that currently operate with greater than 5 wt% U-235 fuel, this data is not available. Power distributions are monitored during core operation, and comparisons between calculated and measured data will be assessed when data is available, consistent with standard reload practices and benchmark requirements.

5.0 RAI 5

Question:

ANP-10353P, Rev. 0, Section 3.3.4, "Detector Sensitivity to Enrichment," states that increasing enrichment will increase the importance of background signals. Please provide an estimation of this increase in importance and discuss how it will be accounted for during operation. Increased enrichment is known to cause spectral hardening. How does spectral hardening affect the functionality of detectors and is this accounted for in the ARCADIA code package?

Response:

The key influence from the spectral hardening to the neutron flux sensitive detectors is manifested in the neutron flux reaction rates. With rhodium detectors as an example, this reaction rate can be expressed as:

$$\int_{emitter} \int_{E} \{ N_{Rh}(\vec{r}) * \sigma_{Rh}(\vec{r}, E) * \Phi(\vec{r}, E) \} d\vec{r} \, dE$$

where,

$N_{Rh}(\vec{r})$ is the rhodium number density at location $\vec{r}$;

$\sigma_{Rh}(\vec{r}, E)$ is the rhodium micro absorption cross section at location $\vec{r}$ for energy E;

$\Phi(\vec{r}, E)$ is the neutron flux at location $\vec{r}$ and with energy E.

Analytically, the generation of this reaction rate is covered by the neutronic design and analysis tools, including the master library, spectrum codes, and core simulator. These design and analysis tools are the key components of the ARCADIA code package and their applicability to the increased enrichment designs is discussed and justified in various sections of Reference 2.

Since the [ ] are not affected by the increased enrichment values, the detector functionality of the hardware is not affected by the spectral hardening.

However, the magnitude of the signals from the rhodium reaction rate will reduce as results from the spectral hardening. On the other hand, the background signals [ ] will not have significant changes [ ] As a result, the 'relative importance' of the background signals in the neutron flux sensitive detectors increases. The same effects occur on the hardware electronic noises.

Though the 'relative importance' of the background signals and noises is still very small for fresh detectors and does not affect the accuracy of the measured results from fresh detectors, its effects will gradually become noticeable with the aging of detectors. This aging will reduce the allowable detector lifetime in term of the [ ] The amount of such reductions depends on many factors, including [ ] Additional discussions on these dependences are given in the response to RAI-6 on setting the [ ] For the current design and system used by Framatome in B&W reactors, the allowable detector lifetime in [ ] will decrease [ ] enriched fuel under consideration.

Nonetheless, due to reductions in the rhodium signals resulting from lower absorption reaction rates with the hardening spectrum, the allowable detector lifetime in terms of [ ] will increase dramatically. This [ ] increase is more than enough to cover [ ] due to the 'relative importance' increase of the

background signals and noises (about [ ] more for all levels of enriched fuel under consideration).

The results from combining these effects ensure that, in practical applications, the current market maximum target lifetime in [ ] will continue to be allowed for higher enriched fuel.

Figure 5-1 provides a graphic presentation, for [ ] enriched fuel, the large margins against the limit on the [ ] which is set according to the descriptions given in the response to RAI-6. The orange curve on this figure clearly indicates that the current market maximum target lifetime in [ ] is not challenging. Furthermore, it indicates that [ ]

Note that the above example with rhodium detectors in B&W reactors is the most challenging 'practical' detector application regarding the aging effects.

Figure 5-1
[ ] vs Material Depletion for Different Enrichment Values

[ ]

On this figure, each curve was plotted up to the 'Current Maximum Market Target Lifetime [ ]

6.0 RAI 6

Question:

ANP-10353P, Rev. 0, Section 3.3.4, "Detector Sensitivity to Enrichment," states that there is a "small, but acceptable, bias in power distributions base on enrichment values" during transition cycles in which there may be a wide range of enrichments. Please provide an estimation of this bias in power distributions.

Response:

In practical applications, Framatome uses monitoring packages that are NRC approved such as the 'Nuclear Application Software Package' (NAS) and the 'Fixed Incore Detector Monitoring System' (FIDMS) for the rhodium detectors in B&W reactors, which are the most challenging 'practical' detector applications regarding the 'aging' effects. The key characteristics of these monitoring packages include:

1. The analytically predicted reaction rates $\int_{emitter} \int_{E} \{ N_{Rh}(\vec{r}) * \sigma_{Rh}(\vec{r}, E) * \Phi(\vec{r}, E) \} d\vec{r}\, dE$ are only performed for new detectors under various possible operating conditions in realistic environments by combining the results from spectrum and core codes to account for both the local and the global variations.
2. The detector depletion effects are extracted from the measured signals obtained from 'aged' detectors to create signals that would be measured with new detectors. This is achieved by using the empirically determined correlation on the relation between the detector relative sensitivity and the detector material depletion.

A simplified neutronic model for the reaction rates and their conversion to signals was developed [

] The results from the simplified neutronic model closely matches the empirical results, as shown in Figure 6-1.

The same model was applied to higher enriched fuels at [ ] The results are also exhibited in Figure 6-1. These results showed that some small biases appear in the curves for higher enriched fuels. [ ] Thus, to be conservative for cores with a wide range of fuel enrichments such as those that may occur during transient cycles [ ] the full spread from the resulting curves will be taken as the extra uncertainty to be included in estimating the accuracy associated with relevant [ ] related parameters.

Figure 6-2 provides a graphic presentation to show that the small bias resulted from having a wide range of fuel enrichments is acceptable because there are still significant margins available to the limit on [ ] after counting not only the additional uncertainties resulted from having a wide range of fuel enrichments but another additional 'Built-in Conservatism' imposed for extra assurances as presented and explained in Figure 6-2.

The limit on the [ ] is derived from the following 'measurement system requirement':

> Measurement system [ ] uncertainty parameter ≤ limiting values set in COLR/TS/FSAR

This 'system limit' sets the limit on the [ ] based on the following detector and methodology related information:

1. Detector related information:

[ ]

2. Methodology related information:

[ ]

With all these complicated dependences, the limit on the [ ] varies with vendors/plants/units. Nonetheless, they all show the same general behavior. Thus, no numerical numbers are given in the x- and y- axes on Figure 6-2 to imply its generality.

Figure 6-1

Detector Sensitivity over Age in Material Depletion [ ]

[ ]

The fuel assembly [ ] fuel rods in it and the curve for it is the lowest one in Figure 6-1. The top curve on Figure 6-1 is for [ ] All other curves on Figure 6-1 from top to bottom are in the order of enrichment values, except for the curve [ ] which goes [ ] all curves [ ]

Figure 6-2
Influence on the [ ] due to Wide Enrichment Spread

[ ]

When developing the processes to set the allowable detector lifetime many years ago, an item ‘Built-in Conservatism’ was imposed at high ‘material depletion’ as extra assurances against the limit on [ ] and caused the green curve on Figure 6-2 to move toward the orange curve location. The magnitude of this extra conservative item can be reduced based on model improvements and experiences obtained over the last couple decades; however, it was kept un-changed since its initial implementation to the Framatome standard processes for determining the allowable detector lifetime.

7.0 RAI 7

Question:

ANP-10353P, Rev. 0, Section 7.1, "SBLOCA (Small Break Loss of Coolant Accident)," states that sensitivity studies, "resulted in small changes to initialized rod parameters." Please identify and quantify the changes in these rod parameters.

Response:

The SBLOCA evaluation model (EM) (References 5 and 6) as supplemented by inclusion of GALILEO as the fuel performance code (FPC) (Reference 7) was reviewed to assess the effect of increasing the fuel rod enrichment up to [ ] U-235 (Reference 2). The review determined that the current SBLOCA EM remains valid for fuel with enrichments greater than 5 wt% U-235. However, one change to the inputs used in the EM was identified. Specifically, the enrichment input to GALILEO should be increased from [ ]

[ ]

Within the SBLOCA EM the fuel rod initial conditions, such as rod internal pressure (RIP), fuel centerline temperature, and pellet average temperature, are calculated by the stand-alone calculation in the FPC. Example variations in the representative rod parameters considering only a change in the UO_2 enrichment specified in GALILEO are provided in Table 7-1 for a Westinghouse 3-loop (W3) plant design and in Table 7-2 for a Combustion Engineering (CE) plant design. In both tables, Analysis "A" represents the GALILEO initialization with [ ] UO_2 enrichment and Analysis "C" represents the GALILEO initialization with [ ] UO_2 enrichment. These results show a maximum increase in RIP of [ ] and a maximum increase in fuel centerline temperature of [ ]

The fuel initial temperature is indicative of the initial stored energy in the fuel rod that must be removed following an SBLOCA. This energy is typically removed early in the transient and well before the peak cladding temperature (PCT) is predicted. Therefore, initial stored energy has little to no effect on the SBLOCA results and the variation in the fuel temperatures due to the change in enrichment will have a negligible effect on the final calculated PCT.

The RIP determines the fuel rod strain and fuel pellet to cladding gap during the transient. Higher RIP values will increase the strain and could affect the final PCT. [ ] Therefore, a change of [ ] will have a negligible effect on the final calculated PCT for higher enrichment applications.

While the change in the initial parameters will have a negligible effect on the SBLOCA transient runs, Framatome has opted to update the guidance for what enrichment to specify in the GALILEO inputs for higher enrichment applications.

Table 7-1
Fuel Performance Code Initialized Parameter Comparison for W3

[ ]

Table 7-2
Fuel Performance Code Initialized Parameter Comparison for CE

[ ]

8.0 RAI 8

Question:

ANP-10353P, Rev. 0, Figure A-9, " [

] appears to show non-conservatisms in the first few seconds of shutdown time. Please discuss these non-conservatisms and provide additional justification that the methodologies used to calculate decay heat remain conservative during the first few seconds after shutdown.

Response:

The dotted lines presented in Figure A-9 show a comparison between the RLBLOCA EM and a Best Estimate Plus Uncertainty (BEPU) model [

] These trends are calculated as **(RLBLOCA EM / BEPU) - 1** for various burnups. This discussion is for fission products only.

- The **RLBLOCA EM** uses [

  ]

- The **BEPU** model uses [

  ]

The differences between these models are discussed below.

[

]

[ ]

9.0 REFERENCES

1. Letter, Ngola Otto (NRC) to Gary Peters (Framatome Inc.), "Request for Additional Information Regarding Framatome Topical Report, ANP-10353P, Revision 0, "Increased Enrichment for PWRs" (EPID°L-2021-TOP-0004)," September 2021.
2. ANP-10353P, Revision 0, "Increased Enrichment for PWRs," January 2021.
3. ANP-10297P-A, Revision 0, Supplement 1P-A, Revision 1, "The ARCADIA® Reactor Analysis System for PWRs Methodology Description and Benchmarking Results," December 2020.
4. ANP-10297P-A, Revision 0, "The ARCADIA® Reactor Analysis System for PWRs Methodology Description and Benchmarking Results," February 2013.
5. EMF-2328(P)(A), Revision 0, "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based," March 2001.
6. EMF-2328(P)(A), Revision 0, Supplement 1(P)(A), Revision 0, "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based," December 2016.
7. ANP-10349P-A, Revision 0, "GALILEO Implementation in LOCA Methods," November 2021.