ENCLOSURE 2

MFN 15-060

NEDO-33257 Supplement 1, "The PRIME Model for Analysis of Fuel Rod Thermal-Mechanical Performance 2015 5-Year Update"

Non-Proprietary Information - Class I (Public)

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This is a non-proprietary version of Enclosure 1, from which has the proprietary information has been removed. Portions of the enclosure that have been removed are indicated by an open and closed bracket as shown here [[ ]].

Global Nuclear Fuel

Global Nuclear Fuel
A Joint Venture of GE,Toshiba, & Hitachi

NEDO-33257 Supplement 1
Revision 0
August 2015

Non-Proprietary Information – Class I (Public)

THE PRIME MODEL FOR ANALYSIS OF FUEL ROD THERMAL-MECHANICAL PERFORMANCE

2015 5-YEAR UPDATE

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ACRONYMS

Term	Definition
BWR	Boiling Water Reactor
FGR	Fission Gas Release
Gd	Gadolinia
GE	General Electric
GENUSA	GNF / ENUSA Nuclear Fuel, S. A.
GNF-A / GNF	Global Nuclear Fuel – Americas, LLC
HBWR	Halden Boiling Heavy Water Reactor
IFA	Instrumented Fuel Assembly
KKL	Kernkraftwerk Leibstadt
L&C	Limitation and Condition
LOCA	Loss-of-Coolant Accident
LTR	Licensing Topical Report
LUA	Lead Use Assembly
LWR	Light Water Reactor
NRC	Nuclear Regulatory Commission
OL1	TVO Olkiluoto Unit 1
PCMI	Pellet-Cladding Mechanical Interaction
PWR	Pressurized Water Reactor
RAI	Request for Additional Information
SCIP	Studsvik Cladding Integrity Project
SE	Safety Evaluation
TVO	Teollisuuden Voima Oyj
WSTOL	Worst Tolerance

1. INTRODUCTION

The PRIME fuel performance model was developed to provide best-estimate predictions of the thermal and mechanical performance of Light Water Reactor (LWR) nuclear fuel rods experiencing variable power histories. The PRIME models, qualification, and application methodology were submitted to the Nuclear Regulatory Commission (NRC) for review in 2007 in the form of three separate Licensing Topical Reports (LTRs) covering each of these respective topics. Based upon review of these LTRs and subsequent responses to NRC Requests for Additional Information (RAIs), the NRC reviewed and approved PRIME for licensing applications in 2010 within the range of applicability specified in, and under the Limitations and Conditions (L&Cs) delineated in, the LTRs and in the NRC Safety Evaluation (SE) (Reference 1).

PRIME SE L&C 4 requires GNF to provide an update to the NRC every five years to confirm the continued applicability of PRIME. GNF has augmented the original PRIME qualification database detailed in NEDC-33257P-A (Reference 1) to include data that has become available to GNF since approval of PRIME in 2010 and has assessed the effect of the added data on PRIME predictive capability. The data was selected by surveying data from industry programs and data from recent GNF programs. Updated qualification results are presented in Section 2 for the fuel rod performance parameters used in PRIME qualification for which new data was added, namely Temperature (Section 2.1), Deformation (Section 2.2), Fission Gas Release (FGR) (Section 2.3), and Rod Internal Pressure (Section 2.4). These sections contain summaries of the new data, including sources of the data, and figures illustrating comparisons of predicted and measured results for the original and expanded qualification data. In these figures, the original and expanded data are differentiated by the use of different symbols. Complete tables of the qualification data, including the original data and the additional data, are included in Appendix A. PRIME material property models for which new data are available since approval of PRIME in 2010 include the cladding irradiation growth model. New data for cladding irradiation growth is presented in and compared to the PRIME irradiated growth model in Section 2.5. Assessment of the PRIME model uncertainty for the expanded qualification data is given in Section 3. The intent of, and specific requirements of, L&C 4 as stated in the SE are included below, together with a *summary* of the response to each requirement included in this report.

<u>SE L&C 4</u>

PRIME models have been calibrated and validated by direct comparison to the existing empirical database. Further, model uncertainties described within the application methodology were derived by direct comparison of model predictions to the existing empirical database. To ensure PRIME's best-estimate predictions and applied uncertainties remain valid, GNF must demonstrate and document, in a letter addressed to the Director, Division of Safety Systems, Office of Nuclear Reactor Regulation, the continued applicability of PRIME every five years starting in 2015.

a. In preparation of this letter, GNF must review available sources for applicable commercial and research reactor fuel performance data which may augment the existing

PRIME qualification database (e.g., international research activities, pool-side examinations, hot-cell programs, power ramp programs).

GNF reviewed and evaluated applicable data available from commercial and research reactors and international fuel performance research programs and augmented the original PRIME experimental qualification database by adding fuel temperature, cladding diametral strain, FGR, and rod internal pressure data.

b. In the letter, sources for new data should be clearly identified. If no new data for a particular model (e.g., FGR model) has been discovered, the letter should state this fact and identify which sources were investigated.

GNF reviewed and evaluated applicable data available from commercial and research reactors and international fuel performance research programs. The data added had to be judged reliable, final and adequately documented to permit analysis with PRIME. The data added to the PRIME experimental qualification database and new material property data are discussed in Section 2. A summary of PRIME qualification data, including the original qualification data and the additional data included in this 5-year update is also provided in Appendix A.

c. PRIME model predictions and uncertainties should be compared against the augmented database. New data should be easily differentiated on the plots. At a minimum, the letter should separately address the following model predictions and their respective uncertainties: (1) fuel temperature, (2) FGR, (3) fuel irradiation swelling, (4) cladding creep, (5) cladding strain (due to over power conditions), and (6) void volume/rod internal pressure.

Comparisons of PRIME predictions to the expanded PRIME qualification data for fuel temperature, cladding diametral strain, fission gas release and rod internal pressure are included in Sections 2.1 - 2.4 (nominal) and Section 3 (perturbed or worst tolerance). In all cases the original and added data are differentiated by the use of different colored symbols.

d. Any data discarded from the augmented qualification database should be identified and dispositioned.

Data evaluated and judged not to meet the criteria above, and thus not included in the expanded PRIME qualification, are summarized and dispositioned in Appendix B.

e. The letter should identify and disposition any bias on model predictions or increase in uncertainty.

The results presented in Section 2 confirm the PRIME best estimate predictive capability for the expanded qualification data. The statistical perturbation and worst tolerance results presented in Section 3 confirm that the existing PRIME statistical, including PRIME model uncertainty, and worst tolerance licensing methodologies are applicable to the expanded qualification data.

f. Since the worst case methodology employed in the [[

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Worst tolerance analyses are performed by perturbing pertinent parameters to their worst-case (bounding) values. The worst tolerance [[ ]] *are specified for each fuel rod consistent with the approved PRIME methodology. GNF has not changed the manufacturing tolerances applied in the worst tolerance analysis.*

Additionally, PRIME SE L&C 3 describes a process by which computer code revisions necessary to correct errors in the source code and return the algorithms to those described in the PRIME LTR may be made. GNF has revised the PRIME code since its approval in accordance with L&C 3. The revisions have been performed under GNF's software maintenance procedures, which comply with industry standards in accordance with GNF's NRC-approved 10 CFR 50 Appendix B quality assurance program (NEDO-11209-A, Reference 2). The code revisions are detailed in Appendix C. The results presented in this update were obtained using the current (corrected) code revision. The results confirm that the current code revision remains adequate relative to the NRC approved methodology, including the constraints, limitations, and uncertainty associated with the use of PRIME methods.

2. ADDITIONAL DATA AND UPDATED QUALIFICATION RESULTS

GNF has augmented the original PRIME qualification data detailed in NEDC-33257P-A (Reference 1) to include data that has become available to GNF since approval of PRIME in 2010. This includes new data not included in the original data used for code development and qualification in 2010. GNF has assessed the effect of the new data on PRIME predictive capability. The data was selected by surveying data from industry programs and recent GNF program data. Results are presented below for the fuel rod performance parameters used in PRIME qualification for which new data was added, namely Temperature (Section 2.1), Deformation (Section 2.2), Fission Gas Release (Section 2.3), and Rod Internal Pressure (Section 2.4). These sections contain summaries of the new data, including the sources of the data, and figures with comparisons of predicted and measured results for the original and expanded qualification data. In these figures data from the original qualification and the data added to the original qualification are differentiated by the use of black symbols for the original qualification data and blue symbols for the new data. The new data is also included in the summary of PRIME qualification data in Appendix A. PRIME material property models for which new data are available since the approval of PRIME in 2010 includes the cladding irradiation growth model. New irradiation growth data is presented in Section 2.5. A comparison of the PRIME cladding irradiation growth model to new data is also presented in Section 2.5.

Not all data identified for potential inclusion in this update were included. Only data judged reliable, final and adequately documented to permit analysis with PRIME were included in the sections below and only results for this data are considered in the assessment of the effect of the new data on PRIME predictive capability and application methodology. Data not meeting these criteria are not included in the results below. These data, and the bases for not including in the update, are summarized in Appendix B.

2.1. FUEL TEMPERATURE

The in-reactor centerline temperature of a fuel rod can be inferred from a temperature indicator, such as the measured radius of a fuel microstructural change (i.e., fuel melting, onset of columnar or equiaxed grain growth) to which a temperature can be assigned, or measured directly by insertion of a thermocouple in the fuel column. Although the fuel melting temperature is known with adequate certainty, this type of data is limited in application, as it can be obtained only from high-power operation, generally beyond the expected operating powers and temperatures for commercial fuel rods. The same problem exists, to a lesser extent, with temperature data inferred from the onset of columnar and equiaxed grain growth, with the added complexity of large uncertainties in assigning a grain growth temperature and irradiation time when the grain growth radius was set. Therefore, to minimize the variability in the experimental data, and to employ data over the entire power range of interest, thermal qualification is

performed by comparison to data obtained by direct in-reactor measurement of fuel temperatures by thermocouple or expansion thermometer[1].

The original PRIME qualification database is detailed in NEDC-33257P-A. The experimental data added to the original PRIME qualification data is summarized in Section 2.1.1. All data was obtained from tests in the Halden Boiling Heavy Water Reactor (HBWR). Comparisons of predicted and measured temperatures for the expanded (original plus added) qualification data are presented in Section 2.1.2.

2.1.1. Measured Data

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[1] An expansion thermometer utilizes the thermal expansion of a metal rod which is located at the centerline of the fuel column and which extends along the full length of the fuel column. Expansion thermometers eliminate the uncertainty arising from transmutation-induced decalibration of conventional thermocouples at high exposure, even at temperatures of interest for normal BWR operation.

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2.1.2. Comparisons of Predicted and Measured Temperatures

An overall comparison of predicted and measured fuel centerline temperature for the expanded data is presented in Figure 2-1. The ratios of predicted/measured fuel centerline temperature as a function of nodal exposure for the expanded data are presented in Figure 2-2. In these figures data from the original qualification and the data added to the original qualification are differentiated by the use of black symbols for the original qualification data and blue symbols for the new data.

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Figure 2-1 Predicted versus Measured Fuel Temperature (Nominal)

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Figure 2-2 Predicted/Measured Fuel Temperature versus Exposure (Nominal)

2.2. CLADDING DEFORMATION

Cladding deformations (strains) consist of elastic, thermal and permanent (plastic plus creep and irradiation growth) components. The cladding diametral and axial thermal strains can be accurately calculated as functions of temperature. The diametral and axial permanent strains are typically determined by comparing pre- and post-irradiation measurements of cladding diameter and length and adjusting for temperature differences (if any) in the measurement conditions. If pre-irradiation characterizations of diameter and length are not available, nominal tubing fabrication specifications can be used, although this introduces uncertainty in addition to that inherent in the various measurement techniques. Alternatively, an approximation of the as-fabricated cladding diameter can be obtained from the measured cladding diameter in the fission gas plenum region of the rod. Additionally, measurements at different exposures may be used to infer incremental changes without introducing the uncertainty associated with lack of pre-irradiation characterizations. In some cases, cladding diametral and axial strains are measured during operation in instrumented test assemblies in test reactors such as the HBWR. These measurements yield total strains and can be used to compare predicted and measured elastic strains provided the measurements are adjusted for the other strain components. Measurements performed at different power levels may be used to infer the effect of incremental power changes on cladding strain response.

The original PRIME qualification database is detailed in NEDC-33257P-A. The experimental data added to the original PRIME qualification data is summarized in Section 2.2.1. Comparisons of predicted and measured cladding strains for the expanded data are presented in Section 2.2.2.

2.2.1. Measured Data

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2.2.2. Comparisons of Predicted and Measured Cladding Deformations

An overall comparison of predicted and measured cladding permanent diametral strains (%ΔD/D) for the expanded data is presented in Figure 2-3. The figure includes data from rapid power ramps. As discussed above, the added data was obtained from high power ramp tests at high exposure. The differences between predicted and measured cladding diametral strains for the same data as a function of nodal exposure is presented in Figure 2-4. In these figures data from the original qualification and the data added to the original qualification are differentiated by the use of black symbols for the original qualification data and blue symbols for the new data.

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Figure 2-3 Predicted versus Measured Permanent Cladding Diametral Strain for Rapid Power Ramps (Nominal)

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Figure 2-4 Predicted Minus Measured Permanent Cladding Diametral Strain versus Exposure for Rapid Power Ramps (Nominal)

2.3. FISSION GAS RELEASE

The single most important prerequisite of experimental data considered for use in qualification of the FGR predictive capability is that the power-exposure history be well-characterized. This prerequisite exists because of the high sensitivity of the FGR model to fuel temperature and exposure. Therefore, of the FGR data currently available, the qualification database has been selected on the basis of availability of well-characterized power histories. Additionally, GNF has historically used only fission gas data obtained by fuel rod puncturing for model calibration and qualification. However, GNF has recently verified the accuracy of fission gas measurements performed by in-pool 85Kr gamma scanning of the fuel rod plenum and is now using fission gas data obtained by this technique, in addition to puncture data, for model calibration and qualification. Use of in-pool gamma scanning has the additional benefit that it permits repeated measurements at increasing exposure on a single rod. GNF has recently performed measurements of this type.

The original PRIME qualification database is detailed in NEDC-33257P-A. The experimental data added to the original PRIME qualification data is summarized in Section 2.3.1. Comparisons of predicted and measured FGRs for the expanded data are presented in Section 2.3.2.

2.3.1. Measured Data

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2.3.2. Comparisons of Predicted and Measured Fission Gas Release

An overall comparison of predicted and measured FGR for the expanded data is presented in Figure 2-5. The differences between predicted and measured FGR as a function of rod average exposure for cases in which the measured release is less than 5% is presented in Figure 2-6. The ratios of predicted/measured FGR as a function for rod average exposure for cases in which the measured release is greater than 5% is presented in Figure 2-7. In these figures data from the original qualification and the data added to the original qualification are differentiated by the use of black symbols for the original qualification data and blue symbols for the new data.

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Figure 2-5 Predicted versus Measured Fission Gas Release (Nominal)

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Figure 2-6 Predicted Minus Measured Fission Gas Release versus Exposure (Nominal) Measured Release < 5%

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Figure 2-7 Predicted/Measured Fission Gas Release versus Exposure (Nominal) Measured Release ≥ 5%

2.4. FUEL ROD INTERNAL PRESSURE

The fuel rod internal pressure is determined by: (1) the as-fabricated fuel rod characteristics such as the as-fabricated void volume and fill gas pressure; (2) the release of gaseous fission products from the fuel pellets to the fuel rod void volume during irradiation; (3) changes in the fuel rod geometry during irradiation; and (4) the temperature of the gas occupying the fuel rod void volume. The as-fabricated fuel rod characteristics are known from the specified fabrication tolerances or available pre-characterization. The PRIME FGR predictive capability is demonstrated in NEDC-33257P-A and Section 2.3 of this report. The predicted changes in the fuel rod geometry are based upon separate effects measurements of the fuel and cladding deformation mechanisms and the integral fuel rod deformation predictive capability for the expanded cladding deformation qualification data as demonstrated in NEDC-33257P-A and Section 2.4 of this report. The purpose of the experimental qualification presented in this section is primarily to address the remaining item regarding the effect of gas temperature due to fuel rod heat-up on rod internal pressure. This pressure is of significance relative to the fuel rod cladding liftoff calculation and to downstream analyses of Loss-of-Coolant Accident (LOCA) response and dry storage of spent fuel.

The fuel rod void volume is comprised of the fuel-cladding diametral gap, radial and transverse pellet cracks, pellet dishes, pellet center hole (hollow pellets), fuel rod plenum, and additional free volume due to pellet chamfers and fuel column loading (stacking) effects. [[

]] The data used to assess PRIME predictive capability for hot operating fuel rod internal pressure is primarily from on-line measurements with fuel rod pressure transducers.

The original PRIME qualification database is detailed in NEDC-33257P-A. The experimental data added to the original qualification data is summarized in Section 2.4.1. All data was obtained from tests in the HBWR. Comparisons of predicted and measured rod internal pressures for the expanded data are presented in Section 2.4.2.

2.4.1. Measured Data

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2.4.2. Comparisons of Predicted and Measured Rod Internal Pressure

An overall comparison of predicted and measured rod internal pressure for the expanded data is presented in Figure 2-8. The differences between predicted and measured rod internal pressures for the same data as a function of rod average exposure are presented in Figure 2-9. In these figures data from the original qualification and the data added to the original qualification are differentiated by the use of black symbols for the original qualification data and blue symbols for the new data.

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Figure 2-8 Predicted versus Measured Fuel Rod Internal Pressure (Nominal)

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Figure 2-9 Predicted Minus Measured Fuel Rod Internal Pressure versus Exposure (Nominal)

2.5. CLADDING IRRADIATION GROWTH

Fuel rod axial growth occurs due to the combined effects of cladding irradiation growth and stress induced creep. PRIME includes irradiation growth in the calculation of the (unfueled) plenum growth, which affects fuel rod void volume and thus rod internal pressure, and includes both irradiation growth and stress induced axial creep in the calculation of overall fuel rod axial growth. [[

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Cladding irradiation growth data previously presented to the NRC are documented in responses to PRIME RAIs (Reference 1). Additional data are summarized in Section 2.5.1. Comparisons of measured data to predictions based on the GNF irradiation growth model included in PRIME are presented in Section 2.5.2.

2.5.1. Measured Data

Zr-2 RX BOR-60 - GNF has continued to expand its irradiation growth database by inclusion of new high fluence growth data from alloy samples irradiated up to very high fluence levels in the Russian BOR-60 materials test reactor as part of an industry research program.

2.5.2. Comparisons of Predicted and Measured Cladding Irradiation Growth

The expanded database is shown in Figure 2-10. In this figure, data previously provided to the NRC are denoted by the black symbols. New data are differentiated by the use of blue symbols. The new data is for recrystallized Zircaloy-2. Only Zircaloy-2 data from the original data previously provided to the NRC in responses to PRIME RAIs is included in Figure 2-10. The new data expands the database for fluence, including fluence above [[

]] and indicates that the PRIME irradiation growth model predicts growth well on a best estimate basis to approximately [[

]]. Note that the fluence of the new data from the BOR-60 materials test reactor and the PWR data have been corrected for the BWR neutron energy distribution (Reference 3).

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Figure 2-10 PRIME Irradiation Growth Model versus Measured Irradiation Growth of Zircaloy

3. MODEL UNCERTAINTY

3.1. MODEL PERTURBATION FOR STATISTICAL ANALYSES

The PRIME code is utilized in procedures described in NEDC-33258P-A (Reference 1), to assure compliance with specified fuel rod licensing thermal-mechanical licensing criteria, also described in NEDC-33258P-A. In all cases that involve statistical analysis, the perturbation procedure summarized in NEDC-33258P-A is utilized to assure [[ ]] compliance of calculated parameters with corresponding criteria. Thus, for the PRIME statistical application methodology, the adequacy of the PRIME model uncertainty for the expanded data (original data plus added data described in Section 2) must be confirmed. The PRIME model uncertainty is determined in NEDC-33258P-A to be [[ ]] plus a FGR model perturbation. The PRIME [[ ]] model perturbation is accomplished by applying a [[ ]] power perturbation and the FGR model perturbation. The derivation and confirmation of this perturbation assures that it explicitly addresses uncertainties in the measured power histories and operating conditions, uncertainties in fuel and cladding fabrication parameters and uncertainties in data collection techniques for data in the PRIME experimental qualification data.

The effect of applying the (existing) PRIME [[ ]] model perturbation on the predicted temperatures for the expanded (original plus added) qualification fuel temperature, FGR, and rod internal pressure data is shown in Figure 3-1 through Figure 3-7. As in Section 2, in these figures data from the original qualification and the data added to the original qualification are differentiated by the use of black symbols for the original qualification data and blue symbols for the new data. These results confirm that the currently approved PRIME [[ ]] model perturbation, which consists of a [[ ]] and a FGR model perturbation, adequately characterizes the PRIME model perturbation for fuel temperature, FGR, and rod internal pressure for the expanded PRIME qualification data by confirming that the [[ ]] model perturbation results in overprediction of approximately [[ ]] of the qualification data.

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Figure 3-1 Predicted versus Measured Fuel Temperature ([[ ]] Model Perturbation)

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Figure 3-2 Predicted/Measured Fuel Temperature versus Exposure ([[ ]] Model Perturbation)

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Figure 3-3 Predicted versus Measured Fission Gas Release ([[ ]] Model Perturbation)

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Figure 3-4 Predicted Minus Measured Fission Gas Release versus Exposure ([[ ]] Model Perturbation) Measured Release < 5%

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Figure 3-5 Predicted/Measured Fission Gas Release versus Exposure ([[ ]] Model Perturbation) Measured Release ≥ 5%

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Figure 3-6 Predicted versus Measured Fuel Rod Internal Pressure ([[ ]] Model Perturbation)

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Figure 3-7 Predicted Minus Measured Fuel Rod Internal Pressure versus Exposure ([[ ]] Model Perturbation)

3.2. WORST TOLERANCE METHODOLOGY

For cladding strain licensing analyses, the worst tolerance methodology summarized in NEDC-33258P-A is utilized to assure compliance of calculated cladding strains with the specified strain limit. [[

]] The effect of applying the Worst Tolerance (WSTOL) methodology on the predicted cladding strains for the expanded cladding deformation data is shown in Figure 3-8 and Figure 3-9. As in Section 2, in these figures data from the original qualification and the data added to the original qualification are differentiated by the use of black symbols for the original qualification data and blue symbols for the new data.

It is noted that the original PRIME cladding deformation data included diameter changes for both steady-state irradiation and power ramp operation. The added data included only power ramp operation, so the original steady-state data is not included in Figure 3-8 and Figure 3-9. The results of the worst tolerance analyses confirm that the worst tolerance methodology results in overprediction of cladding diametral strain for ramp test data, and meets the intent of the methodology to provide conservative cladding strains for PRIME licensing analyses.

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Figure 3-8 Predicted versus Measured Permanent Cladding Diametral Strain for Rapid Power Ramps (WSTOL)

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Figure 3-9 Predicted Minus Measured Permanent Cladding Diametral Strain versus Exposure for Ramps (WSTOL)

4. SUMMARY/CONCLUSION

PRIME SE L&C 4 identifies specific requirements for this five year update. GNF compliance with L&C 4 is summarized as follows.

The PRIME LTR experimental qualification data include fuel temperature, cladding diametral and axial deformation, FGR, rod internal pressure, helium release, and pellet grain growth. For this update, GNF reviewed and evaluated applicable data available from commercial and research reactors and international fuel performance research programs and augmented the original PRIME experimental qualification by adding fuel temperature, cladding diametral strain, FGR, and rod internal pressure data. The data added had to judged reliable, final and adequately documented to permit analysis with PRIME. The data added is summarized in Sections 2.1 - 2.4 and Appendix A. Data evaluated and judged not to meet the criteria above, and thus not included in the expanded PRIME qualification, are summarized and dispositioned in Appendix B. [[

]] Also, GNF considers that the good prediction of the cladding diameter strain for the SCIP high exposure ramp test data, together with the good prediction of the high exposure temperature data, confirms the continued applicability of both the PRIME irradiation swelling and cladding creep models. GNF has expanded the cladding irradiation growth data base by inclusion of new high fluence growth data as discussed in Section 2.5.

Comparisons of PRIME predictions to the expanded PRIME qualification data are included in Section 2 (nominal) and Section 3 (perturbed or worst tolerance). In all cases, data from the original qualification and the data added to the original qualification are differentiated by the use of different colored symbols. The PRIME fuel rod thermal-mechanical performance model has been developed as a best-estimate predictor of fuel performance. Verification of the best-estimate prediction capability is provided by the extensive experimental qualification in NEDC-33257P-A (Reference 1) and is supplemented by this report in Section 2. As discussed in Section 3, the PRIME predictions for the expanded qualification data confirm that the PRIME statistical model perturbation predictions or WSTOL predictions confirm that the statistical and WSTOL licensing methodology calculations remain applicable.

Based upon this summary, GNF concludes that this update complies with the requirements of SE L&C 4. The results and conclusion in this update confirm that the NRC-approved PRIME steady-state models and methodology, including the constraints and limitations associated with the use of the methodology, remain valid.

5. REFERENCES

1. Global Nuclear Fuel, "The PRIME Model for Analysis of Fuel Rod Thermal – Mechanical Performance," Technical Bases - NEDC-33256P-A, Qualification – NEDC-33257P-A, and Application Methodology - NEDC-33258P-A, Revision 1, September 2010.
2. GE Hitachi Nuclear Energy, "GE Hitachi Nuclear Energy Quality Assurance Program Description," NEDO-11209-A, Revision 12, February 11, 2015.
3. EPRI Report 1019098, "NFIR Dimensional Stability Project: A Method for Transposing Test Reactor Irradiation Data for PWR and BWR Applications," October 2009.

APPENDIX A SUMMARY OF QUALIFICATION DATA

This appendix contains a summary of PRIME qualification data, including the original qualification data in NEDC-33257P-A (Reference 1) (includes data added in responses to PRIME RAIs) and the additional data included in this 5-year update.

Fuel Centerline Temperatures

PRIME Original Qualification Data

Identification	Comments	# of Rods	Exposure at Thermocouple GWd/MTU
HRP_IFA21		3	5.0
HRP_IFA116		1	4.5
HRP_IFA117		3	9.5
HRP_IFA410		2	14.2
HRP_IFA411		1	31.3
[[			]]
HRP_IFA429		2	22.4
HRP_IFA431		4	5.5
HRP_IFA432		5	39.5
[[			
			]]
HRP_IFA562	Small diameter; Halden test reactor; expansion thermometer	4	85.4
RISO3	• Five General Electric (GE) rods: 8x8 BWR; base irradiated in commercial BWRs • Three ANF rods: base irradiated in a commercial PWR • Two Riso rods: base irradiated in Halden test reactor • All rods: ramp tested in DR3 test reactor	10	50.1
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Identification	Comments	# of Rods	Exposure at Thermocouple GWd/MTU
			]]

Additional Qualification Data

Identification	Comments	# of Rods	Exposure at Thermocouple GWd/MTU
[[			
			]]

Note:

[[

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Cladding Diametral Deformations

PRIME Original Qualification Data

Identification	Comments	# of Rods	Rod Average Exposure GWd/MTU
[[			
			]]
RISO3	• Four GE rods: 8x8 BWR; base irradiated in commercial BWRs • One Riso rod: base irradiated in Halden test reactor • All rods: ramp tested in DR3 test reactor	5	45.8

Additional Qualification Data

Identification	Comments	# of Rods	Rod Average Exposure GWd/MTU
[[			]]

Fission Gas Release

PRIME Original Qualification Data

Identification	Comments	# of Rods	Rod Average Exposure GWd/MTU
[[			

Identification	Comments	# of Rods	Rod Average Exposure GWd/MTU

Identification	Comments	# of Rods	Rod Average Exposure GWd/MTU
			]]
IFA-432	Halden test reactor; IFA-432	1	42.07
[[			
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RISO3	• Four GE rods: 8x8 BWR; base irradiated in commercial BWRs • One Riso rod: base irradiated in Halden test reactor • All rods: ramp tested in DR3 test reactor	5	45.8

Additional Qualification Data

Identification	Comments	# of Rods	Exposure at Thermocouple GWd/MTU
[[			

Identification	Comments	# of Rods	Exposure at Thermocouple GWd/MTU
			]]

Rod Internal Pressure

PRIME Original Qualification Data

Identification	Comments	# of Rods	Rod Average Exposure GWd/MTU
HRP_IFA116		2	4.6
HRP_IFA117		4	9.0
HRP_IFA429		3	20.5
[[			]]
HRP_IFA562	small diameter; Halden test reactor	2	58.1
RISO3	• Three GE rods: 8x8 BWR; base irradiated in commercial BWRs • Four ANF rods: base irradiated in a commercial PWR • Three Riso rod: base irradiated in Halden test reactor • All rods: ramp tested in DR3 test reactor	10	48.6

Additional Qualification Data

Identification	Comments	# of Rods	Exposure at Thermocouple GWd/MTU
[[			
			]]

APPENDIX B DISPOSITION OF DATA NOT INCLUDED IN UPDATE

Not all data identified for potential inclusion in this update were included. Only data judged reliable, final and adequately documented to permit analysis with PRIME were included and used in the assessment of PRIME adequacy. Data not meeting the criteria above are summarized below.

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APPENDIX C PRIME CODE REVISIONS

Per SE L&C 3, computer code revisions necessitated by errors discovered in the source code and needed to return the algorithms to those described in NEDC-33256P-A (Reference 1) are acceptable. PRIME code revisions to correct errors in the source code and return the algorithms to those described in NEDC-33256P-A are summarized below:

Error	**Description and Correction**
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The revisions have been performed under GNF's software maintenance procedures, which comply with industry standards in accordance with GNF's NRC-approved Quality Assurance program (NEDO-11209-A, Reference 2). The results presented in this update were obtained using the current (corrected) code revision. The results confirm that the current code revision remains adequate relative to the NRC approved methodology, including the constraints, limitations, and uncertainty associated with the use of the methods.