FRAPCON-3: Integral Assessment

Manuscript Completed: November 1997
Date Published: December 1997

Prepared by
D. D. Lanning, C. E. Beyer/PNNL
G. A. Berna/GABC

Pacific Northwest National Laboratory
Richland, WA 99352

Subcontractor:
Gary A. Berna Consulting
2060 Sequoia Drive
Idaho Falls, ID 83404-6131

H. Scott, NRC Project Manager

Prepared for
Division of Systems Technology
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
NRC Job Code W6200

MASTER

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

NUREG/CR-6534, Vol. 3 has been reproduced from the best available copy.

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

**Portions of this document may be illegible
electronic image products. Images are
produced from the best available original
document.**

Abstract

An integral assessment has been performed for the U.S. Nuclear Regulatory Commission by Pacific Northwest National Laboratory to quantify the predictive capabilities of FRAPCON-3, a steady-state fuel behavior code designed to analyze fuel behavior from beginning-of-life to burnup levels of 65 GWd/MTU. FRAPCON-3 code calculations are shown to compare satisfactorily to a pre-selected set of experimental data with steady-state operating conditions.

Contents

Abstract	iii
Executive Summary	xi
Acknowledgments	xv
1.0 Introduction	1.1
2.0 Assessment Data Description	2.1
2.1 Description of the Steady-State Cases	2.1
2.2 Description of the Power-Ramp Cases	2.4
3.0 Thermal Behavior Assessment	3.1
3.1 BOL Fuel Center Temperature Predictions	3.1
3.1.1 Effect of Gap Size	3.2
3.1.2 Effect of Fill Gas Mixture	3.5
3.2 Assessment of Temperature Predictions as a Function of Burnup	3.6
3.2.1 Fuel Thermal Conductivity Degradation from Halden Experimental Rods	3.7
3.2.2 Overall Comparison of Temperature Predictions Versus Fuel Burnup of Halden Rods	3.12
4.0 Fission Gas Release Assessment	4.1
4.1 Assessment of Steady-State FGR Predictions	4.1
4.2 Assessment of Transient FGR Predictions	4.3
5.0 Internal Rod Void Volume Assessment	5.1
5.1 Fuel Rod Void Volume	5.1
5.2 Fuel Swelling	5.2
6.0 Cladding Corrosion and Hydriding Assessment	6.1
6.1 Cladding Oxidation and Hydrogen Uptake	6.1

7.0 Cladding Creep and Axial Growth	7.1
7.1 Cladding Axial Growth	7.1
7.2 Cladding Creepdown	7.2
8.0 Comparison to Independent Data for Fuel Temperature and Fission Gas Release	8.1
8.1 Description of the Independent Data Sets	8.1
8.2 Results of Code-Data Comparisons	8.4
8.2.1 BOL Fuel Temperatures	8.4
8.2.2 Fuel Temperatures at Nominal-to-High Burnup	8.4
8.2.3 FGR at Nominal-to-High Burnup	8.5
9.0 Conclusions	9.1
10.0 References	10.1
Appendix A - Supplement	A1.1
A.1 Description of FRAPCON Case Input for Halden Ultra High Burnup, Rod 18	A1.1
A.2 Description of FRAPCON-3 Case Input for Halden Rods 1, 2, and 3	A2.1
A.3 Description of FRAPCON-3 Case Input for IFA-513 Rods 1 and 6	A3.1
A.4 Description of FRAPCON Case Input for IFA-429, Rod DH Test Case	A4.1
A.5 BR-3 Test Cases	A5.1
A.6 Description of FRAPCON-3 Case Input for BNFL/HBEP Br-3 Test Rod DE	A6.1
A.7 Description of FRAPCON-3 Case Input for NRX PWR Rod LFF	A7.1
A.8 Description of FRAPCON-3 Case Input for NRX Rod CBP	A8.1
A.9 EL-4 Test Cases	A9.1
A.10 Description of FRAPCON-3 Case Input for Arkansas Nuclear-2, Rod TSQ002	A10.1
A.11 Description of FRAPCON-3 Case Input for Oconee 5-Cycle PWR, Rod 15309	A11.1
A.12 Monticello BWR Rod, Corner Position A1	A12.1

A.13 TVO, H8/36-6 Test Case	A13.1
A.14 Description of FRAPCON-3 Case Input for Ramped HBEP Obrigheim/Petten Rodlets D200	A14.1
A.15 Obrigheim/Petten-PK Test Cases for the Super-Ramp Rodlets PK6-2, PK6-3, and PK6-S	A15.1
A.16 Studsvik/Inter-Ramp Test Cases	A16.1
A.17 Ramped Halden/DR-2 Test, Rods F7-3, F9-3, and F14-16	A17.1
A.18 References	A18.1

Figures

3.1	Predicted Versus Measured BOL Centerline Temperatures from Five Halden Rods	3.2
3.2	BOL Centerline Temperature Deviation (Predicted Minus Measured) for Five Halden Rods as a Function of LHGR	3.3
3.3	Predicted Minus Measured BOL Centerline Temperature Versus LHGR for IFA-513 Rod 1 and IFA-432 Rod 1 with Nominal Gap Size	3.4
3.4	Predicted Minus Measured BOL Centerline Temperature Versus LHGR for IFA-432 Rod 3 with Small Gap Size	3.4
3.5	Predicted Minus Measured BOL Centerline Versus LHGR for IFA-432 Rod 2 with Large Gap Size	3.5
3.6	Predicted Minus Measured BOL Centerline Temperature for IFA-513 Rods 1 and 6	3.6
3.7	FRAPCON-3 Predicted and Measured Centerline Temperature for HUHB Assembly Rod 18 as a Function of Time	3.8
3.8	The Differences Between Predicted and Measured Temperatures Versus Time for HUHB Rod 18	3.8
3.9	Measured and FRAPCON-3 Predicted of Centerline Temperature Versus Time for the Upper Thermocouple of IFA-432 Rod 3	3.10
3.10	Measured and FRAPCON-3 Predicted Centerline Temperature Versus Time for the Lower Thermocouple of IFA-432 Rod 3	3.10
3.11	Temperature Differences (Predicted Minus Measured) as a Function of Burnup for IFA-432 Rod 3 and HUHB Rod 18	3.11
3.12	Code-Data Deviations for IFA-432 Rod 3 and HUHB Rod 18 with no Burnup Degradation Factor	3.11
3.13	Predicted Versus Measured Fuel Center Temperatures for Five IFA Rods	3.13
3.14	Predicted Temperature Minus Measured Temperature for Five IFA Rods Through-Life ..	3.13
4.1	Comparison of FRAPCON-3 Predictions to Measured FGR Data for the Experimental and Commercial Rods at Steady-State Power	4.3
4.2	Predicted-Minus-Measured FGR Versus Rod-Average Burnup for Steady-State Power Rods	4.4

4.3	Comparison of Code Predictions and Measured FGR Values for Steady-State and Bumped Power Fuel Rods	4.5
4.4	Measured-Minus-Predicted FGR Versus Rod-Average Burnup for Steady-State and Bumped Power Fuel Rods	4.6
5.1	Measured and Predicted Fuel Pellet Swelling	5.3
6.1	Measured and Predicted Corrosion Layer Thickness as a Function of Axial Position for Oconee 5-Cycle PWR Rod 15309	6.3
6.2	Measured and Predicted Corrosion Layer Thickness as a Function of Axial Position for ANO-2 5-Cycle PWR Rod TSQ002	6.4
7.1	Predicted and Measured Cladding Creepdown from the 2nd and 3rd Cycle Rods in the ANO-2 PWR	7.3
7.2	Predicted and Measured Cladding Creepdown from 3rd, 4th, and 5th Cycle Rods in the Oconee PWR	7.3
8.1	FRAPCON-3 Predicted-Minus-Measured Centerline Fuel Temperature at BOL as a Function of LHGR for Both Independent and Benchmark Data	8.4
8.2	Predicted-Minus-Measured Fuel Center Temperatures as a Function of Burnup for Benchmark Cases and Several (Independent) Data Sets Described in the Text	8.5
8.3	Ratio of FRAPCON-3 Predicted-Minus-Measured Divided by Measured Fuel Centerline-Minus-Coolant Temperatures as a Function of Fuel Burnup	8.6
8.4	Predicted-Minus-Measured FGR as a Function of Burnup for Benchmark Steady-State/Power-Ramp Cases and Several Independent Cases Described in the Text	8.6

Tables

2.1	Steady-State Fuel Rod Data Cases Used for FRAPCON-3 Integral Assessment	2.2
2.2	Steady-State Data Used for Assessment	2.3
2.3	Steady-State Evaluations of Cladding Axial Growth, Creepdown, Oxidation, and Hydrogen Uptake	2.4
2.4	Summary of Fuel Rod Design and Operating Data for Code Integral Assessment Cases Using EOL Power Ramps	2.5
3.1	The As-Fabricated Diametral Gap Size for the Selected Test Rods	3.3
3.2	The Standard Error and Average Bias of the Rods Considered for the Assessment of Gap Size	3.5
3.3	Design Variations for the Selected Test Rods	3.6
4.1	FRAPCON-3 FGR Predictions of Steady-State Rods	4.2
4.2	FRAPCON-3 FGR Predictions of Transient Rods	4.5
5.1	Measured and Calculated Void Volume for Five High Burnup Fuel Rods	5.1
5.2	Measured and Predicted Fuel Pellet Swelling	5.2
6.1	Measured and Calculated Oxidation and Hydrogen Concentration for High Burnup PWR Fuel Rods	6.2
6.2	Measured and Calculated Oxidation for Two High Burnup BWR Fuel Rods	6.3
7.1	Measured and Calculated Cladding Axial Growth	7.1
7.2	Measured and Predicted Rod-Average Cladding Creepdown	7.2
8.1	Independent Data for BOL Fuel Temperatures	8.2
8.2	Independent Data for Fuel Temperatures at Nominal-to-High Burnup	8.2
8.3	Independent Data for FGR at Nominal-to-High Burnup	8.3

Executive Summary

An integral assessment was performed for the Nuclear Regulatory Commission on the newly-developed steady-state fuel behavior code, FRAPCON-3. The code was developed from Version 1, Modification 5 of the FRAPCON-2 code with two objectives: to simplify the code and extend its applicability to high burnup (up to 65 GWd/MTU, depending on code application). There are two other volumes of this report, Volume 1 describes the properties and models updated for high burnup application, and Volume 2 describes the installation of the updated models in the code along with input instructions. The assessment was done by comparing the code predictions for fuel temperatures, fission gas release, rod internal void volume, fuel swelling, cladding creep/growth, and cladding corrosion/hydriding to data from integral irradiation experiments and postirradiation examination programs. In the case of fuel temperatures and gas release, data are scarce, and the primary data sets were actually used to benchmark the thermal models and the fission gas release model. Therefore, the code predictions are also compared herein to additional "independent" data sets for fuel temperatures and high-burnup fission gas release.

The cases used for code assessment were selected on the criteria of having well characterized design and operational data, and spanning the ranges of interest for both design and operating parameters. Thus, the fuel rods represent both boiling water reactor (BWR) and pressurized water reactor (PWR) fuel types, with pellet-to-cladding gap sizes within, above, and below the normal range for power reactor rods. The fill gas is pure helium in most cases, but cases are included with xenon and helium-xenon mixtures. The linear heat generation rates (LHGRs) at beginning-of-life (BOL) range up to 60 kW/m (18 kW/ft), and during end-of-life (EOL) power ramps, they range up to 45 kW/m (14 kW/ft). The rod-average fuel burnups range up to 76 GWd/MTU, but only up to 67 GWd/MTU for power-ramp cases. The EOL fission gas release (FGR) ranges from less than 1% to greater than 30% of the produced quantity.

The primary code assessment data base (used also for benchmarking the thermal and FGR models) consists of 30 well-characterized fuel rods. These include 10 (non-instrumented) test rods that experienced EOL power ramps (used for FGR) and 20 "steady-state" cases (6 instrumented Halden rods used for fuel temperatures, the remaining 14 being non-instrumented rods used for FGR). The 13 steady-state FGR cases include 4 commercial power reactor rods (2 from PWRs and 2 from BWRs), and the remaining 9 come from test reactors (Halden, BR-3, EL-3 and NRX).

The "independent" data base consists of 15 well characterized fuel rods. These include 6 Halden instrumented rods used for BOL fuel temperatures, 4 rods (Halden and DR-2) used for fuel temperatures at significant burnup, and 7 test rods or refabricated commercial rod segments used for FGR. Most of the rods at significant burnup experienced some kind of EOL power ramp. Two of the high burnup rods refabricated from commercial rod segments had both fuel temperature and FGR data.

Six rods from the primary set were used to assess FRAPCON-3 predictions of EOL void volume. The cases selected include 3 full-length power reactor rods and 3 shorter test reactor rods. A mix of test reactor and power-reactor rods was also used to assess the fuel volume change due to densification and swelling.

The FRAPCON model for cladding waterside oxidation was taken from the EPRI ESCORE code without modification and evaluated against PWR rod data in Volume 1 of this report. Here, demonstration comparisons are made to the same PWR rod cladding sections used to demonstrate the oxidation model predictions.

The axial rod growth model is taken without modification from Franklin and is compared to extensive PWR and some BWR data in Volume 1 of this report. Here we demonstrate the model predictions by comparison to 2 PWR rods and 2 BWR rods from the primary data set. The cladding creep model is unmodified (from FRAPCON-2). Here we demonstrate its predictions against data from 2 PWR rod sets 1 BWR rod set.

The following conclusions about FRAPCON-3 were made as a result of this assessment:

Thermal: The comparisons to centerline temperature data are divided into predictions of BOL data and temperature data as a function of burnup. The BOL predictions have been compared to centerline temperature measurements from instrumented rods irradiated in the Halden test reactor that are part of both the benchmarks and independent data sets. The code comparisons to the BOL data at LHGRs > 20 kW/m show a negligible underpredictive bias on average of -3°C and relatively small standard error of 30.5°C in the prediction of fuel centerline temperature for rods with only helium fill gas with the standard error increasing to 37.7°C when xenon filled rods are included. With the exception of the two RISØ rods, the code comparisons to the temperature data as a function of burnup show approximately no bias in the prediction out to a rod average burnup of 40 GWd/MTU with a standard error of 40°C (104°F). The reason for the significant underprediction of the two RISØ rods is unknown. The code underpredicts the two rods above 45 GWd/MTU, Rod 18 from the HUHB assembly, and the ramped Halden rod at 67 GWd/MTU by up to 18%, which are the only rods with measured temperature data above this burnup level.

Fission Gas Release: The comparisons to FGR data are divided into predictions of steady-state data (with steady-state power histories) and those data from fuel rods with power bumps (increase in rod power) at the EOL to simulate operational overpower transients. The predictions of FGR for the experimental fuel rods with steady-state and ramped power histories is very good with a standard error of 5.4% release if the two BWR commercial and the two HBEP ramped fuel rods with high FGR are excluded, and the standard error increases to 8.8% release when these rods are included. The significant underprediction and the increased error introduced by the commercial rods is hypothesized to be due to power uncertainties in commercial reactors caused by control-rod or control-blade movements. The significant underprediction of the HBEP ramped rods is hypothesized to be due to the unstable nature of this fuel that is atypical of today's fuel designs. Therefore, the code predictions of FGR are relatively good if rod powers are known accurately and the fuel is stable (low-densifying).

Internal Void Volume: Comparisons were made to data from four commercial reactor and three test reactor fuel rods. The code predicted the two commercial rods well but underpredicted the BR-3 test rod data by approximately 8.7% (relative) on average.

Cladding Corrosion and Hydriding: Comparisons were made to data from two commercial PWR rods and two commercial BWR rods. The oxide corrosion predictions were very good and tend to bracket the data, depending on the choice of crud layer thickness. The hydrogen concentration was similarly bracketed.

Cladding Creep: Comparisons were made to data from several commercial PWR rods from two fuel assemblies from two different vendors and one BWR commercial rod. The code predictions compare relatively well to the maximum creepdown values measured for the PWR rods and an underprediction of the BWR rod by 0.1% strain (absolute). These predictions are considered to be relatively good based on the usual large scatter in cladding creepdown data, particularly when data are from different fuel vendors. The comparisons to the PWR rod axial growth data were also relatively good while the two BWR rods were underpredicted by 25% (relative).

Acknowledgments

The authors acknowledge Mr. Harold Scott of the U. S. Nuclear Regulatory Commission for his technical guidance on the FRAPCON-3 code improvement project that resulted in this report. Also, we acknowledge K. J. Geelhood of PNNL for his technical support in performing calculations, developing graphics, and supporting data transfer to INEEL and Billie Reagan and Edna Johnsen of INEEL for work on the graphics, typing, and organizing this report.

1.0 Introduction

Fuel rod material properties and performance models have been updated for the FRAPCON steady-state fuel rod performance code to account for changes in behavior due to extended fuel burnup. The updated code is named FRAPCON-3 and is intended to replace the earlier codes FRAPCON-2 (Berna et al. 1981) and GAPCON-THERMAL-2 (Cunningham and Beyer 1984). The property and model updates are described in Volume 1 of this report (Lanning and Beyer 1997). Volume 2 of this report describes the installation of the updated properties and models in the FRAPCON-3 code along with the input instructions (Berna et al. 1997).

This report provides the results of the assessment of the integral code predictions to measured data for various performance parameters. In the case of fuel temperature and fission gas release (FGR) predictions, comparison is made to both benchmark data sets and independent benchmark data sets. The benchmark data sets are described in Section 2.0. Appendix A describes each individual set of benchmark data and gives the code input for each data comparison. The benchmark data are drawn from a wide range of burnup levels and operating conditions that are relevant to commercial operations. Experimental fuel rods were selected on the basis of the linear heat generation rates (LHGRs) to be at or near the maxima for commercial fuel operations because the NRC licenses fuel to the most limiting rod in the core. Not all the data selected are at limiting conditions because some of the cases involve commercial fuel rods that operated at normal commercial operating conditions, which are significantly less than the limiting conditions. Also, it is noted that most of the thermal and FGR benchmark cases are drawn from experimental programs that involved numerous fuel rods, of which only a few were selected as benchmark cases. This was either because the rods in a given group were all irradiated under similar conditions and had similar FGR or because only rods with design parameters and operating conditions similar to current commercial practice were selected. The independent data set were from experimental irradiation programs that were independent of the benchmark experimental data programs and where either FGR or temperatures were measured and rod powers were accurately known.

The integral code assessments include comparison to fuel temperature benchmark data in Section 3.0 and FGR benchmark data in Section 4.0. Comparisons of code predictions to internal void volume/fuel swelling, cladding corrosion and hydriding, and cladding creep and axial growth data are given in Sections 5.0, 6.0, and 7.0, respectively. Comparison of thermal and FGR predictions to independent data sets is given in Section 8.0, and a summary and conclusions are found in Section 9.0.

2.0 Assessment Data Description

A total of 30 benchmark cases (fuel rods) that have post irradiation examination (PIE) were selected to perform the integral assessment of the FRAPCON-3 code. These include 20 fuel rods with steady-state power operation covering a wide range of burnup and 10 fuel rods with steady-state irradiations followed by an EOL power ramp. The purpose of the code assessment was to assess the code against a limited set of well-qualified data that span the range of limiting operational conditions for commercial light water reactors (LWRs). The cases in this relatively limited group were selected using criteria regarding the completeness and the quality of the rod performance data, as follows:

- The cases should all provide pre-irradiation characterization and PIE data of the fuel rods of interest.
- Cases are needed that provide well-qualified fuel rod power and temperature data as a function of time or burnup.
- Cases at both low to high fuel burnup are needed, as well as low to high (limiting) LHGR.
- Cases are needed that provide cladding oxidation, hydriding and deformation under prototypic pressurized water reactor (PWR) and boiling water reactor (BWR) conditions.
- Cases are needed that demonstrate the effects of overpower transients at low and high burnup.

The selected cases fulfill the above criteria, and they provide a mix of well qualified test-reactor data and less qualified power-reactor rod data.

2.1 Description of the Steady-State Cases

The steady-state assessment cases are listed in Table 2.1, together with the EOL burnup for each rod and the major fuel design parameters (gap size, fill gas type/pressure), and major operational parameters (maximum rod-average LHGR, and FGR at EOL). The type of in-reactor instrumentation, if any, is also noted, such as centerline temperature measurement (TCL). For reference, note that the typical gap-to-diameter ratio for commercial power reactor rods is ~2%, and the typical operating rod average LHGR for most rods is 4 to 10 kW/ft for steady-state power operation. Only a few limiting rods in the first cycle of operation attain LHGRs greater than 10 kW/ft.

The rods are listed in the following groupings: test rods with in-reactor instrumentation for fuel temperature measurements (Halden); test reactor rods at nominal to high burnup (BR-3 rods); test reactor rods at low burnup but high LHGR (NRX, EL-3) and, finally, full-length PWR and BWR rods from commercial power reactors. Detailed information on each case is found in Appendix A.

Table 2.1. Steady-State Fuel Rod Data Cases Used for FRAPCON-3 Integral Assessment

Reactor and Type (reference)	Assembly and/or Rod Number	Rod-Average Burnup, GWD/MTU	Fuel-Cladding Diametral Gap Size, mils (gap-to-diameter ratio (%))	Initial Fill-Gas Type and Room Temperature Pressure, psi (Mpa)	Maximum Rod-Average LHGR, kW/ft (kW/m)	FGR at EOL, % of produced	In-Reactor Measurements
Halden HBWR (Wiesenack 1992)	HUHB ^(a) Rod 18	80	4 [1.7]	He, 145 (1.00)	11.56 (37.92)	<1.0	TCL, pressure
Halden HBWR (Lanning 1986)	IFA-432, Rod 1 Rod 2 Rod 3	30 30 40	9 [2.1] 15 [3.6] 3 [0.7]	He, 14.7 (.10) He, 14.7 (.10) He, 14.7 (.10)	11.87 (38.93) 12.54 (41.13) 12.54 (41.13)	20 ± 10 30 ± 10 10 ± 10	TCL, pressure, elongation
Halden HBWR (Bradley et al. 1981)	IFA-513, Rod 1 Rod 6	12	9 [2.1] 9 [2.1]	He, 14.7 (.10) 77%He, 23%Xe, 14.7 (.10)	12.20 (40.02) 12.20 (40.02)	<1.0 2.0	TCL, pressure, elongation
Halden HBWR ^(b)	IFA-429,-519 Rod DH	74	8 [2.2]	He, 376 (2.59)	12.65 (41.49)	24	Pressure
BR-3 Test reactor/PWR conditions (Balfour 1982; Balfour et al. 1982)	Westinghouse Hi bu rods 36-I-8 111-I-5 24-I-6 28-I-6	61.5 48.6 60.1 53.3	7.5 [2.1] 7.5 [2.1] 7.5 [2.1] 7.5 [2.1]	He, 214 (1.48) He, 214 (1.48) He, 200 (1.38) He, 200 (1.38)	12.60 (41.33) 13.94 (45.72) 13.32 (43.69) 10.5 (34.4)	34 14 22 13.2	None
BR-3 Test reactor PWR conditions (Barner et al. 1990)	BNFL-DE	41.5	8.4 [2.3]	He, 14.7 (.10)	14.28 (46.84)	10.7	None
NRX Test reactor (Notley et al. 1967)	LFF	2.2	18 [2.5]	He, 146 (1.01)	17.8 (58.38)	17.3	None
NRX Test reactor (Notley et al. 1965)	CBP	2.6	18 [2.9]	He, 146 (1.01)	16.8 (55.10)	14.1	None
EL-3 Test reactor (Janvier et al. 1967)	4110-AE2 4110-BE2	6.2 6.6	16 [3.2] 14 [2.8]	He, 147 (1.01) He, 147 (1.01)	17.6 (57.73) 17.8 (58.38)	22.1 15.9	None None
ANO-2 PWR (Smith, Jr., et al. 1994)	TSQ002	53	7 [2.2]	He, 380 (2.62)	6.95 (22.80)	<1.0	None
Oconee PWR (Newman 1986)	15309	50	10 [2.7]	He, 480 (3.31)	7.9 (25.91)	0.8	None
Monticello BWR (Baumgartner 1984)	MTAB099 Rod A1	45	9 [2.2]	He, 14.7 (.10)	6.92 (22.70)	30.0	None
TVO-1 BWR (Barner et al. 1990)	HBEP H8/36-6	51.4	8.27 [2.1]	He, 56.6 (.39)	7.14 (23.42)	11.2	None

(a) Halden ultra high burnup.
(b) Halden Reactor Project. 1997. Personal communication with USNRC.

The pellet and rod dimensions for these cases are generally typical of PWR and BWR fuel rods. One exception, noted in Table 2.1, is the small-diameter, highly annular pellets used for the Halden Ultra-High Burnup (HUHB) assembly. Fuel enrichments are in the 3 to 5% range for the PWR and BWR rods, but are higher for the test reactor rods, for example, 6 to 8% for the BR-3 rods and 10 to 13% for the Halden rods.

Note that fuel burnup represented by these cases ranges from 2 to 74 GWd/MTU for FGR cases (and up to 76 GWd/MTU for HUHB, the thermal conductivity case). The beginning-of-life (BOL) LHGRs range up to 18 kW/ft for the low-burnup FGR cases. The FGR data ranges from <1% to >30%.

The code-data comparisons evaluated for fuel-pellet or fuel-rod performance from the various steady-state cases are listed in Table 2.2. This table presents the steady-state fuel behavior phenomena that are assessed in this report and which cases are used for that assessment. An "x" in a table cell indicates that the corresponding data comparison was performed for a particular case to assess code predictions. For example, 2 cases (HUHB and IFA-432 Rod 3) were used to assess fuel thermal predictions (thermal conductivity degradation) with burnup, and 14 cases were used to assess the steady-state FGR predictions.

Table 2.2. Steady-State Data Used for Assessment

Reactor	Assembly and Rod Number	Rod-Average Burnup, GWd/MTU	Fuel Thermal Versus Burnup	BOL Thermal	FGR	Fuel Swelling and Rod Void Volume
Halden HBWR	HUHB	80	x			
Halden HBWR	IFA-432 R1, R2, R3	30 to 40	x (Rod 3)	x		x
Halden HBWR	IFA-513 R1, R6	12		x		
Halden HBWR	IFA-429 Rod DH	74			x	
BR-3 PWR	36-I-8	61.5			x	x
BR-3 PWR	111-I-5	48.6			x	x
BR-3 PWR	24-I-6	60.1			x	x
BR-3 PWR	38-I-6	53.3			x	x
BR-3 PWR	BNFL-DE	41.5			x	
NRX PWR	LFF	2.2			x	
NRX PWR	CBF	2.6			x	
EL-3 PWR	4110-AE2	6.2			x	
EL-3 PWR	4110-BE2	6.6			x	
ANO-2 PWR	TSQ002	53			x	x
Oconee PWR	15309	50			x	x
Monticello BWR	A1	45			x	
TVO-1 BWR	H8/36-6	51.4			x	

The matrix of fuel rod data used to assess cladding oxidation and deformation evaluations is shown in Table 2.3. This matrix is limited to the full-length power reactor rods, because only those rods operated in prototypic neutronic and coolant conditions, both of which affect creepdown, axial growth, and corrosion/hydrating.

Table 2.3. Steady-State Evaluations of Cladding Axial Growth, Creepdown, Oxidation, and Hydrogen Uptake

Reactor	Assembly and Rod Number	Rod-Average Burnup, GWd/MTU	Cladding Axial Growth	Cladding Creepdown	Cladding Oxidation and Hydrogen Uptake
Oconee-1 PWR	15309	50	x	x	x
ANO-2 PWR	TSQ002	53	x	x	x
Monticello BWR	A1	45	x	x	x
TVO-1 BWR	H8/36-6	51.4	x		x

2.2 Description of the Power-Ramp Cases

The major fabrication and operational parameters for the power-ramped rods are listed in Tables 2.2 through 2.4. Note that, for all these cases, the pre-ramp base irradiation occurred at nominal to low LHGRs, and the pre-ramp FGR was low relative to the significant FGR that occurred during the ramp.

Each of these cases was used for transient (ramp) FGR evaluations only; hence, no evaluations corresponding to those of Tables 2.2 and 2.3 are presented here.

Table 2.4. Summary of Fuel Rod Design and Operating Data for Code Integral Assessment Cases Using EOL Power Ramps

Reactor for Base Irradiation/ Reactor for Ramp Test (references)	Assembly and Rod Number	Rod-Average Burnup, Gwd/MTU	Fuel-Cladding Diametral Gap Size, mils (microns)	Fill Gas Type and Pressure, psi (Mpa)	Maximum Rod-Average LHGR, kW/ft (kW/m)	Ramp Terminal Level, kW/ft (and hold time in hours)	FGR Pre-Ramp (and Post-Ramp), % of produced
Obrigheim/Petten (Barner et al. 1990)	D200	25	8 (203)	He, 305 (2.10)	8.26 (27.09)	13.8 (48)	6.6 (38)
	D226	44	6.7 (170)	He, 305 (2.10)	8.32 (27.29)	13.1 (48)	4.2 (44.1)
Obrigheim/Petten (Djurle 1985)	PK6-2	35	5.7 (145)	He, 326 (2.25)	8.2 (26.90)	12.2 (12)	NA (3.5)
	PK6-3	35	5.7 (145)	He, 326 (2.25)	8.2 (26.90)	13.1 (12)	NA (6.7)
	PK6-S	35	5.7 (145)	He, 326 (2.25)	8.2 (26.90)	12.5 (12)	NA (6.1)
Studsvik/studsvik (ramp tested) ^{(a)(b)}	Rod 16	21	5 (127)	He, 14.7 (.10)	13.1 (42.97)	14.6 (24)	NA (16)
	Rod 18	18	5 (127)	He, 14.7 (.10)	10.97 (35.98)	12.5 (24)	NA (4)
Halden DR-2 (ramp tested) (Knudsen et al. 1983)	F7-3	35	7.1 (180)	He, 14.7 (.10)	13.33 (43.72)	13.0 (24)	5.7 (11.5)
	F14-6	27	7.1 (180)	(all rods)	10.43 (34.21)	13.44 (24)	5.8 (22.1)
	F9-3	33	7.1 (180)		13.3 (43.7)	13.3 (43.7)	7.3 (17.5)

(a) H. Mogard, U. Bergenlid, S. Djurle, J. A. Gyllander, E. Larsson, G. Lysell, G. Ronnberg, K. Saltvedt, and H. Tomani. 1979. Final Report of the Inter-Ramp Project, STIR-53, (restricted distribution), Studsvik AB Atomenergi, Studsvik, Sweden.

(b) G. Lysell and S. Birath. 1979. Hot Cell Post-Irradiation Examination of Inter-Ramp Fuel Rods, STIR-51 (restricted distribution) Studsvik AB Atomenergi, Studsvik, Sweden.

3.0 Thermal Behavior Assessment

Thermal predictions are important for calculating initial fuel stored energy, which is used as input to loss-of-coolant-accident (LOCA) analyses. The fuel temperatures are also used to calculate FGRs and EOL rod pressures and to check for fuel melting. In general, PWR LOCA and fuel melting analyses are more limiting at BOL, while the same analyses for BWRs are generally more limiting at burnups between 15 and 25 GWd/MTU.

Predicted and measured fuel center temperatures from instrumented Halden reactor test assemblies have been used to evaluate the code's ability to predict BOL temperatures and through-life temperature histories. The BOL temperature comparisons are needed to establish bias and uncertainties (based on standard deviation) in the code thermal predictions, which will be used to bound initial fuel-stored energy for PWR LOCA and temperatures for PWR fuel-melting analyses. The through-life temperature history comparisons will be used to bound the uncertainties on BWR LOCA initialization and fuel melting analyses. The BOL temperature data base includes not only rods with helium-filled gaps, but also rods with xenon and xenon-helium filled gaps and rods with pellet/cladding gap sizes both larger and smaller than nominal. These variations provide the points for code evaluation beyond the normal ranges for gap size and thermal resistance.

The comparisons of measured and predicted through-life fuel center temperature histories were done with two goals in mind. The first was to check on the trend of thermal conductivity degradation with burnup, which is demonstrated by two rods: Rod 18 from the HUHB, which was specifically designed and operated to enhance the burnup effect, and the small-gap Rod 3 from the NRC Halden test fuel assembly IFA-432. These data along with data from the independent database discussed in Section 8.0 provide an estimate of thermal bias and uncertainty as a function of burnup. The second goal was to check on the effect of thermal feedback caused by (temperature driven) gas release and consequent contamination of the initial helium fill gas with lower-conductivity fission gas. Rod 1 from IFA-432 and Rod 1 from the generally similar IFA-516 were selected to demonstrate this.

The BOL and through-life code-data comparisons are discussed separately below.

3.1 BOL Fuel Center Temperature Predictions

Rods from the NRC-sponsored Halden instrumented test fuel assemblies IFA-432 and IFA-513 were selected to assess code predictions of BOL temperatures. The purpose was to test the long-term steady-state performance of BWR-6 type fuel rods, operated at power levels near the operating limits for commercial reactor rods. The fuel pellets were fabricated at Pacific Northwest National Laboratory (PNNL) and then shipped to Norway; final assembly and rod fabrication were completed at the Halden site, and irradiation began in 1976. Destructive examinations were carried out at Harwell, U.K.

The IFA-513 assembly was irradiated in the Halden Reactor starting from 1978. This assembly was a continuation of the same NRC experimental program and consisted of six instrumented rods with differences in fill-gas type and fill-gas pressure.

The code predictions compare very closely to the measured values at BOL. In Figure 3.1, the predicted fuel temperatures are plotted against the measured values. The same data are plotted in Figure 3.2 in a different form. The predicted temperature minus the measured temperature is plotted as a function of the local LHGR. The bias and standard error can be calculated from this form of the comparison, and the results are a very small bias (-3°C) and a small standard error (28°C). These change to $+2^{\circ}\text{C}$ and 23°C for the helium-only rods. It is noted that the absolute standard error does not change as a function of LHGR above an LGHR of 10 kW/m .

Figure 3.1. Predicted Versus Measured BOL Centerline Temperatures from Five Halden Rods

3.1.1 Effect of Gap Size

Four rods were chosen to demonstrate the effect of varied as-fabricated pellet-cladding gap size on the BOL fuel temperatures. These were Rods 1, 2, and 3 from IFA-432 and Rod 1 from IFA-513. Table 3.1 lists these rods and their gap size and the gap-to-diameter ratios. Note that normal gap-to-diameter ratios for commercial power reactor rods are close to 2.0%.

Figure 3.2. BOL Centerline Temperature Deviation (Predicted Minus Measured) for Five Halden Rods as a Function of LHGR

Table 3.1. The As-Fabricated Diametral Gap Size for the Selected Test Rods

Rod Number	Gap Size, mils	g/D, %
IFA-513 Rod 1	9.05	2.1 (nominal)
IFA-432 Rod 1	9.05	2.1 (nominal)
IFA-432 Rod 2	15.0	3.6 (large)
IFA-432 Rod 3	3.0	0.7 (small)

The predicted fuel temperatures show good agreement with the data for these rods. Figure 3.3 shows the predicted temperature minus the measured temperature as a function of LHGR for the nominal-gap rods (Rod 1, IFA-432 and Rod 1, IFA-513). Figures 3.4 and 3.5 show the same comparison, respectively, for the small-gap Rod 3 of IFA-432 and the large-gap Rod 2 of IFA-432. Table 3.2 gives the standard error and the bias for each of the rods considered.

Figure 3.3. Predicted Minus Measured BOL Centerline Temperature Versus LHGR for IFA-513 Rod 1 and IFA-432 Rod 1 with Nominal Gap Size

Figure 3.4. Predicted Minus Measured BOL Centerline Temperature Versus LHGR for IFA-432 Rod 3 with Small Gap Size

Figure 3.5. Predicted Minus Measured BOL Centerline Versus LHGR for IFA-432 Rod 2 with Large Gap Size

Table 3.2. The Standard Error and Average Bias of the Rods Considered for the Assessment of Gap Size

Rod Number	Standard Error	Average Bias	Type of Gap
IFA-513 Rod 1	18°C	9.0°C	(nominal)
IFA-432 Rod 1	22°C	-2.0°C	(nominal)
IFA-432 Rod 2	29°C	6.4°C	(large)
IFA-432 Rod 3	36°C	-0.5°C	(small)

3.1.2 Effect of Fill Gas Mixture

Two rods were chosen to investigate the ability of FRAPCON-3 to predicted BOL fuel temperatures with varying gas composition: Rods 1 and 6 from the IFA-513 assembly. The differences in these rods are only in the fill gas composition; see Table 3.3. The helium-xenon mixture in Rod 6 was selected to provide about 50% of the gas thermal conductivity of pure helium.

Table 3.3. Design Variations for the Selected Test Rods

Rod Number	Fuel Density (% TD)	Gap Size (mils)	Fill Gas Composition (molar %)
IFA-513 Rod 1	95	9.05	100% He
IFA-513 Rod 6	95	9.05	23% Xe, 77% He

Figure 3.6 shows the predicted temperatures minus the measured temperatures for both rods as a function of LHGR. Above 30 kW/m, Rod 6 is somewhat underpredicted; however, the relative error is still within acceptable bounds. Below 30 kW/m, the scatter and bias in the Rod 6 predictions are comparable small values similar to those for Rod 1.

Figure 3.6. Predicted Minus Measured BOL Centerline Temperature for IFA-513 Rods 1 and 6

3.2 Assessment of Temperature Predictions as a Function of Burnup

Two groups of assessment cases are discussed below: 1) the code-to-data comparisons that reveal fuel thermal conductivity degradation, and 2) those cases that reveal the general effect of "thermal feedback," which is the increase in fuel temperatures due to the contamination of the helium fill gas by lower-conductivity fission gas released from the fuel pellets.

3.2.1 Fuel Thermal Conductivity Degradation from Halden Experimental Rods

The phonon-phonon portion of the fuel thermal conductivity is subject to degradation (reduction) due to lattice damage and distortion caused by the fissioning process. Lucuta made a recommendation for this degradation on the basis of ex-reactor diffusivity measurements on SIMFUEL (a mixture of sintered urania and rare earths meant to simulate the effects of fission-product buildup in the fuel), and based on ex-reactor measurement of fuel diffusivity on high burnup fuel.

Two sets of in-reactor data have been examined in the course of evaluating fuel thermal conductivity as a function of burnup: the temperature/LHGR data verses burnup (up to 76 GWd/MTU) from a representative of the specially-designed HUHB rods; and the temperature/LHGR data (up to 40 GWd/MTU) from the small-gap BWR-sized Rod 3 of assembly IFA-432. The code-to-data comparisons for these sets are presented in the following sections.

3.2.1.1 The HUHB Assembly

The Halden Project has designed and operated a specialized group of fuel rods in Halden Ultra-High Burnup (HUHB) assembly, which are specifically designed to enhance the measurable effect of fuel thermal conductivity degradation (reduction) as a function of burn up. The rods are small in diameter, and the pellets are annular, with a thermal expansion temperature meter placed inside the full length of the pellet column. Thus, a rod-average center temperature is measured, and the fuel temperatures are relatively low, where the thermal conductivity degradation has a greater absolute effect. Furthermore the confounding effects of thermal feedback have been eliminated by minimizing FGR by keeping fuel operating temperatures relatively low (<1000 K).

Two of the six rods in the assembly were equipped with pressure transducers in place of the thermal expansion meters to monitor the FGR. The pressure measurements have confirmed very low gas release ($<1\%$ of produced) and very minor contamination/thermal conductivity degradation of the helium fill gas in these prepressurized rods. Thus, the temperature increases at a given LHGR as a function of burn up in the companion rods are attributed solely to degradation of fuel thermal conductivity and not to thermal feedback effects. This is consistent with the low fuel temperatures, which result from the annular small pellet design and the low LHGR (<25 kW/m) that prevailed through most of the irradiation period.

All four rods that were instrumented with expansion meters demonstrated very similar results for LHGR and temperature as a function of burn up. Rod 18 was selected as a representative rod. The FRAPCON-3 centerline temperature predictions for Rod 18 as a function of time are shown in relationship to the data (obtained from Halden in digitized form) in Figure 3.7. The differences between predicted and measured temperatures for Rod 18 are plotted in Figure 3.8. The predictions near BOL exceed the measured values, approach equality with the measurements over the burn up range from 15 to 35 GWd/MTU, and are less than the measured values from about 40 GWd/MTU to the limit of the reported data (76 GWd/MTU). Because the gap-to-diameter ratio for these rods (1.7%) was large relative to a small-gap rod ($<1\%$) and the fuel/cladding roughnesses are small, the gap is calculated to

Figure 3.7. FRAPCON-3 Predicted and Measured Centerline Temperature for HUIB Assembly Rod 18 as a Function of Time

Figure 3.8. The Differences Between Predicted and Measured Temperatures Versus Time for HUIB Rod 18

have remained open out to a burn up of ~ 60 GWd/MTU. Therefore, uncertainty in the gap size has a significant impact on the degradation factor derived from these data. The FRAPCON-3 overprediction of fuel temperature at BOL is connected to an overprediction of the gap size at BOL, and the underprediction of fuel temperature late in life may be partially due to underprediction of the gap size at that time. The overprediction of BOL is not of particular concern because the large diameter annular hole with respect to the small diameter pellet and the presence of the thermal expansion meter significantly alter the fuel relocation near BOL compared to a commercial fuel rod and, therefore, is considered to be atypical.

3.2.1.2 IFA-432 Rod 3

The IFA-432 experiment was sponsored by NRC and designed by PNNL to demonstrate the thermal effects of variations in various BWR fuel rod parameters, mainly gap size and density. The small-gap rod number 3 in this assembly had an as-fabricated gap of 75 microns ($g/D = 0.7\%$), and consequently operated at temperatures that kept its gas release relatively low. The burn up was relatively extended (40 GWd/MTU) with a surviving centerline thermocouple; therefore, this rod offers data to crosscheck the burn up degradation effect derived from IFA-562.

Because of leakage during the PIE puncturing process for gas analysis and recovery, the gas release could not be determined from the plenum gas analysis. It was estimated instead from multiple measurements of retained fission gas in the fuel pellets. The latter method involved large uncertainty, and the reported FGR for this rod is $10 \pm 10\%$. The calculated FGR by FRAPCON-3 is 16%, at the high end of the reported range.

The thermal effect of the uncertainty in FGR in this rod is significant in spite of its small gap size because the rod was unpressurized (1 atm initial helium fill-gas pressure), and, consequently, FGR quickly contaminates the fill gas. These code predictions for temperature as a function of burn up are compared to the thermocouple data from the upper end of the rod in Figures 3.3 through 3.9 and for the longer-lasting thermocouple in the lower end of the rod in Figure 3.10. As can be seen, these predictions overpredict the measured temperatures at both the upper and lower thermocouples when irradiation time exceeds 500 days. This would indicate that the code overpredicts FGR for this rod, leading to greater thermal feedback and thermal overprediction between 50 to 100°C.

The differences between measured and predicted values are plotted as a function of burn up in Figure 3.11, together with the differences for HUHB Rod 18.

To put these items in perspective, however, consider Figure 3.12, where the code-data deviations for the two rods are shown, assuming no thermal conductivity degradation. Therefore, it is concluded that the current FRAPCON degradation function is clearly an improvement over no degradation. However, further in-reactor and ex-reactor data are needed to determine a more precise estimate of degradation because only the HUHB experimental test provides data above 40 GWd/MTU. In addition, as discussed in Volume 1 of this report, ex-reactor thermal diffusivity measurements from high burn up fuel suggest the burn up degradation is not as large as indicated by the HUHB experimental test.

Figure 3.9. Measured and FRAPCON-3 Predicted Centerline Temperature Versus Time for the Upper Thermocouple of IFA-432 Rod 3

Figure 3.10. Measured and FRAPCON-3 Predicted Centerline Temperature Versus Time for the Lower Thermocouple of IFA-432 Rod 3

Figure 3.12. Code-Data Deviations for IFA-432 Rod 3 and HUH Rod 18 with no Burnup

Figure 3.11. Temperature Differences (Predicted Minus Measured) as a Function of Burnup for IFA-432 Rod 3 and HUH Rod 18

3.2.1.3 Other Halden Rods

The burn up-dependent temperature comparison cases in the assessment database include the two rods discussed from HUHB and IFA-432, plus the following three Halden rods.

Rod 1 from IFA-432: Rod 1 is helium filled with a nominal (9-mil) diametral gap ($g/D = 2.1\%$). EOL rod-average burn up was 30 GWd/MTU. The g/D ratio for commercial reactor fuel rods is $\sim 2\%$, and therefore, Rod 1 is typical of commercial BWR reactor design with the exception that today's designs have initial fill pressures between 4 to 8 atmospheres.

Rods 1 and 6 from the NRC-sponsored IFA-513 Halden test assembly: Rod 1 is a 9-mil-gap, 1-atmosphere helium-filled rod, and Rod 6 is a 9-mil-gap rod filled with 77% He/23% Xe mixed gas to assess the effect of reducing the gas thermal conductivity by a factor of 2.0. Rod-average burnups at the end of the reported data were ~ 8 GWd/MTU.

Plots of measured and predicted fuel temperatures for each of these rods appear in Appendix A.

3.2.2 Overall Comparison of Temperature Predictions Versus Fuel Burn up of Halden Rods

The predicted versus measured fuel centerline temperatures for these five rods are shown in Figure 3.13. We do not consider the extreme overprediction of up to 200°C to be representative of the code performance for nominal-gap, prepressurized fuel rods that are typical of modern commercial reactor fuel design. The reason for the overprediction is because the code has a stronger dependence on thermal feedback due to FGR than observed in the experimental Halden fuel rods with 1-atmosphere helium fill gas pressure.

The differences between predicted and measured temperatures are shown in Figure 3.14. The standard error and bias were calculated as 80°C and 43°C , respectively, for burnups less than 40 GWd/MTU. The standard error and bias do not appear to vary with burn up up to 40 GWd/MTU; however, it is noted that very little thermal data exist above burnups of 20 GWd/MTU, particularly at LHGR/fuel centerline temperatures calculated for BWR LOCA initialization. There is currently too little data to quantitatively estimate code uncertainty above 40 GWd/MTU.

Figure 3.13. Predicted Versus Measured Fuel Center Temperatures for Five IFA Rods

Figure 3.14. Predicted Temperature Minus Measured Temperature for Five IFA Rods Through-Life

4.0 Fission Gas Release Assessment

An accurate prediction of FGR is important for two reasons: 1) it has a significant impact on the prediction of gap conductance and, therefore, fuel temperatures; e.g., as demonstrated in Section 3.0, an overprediction of FGR can result in an overprediction of fuel temperatures and the converse is also true, and 2) it is necessary for the calculation of rod internal pressures that impact LOCA analyses and EOL rod pressures. Currently, the limits on, and analyses of EOL rod pressures determine the LHGR limits for commercial fuel at burnups greater than 30 GWd/MTU. In addition, the NRC requires that these EOL rod pressure analyses include bounding anticipated operational occurrences (AOOs), e.g., overpower transients of several minutes to hours in length. Therefore, the accurate prediction of transient FGR under conditions of power increases above normal operation is important for licensing analyses.

The code's ability to predict FGR has been assessed based on comparisons to FGR data from 14 fuel rods with power histories that are relatively steady-state through the rods' irradiation life and 10 rods with power bumping (increase in rod power) at EOL to simulate an overpower AOO. The assessment in this section has used the MASSIH subroutine in the code that is based on a modified release model proposed by Forsberg and Massih (1985). This release model is described in Volume 1 of this report. An assessment of the ANS5.4 release model that is also an option in the code is provided in Volume 1 of this report. The conclusions from this assessment were that the ANS5.4 model provided a good prediction of FGR for fuel rods with steady-state power histories, but on average underpredicted FGR for fuel rods with power bumping for a few hours duration. This is not too surprising because this model was not intended to predict power transients of short duration.

The following discussions are divided into comparisons of the code predictions to steady-state FGR data and to power bumping (transient) FGR data.

4.1 Assessment of Steady-State FGR Predictions

The assessment of code FGR predictions is based on comparisons to fuel rods with steady-state and measured FGR data from less than 1% to 34% release and rod-average burnups up to 74 GWd/MTU power histories from nine different experimental programs. The code predictions and measured FGR data are provided in Table 4.1. Rod DH from IFA-429/519.9 was selected because FGR was relatively high at 24% release, the rod achieved a very high rod-average burnup level of 74 GWd/MTU, and the relatively high LHGRs of 9 to 12 kW/ft towards the EOL. It should be noted that this rod also experienced power cycling to simulate load follow operation beyond rod-average burnups of 30 GWd/MTU with operation at low powers for several days and higher power for several days.

The majority of the FGR data are from experimental fuel rods irradiated in test reactors with short cores, i.e., less than or equal to 1 meter in length. The test reactor data were selected because the fuel rod power histories in test reactors can be better controlled and more accurately determined than for rods in

Table 4.1. FRAPCON-3 FGR Predictions of Steady-State Rods

Reactor	Assembly and/or Rod Number	Rod-Average Burnup, GWD/MTU	Measured FGR, %	Predicted FGR, %
Halden	IFA-429, IFA-519.9 Rod DH	74	24	42.2
BR-3	24i6	60.1	22	20.6
BR-3	36i8	61.5	34	35.4
BR-3	111i5	48.6	14	13.6
BR-3	28i6	53.3	13.2	12.3
EL-3	4110-AE2	6.2	22.1	18.4
EL-3	4110-BE2	6.6	15.9	20.6
BR-3	BNFL-DE	41.5	10.7	7.70
NRX	CBP	2.6	14.1	11.8
NRX	LFF	2.2	17.3	19.6
ANO-2	TSQ002	53	<1	1.94
Oconee	15309	50	0.8	1.51
Monticello	A1	45	30.0	4.37
TVO-1	H8-36-6	51.4	11.2	0.64

commercial reactors. This is particularly true for commercial BWR rods where control blade movements have significant impact on adjacent fuel rod powers, and the impact is not accurately calculated. The primary driving force for FGR is fuel temperatures that are strongly dependent on fuel rod powers. Therefore, the FGR data from experimental fuel rods are considered to be much more reliable for assessing code FGR predictions than the commercial reactor data.

The FGR assessment data contain the following four fuel rods from commercial reactors: TSQ002 (ANO-2, PWR), 15309 (Oconee-1, PWR), Rod A1 (Monticello, BWR), and H8/36-6 (TVO-1, BWR). Both of the BWR rods had high measured FGR values of 30% and 11.2%, respectively. The code predictions of these two commercial BWR rods were 5.3% and 0.8%, respectively (Table 4.1). Therefore, the code underpredicted the measured FGR for both of the commercial rods with significant FGR. This is not too surprising because the detailed local rod power in commercial BWR reactors is not

accurately documented during control blade movements or during xenon transients, and as noted earlier, the power histories have larger uncertainties than test reactor data. Both Rods A1 and H8/36-6 are known to have resided in a corner position next to a control blade where blade movement has the greatest effect on local rod power.

The predictions of FGR for the ten experimental rods were much better (Table 4.1 and Figure 4.1) with all rods predicted within 5% of the measured release. The one exception was the highest burnup Rod DH from IFA-429/519.9 that was overpredicted by 19% release. The predicted minus the measured values are plotted versus rod-average burnup in Figure 4.2. Examination of Figures 4.1 and 4.2 does not show any bias in the code predictions other than the underprediction for the commercial rods. The standard error on the prediction of the test reactor rods is only 5.6% release and 9.4% release when the commercial rods are included.

It is concluded that the code provides a reasonably good prediction of FGR at steady-state powers when an accurate estimate of the fuel rod power history is known. The error in the code FGR predictions is related to the uncertainty in rod powers.

Figure 4.1. Comparison of FRAPCON-3 Predictions to Measured FGR Data for the Experimental and Commercial Rods at Steady-State Power

4.2 Assessment of Transient FGR Predictions

The assessment of transient FGR predictions is based on comparisons to 10 fuel rods from four different experimental programs with power bumps at the EOL to simulate AOOs (overpower transients) in commercial fuel rods. The peak LHGRs of the power bumps are between 12.2 and 14.6 kW/ft with

Figure 4.2. Predicted-Minus-Measured FGR Versus Rod-Average Burnup for Steady-State Power Rods

hold times between 12 to 48 hours at peak powers. The bumped powers are within the range of the bounding values for overpower AOOs for commercial fuel rods. The rod-average burnup of these data is limited to between 18 to 44 GWd/MTU. This demonstrates the need for further power-bumping data from fuel rods at higher burnup levels.

The code predictions and measured FGR values for the 10 experimental fuel rods are provided in Table 4.2 and plotted in Figure 4.3 along with the steady-state FGR predictions and data. The differences between predicted and measured values for both the transient and steady-state cases are plotted versus burnup in Figure 4.4. These comparisons show that the code does a good prediction of the transient FGR data except for the two High Burnup Effects Program Rods D200 and D226, which are underpredicted by 13% and 21% release, respectively. The fuel in both these rods is considered to be atypical of today's fuel used in commercial rods because it was prone to significant fuel densification (>2.5% TD) rather than the less densification prone (stable) fuel (<1.5% TD) of current fuel designs. In addition, there is evidence that fuel with significant densification releases more fission gas than current stable fuel. Both of these rods were included in the transient FGR data used for FRAPCON-3 assessment because of the lack of transient data from stable fuel. The standard deviation of the steady-state and bumped test rods increases to 9.18% release because of the underprediction of the two HBEP rods and a standard deviation of 10.29% for all rods.

It is concluded that the code prediction of transient FGR due to power bumps is not nearly as good as the code predictions of steady-state power data because of the underprediction of the two densification prone rods. It is also concluded that additional transient FGR data are needed for high-burnup fuel rods.

Table 4.2. FRAPCON-3 FGR Predictions of Transient Rods

Reactor for Base Irradiation/ Reactor for Ramp Test	Assembly and Rod Number	Rod-Average Burnup, Gwd/MTU	Measured FGR, % (Pre-Bump) Post-Bump	Predicted FGR % (Pre-Bump) Post-Bump
Obrigheim/Petten	D200	25	(6.6) 38	(0.2) 13.1
Obrigheim/Petten	D226	44	(4.2) 44.1	(0.9) 27.1
Obrigheim/Studsvik	PK6-2	35	(NA) 3.5	(0.2) 6.0
Obrigheim/Studsvik	PK6-3	35	(NA) 6.7	(0.2) 7.0
Obrigheim/Studsvik	PK6-S	35	(NA) 6.1	(0.2) 6.6
Studsvik/Studsvik	Rod 16	21	(NA) 16	(5.5) 13.7
Studsvik/Studsvik	Rod 18	18	(NA) 4	(1.7) 6.0
Halden/DR-2	F7-3	35	(5.7) 11.5	(1.0) 9.9
Halden/DR-2	F9-3	33	(7.3) 17.5	(1.0) 14.7
Halden/DR-2	F14-6	27	(5.8) 22.1	(0.2) 12.7

Figure 4.3. Comparison of Code Predictions and Measured FGR Values for Steady-State and Bumped Power Fuel Rods

Figure 4.4. Measured-Minus-Predicted FGR Versus Rod-Average Burnup for Steady-State and Bumped Power Fuel Rods

5.0 Internal Rod Void Volume Assessment

5.1 Fuel Rod Void Volume

An accurate prediction of the internal void volume of a fuel rod is important in the calculation of the internal rod pressures along with the FGR prediction. Five well characterized fuel rods were selected to assess the capability of FRAPCON-3 to accurately calculate fuel rod void volumes for high burnup. The cases selected include two full length rods (Rod TSQ002 from ANO-2 and Rod 15309 from Oconee) and three short (44 inches long) rods (36-I-8, 111-I-5, and 24-I-6) that were irradiated in the BR-3 reactor. The set includes only PWR fuel rods with standard Zircaloy-4. The burnup levels achieved on these rods range from 48.6 to 61.5 GWd/MTU.

Table 5.1 presents the measured and FRAPCON-3 calculated void volume at both BOL and EOL for the five fuel rods. The calculations were made at 25°C (77°F), which should be reasonably close to the temperature at which the data were collected. A range of values for void volume is provided for Oconee rod 15309 because this is the range of void volumes measured from 16 sibling fuel rods from the same assembly—including the representative rod 15309. All sixteen rods have very similar EOL burnups and similar power histories. Therefore, the void volume range includes representative uncertainty in the fabricated void volumes, measured rod power histories, and burnup.

The FRAPCON-3 code does a credible job of calculating the integral fuel rod void volumes, particularly for the commercial reactor rods where as-fabricated void volumes were provided. The three BR-3 test rods are overpredicted by 8.7% on average, but this may be due to an overestimation in the as-fabricated void volumes.

Table 5.1. Measured and Calculated Void Volume for Five High Burnup Fuel Rods

Reactor	Rod	Burnup, Gwd/MTU	BOL Void Volume, in. ³		EOL Void Volume, in. ³	
			Measured	Calculated	Measured	Calculated
BR-3	36-I-8	61.5	NA	0.646	0.508	0.535
BR-3	111-I-5	48.6	NA	0.648	0.516	0.573
BR-3	24-I-6	60.1	NA	0.648	0.491	0.539
ANO-2	TSQ002	53.0	1.55	1.55	1.086	1.115
Oconee	15309	49.5 to 49.9	2.14	2.14	1.60 to 1.72	1.53

5.2 Fuel Swelling

A comparison of measured and FRAPCON-3 calculated fuel pellet radial swelling is shown in Table 5.2 for ten rods, taken from test reactor and power reactor benchmark cases. The predicted versus measured radial swelling is plotted in Figure 5.1. The code consistently overpredicts the measured net swelling for the BR-3 rod samples, but not for the other rods. The overall degree of overprediction (averaged over all the examples) is less than 0.3 mils. The model is compared to a much larger data base in Volume 1 of this report.

Table 5.2. Measured and Predicted Fuel Pellet Swelling

Reactor	Test Rod	Sample Burnup, Gwd/MTU	As-Fabrication Density, %TD	Measured Density @EOL, %TD	Estimated Measured Change in Radius mils (see note ^(a))	FRAPCON-3 Net Radius Change mils (see note ^(b))
ANO-2	TSQ002	57.4	95.27	92.74	1.44	1.37
ANO-2	TSQ002	57.6	95.27	93.29	1.13	1.37
ANO-2	TSQ002	51.2	95.27	92.69	1.47	1.3
ANO-2	TSQ022	57.4	95.27	92.89	1.35	1.3
ANO-2	TSQ022	63.1	95.27	92.97	1.30	1.4
Oconee	15309	53.0	95.8	94.53	0.83	1.25
Oconee	15309	52.0	95.8	94.34	0.96	1.25
Oconee	15189	53.6	95.8	93.69	1.35	1.25
Oconee	15335	54.5	95.80	94.09	1.09	1.25
TVO-1	H8/36-6	52.0	95.50	92.70	1.91	1.84
BR-3	24-I-6	68.0	94.77	92.24	1.63	2.3
BR-3	24-I-6	48.1	94.77	93.25	0.98	1.4
BR-3	24-I-6	48.8	94.77	93.7	0.69	1.4
BR-3	111-I-5	53.4	94.77	92.7	1.33	1.6
BR-3	111-I-5	43.1	94.77	93.52	0.80	1.2
BR-3	111-I-5	40.3	94.77	93.52	0.80	1.1
BR-3	36-I-8	69.8	94.77	91.19	2.30	2.14
BR-3	36-I-8	50.8	94.77	93.15	1.04	1.43
BR-3	36-I-8	49.6	94.77	93.15	1.04	1.43
Halden	432r1	34.5	95.00	93.50	1.11	0.9
Halden	432r1	26.8	95.00	94.30	0.52	0.46
(a) Derived from measured density changes.						
(b) Derived from swelling minus densification.

6.0 Cladding Corrosion and Hydriding Assessment

6.1 Cladding Oxidation and Hydrogen Uptake

Four well-characterized fuel rods were selected to demonstrate the capability of FRAPCON-3 to accurately calculate fuel rod waterside oxidation and hydrogen concentration for high burnup. The cases selected include four full-length rods (Rod TSQ002 from ANO-2, Rod 15309 from Oconee, Rod A1 from Monticello bundle MTB99, and Rod H8/36-6 from TVO-1). The set includes both PWR and BWR fuel rods that are either standard Zircaloy-4 in PWRs or Zircaloy-2 in BWRs. (The current FRAPCON-3 modeling contains no provision for reduced oxidation due to niobium or low tin alloys.) The rod-average burnup levels achieved on these rods range from 45 to 53 GWd/MTU.

Both the cladding corrosion and hydrogen uptake models were revised for and incorporated in FRAPCON-3. The cladding waterside corrosion model is based on the uniform oxidation models developed for ESCORE (Fiero et al. 1987), which includes a standard expression for the pre-transition (cubic law) oxidation and a flux-enhanced linear post-transition oxidation. To correct the model and extend it to high burnup, the MATPRO (Hagrman et al. 1981) hydrogen uptake model, CHUPTK, was revised. The post-transition pickup fraction for PWR rods was increased to a constant 0.15, based on paired oxidation and hydrogen concentration measurements in cladding sections from PWR rods with nominal to high burnup.

Table 6.1 shows the measured and FRAPCON-3-calculated peak oxide-layer thickness and peak hydrogen concentration for two selected high burnup PWR rods. Table 6.2 shows the measured and FRAPCON-3-calculated peak oxide-layer thickness for the two selected high burnup BWR rods. The measured and predicted corrosion layer thicknesses as a function of axial position along the rod are shown for the two PWR rods in Figures 6.1 and 6.2. The comparisons indicate satisfactory capability in FRAPCON-3 to predict peak and axial variation in cladding waterside oxidation.

FRAPCON-3 calculated peak oxide-layer thickness and peak hydrogen concentrations are bracketed by the choice of crud-layer thickness for the PWR rods and are in good agreement for the two BWR rods. The purpose of these code-data comparisons is to demonstrate similar predictions as with stand-alone versions of the corrosion/hydriding models. The predictions with zero crud layer are consistent with the stand-alone versions (which had no added temperature rise for crud layer). The peak hydrogen content in the cladding is likewise bracketed for these two example PWR cases by the indicated choices on crud layer. The hydrogen pickup fraction is consistent with the data for both PWR cases. This fraction is derived from a large body of PWR cladding data (see Volume 1 of this report) and proved to be best-estimate; therefore no change to the model is recommended based on the comparisons here. The BWR peak corrosion values are fairly well matched by the FRAPCON-3 predictions, and these predictions are not as sensitive to the crud layer input because of the relatively lower heat fluxes and lower operating temperatures. The conclusion is that the modeling of waterside oxidation and hydrogen pick-up is sufficient in FRAPCON-3 for best estimate analyses.

Table 6.1. Measured and Calculated Oxidation and Hydrogen Concentration for High Burnup PWR Fuel Rods

Reactor	Rod	Burnup (GWd/MTU)	Peak Oxide Layer Thickness, μm		Peak Hydrogen Concentration, ppm		Peak Hydrogen Pickup Fraction	
			Measured	Calculated	Measured	Calculated	Measured	Calculated
ANO-2 0.2 mil crud no crud	TSQ002	53.0	53	47	367	347	0.14	0.15
				40		294		0.15
Oconee 0.2 mil crud no crud	15309	50.0	64	62	408	425	0.14	0.15
				54		370		0.15

Table 6.2. Measured and Calculated Oxidation for Two High Burnup BWR Fuel Rods

Reactor	Rod	Burnup, Gwd/MTU	Peak Oxide Layer Thickness, μm	
			Measured	Calculated
Monticello- 0.2 mil crud no crud	MTB99 Rod A1	45.0	25	
				27
				27
TVO-1 0.2 mil crud no crud	H8/36-6	51.4	12 to 28	19
				19

Figure 6.1. Measured and Predicted Corrosion Layer Thickness as a Function of Axial Position for Oconee 5-Cycle PWR Rod 15309

Figure 6.2. Measured and Predicted Corrosion Layer Thickness as a Function of Axial Position for ANO-2 5-Cycle PWR Rod TSQ002

7.0 Cladding Creep and Axial Growth

The cladding creep model was not altered from the version used in FRAPCON-2 (based on Ibrahim 1973, Fidleris 1973, and Ross-Ross and Hunt 1968 in-reactor creepdown data for Zircaloy-2) as described in Volume 2 of this report. Each fuel vendor has a different fabrication specification for the cladding, which results in differing creep characteristics. Therefore, it is not feasible to have a creep model that is accurate for all the different vendor claddings; but the FRAPCON-2 model predicts a representative creep behavior, as demonstrated below by comparison to creepdown data for rods from three different vendors.

The FRAPCON-2 axial growth model was replaced with the axial growth model from Franklin (Franklin 1982), which was demonstrated to be best-estimate against a large body of PWR data in Volume 1 of this report. In that document, we also showed that the Franklin model times 0.50 fits available BWR cladding growth data quite well. Four rod sets (2 PWR rods and 2 BWR rods) were selected to demonstrate the code predictions here for creepdown and rod axial growth.

7.1 Cladding Axial Growth

The measured and calculated axial growth for the two PWR rods (ANO-2 Rod TSQ002 and Oconee rod 15309) and two BWR rods (Monticello rod A-1 and TVO-1 rod H8/36-6) are shown in Table 7.1. The range of measured data for the TSQ002 rod and for the Oconee rod 15309 represent the range of data for all the rods measured in their respective assemblies; that is, 19 rods from the ANO-2 assembly D040 and 16 rods from Oconee assembly 1D45). There was a 15% relative variation in the burnups for the rods in ANO-2 assembly D040, which results in a range for the calculated rod growths, as shown in Table 7.1. In contrast, there was only a 1% variation in the EOL burnups for the 16 selected rods from the Oconee 1D45 assembly, resulting in a single value quoted for calculated rod growth. The code-data comparison is reasonably good; for the BWR cases, the code underpredicts the measured growth by 0.05 to 0.1% strain, but is closer for the PWR rods. These comparisons are provided to demonstrate that the axial growth model as programmed into FRAPCON-3 is consistent with the predictions of the stand-alone version of the model described in Volume 1 of this report.

Table 7.1. Measured and Calculated Cladding Axial Growth

Reactor	Rod/Assembly Identity	Rod-Average Burnup, GWd/MTU	Measured Rod Growth, %	Predicted Rod growth, %
ANO-2 PWR	TSQ002/D040	53.0	0.83 to 1.11	0.85 to 0.96
Oconee PWR	15309/1D45	50.0	0.792 to 0.907	0.79
Monticello BWR	A1	45.0	0.515	0.39
TVO-1	H8/36-6	51.4	0.30	0.25

7.2 Cladding Creepdown

Two well characterized PWR rod sets and one BWR rod were used to demonstrate the cladding creepdown calculated by FRAPCON-3. The rod-average burnup ranged from 45 to 53 GWd/MTU. The measured and predicted values are shown in Table 7.2. The comparison between FRAPCON-3 predictions and the more extensive set of ANO-2 and Oconee rods is shown in Figures 7.1 and 7.2, respectively. The 14 rods from ANO-2 came from the same assembly, and the calculated creepdown at a given burnup varied by less than 0.05% strain between the rods with highest/lowest LHGR; therefore these calculated results are represented by a single value in Table 7.1. The same is true for the Oconee rods, except that the variation in LHGR between rods was even less. These code-data comparisons do not show a definite trend toward over or under prediction of cladding creepdown. The extent of creepdown is underpredicted for the Oconee rods and for the one Monticello BWR rod, but slightly overpredicted for the ANO-2 rods. It is concluded that the creepdown model is acceptable. Users of this creep model are encouraged to independently verify that this creep model is applicable to a vendors fuel rod cladding for which analyses are being performed.

Table 7.2. Measured and Predicted Rod-Average Cladding Creepdown

Reactor	Rod/Assembly Identity	Rod-Average Burnup, GWd/MTU	Measured Cladding Diameter Reduction Due to Creepdown at EOL, %	Predicted Cladding Diameter Reduction Due to Creepdown at EOL, %
ANO-2 PWR	TSQ002/D040	53.0	-0.6 to -0.85 ^(a)	-0.90 ^(a)
Oconee PWR	15309/1D45	50.0	-0.6 to -1.0 ^(b)	-0.6 to -0.7 ^(b)
Monticello BWR	A1	45.0	-0.37	-0.27
(a) Averaged over 24 inches to 116 inches from rod bottom.				
(b) Averaged over entire rod length.

Figure 7.1. Predicted and Measured Cladding Creepdown from the 2nd and 3rd Cycle Rods in the ANO-2 PWR

Figure 7.2. Predicted and Measured Cladding Creepdown from 3rd, 4th, and 5th Cycle Rods in the Oconee PWR

8.0 Comparison to Independent Data for Fuel Temperature and Fission Gas Release

The code-data comparisons in the previous sections, particularly in the cases of FGR and fuel temperatures, compare code predictions to some of the data used for benchmarking the models, i.e., data used to select parameter values within the models. Thus, for FGR and fuel temperatures and particularly at high burnup, it is important to also compare the code against "independent" data sets, which were not used to tune the models. A number of Halden and RISØ instrumented rods were found for this purpose, as described below.

8.1 Description of the Independent Data Sets

Three groups of independent data were examined:

1. BOL fuel center thermocouple data: These come from helium- and xenon-filled Halden test rods with BWR (8 x 8) radial dimensions and varying gap sizes. These are partially described as Case 3 of the International Atomic Energy Agency's (IAEA's) "FUMEX" data set (Chantoin et al. 1997a) and discussed by Wiesenack 1996 (total of 6 rods).
2. Fuel Temperatures at nominal-to-high burnups. These include one rod refabricated from a section of a commercial BWR rod (67 GWd/MTU section burnup) instrumented and power-ramped in the Halden reactor;^(a) one BWR-sized helium-filled test rod irradiated in Halden to 39 GWd/MTU and then power-ramped (FUMEX Case 4A) (Chantoin et al. 1997a); and two rod segments that were base-irradiated in a U.S. BWR to 22 and 43 GWd/MTU and then instrumented and power-ramped in the DR-2 reactor (RISØ, Denmark) as part of the Third RISØ Fission Gas Release Project (Knudsen et al. 1993; Chantoin et al. 1997b).
3. FGR at nominal-to-high burnup: These include three RISØ-III rods (the two described above plus a third (similar) rod at 42 Gwd/MTU) (Knudsen et al. 1993; Chantoin et al. 1997b); two PWR-type test rods irradiated in the Halden reactor to a total of 49 GWd/MTU, including several months at elevated LHGR (FUMEX Cases 6s and 6f) (Chantoin et al. 1997a); and two rod segments manufactured by Babcock and Wilcox (now Framatome), irradiated in the ANO-1 PWR to 62 GWd/MTU and then power-ramped in the R-2 Reactor (Studsvik, Sweden) (Wesley et al. 1994).

Rod identifications and major design and operating parameters for these test rods are listed in Tables 8.1, 8.2, and 8.3 for the three groups described above. These rods span the ranges of gap size, fill gas composition, LHGR, and burnups of interest.

(a) Halden Reactor Project. 1997. USNRC Private Communication.

Table 8.1. Independent Data for BOL Fuel Temperatures (all BWR-size rods in Halden Reactor)

Gap Size, microns (Gap-to-Diameter Ratio, %)	Initial Fill Gas Type and Room-Temperature Pressure, psia (Mpa)	Maximum Rod-Average LHGR, kW/ft (kWm)
50 (0.47)	He, 14.7 (0.10)	9 (30)
100 (0.94)	He, 14.7 (0.10)	9 (30)
200 (1.9)	He, 14.7 (0.10)	9 (30)
50 (0.47)	Xe, 14.7 (0.10)	9 (30)
100 (0.94)	Xe, 14.7 (0.10)	9 (30)
200 (1.9)	Xe, 14.7 (0.10)	9 (30)

Table 8.2. Independent Data for Fuel Temperatures at Nominal-to-High Burnup

Reactor and Type (and Reactor for Ramping)	Rod-Average Burnup, Gwd/MTU	Rod Identification	Diametral Gap Size, microns (and Gap-to-Diameter ratio, %)	Initial Fill Gas Type and Room-Temperature Pressure, psi (Mpa)	Maximum Rod-Average LHGR, kW/ft (kW/m)
Ringals BWR (Halden)	67	Halden, Rod 2	265 (2.5)	He, 73 (0.50)	7.6 (25)
Quad Cities BWR (DR-2)	43, 22	GE-2, GE-4 from RISØ-III	225 (2.1)	He, 97 (0.66)	12.6 (41)
Halden (Halden)	39	FUMEX-4A	220 (2.1)	He, 44 (0.30)	15.7 (52)

Table 8.3. Independent Data for FGR at Nominal-to-High Burnup

Reactor and Type (and Reactor for Ramping)	Rod-Average Burnup, Gwd/MTU	Rod Identification	Diametral Gap Size, microns (Gap-to-Diameter Ratio, %)	Initial Fill Gas Type and Room-Temperature Pressure, psi (Mpa)	Maximum (Bump Terminal) LHGR, kW/ft (kW/m)	Hold Time, Hours	Measured FGR, %
Quad Cities BWR (DR-2)	43	GE-2 from RISØ-III	225 [2.1]	He, 97 (0.66)	12.6 (40.5)	41	24.6
Quad Cities (DR-2)	22	GE-4 from RISØ-III	225 [2.1]	He, 97 (0.66)	13.2 (43.3)	34	27.0
Quad Cities (DR-2)	42	GE-6 from RISØ-III	225 (2.1)	He, 97 (0.66)	11.6 (37.9)	140	26.0
Halden (Halden)	48.8	FUMEX Case 6s	260	He, 370 (2.5)	~15 (50)	83 days	50
Halden (Halden)	48.8	FUMEX Case 6f	260	He, 370 (2.5)	~12 (40)	150 days	45
ANO-1 (Studsvik)	62.3	R1	188 (2.0)	He, 400 (2.7)	12.0 (39.5)	12	9.3
ANO-1 (Studsvik)	62.3	R3	188 [2.0]	He, 400 (2.7)	12.9 (44)	12	11.2

8.2 Results of Code-Data Comparisons

The code data comparisons are discussed below for each of the three groups of data.

8.2.1 BOL Fuel Temperatures

The predicted temperatures minus the measured temperatures for the independent data sets are plotted as a function of local LHGR and compared to those for the benchmark data in Figure 8.1. The independent data comparisons (solid symbols) plotted at 20 kW/m and 30 kW/m show more scatter (particularly for the xenon-filled rods), but are in most cases within 70°C or less, which is considered to be very good. The standard error at LHGRs greater than 20 kW/m is 37.7 °C for both helium and xenon filled rods and reduces to 30.5 °C for only helium filled rods.

Figure 8.1. FRAPCON-3 Predicted-Minus-Measured Centerline Fuel Temperature at BOL as a Function of LHGR for Both Independent and Benchmark Data

8.2.2 Fuel Temperatures at Nominal-to-High Burnup

The predicted temperatures minus the measured temperatures are plotted versus burnup for both the independent and benchmark data sets in Figure 8.2. These absolute temperature differences are generally much less than 100°C for both data sets. A tendency for underprediction at burnups greater than 45 GWd/MTU is evident. There are large variations in the LHGRs associated with the various points on this plot, and this makes it difficult to assess relative degree of under/overprediction from this kind of plot. Therefore, an alternative presentation is also made.

Figure 8.2. Predicted-Minus-Measured Fuel Center Temperatures as a Function of Burnup for Benchmark Cases and Several (Independent) Data Sets Described in the Text

The ratio of the predicted temperature minus the measured temperature to the measured difference between centerline and coolant temperatures is plotted versus rod-average burnup in Figure 8.3 for both the independent data sets and the benchmark data. The code-data comparison is presented in this way to assess the accuracy of the prediction of total thermal resistance from coolant to centerline, thus normalizing out the differences in LHGR. The two RISØ-III rods from the independent data are significantly underpredicted by 20 to 25% at burnups of 22 and 42 GWd/MTU. The reason for the underprediction is unknown at this time and is inconsistent with the good predictions of the Halden and FUMEX rods included in this figure. With the exception of the two RISØ-III rods, the combined data show a scatter of about 15% relative in this ratio, and the data are underpredicted above 45 GWd/MTU burnup by about 7 to 18%. However, it should be pointed out that all of the data (independent and benchmark) above that burnup come from only two fuel rods so that additional thermal data are needed at high burnups to verify this underprediction.

8.2.3 FGR at Nominal-to-High Burnup

The predicted minus the measured FGR differences are plotted for the independent and benchmark data in Figure 8.4. With the exception of one RISØ-III rod, the independent FGR data are predicted within 5% release, which is equivalent to the prediction with the steady-state benchmark data. This is better than the prediction of the benchmark power-ramped data.

Figure 8.3. Ratio of FRAPCON-3 Predicted-Minus-Measured Temperature Divided by Measured Fuel Centerline-Minus-Coolant Temperatures as a Function of Fuel Burnup

Figure 8.4. Predicted-Minus-Measured FGR as a Function of Burnup for Benchmark Steady-State/Power-Ramp Cases and Several Independent Cases Described in the Text

In summary, FRAPCON-3 predictions of the independent temperature data are consistent with the predictions of the benchmark data with the exception of the two RISØ-III rods, which were significantly underpredicted by 20 to 25%. The FRAPCON-3 predictions of the independent FGR data are as good or better than the comparisons to the benchmark data.

9.0 Conclusions

The FRAPCON-3 steady-state fuel performance code has been assessed against a set of pre-selected data from 45 well characterized fuel rods (includes benchmark and independent sets) where 41 rods were from test reactor irradiations (some test rods were base irradiated in commercial reactors) and the remaining 4 rods from commercial reactor irradiations. The data used for the assessment consisted of measurements of thermal (fuel temperature), FGR, rod internal void volume, cladding corrosion and hydriding, and cladding creep. The fuel rods represent a range of design parameters including different fuel rod diameters, lengths, gap sizes, and fill-gas compositions and a wide range of operating conditions with peak LHGRs varying from 8 to 18 kW/ft, rod-average burnups up to 76 GWd/MTU, and FGRs between less than 1% and greater than 30%. The estimates of code thermal and FGR predictive error are based on code comparisons to both the benchmark and independent data sets.

Thermal: The comparisons to centerline temperature data are divided into predictions of BOL data and temperature data as a function of burnup. The BOL predictions have been compared to centerline temperature measurements from instrumented rods irradiated in the Halden test reactor that are part of both the benchmark and independent data sets. The code comparisons to the BOL data at LHGRS > 20 kW/m show a negligible underpredictive bias on average of -3°C and relatively small standard error of 30.5°C in the prediction of fuel centerline temperature for rods with only helium fill gas with the standard error increasing to 37.7°C when xenon filled rods are included. With the exception of the two RISØ rods, the code comparisons to the temperature data as a function of burnup show approximately no bias in the prediction out to a rod average burnup of 40 GWd/MTU with a standard error of 40°C (104°F). The reason for the significant underprediction of the two RISØ rods is unknown. The code underpredicts the two rods above 45 GWd/MTU, Rod 18 from the HUHB assembly, and the ramped Halden rod at 67 GWd/MTU by up to 18%, which are the only rods with measured temperature data above this burnup level.

Fission Gas Release: The comparisons to FGR data are divided into predictions of steady-state data (with steady-state power histories) and those data from fuel rods with power bumps (increase in rod power) at the EOL to simulate operational overpower transients. The predictions of FGR for the experimental fuel rods with steady-state and ramped power histories is very good with a standard error of 5.4% release if the two BWR commercial and the two HBEP ramped fuel rods with high FGR are excluded, and the standard error increases to 8.8% release when these rods are included. The significant underprediction and the increased error introduced by the commercial rods is hypothesized to be due to power uncertainties in commercial reactors caused by control-rod or control-blade movements. The significant underprediction of the HBEP ramped rods is hypothesized to be due to the unstable nature of this fuel that is atypical of today's fuel designs. Therefore, the code predictions of FGR are relatively good if rod powers are known accurately and the fuel is stable (low-densifying).

Internal Void Volume: Comparisons were made to data from four commercial reactor and three test reactor fuel rods. The code predicted the two commercial rods well but underpredicted the BR-3 test rod data by approximately 8.7% (relative) on average.

Cladding Corrosion and Hydriding: Comparisons were made to data from two commercial PWR rods and two commercial BWR rods. The oxide corrosion predictions were very good and tend to bracket the data, depending on the choice of crud layer thickness. The hydrogen concentration was similarly bracketed.

Cladding Creep: Comparisons were made to data from several commercial PWR rods from two fuel assemblies from two different vendors and one BWR commercial rod. The code predictions compare relatively well to the maximum creepdown values measured for the PWR rods and an underprediction of the BWR rod by 0.1% strain (absolute). These predictions are considered to be relatively good based on the usual large scatter in cladding creepdown data, particularly when data are from different fuel vendors. The comparisons to the PWR rod axial growth data were also relatively good while the two BWR rods were underpredicted by 25% (relative).

10.0 References

- Balfour, M. G. 1982. *BR-3 High burnup Fuel Rod Hot Cell Program, Final Report, Vol. 1*. WCAP 10238, DOE/ET/34073-1, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania.
- Balfour, M. G., W. C. Chubb, and R. F. Boyle. 1982. *BR-3 High Burnup Fuel Rod Hot Cell Program Vol. 2: Data Summary*, WCAP 10238, DOE/ET 34073-2, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania.
- Barner, J. O., M. E. Cunningham, M. D. Freshley, and D. D. Lanning. 1990. *High Burn up Effects Program Summary Report*, DOE/NE/3406-1, HBEP-61, Battelle, Pacific Northwest Laboratories, Richland, Washington.
- Baumgartner, J. A. 1984. *BWR Fuel Bundle Extended Burnup Program Technical Progress Report, January 1983 to December 1983*, DOE/ET/34031-17, GEAP-30643, General Electric Company, San Jose, California.
- Berna, G. A., M. P. Bohn, W. N. Rausch, R. W. Williford, and D. D. Lanning. 1981. *FRAPCON-2: A Computer Code for the Calculation of Steady State Thermal-Mechanical Behavior of Oxide Fuel Rods*, NUREG/CR-1845, prepared for the U.S. Nuclear Regulatory Commission by Pacific Northwest Laboratory, Richland, Washington.
- Berna, G. A., C. E. Beyer, K. L. Davis, and D. D. Lanning. 1997. *FRAPCON-3: A Computer Code for the Calculation of Steady-State, Thermal-Mechanical Behavior of Oxide Fuel Rods for High Burnup*, NUREG/CR-6534, PNNL-11513, Vol. 2, prepared for the U.S. Nuclear Regulatory Commission by Pacific Northwest Laboratory, Richland, Washington.
- Bradley, E. R., M. E. Cunningham, D. D. Lanning, and R. E. Williford. 1981. *Data Report for the Instrumented Fuel Assembly IFA-513*, NUREG/CR-1838, PNL-3637, Pacific Northwest Laboratory, Richland, Washington.
- Chantoin, P. M., J. A. Turnbull, and W. Wiesenack. 1997a. "How Good is Fuel Modeling at Extended Burnup ? - The IAEA's FUMEX Programme Provides Some Answers." *Nuclear Engineering International*, September, 1997.
- Chantoin, P. M., E. Sartori and J. A. Turnbull. 1997b. "The Compilation of a Public Domain Database on Nuclear Fuel Performance for the Purpose of Code Development and Validation," *ANS/ENS 1997 International Topical Meeting on Light Water Reactor Fuel Performance, Portland Oregon*, p. 515. American Nuclear Society.
- Cunningham, M. E., and C. E. Beyer. 1984. *GT2R2: An Updated Version of GAPCON-THERMAL-2*, NUREG/CR-3907, PNL-5178, Pacific Northwest Laboratory, Richland, Washington.

Djurle, S. 1985. *Final Report of the Super-Ramp Project*, DOE/ET/34032-1, U.S. Department of Energy, Washington, D.C.

Fidleris, V. 1976. "Summary of Experimental Results on In-Reactor Creep and Irradiation Growth of Zirconium Alloys." *Atomic Energy Review*, Vol.13, No.1, pp.51-80

Fiero, I. B., M. A. Krammen, and H. R. Freeburn. 1987. *ESCORE - the Steady-State Core Reload Evaluator Code: General Description*, EPRI-NP-5100, Projects 2061-6, -13 Final Report. Prepared for Electric Power Research Institute, Palo Alto, California.

Forsberg, K. and A. R. Massih. 1985. "Diffusion Theory of Fission Gas Migration in Irradiated Nuclear Fuel UO_2 ," *J. of Nucl. Mater.*, Vol.135, pp.140-148.

Franklin, D. G. 1982. "Zircaloy Cladding Deformation during Power Reactor Irradiation." In *Proceedings of the Fifth International Symposium on Zirconium in the Nuclear Industry*, ASTM-STP-754, pp. 235-267.

Hagman, D. L., G. A. Reymann, and G. E. Mason. 1981. *A Handbook of Materials Properties for Use in the Analysis of Light Water Reactor Fuel Rod Behavior*. MATPRO Version 11 (Revision 2). NUREG/CR-0479 (TREE-1280), prepared by EG&G Idaho, Inc., Idaho Falls, Idaho for the U.S. Nuclear Regulatory Commission, Washington, D.C.

Ibrahim, E. F. 1973. "In-Reactor Tubular Creep of Zircaloy-2 at 260 to 300 °C." *J. of Nucl. Mater.*, Vol.46, pp. 169-182.

Janvier, J-C., B. de Bernaday de Sigoyer, and R. Delmas. 1967. *Irradiation of Uranium Oxide in Strong Cladding Effect of Initial Diametral Gap on Overall Behavior*, Program CC-7, 1st and 2nd Sections, CEA-R-3358, Commissariat a l'Energie Atomique, Paris, France.

Knudsen, P., C. Bagger, H. Carlsen, I Misfeldt, and M. Mogensen. 1983. *Riso Fission Gas Release Project Final Report*, DOE/ET/34033-1, U.S. Department of Energy, Washington, D.C.

Knudsen, P., C. Bagger, M. Mogensen, and H. Toftegaard. 1993. "Fission Gas Release and Fuel Temperature During Power Transients in Water Reactor Fuel at Extended Burnup." *Proceedings of a Technical Committee Meeting held in Pembroke, Ontario, Canada. 28 April - 1 May 1992*. IAEA-TECDOC-697. International Atomic Energy Agency, Vienna, Austria.

Lanning, D. D. 1986. *Irradiation History and Final Post-Irradiation Data for IFA-432*, NUREG/CR-4717, PNL-5977, Pacific Northwest Laboratory, Richland, Washington.

Lanning, D. D., and C. E. Beyer. 1997. *Modification to NRC Fuel Rod Material Properties and Performance Models Due to High Burnup*, NUREG/CR-6534 Vol.1, PNNL-11513 Vol. 1, Pacific Northwest National Laboratory, Richland, Washington.

Newman, L. W. 1986. *The Hot Cell Examination of Oconee-1 Fuel Rods after Five Cycles of Irradiation*, DOE/ET/34212-50, BAW-1874, Babcock and Wilcox Company, Lynchburg, Virginia.

Notley, M.J.F., and J. R. MacEwan. 1965. *The Effect of UO_2 Density on Fission Product Gas Release and Sheath Expansion*, AECL-2230, Atomic Energy of Canada, Limited, Chalk River, Canada.

Notley, M.J.F., R. DesHais, and J. R. MacEwan. 1967. *Measurements of the Fission Product Gas Pressures Developed in UO_2 Fuel Elements During Operation*, AECL-2662, Atomic Energy of Canada, Limited, Chalk River, Canada.

Ross-Ross, P. A. and C.E.L. Hunt. 1968. "The In-Reactor Creep of Cold-Worked Zircaloy-2 and Zirconium -2.5 Wt% Niobium Pressure Tubes." *J. of Nucl. Mater.*, Vol.26, pp.2-17.

Smith, Jr., G. P., R. C. Pirek, H. R. Freeburn, and D. Schrire. 1994. *The Evaluation and Demonstration of Methods for Improved Nuclear Fuel Utilization*, DOE/ET/34013-15, Combustion Engineering, Windsor, Connecticut.

Wesley, D. A., K. Mori, and S. Inoue. 1994. "Mark BEB Ramp Testing Program" presented in the proceedings of *The 1994 ANS/ENS International Topical Meeting on Light Water Reactor Fuel Performance, West Palm Beach, Florida*. p. 343. American Nuclear Society.

Wiesenack, W. 1992. "Experimental Techniques and Results Related to High Burn-Up Investigations at the OECD Halden Reactor Project," *Proceedings of a Technical Committee Meeting held in Pembroke, Ontario, Canada, April 28 - May 1, 1992*, IAEA-TECDOC-697, p. 118.

Wiesenack, W. 1996. "Review of Halden Reactor Project High Burnup Fuel Data that can be Used in Safety Analyses" presented at *NRC-Research 23rd Water Reactor Safety Meeting*, NUREG/CR-0145, Vol. 1, p. 127.

Appendix A

Supplement

Appendix A

Supplement

A.1 Description of FRAPCON Case Input for Halden Ultra High Burnup (HUHB), Rod 18

A.1.1 Overview of the HUHB Test

The HUHB test fuel assembly was initiated by the Halden Reactor project to demonstrate the effect of burnup on fuel thermal conductivity. The HUHB configuration of the assembly consisted of six rods, four of which were instrumented with centerline expansion thermometers and two with pressure transducers. The rods have been under irradiation in the Halden Reactor, Norway, from September 1989 to the present (1997). Documented data for fuel center temperatures and linear heat ratings are available through early 1996, to a rod-average burnup of 76 MWd/MTU (Halden 1997).^(a)

Further details are provided below on the following topics: rod design and fabrication, irradiation conditions and history, and major postirradiation examination (PIE) results. The final sections describe the recommended FRAPCON-3 input for Rod 18 and the code-data comparison for fuel center temperature and fission gas release for this rod and its siblings.

A.1.2 Test Rod Design and Fabrication

The general design specifications are shown in Table A1.1. All six rods had the same gap size, i.e., 100 microns, and all rods contained 94% theoretical density (TD) dense, stable, flat-ended sintered annular UO₂ fuel pellets. The diameter of the inner annulus is significant (2 mm) and the diameter of the rods is greatly reduced from that of normal light water reactor rods to achieve lower absolute temperatures and better definition of the thermal conductivity degradation effect.

(a) Halden Reactor Project. 1997. Personal communication with USNRC.

Table A1.1. Fuel Components and Specifications for HUHB

Fuel Rod	
Overall length (in.)	Not specified
Diameter (in.)	0.2762
Fuel stack height (in.)	17.44
Nominal plenum chamber length (in.)	1.28
Number of pellets per rod	59
Fuel/clad diametral gaps (mils)	3.937
Fill gas composition	He
Fill gas pressure (atm)	9.9
Cladding	
Material	Zircaloy-2
Diameter outside (in.)	0.2762
Diameter inside (in.)	0.2368
Clad wall thickness (in.)	0.0197
Ovality	Not specified
Eccentricity	Not specified
Grain size	Not specified
Fuel	
Material	UO ₂
Enrichment (%)	13
Pellet sintered density (% TD)	94
Pellet diameter (in.)	0.2329
Pellet inner diameter (in.)	0.0787 for rods 15, 16, 17, and 18
Pellet length (in.)	0.30
Pellet geometry	Flat end
Grain size (μm)	5 - 30

The rod and assembly instrumentation was different from IFA-432. Four rods (numbers 15, 16, 17, and 18) contained "expansion centerline thermometers." These are tungsten (1.8% ZrO₂) rods that run the full length of the rod on the inside of the pellets and gage the average center temperature of each rod via thermal expansion of the rod detected by resistance change. Two rods (numbers 13 and 14) each contained a pressure transducer for measuring rod internal pressure. The assembly instrumentation included four self-powered neutron detectors (SPNDs), three of which were located coplanar at the top of

the assembly and one near the bottom to define the thermal neutron flux distribution within the assembly.

The behavior of LHGR and measured temperatures were very similar for all four rods with temperature sensors. One rod (number 18) was selected for comparison to FRAPCON-3 predictions.

A.1.3 Test Rod Irradiation History

The Halden Reactor is a boiling heavy water reactor operating at nominal coolant conditions of 500 psi pressure, 459°F inlet temperature, and 464°F saturation temperature. The nominal operating conditions are summarized in Table A1.2. Note that the neutron flux in this heavy water reactor is well thermalized, and the fast neutron flux is significantly less than in a light water power reactor.

Table A1.2. Nominal Halden Reactor Operating Conditions

Parameter and Units	Value
Power level (MW)	12
Reactor pressure (psig)	500
Heavy water saturation temperature (°F)	464
Plenum inlet temperature (°F)	459
Fast flux (> 1 MeV) (N/m ² /s per W/g fuel)	$\sim 5 \times 10^{15}$
Thermal flux (N/m ² /s per W/g fuel)	$\sim 2 \times 10^{16}$

Published temperature and LHGR data extend to a burnup of 76 GWd/MTU. A plot of the rod-average LHGR is shown in Figure A1.1a for Rod 18, with the corresponding (rod-average) measured center temperatures in Figure A1.1b. The axial power profiles are given in Figure A1.2. All six rods experienced similar power and temperature histories, except the central temperatures in Rods 13 and 14 were higher than their sibling rods because these rods contained solid pellets (no inner annulus). The pressure transducer readings from one of these solid pellet rods (number 13) indicates that almost no fission gas was released in that rod (and hence all the rods) during the irradiation period.

A.1.4 FRAPCON-3 Input for Rod 18

The recommended FRAPCON-3 input for Rod 18 appears in Table A1.3. The rod design values in Table A1.2 and the input power history, shown in Figure A1.1a, was provided by the Halden Project.^(a)

(a) W. Wiesenack, Personal communications to D. D. Lanning (PNNL) July 1996.

Figure A1.1. (a) Rod Average LHGR for HUHB, Rod 18, and
(b) Rod Average Temperature Data for HUHB, Rod 18

Figure A1.2. HUSB Axial Profiles as a Function of Burnup

A.1.5 Code-to-Data Comparison for Fuel Centerline Temperature

The rod-average LHGR data were matched by input LHGR history (see Figure A1.1). The measured fuel center temperatures represent rod-average values, since they are deduced from the calibrated "expansion thermometer," which runs the full length of the fuel column. Therefore, calculated rod-average temperature is compared to the temperature data.

The agreement between measured and predicted temperatures is shown in Figure A1.3. This comparison, while fairly close, reveals a trend toward underprediction by the code as the burnup exceeds about 40 GWd/MTU. This trend could be rectified by increasing the "degradation factor" in the fuel thermal conductivity function. However, as discussed in Section 3.2.1, this would result in poorer overall predictions of fuel temperatures from Rod 3 of IFA-432.

For reference, in Figure A1.4, we show the calculated temperature vs. burnup where the code was modified to NOT include the effect of current thermal conductivity degradation as a function of burnup. The overprediction of temperatures in the latter case is taken as firm indication that thermal conductivity degradation is occurring. The FRAPCON-3 code-data comparison with conductivity degradation is shown in Figure A1.5.

Table A1.3. HUHB Rod 18 Input for FRAPCON-3 and Notes on Derivation

Input Variable	Meaning and Units	Input Value	Calculations and Notes	Reference
cpl	Cold plenum length (in.)	100.28	Deliberate large plenum volume input to mitigate FGR effects	(a)
crdt	Initial crud thickness (mils)	0.0	No crud typically seen on Halden test rods	(a)
dco	Cladding outer diameter (in.)	0.2762	Design value	(a)
thkcld	Cladding wall thickness (in.)	0.0197	Design value	(a)
pitch	Rod-to-rod pitch (in.)	0.56	Arbitrary value - boiling condition	
den	Fuel pellet density (% of theoretical)	94	Design value	(a)
dishsd	Dish shoulder width (in.)	--	Flat-ended pellets (hdish = 0.0)	--
thkgap	Gap radial thickens (in.)	0.002	Design value = 100 microns diametral gap	(a)
dspg	Spring diameter (in.)	0.236	Assumed	--
dspgw	Spring wire diameter (in.)	0.04	Assumed	--
enrch	Fuel enrichment (% ²³⁵ U in U)	13	Design value	(a)
fgpav	Fill gas pressure (psi at room temp.)	145	Stated beginning-of-life (BOL) value	(a)
hdish	Dish depth (in.)	0.0	Flat-ended pellets	--
hplt	Pellet length (in.)	0.295	Design value	(a)
icm	Cladding type (2 = zircaloy-2)	2	Design value	(a)
icor	Cladding crud buildup control (=0 for no buildup)	0	No further crud buildup	(a)
idxgas	Gas type (1 = helium)	1	Design value	(a)
iplant	Type of Plant	-4	Heavy Water Plant	
iq	Indicator for axial power shape	0		
jdlpr	Output Control	0		
totl	Fuel column length (ft)	1.35	Design value	(a)
jn	Number of points per axial power shape	5,5,5		
jst	Assignment of shape numbers to each time step	12 × 1 40 × 2 102 × 3		
rc	Inner pellet radius	0.0394		
roughc	Cladding surface roughness (in.)	1.97E-5	Assumed	--
roughf	Pellet surface roughness (in.)	2.36E-5	Assumed	--

Table A1.3. (contd)

Input Variable	Meaning and Units	Input Value	Calculations and Notes	Reference
rsntr	Change in pellet density due to densification (kg/m ³)	150	Stable pellets	(a)
vs	Number of plenum spring turns	8.0	Assumed	--
flux	Fast neutrons (per m ² per second per W/g of fuel)	5E16	Halden reactor conditions	(a)
p2	Reactor system pressure (psi)	500	Halden reactor conditions	(a)
tw	Coolant inlet temperature (°F)	459.0	Halden reactor conditions	(a)
go	Coolant mass flow (lbm/ft ² /h)	0.0	Minimal temperature rises in a boiler	--
qf, x	Parameters of the axial flux shapes	(a)		(a)
qmpy, time	The LHGR history	(a)		(a)
(a) Halden Project personal communication.

Figure A1.3. Agreement Between Measured and Predicted Temperatures for HUIB, Rod 18

Figure A1.4. HUHB, Rod Calculated Temperatures Versus Burnup (where the code was modified to not include the effect of current thermal conductivity degradation as a function of burnup)

Figure A1.5. Temperature Difference with Degradation

The calculated fission gas release at end-of-life (EOL) for this case was 1.2%, and hence this pre-pressurized rod did not experience fill-gas thermal conductivity degradation. This prediction of minimal fission gas release is corroborated by gas pressure transducer readings on the identical sibling rods in the assembly, which indicated negligible gas release ($< 1\%$) throughout the irradiation period.^(a)

(a) Halden Project personal communication.

A.2 Description of FRAPCON-3 Case Input for Halden Rods 1, 2, and 3

A.2.1 Overview of the IFA-432 Test

The IFA-432 test was irradiated under a research program on fuel rod steady-state performance sponsored by the NRC from 1974 to 1986. The IFA-432 test assembly was a heavily instrumented six-rod assembly irradiated in the Halden heavy boiling water reactor, Norway, from 1975 to 1984. The purpose was to test the long-term steady-state performance of BWR-6 type fuel rods, operated at power levels that were at the upper bound for full-length commercial fuel rods. The fuel pellets were fabricated at PNNL and shipped to Norway; final rod and assembly fabrication was completed at the Halden site. Destructive examinations of selected rods were carried out at Harwell Laboratories, UK.

The assembly included six instrumented rods and three replaceable noninstrumented spares. Each instrumented fuel rod had a centerline thermocouple in both the top and the bottom end of the fuel column and a pressure transducer to monitor rod internal pressure. The assembly instrumentation included six vanadium SPNDs and one cobalt neutron detector, together with rod elongation sensors at each rod position, coolant thermocouples at the top and bottom of the assembly, and a coolant flow meter (turbine). The axial locations of the thermocouples and SPNDs relative to the axial thermal neutron flux profile are shown in Figure A2.1.

Figure A2.1. Arrangement of IFA-432 Thermocouples, Neutron Detectors and Fuel Column Relative to the Reference Axial Thermal Neutron Flux Profile

One instrumented rod (Rod 4; xenon-fill gas) was replaced by a noninstrumented rod in 1976. Two instrumented rods from the assembly were removed in 1981 at peak burnups of approximately 36 GWd/MTU and shipped to Harwell Laboratories, UK, for hot cell examinations. The remaining three instrumented rods and one noninstrumented spare rod continued irradiation until 1984 when they were discharged at final peak burnups of 44 to 45 GWd/MTU for the instrumented rod and ~18 GWd/MTU for the noninstrumented rod. These rods were also sent to Harwell for examination. The examinations included: rod axial gamma scanning and length measurements, rod puncture and fill gas analysis, retained gas analysis and fuel burnup determination, and fuel optical ceramography.

Further detail is provided below on the following topics: rod design and fabrication, irradiation conditions and history, and major PIE results. The final section describes the recommended FRAPCON-3 input for three selected cases for which LHGR and fuel center temperature measurements are available through-life: Rods 1, 2, and 3.

A.2.2 Test Rod Design and Fabrication

The test rods were designed to simulate BWR-6 rod cladding type and radial dimensions, with variations in fuel-cladding gap sizes, fuel types, and fill gas compositions. The general design specifications are given in Table A2.1. The fuel rod length was much shorter than full-length (~144-inch) commercial reactor rods to fit well within the short length of the Halden reactor core. Fuel rod overall length was 25 inches, with an active fuel column length of 22.8 inches. The overall void volume was held to 0.5 cubic inches (by selection of a ~1-inch plenum length at the upper end); this was done to approximate the ratio between fuel volume and void volume found in full-length rods. The cladding for all rods was Zircaloy-2.

The fuel and cladding radial dimensions, the fuel pellet type, and the fill gas compositions for the IFA-432 instrumented rods are summarized in Table A2.2. Rods 1, 2, and 3 all had typical high-density (95% TD) stable sintered UO_2 fuel pellets and helium fill gas at one atmosphere pressure; slight differences in the pellet diameters created variations in fuel-cladding gap size among the rods. Rod 4 had xenon fill gas and was specially designed to maintain the pellets surrounding the thermocouple concentric with the cladding in the upper end and eccentric in the lower end. Rod 5 had low-density fuel, stabilized by large pore size produced with a hydrocarbon pore-former. Rod 6 had low-density fuel made susceptible to in-reactor densification by small grain size and small pores.

Because Rods 4, 5, and 6 have experimental design variations that render them atypical of normal power reactor fuel, only Rods 1, 2, and 3 are used for FRAPCON-3 comparison runs.

Table A2.1. IFA-432 Test Rod General Design Specifications

Fuel Rod	
Overall length (in.)	25
Diameter (in.)	0.5035
Fuel stack height (in.)	22.8
Nominal plenum chamber length (in.)	1
Number of pellets per rod	45
Fuel/clad diametral gaps (mils) (microns)	3, 9, 15 (75, 230, 380)
Fill gas compositions	He for Rods 1, 2, 3, 5, 6 Xe for Rod 4
Fill gas pressure (atm)	1
Cladding	
Material	Zircaloy-2
Diameter outside (in.)	0.5035
Diameter inside (in.)	0.4295
Clad wall thickness (in.)	0.0370
Ovality	Not specified
Eccentricity	Not specified
Grain size	Not specified
Fuel	
Material	UO ₂
Enrichment (%)	10
Pellet sintered density (% TD)	95
Pellet outer diameter (in.)	
9 - mil gap	0.4204
15 - mil gap	0.4145
3 - mil gap	0.4265
Pellet inner diameter (in.)	0.069
Pellet length (in.)	0.50
Pellet geometry	Flat end
Grain size (μm)	22 - 77

Table A2.2. IFA-432 Test Rod Design Variations

Rod Number	Fill Gas	Diametral Gap (microns)	Fuel Density (% TD)	Special Features
1	He	230	94.8	Standard rod
2	He	380	94.5	Large gap
3	He	76	95.2	Small gap
4	Xe	230	94.9	Upper/lower TC location pinned to be concentric/eccentric
5	He	230	90.9	Low density fuel stabilized with pore former
6	He	230	91.6	Low density unstable fuel with respect to densification

A.2.3 IFA-432 Power Calibration and Irradiation History

The Halden Reactor is a boiling heavy water reactor operating at nominal coolant conditions of 500 psi pressure, 459°F inlet temperature, and 464°F saturation temperature. The nominal operating conditions are summarized in Table A2.3. Note that the neutron flux in this heavy water reactor is well thermalized, and the fast neutron flux is significantly less than in a light water power reactor.

Table A2.3. Nominal Halden Reactor Operating Conditions

Parameter and Units	Value
Power level (MW)	12
Reactor pressure (psig)	500
Heavy water saturation temperature (°F)	464
Plenum inlet temperature (°F)	459
Fast flux (> 1 MeV) ($\text{N}/\text{m}^2/\text{s}$ per W/g fuel)	$\sim 5 \times 10^{15}$
Thermal flux ($\text{N}/\text{m}^2/\text{s}$ per W/g fuel)	$\sim 2 \times 10^{16}$

Subcooled water can be piped through each test position. It was intended that the IFA-432 SPND output would be calibrated via coolant calorimetry during special "calibration runs" at power, during which sub-cooled water would be admitted into the test position channel, and coolant flow and temperature

rise would be measured. The spatial distribution of the SPND could then be coupled with their outputs to yield both total assembly power and accurate estimates of individual rod LHGR as a function of axial position throughout the irradiation.

Unfortunately, the valve used to admit the sub-cooled water failed to open, and the calibration could not be carried out as planned. However, the calibration was successful for the previously irradiated assembly IFA-431, which was identical in every detail to IFA-432. By comparing the fuel temperatures and corresponding SPND outputs between the two assemblies, especially for the small-gap Rod 3, it was possible to estimate the LHGRs in IFA-432 with limited uncertainty (estimated + 10% relative uncertainty bounds on LHGR at the 95% confidence level).

The LHGRs at the elevations of the upper and lower thermocouples for Rod 1 are shown as a function of operating time in Figure A2.2. The corresponding measured temperatures (taken from Lanning 1986) are shown in Figure A2.3. Note that the time increments from Lanning (1986) have been adjusted to achieve the correct EOL fuel burnup. The peak burnup was measured by radio chemistry Rod 1 as 3.89 atom%, which corresponds to 36.5 GWd/MTU at 200 MeV per fission.

Similar plots of LHGRs and fuel center temperatures are presented in Figures A2.4 and A2.5 for Rod 2 and Figures A2.6 and A2.7 for Rod 3. Peak measured burnup for Rod 3 was 4.70 atom% or 44 GWd/MTU. The estimated peak burnup for Rod 2 is 42 GWd/MTU, based on slightly lower LHGRs derived from SPND readings.

Figure A2.2. Rod 1 Measured LHGR Versus Time

Figure A2.3. Rod 1 Measured Temperature Versus Time

Figure A2.4. Rod 2 Measured LHGR Versus Time

Figure A2.5. Rod 2 Measured Temperature Versus Time

Figure A2.6. Rod 3 Measured LHGR Versus Time

Figure A2.7. Rod 3 Measured Temperature Versus Time

A.2.4 Postirradiation Examination Results

The rods were subjected to nondestructive examinations that included visual exam, length measurements, profilometry, and gamma scanning for ^{137}Cs and ^{95}Zr . Destructive examinations included rod puncture and fill gas analysis, sectioning and ceramography, and analysis of selected pellets for retained fission gas and for burnup. Major results are summarized below.

Pellet-pellet interface ridges were observed on all cladding tubes, with the largest belonging to the small-gap Rod 3. Length changes were small, but indicated pellet-cladding mechanical interaction occurred in all three rods. In the gamma scans, the cesium peaking at pellet-pellet interfaces was in qualitative agreement with the ranking of rods in terms of gap size and fuel temperature. That is, the small-gap, low-temperature Rod 3 displayed no cesium peaking; the large-gap, high-temperature Rod 2 displayed strong peaking along its full length; and Rod 1, with intermediate gap size and fuel temperatures, displayed moderate cesium peaking only in the (higher-powered) upper half.

The appearance of the fuel pellets in ceramography of transverse sections was similarly in qualitative agreement with the fuel temperatures. Rod 3 with the lowest fuel temperatures showed essentially no center restructuring. The upper end of Rod 1 showed moderate grain growth; and the upper end of Rod 2 showed well-developed restructuring, with columnar grain formation and a central void (in a solid pellet as-fabricated), which indicates center temperatures well over 1,700°C.

The puncture results on all three rods indicated leakage of fission gas from the rods before puncture. Estimates of the fission gas release were also made based on retained fission gas analysis of selected pellets. These estimates are listed in Table A2.4. Again, the release is in qualitative agreement with the fuel temperatures, with Rod 3 having the least and Rod 2 the most estimated fission gas release.

Table A2.4. Fission Gas Release Estimates for IFA-432 Rods 1, 2, and 3

Rod Number	As-Fabricated Diametral Gap Size (microns)	Estimated Fission Gas Release (%)	Predicted Fission Gas Release (%)
1	230	20 ± 10	29.0
2	380	30 ± 10	67.2
3	76	10 ± 10	16.1

A.2.5 FRAPCON-3 Input Recommendations

The recommended input for Rods 1, 2, and 3 of IFA-432 is provided in Table A2.5. Most of the FRAPCON-3 input for these rods can be made from the component dimensions and operational data given in this report. Note that in order to match code predictions directly with the thermocouple data, it is necessary to input the thermocouple well diameter (0.069 inch) as the fuel pellet inner radius.

In the recommended input files in Table A2.5, we chose to represent the axial power distribution with four equal-length axial regions, where the LHGRs for the top and bottom region are forced to match respectively the upper and lower thermocouple position LHGRs shown in Figures A2.3, A2.4, and A2.5. The user-input axial power shapes were chosen in conjunction with average-power input to accomplish this.

A.2.6 Code-Data Comparisons for Fuel Centerline Temperatures

Code-data temperature comparisons are discussed for each rod separately in the following subsections.

A2.6.1 Code-Data Fuel Temperature Comparisons for Rod 1

The match between measured and input LHGRs at the lower/upper locations are shown in Figure A2.8a,b. The measured and predicted fuel center temperatures for Rod 1 are plotted as a function of irradiation time in Figure A2.9a,b. From about 75 days onward, the measured temperatures are overpredicted by 100 to 150 K by the code. This is partially due to a probable overprediction of the fission gas release and partially to overprediction of the effective gap size.

The predicted lower/upper BOL temperatures as a function of LHGR are shown in Figure A2.10a and Figure A2.10b, respectively. These temperatures are well predicted by the code.

Table A2.5. IFA-432, Rods 1, 2, and 3 Input for FRAPCON-3 and Notes on Derivation

Input Variable	Meaning and Units	Input Value	Calculations and Notes	Reference
cpl	Cold plenum length (in.)	2.0	Design value	Hann et al. 1988
crdt	Initial crud thickness (mils)	0.0	No crud typically seen on Halden test rods	Hann et al. 1988
dco	Cladding outer diameter (in.)	0.5035	Design value	Hann et al. 1988
thkcld	Cladding wall thickness (in.)	0.037	Design value	Hann et al. 1988
pitch	Rod-to-rod pitch (in.)	0.56	Arbitrary value - boiling condition	
den	Fuel pellet density (% of theoretical)	95.5	Design value	Hann et al. 1988
dishsd	Dish shoulder width (in.)	--	Flat-ended pellets (hdish = 0.0)	Hann et al. 1988
thkgap	Gap radial thickens (in.)	0.0045 (R1) 0.0075 (R2) 0.0015 (R3)	Design value = 100 microns diametral gap	Hann et al. 1988
dspg	Spring diameter (in.)	0.35	Assumed	--
dspgw	Spring wire diameter (in.)	0.03	Assumed	--
enrch	Fuel enrichment (% ²³⁵ U in U)	10	Design value	Hann et al. 1988
fgpav	Fill gas pressure (psi at room temp.)	14.7	Stated BOL value	Hann et al. 1988
hdish	Dish depth (in.)	0.0	Flat-ended pellets	Hann et al. 1988
hplt	Pellet length (in.)	0.5	Nominal design value	Hann et al. 1988
icm	Cladding type (2 = zircaloy-2)	5	Design value	Hann et al. 1988
icor	Cladding crud buildup control (= 0 for no buildup)	0	No further crud buildup	Hann et al. 1988

Table A2.5. (contd)

Input Variable	Meaning and Units	Input Value	Calculations and Notes	Reference
iplant	Type of plant	-4	Heavy water plant	
totl	Fuel column length (ft)	1.9	Design value	Hann et al. 1988
roughc	Cladding surface roughness (in.)	2.5E-5	Measured	--
roughf	Pellet surface roughness (in.)	8.5E-5	Measured	--
rsntr	Change in pellet density due to densification (kg/m ³)	75	Stable pellets; resinter test results	Hann et al. 1988
vs	Number of plenum spring turns	5	Assumed	--
flux	Fast neutrons (per m ² per second per W/g of fuel)	0.18E16	Halden reactor conditions	Lanning and Bradley 1984
p2	Reactor system pressure (psi)	500	Halden reactor conditions	Lanning and Bradley 1984
tw	Coolant inlet temperature (°F)	464	Halden reactor conditions	Lanning and Bradley 1984
go	Coolant mass flow (lbm/ft ² /h)	0.0	Minimal temperature rises in a boiler	--
jn	Number of points per axial power shape	5,5,5,5,5,5,5,5		
jst	Assignment of shape numbers to each time step	1,1,1,1,1,1,6,3,3,2,2,1,4,5,5,6,3,3,2,2,5,2,5,2,2,1,4,5,7,7,7,8,4,4,4,4,4		
qf, x	Parameters of the axial flux shapes	Lanning and Bradley 1984		Lanning and Bradley 1984
qmpy, time	The LHGR history	Lanning and Bradley 1984	Derived from the plotted histories	Lanning and Bradley 1984

Figure A2.8. Rod 1 Match Between Input and Measured LHGR Versus Time (a) Upper TC Position, and (b) Lower TC Position

(a)

— Data - + - FRAPCON-3

(b)

— Data - + - FRAPCON-3

Figure A2.9. Rod 1 Match Between Predicted and Measured Temperature Versus Time, (a) Upper TC Position, and (b) Lower TC Position

Figure A2.10. Rod 1 Match Between Predicted and Measured BOL Temperature Versus Time, (a) Upper TC, and (b) Lower TC

A2.6.2 Code-Data Fuel Temperature Comparisons for Rod 2

The match between measured and input LHGRs is demonstrated in Figure A2.11a,b. The measured and predicted fuel center temperatures for Rod 2 are plotted as a function of irradiation time in Figure A2.12 (lower thermocouple). There are no upper thermocouple position data because a gamma thermometer that failed to operate occupied this position on this rod. Beginning at 50 days, strong thermal feedback is predicted, which leads to predicted complete saturation of the plenum gas with fission gas and a sharp rise in predicted fuel temperatures, far above the measured values. Fuel thermal expansion and gap closure finally result in stabilized temperatures in the 2000°C range. These temperatures are predicted to continue at this high level, about 500°C above the measured values, throughout the irradiation period. Because of the exaggerated thermal feedback effect in this rod caused by the abnormally large gap size, the lack of prepressurization, and the high LHGRs, this overprediction for Rod 2 is not considered representative of the code accuracy for nominal-gap pressurized modern LWR fuel.

The predicted and measured (lower position) BOL fuel temperatures are plotted as a function of LHGR in Figure A2.13. These temperatures are well predicted by the code.

A2.6.3 Code-Data Fuel Temperature Comparisons for Rod 3

The match between input and measured LHGRs is demonstrated in Figure A2.14a,b. The measured and predicted fuel center temperatures for Rod 3 are plotted as a function of irradiation time in Figure A2.15a,b at the lower and upper thermocouple. The temperatures through-life are well matched by the predictions, out to about 400 days for the lower thermocouple and 200 days for the upper thermocouple. Beyond those points, the temperatures are overpredicted by about 100°C. This is probably due, however, to overprediction of the fission gas release, which is 16% at EOL, at the upper end of the estimated range for the EOL FGR. A second prediction of the lower thermocouple temperatures, with no fission gas release permitted throughout the irradiation, is also shown in Figure A2.10a. The true FGR lies somewhere between these extremes.

As discussed in Section 3 of the text, these data provide an estimate of the thermal conductivity degradation, but it is an uncertain estimate due to the uncertainty in the FGR for this rod.

The BOL predicted and measured lower/upper temperatures are plotted as a function of LHGR in Figure A2.16a,b. These temperatures are well predicted by the code.

A2.6.4 Fission Gas Release Predictions

As stated previously, the fission gas release may have been overpredicted for each of the three IFA-432 rods considered. The comparison is summarized in Table A2.4. The largest overprediction is for Rod 2, which also experienced the largest overprediction of temperature.

Figure A2.11. Rod 2 Match Between Input and Measured LHGR Versus Time, (a) Upper TC Position, and (b) Lower TC Position

Figure A2.12. Rod 2 Match Between Predicted and Measured Temperature Versus Time (lower TC)

Figure 2.13. Rod 2 Match Between Predicted and Measured BOL Temperature Versus Time (lower TC)

Figure A2.14. Rod 3 Match Between Input and Measured LHGR Versus Time, (a) Upper TC Position, and (b) Lower TC Position

— Data - + - FRAPCON-3

— Data - + - FRAPCON-3

Figure A2.15. Rod 3 Predicted and Measured Temperature Versus Time,
(a) Lower Thermocouple, and (b) Upper Thermocouple

Figure A2.16. Rod 3 Predicted and Measured BOL Temperature Versus LHGR, (a) Upper TC Position, and (b) Lower TC Position

A.2.7 Bibliography (not called out references)

D. D. Lanning. 1987. *Experimental Support and Development of Single-Rod Fuel Codes Program: Summary Report*, NUREG/CR-4178, PNNL-5972, Pacific Northwest Laboratory, Richland, Washington.

A.3 Description of FRAPCON-3 Case Input for IFA-513 Rods 1 and 6

A.3.1 Overview of the IFA-513 Test

The IFA-513 test fuel assembly was irradiated in the Halden Reactor, Norway, from November 1978 to mid-1981 under a continuation of an NRC program to test the performance of BWR-6 type fuel and the effects of fission gas contamination of the helium fill gas.

Further detail is provided below on the following topics: rod design and fabrication, irradiation conditions and history, and major PIE results. The final sections describe the recommended FRAPCON-3 input for Rods 1 and 6 from the assembly and code-data comparisons for fuel-temperature data and interim fission-gas release.

A.3.2 Test Rod Design and Fabrication

The general design specifications are shown in Table A3.1. All rods had the same diametral fuel-cladding gap size, i.e., 9 mils (230 microns), which is the same as the gap for the standard rods from the IFA-431/432 assemblies. All rods contained 95% TD dense, stable, flat-ended sintered UO_2 fuel pellets. The rods in IFA-513 were slightly longer than the IFA-432 rods, with a fuel column length of 31.5 inches, as compared to 22.5 inches for the IFA-432 rods. The main variation among the rods was fill-gas composition and pressure. These variations are shown in Table A3.2.

The rod and assembly instrumentation was similar to that for IFA-432. Each rod had a fuel centerline thermocouple in both the top and the bottom and carried a pressure transducer. Assembly instrumentation included six vanadium and one cobalt self-powered neutron detectors (SPNDs). Rod axial elongation sensors were carried at each rod position; and the assembly also provided a turbine-type coolant flow meter and multiple coolant thermocouples at the top and bottom. A plot of the general axial power shape, relative to the positions of the SPNDs and thermocouple tips, is shown in Figure A3.1.

A.3.3 Test Rod Irradiation History

The Halden Reactor is a boiling heavy water reactor operating at nominal coolant conditions of 500 psi pressure, 237°C (459°F) inlet temperature, and 240°C (464°F) saturation temperature. The nominal operating conditions are summarized in Table A3.3. Note that the neutron flux in this heavy water reactor is well thermalized, and the fast neutron flux component is significantly less than in light water power reactor.

Published temperature and LHGR data extend through July 1980 when a peak burnup of 10 GWd/MTU was attained. Halden reports contain further irradiation data on some rods. Only the irradiation (to 10 GWd/MTU) data described in the NRC/PNNL references are simulated here.

A plot of the LHGRs at the upper and lower thermocouple locations is shown in Figure A3.2 for Rod 1, and the corresponding measured temperatures are shown in Figure A3.3. Similarly, LHGRs and

Table A3.1. Fuel Rod Components and Specifications for IFA-513, Rods 1 and 6

Fuel Rod	
Overall length (in.)	32.4
Diameter (in.)	0.5039
Fuel stack height (in.)	30.7
Nominal plenum chamber length (in.)	1.142
Number of pellets per rod	62
Fuel/clad diametral gaps (mil)	7.87
Fill gas compositions and pressures: 100% He at 1.0 atm for 1, 3, 5 100% He at 0.3 atm for 2 8% Xe and 92%He at 1 atm for 4 23% Xe and 77% He at 1 atm for 6	He
Cladding	
Material	Zircaloy-2
Diameter outside (in.)	0.5039
Diameter inside (in.)	0.4291
Clad wall thickness (in.)	0.0374
Ovality	Not specified
Eccentricity	Not specified
Fuel	
Material	UO ₂
Enrichment (%)	9.9
Pellet sintered density (% TD)	95.0
Pellet outer diameter (in.)	0.4213
Pellet inner diameter (in.)	0.069
Pellet length (in.)	0.5000
Pellet geometry	Flat end
Grain size (μm)	5 to 12

Table A3.2. IFA-513 Rod Design Variations

Rod Number	Fill Gas	Diametral Gap (microns)	Fuel Density (%)	Special Features
1	100% He at 0.1 MPa	230	95	Standard rod
2	100% He at 0.3 MPa	230	95	Higher fill gas pressure
3	100% He at 0.1 MPa	230	95	Standard rod
4	8% Xe, 92% He at 0.1 MPa	230	95	75% of He fill gas conductivity
5	100% He at 0.1 MPa	230	95	Standard rod
6	23% Xe, 77% He at 0.1 MPa	230	95	50% of He fill gas conductivity

Figure A3.1. General Axial Power Shape, Relative to the Positions of the SPNDs and Thermocouple Tips

temperatures are plotted in Figures A3.4 and A3.5 for Rod 6. The temperatures at a given LHGR are higher for Rod 6 than Rod 1 because the Rod 6 fill gas (77% He, 23% Xe) was chosen to have about half the thermal conductivity of pure helium.

Table A3.3. Nominal Halden Reactor Operating Conditions

Parameter and Units	Value
Power level (MW)	12
Reactor pressure (psig)	500
Heavy water saturation temperature (°F)	464
Plenum inlet temperature (°F)	459
Fast flux (> 1 MeV) (N/m ² /s per W/g fuel)	$\sim 5 \times 10^{15}$
Thermal flux (N/m ² /s per W/g fuel)	$\sim 2 \times 10^{16}$

Figure A3.2. The LHGRs at the Upper and Lower Thermocouple Locations for IFA-513, Rod 1

Neither the measured fuel center temperatures nor the pressure transducer data indicated significant fission gas release for these irradiations. The fission gas release estimates for each rod, derived from pressure transducer data, are shown in Table A3.4.

Figure A3.3. The Measured Temperatures at the Upper and Lower Thermocouple Locations for IFA-513, Rod 1

Figure A3.4. The LHGRs at the Upper and Lower Thermocouple Locations for IFA-513, Rod 6

Figure A3.5. The Measured Temperature at the Upper and Lower Thermocouple Locations of IFA-513, Rod 6

Table A3.4. Fission Gas Release Estimates for IFA-513 Rods

Rod Number	Fission Gas Release (%)
1	0.0
2	2.2
3	0.9
4	1.4
5	Not available
6	2.0

A.3.4 FRAPCON-3 Input

A listing of the recommended FRAPCON-3 input parameters for Rods 1 and 6 appears in Table A3.5. The LHGR histories were derived from Bradley et al. (1981) rod design and as-fabricated data were taken from Bradley et al. (1979).

Table A3.5. Recommended FRAPCON-3 Input for IFA-513, Rods 1 and 6

Input Variable	Meaning and Units	Input Value	Calculations and Notes	Reference
cpl	Cold plenum length (in.)	1.14	Design value	Bradley et al. 1979
crdt	Initial crud thickness (mils)	0.0	No crud observed on test rods in Halden	Lanning and Cunningham 1979
dco	Cladding outer diameter (in.)	0.5035	Design value	Bradley et al. 1979
thkcld	Cladding wall thickness (in.)	0.0372	Design value	Bradley et al. 1979
pitch	Rod-to-rod pitch (in.)	0.56	Arbitrary value - boiling condition	
den	Fuel pellet density (% of theoretical)	95.5	Design value	Bradley et al. 1979
dishsd	Dish shoulder width (in.)	0.0	Flat-ended pellets	Bradley et al. 1979
thkgap	Gap radial thickness (in.)	0.00425	Design value = 230 microns = 9 mils diametral; however minimum values from as-fabricated tolerances could be as low as 0.00425	Bradley et al. 1979
dspg	Spring diameter (in.)	0.42	Assumed	--
dspgw	Spring wire diameter (in.)	0.04	Assumed	--
enrch	Fuel enrichment (% U-235 in U)	9.9	As-fabricated value	Bradley et al. 1979
fgpav	Fill gas pressure (psi at room temp.)	14.7	No pre-pressurization	Bradley et al. 1979
hdish	Dish depth (in.)	0.0	Flat-ended pellets	Bradley et al. 1979
hplt	Pellet length (in.)	0.50	Design value	Bradley et al. 1979
icm	Cladding type (2 = zircaloy-2)	5	Design value	Bradley et al. 1979
icor	Cladding crud buildup (control = 0 for no buildup)	0	No crud buildup noted on Halden rods	Bradley et al. 1979
idxgas	Gas type (1 = helium, 6 = user specified)	Varies 1 (Rod 1) 6 (Rod 6)	Rod 6 AMFHE = 0.77 AMFXE = 0.23 to get xenon = 1 (He) (Rod 1)	Bradley et al. 1979
iplant	Type of plant	-4	Heavy Water Plant	
iq	Indicator for axial power shape	0		
jdlpr	Output control	0		
jn	Number of points per axial power shape	6,6,6,6,6		

Table A3.5. (contd)

Input Variable	Meaning and Units	Input Value	Calculations and Notes	Reference
jst	Assignment of shape numbers to each time step	Rods 1 & 6 10 × 1 4 × 2 4 × 3 4 × 2 4 × 4 4 × 5		
totl	Fuel stack length (ft)	2.56	Design value = 30.7 (in.)	--
rc	Fuel pellet inner diameter (in.)	0.0345	Central holes for centerline thermocouple, from both ends, but not full-length	--
roughc	Cladding surface roughness (in.)	4.5E-5	Measured	Bradley et al. 1979
roughf	Pellet surface roughness (in.)	8.5E-5	Ground to size	Bradley et al. 1979
rsntr	Change in pellet density due to densification (kg/m ³)	75	Stable pellets; resinter test results	Bradley et al. 1979
vs	Number of plenum spring turns	5.0	Assumed	--
flux	Fast neutrons (per m ² per second per W/g of fuel)	5E15	Halden conditions	Bradley et al. 1979
p2	Reactor system pressure (psi)	500	Halden conditions	Bradley et al. 1979
tw	Coolant inlet temperature (°F)	459	Halden conditions	Lanning and Cunningham 1979
go	Coolant mass flow (lbm/ft ² /h)	0.0	Halden is a boiling water reactor	Lanning and Cunningham 1979
qf, x	Parameters of the axial flux shapes	Lanning and Cunningham 1979	Lanning and Cunningham (1979) typical core flux shape combined with fuel position	Lanning and Cunningham 1979
qmpy, time	The LHGR history	Lanning and Cunningham 1979; Bradley et al. 1981	Care taken to get LHGRs at thermocouple tip positions	Lanning and Cunningham 1979; Bradley et al. 1981

A.3.5 Code-Data Comparisons

A.3.5.1 Rod 1

The input LHGRs as a function of time for the upper and lower fuel centerline thermocouple locations closely matched the LHGR data (see Figure A3.6a,b respectively). The centerline temperatures as a function of time were well predicted by the code, as shown in Figure A3.7a,b. This match corresponds with prediction of negligible fission gas release (see Table A3.6).

Figure A3.6. Measured and Input LHGRs for IFA-513, Rod 1, (a) Upper Thermocouple Position, and (b) Lower Thermocouple Position

Figure A3.7. Measured and Predicted Temperatures for IFA-513, Rod 1, (a) Upper Thermocouple, and (b) Lower Thermocouple

Table A3.6. Measured and Predicted Fission Gas Release (FGR)

Rod Number	Measured FGR (%)	Predicted FGR (%)
1	0.0	0.8
6	2.0	6.1

The BOL temperatures were also well predicted as a function of LHGR (Figure A3.8a,b).

A.3.5.2 Rod 6

The pattern for the code-data comparison to centerline temperature data discussed above for Rod 1 was replicated for Rod 6. The LHGR data at the upper and lower thermocouple positions were well replicated by the input LHGRs (Figure A3.9a,b). The predicted centerline temperatures as a function of time matched the data well (Figure A3.10a,b). This result again corresponded with the prediction of low FGR over most of the analyzed irradiation period. The prediction of incipient fission gas release in the last 50 days of the period resulted in an incipient overprediction of the temperatures.

The BOL fuel temperature data for this rod were slightly underpredicted at high LHGRs for both thermocouple positions, as shown in Figure A3.11a,b. This could be a result of a slight overestimate for the early-in-life fuel pellet relocation and gap closure.

Figure A3.8. Measured and Predicted BOL Temperatures for IFA-513, Rod 1,
(a) Upper Thermocouple, and (b) Lower Thermocouple

Figure A3.9. Input and Measured LHGRs for IFA-513, Rod 6, (a) Upper Thermocouple Position and (b) Lower Thermocouple Positions

Figure A3.10. Measured and Predicted Temperatures for IFA-513, Rod 6, (a) Upper Thermocouple, and (b) Lower Thermocouple

Figure A3.11. The BOL Temperatures as a Function of LHGR for IFA-513, Rod 5,
(a) Upper Thermocouple, and (b) Lower Thermocouple

A.4 Description of FRAPCON Case Input for IFA-429, Rod DH Test Case

A.4.1 Overview of the IFA-429 Test

The IFA-429 test fuel assembly was initiated by NRC-Research, and designed and fabricated by INEL (with fuel pellet fabrication by PNNL) to demonstrate the effect of burnup, power level and fuel grain size on fuel thermal behavior and fission gas release. The assembly consisted of 18 original short rods, arranged in three clusters of 6 rods each, and 15 noninstrumented spare and replacement rods. Rod DH is a replacement rod that was re-instrumented with a pressure transducer after it had attained about 30 GWd/MTU burnup at relatively low linear heat generation rate (LHGR); the rod was then irradiated in IFA-519 at much higher and variable LHGR as part of a load-follow test, and eventually attained a peak burnup of 74 GWd/MTU. The estimate for fission gas release for these rods is derived from interpreting the pressure transducer data. Further detail is provided below on the following topics:

- Rod design and fabrication
- Irradiation conditions and history
- Fission gas release results and predictions
- FRAPCON-3 input for Rod DH

A.4.2 Test Rod Design and Fabrication

The assembly instrumentation included vanadium neutron detectors, coolant flowmeters and thermocouples, and moveable absorbing shields by which the power could be raised temporarily on individual rod clusters.

The general rod design specifications are shown in Table A4.1. Rod DH had a nominal gap size of 200 microns (8 mils) and contained 13% enriched, 95% TD dense, stable, dished-end sintered UO₂ fuel pellets. The rod diameter and the cladding type and thickness are typical of PWR-type power reactor fuel rods.

A.4.3 Test Rod Irradiation History

The Halden Reactor is a boiling heavy water reactor operating at nominal coolant conditions of 500 psi pressure, 237°C (459°F) inlet temperature, and 240°C (464°F) saturation temperature. The nominal operating conditions are summarized in Table A4.2. Note that the neutron flux in this heavy water reactor is well thermalized, and the fast neutron flux is significantly less than in a light water power reactor.

Table A4.1. Fuel Rod Components and Specifications for IFA-429, Rods C and DH

Fuel Rod	
Overall length (in.)	15
Diameter (in.)	0.4220
Fuel stack height (in.)	9.61
Nominal plenum chamber length (in.)	4.90
Number of pellets per rod	16
Fuel/clad diametral gaps (mil)	7.9
Fill gas composition	He
Fill gas pressure (atm)	25.9
Cladding	
Material	Zircaloy-4
Diameter outside (in.)	0.4220
Diameter inside (in.)	0.3740
Clad wall thickness (in.)	0.0240
Ovality	Not specified
Eccentricity	Not specified
Fuel	
Material	UO ₂
Enrichment (%)	13.0
Pellet sintered density (% TD)	95.0
Pellet diameter (in.)	0.3660
Pellet length (in.)	0.5984
Pellet geometry	Dished
Dish radius (in.)	0.100
Depth (in.)	0.0584
Grain size (μm)	5.9 (DH), 17 (CD)

Table A4.2. Nominal Halden Reactor Operating Conditions

Parameter and Units	Value
Power level (MW)	12
Reactor pressure (psig)	500
Heavy water saturation temperature (°F)	464
Plenum inlet temperature (°F)	459
Fast flux (> 1 MeV) (N/m ² /s per W/g fuel)	$\sim 5 \times 10^{15}$
Thermal flux (N/m ² /s per W/g fuel)	$\sim 2 \times 10^{16}$

Published LHGR histories and pressure data are contained in Waterman (1978)^(a) for Rod CD, and in Wiesenack et al.^(b) for Rod DH. Figures A4.1 through A4.3, taken from Wiesenack et al.,^(b) contain the following information on Rod DH: the LHGR history (Figure A4.1) and the general axial power profile (Figure A4.2). For comparison, the input LHGR vs. rod-average burnup is shown in Figure A4.4. Note that in Halden reports such as Wiesenack et al.,^(b) burnup is reported in MWd/kg UO₂; the conversion is GWd/MTU = MWd/kg UO₂ × 1.14.

A.4.4 Fission Gas Release Estimates and FRAPCON-3 Predictions

The fission gas release through-life for Rod DH has been estimated by Halden based on the pressure transducer readings taken periodically at low power throughout the irradiation. These estimates are plotted in Figure A4.3, together with the FRAPCON-3 predictions. Note that this estimated FGR is negligible during the first 28 GWd/MTU and increases thereafter in response to the ascension in LHGR, reaching 24% at EOL. These trends are matched by the predictions, except that the rate of FGR increase is greater than the estimated rate after about 43 GWd/MTU, such that the predicted FGR at EOL is 42%.

A.4.5 FRAPCON-3 Input for Rod DH

The FRAPCON-3 input for Rod DH is listed in Table A4.3. For reference, the input LHGR history is shown in Figure A4.4.

(a) Halden Reactor Project. 1997. Personal communication with USNRC.

(b) Wiesenack, W., H. Devold, Y. Kosaka, J. J. Serna, V. Tosi, H. Wallin, and T. Yamaguchi. 1993. *Summary of HBWR Fission Gas Release Data at Medium and High Burnup*, HWR-353 (restricted distribution), Halden Reactor Project, Halden, Norway.

Figure A4.1. The LHGR History on Rod DH

Figure A4.2. The General Axial Power Profile on Rod DH

Figure A4.3. Measured LHGR, and Measured and Predicted Fission Gas Release Versus Rod-Average Burnup for Rod DH of IFA-429

Figure A4.4. The Input LHGR Versus Rod-Average Burnup for Rod DH in Standard Units for FRAPCON-3 Input

Table A4.3. Background Notes on IFA-429 Rod DH Input for FRAPCON-3

Input Variable	Meaning and Units	Input Value	Calculations and Notes	Reference
cpl	Cold plenum length (in.)	4.90	Plenum value quoted after refabrication = 9.2 cm ³ . Fuel ID = 0.95 cm, so plenum length = $9.2 / \{(p/4) \times 0.952\} = 13.0 \text{ cm} = 5.1 \text{ in.}$ Reduced to 4.9 because "plenum volume" ^(a) probably meant "void volume"	(a)
crdt	Initial crud thickness (mils)	0.0	Crud is not observed on Halden rods, even after long exposure	(a)
dco	Cladding outer diameter (in.)	0.4220	Design value	Waterman 1978
thkcld	Cladding wall thickness (in.)	0.024	Design value	Waterman 1978
den	Fuel pellet density (% of theoretical)	95	Design value	(a)
dishsd	Dish shoulder width (in.)	0.122	$d = \sqrt{2(hR - h^2)} = \sqrt{2(0.33)(16.8) - 0.332} = 3.3 \text{ mm.}$ Shoulder = $(9.5 - 3.3)/2 = 3.1 \text{ mm} = 0.122 \text{ in.}$	Waterman 1978
thkgap	Gap radial thickness (in.)	0.00394	Design value = 200 microns diametral = 3.94 mils radial	Waterman 1978 ^(a)
dspg	Spring diameter (in.)	0.358	Assumed	--
dspgw	Spring wire diameter (in.)	0.055	Assumed	--
enrch	Fuel enrichment (% ²³⁵ U in U)	13.0	Design value	Waterman 1978
fgpav	Fill gas pressure (psi at room temp.)	375.6	Design value = 2.58 MPa = 374 psi	Waterman 1978
hdish	Dish depth (in.)	0.013	Design value = 0.33 m	Waterman 1978
hplt	Pellet length (in.)	0.5984	Design value = 15.2 mm	Waterman 1978
icm	Cladding type (4 = zircaloy-4)	4	Design value	Waterman 1978
pitch	Rod-to-rod pitch (in.)	0.56	Arbitrary value - Boiling condition	
icor	Cladding crud buildup (control = 0 for no buildup)	0	No crud buildup noted on Halden rods	(a)
idxgas	Gas type (1 = helium)	1	Design value	Waterman 1978
iplant	Type of plant	-4	Heavy Water Plant	
iq	Indicator for axial power shape	0		
jdlpr	Output control	1		

Table A4.3. (contd)

Input Variable	Meaning and Units	Input Value	Calculations and Notes	Reference
jn	Number of points per axial power shape	7,7,7,7,7		
jst	Assignment of shape numbers to each time step	7 × 1 20 × 2 14 × 3 15 × 4 19 × 5		
totl	Fuel stack length (ft)	.8005	Design value = 30.7 (in.)	--
roughc	Cladding surface roughness (in.)	1.E-5	Assumed	Waterman 1978
roughf	Pellet surface roughness (in.)	8.E-5	Assumed; 80 micro-inches is a standard value for ground pellets	Waterman 1978
rsntr	Change in pellet density due to densification (kg/m ³)	75	Stable pellets	Waterman 1978
vs	Number of plenum spring turns	10	Assumed	--
flux	Fast neutrons (per m ² per second per W/g of fuel)	5E15	Quoted fast flux condition for the well thermalized Halden heavy water reactor	(a)
p2	Reactor system pressure (psi)	500	Halden conditions (test rods were not in a pressurized loop)	(a)
tw	Coolant inlet temperature (°F)	464	Halden conditions	(a)
go	Coolant mass flow (lbm/ft ² /h)	0.0	Use coolant inlet for coolant temperature (Halden is a boiler)	(a)
qf, x	Parameters of the axial flux shapes	Cunningham and Svanholm 1986	The shape reversed at ~30 MWd/kgU burnup	(a)
qmpy, time	The LHGR history	Cunningham and Svanholm (1986)	The input times were calculated from the plotted burnups.	(a)
(a) Halden Reactor Project. 1997. Personal communication with USNRC.				

A.4.6 Bibliography (not called out references)

Cunningham, M. E., and H. Devold. 1986. *Steady-State Irradiation of IFA-419 to Burnup > 50 MWd/kg UO₂*, HPR-319.

Smith, M. R., P. M. Crosby, J. M. Aasgaard, and Y. Minagawa. 1988. *Fission Gas Release During a Second Series of Power Increases on IFA-429 Rods at 40 GWd/TUO₂*, HWR-237, Halden Reactor Project, Halden, Norway.

A.5 BR-3 Test Cases

A.5.1 Description of the BR-3 Test Cases

The U.S. Department of Energy sponsored high-burnup irradiation of five well-characterized PWR-type test rods in the BR-3 reactor located in Mol, Belgium, to demonstrate the feasibility of extending commercial fuel rod burnup and thereby helping minimize radioactive waste disposal. These rods were fabricated by Westinghouse Corporation, whose staff also oversaw the PIEs. The PIE on the rods was carried out in the BR-2 hot cell facility at the Mol site. The rods were of basic PWR radial dimensions. Goal peak burnups exceeded 70 GWd/MTU.

Further detail is provided below on the following topics: rod design and fabrication, irradiation conditions and history, and major PIE results. The final section describes the recommended FRAPCON-3 input for four selected cases.

A.5.2 Test Rod Design and Fabrication

The test rods were designed to simulate Westinghouse PWR (15 x 15) rod cladding type and radial dimensions, with variations in fuel enrichment and rod position providing variations in power history. The general design specifications are given in Table A5.1. The fuel rod length was much shorter than the full-length (~144-inch) commercial reactor rods and fit well within the short length of the BR-3 reactor core. The fuel rod overall length was 44 inches with an active fuel column length of 38.4 inches. Gap size for all rods was 7.5 mils.

Four of the five rods were selected for comparison with FRAPCON-3 predictions: 24-I-6, 36-I-8, and 111-I-5, and 28-I-6.

A.5.3 Irradiation Conditions and History

The BR-3 reactor is a pressurized water reactor (PWR) with a coolant inlet temperature of 255°C (491°F) and a system pressure 2,250 psia. The five high burnup rods began irradiations in 1974, resided in the same 20-rod assembly for three reactor cycles, and were discharged in 1980 with rod-average burnups up to 61 GWd/MTU and maximum burnups of slightly less than 70 GWd/MTU. The peak powers early in life were 15 to 16 kW/ft and decreased by the third cycle to 10 to 12 kW/ft. A plot of the rod-average powers vs. operating time for the four selected rods appears in Figures A5.1 through A5.4. Rod 111-I-5 operated for only two cycles. Rods 24-I-6, 28-I-6, and 36-I-8 operated for three cycles in the same assembly.

The axial power profile at BOL was a chopped cosine distribution with a peak-to-average ratio of about 1.35. The profile gradually flattened with increasing burnup.

Table A5.1. General Design Specifications for BR-3 Rods

Fuel Rod	
Overall length (in.)	44.714
Diameter (in.)	0.4220
Fuel stack height (in.)	38.40
Nominal plenum chamber length (in.)	4.0145
Number of pellets per rod	64
Fuel/clad diametral gaps (mils)	7.5
Fill gas composition	He
Fill gas pressure (atm)	13.61
Cladding	
Material	Zircaloy-4
Diameter outside (in.)	0.4220
Diameter inside (in.)	0.3734
Clad wall thickness (in.)	0.0243
Ovality	Not specified
Eccentricity	Not specified
Grain size	Not specified
Fuel	
Material	UO ₂
Enrichment (%)	5.2, 6.4, and 8.5
Pellet sintered density (% TD)	95.007
Pellet diameter (in.)	0.3659
Pellet length	0.600
Pellet geometry	Dished, both ends
Dish radius (in.)	0.6600
Dish depth (in.)	0.0135
Grain size (μm)	10.9

Figure A5.1. Rod-Average Power Versus Operating Time for Rod 24-I-6

Figure A5.2. Rod-Average Power Versus Operating Time for Rod 36-I-8

Figure A5.3. Rod-Average Power Versus Operating Time for Rod 111-I-5

Figure A5.4. Rod-Average Power Versus Operating Time for Rod 28-I-6

A.5.4 Measured and Predicted Fission Gas Release at EOL

Balfour et al. (1982a and 1982b) present post-irradiation data for rod growth, oxidation and hydriding, fuel swelling, void volume change, burnup, and fission gas release. Of these, however, only fission gas release and swelling are considered to be unaffected by the accelerated nature of the irradiation. The fuel swelling results are noted in Section 5 of the main text. The measured and predicted gas release and end-of-life burnups for the four rods are listed here in Table A5.2. The predictions and measurements compare closely for these high-powered rods with well-defined power histories.

Table A5.2. BR-3 Rod Burnup and Fission Gas Release Measurements

Rod Number	Rod-Average Burnup (GWd/MTU)	Measured Fission Gas Release (%)	Predicted Fission Gas Release (%)
24-I-6	60.1	21.8	20.6
28-I-6	53.3	13.2	12.3
36-I-8	61.5	33.8	35.4
111-I-5	48.6	14.4	13.6

A.5.5 FRAPCON-3 Input

FRAPCON-3 input for the four test rods is listed in Table A5.3 below.

Table A5.3. Background Notes on Westinghouse BR-3, Rod Input for FRAPCON-3

Input Variable	Meaning and Units	Input Value	Calculations and Notes	Reference
cpl	Cold plenum length (in.)	4.0145	Void volume/fuel volume ratio quoted as 0.08 by design	Balfour et al. 1982
crdt	Initial crud thickness (mils)	1.0		Balfour et al. 1982
dco	Cladding outer diameter (in.)	0.4220	Design value	Balfour et al. 1982
pitch	Rod-to-rod pitch (in.)	0.56 (111-I-5) 0.505 (24-I-6) 0.56 (28-I-6) 0.56 (36-I-8)	Arbitrary value - boiling condition	
thkclad	Cladding wall thickness (in.)	.0244 (111-I-5) .0243 (24-I-6) .0243 (28-I-6) .0244 (36-I-8)	Design value	Balfour et al. 1982
den	Fuel pellet density (% of theoretical)	94.77(111-I-5) 94.77(24-I-6) 94.70 (28-I-6) 94.774(36-I-8)	Design value	Balfour et al. 1982
dishsd	Dish shoulder width (in.)	0.0504	Design value	Balfour et al. 1982
thkgap	Gap radial thickens (in.)	0.00375	Design value	Balfour et al. 1982
dspg	Spring diameter (in.)	.358 (111-I-5) .370 (24-I-6, 28-I-6, 36-I-8)	Design value	-
dspgw	Spring wire diameter (in.)	0.055	Design value	-
enrch	Fuel enrichment (% ²³⁵ U in U)	Varies	111-I-5 = 5.2 24-I-6 = 6.42 28-I-6 = 6.42 36-I-8 = 8.53	Balfour et al. 1982
fgpav	Fill gas pressure (psi at room temp.)	Varies	111-I-5 = 214.4 24-I-6 = 200.1 28-I-6 = 200.0 38-I-8 = 214.4	Balfour et al. 1982
hdish	Dish depth (in.)	0.014	Design value	Balfour et al. 1982
hplt	Pellet length (in.)	0.60	Design value	Balfour et al. 1982
icm	Cladding type (4 = zircaloy-4)	4	Design value	Balfour et al. 1982

Table A5.3. (contd)

Input Variable	Meaning and Units	Input Value	Calculations and Notes	Reference
icor	Cladding crud buildup control (= 0 for no buildup)	0	No crud buildup probable in BR-3	Balfour et al. 1982
idxgas	Gas type (1 = helium)	1	Helium fill (all rods)	Balfour et al. 1982
iplant	Type of plant	-2		
iq	Indicator for axial power shape	0		
jdlpr	Output control	0 (111-I-5) 0 (24-I-6) 1 (28-I-6) 0 (36-I-8)		
jn	Number of points per axial power shape	15,15,15,15, 15,15 (111-I-5) 15,15,15,15, 15,15,15,15 (24-I-6, 28-I-6) 11,11,11,11, 11,11 (36-I-8)		
jst	Assignment of shape numbers to each time step	Varies	4*1, 4*2, 5*3, 2*4, 4*5, 6*6 (111-I-5) 11*1, 13*2, 8*3, 2*4, 4*5, 5*6, 5*7, 6*8 (24-I-6) 8*1, 9*2, 6*3, 2*4, 4*5, 5*6, 6*7, 12*8 (28-I-6) 5*1, 3*2, 6*3, 4*4, 7*5, 5*6, 8*5 (36-I-8)	
totl	Fuel column length (ft)	3.2	Design value = 38.4 inches	-
rc	Fuel pellet inner diameter (in.)	0.0	Solid pellets	-
roughc	Cladding surface roughness (in.)	1.97E-5	Assumed	Balfour et al. 1982
roughf	Pellet surface roughness (in.)	2.36E-5	Sintered to size	Balfour et al. 1982
rsntr	Change in pellet density due to densification (kg/m ³)	101.9	Stable pellets	Balfour et al. 1982
vs	Number of plenum spring turns	8.0	Assumed	-
flux	Fast neutrons (per m ² per second per W/g of fuel)	0.21E17	Non-thermalized heterogeneous core	Balfour et al. 1982
p2	Reactor system pressure (psi)	2199.0	BR-3 conditions	Balfour et al. 1982
tw	Coolant inlet temperature (°F)	491	PBF conditions	Balfour et al. 1982

Table A5.3. (contd)

Input Variable	Meaning and Units	Input Value	Calculations and Notes	Reference
go	Coolant mass flow (lbm/ft ² /h)	Varies	1.9E6 (111-I-5) 2.1E6 (24-I-6) 2.1E6 (28-I-6) 1.7E6 (36-I-8)	Balfour et al. 1982
qf, x	Parameters of the axial flux shapes	See Balfour et al. 1982	Profiles flatten with burnup	Balfour et al. 1982
qmpy, time	The LHGR history	See Balfour et al. 1982		Balfour et al. 1982

A.6 Description of FRAPCON-3 Case Input for BNFL/HBEP Br-3 Test Rod DE (Assembly 373)

A.6.1 Overview of the Irradiation Program

Battelle, Pacific Northwest Laboratories administered the international group-sponsored High Burnup Effects Program (HBEP), which continued from 1978 to 1990 (Barner et al. 1990). The objectives of the HBEP were to determine the effects of extended burnup on fuel rod performance, especially fission gas release. A variety of test rods and commercial power reactor rods were irradiated and examined under the HBEP, including 12 PWR-type test rods irradiated in a single assembly in the BR-3 test reactor located at Mol, Belgium. The 12 rods included a variety of fuel pellet material types, distinguished by density and microstructure, and two variations in Zircaloy-4 cladding type and two variations in fill gas pressure.

This report describes the design, operation, and FRAPCON-3 simulation of one of these 12 rods: Rod number DE from fuel assembly 373. This particular rod was "non-pressurized" (1 atm initial helium fill gas pressure), had cold-worked cladding, and had fuel pellets with an 8-micron average grain size and a density of 93% TD.

The rod was irradiated for four cycles in BR-3 and attained a rod-average burnup at EOL of 41.5 GWd/MTU, with an estimated axial peak value of 51.4 GWd/MTU. The LHGR axial profile remained a chopped-cosine like shape throughout life, with a peak-to-average ratio of approximately 1.3. The peak LHGR varied about 45 kW/m (13.7 kW/ft) over the first 1/4 of the irradiation history and dropped to about 15 kW/m (4.6 kW/ft) over the remaining 3/4 of the history. This high LHGR early-in-life, however, resulted in a significant end-of-life fission gas release, determined by rod puncture to be 10.7%.

In addition to rod puncture, plenum gas analysis and void volume measurement, the postirradiation destructive examinations of Rod DE included fuel burnup measurements, fuel ceramography, and cladding metallography.

The nondestructive examinations included rod profilometry and length measurement, and full-length gamma scanning.

These examinations permitted a qualitative estimate of the maximum temperatures the rod experienced, based on the grain growth observed across the radius; and these temperatures are consistent with the gas-release level.

Further detail is provided below on the following topics:

- Rod design and fabrication

- Irradiation conditions and history
- Major PIE results.

The final sections include the FRAPCON-3 input used to simulate the irradiation history of this rod and a comparison of the FRAPCON-3 predictions to the measured data.

A.6.2 Rod Design and Fabrication

The fuel rod general design specifications are shown in Table A6.1. The radial dimensions of the rods were approximately commensurate with 14 x 14 PWR-type rods. The fuel column length was approximately 39 inches, to match the 1-meter active length of the BR-3 reactor core. The helium fill gas pressure was ~1 atmosphere, and the fuel-cladding diametral gap size had a nominal value of 0.0105 inches.

The operating conditions for the BR-3 reactor are similar to that of a PWR except the coolant temperature is somewhat lower; see Table A6.2.

A.6.3 Rod Irradiation History

Assembly 373 operated for four cycles in BR-3, from 1972 to 1980, and accumulated 1,660 EFPD of exposure. The rod-average burnups varied from 36 to 46 GWd/MTU. The rods were arranged in a 5 x 5 square array with a central instrument tube and with the corner rods removed, for a total of 20 fuel rods. Rod DE was next to the central instrument tube.

The power history was characterized by high peak LHGRs (approximately 45 kW/m) over the first 400 days, followed by very low LHGRs (about 15 kW/m) over the remaining 1,260 operating days. The variation of the peak LHGR through time is shown in Figure A6.1.

The axial distribution of the LHGR remained a chopped cosine shape with a peak-to-average ratio of approximately 1.3 throughout the irradiation due to the high initial enrichment and the removal of control rods from the core during operation. This axial distribution is shown in Figure A6.2.

A.6.4 PIE Results and FRAPCON-3 Predictions

The major PIE results are listed in Table A6.3. Fission gas release is surprisingly high at 10.7%, considering the low values of the rod-average LHGRs. The axial-peak burnup is just over 50 GWd/MTU. The fuel microstructure is, however, fairly consistent with the fission gas release and the early-in-life LHGR. Columnar grain structure was seen at the axial midplane in the innermost 55% of the pellet radius. Very light waterside oxidation was seen on the cladding on these rods, which is typical of BR-3 test rods.

**Table A6.1. Fuel Rod and Component Design Parameters
for HBEP Rod DE, Assembly 373**

Fuel Rod	
Overall length (in.)	44.65
Diameter (in.)	0.4244
Fuel stack height (in.)	39.4
Nominal plenum chamber length (in.)	3.84
Number of pellets per rod	88
Fuel/clad diametral gaps (mils)	10.4
Fill gas pressure (atm)	He
Fill gas pressure (atm)	1.0
Cladding	
Material	Zircaloy-4 15% CW
Diameter outside (in.)	0.4244
Diameter inside (in.)	0.3752
Clad wall thickness	0.0246
Ovality	Not specified
Eccentricity	Not specified
Fuel	
Material	UO ₂
Enrichment (%)	9.0
Pellet sintered density (% TD)	93
Pellet diameter (in.)	0.3648
Pellet length (in.)	0.4488
Pellet geometry	Dished, both ends (cylindrical)
Dish radius (in.)	0.1368
Dish depth (in.)	0.015
Grain size (μm)	8

Table A6.2. BR-3 Reactor Operating Conditions

Parameter and Units	Parameter Value
System pressure (psi)	2230
Inlet temperature (°F)	491
Outlet temperature (°F)	534
Fast flux (N/cm ² /s per W/g fuel)	1.6E12
Coolant velocity (m/s)	3.04

Figure A6.1. The Variation of the Peak LHGR Through Time

The FRAPCON-3 prediction of fission gas release at EOL for this rod is 7.7%, which is not far from the measured value. Most of the gas is predicted to have been released during the initial period of high-power operation. At the end of that period (400 days), the rod-average burnup is 22.4 GWd/MTU, and the predicted cumulative gas release is 16%. Over the next 1,220 days, the burnup increases to 42.8 and the cumulative release falls to 7.7%, which is approximately commensurate with zero further gas release during the long period of lower-power operation.

At the end of the high-power operation period, the 1700°C isotherm (within which columnar grain growth could be expected) is predicted to be 32% of the fuel radius, which is somewhat less than the

Figure A6.2. Typical Axial Distribution of LHGR at BOL

Table A6.3. PIE Results

Parameter and Units	Measured Values	Predicted Values
Peak/rod-average burnup (GWd/MTU)	51.4/42	49/42.5
Fission gas release at EOL (% of produced gas)	10.7	7.7
Grain growth in the fuel pellet	Columnar grains to 55% of the radius (2 mm) at rod midplane. Grain growth to ~70% of the radius	-

observed radius for columnar growth, at 55% of the fuel radius. This may be an indication that the maximum peak LHGR's and fuel temperature were even higher than estimated.

A.6.5 FRAPCON-3 Input

The major FRAPCON-3 input parameters are shown and annotated in Table A6.4. The input power history is shown in Figure A6.3.

Table A6.4. FRAPCON-3 Input for BR-3, Rod DE and Notes on Derivation

Input Variable	Meaning and Units	Input Value	Calculations and Notes	Reference
cpl	Cold plenum length (in.)	3.8	Adjusted slightly to get calc. total void volume to match as-fabricated value of 30.9 cm ³	Lanning et al. 1987; Winne 1981
crdt	Initial crud thickness (mils)	0.0	Very low crud found on rods	Lanning et al. 1987
dco	Cladding outer diameter (in.)	0.4244	Design value	Lanning et al. 1987
pitch	Rod-to-rod spacing (in.)	0.556	Not given; literature value for PWR 14 x 14 grid used	-
thkcld	Cladding wall thickness (in.)	0.0246	Design value	Lanning et al. 1987
den	Fuel pellet density (% of theoretical)	93.1	Design value	Lanning et al. 1987
dishsd	Dish shoulder width (in.)	0.0456	Cylindrical dish	Lanning et al. 1987
thkgap	Gap radial thickens (in.)	0.0042	Design value	Lanning et al. 1987
dspg	Spring diameter (in.)	0.37	Estimated	Grimoldy and Crossley 1981
dspgw	Spring wire diameter (in.)	0.055	Estimated	Grimoldy and Crossley 1981
enrch	Fuel enrichment (% ²³⁵ U in U)	8.9	Design value	Lanning et al. 1987
fgpav	Fill gas pressure (psi at room temp.)	14.7	Stated BOL value	Lanning et al. 1987
hdish	Dish depth (in.)	0.015	Stated design value = 0.38 mm	Lanning et al. 1987
hplt	Pellet length (in.)	0.4488	Design value	Winne 1981
icm	Cladding type (2 = zircaloy-2)	4	Design value	Lanning et al. 1987
icor	Cladding crud buildup control (= 0 for no buildup)	0	Small crud allowed	-
idxgas	Gas type (1 = helium)	1	Design value	Lanning et al. 1987
iplant	Type of plant	-2		
iq	Indicator for axial power shape	0		
jdlpr	Output control	1		
jn	Number of points per axial power shape	11,11,11,11, 11,11,11,11		

Table A6.4. (contd)

Input Variable	Meaning and Units	Input Value	Calculations and Notes	Reference
jst	Assignment of shape numbers to each time step	7*1, 4*2, 5*3, 5*4, 4*5, 7*6, 3*7, 1*8		
totl	Fuel column length (ft)	3.3	Design value = 1,000 mm	Grimoldy and Crossley 1981
roughc	Cladding surface roughness (in.)	1E-5	Nominal value	-
roughf	Pellet surface roughness (in.)	3E-5	Nominal value	-
rsntr	Change in pellet density due to densification (kg/m ³)	75	Based on BNWL sintering test (24 hours at 1700°C in hydrogen)	Lanning et al. 1987
vs	Number of plenum spring turns	8.0	Estimated	Grimoldy and Crossley 1981
flux	Fast neutrons (per m ² per second per W/g of fuel)	1.6E16	Average value for in-water portion based on Monticello and PWR information	Lanning et al. 1987
p2	Reactor system pressure (psi)	2230.0	BR-3 conditions	Lanning et al. 1987; Winne 1981
tw	Coolant inlet temperature (°F)	491	BR-3 conditions	Lanning et al. 1987; Winne 1981
go	Coolant mass flow (lbm/ft ² /h)	1.9E7	Estimated	-
qf, x	Parameters of the axial flux shapes	Lanning et al. 1987	Based on reactor conditions and rod gamma scans	Lanning et al. 1987; Winne 1981
qmpy, time	The LHGR history	See Figure A6.3		Lanning et al. 1987; Winne 1981

A.6.6 Bibliography

Garlick, A., L. A. Goldsmith, P. D. Kennedy, P. Mellor, and J.F.W. Thompson. 1982. Destructive Examinations of KWU/CE and BNFL Rods at Windscale - Task 2A, HBEP-24, Pacific Northwest Laboratory, Richland, Washington.

Figure A6.3. Input Power History

A.7 Description of FRAPCON-3 Case Input for NRX PWR Rod LFF

A.7.1 Overview of the Irradiation Program

Twelve UO_2 fuel rods were irradiated in a pressurized water loop in the NRX reactor. The goal of this test was to measure the gas pressures inside the rods with the following objectives:

- To determine the effects of fuel density on gas pressure and fission gas release.
- To determine the effects of element power output variations on gas pressure and fission gas release.
- To obtain data to test the predictions of a model for calculating the variation of gas pressure with power output.

The rods were grouped into four clusters of three rods each. Each cluster contained rods of differing density, either 10.25, 10.5, or 10.75 g/cm³. Each cluster operated at a different power of 580, 737, 810, and 792 W/cm. The rods were irradiated for 108 full power days and achieved a maximum burnup of 3.29 MWd/kgU.

After irradiation, the rods were dimensioned and punctured for fission gas analysis. Samples from the rods were also analyzed for chemical burnup. This document summarizes the design and operating parameters of Rod LFF.

A.7.2 Rod Design and Fabrication

Rod LFF had a density of 10.5 g/cm³ and ran at a peak power of 585 W/cm. The fuel rod design specifications are shown in Table A7.1. The NRX reactor operating conditions are shown in Table A7.2.

A.7.3 Rod Irradiation History

The rod was maintained throughout life at a constant average LHGR of 15.5 kW/ft, with a peak power of 17.8 kW/ft. The peak-to-average ratio was 1.15. The input LHGR was a constant 17.8 kW/ft following a steep ascension to power at BOL.

A.7.4 PIE Results

The major PIE results are listed in Table A7.3. Fission gas release was found to be relatively constant throughout irradiation for all the rods in the test. Significantly higher releases were seen in the lower density fuels for all heat ratings. Most of the gas was observed to come out as the reactor power was lowered and may be associated with fuel cracking.

A very large diameter increase and the extent of structural changes in one of the rods was attributed to its internal pressure exceeding the coolant pressure.

Table A7.1. Fuel Rod Components and Specifications for NRX Test Rod LFF

Fuel Rod	
Overall length (in.)	9.67
Diameter (in.)	0.8108
Fuel stack height (in.)	9.57
Nominal plenum chamber length (in.)	0.1
Number of pellets per rod	Not specified
Fuel/clad diametral gaps (mils)	12
Fill gas pressure (atm)	Ar
Fill gas pressure (atm)	1
Cladding	
Material	304 stainless steel
Diameter outside (in.)	0.8108
Diameter inside (in.)	0.7708
Clad wall thickness	0.0200
Ovality	Not specified
Eccentricity	Not specified
Grain size	Not specified
Fuel	
Material	UO ₂
Enrichment (%)	2.4
Pellet sintered density (% TD)	95.7
Pellet diameter (in.)	0.7588
Pellet geometry	Flat on one end, dished on one end
Dish depth (in.)	0.02
Grain size (μm)	3 - 4

Table A7.2. NRX Operating Conditions

Parameter and Units	Value
Loop	X-5 in NRX reactor
Operating pressure (psi)	994
Temperature (°F)	410
Water flow (l/s)	5.35
Maximum loop pressure (psi)	1419
Reactor power (MW)	40

Table A7.3. PIE Results

Parameter and Units	Measured Value	Predicted Value
Burnup (GWd/MTU)	3.29	3.25
Fission gas release at EOL (% of produced gas)	17.3	19.6
Length change (mm)	+ 0.01	--
Diametral change (mm)	+ 0.038	--

A.7.5 FRAPCON-3 Input

The recommended FRAPCON-3 input for Rod LFF is listed in Table A7.4. The rod design parameters and power history are both drawn from Notley et al. (1967).

Table A7.4. FRAPCON-3 Input for NRX Test Rod LFF and Notes on Derivation

Input Variable	Meaning and Units	Input Value	Calculations and Notes	Reference
cpl	Cold plenum length (in.)	1.1	Assumed	Notley et al. 1967
crdt	Initial crud thickness (mils)	0.0	No initial crud	Notley et al. 1967
dco	Cladding outer diameter (in.)	0.8108	Design value	Notley et al. 1967
thkcld	Cladding wall thickness (in.)	0.030	Design value	Notley et al. 1967
pitch	Rod-to-rod pitch (in.)	0.90	Arbitrary value - boiling condition	
den	Fuel pellet density (% of theoretical)	95.7	Design value	Notley et al. 1967
dishsd	Dish shoulder width (in.)	0.2466	Assumed, given dish depth	Notley et al. 1967
thkgap	Gap radial thickness (in.)	0.009	Design value	Notley et al. 1967
dspg	Spring diameter (in.)	0.72	Assumed	--
dspgw	Spring wire diameter (in.)	0.055	Assumed	--
enrch	Fuel enrichment (% ²³⁵ U in U)	2.4	Design value	Notley et al. 1967
fgpav	Fill gas pressure (psi at room temp.)	146.0	Stated BOL value	Notley et al. 1967
hdish	Dish depth (in.)	0.02	Design value	Notley et al. 1967
hplt	Pellet length (in.)	0.6043	Design value	Notley et al. 1967
icm	Cladding type (4 = zircaloy-4)	4	Design value	Notley et al. 1967
icor	Cladding crud buildup control (= 0 for no buildup)	0	No further crud buildup assumed; little seen in metallographs	Notley et al. 1967
idxgas	Gas type (1 = helium)	1	Design value	Notley et al. 1967
iplant	Type of Plant	-2		
iq	Indicator for axial power shape	0		
jdlpr	Output control	1		
jn	Number of points per axial power shape	4		
jst	Assignment of shape numbers to each time step	12 × 1		
totl	Fuel column length (ft)	0.7975	Design value	Notley et al. 1967
roughc	Cladding surface roughness (in.)	2.0E-5	Assumed	--
roughf	Pellet surface roughness (in.)	5.0E-5	Assumed	--
rsntr	Change in pellet density due to densification (kg/m ³)	150	Stable pellets	Notley et al. 1967
vs	Number of plenum spring turns	8.0	Assumed	--
flux	Fast neutrons (per m ² per second per W/g of fuel)	4*1.8E16	Default value, derived for PWRs	--
p2	Reactor system pressure (psi)	1000	NRX conditions	Notley et al. 1967
tw	Coolant inlet temperature (°F)	550	NRX conditions	Notley et al. 1967
go	Coolant mass flow (lbm/ft ² /h)	1.9E7	Assumed	--
qf, x	Parameters of the axial flux shapes	1, 1, 1, 1	Assumed flat profile	--
qmpy, time	The LHGR history	See deck	Constant power of 17.8 kW/ft with early steps to get up to power	Notley et al. 1967

A.8 Description of FRAPCON-3 Case Input for NRX Rod CBP

A.8.1 Overview of the Irradiation Program

Two assemblies, each containing four rods, were irradiated in the X-5 loop of the NRX reactor from October 27, 1961, to February 22, 1962. The goal of this test was to determine the effect of UO_2 density on both fission gas release and sheath expansion.

The neutron flux in the reactor was such that there was little flux gradient across an assembly, and the rods in the assemblies were placed far enough from each other so that the neutron flux gradient across each rod would be minimal.

The eight rods ranged in density from 95% to 98% TD. Both assemblies were irradiated at power outputs of approximately 17 kW/ft for a total of 84.49 equivalent full-power days. The maximum rod average burnup reached was 2.76 GWd/MTU.

After irradiation, the rods were dimensioned and punctured for fission gas analysis. In addition, each rod was sectioned and examined under a stereomicroscope. Other sections were metallographically examined. Average burnup was determined from the activity of cobalt monitor wires on the surface of the middle pellet in each rod. This document summarizes the design and operating parameters of Rod CBP.

A.8.2 Rod Design and Fabrication

Rod CBP had a density of 10.66 g/cm^3 and ran at a power of 16.8 kW/ft. The fuel rod design specifications are shown in Table A8.1. The NRX reactor operating conditions are shown in Table A8.2.

A.8.3 Rod Irradiation History

The rod was maintained throughout life at a constant average LHGR of 16.8 kW/ft and a flat axial profile. The input LHGR was a flat 16.8 kW/ft with a few small steps to get up to power. After 84.49 full days at power, the rod achieved a burnup of 2.61 GWd/MTU.

A.8.4 PIE Results

The major PIE results are listed in Table A8.3. Gas release and grain growth variations were attributed solely to the change in thermal conductivity of the fuel with density. Thus fission gas release will increase with decreasing density.

It was found that, in general, the higher density rods exhibited a greater diametral expansion while axial expansion was independent of density.

Table A8.1. Fuel Rod Components and Specifications for NRX Test Rod CBP

Fuel Rod	
Overall length (in.)	33.77
Diameter (in.)	0.6885
Fuel stack height (in.)	9.57
Nominal plenum chamber length (in.)	24.2
Number of pellets per rod	Not specified
Fuel/clad diametral gaps (mils)	12
Fill gas composition	
Fill gas pressure (atm)	
Cladding	
Material	Zircaloy-2
Diameter outside (in.)	0.6885
Diameter inside (in.)	0.6485
Clad wall thickness	0.0200
Ovality	Not specified
Eccentricity	Not specified
Grain size	Not specified
Fuel	
Material	UO ₂
Enrichment (%)	4.5
Pellet sintered density (% TD)	97.2
Pellet diameter (in.)	0.6395
Pellet geometry	Dished
Dish depth (in.)	0.01
Grain size (μm)	20

Table A8.2. NRX Operating Conditions

Parameter and Units	Value
Loop	X-5 in NRX reactor
Operating pressure (psi)	1178
Inlet temperature (°F)	480
Water flow (L/s)	5.045
Reactor power (MW)	40

Table A8.3. PIE Results

Parameter and Units	Measured Value	Predicted Value
Burnup (Gwd/MTU)	2.61	2.61
Fission gas release at EOL (% of produced gas)	14.1	11.8
Length change (mm)	+0.28	--
Diametral change (mm)	+0.051	--

A.8.5 FRAPCON-3 Input

The recommended FRAPCON-3 input deck for Rod CBP is listed in Table A8.4. The rod design parameters and power history are both drawn from Notley et al. (1965).

Table A8.4. FRAPCON-3 Input for NRX Test Rod LFF and Notes on Derivation

Input Variable	Meaning and Units	Input Value	Calculations and Notes	Reference
cpl	Cold plenum length (in.)	24.2	Assumed	Notley et al. 1965
crdt	Initial crud thickness (mils)	1.0	Assumed initial crud	Notley et al. 1965
dco	Cladding outer diameter (in.)	0.6885	Design value	Notley et al. 1965
thkcld	Cladding wall thickness (in.)	0.02	Design value	Notley et al. 1965
pitch	Rod-to-rod pitch (in.)	0.90	Arbitrary value - boiling condition	
den	Fuel pellet density (% of theoretical)	97.2	Design value	Notley et al. 1965
dishsd	Dish shoulder width (in.)	0.18675	Assumed, given dish depth	Notley et al. 1965
thkgap	Gap radial thickens (in.)	0.009	Design value	Notley et al. 1965
dspg	Spring diameter (in.)	0.64	Assumed	--
dspgw	Spring wire diameter (in.)	0.055	Assumed	--
enrch	Fuel enrichment (% ²³⁵ U in U)	4.5	Design value	Notley et al. 1965
fgpav	Fill gas pressure (psi at room temp.)	146	Stated BOL value	Notley et al. 1965
hdish	Dish depth (in.)	0.01	Design value	Notley et al. 1965
hplt	Pellet length (in.)	0.6043	Design value	Notley et al. 1965
icm	Cladding type (4 = zircaloy-4)	2	Design value	Notley et al. 1965
icor	Cladding crud buildup control (= 0 for no buildup)	0	No further crud buildup assumed; little seen in metallographs	Notley et al. 1965
idxgas	Gas type (5 = argon)	1	Design value	Notley et al. 1965
iplant	Type of plant	-2		
iq	Indicator for axial power shape	0		
jdlpr	Output control	1		
jn	Number of points per axial power shape	4		
jst	Assignment of shape numbers to each time step	8*1		
totl	Fuel column length (ft)	0.6885	Design value	Notley et al. 1965
roughc	Cladding surface roughness (in.)	2.0E-5	Assumed	--
roughf	Pellet surface roughness (in.)	5.0E-5	Assumed	--
rsntr	Change in pellet density due to densification (kg/m ³)	150	Stable pellets	Notley et al. 1965
vs	Number of plenum spring turns	2.0	Assumed	--
flux	Fast neutrons (per m ² per second per W/g of fuel)	4* 1.8E16	Default value, derived for PWRs	--
p2	Reactor system pressure (psi)	1178.0	NRX conditions	Notley et al. 1965
tw	Coolant inlet temperature (°F)	480	NRX conditions	Notley et al. 1965
go	Coolant mass flow (lbm/ft ² /h)	1.9E7	Assumed	--
qf, x	Parameters of the axial flux shapes	1, 1, 1, 1	Assumed flat profile	--
qmpy, time	The LHGR history	See deck	Constant power of 16.8 kW/ft with early steps to get up to power	Notley et al. 1965

A.9 EL-4 Test Cases

A.9.1 Description of the EL-4 Test Case for PWR Rods 4110-AE2 and 4110-BE2

Sixteen cartridges, each containing two rods, were irradiated in the EL-4 PIE for a varying number of cycles to achieve burnups from 3 to 12 GWd/MTU. The aspects of the rods studied in this project were

- Macroscopic appearances: crack network, material movement, and dimensional changes
- Microscopic appearances: recrystallization, pore redistribution, and new phases
- Migration of fission products: stable gases released by the fuel and distribution of solid fission products.

Each cartridge was constructed of zircaloy-2 and consisted of two separate stages, each containing a stack of UO_2 fuel 123 mm high at each end, and in the central joint, space was provided for cobalt flux indicators. Each stage contained a chromel-alumel thermocouple located in the center of the stack. The cartridges were then filled with helium.

After irradiation, the rods were dimensioned and punctured for fission gas analysis. Gamma scans were done as well as a radiochemical analysis. This document summarizes the design and operating parameters of Rods 4110-AE2 and 4110-BE2.

A.9.2 Rod Design and Fabrication

Both Rods 4110-AE2 and 4110-BE2 contained fuel pellets with an as-fabricated density of 10.52 g/cm^3 . AE2 ran at a power of 17.6 kW/ft while BE2 ran at a power of 17.8 kW/ft. The fuel rod design specifications for 4110-AE2 are shown in Table A9.1. The design specifications for 4110-BE2 are shown in Table A9.2. The EL-4 reactor operating conditions are shown in Table A9.3.

A.9.3 Rod Irradiation History

Rods 4110-AE2 and 4110-BE2 were maintained throughout life at constant average LHGRs of 17.6 and 17.8 kW/ft, respectively. Both ran with a flat axial power profile. The input LHGRs were a flat 17.6 and 17.8 kW/ft with a few steps to get up to power.

Table A9.1. Fuel Rod Components and Specifications for EL-4 Rod 4110-AE2

Fuel Rod	
Overall length (in.)	43.04
Diameter (in.)	0.5564
Fuel stack height (in.)	4.84
Nominal plenum chamber length (in.)	28.2
Number of pellets per rod	Not specified
Fuel/clad diametral gaps (mils)	8
Fill gas composition	He
Fill gas pressure (atm)	10
Cladding	
Material	Zircaloy-2
Diameter outside (in.)	0.5564
Diameter inside (in.)	0.5164
Clad wall thickness	0.0200
Ovality	Not specified
Eccentricity	Not specified
Grain size	Not specified
Fuel	
Material	UO ₂
Enrichment (%)	2.98
Pellet sintered density (% TD)	96.0
Pellet diameter (in.)	0.5084
Pellet geometry	Flat ends
Dish depth (in.)	0.00
Grain size (μm)	< 160

Table A9.2. Fuel Rod Components and Specifications for EL-4 Rod 4110-BE2

Fuel Rod	
Overall length (in.)	42.84
Diameter (in.)	0.5564
Fuel stack height (in.)	4.84
Nominal plenum chamber length (in.)	38.0
Number of pellets per rod	Not specified
Fuel/clad diametral gaps (mils)	7
Fill gas composition	He
Fill gas pressure (atm)	10
Cladding	
Material	Zircaloy-2
Diameter outside (in.)	0.5574
Diameter inside (in.)	0.5164
Clad wall thickness	0.0200
Ovality	Not specified
Eccentricity	Not specified
Grain size	Not specified
Fuel	
Material	UO ₂
Enrichment (%)	2.98
Pellet sintered density (% TD)	96.0
Pellet diameter (in.)	0.5104
Pellet geometry	Flat ends
Dish depth (in.)	0.00
Grain size (μm)	< 160

Table A9.3. NRX Operating Conditions

Parameter and Units	Value
Run	6 cycles in EL-4 reactor
Operating pressure (psi)	100
Inlet temperature (°F)	630

A.9.4 PIE Results

The major PIE results are listed in Table A9.4. Fission gas release was found to be similar in these rods with similar design and operating history. However, an application of pressure on the fuel does not seem to have any influence on the gas release. It was found that for a burnup of 6 GWd/MTU, the release fraction is almost of the same order of magnitude as the fraction of the fuel volume raised above 1,700°C.

Table A9.4. PIE Results

Pameter and Units	Measured Value	Predicted Value
Fission gas release at EOL (% of produced gas)4110-AE2	22.1	18.4
Fission gas release at EOL (% of produced gas)4110-BE2	15.9	20.6

It was also found that the recrystallization modes are clearly influenced by the initial gap. In addition, the thermal conductivity of UO₂ with regards to the increase in density in the recrystallized region reasonably accounts for temperature distribution.

A.9.5 FRAPCON-3 Input

The recommended FRAPCON-3 input for Rod 4110-AE2 is listed in Table A9.5. The input for 4110-BE2 is listed in Table A9.6. The rod design parameters and power history are both drawn from Janvier (1967).

Table A9.5. FRAPCON-3 Input for EL-4 Rod 4110-AE2 and Notes on Derivation

Input Variable	Meaning and Units	Input Value	Calculations and Notes	Reference
cpl	Cold plenum length (in.)	38.2 (AE2) 38.(BE2)	Assumed	Janvier 1967
crdt	Initial crud thickness (mils)	0.0	No initial crud	Janvier 1967
dco	Cladding outer diameter (in.)	0.5564 (AE2) 0.5574 (BE2)	Design value	Janvier 1967
thkcld	Cladding wall thickness (in.)	0.020	Design value	Janvier 1967
pitch	Rod-to-rod pitch (in.)	0.65	Arbitrary value - boiling condition	
den	Fuel pellet density (% of theoretical)	96.0	Design value	Janvier 1967
dishsd	Dish shoulder width (in.)	0.1214 (AE2) 0.1224 (BE2)	Assumed, given dish depth	Janvier 1967
thkgap	Gap radial thickens (in.)	0.0080 (AE2) 0.0070 (BE2)	Design value	Janvier 1967
dspg	Spring diameter (in.)	0.51	Assumed	--
dspgw	Spring wire diameter (in.)	0.055	Assumed	--
enrch	Fuel enrichment (% U-235 in U)	2.98	Design value	Janvier 1967
fgpav	Fill gas pressure (psi at room temp.)	147.0	Stated BOL value	Janvier 1967
hdish	Dish depth (in.)	0.0	Design value	Janvier 1967
hplt	Pellet length (in.)	0.6043	Design value	Janvier 1967
icm	Cladding type (2 = zircaloy-2)	2	Design value	Janvier 1967
icor	Cladding crud buildup control (= 0 for no buildup)	0	No further crud buildup assumed; little seen in metallographs	Janvier 1967
idxgas	Gas type (5 = argon)	1	Design value	Janvier 1967
iplant	Type of plant	-2		
iq	Indicator for axial power shape	0		
jdlpr	Output control	1		
jn	Number of points per axial power shape	4		
jst	Assignment of shape numbers to each time step	11 × 1 (AE2) 9 × 1 (BE2)		
totl	Fuel column length (ft)	0.403	Design value	Janvier 1967
roughc	Cladding surface roughness (in.)	2.0E-5	Assumed	--
roughf	Pellet surface roughness (in.)	5.0E-5	Assumed	--
rsntr	Change in pellet density due to densification (kg/m ³)	150	Stable pellets	Janvier 1967
vs	Number of plenum spring turns	28.0	Assumed	--

Table A9.5. (contd)

Input Variable	Meaning and Units	Input Value	Calculations and Notes	Reference
flux	Fast neutrons (per m ² per second per W/g of fuel)	1.8E16	Default value, derived for PWRs	--
p2	Reactor system pressure (psi)	1000.0	EL-4 conditions	Janvier 1967
tw	Coolant inlet temperature (°F)	630	EL-4 conditions	Janvier 1967
go	Coolant mass flow (lbm/ft ² /h)	1.9E7	Assumed	--
qf, x	Parameters of the axial flux shapes	1,1,1,1	Assumed flat profile	--
qmpy, time	The LHGR history	See deck	Constant power of 17.6 kW/ft with early steps to get up to power	Janvier 1967

Table A9.6. FRAPCON-3 Input for EL-4 Rod 4110-BE2 and Notes on Derivation

Input Variable	Meaning and Units	Input Value	Calculations and Notes	Reference
cpl	Cold plenum length (in.)	38.0	Assumed	Janvier 1967
crdt	Initial crud thickness (mils)	0.0	No initial crud	Janvier 1967
dco	Cladding outer diameter (in.)	0.5574	Design value	Janvier 1967
thkcld	Cladding wall thickness (in.)	0.020	Design value	Janvier 1967
den	Fuel pellet density (% of theoretical)	96.0	Design value	Janvier 1967
dishsd	Dish shoulder width (in.)	0.1224	Assumed, given dish depth	Janvier 1967
thkgap	Gap radial thickens (in.)	0.007	Design value	Janvier 1967
dspg	Spring diameter (in.)	0.51	Assumed	--
dspgw	Spring wire diameter (in.)	0.055	Assumed	--
enrch	Fuel enrichment (% ²³⁵ U in U)	2.98	Design value	Janvier 1967
fgpav	Fill gas pressure (psi at room temp.)	147	Stated BOL value	Janvier 1967
hdish	Dish depth (in.)	0.0	Design value	Janvier 1967
hplt	Pellet length (in.)	0.6043	Design value	Janvier 1967
icm	Cladding type (2 = zircaloy-2)	2	Design value	Janvier 1967
icor	Cladding crud buildup control (= 0 for no buildup)	0	No further crud buildup assumed; little seen in metallographs	Janvier 1967
idxgas	Gas type (5 = argon)	1	Design value	Janvier 1967
totl	Fuel stack length (ft)	0.403	Design value	Janvier 1967
roughc	Cladding surface roughness (in.)	2.0E-5	Assumed	--
roughf	Pellet surface roughness (in.)	5.0E-5	Assumed	--
rsntr	Change in pellet density due to densification (kg/m ³)	150	Stable pellets	Janvier 1967
vs	Number of plenum spring turns	28	Assumed	--
flux	Fast neutrons (per m ² per second per W/g of fuel)	1.8E16	Default value, derived for PWRs	--
p2	Reactor system pressure (psi)	1,000	EL-4 conditions	Janvier 1967
tw	Coolant inlet temperature (°F)	630	EL-4 conditions	Janvier 1967
go	Coolant mass flow (lbm/ft ² /h)	1.9E7	Assumed	--
qf, x	Parameters of the axial flux shapes	1,1,1,1	Assumed flat profile	--
qmpy, time	The LHGR history	See deck	Constant power of 17.8 kW/ft with early steps to get up to power	Janvier 1967

A.10 Description of FRAPCON-3 Case Input for Arkansas Nuclear-2, Rod TSQ002

A.10.1 Overview of the Examined Rods

The U.S. Department of Energy sponsored a program with ABB Combustion Engineering and Energy Operations, Inc. to improve the use of PWR fuel. The scope of this project was to develop more efficient fuel management concepts and an increase in the burnup of discharged fuel.

Two 16 x 16 lead test assemblies were irradiated in the Arkansas Nuclear One-Unit 2 reactor (ANO-2). This is a PWR that operates at 2815 MW thermal. One of the assemblies, D039, was irradiated for three cycles and achieved a burnup of 33 GWd/MTU. The other assembly, number D040, was irradiated for five cycles and achieved a burnup of 52 GWd/MTU.

A.10.2 Rod Design and Fabrication

Rod TSQ002, irradiated in assembly D040, was of standard CE 16 x 16 design and contained solid uranium pellets. The design parameters are summarized in Table A10.1.

A.10.3 Irradiation Conditions and History

Assembly D040 was irradiated from 1979 to 1988 in ANO-2, cycles two through six. It accumulated 52 GWd/MTU assembly-average burnup. Rod TSQ002 accumulated an EOL rod-average burnup of 56.1 GWd/MTU. The rod-average LHGR varied from 2.75 to 6.95 kW/ft, with the higher values near BOL. Detailed cycle-by-cycle LHGR histories are found in Smith et al. (1994), Appendix G, and are reflected in the code input.

The coolant conditions during irradiation remained fairly constant; they are summarized in Table A10.2.

A.10.4 PIE Results and FRAPCON-3 Predictions

The cladding hydrogen content, corrosion layer thickness, and rod axial elongation were measured, in addition to the fission gas release and fuel density. These results are listed in Table A10.3, together with the FRAPCON-3 predictions.

In general, FRAPCON-3 adequately predicted these EOL parameters. The overprediction of hydrogen content may reflect an overestimate of the pickup fraction at high burnup.

A.10.5 FRAPCON-3 Input

An annotated listing of recommended FRAPCON-3 code input parameter values is given in Table A10.4.

Table A10.1. Design Specifications for ANO-2 Rod TSQ002

Fuel Rod	
Overall length (in.)	161.168
Diameter (in.)	0.382
Fuel stack height (in.)	150.0
Nominal plenum chamber length (in.)	11.168
Plenum spring volume (cu. in.)	0.4557
Number of pellets per rod	Not specified
Fuel/clad diametral gaps (mils)	7
Fill gas composition	He
Fill gas pressure (atm)	25.86
Cladding	
Material	Zircaloy-4
Diameter outside (in.)	0.382
Diameter inside (in.)	0.332
Clad wall thickness (in.)	0.025
Ovality	Not specified
Eccentricity	Not specified
Grain size	Not specified
Fuel	
Material	UO ₂
Enrichment (%)	3.48
Pellet sintered density (% TD)	95.0
Pellet diameter (in.)	0.325
Pellet length (in.)	0.390
Pellet geometry	Dished
Dish radius (in.)	0.123
Depth (in.)	Not specified
Grain size (μm)	35

Table A10.2. Coolant Conditions

Parameter and Units	Value
Inlet temperature (°F)	554
Coolant pressure (psia)	2,250
Coolant mass velocity (lbm/ft ² /ht)	1.9E6
Subchannel equivalent hydraulic diameter (in.)	0.5334

Table A10.3. PIE Results and FRAPCON-3 Predictions

Parameter and Units	Measured Values	Predicted Values
Fission gas release (% of produced)	< 1	1.94
Peak hydrogen content (ppm H)	367	294 to 347
Maximum waterside oxide layer thickness (microns)	53	40 to 47
Rod growth (% of initial length)	0.83 to 1.11	0.85 to 0.96
Rod-average burnup (GWd/MTU)	53.2	55
Fuel volume change (net swelling) (% dV/V)	~ 2.4	3.0

Table A10.4. FRAPCON-3 Input for ANO-2, Rod TSQ002 and Notes on Derivation

Input Variable	Meaning and Units	Input Value	Calculations and Notes	Reference
cpl	Cold plenum length (in.)	10.7	Adjusted to achieve stated BOL void volume of 34 cc	Smith et al. 1994
crdt	Initial crud thickness (in.)	0.2		Smith et al. 1994
pitch	Rod-to-rod pitch (in.)	0.56	Chosen to yield 0.5 in. stated equivalent channel diameter	--
dco	Cladding outer diameter (in.)	0.382	Design value	Smith et al. 1994
thkeld	Cladding wall thickness (in.)	0.025	Design value	Smith et al. 1994
den	Fuel pellet density (% of theoretical)	95.0	Design value	Smith et al. 1994
dishsd	Dish shoulder width (in.)	0.1	Design value	--
thkgap	Gap radial thickness (in.)	0.0035	Design value = 7.5 mils diametral	Smith et al. 1994
dspg	Spring diameter (in.)	0.33	Assumed	--
dspgw	Spring wire diameter (in.)	0.055	Assumed	--
enrch	Fuel enrichment (% ²³⁵ U in U)	3.48	Design value	Smith et al. 1994
fgpav	Fill gas pressure (psi at room temp.)	380.0	Stated BOL value	Smith et al. 1994
hdish	Dish depth (in.)	0.0135	Calculated	--
hplt	Pellet length (in.)	0.390	Design value	Smith et al. 1994
icm	Cladding type (4 = zircaloy-4)	4	Design value	Smith et al. 1994
icor	Cladding crud buildup (= 0 for no buildup)	0	No crud buildup assumed	Smith et al. 1994
idxgas	Gas type (1 = helium)	1	Design value	Smith et al. 1994
iplant	Type of plant	-2		
iq	Indicator for axial power shape	0		
jdlpr	Output control	0		
totl	Fuel column length (ft)	12.5	Design value = 150 in.	Smith et al. 1994
roughc	Cladding surface roughness (in.)	2.E-5	Assumed	--
roughf	Pellet surface roughness (in.)	3.E-5	Assumed	--
rsntr	Change in pellet density due to densification (kg/m ³)	150	Stable pellets	Smith et al. 1994
vs	Number of plenum spring turns	8	Assumed	--
flux	fast neutrons (per m ² per second per W/g of fuel)	13 × 0.26E17	Standard value	Smith et al. 1994
p2	Reactor system pressure (psi)	70 × 2,250	Stated PWR conditions	Smith et al. 1994

Table A10.4. (contd)

Input Variable	Meaning and Units	Input Value	Calculations and Notes	Reference
tw	Coolant inlet temperature (°F)	70 × 554	Stated conditions	Smith et al. 1994
go	Coolant mass flux (lbm/ft²/h)	70 × 3.0E6	Stated conditions	--
jn	Number of points per axial power shape	25,25,25,24,24		
jst	Assignment of shape numbers to each time step	9 × 1 11 × 2 17 × 3 14 × 4 19 × 5		
qf, x	Parameters of the axial flux shapes	See Smith et al. 1994		Smith et al. 1994
qmpy, time	The LHGR history	See Smith et al. (1994) App. A	Derived from the tabulated histories	Smith et al. 1994

A.11 Description of FRAPCON-3 Case Input for Oconee 5-Cycle PWR, Rod 15309

A.11.1 Overview of the Irradiation Program

The U.S. Department of Energy (DOE) sponsored a long-term, multi-organizational program on the performance of light water reactor (LWR) fuel rods during operation to extend burnups. As part of that program, Babcock and Wilcox (B&W) 15 x 15-type PWR fuel assemblies were irradiated to 3, 4, and 5 cycles in the Oconee pressurized water reactor (PWR), operated by Duke Power Company. One assembly, 1D45, completed five cycles of irradiation in June 1983, having achieved an assembly-average burnup of 50 GWd/MTU during 1,553 effective full-power days (EFPD).

Several rods from the assembly were nondestructively and destructively examined in the B&W hot cells. This document summarizes the design and operating parameters for one rod, number 15309. Fuel density and microstructure, rod growth, cladding oxidation/hydriding, and diametral strain data are available for this rod together with fission gas release measurement via rod puncture and plenum gas analysis. The fission gas release for this low-powered rod was < 1%; but the cladding oxidation, growth, and diametral strain were significant.

Further detail is provided below on the following topics:

- Rod design and fabrication
- Irradiation conditions and history
- Major PIE results.

A.11.2 Rod Design and Fabrication

The fuel rod general design specifications are shown in Table A11.1. The rods were standard 15 x 15 full-length PWR rods. The Oconee reactor operating conditions are shown in Table A11.2.

A.11.3 Rod Irradiation History

The rod initially had a rod average LHGR of 7 to 8 kW/ft; however, this decreased to ~4 kW/ft by end-of-life. The axial power profile flattened early and remained relatively flat throughout life. A plot of the input LHGR history is shown in Figure A11.1. It should be compared with the peaking factor history in the appendices of Newman (1986).

Table A11.1. Fuel Rod Components and Specifications for Oconee 5-Cycle, Rod 15309

Fuel Rod	
Overall length (in.)	153.69
Diameter (in.)	0.430
Fuel stack height (in.)	141.0
Nominal plenum chamber length (in.)	10.5
Number of pellets per rod	200
Fuel/clad diametral gaps (mils)	10
Fill gas composition	He
Fill gas pressure (atm)	32.6
Cladding	
Material	Zircaloy-4
Diameter outside (in.)	0.4300
Diameter inside (in.)	0.3770
Clad wall thickness (in.)	0.0265
Ovality	Not specified
Eccentricity	Not specified
Grain size	Not specified
Fuel	
Material	UO ₂
Enrichment (%)	3.0
Pellet sintered density (% TD)	95.0
Pellet diameter (in.)	0.3670
Pellet length (in.)	0.70
Pellet geometry	Dished both ends
Dish radius (in.)	0.1385 (assumed)
Depth (in.)	0.014 (assumed)
Grain size (μm)	9.6 - 12.8

Table A11.2. Oconee Reactor Operating Conditions

Parameter and Units	Parameter Value
System pressure (psi)	2,200
Inlet temperature (°F)	555
Outlet temperature (°F)	606
Core average linear heat rate (kW/ft)	5.78
Core average rating (MW thermal)	2,568

Figure A11.1. Input LHGR History for Oconee-1, Rod 15309

A.11.4 PIE Results

The major PIE results are listed in Table A11.3. Fission gas release from this low powered rod is minimal; however, the cladding oxidation and hydriding and the rod growth and diametral strain are significant.

A.11.5 Oconee-1, 15309 Test Case Input Model

Table A11.4 lists the background notes on Oconee PWR Rod 15309 input for FRAPCON-3.

Table A11.3. PIE Results

Parameter and Units	Measured Values	Predicted Values
Rod 15309 burnup (GWd/MTU)	50	50
Fission gas release at end of life (EOL) (% of produced gas)	0.8	1.51
Peak cladding waterside corrosion layer thickness, (microns)	64	54 to 62
Maximum measured hydrogen concentration in cladding (wall-averaged) (ppm)	408	370 to 425
Maximum diametral strain (% $\Delta D/D$ from initial value)	-0.6 to -1.0	-0.6 to -0.7
Rod growth (% $\Delta L/L$)	0.792 to 0.907	0.79

A.11.6 Bibliography (not called out references)

Newman, L. W., T. P. Papazoglou, J. T. Mayer, W. A. McInteer, T. D. Pyecha, P. L. Holman, and P. C. Aadland. 1985. *The Nondestructive Examination of an Oconee-1 Fuel Assembly after Five Cycles of Irradiation*, DOE/ET/34212-48, BAW-1818, Babcock and Wilcox, Lynchburg, Virginia.

Dideon, C. G., and G. M. Bain. 1983. *Fuel Performance Under Extended Burnup: B&W 15 x 15 Design*, BAW-1716, Babcock and Wilcox, Lynchburg, Virginia.

Table A11.4. Oconee PWR Rod 15309 Input for FRAPCON-3 and Notes on Derivation

Input Variable	Meaning and Units	Input Value	Calculations and Notes	Reference
qmpy, time cpl	Cold plenum length (in.)	10.5	Chosen based on obtaining calc. total volume equal to the 35 cc stated beginning of life (BOL) rod void volume. (This value is also reasonable relative to initial rod length, initial fuel column length, and probable end plug configuration.)	Newman 1986
crdt	Initial crud thickness (in.)	0.2	Some allowance for crud	Newman 1986
crdtr	Rate of crud accumulation	0.0		
dco	Cladding outer diameter (in.)	0.430	Design value	Newman 1986
thkcld	Cladding wall thickness (in.)	0.0265	Design value	Newman 1986
pitch	Rod-to-rod pitch (in.)	0.56	Arbitrary value - boiling condition	
den	Fuel pellet density (% of theoretical)	95	Design value	Newman 1986
dishsd	Dish shoulder width (in.)	0.050	Assumed; implies a reasonable value for radius of curvature	--
thkgap	Gap radial thickness (in.)	0.0050	Design value = 10 mils diametral	Newman 1986
dspg	Spring diameter (in.)	0.37	Assumed	--
dspgw	Spring wire diameter (in.)	0.055	Assumed	--
enrch	Fuel enrichment (% ²³⁵ U in U)	3.0	Design value	Newman 1986
fgpav	Fill gas pressure (psi at room temp.)	480	Stated BOL value	Newman 1986
hdish	Dish depth (in.)	0.014	Assumed; typical value	--
hplt	Pellet length (in.)	0.70	Design value = 15.2 mm	Newman 1986
icm	Cladding type (4 = zircaloy-4)	4	Design value	Newman 1986
icor	Cladding crud buildup (= 0 for no buildup)	0	No further crud buildup assumed; little seen in metallographs	Newman 1986
idxgas	Gas type (1 = helium)	1	Design value	Newman 1986
iplant	Type of plant	-2		
iq	Indicator for axial power shape	0		
jdlpr	Output control	0		
totl	Fuel column length (ft)	11.75	Design value = 141 in.	Newman 1986
jn	Number of points per axial power shape	13,13,13, 13,13		

Table A11.4. (contd)

Input Variable	Meaning and Units	Input Value	Calculations and Notes	Reference
jst	Assignment of shape numbers to each time step	7 × 1 10 × 2 2 × 3 5 × 4 10 × 5		
rc	Inner pellet radius	0.0		
roughc	Cladding surface roughness (in.)	1.97E-5	Assumed	--
roughf	Pellet surface roughness (in.)	2.36E-5	Assumed	--
rsntr	Change in pellet density due to densification (kg/m ³)	150	Stable pellets	Newman 1986
vs	Number of plenum spring turns	20.0	Assumed	--
flux	fast neutrons (per m ² per second per W/g of fuel)	13 × 0.25E17	Default value, derived for PWRs	Newman 1986
p2	Reactor system pressure (psi)	2,200.0	Oconee conditions	Newman 1986
tw	Coolant inlet temperature (°F)	555.0	Oconee conditions	Newman 1986
go	Coolant mass flux (lbm/ft ² /h)	2.6E6	Calculated assuming: flow velocity = 15 ft/sec coolant density = 48 lb/ft ³ flow area = pitch ² - rod area	--
qf, x	Parameters of the axial flux shapes	Newman 1986	Shape flattens with burnup	Newman 1986
qmpy, time	The LHGR history	Newman 1986	Derived from the peaking history in Newman (1986)	Newman 1986

A.12 Monticello BWR Rod, Corner Position A1

A.12.1 Description of the Monticello Surveillance Program

A DOE program was completed in 1985 in which nine 8 x 8 fuel assemblies in the Monticello boiling water reactor (BWR) were taken to high burnup (up to 45.6 MWd/MTM assembly-average), and the rods were periodically examined nondestructively and sampled for destructive examinations. Four of the assemblies went for the "full term" from Cycle 3 through Cycle 9, from May 1974 to September 1982.

Because the interim measurements included diametral (creepdown) measurements, and because these fuel rods have typically fully annealed Zircaloy-2 cladding, they were used in verification cases for the cladding creepdown model developed for FRAPCON-2 and used in FRAPCON-3. These rods are also cases for verification of the MASSIH gas release model as well, since they provide fission gas release (FGR) data on the rods' relatively detailed power histories. It is interesting to note that the rods divide into two populations with respect to FGR: the peripheral and especially the corner rods have high FGR (from 8-28%), whereas interior rods uniformly had low FGR (< 3%) even though the quoted power histories between the corner rods and interior rods were not significantly different. The corner rod position A1 is next to the control blade and the power changes in the corner rod due to control blade movement are not accurately determined.

The general design, power history, and PIE results are documented in a series of DOE/GE reports (Baumgartner 1983a, 1983b, 1984; West et al. 1983). For the detailed power/power profile history, however, it was necessary and efficient for PNNL to obtain magnetic tape compilations (24 node x 60-70 step histories) from the Electric Power Research Institute (EPRI) Fuel Performance Data Base (FPDB) for the two rods we selected for study. EPRI graciously supplied these data at minimal cost through Saul Levy, Inc. (the FPDB administrator) and their public outlet, Radford Associates.

There were other necessary details (such as axial-dependent void fraction and axial-dependent fast-flux/power ratio), which were supplied directly by GE through the program manager, Mr. John Baumgartner.

This section describes the assemblies/rods in some detail, indicates the complexity of their power histories, and documents the FRAPCON-3 input generated for the two rods selected for analysis. The PIE results are also referenced and discussed.

A.12.2 Design and Operation of the Rods

The four "full-term" assemblies were identical rod-for-rod in design and were very nearly identical in power history, as they occupied symmetric core positions through life. In Table A12.1, the calculated end-of-cycle assembly average burnups are shown (same for all four assemblies) together with the discharge dates. As noted, MTB072 was discharged after Cycle 7, MTB071 after Cycle 8, and MTB099/MTB048 after Cycle 9.

Table A12.1. Summary of Lead Burnup Fuel Bundle Operation

Cycle	Date	BOC Bundle Average Burnup (GWd/MTU)	Discharged
BOC	May 21, 1974		
EOC	January 9, 1975	3.9	--
BOC	February 7, 1975		
EOC	September 11, 1975	7.8	--
BOC	November 19, 1975		
EOC	September 9, 1977	20.5	--
BOC	November 10, 1977		
EOC	October 13, 1978	26.0	--
BOC	November 16, 1978		
EOC	February 22, 1980	33.8	MTB072
BOC	April 5, 1980		
EOC	April 20, 1981	39.6	MTB071
BOC	May 11, 1981		
EOC	September 1, 1982	45.6	MTB048 and MTB099

We have chosen to study rod position A1; however, rod position F6 is included in the discussions to capture the range of behavior (A1 is a corner position and F6 an internal position). These are representative of the two populations of rods for these positions. We received MTB099 power histories through Cycle 8 from EPRI/SAI (Scientific Advances, Inc.). In Table A12.2 and Figure A12.1a,b, we

Table A12.2. Power History Summaries for Rods A1 and F6, MTB099

Input Step Numbers	Cumulative Time (days)	Power (W/cm)			
		A1 - Peak	A1 - Average	F6 - Average	F6 - Average
1-4	0-52	340	100	230	150
5-6	105	280	227	205	165
6-9	144	290	210	240	175
10-12	200	300	120	230	159
13-15	262	270	205	200	155
16-17	369	240	90	230	157
18-20	487	290	195	230	160
21-30	801	250	200	240	160
31-33	1010	180	66	157	133
34-40	1200	166	200	180	145
41-43	1272	130	50	150	123
44-61	1696	210	160	175	140

Figure A12.1. Peak Power Histories for (a) Rod A1 and (b) Rod F6

summarize the average/peak power histories for these two rods. Note that the ratio of peak-to-average power varies through life (related to control rod positioning). The axial position of the power peak also varies, as shown in Figures A12.2 and A12.3.

The geometric and material design of the rods is that of standard GE-BWR fuel assemblies and is shown in Table A12.3. General operating conditions are summarized in Table A12.4. Specific operating conditions, which are highly dependent on axial node position, are the fast flux-to-power ratio and the water density or void fraction. These are shown in Table A12.5.

A.12.3 Monticello, A1 Test Case Input Model

Some approximations and assumptions had to be made to translate the above information into FRAPCON-3 input. These are documented below.

Assumptions and Approximations

The power history as-delivered from EPRI consisted of 61 time steps and a 24-node axial power profile unique for each step. This could not be input directly into FRAPCON-3 since the code is limited to

Figure A12.2. Axial Position of the Power Peak for Rod A1

Figure A12.3. Axial Position of the Power Peak for Rod F6

Table A12.3. Original Design Characteristics of GE Fuel Bundles

Active fuel length (in.)	144.0
Plenum length (in.)	11.24
Fuel pellet form	Flat ended with chamfered edges
Fuel pellet diameter (in.)	0.416
Tube outside diameter (in.)	0.493
Tube wall thickness (in.)	0.034
Nominal diametral gap (in.)	0.009
Fuel enrichment (%) range	1.45 - 2.87 wt% ²³⁵ U
Average fuel enrichment	2.62
Number of gadolinium bearing rods	4
Fill gas	Helium (99.9%)
Fill gas pressure (psig)	15.0
Rod-to-rod pitch (in.)	0.64

Table A12.4. Operating Conditions for the Monticello Assemblies

Parameter and Units	Value
System pressure (psia)	1,038
Coolant inlet temperature (°F)	516
Coolant temperature rise (°F)	30
Fast neutron flux (N/cm ² -sec)	See Table A12.5
Water density (fractional)	See Table A12.5
Water level (position)	Top node 5 (of 24 nodes)
Mass flow rate (lb/h-ft ²)	9.92×10^5
Equivalent hydraulic diameter (in.)	0.5650

Table A12.5. Fast Neutron Flux and Fractional Water Density Versus Axial Position

Axial Node Number	Mid-Node Elevation Above Fuel Column Bottom (in.)	Fast Flux (N/cm²/sec per W/g UO₂ x 10¹²)	Fractional Water Density
1	6	2.05	1.02
2	18	2.227	1.02
3	30	2.41	1.02
4	42	2.52	0.99
5	54	2.74	0.92
6	66	2.89	0.84
7	78	3.03	0.75
8	90	3.12	0.70
9	102	3.22	0.64
10	114	3.20	0.60
11	126	3.12	0.57
12	138	2.88	0.55

a maximum of 20 axial nodes and 20 axial power shapes. Seven axial power shapes were constructed to represent the through-life variations. The original 24 axial positions (plus top/bottom end points) were input for these seven axial shapes, since the code will permit up to 40 position/power pairs for each shape, but the code was run with 12 axial nodes. In addition, the EPRI power shape data as-received were normalized to peak = 1.0; this had to be changed to average = 1.0 for FRAPCON-3 input in conjunction with the power/time history, which was in terms of rod-average power. Finally, since the original power history only proceeded to a burnup of 39 GWd/MTU (end-of-cycle 8), the history was extended to 74 steps with constant rod-average power of 4.8 kW/ft to initiate cycle 9 and achieve the measured 45 GWd/MTU burnup at end-of-cycle.

In addition to the above, the following are major input assumptions:

- The water density profile presented in Table A12.5 is constant through life (only untrue in major power changes, which are few; in any case, this distribution is only used to calculate the neutron cross-sections for use in the RADAR subroutine for radial power shapes).
- The fast flux-to-power ratio distribution in Table A12.5 is invariant through-life. (This is purported to be a fairly good assumption for calculating end-of-cycle fast fluences.)

A.12.4 Creepdown and Oxidation Data

The measured creepdown for positions A1 and F6 are summarized in Table A12.6. These values are for uncorrected oxide thickness, values which are also listed in Table A12.6.

Table A12.6. Creepdown and OD Oxidation Data

Cycle Number	Maximum Creepdown($\Delta D/D$, %)	Oxide Thickness, (mils)
3	0.05	--
4	--	--
5	0.20	--
6	--	--
7	0.30	1.0
8	--	--
9	0.37	1.5 - 2.0

The fission gas release values for rods in positions A1 and F6 are listed in Table A12.7. The FRAPCON-3 prediction of FGR at end-of-cycle 9 for position A1 is 4.4%, which is consistent with the reported power history, but much less than the measured result. Note that the large difference in gas release between the two positions may be related to the difference in position and hence in power history due to control blade movements. The enrichment and powder lot for the rods in the two positions were different, and hence it is possible that there were differences in as-fabricated parameters such as grain size, porosity, open porosity, and fuel stability against in-reactor densification, all of which could have influenced the gas release.

A.12.5 FRAPCON-3 Input

Table A12.8 presents the notes used for the Monticello BWR Rod A1 to derive the input for FRAPCON-3.

Table A12.7. Fission Gas Release Results

Position in Assembly	End of Cycle 7 (MTB071)	End of Cycle 8 (MTB072)	End of Cycle 9 (MTB099)
A1 (corner)	27.7	27.9	26.5, 30.0
F6 (interior)	0.35	0.4	0.5, 1.2

Table A12.8. Monticello BWR Rod A1 Input for FRAPCON-3 and Notes on Derivation

Input Variable	Meaning and Units	Input Value	Calculations and Notes	Reference
cpl	Cold plenum length (in.)	11.24	Chosen based on obtaining calculation of total volume equal to the 35 cc stated BOL rod void volume. (This value is also reasonable relative to initial rod length, initial fuel column length, and probable end plug configuration)	Baumgartner 1983
crdt	Initial crud thickness (mils)	0.0	Some allowance for crud	Baumgartner 1983
dco	Cladding outer diameter (in.)	0.493	Design value	Baumgartner 1983
thkelad	Cladding wall thickness (in.)	0.034	Design value	Baumgartner 1983
pitch	Rod-to-rod pitch (in.)	0.64	Arbitrary value - boiling condition	
den	Fuel pellet density (% of theoretical)	95	Design value	Baumgartner 1983
dishsd	Dish should width (in.)	0.0	Flat-ended pellets	--
thkgap	Gap radial thickness (in.)	0.0045	Design value = 9 mils diametral	Baumgartner 1983
dspg	Spring diameter (in.)	0.4	Assumed	--
dspgw	Spring wire diameter (in.)	0.04	Assumed	--
enrch	Fuel enrichment (% ²³⁵ U in U)	1.45	Design value	Baumgartner 1983
fgpav	Fill gas pressure (psi at room temp.)	14.7	Stated BOL value	Baumgartner 1983
hdish	Dish depth (in.)	0.0	Flat-ended pellets	--
hplt	Pellet length (in.)	0.5	Design value	Baumgartner 1983
icm	Cladding type (5 = recrystallized zircaloy-2)	5	Design value	Baumgartner 1983
icor	Cladding crud buildup (= 0 for no buildup)	2	Crud buildup assumed	Baumgartner 1983
idxgas	Gas type (1 = helium)	1	Design value	Baumgartner 1983
iplant	Type of plant	-2		
iq	Indicator for axial power shape	0		
ivoid		1		
jdlpr	Output control	0		
jn	Number of points per axial power shape	26,26,26,26, 26,26,26		

Table A12.8. (contd)

Input Variable	Meaning and Units	Input Value	Calculations and Notes	Reference
jst	Assignment of shape numbers to each time step	6 × 1, 3 × 2, 8 × 1, 5 × 3, 1, 2 × 4, 1, 4, 1, 2 × 5, 5 × 1, 5 × 6, 3 × 1, 10 × 7, 1 × 3, 18 × 7		
totl	Fuel column length (in.)	12.0	Design value	Baumgartner 1983
roughc	Cladding surface roughness (in.)	1.97E-5	Assumed	--
roughf	Pellet surface roughness (in.)	8.3E-5	Assumed (ground pellets)	-
rsntr	Change in pellet density due to densification (kg/m ³)	150	Stable pellets (assumed)	Baumgartner 1983
vs	Number of plenum spring turns	100	Assumed	--
flux	Fast neutrons (per m ² per second per W/g of fuel)	1.95E16 2.20E16 2.40E16 2.50E16 2.60E16 2.80E16 2.95E16 3.08E16 3.17E16 3.21E16 3.15E16 2.95E16	Communication with GE	(a)
p2	Reactor system pressure (psi)	1,037.0	Monticello conditions	Baumgartner 1983
tw	Coolant inlet temperature (°F)	516	Monticello conditions	Baumgartner 1983
rhoh2o	relative water density (1.0 = 62.4 lbm/ft ³)	1.0241, 1.022, 1.017, .99, .92, .84, .75, .70, .64, .60, .575, .550	Communication from GE	(a)
go	Coolant mass flux (lbm/ft ² /h)	9.92E5		Baumgartner 1983
qf, x	Parameters of the axial flux shapes	Baumgartner 1983	Shape flattens with burnup, but returns to skewed at start of each cycle	Baumgartner 1983
qmpy, time	The LHGR history	Baumgartner 1983	Derived from the reactor power and assembly peaking factor history; extended to include cycle 9	Baumgartner 1983

(a) Correspondence, J. A. Baumgartner (GE - San Jose) to D. D. Lanning (PNNL), January 7, 1985.

A.13 TVO, H8/36-6 Test Case

A.13.1 Overview of the Irradiation Program

Battelle, Pacific Northwest Laboratories (BNWL) administered the international group-sponsored High Burnup Effects Program (HBEP), which continued from 1978 to 1990. The objectives of the HBEP were to determine the effects of extended burnup on fuel rod performance, especially fission gas release. A variety of test rods and commercial power reactor rods was irradiated and examined under the HBEP, including nine full-length 5- and 6-cycle rods from the TVO-1 boiling water reactor (BWR) in Finland. This section contains the design, operation, and FRAPCON-3 simulation input data of one of these nine rods: rod number H8/36-6 from 5-cycle fuel assembly 6116.

The rod occupied position H8, which was the control blade corner position (see Figure A13.1). The rod-average burnup at EOL was 44.6 GWd/MTU, with a peak value (confirmed by chemical burnup

38907144.1

Figure A13.1. Control Blade Corner Position

analysis) of 50.9 GWd/MTU. The rod-average LHGR varied between 12 and 24 kW/m (3.3 to 7.6 kW/ft), but large variations in the peak-to-average LHGR ratio occurred due to control blade movements. These variations resulted in significant local temperature variations and in a significant EOL fission gas release, determined by rod puncture to be 11.2%.

In addition to rod puncture, plenum gas analysis, and void volume measurement, the postirradiation destructive examinations of Rod H8/36-6 included fuel density and burnup measurements, fuel ceramography, cladding metallography, scanning electron microscope examination, and determination of radial xenon cesium and neodymium distributions via electron microprobe. The total xenon concentration was also measured by x-ray fluorescence.

The nondestructive examinations included rod profilometry and length measurement and partial-length gamma scanning.

These examinations made it possible to determine the rod's total FGR and to qualitatively determine where in the fuel the gas was released.

Further detail is provided below on the following topics:

- Rod design and fabrication
- Irradiation conditions and history
- Major PIE results.

The final sections include the FRAPCON-3 used to simulate the irradiation history of this rod and a comparison of the FRAPCON-3 predictions to the measured data.

A.13.2 Rod Design and Fabrication

The fuel rod general design specifications are shown in Table A13.1. The rods were standard 8 x 8 full-length BWR rods, manufactured by ABB-Atom. The helium fill gas pressure was ~4 atmospheres, and the fuel-cladding diametral gap was a standard value of 0.0083 inches.

Table A13.1. Fuel Rod and Component Design Parameters

Fuel Rod	
Overall length (in.)	150
Diameter (in.)	0.4626
Fuel stack height (in.)	144.9
Nominal plenum chamber length (in.)	7.69
Number of pellets per rod	
Fuel/clad diametral gaps (mils)	8.23
Fill gas composition	He
Fill gas pressure (atm)	3.85
Cladding	
Material	Zircaloy-2
Diameter outside (in.)	0.4626
Diameter inside (in.)	0.3996
Clad wall thickness (in.)	0.0315
Ovality	Not specified
Eccentricity	Not specified
Fuel	
Material	UO ₂
Enrichment (%)	1.39
Pellet sintered density (% TD)	95.7
Pellet diameter (in.)	0.3913
Pellet length (in.)	
Pellet geometry	Dished both ends
Dish radius (in.)	0.098
Depth (in.)	0.0039
Grain size (μm)	6.6

The operating conditions for the TVO-1 reactor are summarized in Table A13.2. It is a standard 716 MWe BWR. Fluence estimates or data were not directly available; however, based on the information we received from the Monticello BWR, the fast flux ratio probably varied from 1.0 times the normal PWR value in the lower 1/3 of the rod length to ~1.5 times that value over the upper 2/3 of the length.

Table A13.2. TVO-1 Reactor Operating Conditions

Parameter and Units	Parameter Value
System pressure (psi)	1015
Inlet temperature (°F)	523
Outlet temperature (°F)	547
Fast flux	Not given
Core total coolant mass flow rate (kg/s)	7250

A.13.3 Rod Irradiation History

Assembly 6116 operated for five commercial cycles in the TVO-1 reactor, from June 1980 to June 1985. The power history that TVO provided for Rod H8/36-6 is summarized in Figure A13.2, and the input history is shown in Figure A13.3. The rod had its highest rod-average LHGR values in the first cycle (about 24 kW/m), then had much lower LHGRs in the second cycle (~12 kW/m), followed by an increase back to 18 kW/m at the start of the third cycle and a slow decline thereafter for the remainder of the irradiation.

Figure A13.2. Reported Power History for Rod H8/36-6

Figure A13.3. Input Power History for Rod H8/36-6

Representative axial LHGR profiles for Assembly 6116 are shown in Figure A13.4. The profiles shifted from peaking at the bottom of the rod at BOL to peaking at the top at EOL.

A.13.4 PIE Results and FRAPCON-3 Predictions

The major PIE results are listed in Table A13.3, together with the FRAPCON-3 predictions. Fission gas release is surprisingly high at 11.2%, considering the low values of the rod-average LHGRs. The axial-peak burnup is just over 50 GWd/MTU. The rod growth (0.3%) is consistent with the exposure and the cladding type. The degree of apparent swelling is consistent with normal (solid fission product-induced) swelling, starting with high-density, stable fuel pellets.

Except for the fission gas release, FRAPCON-3 predicts the PIE results fairly well for this rod. The underprediction of the fission gas release may be related to an incomplete detailed power history for this very complex operational position. The prediction for the hydrogen uptake, however, should prompt closer evaluation of the BWR hydrogen uptake model, which has not been changed from the original MATPRO model.

Figure A13.4. Representative Axial LHGR Profiles for Assembly 6116

Table A13.3. PIE Results

Parameter and Units	Measured Values	Predicted Values
Fission gas release at EOL (% of produced gas)	11.2	0.8
Peak cladding waterside corrosion layer thickness (microns)	12-28 uniform, 60-120 at nodules	19
Maximum measured hydrogen concentration in cladding wall averaged (ppm)	not measured; light hydriding observed	680
Maximum diametral strain (% $\Delta D/D$ from initial value)	~ 0 (and minimal ridging)	--
Change in fuel density (% TD)	3.1	2.7
Rod axial growth (% $\Delta L/L$)	0.3	0.25

A.13.5 FRAPCON-3 Input

The major FRAPCON-3 input parameters are shown and annotated in Table A13.4. The input LHGR values have been increased by a factor of 1.15 over those given in Hallstadius (1989) to bring the peak predicted burnup (at 2200 mm above the bottom of the fuel column) into agreement with the measured burnup from a fuel sample taken at that position.

Table A13.4. TVO-1 BWR Rod H8/36-6 Input for FRAPCON-3 and Notes on Derivation

Input Variable	Meaning and Units	Input Value	Calculations and Notes	Reference
cpl	Cold plenum length (in.)	7.03	Adjusted slightly to get calc. total void volume to match as-fabricated value of 30.9 cm ³	Cunningham et al. 1990; Hallstadius 1989
crdt	Initial crud thickness (in.)	1	Very low crud found on rods	Cunningham et al. 1990
dco	Cladding outer diameter (in.)	0.4626	Design value = 11.85 mm	Cunningham et al. 1990
pitch	Rod-to-rod spacing (in.)	0.56	Not given; literature value for TVO-1 for 8 x 8 fuel assembly used (1.63 cm)	--
thkcl	Cladding wall thickness (in.)	0.0315	Design value = 1.60 mm	Cunningham et al. 1990
den	Fuel pellet density (% of theoretical)	95.5	Design value = 95.5; as-fabricated = 95.7	Cunningham et al. 1990
dishsd	Dish shoulder width (in.)	0.097	Dish diameter = 5.0 mm, pellet diameter = 9.94 mm	Cunningham et al. 1990
thkgap	Gap radial thickness (in.)	0.004135	Design value = 210 microns. Fuel pellet OD and cladding ID as-fabricated values very close to nominal. Minimum/maximum gaps would only be 10-20 microns different than nominal (in diametral measure)	Cunningham et al. 1990
dspg	Spring diameter (in.)	0.395	Design value	Hallstadius 1989
dspgw	Spring wire diameter (in.)	0.039	Design value	Hallstadius 1989
enrch	Fuel enrichment (% ²³⁵ U in U)	1.38	Design value	Cunningham et al. 1990
fgpav	Fill gas pressure (psi at room temp.)	56.6	Stated BOL value = 3.6 atmos. at 0 °C	Cunningham et al. 1990
hdish	Dish depth (in.)	0.0135	Stated design value = 0.1 mm	--
hplt	Pellet length (in.)	0.390	Design value	Hallstadius 1989
icm	Cladding type (2 = zircaloy-2)	2	Design value	Cunningham et al. 1990

Table A13.4. (contd)

Input Variable	Meaning and Units	Input Value	Calculations and Notes	Reference
icor	Cladding crud buildup (= 0 for no buildup)	2	Small crud allowed	--
idxgas	Gas type (1 = helium)	1	Design value	Cunningham et al. 1990
iplant	Type of plant	-2		
iq	Axial power shape	0		
jdlpr		0		
totl	Fuel column length (ft.)	12.075	Design value = 3,680 mm	Cunningham et al. 1990
roughc	Cladding surface roughness (in.)	1.978E-5	Nominal value	Hallstadius 1989
roughf	Pellet surface roughness (in.)	8.5E-5	Nominal value	Hallstadius 1989
rsntr	Change in pellet density due to densification (kg/m ³)	47	Stable pellets; resinter density change = 0.43% TD	Cunningham et al. 1990
vs	Number of plenum spring turns	20	Design value	Hallstadius 1989
flux	fast neutrons (per m ² per second per W/g of fuel)	1.8E16	Average value for in-water portion based on Monticello and PWR information	Cunningham et al. 1990
p2	Reactor system pressure (psi)	1015.0	TVO-1 conditions	Cunningham et al. 1990; Hallstadius 1989
tw	Coolant inlet temperature (°F)	523	TVO-1 conditions	Cunningham et al. 1990; Hallstadius 1989
go	Coolant mass flux (lbm/ft ² /h)	0.0	Mass flow quoted at an average of 1,240 kg/m ² -s	Hallstadius 1989
jn	Number of points per axial power shape	13,13,13, 13,13,13		
jst	Assignment of shape numbers to each time step	6 × 1 6 × 2 6 × 3 6 × 4 6 × 5 7 × 6		
qf, x	Parameters of the axial flux shapes	Cunningham et al. 1990	Assumed assembly-average flux profiles. May be steeper for Rod H8 (see Hallstadius 1989)	Cunningham et al. 1990; Hallstadius 1989
qmpy, time	The LHGR history	Cunningham et al. 1990	Increased by 15% over reference values to normalize to the measured peak burnup	Cunningham et al. 1990; Hallstadius 1989

A.14 Description of FRAPCON-3 Case Input for Ramped HBEP Obrigheim/ Petten Rodlets D200 (25 GWd/MTU and D226 (45 GWd/MTU)

A.14.1 Overview of the Test Rods

The HBEP was an international, group-sponsored program administrated by BPNL from 1979 to 1989. The objective was to investigate the impact of extended burnup upon fuel rod performance, especially fission gas release. A total of 81 rods was irradiated and examined under the program, of both BWR and PWR types, with rod-average burnups ranging up to 69 GWd/tU and peak pellet burnups up to 83 GWd/tU.

Under Task 2 of the program, full-length segmented rods were irradiated in commercial power reactors and then subjected to power ramps in test reactors. The rod segments comprised "rodlets" that were individual short-length fuel rods, mated end-to-end to form the full-length rods. Following irradiation to a variety of burnup levels, the rods were disassembled into the individual rodlets, and the rodlets were ramp-tested in test reactors. The peak LHGRs in these ramps ranged from 35 to 50 kW/m, and hold times ranged from 48 to 196 hours. The FGR during bumping was a function of the peak LHGRs and ranged from 10 to 45%. The pre-bump LHGRs ranged from 15 to 35 kW/m, as confirmed by calibrated nondestructive ⁸⁵Kr activity determinations for the plenum gas, and the pre-bump FGRs were generally low (1 to 5%).

Two PWR-type ramped rodlets were chosen for comparison to FRAPCON-3 predictions. Both were fabricated by Kraftwerk Union (KWU), irradiated in the same fuel assembly in the Obrigheim PWR, Germany, and then power-ramped to approximately the same peak LHGR (41 to 43 kW/m) in the JRC-Petten test reactor, The Netherlands. The power histories of these rodlets are shown in Figure A14.1; Rodlet D200 attained 25 GWd/tU burnup in one reactor cycle at LHGRs of 25 ± 2 kW/m. Rodlet D226 attained 45 GWd/tU by further irradiation in the same assembly for two more cycles, with LHGR generally decreasing with time from 25 kW/m at BOL to ~17 kW/m during the final cycle.

The post-bump FGR is greater for the higher-burnup Rodlet D226 than for Rodlet D200 (44 vs. 38%), in spite of the fact that D226 had a smaller as-fabricated fuel cladding gap. The pre-ramp FGRs, based on krypton-85 activity in the plenums, were very similar: 4.2 and 6.6%, respectively. Therefore, the net FGR during ramping is greater for Rodlet D226, and this was attributed to burnup effects. This rodlet pair thus provides a test of the burnup effects inherent in the modified FRAPCON-3 code and its new FGR model, MASSIH.

A.14.2 Test Rod Design and Fabrication

The segmented PWR rods containing Rodlets D200 and D226 were designed in basic conformance with KWU's 15 x 15 PWR fuel design. The general design specifications are given in Table A14.1. The fuel rodlet length was short, 39 cm overall and 31.8 cm active fuel length, to match well within the short length of the Petten reactor core. The diametral fuel-cladding gap size for Rodlet D200 was 233 microns, but only 170 microns for D226.

Figure A14.1. Power Histories, (a) D200 and (b) D226

Table A14.1. General Design Specifications for KWU/Obrigheim HBEP Rodlets D200 and D226

Fuel Rod	
Overall length (in.)	15.35
Diameter (in.)	0.4236
Fuel stack height (in.)	12.5
Nominal plenum chamber length (in.)	2.5
Number of pellets per rod	29
Fuel/clad diametral gaps (mils)	9.5 (D200) 6.8 (D226)
Fill gas composition	He
Fill gas pressure (atm)	20.7
Cladding	
Material	Zircaloy-4
Diameter outside (in.)	0.4236
Diameter inside (in.)	0.3655
Clad wall thickness (in.)	0.0291
Ovality	Not specified
Eccentricity	Not specified
Grain size	Not specified
Fuel	
Material	UO ₂
Enrichment (% ²³⁵ U)	3.2
Pellet sintered density (% TD)	94.3
Pellet diameter (in.)	0.3559 (D200) 0.3587 (D226)
Pellet length (in.)	0.44
Pellet geometry	Dished and chamfered, both ends
Dish radius (in.)	Total volume is 28 mm ³ (D200) 16 mm ³ (D226)
Grain size (μm)	5.5

The fuel pellet density is slightly low (94% TD), and the standard KWU densification test is only 2.2 hours at 1700°C. Therefore the quoted maximum densification for this fuel (0.7%) may be low—it may instead be as great as 2%, and the latter figure is used as FRAPCON-3 input.

A.14.3 Test Rod Design and Fabrication

The operating conditions in the Obrigheim PWR are relatively standard for power reactors and are listed in Table A14.2. The peak-to-average ratio for these short segments was 1.08 or less. Power histories are shown in Figure A14.1.

Table A14.2. Obrigheim Reactor Operating Conditions

Parameter and Units	Value
Power level (MW)	1045
Coolant inlet/outlet temperature (°C)	283/312
Coolant velocity (m/s)	3.39
System pressure (bar)	145.2
Fast neutron flux (> 1 MeV) (N/cm ² /s)	0.8×10^{14}

The conditions during bumping in the Petten facility are listed in Table A14.3, and a sketch of a typical ramping plan is shown in Figure A14.2.

Table A14.3. JRC-Petten Ramp Facility Conditions

Parameter and Units	Value
Coolant inlet/outlet temperature (°C)	338°C (cladding surface temperature)
Coolant pressure (bar)	145
Peak-to-average LHGR ratio	1.09

Figure A14.2. Sketch of a Typical Ramping Plan

A.14.4 Measured and Predicted FGR at EOL

The KWU rodlets were measured extensively after irradiation, including pre/post ramp ^{85}Kr plenum activity determination, pre/post ramp gamma scans and profilometry, post-ramp puncture and gas analysis, and sectioning and ceramography. A major conclusion from the references, based on radial xenon distribution determinations on cross sections of bumped rods, including these rods, was that the onset of grain boundary bubble precipitation and gas release within the pellet occurred at a fuel local temperature of $1100 \pm 100^\circ\text{C}$. This is highly consistent with the way the MASSIH fission gas release model performs in FRAPCON-3. Grain growth was observed to have occurred as a result of the ramping; however, grain boundary sweeping was judged to not be a major gas release mechanism during the ramping. This was because of rapid gas release in the high-temperature regions and the fact that these regions were not capable of grain growth within the limited time span of the ramp.

The measured and predicted FGR data are listed in Table A14.4. In spite of the good qualitative code-data agreement regarding the temperature threshold for the onset of fission gas release, the net release upon bumping is underpredicted for these rods. This may be because of the relatively unstable nature of this fuel with regard to densification and porosity redistribution.

The measured fractional fuel pellet radius at which the onset of grain growth was discernible was 0.55 for Rod D200 and 0.7 for Rod D226. This boundary is traditionally associated with approximately 1400°C . In contrast, the end-of-bump 1400°C isotherms predicted by FRAPCON-3 were at fractional pellet radii of 0.4 and 0.5 for Rods D200 and D226, respectively. This indicates that the through-bump fuel temperatures were underestimated for these rods by as much as 300°C ; this is consistent with the

Table A14.4. Measured and Predicted FGR at EOL

Rod and Condition	Measured FGR, %	Predicted FGR, %
D200 Pre-bump	4.2	0.2
D200 Post-bump	38.0	13.1
D226 Pre-bump	6.4	0.4
D226 Post-bump	44.0	27.1

underestimate of gas release. If through-bump gap dimensions were adjusted to bring the predicted fuel temperatures into agreement with the microstructure, the predicted gas release would be much closer to the measured values.

A.14.5 FRAPCON-3 Input

A listing of the recommended FRAPCON-3 input parameters for Rods D226 and D200 appear in Table A14.5. See footnotes for sources of LHGR histories.^{(a)(b)} Rod design and as-fabricated data were taken from Barner et al. (1990).

Table A14.5. Recommended FRAPCON-3 Input for HBEP/KWU Ramped Rodlets D226 and D200

Input Variable	Meaning and Units	Input Value	Calculations and Notes	Reference
cpl	Cold plenum length (in.)	0.974 (D200) 1.05 (D226)	Assumed	Barner et al. 1990
crdt	Initial crud thickness (mils)	0.0	No crud observed on test rods	(a)
dco	Cladding outer diameter (in.)	0.4236 (D200) 0.4240 (D226)	Design value	Barner et al. 1990
thkcl	Cladding wall thickness (in.)	0.02905 (D200) 0.02925 (D226)	Design value	Barner et al. 1990
pitch	Rod-to-rod pitch (in.)	0.56 (D200) 0.55 (D226)	Arbitrary value - boiling condition	
den	Fuel pellet density (% of theoretical)	94.3	Design value	Barner et al. 1990
dishsd	Dish shoulder width (in.)	0.02185	Dished pellets	Barner et al. 1990

(a) For Rod D200: HBEP-2, 12 (Fabrication); 3, 14 (Irradiation); 16, 21 (PIE); 35, 38, 42 (Special PIE).

(b) For Rod D226: HBEP-11, 12 (Fabrication); 20, 26 (Irradiation); 3, 15, 16, 22 (PIE); 35, 38, 42 (Special PIE).

Table A14.5. (contd)

Input Variable	Meaning and Units	Input Value	Calculations and Notes	Reference
thkgap	Gap radial thickens (in.)	0.00400 (D200) 0.00355 (D226)	Design value	Barner et al. 1990
dspg	Spring diameter (in.)	0.36	Assumed	--
dspgw	Spring wire diameter (in.)	0.02	Assumed	--
enrch	Fuel enrichment (% ^{235}U in U)	3.20	As-fabricated value	Barner et al. 1990
fgpav	Fill gas pressure (psi at room temp.)	304.6	As-fabricated value	Barner et al. 1990
hdish	Dish depth (in.)	0.0284		Barner et al. 1990
hplt	Pellet length (in.)	0.3588	Design value	Barner et al. 1990
icm	Cladding type (4 = zircaloy-4)	4	Design value	Barner et al. 1990
icor	Cladding crud buildup control (= 0 for no buildup)	0	No crud buildup noted	Barner et al. 1990
idxgas	Gas type (1 = helium)	1	Design value	Barner et al. 1990
iplant	Type of plant	-2		
iq	Indicator for axial power shape	0		
jdlpr	Output control	0		
totl	Fuel stack length (ft)	1.0417	Design value	--
rc	Fuel pellet inner diameter (in.)	0.0		--
roughc	Cladding surface roughness (in.)	2.5E-5	Measured	Barner et al. 1990
roughf	Pellet surface roughness (in.)	4.5e-5 (D200) 8.5E-5 (D226)	Ground to size	Barner et al. 1990
rsntr	Change in pellet density due to densification (kg/m ³)	150 (D200) 250 (D226)	Low-density pellets; KWU sintering test is inconclusive	Barner et al. 1990
vs	Number of plenum spring turns	8	Assumed	--
flux	Fast neutrons (per m ² per second per W/g of fuel)	0.18E17	Standard PWR conditions	Barner et al. 1990
p2	Reactor system pressure (psi)	2106	Obrigheim conditions	Barner et al. 1990
tw	Coolant inlet temperature (°F)	580(Steady State) 611 (Bump)	Obrigheim conditions (at the rodlet positions)	(a)
go	Coolant mass flow (lbm/ft ² /h)	0.0	Need to hold coolant temperature constant over short rodlet length	(a)
jn	Number of points per axial power shape	6,6 (D200) 6 (D226)		
jst	Assignment of shape numbers to each time step	17 × 1 (D200) 12 × 2 (D200) 45 × 1 (D226)		

Table A14.5. (contd)

Input Variable	Meaning and Units	Input Value	Calculations and Notes	Reference
qf, x	Parameters of the axial flux shapes	(a)	(a) typical core flux shape	(a)
qmpy, time	The LHGR history	(a) and (b)		(b)
(a) For Rod D200: HBEP-2, 12 (Fabrication); 3, 14 (Irradiation); 16, 21 (PIE); 35, 38, 42 (Special PIE).				
(b) For Rod D226: HBEP-11, 12 (Fabrication); 20, 26 (Irradiation); 3, 15, 16, 22 (PIE); 35, 38, 42 (Special PIE).				

A.15 Obrigheim/Petten-PK Test Cases for the Super-Ramp Rodlets PK6-2, PK6-3, and PK6-S

A.15.1 Overview of the Test Rods

The Super-Ramp project was an international group-sponsored program that involved base-irradiation of segmented full-length BWR and PWR rods in various power reactors, followed by ramp-testing of the rod segments in the Studsvik R-2 test reactor, Sweden. The purpose of the project was to establish the failure threshold for rods of varying types and burnup, and some of the rod segments did indeed fail during high-power ramp testing. Those rod segments that did not fail, however, provided data on FGR during EOL power transients.

The three rod segments selected as assessment cases here were non-failed PWR rod segments, which had been base-irradiated in the Obrigheim PWR, Germany, for three cycles up to a burnup of 34 to 37 GWd/MTU. The segments were then ramp-tested in the Studsvik reactor to ramp terminal levels (RTLs) as high as 43 kW/m. The fission gas was measured following the ramp test and ranged from 3.5 to 6.7%. These rods were selected over other rods in the test program because the fuel grain size of 22 μm is closer to the grain size of commercial fuel fabricated today with a grain size between 10 and 25 μm , and the measured fission gas following was greater than 2% release. A large portion of the fuel in the program had a small grain size of 5.5 to 6 μm , and therefore, was not selected.

A.15.2 Test Rod Design and Fabrication

The segmented PWR rods were designed in basic conformance with KWU's 15 x 15 PWR fuel design. The general design specifications are given in Table A15.1. The fuel segment length was short, 39 cm overall and 31.5 cm active fuel length, to match well within the ~1 meter active length of the Studsvik reactor core. The diametral fuel-cladding gap was 145 microns (5.7 mils).

The fuel pellet density is 95% TD, and the standard KWU densification test is only 2.2 hours at 1,700°C rather than the 24-hour densification test at 1,700°C required by the U.S. Nuclear Regulatory Commission (NRC) as a measure of maximum densification. Therefore, the quoted maximum densification for this fuel "none" may be low, and it may be as great as 1% TD—the latter figure is used as FRAPCON-3 input.

A.15.3 Irradiation Conditions and History

The operating conditions in the Obrigheim PWR are relatively standard for power reactors and are listed in Table A15.2. The peak-to-average ratio for these short segments was 1.08 or less. Power histories are shown in Figure A15.1a, b, c.

The conditions during bumping in the Studsvik facility are listed in Table A15.3, and a sketch of a typical ramping plan is shown in Figure A15.2.

Table A15.1. General Design Specifications for Obrigheim/Studsvik
Super-Ramp Rodlets PK6-2, PK6-3, and PK6-S

Fuel Rod	
Overall length (in.)	15.35
Diameter (in.)	0.4232
Fuel stack height (in.)	12.4
Nominal plenum chamber length (in.)	1.3
Number of pellets per rod	28
Fuel/clad diametral gaps (mils)	5.7
Fill gas composition	He
Fill gas pressure (atm)	22.5
Cladding	
Material	Zircaloy-4
Diameter outside (in.)	0.4232
Diameter inside (in.)	0.3658
Clad wall thickness (in.)	0.0287
Ovality	Not specified
Eccentricity	Not specified
Grain size	Not specified
Fuel	
Material	UO ₂
Enrichment (% ²³⁵ U)	3.0
Pellet sintered density (% TD)	95.0
Pellet diameter (in.)	0.3600
Pellet length (in.)	0.44
Pellet geometry	Dished and chamfered, both ends total volume of 28 mm ³ per pellet
Dish radius (in.)	0.06931
Grain size (μm)	22

Table A15.2. Obrigheim Reactor Operating Conditions

Parameter and Units	Value
Power level (MW)	1045
Coolant inlet/outlet temperature (°C)	283/312
Coolant velocity (m/s)	3.39
System pressure (bar)	145.2
Fast neutron flux (> 1 MeV) (N/cm ² /s)	0.8×10^{14}

Figure A15.1. Input Power Histories, (a) PK6-2, (b) PK6-3, and (c) PK6-S

Table A15.3. Studsvik Ramp Facility Conditions

Parameter and Units	Value
Coolant inlet/outlet temperature (°F)	550
Coolant pressure (bar)	145
Peak-to-average LHGR ratio	1.3

Figure A15.2. Typical Ramping Plan

A.15.4 FRAPCON-3 Input

A listing of the recommended FRAPCON-3 input parameters for Rods PK6-2, PK6-3, and PK6-S is shown in Table A15.4. The LHGR histories were derived from Djurle (1985), and rod design and as-fabricated data were taken from Djurle (1985).

A.15.5 Postirradiation Examination Results

The KWU rodlets were extensively measured after irradiation: ^{85}Kr plenum activity determination, pre/post ramp gamma scans and profilometry, post-ramp puncture and gas analysis, and sectioning and ceramography. Major observations are listed in Table A15.5. A major conclusion from the references was that the onset of grain boundary bubble precipitation and gas release within the pellet occurred at a fuel local temperature of $1100 \pm 100^\circ\text{C}$. This is consistent with other experimental program results and with FRAPCON-3/MASSI model predictions of the threshold temperature for gas release at nominal to high burnups. Grain growth was observed to have occurred as a result of the ramping; however, grain boundary

**Table A15.4. Recommended FRAPCON-3 Input for the Obrigheim/Studsvik
Super-Ramp Rodlets PK6-2, PK6-3, and PK6-S**

Input Variable	Meaning and Units	Input Value	Calculations and Notes	Reference
cpl	Cold plenum length (in.)	1.28	Design value	Djurle 1985
crdt	Initial crud thickness (mils)	1.0	No crud observed on test rods	Djurle 1985
dco	Cladding outer diameter (in.)	0.4230	Design value	Djurle 1985
thkcld	Cladding wall thickness (in.)	0.0286	Design value	Djurle 1985
den	Fuel pellet density (% of theoretical)	95.0	Design value	Djurle 1985
dishsd	Dish shoulder width (in.)	0.0732	Dished pellets	Djurle 1985
pitch	Rod-to-rod pitch (in.)	0.56	Arbitrary value - boiling condition	
thkgap	Gap radial thickness (in.)	0.00285	Design value = 0.145 mm diametral	Djurle 1985
dspg	Spring diameter (in.)	0.35	Assumed	--
dspgw	Spring wire diameter (in)	0.03	Assumed	--
enrch	Fuel enrichment (% ²³⁵ U)	2.98	As-fabricated value	Djurle 1985
fgpav	Fill gas pressure (psi at room temp.)	326	As-fabricated value	Djurle 1985
hdish	Dish depth (in.)	0.0228	Design value	Djurle 1985
hplt	Pellet length (in.)	0.437	Design value	Djurle 1985
icm	Cladding type (4 = zircaloy-4)	4	Design value	Djurle 1985
icor	Cladding crud buildup (= 0 for no buildup)	0	No crud buildup noted	Djurle 1985
idxgas	Gas type (1 = helium)	1	Design value	Djurle 1985
iplant	Type of Plant	-2		
iq	Indicator for axial power shape	0		
jdlpr	Output control	1		
jn	Number of points per axial power shape	9,9		
jst	Assignment of shape numbers to each time step	19 × 1 (PK6-2,s) 15 × 2 (PK6-2,s) 19 × 1 (PK6-3) 13 × 2 (PK6-3)		
totl	Fuel stack length (ft)	1.022	Design value	--
rc	Fuel pellet inner diameter (in.)	0.0	Solid pellets	--
roughc	Cladding surface roughness (in.)	3.0E-5	Measured	Djurle 1985
roughf	Pellet surface roughness (in.)	8.0E-5	Ground to size	Djurle 1985
rsntr	Change in pellet density due to densification (kg/m ³)	100.	Stable pellets	Djurle 1985
vs	Number of plenum spring turns	8	Assumed	--
flux	fast neutrons (per m ² per second per W/g of fuel)	0.18E17	Standard PWR conditions	Djurle 1985
p2	Reactor system pressure (psi)	2,106	Obrigheim conditions	Djurle 1985
tw	Coolant inlet temperature (°F)	580	Obrigheim conditions (at the rodlet positions)	Djurle 1985
go	Coolant mass flow (lbm/ft ² /h)	0.0	Need to hold coolant temperature constant over short rodlet length	Djurle 1985

Table A15.4. (contd)

Input Variable	Meaning and Units	Input Value	Calculations and Notes	Reference
qf, x	Parameters of the axial flux shapes	See Djurle 1985	Djurle (1985) typical core flux shape. See attachments ???	Djurle 1985
qmpy, time	The LHGR history	See Djurle (1985)		Djurle 1985

Table A15.5. Post-Irradiation Results

Rod Number	Post-Bump FGR (%)	Ramp Terminal Level (kW/m)	Burnup (GWd/MTU)
PK6-2	3.5	40	36.8
PK6-3	6.7	43	36.5
PK6-S	6.1	41	35.9

sweeping was judged to not be a major gas release mechanism during the ramping because the rapid gas release in the high-temperature regions was faster than the rate of grain growth.

A.15.6 Code-Data Comparison

The predicted and measured FGR values, before and after the ramp test, are listed in Table A15.6 for the three rods. As can be seen, the code overpredicts the fission gas release slightly, for PK6 rods, but consistently predicts the majority of the fission gas to be released during the EOL power ramping, which is qualitatively and semi-quantitatively consistent with the measurements.

Table A15.6. Code-Data Comparison EOL (Post-Bump) Fission Gas Release

Rod Number	Measured FGR, %	Predicted FGR, %
PK6-2	3.5	6.03
PK6-3	6.7	6.96
PK6-S	6.1	6.63

A.16 Studsvik/Inter-Ramp Test Cases

A.16.1 Test Overview

The Studsvik Inter-Ramp Project objectives were to investigate the mechanical failure threshold for BWR 8 x 8 type fuel rods. Short rodlets with standard BWR 8 x 8 dimensions and components were fabricated by ABB/Atom specifically for the project and were base-irradiated to ~20 GWd/MTU at low LHGRs before EOL ramping at rapid rate to high LHGRs to probe for cladding failure. Hold times at the ramp terminal (LHGR) level were 24 hours for non-failed rods. For the non-failed rods, post-ramp fission gas release was determined by rod puncture.

Two of the non-failed ramp-tested Inter-Ramp rods, numbers 16 and 18, are described in this section. The following sections provide information on rod design and fabrication, irradiation conditions and test conduct, and post-test examination results.

A.16.2 Rod Design and Fabrication

Twenty short 21-inch rodlets were fabricated for the test, with nominal 8 x 8 BWR fuel rod characteristics, and there were some departures from these characteristics. Rods 16 and 18 were both "nominal rods" with 6-mil diametral gaps, 1 atm helium fill, and 95% TD solid, dished fuel pellets. The design parameters are listed in Table A16.1.

The rods were irradiated in approximate BWR coolant conditions in pressurized loops within the Studsvik reactor. Rods 16 and 18 operated for ~550 days at LHGRs ranging from 20 to 40 kW/m and achieved burnups of 21 and 18 GWd/MTU, respectively. Plots of the rod irradiation histories are shown in Figure A16.1, taken from H. Mogard et al. (1979) (*Final Report of the Inter-Ramp Project*, STIR-53, [restricted distribution]). The normal axial neutron flux profile for Studsvik is shown in Figure A16.2. Corresponding plots of the input power histories for FRAPCON-3 are shown in Figure A16.3 and Figure A16.4. Interim nondestructive measurements taken at the end of this base-irradiation period revealed nominal (as-expected) indications for cladding, ridging, oxidation, creepdown, and length increase.

Rods 16 and 18 were then preconditioned for 24 hours at 29 and 25 kW/m, respectively, and then ramped at a rate of ~70 W/m per second to terminal levels (maximum peak LHGR values) of 48 and 41 kW/m, respectively, where they were each held for 24 hours, during which time the coolant was monitored for added radioactivity (indicating rod failure). Neither of the rods failed. Therefore, puncture and fission gas release determinations were feasible, and were performed.

A.16.3 Post-Irradiation Examination Results

The deduced rod burnups, ramp terminal levels, and fission gas release values for the two rods are listed in Table A16.2.

Table A16-1. General Design Specifications for Inter-Ramp Rods

Fuel Rod	
Overall length (in.)	21.2
Diameter (in.)	0.4929
Fuel stack height (in.)	16.7
Nominal plenum chamber length (in.)	1.9
Number of pellets per rod	31
Fuel/clad diametral gaps (mils)	6.0 (nominal)
Fill gas composition	He
Fill gas pressure (atm)	1.0
Cladding	
Material	Zircaloy-2
Diameter outside (in.)	0.4929
Diameter inside (in.)	0.4250
Clad wall thickness (in.)	0.034
Ovality	Not specified
Eccentricity	Not specified
Grain size	Not specified
Roughness, microinch	31
Fuel	
Material	UO ₂
Enrichment (%)	3.5
Pellet sintered density (% TD)	95
Pellet diameter (in.)	0.4191
Pellet length (in.)	0.51
Pellet geometry	Dished, both ends
Dish radius (in.)	0.161
Dish depth (in.)	0.0102
Grain size (μm)	8 to 10
Roughness (microinch)	59

Figure A16.1. Rod Irradiation Histories for Inter-Ramp Rods 16 and 18

Figure A16.2. Normal Axial Neutron Flux Profile for Studsvik

Figure A16.3. Input Power Histories for FRAPCON3 for Rod 16

Figure A16.4. Input Power History for FRAPCON-3 for Rod 18

Table A16.2. Inter-Ramp Rod Burnup and Fission Gas Release Measurements

Rod Number	Rod-Average Burnup (GWd/MTU)	Ramp Terminal Level (kW/m)	Fission Gas Release (%)
16	21.0	47.9	16.0
18	17.7	41.0	3.5

A.16.4 FRAPCON-3 Input

The following table, Table A16.3, contains the background notes on the Studsvik Inter-Ramp rods input for FRAPCON-3.

Table A16.3. Background Notes on the Studsvik Inter-Ramp Rods Input for FRAPCON-3

Input Variable	Meaning and Units	Input Value	Calculations and Notes	Reference
cpl	Cold plenum length (in.)	1.93	Design value = 49 mm	(a)
crdt	Initial crud thickness (mils)	0.0	Little or no crud buildup	(a)
dco	Cladding outer diameter (in.)	0.4929	Design value	(a)
thkcld	Cladding wall thickness (in.)	0.034	Design value	(a)
den	Fuel pellet density (% of theoretical)	95	Design value	(a)
dishsd	Dish shoulder width (in.)	0.049	Design value	(a)
thkgap	Gap radial thickness (in.)	0.0025	Design value = 150 mm nominal; max/min values range from 123 to 159 mm (as calculated from as-fabricated tubing and pellet data)	(a)
dspg	Spring diameter (in.)	0.42	Design value	--
dspgw	Spring wire diameter (in.)	0.055	Design value	--
enrch	Fuel enrichment (% ²³⁵ U)	3.5	Design value	(a)
fgpav	Fill gas pressure (psi at room temp.)	14.7	Design value	(a)
hdish	Dish depth (in.)	0.0102	Design value = 0.26 mm	(a)
hplt	Pellet length (in.)	0.51	Design value = 13 mm	(a)
icm	Cladding type (2 = zircaloy-2)	2 (Rods 18,20,7) 5 (Rod 16)	Rod 16 = 5 (recrystallized) Rod 18 = 2 (stress-relieved)	(a)
icor	Cladding crud buildup (= 0 for no buildup)	0	No crud buildup probably in BOCA loops in Studsvik	(a)
idxgas	Gas type (1 = helium)	1	Helium fill (all rods)	(a)
iplant	Type of plant	-3	Heavy water plant	
iq	Indicator for axial power shape	1		
jdlpr	Output control	1		
totl	Fuel stack length (ft)	1.322	Design value = 403 mm enriched pellets	--
pitch	Rod-to-rod pitch (in.)	0.56	Arbitrary value - boiling condition	
rc	Fuel pellet inner diameter (in.)	0.0	Solid pellets	--
roughc	Cladding surface roughness (in.)	3.1E-5	Design value	(a)
roughf	Pellet surface roughness (in.)	5.96E-5	Design value; ground pellets	(a)
rsntr	Change in pellet density due to densification (kg/m ³)	120.	Resinter test data: 1,700°C, 24 hours	(a)
vs	Number of plenum spring turns	8	Assumed	--
flux	fast neutrons (per m ² per second per W/g of fuel)	0.18E17	Default value, but checked against fluence data	(a)
p2	Reactor system pressure (psi)	1,068	Normal BWR coolant conditions	(a)
tw	Coolant inlet temperature (°F)	500.0	Estimated; subcooled boiling conditions retained	(a)

Table A16.3. (contd)

Input Variable	Meaning and Units	Input Value	Calculations and Notes	Reference
go	Coolant mass flow (lbm/ft ² /h)	6E5	Equivalent mass flow unknown; given capsule mass flow must also cool capsule itself	(a)
qf, x	Parameters of the axial flux shapes	(a)	Profiles flatten with burnup; not accounted for in these low-burnup cases	(a)
qmpy, time	The LHGR history	(a)	LHGR levels in two-level history estimated from the appropriate curves in Figure A16.1	(a)
(a) H. Mogard et al., <i>Final Report of the Inter-Ramp Project</i> , STIR-53, 1979 (restricted distribution). (b) G. Lysell and S. Birath, <i>Hot Cell Post-Irradiation Examination of Inter-Ramp Fuel Rods</i> , STIR-51, 1979, (restricted distribution).				

A.17 Ramped Halden/DR-2 Test, Rods F7-3, F9-3, and F14-16

A.17.1 Overview of the Test Rods

The Riso Fission Gas Release Project was an international, group-sponsored program administrated by Riso Laboratories, Denmark, from 1980 to 1981. The objective was to investigate the impact of extended burnup and EOL power ramping on FGR in BWR-type fuel rods. This was done by performing power-bumping tests in the DR-2 reactor (Denmark) on 9 of the 14 rods irradiated in test fuel Assembly IFA-148. This assembly was operated in the Halden Reactor, Norway, from 1968 to 1979. The power ramps featured 24-hour hold periods at the peak power level, with the peak power level varied among the tests. These tests were supplemented by nondestructive examinations before and after the bumping irradiations, rod puncturing/gas analysis on all tested rods, and detailed destructive examinations on selected rods.

Three of the bumped rods were selected as FRAPCON-3 assessment cases: Rods F7-3, F9-3, and F14-6, which had rod-average burnups of 35, 33, and 27 GWd/MTU, respectively. The analyses of plenum gas ^{85}Kr activity before and after bumping were performed, and these were calibrated against the post-bump rod puncture results to yield an estimate of the net fission gas release caused by the power bumping. Thus, these cases provide the opportunity to assess the short-term FGR predictions of the FRAPCON-3 FGR model.

A.17.2 Test Rod Design and Fabrication

The IFA-148 assembly contained a total of 14 short BWR-type test rodlets, with 7 rods each in two clusters (upper and lower clusters). The fuel pellets were 5% enriched sintered uranium, with some variations in density and grain size. These two assessment cases, the nominal grain size and the fuel pellet densities, are equal (13-16 micron grain size and density = 93.4% TD, with a 0.6% increase upon resinter). The rods were initially filled with 1 atm helium fill gas. The design features are summarized in Table A17.1.

A.17.3 Irradiation Conditions and History

The operating conditions in the Halden boiling water reactor are summarized in Table A17.2. The peak-to-average power ratio for these segments was approximately 1.4, distributed as a chopped cosine. The LHGR levels were briefly at ~390 W/cm at BOL, quickly reduced to ~320 W/cm, and falling slowly to ~140 W/cm thereafter. The rod-average LHGR history for the assembly is shown in Figure A17.1, taken from Knudsen et al. (1983).

The assembly-average burnup at the end of 1,700 operating days was 31.7 GWd/MTU. The input power histories for the two rods are adjusted to the rod-to-assembly burnup ratios deduced for the individual rods from end-of-irradiation (EOI) gamma scans: 0.86 for F7-3 and F9-3 and 1.10 for F14-6. The base irradiation input LHGRs for these rods are shown in Figure A17.2a,b.

Table A17.1. General Design Specifications for Halden/Riso Rods F7-3, F9-3, and F14-6

Fuel Rod	
Overall length (in.)	35.0
Diameter (in.)	0.5472
Fuel stack height (in.)	31.53 (active)
Nominal plenum chamber length (in.)	1.79
Number of pellets per rod	62
Fuel/clad diametral gaps (mils)	7.1
Fill gas composition	He
Fill gas pressure (atm)	1.0
Cladding	
Material	Zircaloy-2
Diameter outside (in.)	0.5472
Diameter inside (in.)	0.5067 (F7-3) 0.5070 (F9-3) 0.5043 (F14-6)
Clad wall thickness (in.)	0.0216 (F7-3) 0.0213 (F9-3) 0.0205 (F14-6)
Ovality	Not specified
Eccentricity	Not specified
Grain size	Not specified
Fuel	
Material	UO ₂
Enrichment (% ²³⁵ U)	5.0
Pellet sintered density (% TD)	94.3
Pellet diameter (in.)	0.4996 (F7-3) 0.4976 (F14-6)
Pellet length (in.)	0.51
Pellet geometry	Dished, both ends
Dish radius (in.)	2.02 (curv.) 0.218 (dish)
Dish depth (in.)	0.0118
Grain size (m)	14

Table A17.2. Nominal Halden Reactor Operating Conditions

Parameter and units	Value
Power level (MW)	12
Reactor pressure (psig)	500
Heavy water saturation temperature (°F)	464
Plenum inlet temperature (°F)	459
Fast flux (> 1 MeV) (N/m ² /s per W/g fuel)	$\sim 5 \times 10^{15}$
Thermal flux (N/m ² /s per W/g fuel)	$\sim 2 \times 10^{16}$

Figure A17.1. Assembly Average LHGR History

Figure A17.2. The Base Irradiation Input LHGRs for Riso, (a) Rod F14-6, (b) Rod F7-3, and (c) Rod F9-3

The coolant conditions during bumping in the DR-2 facility are pressure = 1,000 psi, water near saturation temperature. A sketch of the typical power bumping test is shown in Figure A17.3. The time/LHGR values for this plot for the two rods are shown in Table A17.3.

Note that Rod F7-3 was bumped twice: first to a peak LHGR of 346 W/cm and finally to a peak LHGR of 441 W/cm, with 24-hour hold periods at the peak LHGR level in both cases.

Figure A17.3. A Sketch of the Typical Power Bumping Test

Table A17.3. DR-2 Conditions (see Figure A17.3). Note that LHGRs are axial peak values.

Parameter and Units	Rod F7-3	Rod F9-3	Rod F14-6
T1 (hours)	1, 0.5	1	0.5
T2 (hours)	6, 8	7	13.5
T3 (hours)	42, 44	47	41.5
T4 (hours)	66, 68	71	113.5
P1 (W/cm)	127, 39	91	29
P2 (W/cm)	178, 222	222	245
P3 (W/cm)	346, 426	451	441

A.17.4 Postirradiation Examination Results

Extensive postirradiation measurements were performed, including pre/post ramp ^{85}Kr plenum activity determination, pre/post ramp gamma scans and profilometry, post-ramp puncture and gas analysis, and sectioning and ceramography. Major observations are listed in Table A17.4.

Note that no additional FGR was recorded on the first bump test for Rod F7-3, whereas net FGR of ~5% was recorded for the second bump test. This is consistent with a general conclusion in the final report of a threshold of ~350 W/cm for the onset of bumping FGR.

A.17.5 FRAPCON-3 Input

A listing of the recommended FRAPCON-3 input parameters for Rods F7-3 and F14-6 appears in Table A17.5. The LHGR histories were derived from Misfeldt et al. (1981), and rod design and as-fabricated data were taken from Knudsen et al. (1983).

Table A17.4. Postirradiation Results

Item and Units	Value
Fission gas release (%) pre-bump	5.7 (F7-3) 3.8 (F9-3) 5.8 (F14-6)
Fission gas release (%) post-bump	5.7, 11.5 (F7-3) 8.5 (F9-3) 22.1 (F14-6)

Table A17.5. Recommended FRAPCON-3 Input for Halden/DR-2 Ramped Rods F7-3 and F14-6

Input Variable	Meaning and Units	Input Value	Calculations and Notes	Reference
cpl	Cold plenum length (in.)	1.79	Design value	Knudsen et al. 1983
crdt	Initial crud thickness (mils)	0.0	No crud observed on test rods	Knudsen et al. 1983
dco	Cladding outer diameter (in.)	0.5472	Design value	Knudsen et al. 1983
thkcl	Cladding wall thickness (in.)	.0213 (7-3) .0213 (9-3) .0205(14-6)	As-fabricated values for F7-3, F9-3, and F14-6	Knudsen et al. 1983
den	Fuel pellet density (% of theoretical)	93.4	Design value	Knudsen et al. 1983
pitch	Rod-to-rod pitch (in.)	0.6	Arbitrary value - boiling condition	

Table A17.5. (contd)

Input Variable	Meaning and Units	Input Value	Calculations and Notes	Reference
dishsd	Dish shoulder width (in.)	0.0	Dished pellets; shoulder = 0.75 mm	Knudsen et al. 1983
thkgap	Gap radial thickness (in.)	0.00355	180 micron diametral gap, all rods	Knudsen et al. 1983
dspg	Spring diameter (in.)	0.50	Assumed	--
dspgw	Spring wire diameter (in.)	0.030	Assumed	--
enrch	Fuel enrichment (% ²³⁵ U in U)	5.0	As-fabricated value	Knudsen et al. 1983
fgpav	Fill gas pressure (psi at room temp.)	14.7	As-fabricated value	Knudsen et al. 1983
hdish	Dish depth (in.)	0.0	Curvature = 51.5 mm; calculation depth = 0.3 mm	Knudsen et al. 1983
hplt	Pellet length (in.)	0.51	Design value assumed = 13 mm	Knudsen et al. 1983
icm	Cladding type (2 = zircaloy-2)	2	Design value assumed	Knudsen et al. 1983
icor	Cladding crud buildup (control = 0 for no buildup)	0	No crud buildup noted	Knudsen et al. 1983
idxgas	Gas type (1 = helium)	1	Design value	Knudsen et al. 1983
iplant	Type of plant	-4	Heavy water plant	
iq	Indicator for axial power shape	0		
jdlpr	Output control	1		
totl	Fuel stack length (ft)	2.627	Design value for active fuel length = 827 - 2 x 13 = 801 mm	--
rc	Fuel pellet inner diameter (in.)	0.0	Solid pellets	--
roughc	Cladding surface roughness (in.)	2.5E-5	Assumed	Knudsen et al. 1983
roughf	Pellet surface roughness (in.)	2.5E-5	Ground to size	Knudsen et al. 1983
rsntr	Change in pellet density due to densification (kg/m ³)	66.	2 hour sintering; resinter test result was 0.6% TD	Knudsen et al. 1983
vs	Number of plenum spring turns	8.0	Assumed	--
flux	Fast neutrons per m ² per second per W/g of fuel	5E15	Halden thermalized conditions	Knudsen et al. 1983

Table A17.5. (contd)

Input Variable	Meaning and Units	Input Value	Calculations and Notes	Reference
p2	Reactor system pressure (psi)	7-3, 9-3 21 × 500, 6 × 1030 14-6 21 × 500, 7 × 1030	Halden conditions; DR-2 test conditions	Knudsen et al. 1983
tw	Coolant inlet temperature (°F)	7-3, 9-3 21 × 464, 6 × 500 14-6 21 × 464 7 × 500	Halden conditions; DR-2 assumed test condition	Misfeldt et al. 1981
go	Coolant mass flow (lbm/ft ² /h)	0.0	Need to hold coolant temperature constant over short rodlet length	Misfeldt et al. 1981
jn	Number of points per axial power shape	7,7,7,7		
jst	Assignment of shape numbers to each time step	1,1,2, 5 × 3, 3 × 4, 5 × 3, 4,1, 1,2,3, (7-3, 9-3, 14-6) 6 × 5 (7-3, 9-3) 7 × 5 (14-6)		
qf, x	Parameters of the axial flux shapes	Misfeldt et al. 1981	Barner et al. (1990) typical flux shape	Misfeldt et al. 1981
qmpy, time	The LHGR history	Knudsen et al. 1983; Misfeldt et al. 1981	Assumed peak values correspond with LHGRs on temperature data plots	Knudsen et al. 1983; Misfeldt et al. 1981

A.17.6 Code-Data Comparison

The measured and predicted FGR for the three selected rods are compared in Table A17.6. The code correctly predicts the majority of the gas release to occur during the power bump, and the net release is very closely predicted in each case. The underprediction of total post-bump release seems to be related in these cases to underprediction of the pre-bump release.

Table A17.6. Measured and Predicted FGR

Rod Identity	Parameter and Units	Measured Value	Predicted Value
F7-3	Pre-bump FGR (%)	5.7	0.4
F7-3	Post-bump FGR (%) 1st/2nd bump	5.7/11.5	4.6/9.91
F14-6	Pre-bump FGR (%)	5.8	0.3
F14-6	Post-bump FGR (%)	22.1	12.7
F9-3	Pre-bump FGR (%)	7.3	0.4
F9-3	Post-bump FGR (%)	17.5	14.7

A.18 References

Balfour, M. G. 1982a. *BR-3 High Burnup Fuel Rod Hot Cell Program Volume 1: Final Report*, WCAP 10238, DOE/ET 34073-1, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania.

Balfour, M. G., W. C. Chubb, and R. F. Boyle. 1982b. *BR-3 High Burnup Fuel Rod Hot Cell Program Volume 2: Final Report*, WCAP 10238, DOE/ET 34073-1, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania.

Barner, J. O., M. E. Cunningham, M. D. Freshley, and D. D. Lanning. 1990. *High Burnup Effects Program-Final Report*, DOE/NE/34046-1, HBEP-61, Pacific Northwest National Laboratory, Richland, Washington.

Baumgartner, J. A. 1983a. *BWR Fuel Bundle Extended Burnup Program Technical Progress Report, January 1981 - December 1981*, DOE/ET/30431-12, GEAP-30083, Pacific Northwest National Laboratory, Richland, Washington.

Baumgartner, J. A. 1983b. *BWR Fuel Bundle Extended Burnup Program Technical Progress Report, January 1982 - December 1982*, DOE/ET/30431-12, GEAP-30268, Pacific Northwest National Laboratory, Richland, Washington.

Baumgartner, J. A. 1984. *BWR Fuel Bundle Extended Burnup Program Technical Progress Report, January 1983 - December 1983*, DOE/ET/30431-17, GEAP-30643, Pacific Northwest National Laboratory, Richland, Washington.

Bradley, E. R., M. E. Cunningham, J. L. Daniel, N. C. Davis, D. D. Lanning, and R. E. Williford. 1979. *Precharacterization Report for Instrumented Fuel Assembly IFA-513*, NUREG/CR-1077, PNL-3356, Pacific Northwest National Laboratory, Richland, Washington.

Bradley, E. R., M. E. Cunningham, D. D. Lanning, and R. E. Williford. 1981. *Data Report for the Instrumented Fuel Assembly IFA-513*, NUREG/CR-1838, PNL-3637, Pacific Northwest National Laboratory, Richland, Washington.

Cunningham, M. E., and K. Svanholm. 1986. *Fission Gas Release from PWR-Type Rods during Rapid and Short Power Increases*, HWR-159, Halden Reactor Project, Halden, Norway.

Cunningham, M. E., D. D. Lanning, and J. O. Barner. 1990. *Qualification of Fission Gas Release Data from Task 3 Rods*, HBEP-60, Pacific Northwest National Laboratory, Richland, Washington.

Djurle, S. 1985. *Final Report of the Super-Ramp Project*, DOE/ET/34032-1, U.S. Department of Energy, Washington, D.C.

Grimoldy, R. D., and B. Crossley. 1981. *Pre-irradiation Characterization of BNFL Rods - Task 2A*, HBEP-04, Pacific Northwest National Laboratory, Richland, Washington.

Hallstadius, L. 1989. *Fabrication, Pre-irradiation Characterization, and Irradiation History for TVO-1 Rods, Task 3*, HBEP-51.

Hann, C. R., J. L. Bates, D. W. Brite, J. L. Daniel, N. C. Davis, P. E. Hart, R. K. Marshall, G. B. Mellinger, and R. E. Williford. 1988. *Test Design, Precharacterization, and Fuel Assembly Fabrication for Instrumented Fuel Assemblies IFA-432 and IFA-432*, NUREG/CR-0332, Pacific Northwest National Laboratory, Richland, Washington.

Janvier, J-C., B. de Bernaday de Sigoyer, and R. Delmas. 1967. *Irradiation of Uranium Oxide in Strong Cladding Effect of Initial Diametral Gap on Overall Behavior*, Program CC-7, 1st and 2nd Sections, CEA-R-3358, Commissariat a l'Energie Atomique, Paris, France.

Knudsen, P., C. Bagger, H. Carlsen, I. Misfeldt, and M. Mogensen. 1983. *RISØ Fission Gas Release Project Final Report*, DOE/ET/34033-1, U.S. Department of Energy, Washington, D.C.

Kolstad, E., H. Devold, and V. Tossi. 1991. "High Burn-up Fuel Behavior Studies by In-Pile Measurements," *ANS/ENS International Topical Meeting on LWR Fuel Performance*, Avignon, France, pp. 838-849.

Lanning, D. D., and E. R. Bradley. 1984. *Irradiation History and Interim Postirradiation Data for IFA-432*, NUREG/CR-3071, PNL-4543, Pacific Northwest National Laboratory, Richland, Washington.

Lanning, D. D., and M. E. Cunningham. 1979. *Startup Data Report for NRC/PNL Assembly IFA-513*, NUREG/CR-0862, PNL-2948, Pacific Northwest National Laboratory, Richland, Washington.

Lanning, D. D. 1986. *Irradiation History and Final Postirradiation Data for IFA-432*, NUREG/CR-4717, PNL-5971, Pacific Northwest National Laboratory, Richland, Washington.

Lanning, D. D., M. E. Cunningham, E. R. Bradley, and J. O. Barner. 1987. *Qualification of Fission Gas Release Data from Task 2 Rods*, HBEP-25, Final Report, Pacific Northwest National Laboratory, Richland, Washington.

Misfeldt I., J. Aukdal, and P. Knudsen. 1981. *Halden Irradiation of IFA-148*, RISØ-FGP-12, RISØ Laboratory, Roskilde, Denmark.

Newman, L. W. 1986. *The Hot Cell Examination of Oconee-1 Fuel Rods after Five Cycles of Irradiation*, DOE/ET/34212-50, BAW-1874.

Notley, M. J. F., and J. R. MacEwan. 1965. *The Effect of UO₂ Density on Fission Product Gas Release and Sheath Expansion*, AECL-2230, Atomic Energy of Canada, Limited, Chalk River, Canada.

Notley, M. J. F., R. DesHais, and J. R. MacEwan. 1967. *Measurements of the Fission Product Gas Pressures Developed in UO₂ Fuel Elements during Operation*, AECL-2662, Atomic Energy of Canada, Limited, Chalk River, Canada.

Smith Jr., G. P., R. C. Pirek, H. R. Freeburn, and D. Schrire. 1994. *The Evaluation and Demonstration of Methods for Improved Nuclear Fuel Utilization*, DOE/ET/34013-15, Combustion Engineering, Windsor, Connecticut.

Waterman, M. E. 1978. *Corrected and Updated Data for IFA-429 from Beginning of Life Through June 1978*, NUREG/CR-0478, Tree-1269, Idaho National Engineering Laboratory, Idaho Falls, Idaho.

West, J. S., S. E. Kasib, and S. Y. Ogawa. 1983. *BWR Fuel Bundle Extended Burnup Program, EOC-9 Final Fuel Bundle Examination at Monticello Nuclear Generating Plant*, DOE/ET/34031-16, General Electric Company, San Jose, California.

Wiesenack, W. 1995. "High Burn-up Phenomena - Results from Experiments in the Halden Reactor," presented at the *CSNI Specialists' Meeting on Transient Behavior of High Burn-up Fuel*, September 12-14, 1995 at Cadarache, France, OCDE/GD(96) 197, NEA/CSNI/R(95)22, pp. 405-416.

Winne, C. 1981. *Irradiation History of BNFL Rods - Task 2A*, HBEP-05, Pacific Northwest National Laboratory, Richland, Washington.

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER
(Assigned by NRC, Add Vol., Supp., Rev.,
and Addendum Numbers, if any.)

NUREG/CR-6534 Vol 3

PNNL - 11513

2. TITLE AND SUBTITLE

FRAPCON-3: Integral Assessment

3. DATE REPORT PUBLISHED

MONTH

YEAR

December

1997

4. FIN OR GRANT NUMBER

W6200

5. AUTHOR(S)

D. D. LANNING (PNNL), C. E. BEYER (PNNL), G. A. BERNA (GABC)

6. TYPE OF REPORT

computer code manual

7. PERIOD COVERED (Inclusive Dates)

1997

8. PERFORMING ORGANIZATION - NAME AND ADDRESS (If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

Pacific Northwest National Laboratory (PNNL)
Richland, Washington 99352-0999

Subcontractor:
Gary A. Berna Consulting (GABC)
Idaho Falls, Idaho 83404-6131

9. SPONSORING ORGANIZATION - NAME AND ADDRESS (If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)

Division of Systems Technology
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

H. Scott, NRC Project Manager

11. ABSTRACT (200 words or less)

Fuel rod material properties and performance models have been updated for the FRAPCON steady-state fuel rod performance code to account for changes in behavior due to extended fuel burnup. The updated code is named FRAPCON-3 and is intended to replace the earlier codes FRAPCON-2 and GAPCON-THERMAL-2. The property and model updates are described in Volume 1 of this report. Volume 2 of this report constitutes the code description document and includes the input instructions.

This document (Volume 3) provides the results of the assessment of the integral code predictions to measured data for various performance parameters. In the case of fuel temperature and fission gas release (FGR) predictions, comparison is made to both benchmark data sets and independent benchmark data sets. The benchmark data sets are described in Section 2.0. Appendix A describes each individual set of benchmark data and gives the code input for each data comparison. The data are drawn from a wide range of burnup levels and operating conditions for both PWR and BWR type rods.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

Computer Code	Code Assessment	Benchmark Data
Fission Gas Release	Fuel Rod Performance	Cladding Corrosion
Cladding Creepdown	High Burnup	Pellet Centerline Temperature
Fuel Swelling		

13. AVAILABILITY STATEMENT

unlimited

14. SECURITY CLASSIFICATION

(This Page)

unclassified

(This Report)

unclassified

15. NUMBER OF PAGES

16. PRICE