ATTACHMENT B
MODIFICATIONS TO FRAPCON-3.2 FOR MOX FUEL

## 1.0 FUEL THERMAL CONDUCTIVITY

The major modification to FRAPCON-3.2 for application to MOX fuel was the addition of a fuel thermal conductivity model specific to MOX fuel. This was selected as a combination of the Duriez stoichiometry-dependent correlation, derived from diffusivity measurements on unirradiated fuel pellets (Reference 1), plus the burnup degradation contained in a modified version of the NFI fuel thermal conductivity model (Reference 2). The combined model was described in PNNL's paper for the Halden EHPG meeting in Storefjell, Norway (Reference 3). In that paper, code-data comparisons were made with the new model added, for three instrumented MOX fuel tests in Halden Reactor: IFA-629.1, IFA-610.2,4 and IFA-648.1. Since then, comparisons have also been made to IFA-629.3 (the ramp-test extension of IFA-648.1), and to IFA-606. All these tests and their reference documents are briefly summarized in Table 1.

Predicted-vs.-measured results for all the comparisons are shown in Figure 1. The normalized temperature differences (predicted-minus measured divided by measured minus coolant temperature) are shown as a function of LHGR in Figures 2 and 3. As can be seen, the predictions are very close to the data for IFA's 629.1, 606, and 610.2,4 (Figure 2). They deviate about 5% above the data for IFA 629.3 and 5% below the data for IFA-648.1 (Figure 3). Since the same rods and thermocouples are used in both tests, it may be that the LHGR associated with measured temperature may deviate from true values in one or both tests. Halden Project has been requested to investigate this possibility.

Overall, the addition of comparisons to the extensive raw data files from IFA's 648/629.3, and the digitized data from IFA 606, has extended the data base but provided no net incentive to change in the model from that presented at the Storefjell meeting.

## 2.0 FISSION GAS RELEASE

Design, operation, and FGR data provided by Halden has provided opportunity to compare code predictions to the steady-state FGR from three full-length MOX PWR rods (the 'mother rods" N06, N12, and P16 for instrumented sections tested in IFA's 610.2,4 and IFA-648.1/629.3). Comparison has also been made to end-of-ramp FGR for the power-ramp tested instrumented fuel rod sections in IFA's 629.1, 629.3, and 606. The results, with no modification to the FGR model, are shown in Figure 4. It is clear that FRAPCON-3.2 is generally under predicting the FGR for these 6 cases. Multiplying the diffusion constant by 1.75 raises the FGR to a closer overall comparison with this available data (see Figure 5), and has been incorporated for MOX into FRAPCON-3.2.

## 3.0 HELIUM PRODUCTION AND RELEASE

Puncture data and gas analysis was provided for two of the three mother rods, N12 and P16 (Reference 4). This permits evaluation of the change to rod helium inventory from beginning of life (BOL) to of life (EOL) The results indicate negligible change ~3% relative) in the helium inventory from beginning to end of life. These results are summarized in Table 2. This is consistent with current FRAPCON-3.2 predictions, and no change to FRAPCON-3.2 regarding helium release is recommended at this time. It should be noted that the initial fill gas pressure for these rods was relatively high at 363 psia, vs. a somewhat smaller value probable for MOX rods used in the U.S. for plutonium disposition. There is some evidence and theory that suggests higher fill gas pressure will reduce helium release.

## 4.0 ADJUSTMENTS FOR PLUTONIUM ISOTOPES

Input parameters have been added to signal when MOX fuel is being analyzed, and to initialize the concentrations of plutonium isotopes in the TUBRNP subcode, which calculates radial power and burnup profiles within the fuel pellets. Given this initialization, TUBRNP appears to calculate the radial profiles for LWR MOX fuel with acceptable accuracy. This was assessed by comparing code calculations to MCNP code calculations for radial power profiles (where the MCNP results were provided by ORNL). An example of this comparison is shown in Figure 7.

## 5.0 Xe/Kr RATIO

Fission gas is partitioned into krypton and xenon fractions within the code. Currently, the code uses Xe/Kr ratio of 5.67 in making this partition, which is appropriate for urania fuel. For MOX fuel, the majority of fissions occur in plutonium, and the xenon stable isotope yields are higher. Gas analysis data from MOX rod punctures at nominal to high burnup indicates Xe/Kr ratios of approximately 19 (Reference 4), however, Xe/Kr fission yields for plutonium indicate a value of 16 (see Reference 5 for example). The code has been altered to use the ratio of 16 when MOX fuel is being analyzed. The effects of this change are a small decrease in gas conductivity and a very small decrease in gap conductance for cases where fission gas concentration in the plenum gas becomes significant. However, the output gas species ratios now reflect a more realistic Xe/Kr ratio for MOX.

## REFERENCES


1. Duriez, C., J.-P. Allesandri, T. Gervais, and Y. Philipponneau. 2000. "Thermal Conductivity of Hypostoichiometric Low Pu Content $(U,Pu)O_{2-x}$ Mixed Oxide." *Journal of Nuclear Materials* 277:143-158.
2. Ohira, K., and N. Itagaki. 1997. "Thermal Conductivity Measurements of High Burnup $UO_2$ Pellet and a Benchmark Calculation of Fuel Center Temperature." *Proceedings of the ANS International Topical Meeting on LWR Fuel Performance***,** pp. 541-549. March 2-6, 1997, Portland, Oregon.
3. Lanning, D.D., and C.E. Beyer. 2002. "Revised $UO_2$ Thermal Conductivity for NRC Fuel Performance Codes." *Transactions of the American Nuclear Society, 2002 Annual Meeting*, volume 86, pp 285-287. June 9-13, 2002, Hollywood, Florida.
4. Halden Project 2002. Private Communication
5. White, R.J., et al., 2001. "Measurement and Analysis of Fission Gas Release from BNFL's SBR MOX Fuel" *Journal of Nuclear Materials Vol. 288*, pages 43 to 56.

**Halden Reactor Project Reports**:

HPR 349/30 .. "The FIGARO Programme: The Behaviour of Irradiated MOX Fuel Tested in the IFA-606 Experiment, Description of Results and Comparison with COMETHE Calculation" L.Mertens, M. Lippens and J Alvis, March 1998.

HWR-586 "The Re-Irradiation of MIMAS MOX Fuel in IFA-629.1" Rodney J. White, March 1999.

HWR-603 "The Lift-Off Experiment with MOX Fuel Rod in IFA-610.2; Initial Results" Stephane Beguin, April 1999.

HWR-650 "The Lift-Off Experiments IFA 610.3 ($UO_2$) and IFA610.4 (MOX): Evaluation of In-Pile Measurement Data". Hajimi Fuji, Julien Claudel, February 2001.

HWR-651 "Results from the Burnup Accumulation Test with High Exposure (63 GWd/kg HM) MOX Fuel (IFA-648)". Julien Claudel, Francois Huet, February. 2001

HWR-664 "Summary of Pre-Irradiation Data on Fuel Segments Supplied by EdF/FRAMATOME and Tested in FA-610, 629 and 648". Masahiro Nishi, and Byung-Ho Lee, February, 2001.

HWR-714 "Ramp Tests with Two high Burnup MOX Fuel Rods in IFA-629.3". Benoit PetitPrez, June 2002

Table 1. Instrumented MOX Tests in Halden. All with re-fabricated PWR rod sections containing MIMAS MOX Fuel
(Calculated FGR values are code predictions with diffusion constant multiplier = 1.0)

| Reactor/Full Length Rod (Rod Diameter in mm) | Base Irradiation Cycles | Burnup, GWd/MTM (and FGR%) at end of base irradiation | Sponsor | Halden Test (IFA No.) and report (HWR No.)** | Test Type and Max. Rod-Average LHGR , kW/m | End of Test FGR % And Measurement Type |
|---|---|---|---|---|---|---|
| St. LaurentB1/J09 (9.35) | 2 | 27 (low) | Halden Group | 629.1 HWR-586 | Ramp (35) | 25% (Puncture) 26% PT[(b)] 17% calculated |
| Gravelines-4/N06 (9.35) | 4 | 48 (4.12) (2.6% calculated) | Halden Group | 610.2,4* HWR-603,650 | Lift-off (10) | -- |
| Gravelines-4/N12 (9.35) | 4 | 50 (4.86) (3.0% calculated) | Halden Group | 648* (629.3) HWR-651 (HWR-714) | SS (10) (Ramp, 25) | -- |
| Gravelines-4/P16 (9.35) | 4 | 47 (2.58) (1.7% calculated) | Halden Group | 648* (629.3) HWR-651 (HWR-714) | SS (10) (Ramp, 25) | 7% (PT) 2.3% calculated |
| Beznau-1 (10.7) | 5 | 50 (low) | Belgo-Nucleaire[(a)] (FIGRARO) | 606 (HPR-349/30) | Ramp (32) | 13% (PT and puncture) 19% calculated. |

*Note that IFA's 610.2,4 and IFA-648.1 operated in a PWR-condition loop within the HBWR, thus at a coolant temperature and pressure of 310 C and 2250 psia, instead of normal HBWR conditions (240 C, 500 psia)

**HWR-664 contains design, precharacterization, and base irradiation data for the St.Laurent and Gravelines EdF rods.

[(a)] Note this is proprietary data
[(b)] Pressure transducer

Table 2. Helium Results from Halden Test High-Burnup PWR MOX "Mother Rods"

| Reactor/Full Length Rod (Rod Diameter in mm) | Base Irradiation Cycles | Burnup, GWd/MTM | BOL/EOL Helium inventory, STPcc |
|---|---|---|---|
| Gravelines-4/N12 (9.35) | 4 | 50 | 449/454 |
| Gravelines-4/P16 (9.35) | 4 | 47 | 417/422 |

Figure 1. Predicted vs. Measured Fuel Center Temperatures for Halden MOX Tests

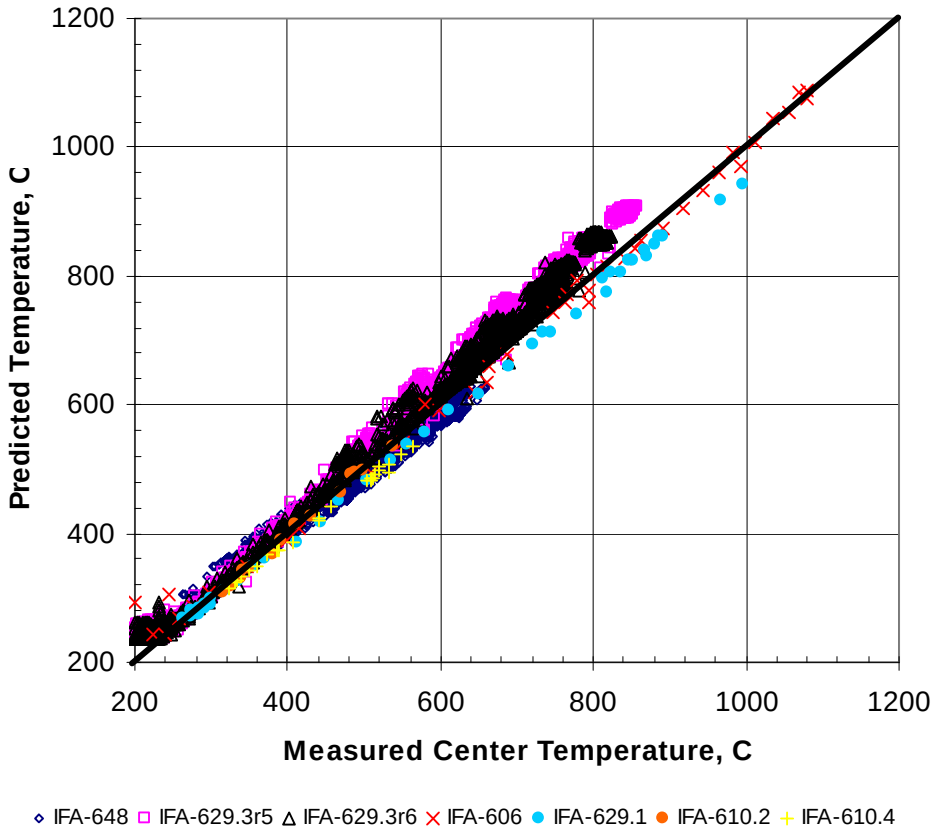


Figure 2. Normalized Predicted-minus-Measured Temperature Difference (IFA's 629.1, 610.2,4 and 606)

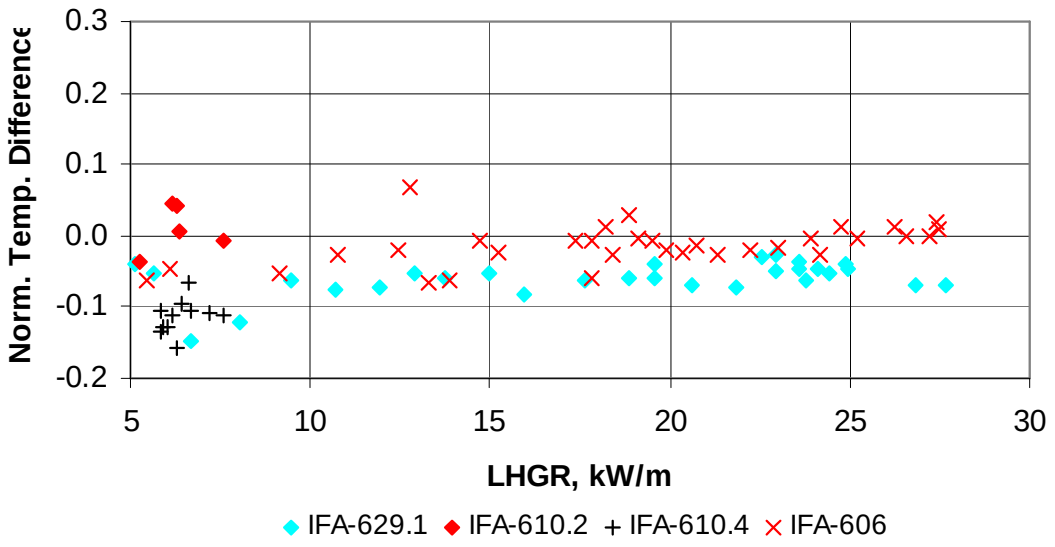



Figure 3. Normalized Predicted-minus-Measured Temperature Difference (IFA's 648.1 and 629.3 [same rods, 629.3 subsequent to 648.1])

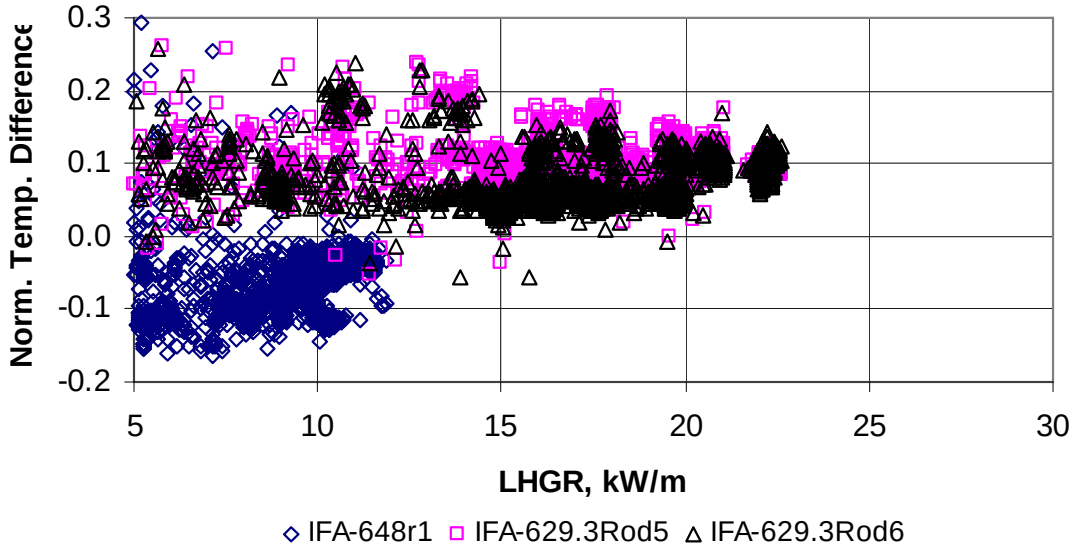


Figure 4. Predicted vs. Measured FGR, unmodified FGR model

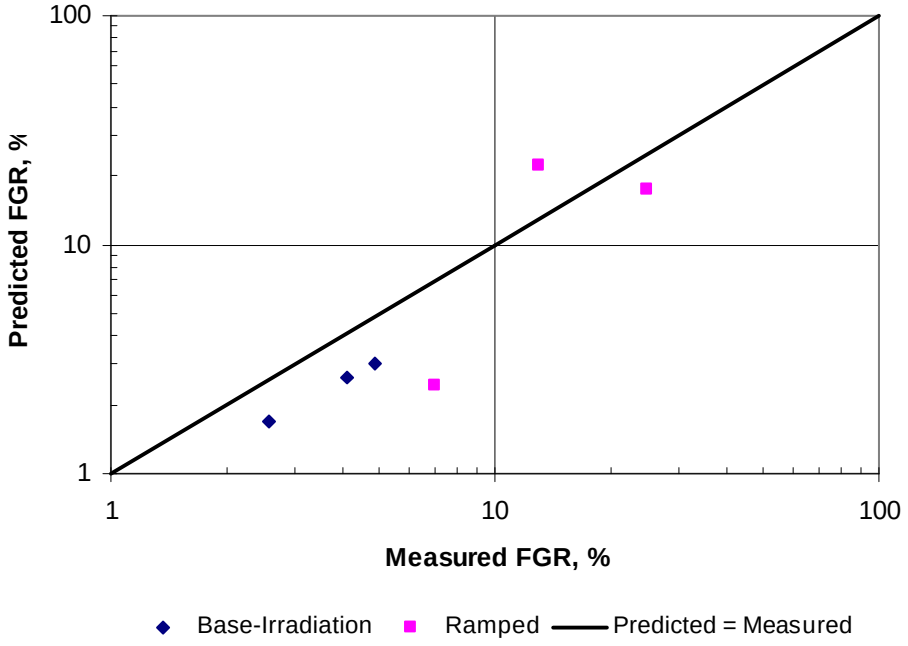



Figure 5 Predicted vs. Measured FGR, Diffusion constant x 1.75

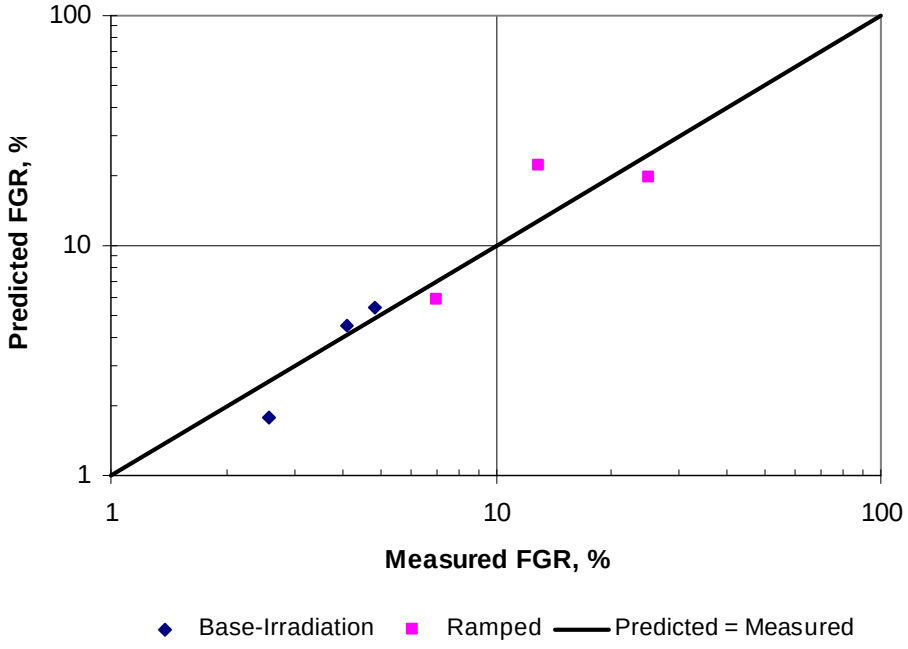
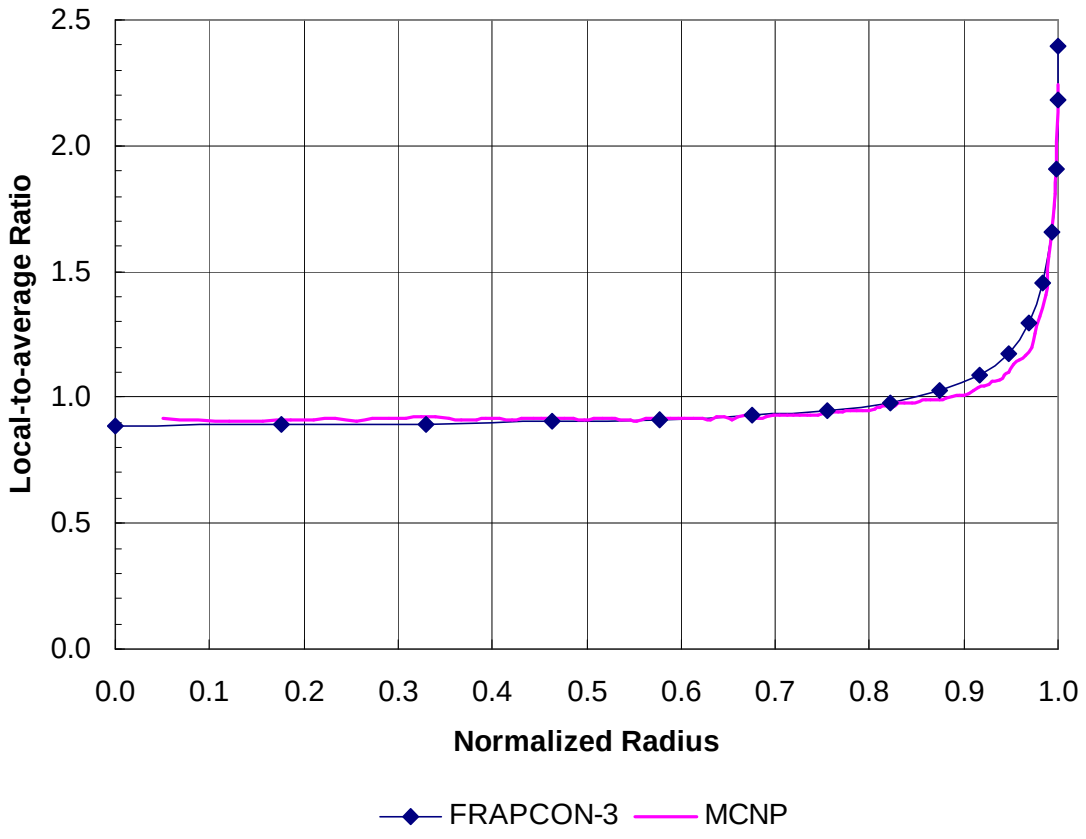


Figure 6

MCNP and FRAPCON-3 Calculations of normalized radial power profile in PWR MOX Fuel at 50 GWd/MTHM (Initial content = 5 wt.% WG Plutonia in MOX)