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# Consolidator's Report for the SPERT-III Benchmark

Nuclear Engineering Division

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## Introduction

The SPERT-III reactor facility historical documentation was assembled into a Benchmark Specification. Recommendations were made in that specification as to which tests would be of most interest for analysis. This Benchmark Specification was used by analysts from Romania and the USA to construct computational models of the experiments. The work required 3-dimensional neutronics, to generate point reactor kinetics parameters, power shapes, and reactivity feedback coefficients. It then required coupled space-time kinetics in the point-kinetics mode, using reactivity feedback from the Doppler Effect caused by heat-up of the fuel meat, reactivity feedback from coolant heat-up, and feedback from void production.

Selected SPERT-III Experiments were analyzed by two organizations:

1. M. Mladin, D. Mladin, S. Dulugeac, G. Budriman – INR, Romania, Final Report, Contract no. 15350, February 8, 2013
2. Arne P. Olson, NEUTRONICS CALCULATIONS for SPERT-III, E-CORE, Argonne, IL 60439 USA**,** April 15, 2013

The purpose of this report is to compare the work of these two organizations and to summarize overall findings and recommendations for future work.

## Remarks Concerning the Benchmark Specification

Detailed Specifications created by USA are given in the companion document: IAEA CRP: Innovative Methods for Research Reactors, SPERT III E-CORE Reactor Specification, IAEA, 2012.

Early on there were a few requests for clarification of items in the benchmark specification, but none in 2012. Romania was able to proceed with their analysis without any need for more information. Their success in producing good agreement with measurement indicates that the specification was adequate. The USA found that the lack of engineering drawings for the control rods created some uncertainty as to where the top of the fuel pellets were located (and the top of the active core) relative to the boron-steel follower box. Consequently the USA assumed that the boron box bottom was at the plane of the top of the fuel pellets. It is hoped that an NEA benchmark case for SPERT III could help clarify this issue.

## Remarks Concerning Analysis of the Experimental Program

The E-core experimental program was divided into low-initial-power and high-initial-power test phases. Low-initial-power (≈50 W) excursions were performed for cold- and hot-startup conditions. High-initial-power excursions were performed for hot-standby and operating-power conditions. Reactor physics/thermal hydraulic analyses were performed for three different reactor conditions of temperature, pressure, coolant flow rate, and initial power. Table 1 lists the estimated standard deviation for key measured parameters.

**Table 1. Estimated standard deviation for measured parameters**

- Reactor period 2 %
- Reduced prompt neutron generation time 2.5 %
- Delayed neutron parameters 7-15 %
- Derived reactivity insertion 4 %
- Reactivity compensation at peak power 11 %

It is suspected that the reactivity insertion was actually known more accurately that the quoted standard deviation of 4% because the effect of this apparently small range is so huge. It is also noted that the reactor period is known with half the uncertainty of the reactivity insertion. Analysis codes such as PARET cannot search for a desired period. It is clearly significantly more work, but it is recommended that future analysts first calculate reactor period vs. reactivity for a class of experiments. Second, they could interpolate on period to find the reactivity that should match the particular experimental reactivity insertion. Finally, they could run the case and refine it to match the expected period. This is a rather complex and multi-step process.

The analysis by Romania for case T-86 was carried out using a reactivity input of 1.17 $, as the base. They also performed analyses using uncertainty limits on the low side of ±0.03, and showed that their results compared very well with experiment when they included direct heating to the moderator. These results showed the value in not using the upper limits on the reactivity uncertainty when performing bounding calculations for these tests.

## Comments on the MCNP Neutronics Models

The MCNP5 code used for the analysis by USA was version 1.60, with standard libraries (ENDF-B/VII). It is documented in LA-UR003-1987, MCNP— A General Monte Carlo N-Particle Transport Code, Version 5, Volume I: Overview and Theory, X-5 Monte Carlo Team, April 24, 2003 (Revised 2/1/2008). This version has the ability to calculate point kinetics parameters directly. Comparisons have been made recently at ANL between the new option, and traditional methods of obtaining $\beta_{eff}$ etc. This comparison validated the new option.

Romania also used MCNP, but did not identify the version.

Direct heating to the moderator was not calculated by the USA or by Romania. Instead, the USA used a value of 2.6%, which was assumed as typical of a PWR $UO_2$ fuel rod [Shigeaki Aoki, Takayuki Suemura, Junto Ogawa and Toshikazu Takeda, Analysis of the SPERT-III E-Core Using ANCK Code with the Chord Weighting Method, Journal of NUCLEAR SCIENCE and TECHNOLOGY, Vol. 46, No. 3, p. 239–251 (2009)]. Subsequent PARET calculations for experiment T-86 confirmed that direct heating was significant. As a result, experiments T-79 through T-86 were recomputed using direct heating. The other cases assumed no direct heating, because the coolant temperature rise in those tests was so small as to make negligible the effect of direct heating.

## Comments on the PARET Model Created by USA

Analysts at ANL are divided as to what is the best procedure to follow when creating a PARET model. Some believe that a two-channel model is best when one only has reactor-averaged feedback coefficients. In that case, one would use one channel to represent the hottest fuel rod or plate, and the 2$^{nd}$ channel to represent the remainder of the core. USA created a 5-channel core representation in order to attempt to follow the consequences of heat up of smaller groups of channels, rather than one representing the core average. The 5-channel model would theoretically be even better if channel-dependant feedback coefficients were available (this

requires much more analysis). Future studies of the effect of multi-channel analysis, to account for spatial effects on reactivity feedback, are recommended.

## Discussion of Results Concerning Reactivity Feedback, and Clad Surface Temperature Rise

There is some uncertainty as to the initial power of each test. One can see that typical calculations of power show about the same slope, indicating that the period is correct, but that there may be a time offset caused by this uncertainty in initial power. If the calculated initial power is assumed to be 5 W, but it actually was 50 W, then there will be a time lapse between calculation and experiment.

It is noted that the PARET results are always conservative: they predict too high a peak power, and too high an energy release. This leads to predicting too high a temperature rise in the clad. This comparison is somewhat imprecise because the axial locations of the measurement may not be quite the same as computed (for example, PARET reports the absolute maximum found), and because the axial power shape in the calculations is sensitive to the position of the control rods. As modeled, the control rods are quite close to the correct initial condition locations for each experiment, but are not precise. It is concluded that the Cold-Startup tests, which had no flow at the start of each transient, significantly over-predicts temperature rise in the clad. This may be due, in part, to insufficient induced natural convection flow, which in turn is a consequence of inadequate modeling of the flow circuit by a 1-dimensional model without recirculation (PARET). The other test conditions with flow also over-predict, but appear to be quite reasonable.

One can also observe that the trends in power vs. reactivity are excellent. This is fine for reactor safety and licensing because then the reactor performance can conservatively be predicted for similar designs, for similar conditions covered by the test envelope.

Clearly, the Doppler Effect from heat-up of the $UO_2$ dominates the shape of each test's power vs. time curve. There is so little temperature rise in the low-power tests that there is no void production, and the temperature coefficient for the water is quite small.

Both the USA and Romania reported that their calculated Doppler feedback coefficients were too small compared to the evolution of the experiments. This is an analysis area needing further study. One consideration to investigate is the reactivity feedback effect from swelling of the fuel rod cladding as they heat up. This reduces the water volume in the coolant because the lattice pitch does not change during a short transient. The pitch is constrained by spacers that do not heat up very much compared to the clad.

## Conclusions

The Final Report by Romania is complete regarding tables and graphics for a large number of tests. It could be improved by expanding the discussion of their methodology. The Final Report provides results for clad surface temperature rise (the peak temperature rise measured at any time during the event). Also, they have provided the computed amount of reactivity feedback at the time of peak power.

The Report by USA has covered all the planned tests and more. Graphics comparing calculation with experiment are provided. Peak clad surface temperature rise results are provided. The USA also provided tabular results for the amount of reactivity feedback at the time of peak power. It is concluded that the Final Reports by Romania and by USA are complete.

It is concluded that reactivity compensation at peak power compares quite well with values deduced by the experimentalists (it is inferred, not explicitly measured). USA updated their draft Report by improving their MCNP model for the control rods, and by calculating axial power shapes for the transient rod out, with about 1$ of excess reactivity, for temperatures of 294, 400, and 533 K. Romania made the approximation that the power shape change between 294 and 400K was not expected to be very large, so they used a single shape at 294 K.

All results obtained by the USA, and by Romania, used the nominal 15$/s reactivity insertion rate recommended by the original analysts. As a check on this assumption, the USA performed a sensitivity study of the transient rod worth vs. time, during its ejection from the core. This was accomplished by computing the reactivity for the control rod located at many positions in the core. Knowing the design acceleration of the transient rod, it was possible to convert change in position to change in time. This was based on the assumption that the transient rod was ejected with the design acceleration of 787.4 cm/s/s. It was shown, for the cases investigated, that the effects of deviation from a linear ramp rate were quite small. It is recommended that the problem of determining reactivity insertion vs. time receive further study. It is also recommended that the reactivity feedback effect of clad heat-up on change in coolant water volume, and the effect of fuel heat-up (Doppler), also receive further study.

The user community is asked to provide any additional documentation that they may have regarding fuel assembly and control assembly design drawings and specifications. Dimensional details of the junction between the boron-steel absorber box and the fueled follower at this time are ill-defined. With more information about that junction, it will be possible to locate the control and transient rods more precisely for initial criticality and for each class of test.

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