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WSRC-TR-92-439

**NUCLEAR REACTOR TECHNOLOGY
AND SCIENTIFIC COMPUTATIONS**

KEY WORDS: Code Validation
Technical Specification
Reactor Restart

RETENTION PERIOD: Lifetime

**FURTHER CODE VALIDATION
OF
TECHNICAL SPECIFICATION BASES(U)**

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DOCUMENT: **WSRC-TR-92-439**

TITLE: **FURTHER CODE VALIDATION OF TECHNICAL
SPECIFICATION BASES (U)**

TASK:

APPROVALS

M. R. BUCKNER, MANAGER
SCIENTIFIC COMPUTATIONS SECTION

DATE: 9-15-92

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DATE: 9-11-92

1.0 INTRODUCTION AND SUMMARY

Personnel from the Applied Physics Group (APG) and from the Reactor Technology Section devised the K-14 reactor restart test program. The primary purpose of that program was to acquire data which could be used to further validate the computer codes which supported the Technical Specification Bases. The test program was highly successful and a large amount of useful data was obtained. DOE has now requested (Ref. 1) the schedule and plan for the additional code validation activities. A response to DOE was made (Ref. 2) in which an outline of WSRC's planned activities was presented, but specifics were not discussed. This memorandum is intended to provide additional details on these activities.

The WSRC plan consists of five major activities. The first activity is to catalog and document the measured data. The second activity is to investigate the radial and axial neutron flux distributions. The results from this task will complement the analyses being performed in the final three activities: steady state reactivity effects, transient reactivity effects, and planned and unplanned shutdowns. Work on these activities is scheduled to be completed by the end of fiscal year 1993.

2.0 DISCUSSION

2.1 CATALOGING AND DOCUMENTING TEST DATA

The Power Ascension Data Acquisition System (PADAS) was installed to collect data during the K-reactor power ascension test program. The Applied Physics Group (APG) is working with the Equipment Engineering Section (EES) to extract data from the PADAS and transfer it to CFS for permanent storage. CFS is a centralized file management and storage system that provides permanent file space. The files on CFS can then be accessed from the various computing environments at SRS. A checksum system was specified by EES to ensure that the PADAS data was not corrupted during the transfer.

Test data of interest for code validation includes the data from two unplanned reactor SCRAMs and the data from various restart tests:

- . RSP 90-007-03 Initial Criticality
- . RSP 90-007-04 Reactivity Computer Calibration
- . RSP 90-007-05 Differential Rod Worths (Partial)
- . RSP 90-007-07 Single Partial Rod Worth Measurements

- RSP 90-007-08 Power Ascension (Reactivity and Flux Changes)
- RSP 90-007-12 Partial Rod Gang Insertion Tests
- RSP 90-007-14 Axial Power Monitor Rod Response Test
- RSP 90-007-15 Safety Rod SCRAM using REMACS
- RSP 90-007-18 Single Side Injection of SSS

It is expected that future analyses will be interested primarily in the PADAS data associated with these tests - hence the emphasis placed in moving that data to CFS. However, steps are also being taken to place all additional electronic data on CFS. This additional data includes digital Travelling Wire Flux Monitor (TWFM) traces and the data from the control computer (the TaskCommon, EXPWR, and DELTAP files) along with the Control Rod Position Log (CRPL). Consequently, all electronic data from the restart test program will be placed in a central location (on CFS) using a file naming structure that has been previously agreed upon. Analysts will then be able to access this data by different methods.

Utilities currently exist to access and display data from these files, and more utilities will be developed as needs arise. Utilities that have been used in the past to access the control computer data include FACEMAP (Ref. 3), RMS (Ref. 4), TCMONITOR, (Ref. 5), and RXBETA_TCCAL (Ref. 6). TWFM Eval (Ref. 7) is a new utility developed to evaluate TWFM data. Tools for reviewing the PADAS data are also under development (Ref. 8). In addition, several references are available to aid analysts in identifying and retrieving the data from the various files (Refs. 8 - 10).

In summary, the focus of this task is not to develop new utilities or enhance old utilities that access the electronic data. Instead the goal is to consolidate the data and provide a means for permanent storage. The documents that describe how this has been done will be the deliverables from this task. At least two documents are expected from this task.

Milestones

- ♦ Final documentation - PADAS data to CFS (Nov 1992)
- ♦ Final documentation - Non-PADAS data to CFS (Dec 1992)

2.2 INVESTIGATE NEUTRON FLUX DISTRIBUTION

The reactor power ascension test program provided clear evidence that calculations of the axial flux shapes using observed control rod positions often do not agree with the axial flux shapes measured in the reactor. Those calculations are made with GRIMHX (Ref. 11), a coarse-mesh, finite difference reactor code that is the standard computational tool for reactor physics analysis at SRS. There are several reactor states that illustrate this axial flux shape deficiency, and those states would be ideal as benchmark states for GRIMHX.

There are three different observed critical control rod configurations under the same reactor conditions: essentially zero power with zero fission product inventory. In these tests (RSP 90-007-05) the partial rods were positioned at 700, 800, and 950 vu while compensating with the full length rods. Though GRIMHX predictions of the initial critical position (partials at 800 vu) were in close agreement with observation, GRIMHX overestimated the amount of full rod necessary to compensate for a partial rod repositioning to 700 vu while underestimating the amount necessary in moving to 950 vu. These three states provide clean, unambiguous data uncomplicated by any considerations for burnup, fission product change, or temperature change.

One can deduce from the above tests that GRIMHX computes a flux shape that is more bottom heavy than is actually present. Direct confirmation of this can not be obtained for these tests since flux levels were below the operability range of the axial power monitors (APMs) and the internal hafnium-rodium neutron detectors. However, in a later test (RSP 90-007-12) this same bias was observed. In that test a movement of gang 1 partials from 800 to 850 vu was predicted to require compensation by withdrawing full rods. Instead, full rods had to be inserted. It should be noted that Reactor Operations personnel determined prior to executing the procedure that rods would have to be inserted, not withdrawn as predicted. Their determination was based upon the observed axial flux shape. The axial flux shape was in reality significantly more top heavy than predicted by GRIMHX. Unlike the three earlier measurements, this test does have measured axial flux data. However, the test analysis is more difficult because of differential fission product build-in considerations.

Investigation into this discrepancy is continuing. One report (Ref. 12) discusses some of the work that has been done in this arena. Studies performed to date by APG indicate the cause

of this disagreement is not due to any deficiency in GRIMHX's numerical models. Instead, there are indications that there is greater variation in core materials at the ends of the target and fuel tubes than previously assumed. The investigation has focussed on this, and a report on the findings will constitute the deliverable for this task.

Milestones

- Final documentation - Mark 22 axial geometry (Nov 1992)
- Recommendations to correct observed discrepancy (Mar 1993)

2.3 STEADY STATE REACTIVITY EFFECTS

Data from a number of different reactor states was cataloged during the power ascension testing. This includes the critical full rod positions under various conditions:

- with partials at different positions,
- with different partial rod complements, and
- at various "wait" points during the ascension to full power where full rods had to be adjusted to compensate for temperature and fission product build-in.

In addition to axial and radial flux shapes and the critical rod positions, there is a multitude of data associated with each reactor state. The number of reactor states is too large to consider them all, and much of the recorded data is not of interest to the analysts. Consequently, as part of this task, it is our plan to examine only some subset of the available data.

It is important to note that, in essence, the analysis of the steady state events has already been performed. Predictions of anticipated control rod moves for the various tests had been made (Refs. 13 - 15) prior to K-14 restart. These predictions, which were generally in the form of control rod worth curves, were the basis for estimating critical rod positions and for inferring temperature and fission product worths. Moreover, the predictions were made using the same computational tools (primarily GLASS (Refs. 16 and 17) and GRIMHX) that will be used in analyzing the test data. Hence, any test data analysis will be of limited use unless that analysis uses the improved axial geometry findings from the previous task.

The thrust of this task will be to examine several steady state reactor conditions using the improved axial geometry. The

exact number of states which will be considered has not been determined, but will include the three initial critical rod positions discussed earlier as well as the RSP 90-07-012 gang 1 partial rod insertion test. The analysis tool will be the GRIMHX reactor code using cross sections generated with GLASS. The results from the analysis will be used to better quantify uncertainties associated with this methodology. At this point, only a single document is anticipated from this task.

Milestones

- ♦ Final documentation - Analysis of selected steady state reactor conditions (Mar 1993)

2.4 TRANSIENT REACTIVITY EFFECTS

Several restart tests involved following the reactor power increase after inserting positive reactivity via a control rod move. As with the previous task, this task will involve examining several of these tests. Computational tools to be used in the analysis will be GRIMHX (using GLASS-generated cross sections) and the AA3 (Ref. 18) accident analysis code. Various small PC-based point kinetics codes may also be used. It is our intention to use the new axial geometry model in all GRIMHX calculations.

The control rod worth transients discussed above were a consequence of the planned restart test program. An additional transient, which was not planned, will be examined as part of this task. On July 27, the partial length rods in gang 3 continued to drive into the reactor after the operator released the controlling toggle switch. This accident scenario was not previously considered in the analysis for Chapter 15 of the Safety Analysis Report (SAR). The physics analysis of this event (Ref. 19) determined that the event was bounded by the gang rod withdrawal accident which had been considered in the SAR. A reanalysis of this event using the 3-D transient code TRIMHX (Ref. 20) is planned.

Milestones

- ♦ Final documentation - Analysis of selected control rod transients (Jul 1993)
- ♦ Final documentation - Transient analysis of the gang 3 partial rod rod drive event (Aug 1993)

2.5 PLANNED AND UNPLANNED SHUTDOWNS

The revised and abbreviated final restart test plan had provisions for only two planned shutdowns - a safety rod SCRAM (using REMACS) from a distorted axial flux shape and a shutdown via activation of a single side of the Supplementary Safety System (SSS). In addition to these two planned shutdowns (RSP 90-007-15 and RSP 90-007-18), there were three unplanned SCRAMS. There were two automatic SCRAMs caused by failure of both safety computers, and there was an operator initiated SCRAM when a relay failure caused a gang 3 partial rods drive-in. Data from these three safety rod shutdowns was also recorded on the PADAS.

In this task, as a minimum, the data from the planned safety rod SCRAM and the data from the SSS activation will be analyzed. The approach used to analyze the SSS test will be different than that used in reviewing the safety rod tests. The different approaches are mandated by the difficulties in attempting to calculate the worth of ink in SSS tests. The approaches that will be used are described below.

In analyzing the safety rod test(s), neutron flux data from the internal and/or external flux detectors will be combined to form a measured global neutron flux response. The measured global flux will be input into an inverse kinetics code to compute reactivity as a function of time. This inferred reactivity transient will then become a comparison criteria for the physics analysis tools. Specifically, GRIMHX will be used to calculate a "computed" reactivity transient for comparison against the inferred reactivity. The initial pre-shutdown state with the observed control rod configuration will be modelled using GRIMHX, followed by additional GRIMHX calculations as the safety rods insert into the core.

Though safety rod worth curves under different pre-incident flux shapes are of interest, at this time there are plans to examine only one of the cases. The most limiting case (shutdown from distorted axial flux shape) will be analyzed first, with the additional cases to follow as time permits.

The approach which will be used in analyzing the SSS test will be similar to the safety rod tests. Namely, the reactivity worth of the SSS will be computed with an inverse kinetics code using an inferred global flux response. Since the SSS activation was performed under limiting conditions (single side activation only, ink concentration at lower bound, etc) the inferred reactivity will become, after consideration for uncertainties,

the expected worth of the SSS. The current estimated worth of the SSS which was used in the K-14 SAR was based upon the last operational test of the SSS (in 1975). The analysis of that test data followed similar lines.

A direct calculation of the worth of the SSS system is not feasible and will not be attempted. Such a calculation would require detailed information on the concentration of ink throughout the core as a function of time. That information is not known, and would be impractical (if not impossible) to obtain. Consequently, the worth of the SSS will continue to be inferred from the neutron flux response.

Milestones

- ◆ Final documentation - Analysis of safety rod and SSS Shutdowns (Apr 1993)

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