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MEMORANDUM FOR: Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Robert B. Minogue, Director
Office of Standards Development

FROM: Saul Levine, Director
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SUBJECT: RESEARCH INFORMATION LETTER #59 - TRANSIENT FUEL ROD BEHAVIOR CODE: FRAP-T4

Reference:

1. Research Information Letter #29, "Fuel Rod Analysis Computer Code: FRAP-T3," June, 1978.
2. "FRAIL-4: A Fuel Rod Failure Subcode," by J. D. Kerrigan, D. L. Hagrman, and S. O. Peck, CDAP-TR-012, April, 1978.
3. "MATPRO-Version 10 - A Handbook of Materials Properties for Use in the Analysis of Light Water Reactor Fuel Rod Behavior," ed. by G. L. Reymann and D. L. Hagrman, TREE-NUREG-1180, February, 1978.
4. "FRAP-T3 - A Computer Code for the Transient Analysis of Oxide Fuel Rods," TFBP-TR-194, August, 1977.

This Research Information Letter transmits the description and assessment documentation of the latest version of the transient fuel rod behavior computer code - FRAP-T4.

1. INTRODUCTION

FRAP-T4 is a best-estimate computer code that calculates the thermal and mechanical response of a nuclear fuel rod during normal, off-normal, and transient conditions. These conditions include such events as Loss-of-Coolant Accidents (LOCA), Reactivity-Initiated-Accidents (RIA), Power-Cooling-Mismatch Accidents (PCM), Inlet Flow Blockage Accidents (IFB), and Anticipated Transients Without Scram (ATWS). The code is utilized to analyze such events for the Power Burst Facility (PBF) test program, and the Loss of Fluid Test (LOFT) program. The models developed for the code are also needed and used in systems codes such as RELAP and TRAC, either by direct linking of the codes, or by insertion of specific FRAP-T4 models into the fuel behavior modules of these codes. Finally, because of its modular structure, the code can and will be used in systems codes

adapted for licensing applications by substitution of appropriate evaluation models in place of the best-estimate models. One example of this is the FRAP-T4 LACE code to be used in the WRAP computer code package currently under development by RES for the use of the Office of Nuclear Reactor Regulation. Another example of NRR use of the code was the recent study of operational transients by the Core Performance Branch of the Division of Systems Safety. The overall purpose of that study was to provide a better understanding of fuel duty under conditions where current licensing criteria are applied.

2. RESULTS AND EVALUATION

A detailed description of the code is provided in Enclosure 2 and in Research Information Letter #29 on the previous version of the code (FRAP-T3) (Reference 1). Table I of Appendix A shows the models contained in the last three versions of the code, and Table II summarizes the improvements and added models to the T4 version of the code over previous versions.

Through the addition of models and correlations not in FRAP-T3, the FRAP-T4 version now has the capability to analyze the entire sequence of a LOCA event through reflood. It also contains improved thermal and mechanical models shown to be necessary in the assessment of FRAP-T3. These improvements yield better accuracy in the prediction of overall fuel rod behavior.

A major emphasis in the production of NRC fuel codes is placed on independent code assessment. In this task, the code is given to a group other than the code developers to assess its performance against data, most of which are not available to the code developers. The primary objective is the documentation of the code capabilities to the extent allowed by available data. The code responses chosen to be assessed correspond to fuel behavior phenomena that are amenable to experimentation, and which are important to safety analyses. A secondary objective is to identify those code predictions which compare poorly with experiment and, therefore, require further study.

The assessment of the transient capabilities of FRAP-T4 was broken down into three categories; namely, (a) off-normal data comparisons, (b) transient data comparisons, and (c) out-of-pile tube rupture comparisons. Over 800 computer cases were run for the 200 rods and 400 cladding tubes considered. A brief summary of the results of each assessment category is given below. More specific details are available in Enclosure 3.

a. Off-Normal Data Comparisons

Off-normal data comparisons were assessed with respect to two characteristics: (1) the ability of the code to predict the flow and power conditions which cause departure from nucleate boiling (DNB), and (2) the ability of the code to calculate ramp-induced cladding failures caused by pellet-cladding interaction (PCI).

The former (i.e., DNB data comparison) utilized data from approximately 90 rods from the PBF test series, and tested the B&W-2, W-3, LOFT, and the CE-1 critical heat flux correlations (CHF) available as options in the code. The results are detailed in Enclosure 3, and can be summarized by the statement that the LOFT correlation best predicts the onset of DNB (within 25% of the data for 92% of the cases) followed by the W-3, B&W-2, and the CE-1 correlations.

The second off-normal characteristic evaluated was the ability of the code to compute power-ramp induced cladding failures. Although the FRAP code has no stress-corrosion cracking model in its structure, the FRAIL subcode (Reference 2) provides a failure algorithm based upon cladding stress conditions. In brief, the results show that the code reasonably predicts clad failure probabilities (90 power ramp tests were used) using the FRACAS-I (see Enclosure 2 for details) mechanical subcode in conjunction with FRAIL. Improved relocation modeling and cracked fuel deformation models, scheduled for testing in the steady-state code FRAPCON-2 by both INEL and BNWL, should yield even better predictions when incorporated into a later version of FRAP-T.

b. Transient Data Comparisons

To evaluate transient data comparisons, three fuel temperature response characteristics were assessed: (1) initial fuel temperature after shutdown, (2) equilibrium fuel temperature, and (3) the thermal decay constant. The initial temperature reflects the code's ability to approximate fuel temperatures prior to a transient. For best results, however, the code should be linked to the best-estimate, steady-state code FRAP-S3, which provides more accurate initial conditions that reflect the effects of burnup and power history (e.g., fission gas release, cladding creep, etc.). As a stand-alone code, FRAP-T4 was able to predict the initial temperature of Halden and PBF rods with a standard error of ± 280°K. Figure 1 of Appendix A shows the results.

The equilibrium temperature of the fuel corresponds to the leveling off of the fuel center temperature after scram (no coolant loss). The data from the PBF test program show that the equilibrium temperature can be predicted to within a standard error of ± 54°K (see figure 2 of Appendix A). This low error value is a significant improvement over FRAP-T3 results and reflects the effect of the new fuel relocation on the gap size, crack geometry, and gap heat transfer resistance.

The thermal decay constant is a measure of the rate of heat loss of the fuel after reactor shutdown. It will vary according to the shutdown ramp rate and can be calculated from fuel centerline temperature data versus time. For the PBF rods, FRAP-T4 predicted the decay constant to within a standard error of ± 5.4 seconds for ramp rates, which caused a range in measured decay constants from 12 to 140 seconds.

c. Out-of-Pile Cladding Tube Rupture Comparisons

The FRAP-T code has two models available for analyzing cladding rupture data. The first is an empirical model using the FRAIL subcode and data correlations, whereas the second model is based upon the MATPRO (Reference 3) stress-strain correlation, instability strain criteria, and the BALLOON subcode. The empirical model is unchanged from that in FRAP-T3, but the MATPRO-based model has an improved strain-rate model and now uses a true-stress, true-strain law for Zircaloy as opposed to the previous engineering stress-strain law.

The data comparisons included 158 single rod tests covering a wide range of heating rate, pressure, and temperature conditions. Cladding temperature and pressure histories were directly input to the code so that the mechanical models could be assessed separately from any other code uncertainties.

Figure 4 in Appendix A shows the results of those tests in which nominally constant pressure rods were ramp heated to failure using the BALLOON/MATPRO model. The standard error in the burst temperature was ± 290°K. The corresponding curve for the FRAIL model (from the verification document for FRAP-T3 - Reference 4) is shown in Figure 5. Comparing the two figures shows that the empirical model is currently a better "fit" to the data than the more deterministic model. Figures 6 and 7 show the corresponding results for those tests in which the pressure was ramped at constant temperature. Again the FRAIL model gives a better "fit" to the data.

The code's ability to predict the circumferential burst strain was not well represented (a standard error of ± 57% strain was determined). The reason for this, however, is not necessarily the fault of the models, since measured burst strains have data scatter ranging to over 100% (see Figure 8 in Appendix A). It is expected that the implementation of the Multi-Rod Burst Test Data (MRBT) into the BALLOON subcode, and the new stress-strain data for Zircaloy from the University of Florida will improve the predictive capability of the models and refine the data base used.

3. USER RECOMMENDATIONS

It was found during the code assessment that for rods characteristic of standard BWR and PWR reactors the best options to be used in the FRAP-T4 code are the FRACAS I deformation package, and the annular gap conductance model (Ross-Stoute) using the pellet relocation and cracked fuel thermal conductivity models. The following assessment results should be considered when analyzing code output.

a. When used as a stand-alone code, FRAP-T4 steady-state centerline temperature and pressure levels are likely to be overpredicted for rods with large cold annular gaps (greater than 3% of pellet O.D.) and prior burnup.

b. High temperature cladding rupture conditions are currently better represented by the FRAIL subcode option than by the BALLOON/MATPRO option.

A further aid to the user is provided in Table III of Appendix A which contains the standard error predictions of the currently assessed steady-state and transient NRC fuel behavior codes (FRAP-S3 and FRAP-T4). This table should be used when assessing the significance and range of applicability of code results. Model improvements are continually being made to remove some of the present limitations, and these improvements will be incorporated in future versions or modifications of the code.

In summary, the FRAP-T4 code is considered to be an excellent analysis tool for fuel behavior under transient conditions as evidenced by this report and its performance in predicting PBF and LOFT test results. When used with the above recommendations, the user can be assured of obtaining current best-estimate predictions of fuel behavior.

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Enclosures:
1. Appendix A (contains three tables and eight figures referred to in the text).
2. CDAP-TR-75-027, "FRAP-T4: A computer Code for the Transient Analysis of Oxide Fuel Rods," July, 1978.
3. CVAP-TR-78-18/R1, "FRAP-T4: A Computer Code for the Transient Analysis of Oxide Fuel Rods - Model Assessment Report," December, 1975.