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

FROM: Saul Levine, Director
Office of Nuclear Regulatory Research

SUBJECT: RESEARCH INFORMATION LETTER #36 EVALUATION OF GENERAL ATOMIC CODES: OXIDE-3, SORS, TAP, AND RECA

Introduction and Summary

This memorandum transmits the results of completed research to evaluate the General Atomic Company's (GAC) computer codes, OXIDE-3, SORS, TAP, and RECA. This work was completed as part of the High Temperature Gas-Cooled Reactor (HTGR) Safety Program at Brookhaven National Laboratory and Oak Ridge National Laboratory under the direction and sponsorship of the Advanced Reactor Safety Research, Office of RES. This particular research was initiated by RES, three years ago, to evaluate the applicability and utility of the codes for the ARSR/Gas-Cooled Reactor Safety Program. The objectives of the evaluations also included an assessment of the models and numerics used in the codes and to note under what conditions or for what scenarios the codes were useful. Since these codes were developed by the single vendor for HTGR's in the United States, it is felt that these code evaluations could be useful to NRR. The results of these evaluations have been discussed at Research Review Group meetings, Mid-year reviews, and other meetings with participation by the NRR staff.

OXIDE-3 was developed to analyze the transient response of certain state variables of an HTGR during an accident involving the inleakage of steam and/or air into the helium primary coolant system. Primary tasks of the code are to calculate the primary coolant constituents as a function of time, their resultant chemical interactions with graphite, and their possible egress into the containment building.

SORS was written to calculate the release of fission products from the fuel into the primary coolant during a hypothetical sustained loss of forced circulation (LOFC). The code calculates the release of fission products, both volatile and non-volatile.

TAP was developed as a means of predicting the dynamic behavior of HTGR nuclear steam supply systems (NSSSs) for a variety of operational and accident cases. It is a flexible, general-purpose code that evolved from several component codes.

RECA was developed to analyze postulated accidents which result in a loss of normal core cooling and a subsequent reliance on the emergency cooling systems. For such accidents, it is essential that the analysis include a three-dimensional model of the thermal-hydraulic behavior of the core.

Evaluation Results

OXIDE-3[1]: The code has been compared with the GOPTWO code for several steady state problems involving constant steam ingress at full operating power. Taking into consideration differences in the two codes, the agreement was very good. The code was evaluated with respect to the results obtained for several other problems (e.g. steam ingress with and without scram, air and air/steam ingress following depressurization). The accuracy of the solution and the adequacy of the models depend strongly on the accident scenario under investigation. Additional weaknesses noted in the code include: (1) existence of the core support post and blocks is not accounted for and these relatively reactive graphites should have an effect on the primary coolant impurity level and on the thermal inertia; (2) the selection of the mesh size and nodal point distribution can lead to significant errors when the chemical reaction rates are high (e.g. high temperatures, air ingress); (3) the time step selected for use in OXIDE-3 is intimately linked to the particular problem being investigated; (4) the analysis for the possibility of flammable mixtures entering the containment vessel is in error because of discrepancies in the mass balance.

SORS[2]: The models and numerics for the release of the gaseous and halogen fission products wherein no transport delay from the fuel to the coolant channel is assumed are probably as accurate as needed. The error in the results is dominated by the uncertainties in the empirical fuel failure model, in the experimental data on release rates from fuel particles as a function of temperature and in the predicted temperature history.

For the release of the remainder of the fission products, the SORS-G code introduces an explicit treatment of the fission product diffusion through the graphite fuel element web. If the rates of decay and of transport through the web are comparable, the procedure used in SORS-G to decouple these processes is questionable. The expression used to calculate the rates of evaporation of the fission products from the graphite surface of the coolant channels is in error. The net effect of this error and the numerical treatment of the fission product transport in the coolant channels is indeterminate. A comparison of the evaporation and transport along the coolant channels was made with another code using the same fission product release and graphite diffusion data, a corrected evaporator expression and a different coolant channel transport model which is believed to be as rational as the approach implemented in SORS-G. For the particular case examined the release curves from the core were displaced by up to a few hours.

TAP[3]: Direct comparisons were made for certain component models in the code; however, no direct comparison of TAP predictions with ORTAP-FSV output could be made since the available version of TAP was set up to simulate LHTGRs. The general features of ORTAP-FSV and TAP were compared and gave equivalent results. Comparisons were made with the CORTAP code for steady-state temperatures at 100% power and the agreement was excellent. The core model is primarily intended for at-power simulations and is not valid for analyses of very low-core-flow accidents, which are relegated to the RECA code. Accidents involving very large and sudden reactivity insertions would also be beyond the domain of TAP because of point neutron-kinetics assumptions. The steam generator model in TAP was compared with analyses performed using the BLAST code and the agreement was very good in spite of the fact the models used are quite different. Recent analysis on FSV rod pair withdrawal accidents gave good results for those cases where comparisons could be made.

The TAP code employs fairly complex and sophisticated modelling and as such is susceptible to misuse and errors in setting up to run various analyses. Other possible problem areas in using the code are: (1) the coolant-channel-centered cylindrical model for the core may give unrealistic predictions of fuel and moderator temperature; (2) the limitations of using a simplified, algebraic input-output model for the BOP are not clearly identified; and (3) neglecting the rotary moment of inertia for the circulator-turbine may cause underestimation of the primary system flow during coastdowns.

RECA[3]: Comparisons of RECA predictions were made with the ORNL codes ORECA and FLODIS for certain Design Basis Depressurization Accidents. The RECA calculations were performed by GAC because of problems in getting the code to run on the ORNL computer system. Core temperatures and net flow histories were generated from TAP runs performed by GAC. The predictions by RECA and ORECA indicate good agreement and the comparison with FLODIS was acceptable. FLODIS predicted slightly higher outlet temperatures. This is accounted for by the finer mesh spacing in FLODIS which permits representation of temperature gradients and flow distributions within the refueling regions.

The RECA code has considerable flexibility built in to allow for simulation of a variety of component designs, for adjustment of node spacing, for variation of model and controller parameters, etc. Because of this flexibility and the code's sophistication it is highly susceptible to user input errors. Other significant problems identified in RECA include: (1) the potential problems with interfacing RECA and TAP are not clearly identified; (2) the afterheat generation curve following a reactor trip appears to decrease too rapidly; (3) the assumption of instantaneous depressurization for all blowdown accidents limits the scope of accidents that can be analyzed; and (4) the assumption of a single ring-shaped node for the side reflector at each axial level may lead to nonconservative peak temperature predictions.

Future Work

All work in this area has been terminated.

Conclusions and Recommendations

The applicability and utility of the GAC codes for RSR-HTGR Safety programs was found to be very limited. The OXIDE-3 code appears to handle oxidation of graphite by moisture appropriately under normal operating conditions. The code will require further experimental verification before the limits of accuracy can be established. The SORS code in the form presented for our evaluation appears to have some serious deficiencies in the models which places doubts on the analysis performed with the code. The TAP and RECA codes give good agreement with other analytical tools and appear to be useful and appropriate for their designed applications. Quantification of the accuracy of the codes will require further comparison with operating reactor conditions. A vendor verification program for the codes is recommended. For further information on the results of the evaluations of the these codes, contact Dr. John Larkins, in RES.

Saul Levine

Saul Levine, Director
Office of Nuclear Regulatory Research

Enclosed References:

(1) J. Skalyo, Jr., L.G. Epel and C. Sastre, "An Analysis of the Methods Utilized in OXIDE-3", BNL-NUREG-50810, (April 1978)

(2) J.M. Dickey, "An Analysis of SORS: A Computer Program for Analyzing Fission Product Release From HTGR Cores During Transient Temperature Excursions", BNL-NUREG-50806, (April 1978)

(3) S.J. Ball, J.C. Cleveland, and J.P. Sanders", Evaluation of the General Atomic Codes TAP and RECA for HTGR Accident Analyses", ORNL/NUREG/TM-178 (May 1978)

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C Gupta 9/29/78

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