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INVESTIGATIONS OF POSTULATED ACCIDENT SEQUENCES FOR THE FORT ST. VRAIN HTGR

S. J. Ball   J. C. Cleveland   J. C. Conklin

M. Hatta*   J. P. Sanders

Oak Ridge National Laboratory

Oak Ridge, Tennessee, USA

* Guest Researcher from IHI, Japan

DISCLAIMER

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ABSTRACT

The systems analysis capability of the ORNL HTGR Safety analysis research program includes a family of computer codes--an overall plant NSSS simulation (ORTAP), and detailed component codes for investigating core neutronic accidents (CORTAP), shutdown emergency-cooling accidents via a 3-dimensional core model (ORECA), and once-through steam generator transients (BLAST). The component codes can either be run independently or in the overall NSSS code.

Verification efforts have consisted primarily of using existing Fort St. Vrain reactor dynamics data to compare against code predictions. Comparisons of core thermal conditions made for reactor scrams from power levels between 30 and 50% showed good agreement. An optimization program was used to rationalize the differences between the predicted and measured refueling region outlet temperatures, and, in general, excellent agreement was attained by adjustment of models and parameters within their uncertainty ranges. However, more work is required to establish a unique and valid set of models.

Several postulated accident sequences have been analyzed, including rod pair withdrawal accidents, design basis depressurization accidents, and loss of forced-convection cooling accidents. Sensitivity studies are run in conjunction with each accident to determine the importance of both model and parameter uncertainties.

INTRODUCTION

The Reactor Safety Research Division of the U. S. Nuclear Regulatory Commission (USNRC) has sponsored a high temperature gas-cooled reactor (HTGR) safety research program at Oak Ridge National Laboratory (ORNL) which began in July 1974. The goals at that time were to develop independent capabilities for assessing and confirming analyses of large HTGR plant safety. Subsequently, the work was redirected to specific licensing-related accident analyses of the Fort St. Vrain (FSV) HTGR.

Descriptions of the ORNL-developed codes, results of some code verification work, and several analyses of postulated FSV accidents performed under this program are presented.

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DESCRIPTION OF THE ORNL CODES

Because of the unique coupling features of the FSV reactor, it is necessary to have a detailed simulation of the entire nuclear steam supply system (NSSS) to determine the course of many of the postulated accidents. As seen in the flow diagram (Fig. 1) the primary system is coupled to the secondary and balance of plant (BOP) both via heat transfer through the steam generators and due to the fact that the turbine-driven primary circulators derive their steam from the high pressure turbine exhaust. In addition, the major plant control loops use BOP parameters to control core power and flow, and several safety trip signals originate in the BOP.

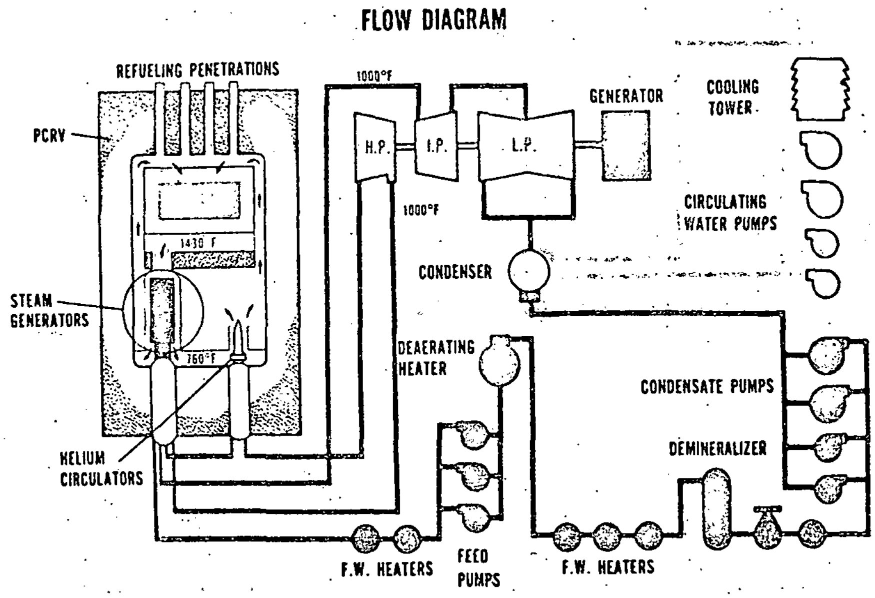


Fig. 1. FSV reactor flow diagram.

The computer codes developed at ORNL are designed to analyze many of the major design basis accidents postulated for FSV. The primary ORNL HTGR accident code is called ORTAP,[1,2] which is similar in purpose to a combination of the General Atomic Co. (GAC) codes TAP[3] and RECA[4], although modeling techniques and assumptions used in the ORNL and GAC codes differ significantly. ORTAP has been operational for about 2 1/2 years. In the ORTAP code, the core is normally simulated by a coupled heat transfer-neutron kinetics single channel model (CORTAP[5]). An alternate core model (ORECA[6]) is used for transients involving post-trip power and flow conditions. The ORECA model includes 3-dimensional temperature distribution calculations, accounts for variations in flow distribution among the individual refueling regions, and can simulate reverse flows.

A steam generator and reheater module is simulated with a multinode, fixed-boundary, homogeneous-flow model (BLAST[7]). Time dependent equations for conservation of energy, mass, and momentum are solved for both the helium and water/steam sides by an implicit integration technique. Transients involving both startup and flood-out can be simulated.

The detailed turbine-generator plant simulation calculates enthalpies, pressures, and flows in each stage and at extraction points of the high, low, and intermediate pressure turbines. The dynamic responses of the main steam bypass system, each of the feedwater heaters, and the deaerator are all modeled explicitly. The circulator turbine model includes calculations of steam side turbine enthalpies, pressures, and flows as well as the circulator side performance. The control systems for the overall plant are also modeled in detail.

ORTAP's component codes can either be used in the overall plant simulations or independently. For example, CORTAP is often used to calculate hot-channel behavior, given independently-generated power, flow, and inlet temperature functions as inputs.

ORTAP and its component parts were developed as best estimate rather than conservative or worst-case models. Occasionally it has been found that the use of what one would think to be conservative values of a parameter in an accident simulation leads to non-conservative results. Hence the typical procedure used to determine worst-case results is to vary the least certain parameters and models (within their uncertainty ranges) and note the sensitivity of the response for each type of accident.

CODE VERIFICATION

Complete or absolute verification of accident codes is difficult due to the understandable reluctance of plant operators to generate the needed data. Hence our attempts to provide partial verification are limited to: 1) comparisons of results with those of independently-developed codes; 2) comparisons with plant data from normal transients; 3) comparisons with specially-designed plant tests; and 4) use of scale model experiments.

The ORTAP code and most of its component codes have been compared collectively and individually with their GAC counterparts for a variety of transients. In general, the comparisons have been found to be good. Several specific comparisons are discussed in the following section.

Most of the experimental verification efforts to date have been comparisons of ORECA code predictions with data from FSV scrams. Four different scrams from power levels between 30 and 50% have been used. GAC has supplied the necessary input data, including circulator inlet temperature, core flow, pressure, and power (afterheat) vs. time after the scram. The initial conditions are the 37 measured refueling region outlet temperatures and estimates of each region's power peaking factor. Comparisons are then made of the computed and measured outlet temperature transients for all regions. Although there is no unique combination of models and parameters that will produce a good fit to the data, (and hence no guarantee that an optimized model is valid), optimization schemes do suggest areas where the code may need improvement.

The original ORECA best estimate calculations of the scram tests were usually in reasonably good agreement with the data; however, there were some significant differences. In Fig. 2, a typical ORECA prediction of measured region outlet gas temperature (TGO) is seen to be low after 20–30 min following the scram. This has been attributed both to an overestimate of core flow (or underestimate of bypass flows) and to deficiencies in the original dynamic model of the TGO thermocouples.

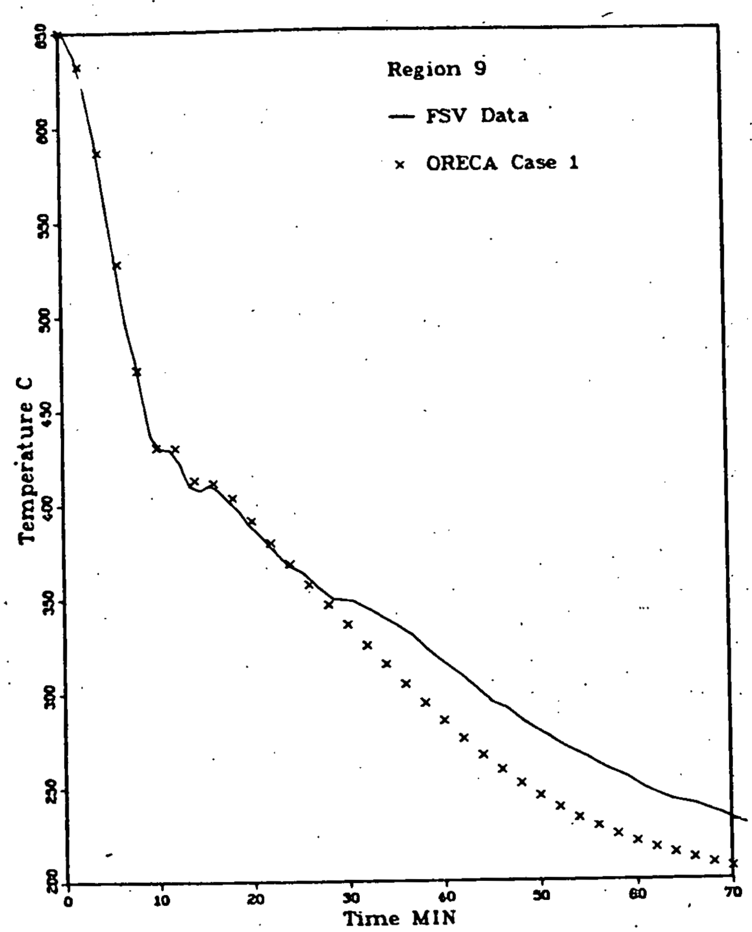


Fig. 2. FSV scram test of Aug. 6, 1977 from 28% power — comparison of best estimate reference ORECA code predictions of measured gas outlet temperature from region 9 vs plant data.

The core bypass flows include those through gaps in the refueling region and side reflector blocks, as well as flows bypassing the core barrel entirely. None of these is directly measurable.

The TGO thermocouples are in large graphite sleeves, and have time constants of ~2 min at rated conditions. Several versions of the thermocouple model have been used subsequently, with the most significant improvement being the addition of T^4 radiation effects. In order to properly account for these effects, however, ORECA had to be revised to model the lower part of the core support blocks separately (rather than lumped with the lower reflector as before) since the support blocks cool down much more slowly after a scram.

Other modifications required to produce a good fit were adjustments in the assumed peaking factors for many of the regions, especially those near the outer ring; and adjustments of the assumed temperature rise of the helium between circulator inlet (measured data) and the core inlet. This rise is due both to the heat of compression from the circulators and to heat transfer to structures between the circulators and the core inlet plenum.

An optimization code was used to find the ORECA parameters that give the best least-squares fit to the data. The optimization code utilizes the differences in the responses generated by ORECA for several selected parameter-variation cases. By comparing these responses to the FSV data, the optimization code computes a set of optimized parameters. This set is limited by what are judged to be reasonable uncertainty ranges. After these parameter adjustments are incorporated into the ORECA code, the agreement is generally excellent, with typical results shown in Fig. 3. One discrepancy still remaining (especially in the higher-power tests) is a distinct difference in the shape of the curve for several regions adjacent to the side reflector (Fig. 4). These differences are thought to be due to interactions with the side reflectors that are not yet explicitly modeled. Work on the optimization is still in progress.

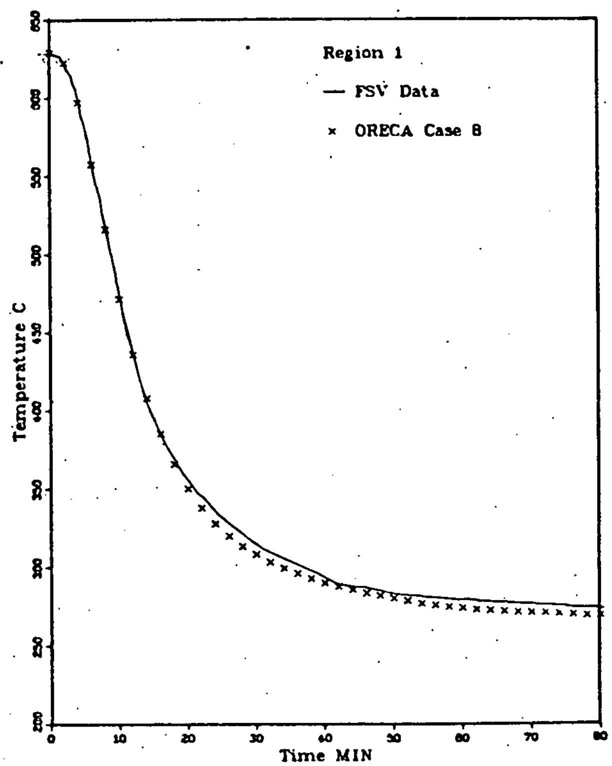


Fig. 3. FSV scram test of Oct. 25, 1977, from 40% power — comparison of optimized ORECA code predictions of measured gas outlet temperature from region 1 vs plant data.

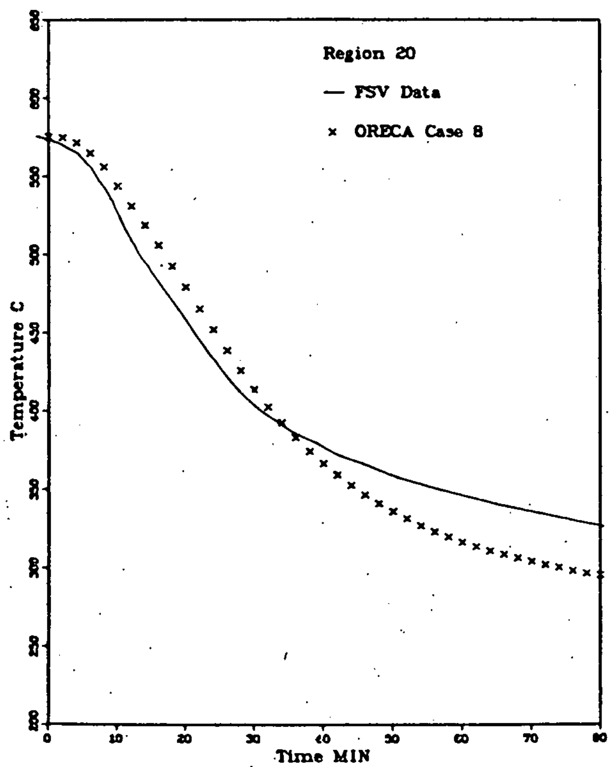


Fig. 4. FSV scram test of Oct. 25, 1977, from 40% power — comparison of optimized ORECA code predictions of measured gas outlet temperature from region 20 vs plant data.

Other verification efforts include proposed special FSV tests to check the predictability of reverse flow in the refueling regions, in-situ thermocouple time constant tests, and scale model (air) experiments to determine the effects of reverse flow plume impingement on the upper plenum cover plate temperatures.

POSTULATED ACCIDENT ANALYSIS EXAMPLES

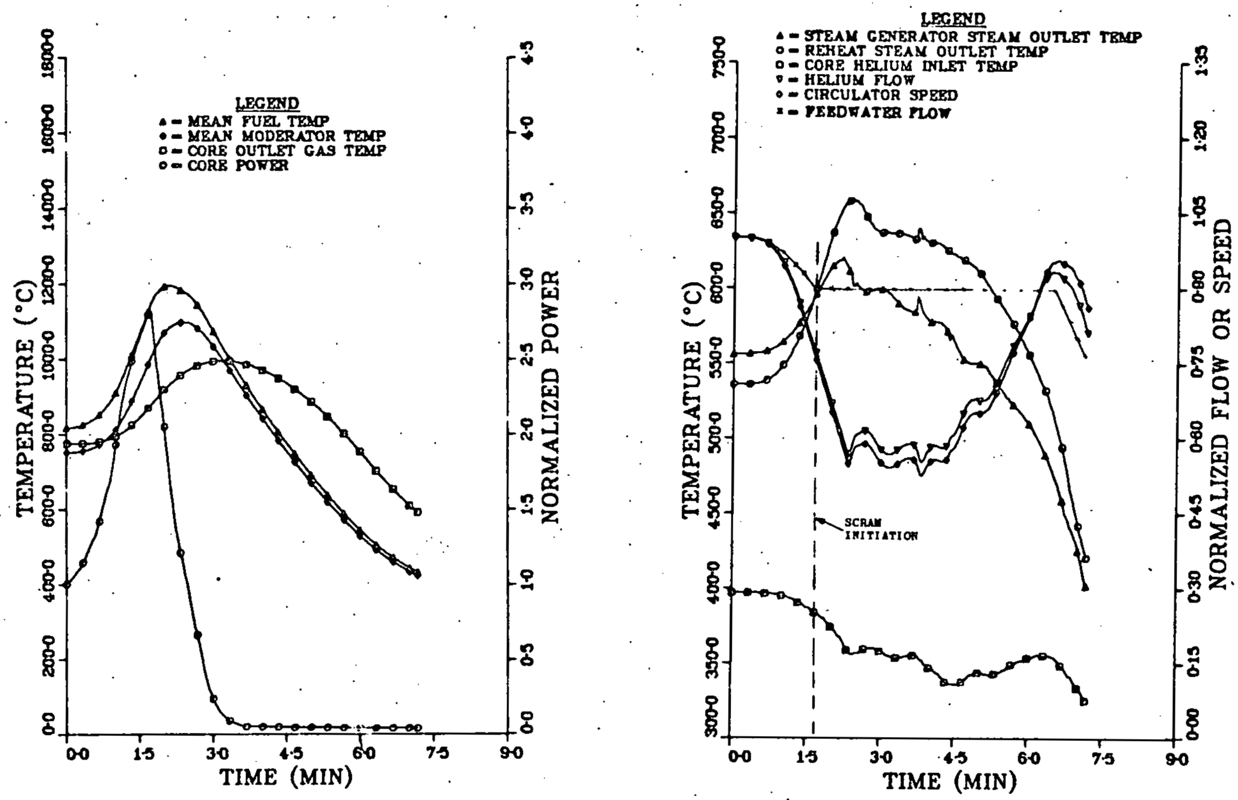
CONTROL ROD PAIR WITHDRAWAL ACCIDENT

Various types of postulated FSV accidents have been analyzed using ORTAP and its component subroutines. One study was made to determine what conditions and assumptions lead to the most severe predicted core temperatures for a control rod pair withdrawal accident.[8] A thorough analysis of this accident was also given in the FSV Final Safety Analysis Report[9] (FSAR) and backup documents, so the two sets of results could be compared in detail.

It had been determined in the FSAR that the rod pair withdrawal accident leads to reactivity insertions and insertion rates that are greater than those for any other credible occurrence. The plant defense mechanisms for limiting the accident consequences include operator and control system responses, plus automatic scrams both at 140% of rated power and when the measured reheat steam temperature exceeds its rating by 42°C.

Maximum fuel temperatures were calculated both for the 140% power scram and for the scram at high reheat steam temperature (RHST), the latter case assuming that the 140% power scram was inoperative. Other variations included use of nuclear parameters from various times in the core life cycles, and assumptions of different initial power levels, rod pair initial positions and worths, plant control system configurations, and fuel, gap, and moderator conductances. Very significant effects on calculated peak temperatures are due to assumptions of the power peaking factors (vs. time) that are applied to the refueling region experiencing the rod withdrawal. A followup study[10] on the peaking factor question indicated that application of less conservative assumptions mitigated the results significantly.

Some results of an example ORTAP run are shown in Fig. 5 for an RHST scram case, which assumed end of cycle equilibrium (EOC-EQ) core conditions and rod pair withdrawal from an initial fully inserted position. A reduction in helium flow is seen to occur prior to the reactor scram as a result of the main steam temperature controller's attempts to maintain the main steam temperature at its setpoint. This reduction in helium flow decreases the rate at which reheat steam temperature increases, and therefore delays the scram signal. The reduction in helium flow also causes the steam generator helium outlet temperature, and therefore the core inlet helium temperature, to drop because it results in a closer approach of the helium temperature to the near-constant feedwater temperature. Prior to scram, the flow through the high pressure turbine is controlled to meet a constant electrical demand. Because the enthalpy of the main steam is increasing, the high pressure turbine valve is throttled to reduce the steam flow, which results in an increase in the throttle pressure. Figure 5 also shows the reduction in feedwater flow prior to the reactor trip, which is due to the high pressure turbine throttle pressure controller action attempting to maintain the throttle pressure at its



(a) (b)

Fig. 5. Response of system parameters to a FSV rod-pair-withdrawal transient from an initially fully inserted position at end-of-equilibrium-cycle conditions; trip signal 42°C increase in measured reheat steam temperature.

setpoint. At the time of the scram, when the measured reheat steam temperature has increased by 42°C, the actual reheat steam temperature is 68°C above the trip point, the difference being due to the flow dependent measurement lag. Following the scram, the feedwater flow is held constant until the reheat steam temperature falls below 524°C. This interlock is to prevent additional increases in reheater and steam generator tube temperatures after the trip. After the reheat steam temperature falls below 524°C, the feedwater flow is ramped to 25% at the rate of 0.5% per sec.

A brief summary of the results of the rod pair withdrawal study is shown in Table I.

In the reference case, the nuclear parameters used are for an end-of-cycle equilibrium core, and the rod worth is 0.012 Δk. It should be noted that the calculations neglect the latent heat of fusion of UC_2 kernels at 2500°C by extrapolating a homogenized fuel stick heat capacity beyond 2500°C.

Table I. Sample Results of the FSV Rod-Pair Withdrawal Accident Study

	Reference Case RHST Scram	140% Power Scram	Rods Initially Half Inserted RHST Scram	Core Conductances Increased 50% RHST Scram
Time at scram initiation, sec	102.5 (105)	39.2	76.2	98.3
Power level at scram initiation, %	282	140	210	291
Max. core average fuel temp., °C	1195 (1225)	861 (870)	1098	1147
Max. mixed mean core outlet temp., °C	994 (1062)	804 (796)	927	993
Region experiencing withdrawal: Peak fuel centerline temp., °C	3057 (2870)	1137 (1183)	2082	2981
Peak region outlet temp., °C	1654 (1650)	862 (914)	1224	1604

NOTE: FSAR values, where available, are shown in parentheses

A major conclusion of the study was that several plant control and safety systems including the 140% power trip must be inoperative for fuel temperatures to exceed 1600°C, the temperature below which no fuel failure is expected.

DESIGN BASIS DEPRESSURIZATION ACCIDENT (DBDA)

In April 1978, ORNL was requested by NRC to provide independent calculations of both DBDA and loss of forced convection (LOFC) accidents to assist in evaluating a 100% power operating license application for FSV. Reactor operating parameters were supplied by GAC, and worst-case equilibrium core conditions were assumed. The reference case DBDA analysis, using ORECA, assumed a 5-min delay in the startup of the emergency cooling system, and then assumed the availability of only one loop (two of the four circulators).

The primary concern with the DBDA is with overheating the steel liners and ducting to the steam generators, rather than with fuel damage, since peak predicted fuel temperatures were well below 1600°C. Calculation of the steel liner temperatures is complicated primarily by the uncertainties in the estimates of "streaking factors", which relate the maximum gas temperatures impinging on the liners to the maximum refueling region gas exit temperatures. Using a conservative value of the streaking factor derived from GAC air model tests, the predictions indicated that the 1093°C damage limit would not be reached. Figure 6 shows some results of the reference case ORECA prediction.

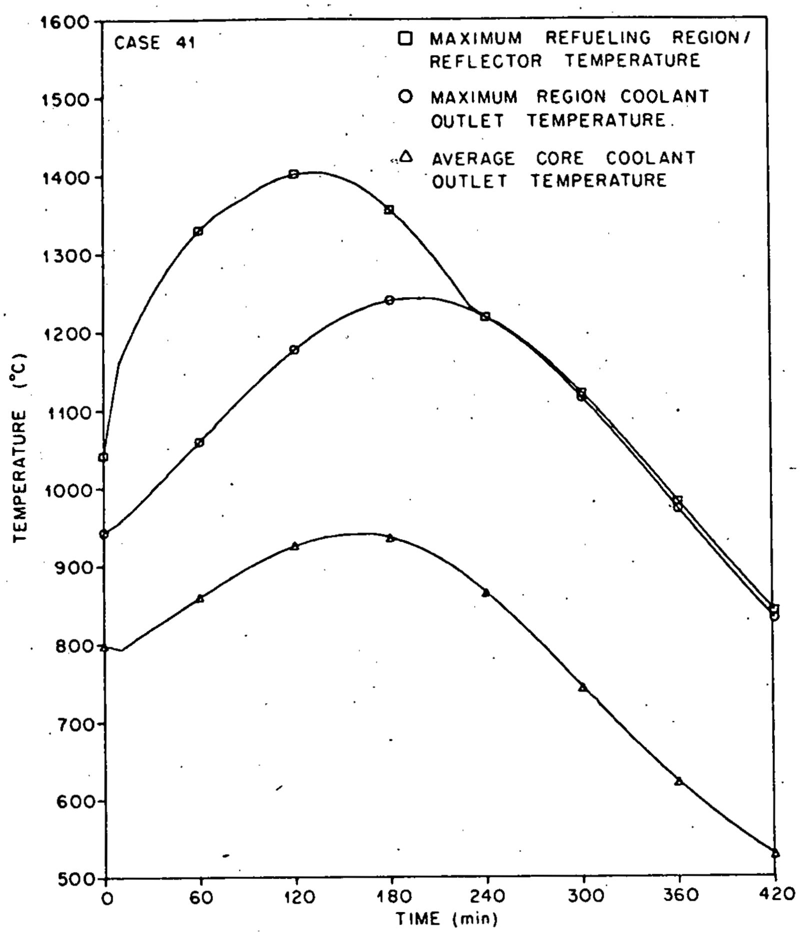


Fig. 6. Sample results of a postulated FSV reactor DBDA using the ORECA code.

Sensitivity studies were also run to determine the effects of various assumptions on peak temperatures, and no surprises were encountered. Table II, which gives comparisons of ORECA results with those generated by GAC (RECA3 code) indicate generally good agreement. It should be noted that FSAR afterheat equations were used in both analyses.

Table II. Results of FSV DBDA Study

Reference DBDA	GAC/RECA3	ORNL/ORECA
Peak Fuel Temperature, °C	~1427	1403
Maximum Average Core outlet temperature, °C	~ 927	940
Maximum Refueling region outlet temperature, °C	~1288	1243

LOSS OF FORCED CONVECTION (LOFC) - FIREWATER COOLDOWN (FWCD) ACCIDENTS

The LOFC accident calculations for the FSV 100% power license evaluation centered on the question of how much time the operators would actually need to start up the emergency cooling system following a postulated design basis earthquake. In this case, an LOFC is followed by use of the earthquake proof firewater system to provide both the motive force for the circulators' Pelton wheel drives and the cooling water for the steam generators. Calculations were done both for worst-case initial and equilibrium cores, the latter giving the higher peak temperatures.

The main concern during the LOFC period is the ability of the carbon steel upper plenum thermal barrier cover plates to withstand the heat from the hot plumes which emanate from refueling regions experiencing reverse flows. Calculations were done for postulated delays of up to 2 hrs. in initiation of the FWCD system. A major uncertainty in the model is the effective plume heat transfer coefficient (h-plume), and a detailed model of the plumes and cover plates[11] was added to the ORECA code. Depending on h-plume assumptions, the calculations indicated that some of the cover plates would be likely to exceed failure limit temperatures for extensive LOFC periods.

The major problem following initiation of the FWCD system is, like the DBDA, possible damage to the steam generator inlet ducts. As before, using the GAC-derived streaking factor, the damage limit was not exceeded for any of the cases analyzed.

Sample results of a LOFC/FWCD ORECA calculation are shown in Fig. 7, and comparisons with some GAC results for the case of no delay in initiation of FWCD (as in the FSAR) are shown in Table III.

Table III. Results of FSV LOFC/FWCD Study

Zero-Delay FWCD	GAC/RECA3	ORNL/ORECA
Equilibrium Core:		
Maximum Average Core outlet temperature, °C	~ 829	821
Maximum Refueling region outlet temperature, °C	~1038	1023
Initial Core:		
Maximum Average Core outlet temperature, °C	~ 816	804
Maximum Refueling region outlet temperature, °C	~1038	1038

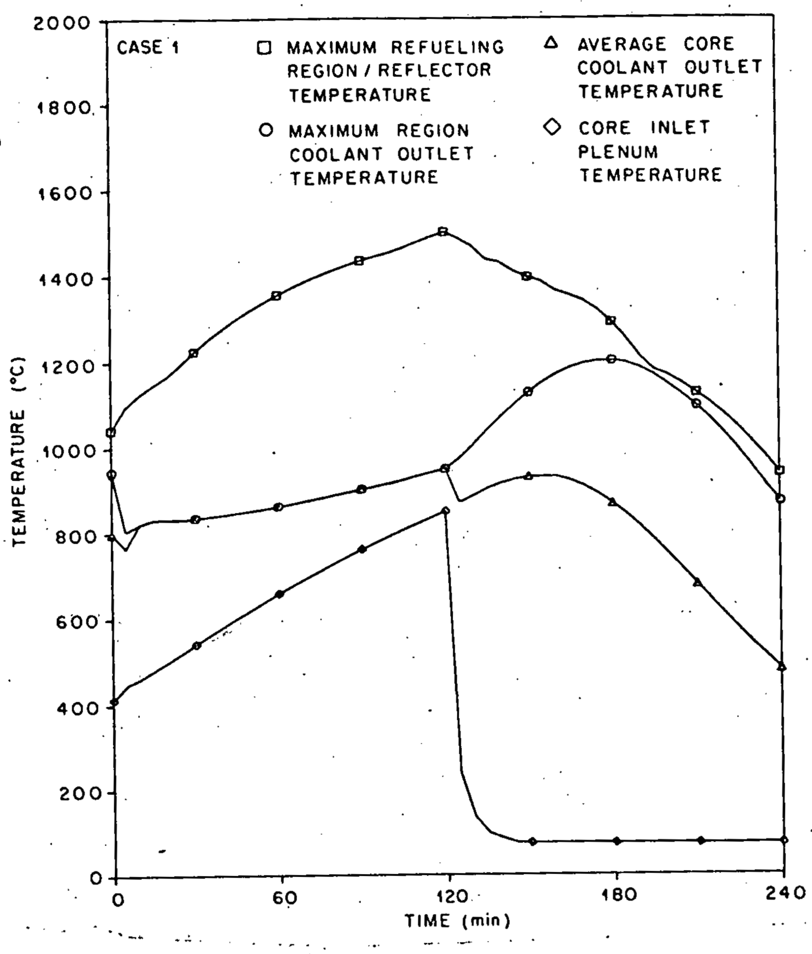


Fig. 7. Sample results of apostulated FSV reactor LOFC-FWCD accident using the ORECA code.

CONCLUSIONS AND FUTURE PLANS

The ORTAP code and its component subroutines are still under development, and are regularly being updated and improved. The results of verification studies are also being included and are resulting in a reduction of the initially assumed uncertainty ranges.

It is expected that the major near-term efforts will include model verification tests, code upgrading and documentation, and analyses in support of other FSV licensing questions.

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