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ORNL'S NRC-SPONSORED HTGR SAFETY AND LICENSING ANALYSIS ACTIVITIES FOR FORT ST. VRAIN AND ADVANCED REACTORS*

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ABSTRACT

The ORNL safety analysis program for the HTGR was established in 1974 to provide technical assistance to the USNRC on licensing questions for both Fort St. Vrain and advanced plant concepts. The emphasis has been on development of major component and system dynamic simulation codes, and use of these codes to analyze specific licensing-related scenarios. The program has also emphasized code verification, using Fort St. Vrain data where applicable, and comparing results with industry-generated codes. By the use of model and parameter adjustment routines, safety-significant uncertainties have been identified.

A major part of the analysis work has been done for the Fort St. Vrain HTGR, and has included analyses of FSAR accident scenario re-evaluations, the core block oscillation problem, core support thermal stress questions, technical specification upgrade review, and TMI action plan applicability studies. The large, 2240-MW(t) cogeneration lead plant design was analyzed in a multi-laboratory cooperative effort to estimate fission product source terms from postulated severe accidents.

1.0 INTRODUCTION

The safety research program for high-temperature gas-cooled reactors (HTGRs) at Oak Ridge National Laboratory (ORNL), sponsored by the U. S. Nuclear Regulatory Commission (NRC), has been directed mainly at the analysis of postulated HTGR accident sequences. A continuing effort has been on providing technical assistance to NRC on licensing-related questions for the Fort St. Vrain (FSV) HTGR located near Denver, Colorado. Other advanced HTGR concepts studied over the course of the program have been the GA Technologies (formerly General Atomic Co.) large commercial HTGRs that were planned in the 1970s, a more recent DOE-sponsored lead plant design adaptable to cogeneration rated at 2240-MW(t), and the current modular HTGR designs. In all of these cases, detailed dynamic simulations were developed to analyze accident scenarios and assist NRC in evaluating licensing-related problems.

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Another major effort in the program has been on computer code verification. This work has been approached from many angles: 1) internal and external reviews; 2) applications to a variety of problems bv many users; 3) sensitivity studies to show effects of model and parameter variations; 4) comparisons with other similar codes, and 5) comparisons with experimental data.

This paper describes the FSV simulation, analysis, and licensing support work, the 2240-MW(t) HTGR source term study, and code verification activities. The recent work on modular HTGRs is described in a companion paper.[1]

2. ORNL HTGR SYSTEM SIMULATION CODES

The code development work has been geared to the various HTGR designs as interest in the particular concepts has waxed and waned. One constant, however, has been the FSV HTGR, for which ORNL has been providing technical assistance to NRC over the entire course of the program. The overall system simulator developed for FSV is the ORTAP code.[2,3]

ORTAP contains coupled component simulations of the core (CORTAP,[4] ORECA[5]), the reheater and steam generator (BLAST[6]), the regenerative steam turbines (ORTURB[7]), the helium circulator and circulator turbine, and the balance-of-plant. The major plant control systems are also modeled.

The core is normally simulated by a single-channel thermal-hydraulics model incorporating coupled heat-transfer and neutron-point-kinetics equations (CORTAP) for at-power conditions. The other core model (ORECA) is used to simulate transients involving post trip power and flow conditions. The ORECA model includes three-dimensional temperature distribution calculations, accounts for the varying flow distribution among the individual refueling regions, and simulates flow reversals.

The reheater and steam generator are simulated by a multi-node, fixed-boundary, homogeneous-flow model (BLAST). Time-dependent equations for conservation of energy, mass, and momentum for the water/steam side, and for conservation of energy for the helium side and the tube are solved by an implicit integration technique. Transients involving both start up and flood-out of the steam generator can be simulated.

A detailed model of the regenerative steam turbines is necessary to accurately predict primary system component response because of the close coupling in FSV between the primary and secondary systems. The steam turbine model (ORTURB) calculates pressures, enthalpies, and flows at several points, including extraction and exhaust lines, in the high-, intermediate-, and low-pressure turbines. The dynamic response of each feedwater heater and the deaerator is specifically calculated. The detailed circulator-turbine model includes the turbine speed and pressure ratio controls. The main steam bypass system, desuperheater and flash tank are also modeled.

Though the present version of ORTAP is developed specifically for the FSV plant, changes in input and minor program modifications could adapt ORTAP to be used to simulate other HTGRs. The component routines can be either run independently or as a part of ORTAP. All of the code verification work has been done with the component routines being run independently.

3. FSV LICENSING STUDIES

Examples of several licensing-related FSV studies are presented. One unique example is the FSV oscillation problem. Unexpected oscillations were observed in the core outlet temperature, steam generator helium inlet (and steam outlet) temperature, and neutron detector measurements. At certain operating conditions, fluctuations in individual neutron channels would be as much as ± 5%, and helium temperature excursions from individual refueling regions and to steam generator modules would be as large as 100 and 50°C, respectively. Fluctuations were more likely to occur at higher powers and flows and high core flow resistance (which can be adjusted by the refueling-region orifices valves). They could be terminated by reducing the power and flow. The fluctuations had random spatial and temporal characteristics, although a dominant periodicity of ~10 min was observed in many instances.

The oscillations first occurred on Oct. 31, 1977. Over 100 h were spent in an oscillation mode at power levels ranging from 30 to 68%. Subsequent installation of region constraint devices (RCDs) to the top layer of plenum elements after the October 1979 outage was successful in stopping the oscillations for power levels up to 100%.

The ORNL involvement in the oscillation problem included 1) technical support during the initial stages of analyses, 2) assessment of related safety analyses and test program plans, 3) noise analyses of various core instrumentation signals, 4) review of the special in-core instrumentation [Instrumented Control Rod Drives (ICRDs)], and 5) safety assessments of the proposed fixes, including the RCDs. Our involvement continued through the post-RCD tests and the 70 to 100% power tests. The major analytical effort was an evaluation of the "jaws" theory, which postulates that periodic tilting of fuel element blocks near the top of the core opens alternative flow paths through the jaws so formed and that the resulting additional flow through a region's coolant channels could cause a substantial and rapid decrease in its outlet temperature.[8] The ORNL analyses corroborated the explanations advanced by the FSV utility.

A second licensing study addressed the possibilities of high core support block (CSB) thermal stresses during the earthquake accident sequence. This scenario includes a postulated loss of forced convection (LOFC) followed by a "firewater cooldown" (FWCD). (The FWCD refers to the use of emergency quake-proof fire protection system water to power the Pelton wheel drives on the helium circulators. It is postulated that FWCD can be implemented within 90 min of the onset of the LOFC.)

Results of ORECA code analyses of the postulated LOFC/FWCD scenarios were used by Los Alamos National Laboratory (LANL) to calculate thermal stresses in parts of the core-support structure. These stresses result from large temperature differences between adjacent refueling regions caused by mismatches in the heating and cooling of the regions caused by flow redistributions during the LOFC and FWCD phases of the accident. LANL calculations of maximum stresses in the CSBs indicated that the stresses were large enough to warrant some concern about possible crack formation and propagation in the support blocks. Several significant uncertainties, however, in both the thermal analyses and the stress analyses required refinements in both analyses. The outcome of the analyses concluded that the maximum predicted stresses would be somewhat less than the allowable limits; therefore, the existing core design and accident mitigation system were judged acceptable.[9]

Two other FSV technical assistance projects at ORNL have been sponsored by NRC's Office of Nuclear Reactor Regulation. The first was a review of the applicability of the NRC's TMI Action Plan requirements (primarily directed to water reactors) to FSV. The second is an ongoing review of the FSV Technical Specifications that ensure acceptable limits on core temperatures during low-power, low-flow operation. Special modifications were made to the ORECA code to predict intra-region flow redistributions (and stagnation) due to non-uniform radial heating ("tilts"). Recommendations for simplifying the tech specs have been made.

4. 2240-MW(t) HTGR STEAM CYCLE COGENERATION PLANT SOURCE TERM STUDIES

As part of a multi-laboratory collaborative study[10] of the fission product source terms from postulated severe accidents in the 2240-MW(t) HTGR, the ORECA code was modified extensively to model both the core and the available shutdown cooling mechanisms. In this version of ORECA, the core was represented by 14 axial nodes for each of the 85 active refueling regions and 24 side reflector regions (for a total of 1526 nodes). An improved time-at-temperature fuel failure model by GAT was also incorporated. Detailed models were developed for the upper (core inlet) plenum and lower (core outlet) plenum. In severe transients, radiative heat transfer to the liner cooling system (LCS) is significant, and the variations in the temperature between neighboring refueling regions can be large. Hence, a model is used which accounts for radiant heat exchange between individual refueling regions (upper and lower surfaces) and the coverplates (above or below) associated with individual regions. For example, in the upper plenum, each refueling and side reflector region's upper surface exchanges heat with the 109 upper plenum coverplates. Each coverplate is modeled dynamically; i.e., its heat capacity is included. Radiation to the side walls in both the upper and lower plenums is also modeled. The liner cooling system is modeled, and distributed models for the concrete include release of H_2O and CO_2 with concrete degradation. The code also allows for core auxiliary cooling system (CACS) operation.

The analyses done for the source term study included estimates of maximum time to restore cooling (MTRC). In an unrestricted core heatup accident (UCHA), the core heats up to a point beyond which attempts to cool it with the CACS would result in damage by the overheated coolant to ducting, support structures, cooling tubes, and circulators. The simulations showed the MTRC times to be 12-13 h, and that a satisfactory cooldown was possible (after depressurization) using only one of the three CACS loops.

Longer term (many days) UCHA scenarios were also studied to determine how long the operators would have to restore cooling to the LCS after a station blackout. Depending on various assumptions made, the results showed the PCRV remaining intact if liner cooling were restored before 40 to 60 h.

In support of the source term analyses both in this and the FSV severe accidents, ORNL has reviewed the status of fuel failure models and the information available on fission product release and transport. As a result, several key uncertainties were identified, and experiments have been run to expand the fission product transport properties data base.

5. CODE VERIFICATION STUDIES

Code verification is approached from several angles: 1) internal and external reviews; 2) application of codes by many users to a variety of problems; 3) sensitivity studies; 4) comparisons with other codes; and 5) comparisons with experimental data.

Comparisons of the ORECA core model code predictions were made with data from the FSV reactor scrams. Power levels just preceeding the scrams ranged from about 30 to 50%. The core was treated as an "isolated" component, so ORECA could be used independently of ORTAP for these comparisons. The time dependent input data included circulator inlet temperature, total primary system helium flow, primary pressure, and power (afterheat estimates). The initial conditions were estimates of each of the 37 refueling region power peaking factors, measured values of the region outlet temperatures, and the region orifice positions. With this input data, ORECA can compute the core transient response, including predictions for the measured gas outlet temperatures from all the regions, which are compared with data from the plant data logger.

The first comparisons were made for a scram test from 28% power on Aug. 6, 1977. Of particular interest in this test were the two 2-min periods of no-flow (at ~10 and 14 min after the scram) which caused significant delays in the core cooldown. The ORECA prediction of the measured central region outlet temperature was in excellent agreement with the data for the first 30 min and low thereafter, as shown in Fig. 1 Considering the complexity of the core thermal-hydraulics, the approximations used in ORECA, and the uncertainty ranges of both the model parameters and the measurements, comparisons of these and other initial "best-estimate" ORECA predictions with the data were quite good.

The other region outlet temperatures showed similar behavior, with slower responses in the lower-peaking-factor, lower-flow regions. The discrepancies with longer-term responses were also similar. In order to rationalize these discrepancies, sensitivity studies were made where many parameter and model variations were considered and tested. The sensitivity studies, along with the use of an automatic parameter optimization scheme, showed that modification of several models and parameters within reasonable uncertainty limits could affect the results enough to obtain excellent agreement with the measurements. The question then becomes which of the models and parameters should be varied and by how much.

The most significant differences between the initial, or "best estimate", predictions and the data are the long-term errors, where the predicted region gas outlet temperatures typically fall below the data ~20 to 30 min after the scram. Of all the different types of code "adjustments" tried, only two were found to reduce these errors to near zero: 1) the assumption of a greater-than-expected core bypass flow fraction; or 2) the assumption of a greater-than-expected response time for the region outlet temperature thermocouple assembly. An example of the first type of adjustment is seen in Fig. 2, which shows the same test and region response as in Fig. 1 but with an optimized ORECA calculation, the primary difference being the assumption of a relatively large (~19%) flow bypassing the core's main coolant paths. The resolution of this question is safety related in that the assumption of larger core bypass fractions leads to higher predicted core temperatures in postulated accident studies, while moderate increases in outlet thermocouple response times have no safety implications. Tests have been proposed to resolve this question.

Extensive verification work has also been done on the BLAST steam generator code, using data from both FSV and AVR.[11] In the latter case, the cooperative effort with FRG has been most beneficial. A rapid core outlet temperature change that occurred at FSV during an "oscillation transient" provided a good opportunity for BLAST code verification. One steam generator module experienced a 45°C drop in helium inlet temperature over a 5 min period, causing the outlet steam temperature to decrease about 67°C. The measured and computed steam temperature transients are shown in Fig. 3. Note that since the individual module feedwater flows are not recorded, initial offsets such as that shown are to be expected. The comparison shows that the size and shape of the responses agree. The measured increased steam temperature at the end of the transient is due to automatic throttling of the feedwater, not modeled in BLAST because module flows were not available.

A CORTAP core code validation excercise using FSV rod jog test data showed excellent agreement.[12] Results of ORTURB turbine plant code verification studies, including data from large ramp changes in power level (70 to 25%), appear in Refs. 13 and 14. A variety of FSV verification tests, including measurements of core flow redistribution and liner cooling system performance, have been proposed.

6. CONCLUSIONS

The ORNL HTGR safety research program for NRC has concentrated on building simulation capabilities for FSV and advanced HTGR designs. The current shift in emphasis is to modular pebble bed designs. Verification efforts have played, and will continue to play, a major role.

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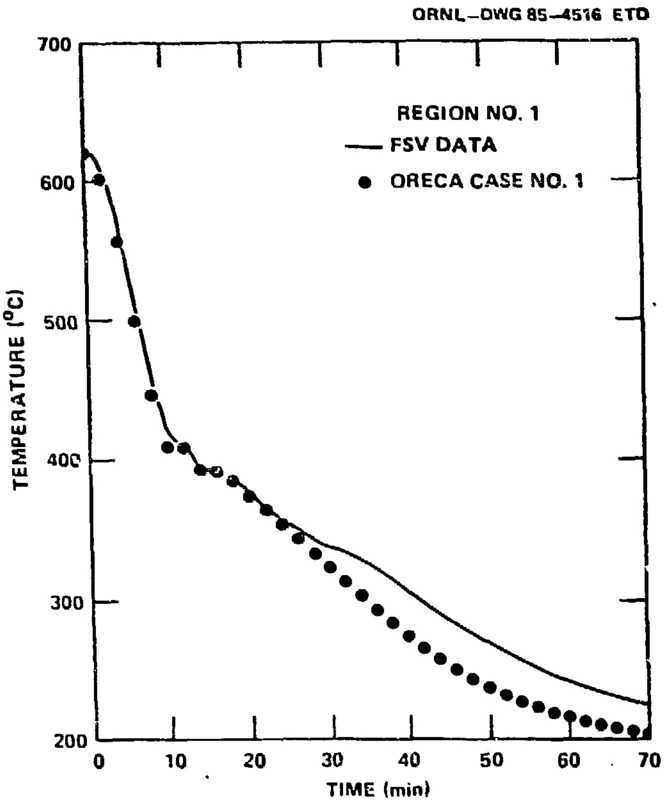

Fig. 1. FSV scram test of Aug. 6, 1977 (28% power) - comparison of reference case ORECA code predictions of measured gas outlet temperature from region 1 vs plant data.

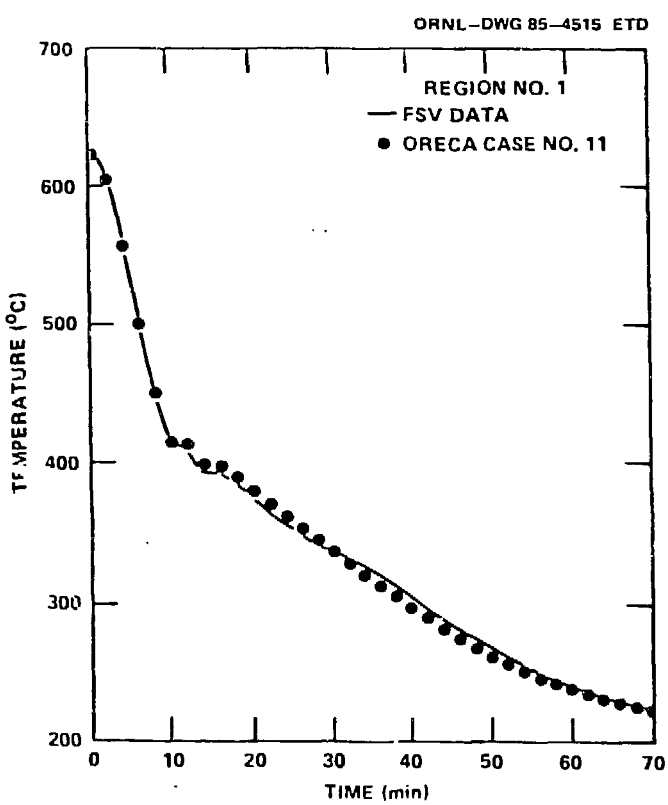

Fig. 2. FSV scram test of Aug. 6, 1977 (28% power) - comparison of optimized ORECA code predictions of measured gas outlet temperature from region 1 vs plant data.

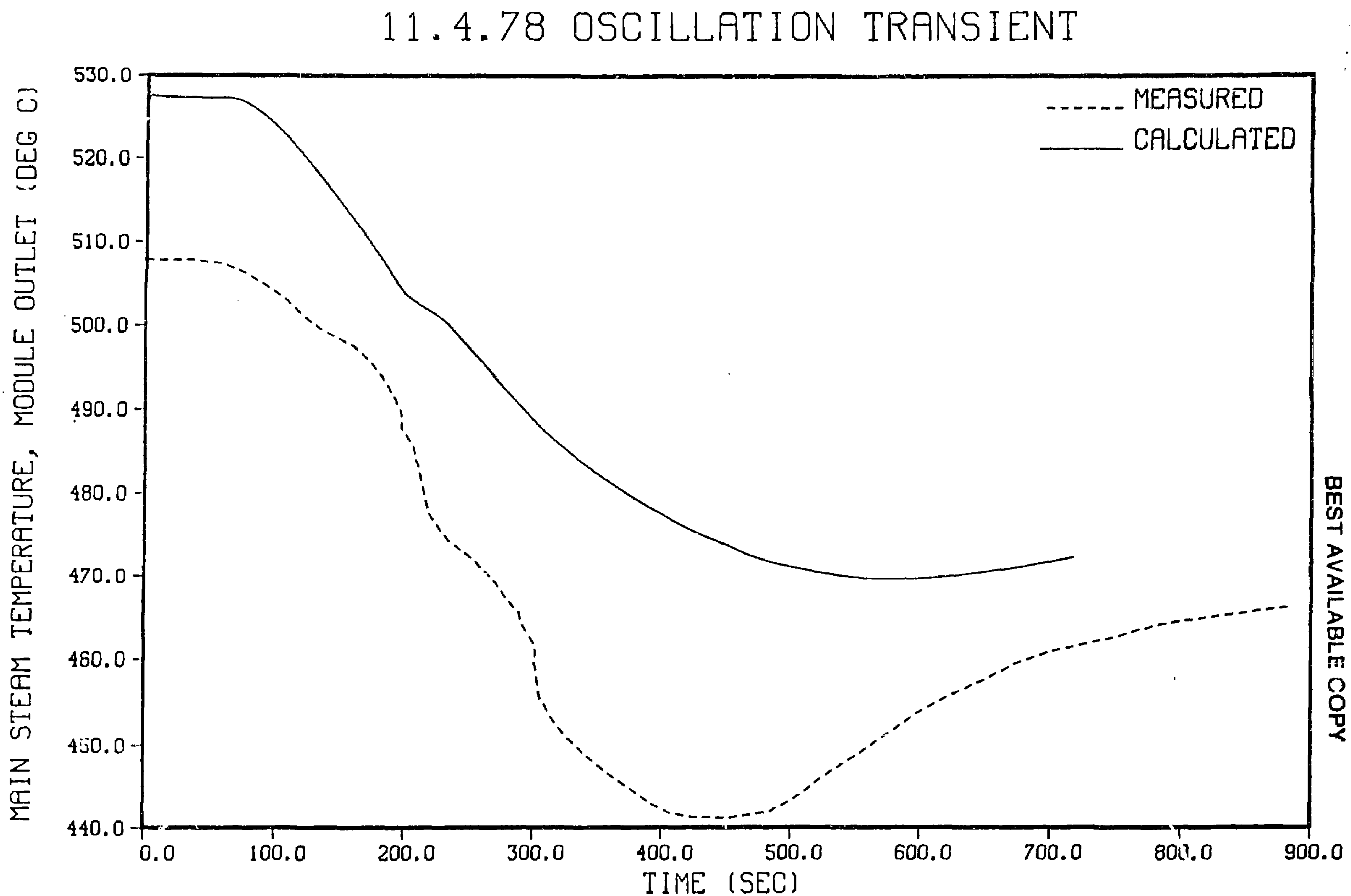

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Fig. 3. Comparison of BLAST prediction of FSV module outlet main steam temperature with measured data for the Nov. 4, 1978 oscillation transient.