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High-Temperature Gas-Cooled Reactor Safety Studies for the Division of Accident Evaluation Quarterly Progress Report, April 1–June 30, 1982

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FOREWORD

High-temperature gas-cooled reactor safety studies at Oak Ridge National Laboratory are sponsored by the Division of Accident Evaluation, (formerly the Division of Reactor Safety Research), which is part of the Office of Nuclear Regulatory Research of the Nuclear Regulatory Commission.

This report covers work performed from April 1–June 30, 1982. Previous quarterly reports and topical reports published to date are listed on pages v and vi. Copies of the reports are available from the Technical Information Center, U.S. Department of Energy, Oak Ridge, TN 37830.

HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION QUARTERLY PROGRESS REPORT, APRIL 1-JUNE 30, 1982

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ABSTRACT

Continuing work on High-Temperature Gas-Cooled Reactor (HTGR) severe accident analyses included a study of a hypothetical large-scale release following a permanent loss-of-coolant accident at the Fort St. Vrain reactor and further development of the CRECA code for siting studies of the 2240-MW(t) cogeneration plant HTGR. Work on fission-product release and transport included investigations of alternative iodine chemistry scenarios and an analysis of the major areas of uncertainties in release predictions during severe accidents. Code development work showed further progress in steam generator modeling, development of a multiloop HTGR simulation, and testing of an alternative simplified core model.

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1. HTGR SYSTEMS AND SAFETY ANALYSIS

S. J. Ball

Work for the Division of Accident Evaluation (formerly Reactor Safety Research) under the High-Temperature Gas-Cooled Reactor (HTGR) Systems and Safety Analysis Program began in July 1974, and progress is reported quarterly. Work during this quarter included development of the Oak Ridge National Laboratory (ORNL) HTGR Safety Codes, their applications to accident analyses, and reviews of fission-product release and transport methodology.

1.1 HTGR Safety Program Overview

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J. C. Conklin
R. M. Harrington
J. C. Cleveland
N. E. Clapp, Jr.

An overview paper on the program's progress and accomplishments over the past four years was written and presented at the Third Japan—

U.S. Seminar on HTGR Safety Technology held at Brookhaven National Laboratory (BNL) on June 2–3, 1982. The paper, "Safety and Licensing Analyses for the Fort St. Vrain HTGR," will be published in the proceedings of the meeting in the fall of 1982. The abstract is the following.

> The ORNL safety analysis program for the HTGR includes development and verification of system response simulation codes and applications of these codes to specific Fort St. Vrain reactor licensing problems. Licensing studies addressed the oscillation problems and the concerns about large thermal stresses in the core support blocks during a postulated accident. Other work includes proposed experiment planning, Three Mile Island (TMI) action plan applicability studies, and a new siting study on the 2240-MW(t) HTGR design.

1.2 HTGR Severe Accident Sequence Analysis

P. M. Harrington F. C. Kornegay

The analysis of a hypothetical large-scale release of noble gases and iodine occurring 24 h after a permanent loss-of-forced-convection (LOFC) accident was completed. These are very low probability, worst-case releases. For a permanent LOFC to happen, all four helium circulators would have to trip and not be restarted or repaired. For a subsequent large release of radioactivity to occur, the prestressed concrete reactor vessel (PCRV) pressure boundary would have to develop a major failure such as a failed penetration or a stuck-open safety valve. Two types of release were considered: (1) depressurization of the PCRV by leakage to the reactor building (with subsequent leakage from the reactor building to outdoor air) and (2) depressurization of the PCRV by opening of a PCRV relief valve (which discharges directly to the outdoor air). For each release, calculations were performed to predict the rate of release of radioactive primary coolant to outdoor air, dispersion of the released radioactivity throughout the vicinity of the plant site, and the resulting whole body and thyroid doses at fixed receptor locations. These calculations are described very thoroughly in a paper[1] presented at the Third Japan-U.S. Seminar on HTGR Safety Technology sponsored by the U.S. Nuclear Regulatory Commission (NRC) at BNL on June 2–3, 1982. Major conclusions discussed in the paper are as follows.

1. For both the safety valve release and the release via the reactor building, the calculated peak whole body doses are well below the lower protective action guideline (PAG). This is significant because, in both cases, almost the entire core inventory of noble gases was released to the environment. The small whole body doses can be explained by three factors: decay of short-lived noble gas isotopes, greater dilution caused by the elevated release points, and the additional dispersion caused by the gradual wind shift that was assumed to occur during the 4-h duration of the calculation. Simplified hand calculational models, such as those employed in reactor station radiological emergency response plans, would predict doses approximately a factor of 100 larger because they assume ground level release and because they cannot effectively account for changing wind direction.

2. The thyroid exposures calculated for the hypothetical Fort St. Vrain (FSV) releases are much more significant. For the release via the reactor building, the highest offsite thyroid doses are caused by the building leakage component of the iodine release; these doses exceed the lower PAG in the vicinity of the south side of the exclusion area boundary (EAB). Protective action would be required for individuals in the downwind quadrant near the EAB for this essentially nonelevated release of iodine. For the safety valve release, thyroid doses exceeding the lower PAG are predicted to about 6 km. Therefore, protective action would be required for populace within the affected quadrant to about 6 km.

3. The offsite doses calculated for these hypothetical FSV releases are not life threatening. The protective actions noted above would maintain health standards rather than save lives. A significantly greater fraction of the core radioactivity would have to be released to the environment to cause life-threatening offsite doses. This analysis was restricted to iodine and noble gas because an approximation of the maximum consequences of accidental releases during the first 24 h following a permanent LOFC was desired. The potential exists for more serious consequences than shown by this analysis because of the inventory of other hazardous but less volatile fission products in the fuel. According to results published[2] for a worst-case pressurized water reactor severe accident release, noble gases account for ~1% of the potential radiological health effects, and iodine accounts for about 10%. The remaining 90% of the potential radiological effects are caused by exposure to the cesium-rubidium, tellurium, barium-strontium, ruthenium, and lanthanum groups. These nuclides would be released in significant quantities in the latter stages (i.e., well after 24 h) of an LOFC accident. Current work at ORNL is directed toward quantifying the release from fuel and transport within the PCRV of all hazardous nuclides.

A model for calculation of heat transfer from the reactor core to the PCRV insulation in the core inlet plenum was completed, compiled, and executed with the ORECA program.[3] The objective of this modification is to enhance the capability of ORECA to calculate core and PCRV temperatures during extended LOFC accidents. Heat transfer from the core to PCRV becomes important in the latter stages of a permanent LOFC accident.

As a basis, the new model used the existing ORECA subroutines for inlet plenum heat transfer.[4] The existing programming for radiant heat transfer from the upper core surface to PCRV insulation cover plates and the existing program for convective heat transfer from reverse-flow plumes (which arise in hotter refueling regions in the LOFC accident) to the cover plates were used without modification. As in the existing program, the PCRV surface above the core was divided into 37 regions, each region equal in size and located directly above a refueling region.

Substantial modifications were made to the existing program to improve the conservation of energy:

1. A calculation of the temperature of the plenum elements atop each fuel region was added. The existing model simply set the plenum element temperature equal to the core exit temperatures for regions in reverse flow or equal to the mixed inlet plenum temperature for regions

in forward (i.e., downward) flow. Plenum element temperature is important to calculation of radiant heat losses from the core because the plenum element is actually the upper surface of the core.

2. Conductive and radiative heat losses from the core upper reflector were calculated and subtracted from the upper reflector heat balances. The existing model did not calculate or subtract these heat losses from the core heat balance. This effect was not considered in the existing model because radiative heat losses during the first several hours of an LOFC accident are negligible.

3. Convective heat loss from the inlet plenum helium was accounted for in the calculation of bulk (mixed) inlet plenum helium temperature. This heat loss was not subtracted in the existing model.

4. The calculation of plenum element surface and insulation cover plate temperatures was changed from an instantaneous algebraic calculation to a differential energy balance (solved by integration). This improves the conservation of energy by taking into account the nonnegligible internal energy of these components and has the effect of slowing somewhat the heatup of these components during an LOFC accident.

Selected results of ORECA calculation of the first 8 h of a hypothetical LOFC accident are shown on Figs. 1 and 2. The accident was initiated by coincident reactor trip and trip of all four helium circulators; the circulators were assumed not to restart, and a depressurization of the primary coolant began after 2 h. The average temperature of coolant in the core inlet plenum (Fig. 1), as calculated by the new model, was much lower than predicted by the existing model. This difference was caused by the modification described in item 3 whereby convective heat transfer from coolant to PCRV insulation was subtracted from the coolant energy balance.

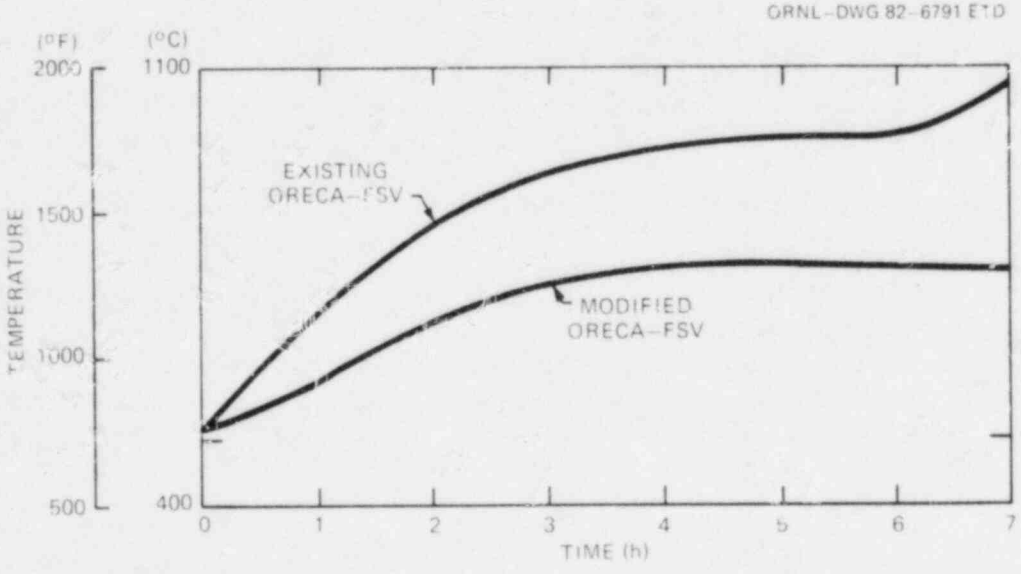

Fig. 1. Average gas temperature of core inlet plenum as calculated by existing and modified versions of ORECA-FSV.

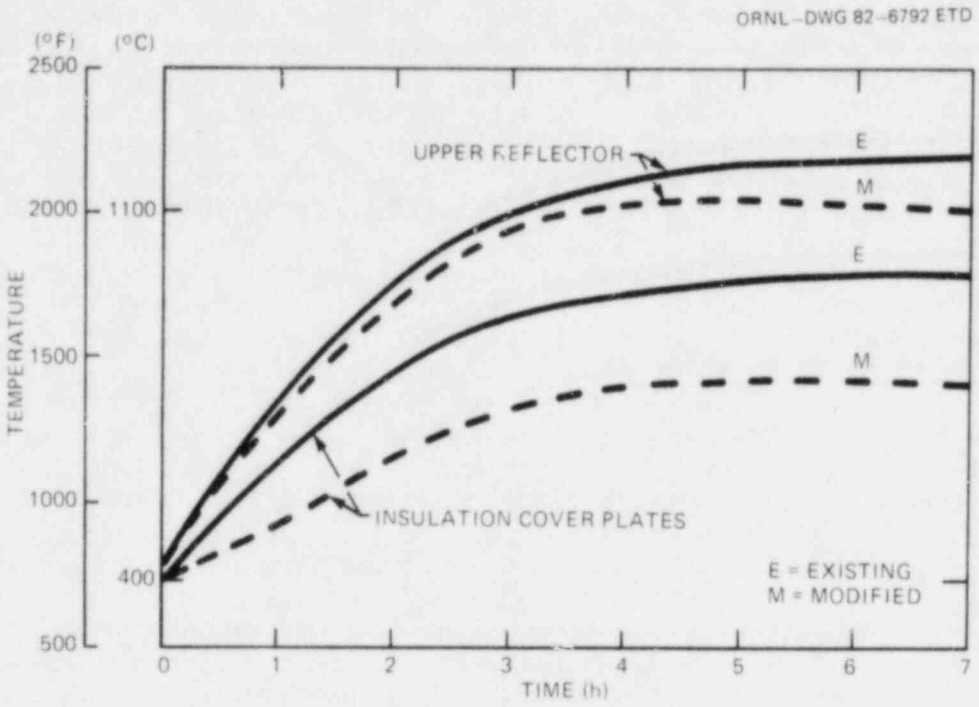

Fig. 2. Upper reflector and inlet plenum insulation cover plate temperature (refueling region 19) as calculated by existing and modified versions of ORECA-FSV.

Figure 2 shows the core upper reflector temperature for refueling region 19 and the PCRV insulation cover plate temperature at a location in the inlet plenum directly above refueling region 19. This region was selected because it had the highest power peaking factor (and therefore the highest decay heat level after scram). The upper reflector temperatures calculated by both the new and existing models were nearly identical during the first 2 h but were beginning to diverge after 8 h.

The new model subtracted the radiative and conductive heat losses from the upper reflector heat balance and thus was expected to predict lower temperatures. The region 19 PCRV insulation cover plate temperature predicted by the new model was significantly lower than that of the existing model, primarily because of the lower coolant temperatures that are now predicted (but also because of the insulative effect of the plenum elements, which were not included in the existing model).

1.3 Development of the ORECA Code for Simulating 2240-MW(t) SC/C HTGR Core Emergency Cooling Transients

S. J. Ball

Development work continued on adapting the ORECA code[3] used for simulating three-dimensional core thermal hydraulic dynamics to the new General Atomic Company (GA) design of a 2240-MW(t) steam cycle/cogeneration (SC/C) HTGR plant design. The code is to be used in the NRC-sponsored cooperative effort on siting studies being carried out jointly

with Idaho Nuclear Engineering Laboratory (INEL), BNL, Los Alamos National Laboratory (LANL), and ORNL. ORNL's role in the study was defined as contributing to the uncontrolled core heatup accident (UCHA) analysis. This is a highly unlikely, beyond design basis accident scenario, which assumes failure of the main cooling systems and all three independent core auxiliary cooling systems. Analyses of temporary or partial losses of the PCRV liner cooling system (LCS) are also of interest.

Code development work completed this quarter included conversion and debugging of the FSV version of ORECA for the 2240-MW(t) plant design. Modeling of the radiant heat transfer from the core to the adjacent LCSs was also completed and implemented. This included detailed models for both the upper (core inlet) and lower (core outlet) plenums, where radiation was accounted for between individual refueling regions (upper and lower surfaces) and the cover plates (above or below) associated with the individual regions. For example, in the upper plenum, each refueling region's upper surface exchanged heat with the 85 upper plenum cover plates. Each cover plate was modeled dynamically; that is, its heat capacity was included. Radiation to the side walls in both the upper and lower plenums was also important, so the modeling for radiant heat exchange between the refueling region surfaces and an "average temperature" plenum sidewall was included. In this case, rather than having each region-to-sidewall heat exchange modeled, a weighted average temperature for each ring of regions was used. This approximation was justified on the basis that some smearing of the exchange would be done by the control rod drive tubes (upper plenum) or core support posts (lower plenum). The effects of these obstructions are not otherwise considered. In addition to radiation, convection heat transfer between an average plenum gas temperature and the individual cover plates was also modeled. The calculation of the average upper and lower plenum helium temperatures was modified to include heat losses from the original FSV model.

Because the upper plenum cover plates are made of carbon steel, they are assumed to fall away and leave the liner areas behind them exposed if their temperatures exceed 816°C (1500°F). Because there is only a single-thickness cover plate in the 2240-MW(t) design (as opposed to the double plate design for FSV), all of the Kaowool insulation is assumed to disappear with the loss of the cover plate. This phenomenon was modeled for each of the 85 upper plenum cover plates. The lower plenum (floor) cover plates, which are made of ceramic and graphite materials (and which would have no place to fall), are assumed to stay intact regardless of their temperatures.

1.4 Simulation of an ATWS Transient for the 2240-MW(t) SC/C HTGR Core

J. C. Conklin

The nuclear core of the 2240-MW(t) reactor was modeled with the single-channel computer code CORTAP (Ref. 5). An anticipated transient without scram (ATWS) depressurization transient from full operating

power was simulated; that is, no control rod motion was assumed after initiating the transient. It was also assumed that circulator speed remained constant and that the steam generators provided a constant core inlet helium temperature. The negative temperature reactivity coefficient of the fuel reduced the core power to about 5% of full operating power (including the afterheat). The results are shown in Fig. 3. The calculated average fuel temperature reached a maximum of 1093°C (2000°F) at ~500 s.

The active core height was divided into six computational axial segments, the sixth being the lowest. The calculated fuel centerline temperature is plotted on Fig. 3 for the fifth and sixth segments. The fifth segment has the highest peak temperature initially, but the sixth segment has the highest peak temperature for the transient of 1316°C (2400°F) at 2000 s.

The initial portion of this hypothetical transient was also modeled with the simplified single-channel core computer code SCORE presently under development. The agreement with the CORTAP results was very good for the initial sequence of the transient, with a much lower computational cost.

The large heat capacity of the lower support block delayed the core outlet helium temperature peak of 1149°C (2100°F) to 6000 s after accident initiation. The helium core outlet temperature remained approximately at that high level.

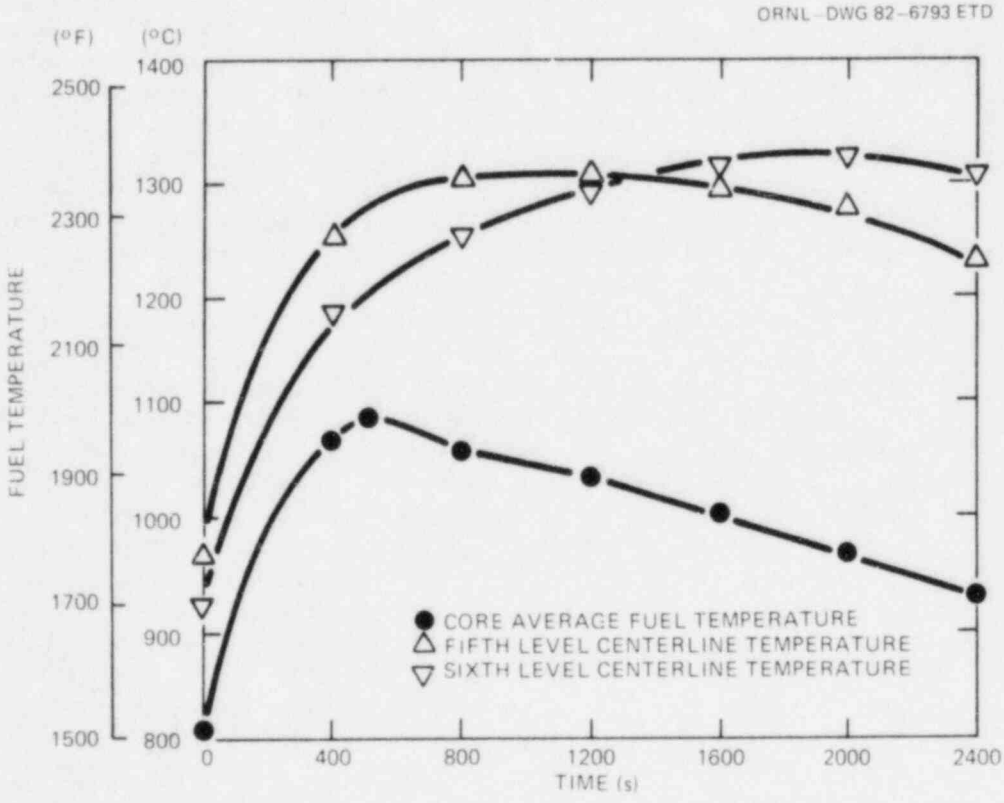

Fig. 3. Fuel temperature for ATWS depressurization.

1.5 Fission-Product Release from HTGRs

A. D. Kelmers

Work continued on the identification of problem areas in the prediction of fission-product releases from HTGRs during postulated severe accident sequences. A report was written that identified three areas where further research would be useful in determining and narrowing the uncertainties. These areas, fission-product release from the PCRV, aerosol production, and fission-product release from failed fuel, are summarized below. A report and proposal was also written on the questions relating to fission-product iodine behavior in severe accident scenarios.

1.5.1 Fission-product release from the PCRV

During an HTGR severe accident scenario, fission products must sequentially escape through three barriers [(1) failed fuel, (2) the graphite core and reflector, and (3) the PCRV] before they are available for release to the containment or the environment. Experimental measurements of release rates and release mechanisms have been studied in the past, primarily at GA and also at other laboratories. This information has been used to help establish codes for the calculation of fission-product release under various potential accident sequences and, thus, the resulting calculated exposure of the plant operators and/or public. In general, fission-product release information is best developed for failed fuel, less well for the graphite core and reflector, and least rigorously for the PCRV. Because the PCRV escape fraction value (fraction of the fission products not subject to plateout on PCRV internals) is a very significant or dominant value in the release calculations for many fission products, we suggest that it is important to improve the data base that supports the present PCRV escape fraction values. This would involve obtaining new experimental measurements that may more precisely reflect the situations to be encountered in various potential accident sequences. This will permit an improvement of the estimated fraction of fission products to be released during hypothetical accident sequences for FSV and/or a more accurate calculation of anticipated fission-product release during accident scenarios for future HTGRs.

Under either normal operating conditions or accident conditions, plateout or deposition of condensible or chemically reactive fission-product species on the structural surface of the primary circuit (PCRV internals) greatly reduces the fraction or quantity of these fission products calculated to be available for release to the environment by leakage. During postulated accident scenarios, plateout is a major mitigating factor. [For example, in the FSV Final Safety Analysis Report (FSAR),[6] 0.77 fraction of the strontium core inventory was calculated to be released from the core and top reflector during the design base accident (DBA); however, the PCRV escape fraction (the fraction released from the core and reflector that is not retained by plateout) for strontium was taken as 3×10^{-4}; thus only 2.3×10^{-4} of the core inventory of strontium was calculated to be available for leakage from

the PCRV and release to the environment. Likewise, although 0.98 of the iodine inventory was calculated to be released from the core and top reflector, this was similarly reduced to only 0.055 inventory fraction available for leakage.] In fact, for all fission products other than the noble gases krypton and xenon, the plateout values (or PCRV escape fraction values) were the dominant and controlling values in calculations of the fraction (or quantity) of fission products available for release to the environment.[6]

Therefore, in improving source term estimates, it would seem to be desirable to improve the plateout values used in the release computer programs. Two recent GA documents are pertinent. One reviews the history of plateout investigations at GA, and the other presents a new computer program for plateout calculations.

A new program, MULTI*PADLOC, for the calculation of plateout concentrations has recently been released by GA (Ref. 7). This is a multiple species plateout computer program that accommodates fission-product behavior such as precursor effects, isotopic exchange on surfaces, and chemical reactions on surfaces. It is an expanded multiple species version of the original PADLOC (Ref. 8), which was derived from the code PAD (Ref. 9). It is capable of analyzing coupled effects and solves problems of mass transport in terms of time and one-dimensional spatial dependence of the concentrations of fission-products in the carrier gas and on the surface for several impurity species, including the effects of sources in the gas and on the surface; convection along the flow paths; decay interaction; sorption interaction on the wall surfaces; and chemical reaction.

MULTI*PADLOC seems to be a substantial improvement over the previous programs. Because it is recent, it was not, of course, used in the Accident Initiation and Progression Analysis (AIPA) calculations.[10] For those calculations, plateout and purification rates were used from the PAD code to obtain total plateout activities, which, in turn, were combined with circulating activities in the PAD code to obtain plateout distributions.

Earlier, in the FSV-FSAR, the PCRV escape fractions during the DBA were obtained by multiplying the largest values for activity released from the test crucible in a series of King Furnace tests by the fraction of core inventory released, as calculated by the FREVAP code.

MULTI*PADLOC is a more sophisticated computer treatment and, if applied retroactively to the AIPA and FSV-FSAR DBA events, probably would lead to different calculated fission-product release values. Any computer program is constrained by the input values used from the data base, however, and discussion of a GA report that addresses this relative to plateout values follows.

In 1977, GA issued a review of their work on fission-product plateout.[11] The conclusion was that the data base that supports plateout calculations is not completely adequate, especially for accident conditions.

Improved estimates of PCRV release fractions will require experiments that measure the deposition or adsorption isotherms of the important fission products on the various structural materials under realistic accident-related conditions. The successful completion of such testing would seem to be desirable and necessary to make substantial reductions

in the plateout value uncertainties. There are many phenomena that would need to be considered in such tests. The fission products available for release by leakage from the PCRV following escape from failed fuel and the core and reflector will be retarded or reduced by adsorption phenomena on the PCRV internals. Dominant sorption phenomena are expected to be chemisorption and chemical reaction of the fission product with a PCRV component. This would be dependent on the PCRV component surface area as well as temperature and chemical nature of the component. An example of chemical reaction would be the reaction of iodine with the steel to form less volatile metal iodides. Tellurium is another chemically reactive fission-product that could undergo similar reactions. Chemical interaction of some fission products (such as cesium and iodine) in the temperature range to be expected in the PCRV could also lead to reduced mobilities through chemical reactions leading to less volatile compounds. Tests with mixed fission-products may also be necessary; these could be important because reactions among fission products or competition between them could alter the sorption behavior.

The results of such an experimental program could be expressed as adsorption isotherms or reduced apparent vapor pressures for the important fission products and mixture of fission products in the presence of PCRV components materials. The materials include steel and nickel-base alloy blanket hold-down plates, silica-alumina fibers in the blankets, the steel PCRV liner, and the concrete PCRV. These values would then be used to establish revised PCRV release fraction values under specific accident scenarios. The experiments would be conducted under temperature and atmosphere parameters that would be representative of probable accident sequence conditions. Oxygen would be included in some tests (as H_2O, O_2, and/or CO) to model steam or air ingress events.

The experimental parameters would involve high temperatures, controlled atmospheres, and, probably, the use of radioisotopes, representing a significant laboratory effort.

1.5.2 Aerosol production

Aerosol transport of less volatile fission products is considered as a major release mechanism for such fission products in light-water reactor (LWR) severe accident scenarios and may also be so for HTGRs. The possible formation of particulates was briefly mentioned in the 1977 GA review.[11] High temperatures (thousands of degrees Celsius) are postulated in the core during some LOFC accidents. These high temperatures offer potential mechanisms for aerosol generation, as well as the more generally recognized release of the gaseous fission-product inventory from the failed fuel. Two general types of aerosol formation mechanisms during HTGR accident scenarios appear possible. These are aerosol generated from the PCRV components and aerosol formed by condensation of volatile fission products. Under extreme high-temperature conditions in the PCRV, a portion of the steel liner and/or alumina-silica blanket and associated nickel-base alloy or steel alloy blanket hold-down plates could melt and undergo thermal decomposition. Some of these decomposition products could vaporize in the hottest regions and then condense to form dispersed particulates that could, in turn, sequester and transport the less volatile fission products. Sequestration

could result from either (or both) chemisorption and physical adsorption phenomena. Also, failure of the PCRV blankets and cover plates and the liner would expose the concrete PCRV. Particulates and gases generated by the thermal decomposition of exposed concrete could also generate aerosol particles that could adsorb and transport less volatile fission products.

For the case of aerosol formed by condensation of volatile fission-products at the high temperatures postulated in the core and reflector during LOFC accidents, many fission products would be present as gaseous elements, even in the presence of appreciable partial pressures of oxygen (as air, steam, or CO). For example, cesium, strontium, and iodine would probably be released from the failed fuel, core, and top reflector as elemental gases. Then, in cooler regions inside the PCRV, chemical equilibria such as

$$CsI \rightleftarrows Cs + I$$

or

$$SrO \rightleftarrows Sr + 1/2O_2$$

could be shifted to the left, and the elements could combine and condense, possibly forming aerosols. These, then, could offer an aerosol transport mechanism for the fission products that is independent of the carried fission products described in the previous case. Similar volatilization-as-elements condensation-as-compound scenarios likely could exist for other fission products such as silver, barium, lanthanum, and tellurium. Some of these have large hazard indices in LWR accident release calculations but are not usually considered as major HTGR release sources.

Experiments to evaluate these possible means of aerosol generation and resulting fission-product transport would be useful. Estimation of the significance of possible mechanisms leading to aerosol production and quantity and rate of production to be expected is important. Characterization of the chemical type and physical-chemical determination of the fission-product form (independent or adsorbed on the aerosol) should also be carried out. Tests could be conducted in flowing atmospheres under temperature gradients and oxygen partial pressures representative of various accident conditions. Both fission-product mixtures and individual fission products could be measured in the presence and absence of exogenous simulated PCRV-component-derived aerosols. The actual chemical form of the aerosol and/or the sorption mechanism would be established where possible.

1.5.3 Fission-product release from failed fuel

Considerable work has been done in the past (primarily at GA) both to measure and empirically model fission-product release from failed fuel. Since the time of the FSV-FSAR[6] and the AIPA[11] studies, the

models used for fission-product release from fuel have been upgraded.[12,13] These models could probably be relatively easily combined to yield a new improved computer code for time-temperature dependent release to some fission products from fuel during accident sequences. The new models are somewhat limited in scope in that much of the experimental work was restricted primarily to krypton, xenon, tellurium, and iodine (some attention has also been given to cesium and has considered only chemical regimes in the absence of oxygen (as H_2O, O_2, or CO). Some accident sequences involve the ingress of steam and/or air. In an oxidizing environment, the chemical form of some of the fission products (and thus the vapor pressure and mobility) would be changed. To model accident scenarios that involve oxygen ingress, additional information would be desirable on the chemical effects of oxygen (CO, H_2O, or O_2) on the fuel particle coating and the result of such chemical attack on the particle integrity. A modified form of the Goodin fuel failure model[12] may be needed to adequately account for the oxygen effect on fission-product release. Also, experimental work with additional fission products not covered in the earlier experimental work could be desirable to establish that the similitude modeling used was adequate or else to develop new release models for these fission products.

Alternatively, a modeling approach other than that developed at GA could be considered. For example, recent German work[14] presents different fuel failure mechanisms and fission-product release information. In their mechanisms, time at intermediate temperatures seems more significant in controlling fission-product release than in the GA model[12] for fuel failure. The German work indicated some release of metallic fission product from fuel particles prior to fuel particle failure, while the most recent GA model[13] is restricted to failed fuel particles. Also, the Germans indicate differences in behavior of oxide and carbide fuels, while GA saw no significant difference.[12] Additional work should be done to improve the understanding of the mechanisms and time-temperature dependent release of various fission products from whole or failed fuel and to resolve or combine the features of the GA and German models.

Possible experimental work would include measurement of fission-product vapor pressures as a function of temperature and oxygen partial pressure. Vapor pressure data as a function of temperature could be obtained under four atmospheric conditions to address different potential accident scenarios in: (1) a reducing environment (i.e., helium containing contaminants at typical trace levels), (2) helium containing H_2O levels characteristic of a steam ingress accident scenario, (3) helium containing N_2 and O_2 concentrations representative of air ingress conditions, and (4) helium containing CO concentrations representative of graphite oxidation events.

Emphasis should be directed primarily toward the more chemically reactive fission products such as iodine, tellurium, cesium, antimony, and ruthenium where chemical reactions as a function of oxygen partial pressure could be expected to change the chemical form, and thus the volatility, in the different atmospheres. Measurements could also be made in the presence of an excess of graphite to include graphite adsorption phenomena effects on the vapor pressure.

In the area of fission-product release, most of the reported experimental measurements and empirical models are based on krypton, xenon,

tellurium, and iodine data. Because the most recent GA model[13] is empirically derived, it is difficult to estimate how applicable it may be to other, chemically different fission products such as cesium, antimony, silver, strontium, and barium. Some of these fission products have large hazard indices in LWR release studies. To better understand this problem, careful review of the published information would be necessary. While relevant work has been done previously, both the fuel composition and coating have changed with time. After evaluation of this information, an experimental program would be required to fill voids in data for the development and verification of a release model(s). The experimental work probably would involve measurements of time-temperature dependent release rates for the fission products of interest from failed irradiated fuel particles.

1.5.4 Fission-product iodine chemistry

A proposal was also drafted and sent to NRC for comment on resolving crucial questions relating to the behavior of fission-product iodine in HTGR severe accident scenarios. Iodine is a chemically active element that can undergo various reactions under accident conditions. Two views of the chemical behavior of iodine in the primary coolant can be found in the literature. In the majority view in the past, it has generally been assumed that iodine diffuses from the failed fuel and graphite core and reflector as atomic iodine, which then reacts with the steel in the PCRV to form iron iodine. This view of iodine plateout is discussed first. Relevant reactions are summarized and discussed below, based primarily on information from an excellent review article by Hoinkis.[15] Later summary reports were also consulted.[16,17] Some additional reactions are also discussed for which insufficient data exist to establish their importance or which are less likely to be important under severe accident conditions.

An alternative minority view of iodine chemistry is described in a few recent reports.[18,19,20] Chemical thermodynamics show that essentially all the fission-product iodine in the fuel particle should exist as the stable compound cesium iodine, CsI.[18] After fuel failure in an accident scenario, if iodine is released from the fuel as CsI, its chemistry and behavior on diffusion through the graphite core and reflector and its plateout in the PCRV would be considerably different than that conventially assumed for iodine release and diffusion and iron iodide plateout. Thermodynamic data for iodine compounds in the primary coolant were compiled, but no conclusions relative to expected behavior were drawn.[19] The reaction of CsI with high-chromium steel alloys (not typically used in HTGRs) has been briefly investigated.[20]

It is important to resolve the apparent differences of these two views of iodine chemistry in the PCRV. The temperatures at which plateout would stabilize iodine and where revolatilization and availability for release could occur under accident scenarios appear to be quite different for FeI_2 and CsI. A careful review of the existing information would be desirable to see if sufficient data exist to resolve this question. If not, then an appropriate experimental program could be devised.

In developing a release code for iodine, it would be necessary to develop mathematical expressions for the relevant reactions and to supply the appropriate input of data base values. An examination of the recent GA computer program MULTI*PADLOC (Ref. 7) was proposed to see if it is capable of handling these reactions or if modifications to the program would be necessary, and to determine what data inputs are required. It would also be interesting to see if all the data base values and/or mathematic expressions exist that could be required to support the computer program, or, if not, what additional experiments or work would be necessary to develop these.

The iron iodine view of iodine chemistry is the conventional treatment and has been considered for many years. This chemistry underlies the philosophy supporting the iodine release calculations in FSV-FSAR, Delmarva PSAR, and Safety/Licensing Assessment of the 2240-MW(t) SC/C documents. Iodine is assumed to diffuse into the PCRV as atomic iodine and then plateout as metal iodides, primarily iron iodide.

Eight important reaction possibilities have been characterized, including molecular iodine dissociation, plateout of iodine vapor as an iron iodide (FeI_2) solid, sublimation and vapor transport of FeI_2, dissociation of FeI_2, atomic iodine adsorption, air oxidation of FeI_2, and formation of other metal iodides. The reaction of iodine or of iron iodide with other steel alloy constituents or with nickel-base alloy constituents could be important, because these are known to form more stable iodides than iron. Iodine plateout has apparently not been treated from this viewpoint, and insufficient information may exist to estimate what role these or similar reactions may play during accident scenarios.

Cesium-iodine chemistry is the relatively new minority view of iodine chemistry and has not been considered as extensively with respect to HTGR accident scenarios. Thus, data base values needed for calculations and for comparison with the previous view may be limited. Many fission-product elements accumulate in intact fuel particles as fuel burnup continues. At the fuel temperature during reactor operation (600—1000°C) and long-operating times (up to 3—4 years), most chemical reactions would be expected to go to completion and the most thermodynamically stable compounds would be expected to form from all the possible combinations of fuel, fission products, and fuel-coating elements. Of the many compounds that iodine could form, CsI is reported[18] to be the most stable. The fission yield generates a large stoichiometric excess of cesium compared with iodine, so adequate cesium should exist to essentially complete the formation of CsI. Thus, fission-product iodine in the intact fuel particle could be expected to accumulate as CsI.

A number of chemical reactions could become important if CsI is released from failed fuel in an accident scenario. These are briefly discussed below. Because the possible existence of CsI in the fuel is a relatively recent concept, this discussion is somewhat speculative.

$$CsI_{(s)} \rightarrow CsI_{(\ell)} \rightarrow CsI_{(g)} .$$

Chemical handbooks show the boiling point of CsI as 1280°C. Thus, accident conditions that generate higher temperatures would yield $CsI_{(g)}$, which might be expected to diffuse rapidly from failed fuel through the core and reflector and be released to the primary coolant in the PCRV. The melting point of CsI is 626°C; thus, in the temperature range 626–1280°C, liquid CsI could be present, which might diffuse fairly rapidly through the core and reflector. The temperature profile of the core and reflector thus likely would control the behavior and rate of release of CsI to the primary coolant in the PCRV, and knowledge of the temperature profile would be a necessary first step in understanding CsI behavior.

Evaluation of the actual situation in the fuel particle and in the graphite core and reflector is reported[18] to be furhter complicated by the possible formations of BaI_2, as well as CsI, and by the possible stability of cesium-carbon intercalation compounds (C_nCs), which could reduce the availability of cesium for CsI formation or stabilization. For the purposes of this report, however, CsI will be assumed to be the chemical form released from the top reflector into the primary coolant and into the PCRV in an accident.

A further potential complexity is the known prevalance of stable metal halide double salts. For example, the phase diagram for the system $CsI\text{-}TeI_4$ is dominated by a wide stability region for the compound Cs_2TeI_6.[21] The existence of such types of compounds has apparently not been previously evaluated relative to fission-produce plateout.

Reaction of cesium iodide with PCRV constituents is shown as

$$CsI + Fe,\ Ni,\ SiO_2\text{-}Al_2O_3,\ \text{etc.} \rightarrow ? \ .$$

The thermodynamic data indicate that CsI is more stable than FeI_2 and NiI_2; thus, CsI wuld not be expected to react with the steel or nickel-base alloy blanket hold-down plates or with the steel PCRV liner. Similar calculations have not been made for the silica-alumina blanket or for the concrete PCRV. CaI_2 might be formed by reactor with the cement, and thus, it is at least conceivable that the PCRV itself could become an iodine sink.

One brief report[20] suggests that CsI does not react with stainless steel and deposits on stainless steel from a flowing helium stream at 540–630°C as CsI. Such behavior is consistent with the calculated thermodynamic stability of CsI.

Cesium iodide plateout in the PCRV is shown as

$$CsI_{(g)} \xrightarrow[1280^\circ C]{} CsI_{(\ell)} \xrightarrow[626^\circ C]{} CsI_{(s)} \ .$$

This is simply the reverse of the first action. The information suggests that CsI discharged from the top reflector would plate out on the cooler regions of the PCRV and not be available for release to the environment. A more detailed investigation of this point may be desirable. In particular, the temperature profile of the PCRV under accident conditions will likely be an important factor. Vapor phase transport and deposition

of CsI would be dependent on the pressure as well as the temperature. In tests,[20] CsI volatilized at 800°C in a flowing helium atmosphere subsequently redeposited on quartz or stainless steel at 540–630°C. Thus, CsI might be expected to accumulate in regions of the PCRV that are in this temperature range.

Dissociation of CsI is shown as

$$CsI_{(g)} \rightarrow Cs_{(g)} + I_{(g)}.$$

This reaction probably would not become important at the temperatures in the PCRV. If this is correct, fission-product cesium iodine PCRV plateout would not become an iodine source on heatup during an accident. Additional work is likely needed to substantiate this hypothesis.

Iodine release during an air ingress accident is shown as

$$2CsI_{(\ell)} + Cr_{(alloy)} + 2O_{2(g)} \rightarrow Cs_2CrO_{4(s)} + I_{2(g)}.$$

The above reaction is reported[20] to proceed at 580°C on stainless steel. It is not known if a similar reaction will occur on the mild steel or nickel-base alloy materials used as the blanket hold-down plates and the liner in FSV or other HTGRs. The compound $CsFeO_2$ may offer a route for a similar reaction with mild steel via the reaction:

$$CsI_{(\ell)} + Fe_{(s)} + O_2 \rightarrow CsFeO_{2(s)} + I_{(g)}.$$

Thus in an air ingress accident, plateout CsI on metal surfaces might decompose rapidly and release elemental iodine gas to the primary coolant. Because air ingress implies a breach in the PCRV, this reaction offers a potential mechanism for a sudden release of fission-product iodine to the environment.

Iodine release during a steam ingress accident is shown as

$$CsI_{(\ell)} + Fe_{(s)} + H_2O \rightarrow ?? .$$

There may be insufficient information to describe this reaction under steam ingress accident conditions. If it proceeds, it could be a source of iodine for release, similar to the previous reaction.

1.6 Development of the BLAST Steam Generator Code

J. C. Conklin

Actual FSV plant operating data from the 100% power condition achieved on November 8, 1981, and as-built steam generator construction data are being used to develop modeling parameters for use in the BLAST dynamic simulation of the FSV steam generator/reheater.[22] The computational modeling was improved by including an increase in the water-side heat transfer coefficient for the superheat nodes because of the

tube helical curvature. This modeling improvement resulted in better agreement with the operating data for the steam and helium reheater outlet temperature when using the measured cold reheat attemperation water flow as input. However, the calculated main steam outlet temperature was too high and the module exit helium temperature was too low when compared with the data. This discrepancy is caused by at least two FSV specific design conditions that will require additions to the model: (1) cool helium from the circulators passes through the steam generator support shroud, which also contains the main steam downcomers; and (2) the module exit helium flow passes over the main steam downcomers and subheaders. These two considerations will bring the calculated main steam outlet and module helium outlet temperatures more in agreement with the plant data. Regenerative effects in the penetration also need to be modeled. Additional design data are being requested from Public Service Company of Colorado.

1.7 Development of Multiloop Capability for ORTAP

J. C. Cleveland

The Kernforschungsanlage (KFA) "multiloop" version of the BLAST steam generator simulation program has been implemented on the ORNL computer system. This version was developed at KFA from their improved version of the original ORNL stand-alone (single steam generator) BLAST code. This multiloop steam generator simulation is being used by KFA and Rheinisch-Westfalischer Technischer Uberwachungs e.V. (RWTUV) as a subprogram in their thorium high-temperature reactor (THTR) system simulation program to analyze off-normal and accident conditions as necessary in the licensing process for THTR.

Incorporation of this multiloop version of BLAST into ORNL's FSV system simulation (ORTAP) (Ref. 23) will allow multiloop steady-state and transient simulation of the two FSV primary loops with the capability of treating from 2 to all 12 of the steam generator modules separately. The ability to simulate only a single "average" steam generator module is also retained in the KFA multiloop BLAST version.

The KFA-developed "driver" subroutine for this multiloop BLAST is designed specifically for use with the various subroutines of the KFA (stand-alone) BLAST version, which have been installed and tested on ORNL's computer system during the past three to four months. This driver subroutine is not designed for use with the BLAST subroutines currently employed in ORTAP. Rather, it will be necessary to incorporate the KFA BLAST subroutines into ORTAP.

Several test cases on the KFA BLAST version were successfully run to exercise the various new features.

1.8 Development of a Simplified Core Model for ORTAP

N. E. Clapp, Jr.

Development continued on the simplified core model code SCORE, which is designed to be an optional replacement for the CORTAP code[5] in

the overall HTGR system code ORTAP (Ref. 23). SCORE uses LSODE (Livermore Solver for Ordinary Differential Equations) to perform the numerical integration. Because the system of equations contains both thermal and neutronic equations, LSODE's stiff option was used. The stiff option requires an additional subroutine, which contains the Jacobian of the set of equations used to define the system dynamic behavior. This subroutine was written and included in the code.

Two tests were selected to verify the performance of SCORE: a 9.5¢ reactivity increase with a ramp time of 6 s and a 6.3¢ reactivity decrease with a ramp time of 21 s. These tests were chosen because of the availability of experimental data. To simulate these conditions using SCORE, an additional equation was required. This equation defined the dynamic behavior of the control rods, approximated by a first-order lag with a time constant proportional to the ramp time of the experimental data.

The reactor power response for the 9.5¢ reactivity insertion is shown in Fig. 4. Both the experimental data and the results of SCORE are shown in this figure, and the comparison for this test is good

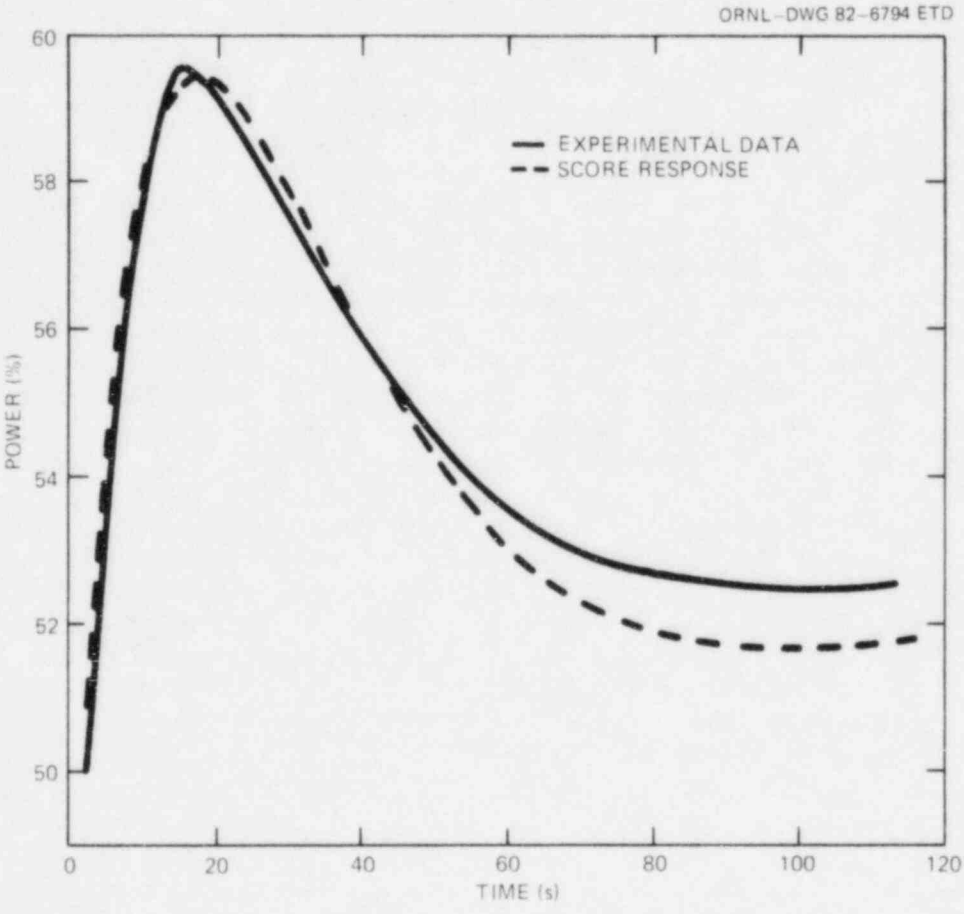

Fig. 4. Reactor power response for +9.5¢ reactivity insertion; ramp time = 6 s.

enough for control studies. The reactor power response for the -6.3¢ reactivity insertion is shown in Fig. 5. As can be seen, SCORE results correspond to the experimental data within a reasonable uncertainty band for control studies.

The simplified code SCORE will be evaluated by using different transient conditions and by verifying the results using CORTAP. The preliminary results indicate that SCORE runs about 100 times faster than CORTAP for the cases presented.

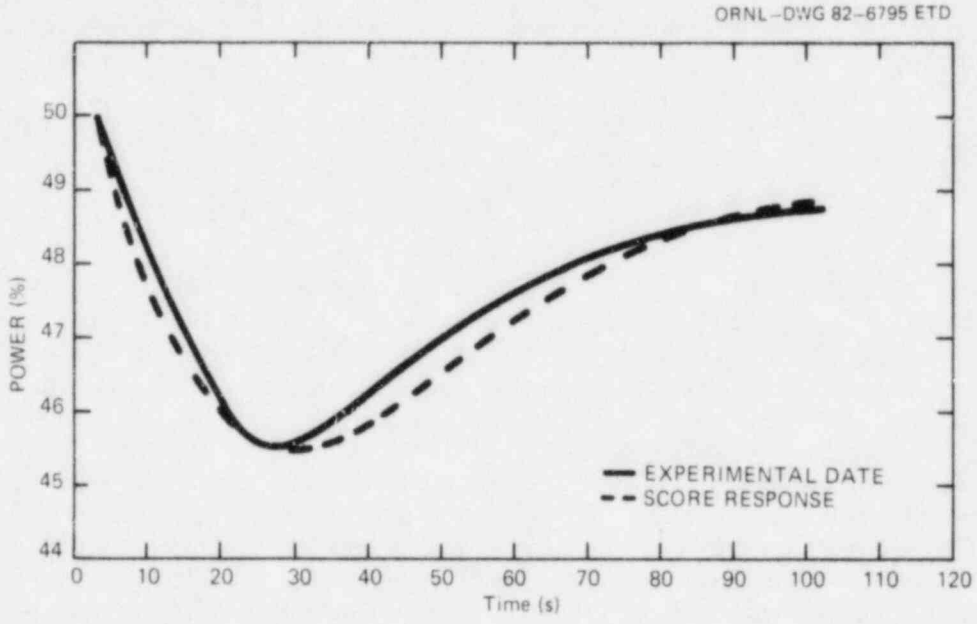

Fig. 5. Reactor power response for -6.3¢ reactivity insertion; ramp time = 21 s.

2. TRIPS MADE UNDER PROGRAM SPONSORSHIP

2.1 HTGR Siting Study Meeting at INEL, Idaho Falls, Idaho, May 5, 1982

S. J. Ball

The purpose of the meeting was to plan the cooperative effort on the siting studies of the 2240-MW(t) SC/C HTGR. Schedules and division of effort between participating laboratories (INEL, BNL, LANL, and ORNL) were agreed on and approved by NRC. ORNL's part included a commitment to develop the ORECA code to simulate postulated UCHA sequences for the 2240-MW(t) design HTGR in parallel with a similar effort at BNL.

2.2 Third Japan–U.S. Seminar on HTGR Safety Technology at BNL, Upton, N.Y., June 2-3, 1982

S. J. Ball R. M. Harrington A. D. Kelmers

The purpose of the meeting was to exchange current information on HTGR safety topics and on-going projects in Japan and the United States. Two papers written with program sponsorship were presented: "HTGR Severe Accident Sequence Analysis," and "Safety and Licensing Analyses for the Fort St. Vrain HTGR".

Items of major interest were (1) the Japanese shift from their previous primary HTGR interest in nuclear steelmaking to lower-temperature process heat applications; (2) construction that is to start on Japan's 30-MW(t) very high temperature gas-cooled reactor in 1986, with the design core outlet temperature lowered from 1000 to 950°C; (3) the considerable interest shown by most participating organizations in fission-product release, chemistry, and plateout research; and (4) Japan's HENDEL loop that is operational and has the potential for generating much interesting data.

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