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TITLE: HTGR SAFETY RESEARCH AT THE LOS ALAMOS NATIONAL LABORATORY

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HTGR safety research at the Los Alamos National Laboratory

This paper summarizes activities undertaken at the Los Alamos National Laboratory as part of the High-Temperature Gas-Cooled Reactor (HTGR) Safety Research Program sponsored by the US Nuclear Regulatory Commission. Technical accomplishments and analysis capabilities in six broad-based task areas are described. These tasks are: Fission-Product Technology, Primary-Coolant Impurities, Structural Investigations, Safety Instrumentation and Control Systems, Accident Delineation, and Phenomena Modeling and Systems Analysis.

INTRODUCTION

1. The Los Alamos National Laboratory has been conducting a broad program of research in High-Temperature Gas-Cooled Reactor (HTGR) safety technology since March 1974. This work has been performed under the auspices of the US Nuclear Regulatory Commission, Office of Nuclear Regulatory Research (NRC/RES). The program scope and level of effort have varied widely over the years. At the height of the program the Los Alamos effort included the following major task areas:

A. Fission-Product Technology,
B. Primary-Coolant Impurities,
C. Structural Investigations,
D. Safety Instrumentation and Control Systems,
E. Accident Delineation, and
F. Phenomena Modeling and Systems Analysis.

2. At the beginning of 1975, 28 Los Alamos staff members worked either full or part time on the HTGR Safety Research Program. In mid 1975, the General Atomic Company (GA) withdrew from the commercial market, which removed the immediate licensing pressure from the NRC. In the following years, our task schedules were modified, some tasks were terminated, and others were reduced in scope. The program cutbacks were designed to allow a continuing level of effort on those tasks considered most important by the Laboratory and by the NRC. Approximate levels of effort by year can be found in Table 1.

3. Some tasks were terminated before any significant new knowledge or capabilities could be developed; the sudden termination of other tasks prevented the complete documentation or analysis of preliminary results. However, many of the personnel originally assigned to the program are still with the Laboratory and much of the experimental equipment is still available.

4. This report is a summary of the work done at Los Alamos in the above task areas from March 1974 through mid 1982. During the period from March 1974 through September 1974, progress was reported in informal monthly letter reports. Task accomplishments since October 1974 have been reported in published quarterly progress reports. A bibliography of the more than 80 Los Alamos HTGR safety publications can be found in Ref. 1. The reader desiring additional information on topics discussed in this paper can obtain specific reports from the Los Alamos National Laboratory.

Table 1. Approximate task level of effort by year in full-time staff-member equivalents

Year	TaskA	TaskB	TaskC	TaskD	TaskE	TaskF	Total
1974	6		6			4	16
1975	6	5	6	3	1	4	25
1976	4	3	5	2	1	3	19
1977	3	2	4		1	4	14
1978	2		4		1	5	12
1979			3			1	4
1980			3			1	4
1981			3			2	5
1982	0.5		2.5			2	5

5. The first undertaking was the joint development by the Los Alamos National Laboratory, the Brookhaven National Laboratory (BNL), and the NRC of a comprehensive long-range plan for performing HTGR safety-related research. This unpublished plan specified those areas of HTGR technology where additional information was deemed necessary to assure that HTGRs could be designed, constructed, and operated safely. The plan covered both phenomenological research and safety-related analytical research (this plan recently has been updated in draft form by the NRC/RES).

6. Phenomenological research needs were divided into five areas: Fission-Product Release and Transport, Primary-Coolant Impurities, Rapid Graphite Oxidation, Structural Evaluation, and Safety Instrumentation and Control. Safety-related analytical research needs were divided

into three areas: Accident Delineation, Probabilistic Analysis, and Phenomena Modeling and Systems Analysis. Each of these research areas had one or more associated tasks for which the plan specified objectives, the existing state of the art, program scope, subtasks and schedules, and task relationships to other program elements. The six management task areas of the Los Alamos HTGR Safety Research Program touched on all eight areas of the long-range plan. The summary of tasks and accomplishments in the following sections discusses the relationship of the work done at Los Alamos to the tasks in the long-range plan. The summary also enumerates accomplishments, and discusses task completion status and lessons learned.

FISSION PRODUCT TECHNOLOGY

7. The Los Alamos task in Fission Product Technology included both experimental and theoretical studies of fission-product release, transport and deposition for HTGR normal operation and accident sequences. The research concentrated on the development of computer codes to calculate fission-product distribution and transport in the core structure, fission-product distribution in the coolant circuit, and fission-product transport in the containment building and in the atmosphere, and on an experimental investigation of coated fuel-particle failure and release rates. These areas covered all the subtopics of the original long-range plan for Fission-Product Release and Transport research.

8. The Los Alamos task began in late 1974 and was terminated at the end of 1976. The funding was sufficient to conduct several experiments and calculations, and to develop several useful computer codes.

Code development

9. Computer codes (AYERM, LARC, LARC-2) were developed to predict the time-dependent fuel failure and subsequent release of volatile fission products from an HTGR during core heatup events. The LEAF code was developed to calculate fission-product release from a reactor containment building. Multicomponent time-dependent in-core diffusion with radioactive decay could be analyzed by the DASH code. An additional code, named SUVIUS, was developed to calculate HTGR primary-coolant circulating activity and the plateout of volatile nuclides on cold surfaces.

10. The codes mentioned above can be obtained from the BNL HTGR Safety Code Library. They are modular and additional phenomena and component models can be included easily. These codes incorporate existing fuel models, temperature estimates, published filter efficiencies, and so forth, and allow the user to perform simple calculations to verify and understand the much more extensive calculations of the HTGR vendor.

Experimental studies

11. An experimental study of the failure statistics of irradiated, coated fuel particles was performed. Fuel-particle failure was induced by thermal ramping from room temperature to high temperatures and detected by measuring ^{85}Kr release. The data suggested that the TRISO particles failed instantaneously and completely at high temperatures. However, fission-product release from failed BISO particles appeared to be diffusion-controlled beginning at lower temperatures. A simple method for calculating the time-dependent fuel-particle survival fraction was developed with appropriate empirical constants.

12. The question was examined of whether metals could be transported in the vapor phase by unstable compounds such as metal carbonyls. The main conclusions were that a metal carbonyl could be formed within the fuel particle but that transport out of the fuel particles was unlikely.

13. In 1982, a task was begun for the NRC Office of Nuclear Reactor Regulation (NRC/NRR) to evaluate Fort St. Vrain (FSV) Nuclear Generating Station plateout probe data. A task to examine the most recent NRC fuel-failure model (NUREG-0111) in light of current data from FSV and other experiments is being considered for 1983.

PRIMARY COOLANT IMPURITIES

14. The Los Alamos research in Primary Coolant Impurities was divided between code development and study of impurity reactions with graphite and metallic components. The long-range objective was to obtain an understanding of chemical reactions between primary-coolant impurities and the various materials in the primary-coolant circuit. The impurities considered were steam; air; the products of reactions of these impurities with graphite such as hydrogen, carbon monoxide, carbon dioxide, and methane; and fission products. The program concentrated on the development of a computer code for the analysis of chemical reactions between graphite and flowing gases, experimental studies of reactions between impure helium and graphite or alloy structural materials, and development of codes for the calculation of complex chemical kinetics and equilibria. The task began in 1975 and was terminated in 1977.

Code development

15. The most significant accomplishment in the code development area was the development of the QUIL and QUIC codes. QUIL calculates complex chemical thermodynamic equilibria; QUIC, a rate code, calculates the time dependency of chemical reactions. The appropriate rate equations and rate constants for a limited number of HTGR primary-coolant impurity reactions were programmed into QUIC. The temperature at which a reaction occurs could be varied with time. A cyclic temperature mode also could be used to simulate the temperature variations of the primary coolant as it flowed around the HTGR primary loop. Code thermodynamic data for selected gas impurities in the primary coolant of an HTGR were collected from the literature and were supplemented by calculations.

16. These codes (available from the BNL HTGR Safety Code Library) can analyze many HTGR chemistry problems and could be applied to recent questions regarding the chemical behavior of radioiodine in HTGR accidents. Because the rate equations and rate constants are not known for many of the chemical reactions expected to occur in the primary loop of the HTGR, the equilibrium code QUIL must be used for much of the analysis. QUIC and QUIL operate symbiotically. QUIC is used for those chemical reactions for which the rate equations and rate constants are known and QUIL is used for instances where the rate equations or the rate constants are unknown.

Coolant impurity reactions with structures

17. Literature surveys and small-scale experiments were used to obtain information on reactions between coolant impurities and structural materials. Some analyses also were performed. Major emphasis was placed on the reactions between fission products and structural alloys. Chemical reactions can produce volatile compounds and result in the selective vapor transport of alloying elements. They also may form condensed phases with resultant penetration and diffusion of fission products through the alloys. Carbon/water equilibria in HTGRs also were analyzed.

STRUCTURAL INVESTIGATIONS

18. The Los Alamos Structural Investigations research concentrated on core seismic behavior, prestressed concrete reactor vessel (PCRV) analysis, and graphite component stress analysis. A fourth area evolved in the later stages of the program and applied techniques that had been developed previously to structural safety questions arising at FSV. Although the original program plan formulated by Los Alamos, BNL, and the NRC covered many facets of the structural evaluation of specific HTGR components for design adequacy and margin of safety, the program evolved over the years to a consideration of generic issues (for example, validity of scaling laws for seismic core behavior and development of a code for structural analysis of arbitrary concrete containments) or to investigation of operational and safety problems. This ongoing task began in late 1974.

19. The Structural Investigations task at Los Alamos has addressed several important generic safety issues. Significant analysis capabilities have been developed and several important results have been obtained.

Core seismic behavior

20. Scaling laws for the seismic behavior of an HTGR block core were derived and verified by a series of seismic experiments of increasing complexity using the White Sands Missile Range Seismic Simulator. In particular, the GA seismic test of a 1/5-scale graphite core model was shown to be conservative (that is, forces measured during the seismic testing of the model were greater than the forces that would exist in a full-scale core with the same peak input seismic acceleration.)

21. Another effort was an investigation of the torsional seismic behavior of the core using an analytical model to describe its three-dimensional seismic response. Because this was an extremely large and complex problem with only a modest chance of solution even with large computers, the task was terminated when it appeared that the program was destined for only constant or perhaps reduced funding.

22. As part of the HTGR seismic program, a Los Alamos staff member was assigned during 1979 to the Japanese Atomic Energy Research Institute (JAERI) at Tokai-Mura in Japan to participate in and observe Japanese seismic tests on Very-High-Temperature Reactor (VHTR) core models. This interaction proved to be extremely useful in providing the NRC and the Los Alamos National Laboratory with information on the Japanese testing program.

PCRV analysis

23. A finite-element code (NONSAP-C) for the structural analysis of PCRVs under static, dynamic, and long-term loads (where creep is the main issue) was developed, verified on test problems, documented, and distributed. Extensive verification problems have been run to check NONSAP-C in all of its operating modes (static, dynamic, and long-term creep).

24. For future problems related to PCRV structural behavior, the NONSAP-C code will provide a state-of-the-art capability that cannot be supplied by other structural analysis codes. Although other finite-element codes (for example, ADINA and MARC) have been applied to the analysis of concrete structures, the NONSAP-C code is unique in that it contains a variety of constitutive laws and an extensive element library. No other US code possesses a three-dimensional, aging-viscoelastic-creep constitutive model. The multinode tendon element in NONSAP-C's element library also is unique. The membrane element differs from the usual two-dimensional element in that its stiffness matrix is derived directly from the equations of elasticity in general curvilinear coordinates.

25. Finally, a reliability analysis procedure has been developed for PCRVs that uses large computer codes for strength analysis.

Graphite component stress analyses

26. Stress analyses have been performed for HTGR core-support columns under various static loading conditions. Cracked core-support posts were examined both analytically (that is, using fracture mechanics theory) and using a finite-element model. For shallow cracks, both methods gave approximately the same results.

27. A program is under way to develop design methods and failure criteria for graphite structures that take secondary (thermal) stresses into account.

FSV technical assistance

28. Procedures developed in the HTGR Safety Research Program have been applied to structural safety questions arising at the FSV station. These investigations have been wholly or partially supported by NRC/NRR.

29. During 1979 the mechanical aspects of the FSV fluctuation problem were investigated. Specific topics included a determination of the energy involved in core-block collisions and the maximum forces that could be transmitted to the massive PCRV from colliding blocks and columns. Instrumentation was specified that could identify the causes and consequences of the fluctuation. Also, a study was made of the effect of the region-constraint devices on the seismic behavior of the core.

30. During 1980 most of the FSV technical assistance was directed at the investigation of the thermal stresses developed in the core-support blocks during the firewater cooldown accident. In 1981, the major focus was the investigation of thermal anomalies at the liner-core interface and their effect on PCRV creep and shrinkage.

SAFETY INSTRUMENTATION AND CONTROL SYSTEMS

31. The Los Alamos Safety Instrumentation and Control Systems research concentrated on the evaluation and development of temperature sensors and coolant impurity monitors. The long-range objective was to evaluate and develop the instruments, techniques, and control systems necessary to assure the safety of HTGR operation during normal, upset, and accident conditions. In addition to the Los Alamos task areas, the long-range plan called for research on failed fuel detection and location, post-accident monitoring systems development, noise analysis and dynamic testing, and data acquisition and processing systems development. It also called for research into plant control systems, human factors engineering, and plant simulators. The task began in 1975 and was terminated in 1976.

Temperature sensors

32. The effects of the thermal, mechanical, and radiation environment on various thermocouple junctions, leads, insulation, and sheaths were examined. Thermocouple irradiation data were reviewed to characterize the potential for dimensional changes and decalibration. Measurement errors caused by gamma heating of the thermocouple were shown to be insignificant. When the task was phased out, it was not clear that Chromel-Alumel thermocouples could meet the operating goal of 10 000 h at temperatures to 1 023 K and neutron fluences to 8 x 10^{21} nvt (E>0.18 MeV) and 8 x 10^{21} nvt (thermal).

33. A preliminary investigation into the use of kryptonates for maximum temperature determination was made. A kryptonate is formed when radioactive ^{85}Kr is diffused into graphites or solids and is stabilized. These solids then are subjected to a thermal environment. Following this exposure, the kryptonate is thermally annealed and krypton release or retention is measured as a function of time and annealing temperature, allowing the inference of the maximum temperature to which the solid had been exposed. When this task was terminated, the suitability of using kryptonates as passive, maximum- temperature recorders in HTGRs had not yet been determined.

Coolant impurity monitors

34. A coolant impurity monitor based on the emissions of excited molecules produced in energy-transfer collisions between impurities and metastable helium atoms was designed, built, and tested at FSV. The Afterglow Helium Monitor measured concentrations of water, carbon monoxide, carbon dioxide, nitrogen, oxygen, and hydrogen. Tests at FSV showed the device could determine impurity concentrations in the low-ppm range. The lower detection limit was ~0.1 ppm for carbon dioxide and hydrogen; ~1.0 ppm for methane, nitrogen, and carbon monoxide; and ~10 ppm for oxygen.

35. The possibility of using the $^{16}O(n,p)^{16}N$ reaction to detect moisture in HTGR coolant was examined. The isotope ^{16}N decays with a 7.1-s half life, producing a dominant 6.13-MeV gamma. It appeared feasible to use this technique to detect rapid changes in moisture concentration, with a lower detection limit of ~2 ppm.

36. Although the Los Alamos effort was terminated, the completed work demonstrated that the Laboratory can design and fabricate innovative instruments.

ACCIDENT DELINEATION

37. The Los Alamos effort on Accident Delineation began as a cataloging and classifying exercise but eventually produced an independent probabilistic risk assessment of a large HTGR. The objective described in the long-range plan was to justify and improve the methods used for identifying accident sequences. A formal justification of the completeness of the sequence set produced was deemed necessary.

38. One staff member was assigned to this task from early 1975 through mid 1978. This task built upon the methodology of the Reactor Safety Study (WASH-1400). Consequence models were used to delineate those parameters that are important in the calculation of accident consequences. Event trees (unique accident sequences) were determined by identifying those consequence-model parameters that could be described by probability density functions. Fault-tree methodology was used to trace event sequences back to possible initiating conditions.

39. One innovation in the Los Alamos study allowed not only a branch for either complete success or complete failure but also allowed a branch for partial failure. A significance test using latent-hazard indices was developed. These latent-hazard indices quantify the relative potential of the various radionuclides for producing latent fatalities in the exposed population. This parameter was designed to establish the relative importance of possible accidents and accident sequences and to provide a basis for selecting the most important

sequences for detailed analysis. A report describing the detailed methodology was published that discussed application of the methods to a reference-design large HTGR and the differences between and similarities to the vendor-produced Accident Initiation and Progression Analysis (AIPA) study.

40. The Los Alamos study used the data base developed in the Reactor Safety Study. The AIPA study used a different data base, leading to different branch probabilities. The Los Alamos study considered event trees leading to core-heatup accidents, whereas the AIPA study considered specific initiating conditions. This difference in approach resulted in event sequences with dissimilar branches. Consequently, point estimates of the branch probabilities are not comparable.

PHENOMENA MODELING AND SYSTEMS ANALYSIS

41. The Los Alamos Phenomena Modeling and Systems Analysis research concentrated on HTGR neutronic analysis and systems code development. The original long-range objective was to provide basic data on the physical, chemical, and nuclear processes involved in accident sequences. Over the years the program evolved toward the development of a generic HTGR whole-plant systems analysis code and the application of that code to the simulation of the FSV station. This ongoing task began in 1974.

Neutronic analysis

42. A large effort in the early part of the program involved the calculation of the safety-related neutronic characteristics of the Fulton Generating Station (FGS) HTGR. These calculations were performed using codes and cross-section libraries that were developed independent of the HTGR vendor, whenever possible.

43. Extensive calculations were performed to determine the safety-related neutronic characteristics of the FGS initial core. Two-dimensional material-worth distributions were computed for a large HTGR. Isothermal temperature coefficients were computed for conditions at the end-of-equilibrium cycle (EOEC). Burnup calculations were performed to determine the average fuel, actinide, and fission product concentrations at various points in the fuel cycle. Separate fuel, moderator, and reflector temperature coefficients were calculated for EOEC conditions. Other calculations included the effects of burnable poison, control rods, xenon, samarium, and ^{233}U on the isothermal temperature coefficient. Additional calculations were performed to determine control-rod worths, the reactivity change caused by core-support-post failure and subsequent central refueling region collapse and neutron-kinetics parameters. This subtask was terminated in [illegible] 1977.

Systems code development

44. The objective of the systems code development effort was to develop, verify, and apply gas-cooled-reactor whole-plant dynamic-simulation computer programs for the analysis of system transients and accidents. The initial code developed was the Composite HTGR Analysis Program (CHAP-1). This code contained models for most of the major components and associated controls for a 3000-MW(t) HTGR power plant. It was based on the Los Alamos-developed Transient and Frequency (TAF) code.

45. The current version of the code, CHAP-2, is written in standard FORTRAN and has many improvements both in the simulation mainframe and in the generic HTGR models. CHAP-2 models both the large HTGR and the FSV station. The CHAP-2 code contains simulation models for the thermal, hydraulic, and neutronic behavior, and control and plant-protective actions of an entire HTGR generating station. The code can simulate a wide range of system transients and accidents with no change in coding. The plant model, initiating events, and subsequent system failures or modifying events can be specified through input. Operator intervention can be simulated when the code is executed in an interactive mode.

46. Approximate numerical solutions to model differential equations are provided by the Los Alamos System Analysis (LASAN) Code. The model-dependent portion of CHAP-2 provides LASAN with system time derivatives, transfer functions and state variable limits. LASAN can provide steady-state, transient, and frequency-response solutions. LASAN methods are robust and well-protected. The code can explore time reversibly, allowing efficient selection of the transient time step. It also can handle nonlinearities such as those associated with controller deadbands. The LASAN code has been thoroughly tested and documented. Comments from an earlier BNL review of CHAP and LASAN have been addressed in the current versions of these codes. CHAP-2 and LASAN will be released at the end of 1982. CHAP-2 and LASAN are already running at a number of installations in the US as well as in the the Federal Republic of Germany.

47. CHAP-2 models for the proposed HTGR lead plant are being developed using the generic component models already in the code. LASAN and the CHAP-2 methodology have potential applications to other nuclear, fossil, or alternative energy systems. Any process or system whose time response can be described by coupled first-order, nonlinear, ordinary differential equations can be simulated using these techniques.

CONCLUSIONS

48. The HTGR Safety Research Program at the Los Alamos National Laboratory began in 1974 with six broad-based tasks: Fission-Product Technology, Primary-Coolant Impurities, Structural Investigations, Safety Instrumentation and Control Systems, Accident Delineation, and Phenomena Modeling and Systems Analysis. Task activities included hardware development as well as experimental and analytical work. The overall program covered the spectrum of safety issues that arose from the introduction of a new

commercial reactor type. The uncertainty about the continued development of HTGRs after mid 1975 resulted in the termination or delay of many of the original tasks. Even with the reduced level of effort, significant HTGR analysis capabilities and technical expertise have been developed. In addition to the specific capabilities for HTGR analysis and safety research described in this paper, the Los Alamos National Laboratory has the ability to develop major computer codes and to design, build and test prototype devices, special equipment, and advanced instrumentation. Existing Laboratory experimental facilities (such as the high-pressure gas-handling and graphite-heating facility developed for the nuclear rocket program and later used for US Department of Energy gas-cooled fast-reactor experiments) are suitable for HTGR safety research programs. It is our hope that the US will continue the development of gas-cooled reactors and that the technological and staff resources of the Los Alamos National Laboratory can contribute to this development.

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