INTERIM REPORT

NRC Research and Technical
Assistance Report

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DEPARTMENT OF ENERGY

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ABSTRACT

Highlights of technical progress during March 1981 are presented for nineteen selected ORNL research programs for the Office of Nuclear Regulatory Research.

NRC Research and Technical Assistance Report

PROGRAM TITLE: Advanced Instrumentation for Redflood Studies (AIRS)

PROGRAM MANAGER: M. B. Herskovitz

ACTIVITY NUMBER: ORNL# 41 89 55 11 8 (189# B0413)/NCR #60 19 11 01

TECHNICAL HIGHLIGHTS:

The PKL-II impedance and film probe electronics and vent tube pressure controller have been shipped. Fabrication of the remaining PKL sensors is continuing. It is presently anticipated that ORNL personnel will supervise the installation and start up of these systems at PKL beginning around July of this year.

In Japan, installation of the in-core flag probes into the CCTF-II core bundles was started. Two ORNL personnel traveled to Takahagi to perform this work. In addition, they have also spent some time at the SCTF facility in Tokai investigating a potential corrosion problem on the stainless steel sensor cables of probes installed in SCTF. A series of inspections and tests are underway at JAERI and ORNL to determine what, if any, corrective action needs to be taken.

Testing of the CCTF-II in-core probe bundle has been completed. As indicated in previous reports, this bundle contained a probe of the new "single electrode" design which was being considered for use in CCTF-II. The bundle also contained a prototype flag probe of the SCTF design which had been used in earlier tests. This probe was fabricated from various components and subassemblies which did not meet all the QA requirements for SCTF. It had failed near the end of the previous test series and was repaired for the current tests. The purpose of these tests was to determine if the void fraction and fluid velocity correlation developed for flag probes would also work for the single electrode probes. Also it was planned to test the correlation at some lower steam pressures than had been used before.

Unfortunately, the repaired flag probe failed early in the test, leaving only the new style probe operating. The test series was completed on this probe. Preliminary analysis of the test data indicates that the new probe does not follow the existing flag probe correlation under identical flow conditions. Further analysis of the data will, hopefully, indicate the extent and nature of the disagreement.

It is undecided at present whether it will be necessary to test another flag probe to extend the correlation to lower steam pressures. This decision will be made after it is determined what the effect of lower pressures and temperatures was upon the new style probe in the tests just completed.

PROGRAM TITLE: Advanced Two-Phase Instrumentation

PROGRAM MANAGER: D. G. Thomas

ACTIVITY NUMBER: 40 89 55 11 5 (189 #B0401)/ NRC 60 19 01 30

TECHNICAL HIGHLIGHTS:

A multiple position heated junction thermocouple (MPHJTC) rod was fabricated and tested under various temperatures and pressures. The rod consisted of 4 pairs of heated and unheated thermocouples spaced along a 36-in. length. This array afforded four level monitoring stations. The rod will be installed in Semiscale in late April and testing should begin in May. The MPHJTC was tested in the pressurizer at the following conditions:

1) Ambient pressure and temperature
2) 395 psia, 440°F
3) 1545 psia, 600°F

The instrument survived all conditions, responded to liquid level as expected and exhibited adequate time response to changing liquid level.

At the mid-year review, February 1981, the pressure seal of the Torsional Ultrasonic Probe in a reactor vessel was identified as a high priority problem. At a subsequent meeting with the staff of one reactor vendor, a variety of seal techniques were discussed and no faults were found with the proposed conceptual designs.

At the mid-year review, February 1981, results were reported that indicated that with moderate air flow rates, minimal amounts of liquid were required to cool shielded heated thermocouple in air/water IDL upper plenum, i.e., appreciable cooling occurred at a void fraction of 0.979. We have now conducted tests in which a fine mist was sprayed transverse to the shielded heated thermocouple well above the location of the heated thermocouple. This meant that whatever mist droplets were collected on the HTC would flow down over the heater region and provide cooling. Detectable cooling was observed when the flow was only 0.03 mℓ/sec corresponding to a Reynolds number of 3.5.

PROGRAM TITLE: Aerosol Release and Transport from LMFBR Fuel

PROGRAM MANAGER: T. S. Kress

ACTIVITY NUMBER: ORNL # 41 89 55 11 1 (189 #B0121)/NRC # 60 19 20 01

TECHNICAL HIGHLIGHTS:

NSPP:

Complete analytical results were received from the laboratory for all aerosol samples taken during the recent Run 401 in which U_3O_8 aerosol was generated and released into a steam environment at 364 K and 0.093 MPa. This environment was maintained for two hours after start of aerosol generation and then steam injection was terminated and the vessel and contents allowed to cool. The rate of mass concentration decrease was higher in the steam environment as contrasted with U_3O_8 aerosol behavior noted under dry conditions at comparable concentration levels. This behavior in steam was also reflected in accelerated aerosol fallout and plateout rates noted over the first few hours of the run.

A second aerosol experiment in this series (Run 402) was conducted in late March. This run was conducted in a manner similar to Run 401 except that the initial aerosol mass concentration was lowered by a factor of 10, or so. Analytical results from this run should be available by late April.

CORE MELT EXPERIMENTS:

Instrumentation of the 0.5 kg split crucible rf induction furnace which is designed to permit direct coupling to melt a small bundle of Zr-4 clad UO_2 fuel pins, is continuing in order to provide accurate readout of temperature and hydrogen release from the Zircaloy metal-water reaction. Additional measurements of cooling water flow-rate and temperature change will provide an approximate value for the rate of power input to the melt being in error by the losses due to radiation and evaporation.

Design for the larger meltdown experiments, following a successful increase in scale in the bench facility to 1.0 kg, is expected to be based on a minor modification of the housing of the existing D.C. electric Arc furnace.

An aerosol collection unit is being added to the bench facility by connecting an air bag fitted inside a steel drum. Most of the carrier gas (N_2) and the evolved hydrogen will be directed to the bag while a small bypass stream is diverted to a prefilter and a thermal conductivity hydrogen monitor. Aerosol characterization of the added tracers including the structural and control rod components with whatever fuel aerosol is generated will be performed by sampling from the air bag.

FAST/CRI-III:

Work has continued on converting the FAST facility to permit performing under-sodium experiments.

Related to acoustic bubble detection development, bench-scale tests indicated that the wave-guides for the acoustic transducers can be welded to the vessel without introducing a major amount of acoustic attenuation at the wave-guide - vessel wall interface. Six pairs of wave-guides are being made; these will be welded to the vessel wall when fabrication is completed.

Work on the revised sodium test plan is continuing. A preliminary decision on the tests to be performed has been made. Related to scaling of test results to reactor accident cases, an evaluation of how to scale estimates of radiation heat loss mechanisms has been initiated.

ANALYTICAL:

A series of survey calculations has been started using the UVABUBL code. The recently added plotting capability makes it possible to evaluate the results rapidly and make greater judgements on appropriate ranges for parameter variation.

One of the NSPP data analysis codes has been modified to ensure that correct conversion of gauge to absolute pressure is made in the treatment of sampling flow-rate data.

PROGRAM TITLE: Continuous On-Line Reactor Surveillance System

PROGRAM MANAGER: D. N. Fry

ACTIVITY NUMBER: ORNL #41 89 55 12 8 (189 #B0442)/NRC #60 19 11 01

TECHNICAL HIGHLIGHTS:

Surveillance System at Sequoyah. The dedicated phone link between ORNL and Sequoyah which provides the ability to control the system from ORNL has proved to be essential during the plant start-up. The surveillance system requires a long period of steady-state plant operation to calculate statistical limits for the pattern recognition discriminants. To insure that this condition is met, the system is being monitored daily via the phone link to ORNL.

Initial plant signal PSDs have been obtained using the system at Sequoyah. This data is stored on a disk by the computer at Sequoyah. To provide the capability of processing and analyzing the PSD data,a program was written for the computer at ORNL to retrieve and process the data stored by the automated system.

Advanced System. The digital system was received from Digital Equipment Corporation (DEC). DEC installed the software operating system and verified operation of the computer.

Modification of the present surveillance software to make it compatable with the new system configuration was initiated. A program was written to perform operational verification of the prototype programmable gain amplifier. The prototype programmable gain amplifier has been operationally verified. The programmable filters have been received and their performance and computer controllability has been verified using the new NRC computer.

Data Analysis. Baseline PSDs were obtained at ∿40, ∿60 and ∿100% power during an ascent to full power in late February. We have performed a preliminary analysis of flow, pressure and power PSDs. Based on this analysis, we have selected three ex-core neutron detectors plus the primary pressure for continuous monitoring during the next month. This signal selection will be reassessed when additional data is available.

The core exit thermocouple signals are contaminated with electrical noise pulses of unknown origin. These signals will be studied in April to determine the source of electrical noise in order to eliminate it. We are also developing a method of signal preprocessing to remove the unwanted pulses from the data before the PSD is calculated.

PROGRAM TITLE: Fission Product Release from LWR Fuel

PROGRAM MANAGER: R. P. Wichner, M. F. Osborne

ACTIVITY NUMBER: ORNL # 41 89 55 10 8 (189 #B0127)/NRC # 60 19 01 40

TECHNICAL HIGHLIGHTS:

1. Furnace Concept Testing

Tests of the induction furnace concept have been extended to >1800°C. Based on heating rate vs power input data, test temperatures <2000°C should be easily attainable with either the tungsten or the graphite susceptor. The ZrO_2 furnace tubes survived four tests in the 1600-1800°C range before the first (relatively minor) cracking appeared. If no serious problems occur during the impending tests to 2000°C, the furnace design will be finalized based on these tests.

A set of six new rectifier tubes and a new rheostat control for the RF generator were procured and installed. If these new components result in stable operation, this power supply will be proven adequate for this test program.

2. Furnace Design

The general design of the new induction furnace is compatible with the existing test apparatus. Detailed design of seals at the ends of the furnace tubes, high temperature thermocouple installation, and assembly/disassembly procedures is in process. Construction of this new furnace should begin during April.

3. Fuel Specimens

The fuel rod sections have been removed from storage to a hot cell in Bldg. 4501. They will be packaged and shipped to Bldg. 3525 (High Radiation Level Examination Laboratory) for preparation of our test specimens. This work will include cutting to length, capping the ends, drilling a small hole in the cladding, gamma scanning to determine the level and distribution of radioactivity (mostly ^{137}Cs), and examination of the fuel and cladding.

4. Data Acquisition and Analysis

Orders have been submitted for (1) a two color optical pyrometer, to measure specimen temperature during testing, and (2) a data acquisition and control unit to be used with the HP 9825A computer. The software needed to acquire and analyze test data by computer is being prepared; this work includes compilation of a new fission product data library for the TP-5000 spectrometer system and verification of the data transfer and storage procedures.

PROGRAM TITLE: Heavy-Section Steel Technology Program

PROGRAM MANAGER: G. D. Whitman

ACTIVITY NUMBER: ORNL #41 89 55 10 1 (189 #B0119)/NRC #60 19 01 30

TECHNICAL HIGHLIGHTS:

Task 1: Program Administration — G. D. Whitman visited NRC offices in Silver Spring, MD, on March 5, to review program planning and discuss the June information meeting.

Eight members of the UK Marshall Committee headed by John Collier of the UKAEA visited ORNL on March 20, to review major program accomplishments since their last visit in 1974.

G. D. Whitman made a midyear presentation at Silver Spring, MD, on March 24.

The Defense Industry Analysis Field Study Group of the Industrial College of the Armed Forces visited ORNL on March 24 and was briefed on the HSST program.

Approximately 250 invitations have been sent out for the June 1–5 information meeting and workshops.

Task 2: Fracture Mechanics and Analysis — A report describing and giving user instructions for the NOZ-FLAW finite-element computer code was published.

Work continues on the development of the ORVIRT-3D Code for 3-D linear or nonlinear fracture calculations. Excellent agreement was obtained with results published by deLorenzi and Newman for the elastic analysis of a standard compact tension specimen. For comparison purposes, ORVIRT-3D elastic calculations for a double edge cracked bar in tension are presently being compared with similar results from the ADINA-J and NOZ-FLAW finite element computer codes.

Two-dimensional elastic-plastic computations were performed with the ADINA-J code for a compact tension specimen previously studied by Atluri et al. A von Mises yield condition with an isotropic strain-hardening rule was employed. The ADINA-J calculations gave excellent agreement with experimental results published by Bucci et al. and by Begley and Landes.

Task 3: Irradiation Effects — Initial testing was conducted on Charpy V-notch (C_V) specimens from the first capsule of the Fourth HSST Irradiation Study. The specimens in this capsule were A533, grade B, class 1 plate, HSST-02, and were irradiated at 288°C to estimated fast neutron fluences of 8 to 20 x 10^{18} n/cm^2 (E > 1 MeV). At 2 x 10^{19} n/cm^2 fluence the shift in C_V transition temperature was about 90°C and the upper-shelf energy drop was about 9% (133 to 114 J). Final neutron dosimetry results should be available shortly.

Irradiation of the second capsule in this study was completed this month with an accumulated exposure of 4330 h (BSR at 2 MW). Disassembly is planned for April.

The third capsule was placed in the BSR facility at the end of this month and irradiation will start in April.

Preparations for the fourth capsule are in progress. Assembly will start shortly after the specimens are received from the Federal Republic of Germany (FRG). The specimens are expected in May.

Work on preparing the tensile machine in the hot cell is continuing.

Task 4: Thermal Shock — Final modifications were made to the OCA-I computer code, and a final draft of a report describing OCA-I was prepared and submitted for review.

A memo describing the parametric analysis of the Rancho Seco overcooling accident was completed and submitted to NRC.

The design of new major test components for TSE-6 was completed and fabrication of these components, including the TSE-6 test cylinder, was begun.

The materials investigations are continuing and the C_V specimens for the dynamic precracked Charpy V-notch tests have been inspected. We also received the 1T compact specimens from thermal shock vessel TSC-2 and precracking has been started.

Task 5: Intermediate Vessel Test, V-8A — Preparations for the test of intermediate vessel V-8A including subcontract work by the Babcock & Wilcox Company (B&W) on the special low-upper-shelf welds and ORNL planning are continuing.

The flawing practice weldment was shipped and has just arrived at our Receiving Department. B&W (Barberton) had encountered problems at both ends of the weld seam which were noted by them on removal of the run-off tabs and backing bar. Tab warpage resulted in insufficient weld deposits both at the top and bottom of the weld. Defects were ground out and the weldment passed an X-ray examination. It is planned to shift the location of the practice flaw in order to utilize acceptable material.

Delays were encountered on the V-8A prolongation-characterization weld seams at B&W (Barberton). The four welds have just been completed and are awaiting preliminary X-ray examinations.

Actual work on the vessel is also running slightly behind schedule. Preparations have been completed for repairing the original test cavity. The cavity for the special low-upper-shelf weld was machined in the vessel.

The contract with B&W was amended to provide additional Charpy impact and J-integral tests of the characterization welds, including 2T compact specimens and precracked Charpy specimens.

Work is under way in our Model Shop to build a 1/5-scale plexiglass model of an intermediate pressure vessel graphite ballast arrangement. The model will serve to develop graphite block securing, loading and unloading schemes, and to assure proper clearance and routing for all internal vessel instrumentation.

Task 6: Pressurized-Thermal-Shock Studies — Fracture mechanics analyses of flaws under combined pressurized-thermal-shock loadings are progressing. It had been estimated previously that crack initiation could be achieved in an intermediate test vessel, based upon assumptions of a uniformly applied shock. Further finite element analyses were made for non-uniform conditions to estimate the magnitude of the effect of the uncooled ends of the test vessel. The effect on initiation of short flaws is estimated to be minor. Study of effects of flaw size and shape is continuing.

Parametric heat transfer and fluid flow calculations have been made of a preliminary layout of a pressurized thermal-shock facility to establish the effect of line size, cooled-fluid storage capacity, intermediate vessel cooled-surface hydraulic diamater and mass flow upon performance. A loop concept has been patterned after the dismantled loop used for the first series of thermal-shock tests utilizing a test tank to direct flow over the outside cylindrical surface of an intermediate test vessel.

Analysis has been initiated to establish the acceptability of subjecting an intermediate vessel to combined thermal and pressure loadings including cycling loadings. Potential problem areas being investigated are the effect of loading on the behavior of the closure seal and on the closure bolting. An axisymmetric 2-D analysis using ADINA and ADINAT is being employed.

PROGRAM TITLE: HTGR Safety Analysis and Research

PROGRAM MANAGER: S. J. Ball

ACTIVITY NUMBER: ORNL #41 89 55 11 2 (189 #B0122)/NRC #60 19 13 02

TECHNICAL HIGHLIGHTS:

Fort St. Vrain (FSV) Reactor Licensing Support: Technical support on questions relating to final approval of RT-500 was provided to NRR. RT-500 is the FSV test procedure for oscillation testing at powers between 70 and 100%. An option was added to the ORECA code to allow for core heatup calculations during a refueling operation. The question addressed was whether the excess flow through a "removed" refueling region would deplete the cooling in the rest of the core to such an extent that the rest of the core would overheat. Preliminary analyses indicated that the core temperature rise would not be excessive.

Code Development: Further work was done on improving the models and user options relating to core bypass flow in the ORECA code. The current options now allow specification of one hot and two cold bypass flow streams. These improvements are being made so that the proposed FSV test plans can be developed.

Miscellaneous: The program mid-year review was held in Silver Spring on March 24, in a joint meeting with other labs (BNL and LASL) and agencies (DOE and NRR). Accomplishments for the past two years were presented, along with plans for the rest of FY'81 and for FY'82. The first draft of a TM summary report on all program work on NRC-sponsored HTGR safety research through September 1980 was completed and distributed for comment.

PROGRAM TITLE: Improved Eddy Current In-Service Inspection for Steam Generator Tubing

PROGRAM MANAGER: Robert W. McClung

ACTIVITY NUMBER: ORNL 41 89 55 12 1 (189 B0417-8)/NRC 60 19 11 05

TECHNICAL HIGHLIGHTS:

We are continuing our task to improve the inspection of steam generator tubing using multiple-frequency techniques.

We participated in a mid-year review held in Silver Spring, Maryland on March 24. The entire program was discussed including the future plans. A list of proposed work has been assembled and sent to the program manager, J. Muscara.

PROGRAM TITLE: Light Water Reactor Pressure Vessel Irradiation Program

PROGRAM MANAGER: F. B. K. Kam

ACTIVITY NUMBER: ORNL 41 89 55 12 (189 #B0415)/NRC #60 19 10 05

TECHNICAL HIGHLIGHTS:

Task 1: Program Administration - F. B. K. Kam presented the NRC Midyear Review of ORNL's LWR Pressure Vessel Irradiation Program in Silver Springs, Maryland. Summary of costs for FY-81 and FY-82, technical highlights, and projected milestones were presented. The need to proceed with the establishment of the Surveillance Dosimetry Measurement Benchmark Facility and with the calculational "Blind Test" of an actual power reactor were emphasized.

Task 2: Benchmark Fields -

A. PCA - Transport Calculations and Dosimetry -

Preparation of ORNL's contribution to a NUREG report on the PCA "Blind Test" has been completed.

The recently developed logarithmic least squares adjustment program (tentatively called LSL) was applied to the data of the REAL80 Project. The 100 group energy structure was reduced to sets of 50, 25 20, and 10 energy groups, respectively. Very little difference was noted in the estimated integral parameters and their variances for different group structures. The results will be submitted to IAEA during April 1981.

B. ORR-PSF -

Irradiation of Simulated Pressure Vessel Capsule (SPVC) is continuing. Testing of thermal characteristics of SSC-2 capsule is being rescheduled from May 1981 to April 1981.

C. ORR-SDMBF -

The Surveillance Dosimetry Measurement Benchmark which was planned for the Bulk Shielding Reactor is now planned for the ORR after completion of the metallurgical irradiation experiment.

Task 3: ASTM Recommended Procedures for LWR-PV Surveillance Embrittlement Program -

A new guide "Application of Neutron Spectrum Least Squares Adjustment Methods" will be completed in June 1981 to replace the Neutron Spectrum Unfolding Methods Guide.

PROGRAM TITLE: LWR Severe Accident Sequence Analysis (SASA)

PROGRAM MANAGER: S. A. Hodge

ACTIVITY NUMBER: ORNL # 41 89 55 13 4 (189 #B0452)/NRC 60 19 01 3 0

TECHNICAL HIGHLIGHTS:

The personnel contributing to the SASA program at ORNL are divided into three working groups. The mission and progress of each of these groups is summarized below. The Severe Accident sequence currently under study is Station Blackout at Browns Ferry Unit One (Loss of Offsite Power and Failure of the Onsite Diesel-Generators to start and load).

Group 1: determines and analyzes the events of the accident sequence which would occur prior to core uncovery, using efficient coding. An efficient set of "mechanized hand calculations" specific to Browns Ferry Unit One has been developed at ORNL for use in obtaining quick-plotted studies of the plant response to hypothetical operator actions. This program for plant simulation is used to provide answers wherein the level of detail provided by the more sophisticated codes does not justify their added expense, nor the additional time required to prepare input and obtain the results.

Work on the Containment Model for the plant simulation coding continued during March. A February meeting at the TVA offices in Knoxville resulted in an invitation for ORNL to review the Nuclear Engineering Branch file on Browns Ferry drywell heat sources. This file exists because Unit 1 has had unusually high measured drywell temperatures (above 150°F) during normal power operation. The data in this TVA file concerns full power operation and thus is not directly applicable to Station Blackout. However, several useful data were found: (1) results of a field test conducted at Browns Ferry to actually measure drywell heat load and (2) an estimate that about half of the full power heat load is due to operation of the AC powered equipment in the drywell. These data have shown that an estimate of 1 Mw for initial drywell heat load following a Station Blackout is reasonable. This estimate was programmed into the containment model; runs were made which show that excessively high drywell atmosphere temperature (because of the loss of the drywell coolers) can be avoided by depressurizing the reactor within 60 minutes after the inception of a Station Blackout.

Various activities for verification of the ORNL plant simulation code were pursued during March. Cases were run for comparison to the transient results presented in Chapter 14 of the Browns Ferry FSAR. The results show a satisfactory agreement. A working visit to the Browns Ferry Simulator at Soddy-Daisy, Tennessee produced Station Blackout results obtained under controlled input conditions for comparison to other codings. A comparison of these results to the results provided by RELAP and the ORNL plant simulation code for the same Station Blackout sequence is in progress and will be reported.

Group II: determines and analyzes the events of the accident sequence which would occur following core uncovery, including core melt and containment failure, using the sophisticated codes.

A. MARCH runs. Several exploratory runs using the MARCH code were completed to determine the range of possible degraded core scenarios following a Station Blackout. It has been determined that the most probable loss of primary containment integrity would be by failure of the electrical penetrations in the drywell.

The results for ten MARCH runs are summarized in Table 1. The versions of MARCH used in these runs are:

MARCH 1.3	developed at BNL	9 Feb 1981
MARCH 1.4	developed at BNL	30 Mar 1981
MARCH 1.4A	developed at ORNL	30 Mar 1981

The MARCH 1.4A version corrects an error in the MARCH coding as discussed in subparagraph B of this progress report for Group II.

The failures of the electrical penetrations in the primary containment (drywell) are assumed to proceed in two stages. First, that a significant containment venting occurs when the drywell pressure reaches 125 psig, as test data have indicated. A realistic assumption for the total opened area around the electrical penetrations over the surface of the drywell at this pressure is 1 in^2, assuming deformation of the penetration seals.

The total drywell surface area devoted to electrical penetration seals is 21 ft^2. As a worst case, it can be assumed that ten percent of this area would be opened at a drywell pressure of 125 psig due to the proportionate number of seals being physically blown from their settings (drywell design pressure is 62 psig). Runs representing both a 1 in^2 and a 2.1 ft^2 electrical penetration failure ("Drywell Venting") at 125 psig drywell internal pressure are summarized in Table 1. However, the 1 in^2 case is considered the more realistic. The results of proprietary tests support this assessment.

The second stage of containment (drywell) failure involves the blow-out of all electrical penetration modules and is estimated to occur if the drywell pressure increases to 175 psig, despite the containment venting around electrical penetrations which becomes significant at a drywell pressure of 125 psig. This second stage is denoted "Containment Failure" in Table 1. However, in the cases where a 2.1 ft^2 area is assumed to open at 125 psig, this is tantamount to containment failure; containment venting and containment failure would be synonymous at 125 psig.

Table 1

Completed MARCH Runs for Station Blackout
at Browns Ferry Unit One

Run No.	Sequence	MARCH Version	Assumed Venting Area/Pressure Drywell Electrical Penetrations	Event Timing (minutes)					Notes
				Core Uncovery	Core Melting	Bottom Head Failure	Drywell Venting	Containment Failure	
1.	No Injection	1.3	1 in^2 @ 125 psia	40	94	183	247	247	1
2.	No Injection and Relief Valve Failed Open	1.3	2.1 ft^2 @ 125 psia	17	71	—	250	—	2
3.	Same as 2	1.4A	1 in^2 @ 125 psia	17	75	220	284	329	
4.	With Injection (Injection Failure after 3 hours)	1.3	2.1 ft^2 @ 125 psia	271	357	433	415	415	3
5.	Same as 4, except relief valve failed open	1.3	2.1 ft^2 @ 125 psia	272	427	—	446	—	2,4
6.	With Injection (Injection Failure after 4 hours)	1.3	1 in^2 @ 125 psia	345	436	556	464	556	
7.	Same as 6	1.4A	1 in^2 @ 125 psia	345	436	547	462	547	5
8.	Same as 6, except relief valve failed open	1.3	1 in^2 @ 125 psia	345	478	—	449	—	2,4

Table 1 (cont'd)

Completed MARCH Runs for Station Blackout at Browns Ferry Unit One

Run No.	Sequence	MARCH Version	Assumed Venting Area/Pressure Drywell Electrical Penetrations	Event Timing (minutes) Core Uncovery	Core Melting	Bottom Head Failure	Drywell Venting	Containment Failure	Notes
9.	With Injection, One Relief Valve Left Open After 15 Minutes. (Injection Failure after 3 hours)	1.3	2.1 ft^2 @ 125 psia	350	450	581	581	581	3,6
10.	With Injection, Six Relief Valves Left Open After 15 Minutes. (Injection Failure after 4 hours)	1.3	2.1 ft^2 @ 125 psia	340	511	820	261	261	3,7

Notes to Table 1

1. Initial Conditions For All Runs

    100% power at inception of Station Blackout
    Drywell Temperature = 150°F, Relative Humidity = 20%
    Torus Temperature = 95°F, Airspace Relative Humidity = 100%
    Realistic decay heat model.

2. Reactor vessel bottom head failure not predicted due to error in MARCH coding. This also precludes predictions of containment failure. MARCH version 1.4A was developed at ORNL for correction of this error.

3. Drywell venting is tantamount to containment failure for cases where a venting area of 2.1 ft^2 is assumed.

4. Core melting delayed with respect to previous run due to lower reactor vessel pressure during and after core uncovery.

5. The difference between results obtained with MARCH version 1.4A and version 1.3 is illustrated by comparison of these results with those for run 6.

6. Injection provided by RCIC system only. Core uncovers briefly (1 min) at about time 46 minutes and then recovers.

7. Injection provided by RCIC system only. Core uncovers briefly (10 min) at time 23 minutes and then recovers. MARCH indicates a steady pressure of about 55 psig attained with six relief valves open. This is sufficient (barely) for RCIC system operation.

B. Correction to the MARCH Code. An error was found in the MARCH coding. For cases where the drywell pressure is greater than the reactor vessel pressure, the failure of the bottom reactor vessel head is not correctly calculated. In this case, MARCH creates an artificial support for the bottom head, so that failure does not occur, even when the code has established that the debris has completely melted through the thickness of the bottom head. The code developers, Battelle/Columbus, have been informed of this code discrepancy. A local ORNL version of MARCH, which corrects this problem, has been established and designated Version 1.4A.

C. Local MARCH Capability. ORNL is currently doing MARCH runs on the CDC computers at Brookhaven (BNL). Local plotting capability has been established, and the results for all future MARCH runs will be plotted locally.

D. MARCH Workshop. Four ORNL SASA personnel responsible for operations with the MARCH code attended a three-day workshop conducted by Battelle/Columbus at TVA headquarters, Knoxville. This attendance was by courtesy of the TVA. The workshop covered MARCH, CORRAL, and other codes for dose rate calculations related to emergency planning procedures.

E. Plant State Recognition. Consideration is being given to measures which might be used by the operator to determine the status, or plant state, of the reactor during the latter stages of a degraded core accident, when the normal instrumentation is not available. Preliminary investigation indicates that the Kalman filter technique may be useful for this purpose. This is a method for the prediction of plant state at any time into the transient based on previous plant status information.[1,2]

F. Suppression Pool Calculations. Satisfactory models for the phenomenology of the Pressure Suppression pool under severe accident conditions do not now exist, at least in the non-proprietary literature. The matter is important because thermal stratification in the pool can reduce the steam-quenching effectiveness of the water by a factor of ten or more. An approach to pressure suppression pool calculations is being undertaken at ORNL, using TRAC-P1A.

TRAC-P1A has been installed and is currently operational on the ORNL IBM computer system. Several runs have been made in which errors in the Browns Ferry suppression pool model data base have been identified and eliminated. Current analysis problems may be related to the structure of the initial suppression pool model or to the assumed initial conditions prior to the transient. Close coordination is being maintained with the TRAC user's advisory group at Los Alamos National Laboratory in the effort to identify and eliminate problems as they arise.

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[1]R. E. Kalman, "A New Approach to Linear Filtering and Prediction Problems," *Trans. ASME, Journal of Basic Engineering*, Vol. 82D, pp 35-45, 1960.

[2]R. E. Kalman, R. Bucy, "New Results in Linear Filtering and Predictions," *Trans. ASME, Journal of Basic Engineering*, Vol. 83D, pp 95-108, 1961.

Group III: analyzes pathways for fission product release to the atmosphere during a Severe Accident, and to determine the magnitude of such releases.

Fission Product Release from Fuel

The rate of release of fission products from melted fuel has been examined in more detail. The results from three types of melting experiments show good consistency in the relative release rates for the various fission products and UO_2. These tests were: melting in helium in tungsten crucibles for 0.4 to 10 min, melting in the Oak Ridge Research Reactor in helium or steam-helium mixtures for 5 min, and transient melting in the pulse-burst TREAT reactor at Idaho for <1 min. The relative release rates were I > Cs >> Ba > Sr > Ru > (UO_2, Ce, and Zr). The release of Te varied and probably depended on capture by Zircaloy cladding. The volatility range (fraction of I and Cs release divided by that for UO_2, Ce, and Zr during the same time period) was $\sim 10^3$.

Results from SASCHA melting tests performed in air by Kernforschungszentrum Karlsruhe, Germany, show similar relative release rates at 2400 to 2750°C except that the volatility range was $>10^5$; at these temperatures extrapolation to the I and Cs release rates was necessary. The Te release rates were midway between Cs and Ba suggesting capture by the Zircaloy cladding.

The difference in the range of volatilities noted above must be understood in order to apply the data with confidence to the analysis of reactor scale melting for use in the SASA program. It is probable that the surface-to-volume ratio of the melt contributed to this difference.

The rates of release of fission products by diffusion from unmelted fuel pieces were examined in a similar manner. The volatility range was as low as 10 for tests conducted at 2000°C in helium. These particular tests used small fuel samples with a high surface-to-volume ratio and long heating times (1.5 to 5.5 h).

Fission Product Transport Calculations

Preparation for the transport calculation has been progressing smoothly. The results of this calculation will indicate where the fission products should be found at any time during the accident. The reactor core and reactor vessel are modeled as ten parallel paths through the core which converge in the lower head and upper plenum. Flow to the Pressure Suppression Pool is through the upper core structure and the pressure relief valves. The primary and secondary containment structures are modeled as a number of well-mixed holdup volumes with assumed leakage and intermixing rates.

A partial interface between the MARCH program and the transport program has been written, and several questions concerning the application of the MARCH results have been resolved. The MARCH calculation gives many useful results, but the program is not designed to determine information such as where the primary containment will fail, or the secondary containment leakage rate in the absence of ventilation fans. Presently, the initial transport calculation is made using a series of necessary assumptions; the results will be refined when better information is available.

Work is continuing on adapting the MARCH code to provide the necessary output for fission product transport calculations. The MARCH workshop with Battelle personnel at TVA was very helpful in understanding the code itself, its limitations, and its liabilities. Perhaps the greatest aid was in learning of several containment flow rates which are needed for transport calculations but are not output by MARCH or mentioned in the user's manual.

Study of Alternate Release Pathways

This subtask is being reactivated as work on the Iodine State of Technology Report winds down. We are currently examining potential fission product leakage pathways which could result from passage through the drywell by means of the installed piping penetrations. In this examination we are assuming that there are no independent component failures unless excess temperature or pressure conditions are imposed; i.e., (1) valves are in their proper positions, (2) check valves function as designed (unless overstressed), and (3) no pipe or tubing failures (unless overstressed).

We have partially completed examination of potential leakage pathways resulting from the main steam line penetrations, and the HPCI and RCIC steam line penetrations. The status of this examination is as follows:

We note that the HPCI and RCIC systems are similar but of different size. Both these systems contain steam trap drains via normally-open DC-motor-operated valves to the main condenser. The purpose of these drains is to keep the steam line to the turbine free of water when these systems are in standby status. These drains will remain open during the latter part of a Station Blackout sequence after DC power is lost due to depletion of the unit battery, allowing direct communication from the reactor vessel to the main condenser. Leakages from the HPCI and RCIC pump seals must be estimated as these are passed by a direct route from the primary system to the floor drain collecting tank in the liquid radwaste system.

We also note that although the MSIV's will be shut during the Station Blackout sequence, each MSIV is designed for a nominal leakage rate in the shut position of about 0.1 liter of steam per second at 50 psi. The significance of this leakage rate is being assessed.

Physical, Chemical Properties, and Precursor Effects

We are collecting data on iodine chemistry and transport rate parameters in preparation for our pathways calculations. In this regard, the State-of-the-Technology Report is quite helpful and is being used as a source book.

PROGRAM TITLE: Multirod Burst Tests

PROGRAM MANAGER: R. H. Chapman

ACTIVITY NUMBER: ORNL # 41 89 55 10 6 (189 # B0120)/NRC # 60 19 11 04 1

TECHNICAL HIGHLIGHTS:

R. H. Chapman and J. L. Crowley visited NRC/RSR offices in Silver Spring, MD, on March 24 and 25 to participate in the FY 1981 midyear review and to hold programmatic and technical discussions with RSR personnel. Based on extensive preliminary analysis of the B-5 (8×8) burst test data, a decision was made to increase the size of the last test (B-6) planned for this program from a 6×6 array to an 8×8 array. Also, since the test will incorporate an unheated, closely fitted shroud and since the outer ring of simulators function primarily as deforming guard heaters, it was decided to include temperature measuring instrumentation in only eight of the 28 simulators in the ring. Test conditions, i.e., heating rate of ~7 K/s with bursts in the temperature range of 925–950°C, remain the same. Results of the test should establish a lower limit for the deformation and flow blockage expected in large arrays under conditions tested in this program.

As a result of this change, the B-6 test will be delayed approximately three months; current plans are for the test to be conducted in December.

Three single rod tests were conducted to compare the effect on deformation of temperature measuring instrumentation as used in MRBT and JAERI bundle tests. The lack of good agreement in results from the two test programs is believed to be associated with the differences in instrumentation and fuel simulator designs. Results of the tests are not yet available to confirm this belief.

The B-4 (6×6) bundle was disassembled by slitting the grids to remove one layer of tubes at a time. Additional photographs were made of the test array (after each layer of tubes was removed) to document the deformation. The individual tubes will be measured to determine strain profiles. The tube (No. 14) exhibiting the largest volume increase (~48%) and, hence, average strain (~22%) was found to have a localized ballooned region, having a maximum strain of ~45%.

Flow tests on the B-5 bundle are underway at the B&W Alliance Research Center. Tests at three different flow rates have been conducted thus far; two additional flow rates will be investigated. The bundle will be returned to ORNL in late April, at which time it will be cast in an epoxy matrix for sectioning.

PROGRAM TITLE: Noise Diagnostics for Safety Assessment

PROGRAM MANAGER: D. N. Fry

ACTIVITY NUMBER: ORNL #41 89 55 11 4 (189 #B0191)/NRC #60 19 11 01

TECHNICAL HIGHLIGHTS:

PWR In-Core Vibration Detection Study. One-dimensional adjoint calculations of ex-core neutron detector response were performed using the TASK code. The results of these calculations will be used to interpret and assess the validity of two-dimensional diffusion calculations performed with the JPRKINETICS code.

Non-Neutron Signals for Safety Assessment. We have completed a review of the literature on use of pressure signals for reactor system fault detection. Based on this review we prepared an initial program plan to assess the use of pressure signals for surveillance and diagnosis of safety problems. The overall objectives of this program would be to: (1) analyze existing data to determine the sources of pressure noise in PWRs; (2) study pressure sensor response; (3) study propagation of pressure waves in PWRs; and (4) conduct tests in out-of-reactor facilities to assess use of pressure signals for fault detection.

Loose Parts Monitoring. We received comments from NRC and other reviewers on a draft NUREG report, "Summary of Fundamental Studies on Methods for Detecting, Locating, and Characterizing Metallic Loose Parts in Nuclear Reactor Coolant Systems". A revised draft is being prepared.

Baseline Signature Acquisition. We obtained baseline data from two four-loop Westinghouse PWRs. Neutron and accelerometer noise measurements were made at the Trojan plant by personnel at the University of Washington under subcontract to ORNL. In addition to supplying ORNL with analog tape recordings, University of Washington noise analysts submitted a letter report summarizing the results of data analysis and their speculation on the sources of normal neutron noise in Trojan.

We also obtained baseline data from Sequoyah-1 in connection with the demonstration of an on-line noise surveillance system (FIN #B0442). Data was obtained at $\sim$40, $\sim$60, and $\sim$100% power during plant startup. Signatures were obtained for two excore neutron detectors, primary flow and primary pressure. This data is being analyzed and compared with baseline data from other plants.

Primary Coolant Inventory Monitoring. We expect to meet with TVA design engineers in April to review instrumentation for primary coolant inventory monitoring.

Meetings. We will attend and present a paper at the 14th Informal Meeting on Reactor Noise to be held in Munich, Germany, April 28-30, 1981.

A summary paper, "Advances in Automated Noise Data Acquisition and Noise Source Modeling for Power Reactors," was submitted to the paper review committee for the Third Specialists' Meeting on Reactor Noise (SMORN-III) to be held in Tokyo, Japan in October 1981.

We described FY 1981 program accomplishments, schedule, and future directions at a mid-year review meeting at NRC on March 25th.

PROGRAM TITLE: Nuclear Safety Information Center

PROGRAM MANAGER: W. B. Cottrell

ACTIVITY NUMBER: ORNL #41 89 55 10 4 (189 #B0126)/NRC #60 19 11 01

TECHNICAL HIGHLIGHTS:

The purpose of NSIC is to collect, evaluate, and disseminate information on the safety of nuclear facilities to the nuclear community through reports and the bimonthly technical progress review, *Nuclear Safety*, and to NRC staff members and their contractors through SDI, searches, and consultation. (The work reported herein is that with which NRC is especially concerned and for which the NRC RES Program supported with 500 K, i.e., 66% of the total NSIC budget.)

During the month of March, the staff of the Nuclear Safety Information Center (a) processed 1572 documents, (b) responded to 45 inquiries (of which 19 involved the technical staff), and (c) made 18 computer searches. The RECON System, which now has over 200 remote terminals, reports that the NSIC data file was accessed 204 times between February 2 to 27 making it the fifth most utilized of the 26 data bases on RECON (see attached Table 1). During the past month, the NSIC staff received 8 visitors and participated in 6 meetings.

Several NSIC reports are in various stages of preparation, including *Nuclear Power and Radiation in Perspective* (ORNL/NUREG/NSIC-161); *Role of Probability in Risk and Safety Analysis* (ORNL/NUREG/NSIC-167); *Annotated Bibliography on Fire and Fire Protection in Nuclear Facilities* (ORNL/NUREG/NSIC-172); *Summary and Bibliography of Safety-Related Events at Boiling Water Nuclear Power Plants as Reported in 1979* (ORNL/NUREG/NSIC-178); *Summary and Bibliography of Safety-Related Events at Pressurized Water Nuclear Power Plants as Reported in 1979* (ORNL/NUREG/NSIC-179); *Nuclear Power Plant Operating Experience — 1979 Annual Report* (ORNL/NUREG/NSIC-180); and *Index to Nuclecr Safety, Vol. 11(1) through Vol. 21(6)* (ORNL/NUREG/NSIC-186).

During the month of March, we received 17 foreign documents (16 German a d 1 United Kingdom). In accordance with the arrangements effective January 1, 1979, a copy of each of these have been sent to R. S. Scott (DDC) for microfiche processing. In addition, the foreign language documents were reviewed for translation (see letter of March 31, 1981, to H. H. Scott, RSR).

During the month of March, NSIC's Selective Dissemination of Information (SDI) was providing service to a total of 329 users, including 1 new user. It should be noted that consistent with our FY-81 budget, "cost-recovery" services have been eliminated so that as "paid" subscriptions expire they are not renewed. (Note: There are 8 paid subscriptions yet to expire.) However, this and other services to NRC, DOE and their subcontract personnel are not affected.

All technical articles for *Nuclear Safety* 22(4) were completed and mailed to NRC, DOE and TIC on March 20th. The "current events" material (covering events which occurred during January and February) for *Nuclear Safety* 22(3) was completed by March 16 (except for the data on operating power reactors which was not yet available from NRC). Most technical articles for *Nuclear Safety* 22(5) have been received, submitted to peer review, and are in various stages of preparation. Final copies of *Nuclear Safety* 22(2) are expected from the printer (via TIC) momentarily.

The NRC Midyear Review of the Nuclear Safety Information Center was held at NRC's Silver Spring offices on March 26th with about 20 NRC staffers present. This review differed from previous ones principally in that the responsibility for NSIC will be transferred at the end of this fiscal year from Research to the Office For Analysis and Evaluation of Operational Data. While there were no comments regarding NSIC scope and general activities, NSIC was directed to undertake some housekeeping activities (most notably the survey of SDI users) and to provide certain specific information to Headquarters personnel. The desirability of both advising NRC personnel of the availability of certain NSIC services and of establishing an advisory committee was discussed, but no decisions were reached.

TABLE 1 RECON DATA BASE ACTIVITY FROM 02-02-81 TO 02-27-81
(21 OPERATING DAYS)

DATA BASE IDENT.	DATA BASE NAME AND SUPPORTING INSTALLATION IDENTIFICATION	NO. OF SESSIONS	NO. OF EXPANDS	CITATIONS PRINTED
EDB	(TIC) DOE ENERGY DATABASE	3774	4396	123800
NSA	(TIC) NUCLEAR SCIENCE ABSTRACTS	569	638	7266
WRA	(WRSIC) WATER RESOURCES ABSTRACTS	291	794	14033
EMI	(EMIC) ENV. MUTAGENS INFO.	210	343	15008
NSC	(NSIC) NUCLEAR SAFETY INFO. CENTER	204	209	13025
RIP	(DOE) ENERGY RESEARCH IN PROGRESS	172	158	2094
GAP	(DOE) GENERAL AND PRACTICAL INFO.	137	108	1380
FED	(DOE/EIA) FEDERAL ENERGY DATA INDEX	132	135	1494
ESI	(EIC) ENV. SCIENCE INDEX	103	198	1844
ETI	(ETIC) ENVIRONMENTAL TERATOLOGY	87	56	1826
EIA	(EIC) ENERGY INFO. ABSTRACTS	82	113	278
WRE	(WRSIC) WATER RESOURCE RESEARCH	72	111	821
IPS	(TIC) ISSUES AND POLICY SUMMARIES	32	32	28
SLR	(FRANKLIN) SOLAR DATA BASE	32	22	-
ERG	(BERC) ENHANCED OIL AND GAS RECOVERY	29	24	271
RSI	(RSIC) RADIATION SHIELDING INFO.	29	57	487
PRD	(TIC/NRC) POWER REACTOR DOCKETS	28	18	1167
NES	(NESC) NATIONAL ENERGY SOFTWARE	27	31	62
GID	GOVERNMENT & INDUSTRY DATA EXCHANGE	26	69	371
NTB	(NASA) NASA TECH BRIEF FILE	26	60	27
CIM	(DOE) CENTRAL INVENTORY OF MODELS	23	23	-
NRC	(LC) NATIONAL REFERRAL CENTER	22	28	105
EIS	(TIRC) EPIDEMIOLOGY INFO. SYSTEM	0	8	74
TUL	(U. TULSA) TULSA DATA BASE	18	28	259
SUP	(TIC) THESAURUS SUPPLEMENT	16	9	-
SER	(TIC) SERIAL TITLES DATA BASE	13	15	-
OGR	OIL AND GAS RESERVE FILE	9	5	-
RSC	(RSIC) RADIATION SHIELDING CODES	9	9	44
NSR	(NDP) NUCLEAR STRUCTURE REFERENCE	7	4	-
ARF	(EMIC/ETIC) AGENT REGISTRY FILE	2	-	-

PROGRAM TITLE: Operational Aids for Reactor Operators

PROGRAM MANAGER: G. F. Flanagan

ACTIVITY NUMBER: ORNL #41 88 55 02 8 (189 #B0438-9) NRC #60 19 50 01 0

TECHNICAL HIGHLIGHTS:

During the month of March, staff members toured the Oak Ridge Gaseous Diffusion Plant Central Control Facility. The facility functions as a process monitor for the gaseous diffusion process and as a control center for electrical power distribution. The facility has been in operation for about one year. Further details are available from a trip report.

R. A. Kisner presented a seminar at Virginia Polytechnic Institute and State University on human factors in nuclear power plant operation. The seminar was given to mechanical and nuclear engineering graduate students to acquaint them with plant operational problems and ongoing work on human performance.

Staff members attended a five day course on human factors presented by The University of Tennessee Industrial Engineering Department. The course offered a good perspective on human factors and promoted discussion of current problems facing control room designers.

The NRC Mid-Year Program Review was held at Silver Spring on March 25. R. DiSalvo and his staff were in attendance. NRC representatives commented that the results of the operational aids for NPP Operators Program may not have been directly usable by Regulatory Staff. The content of the program has been qualitative, consisting of technology transfer of information from non-nuclear to nuclear and evaluation of behavioral theories related to operator performance. Many of the results of the program are subjective and tend toward generalizations by their very nature. Such results are difficult to use directly in licensing, but connected with an experimental program, much can be learned. The following actions are being taken immediately to bring the program abreast of user needs:

1) Prepare program brief for FY 1982, approximately 2-3 pages by April 7.

2) Begin concentrating efforts in those areas where ORNL has expertise, and rely on consultants and not R&D subcontracts.

3) Concentrate on effort to produce operator role report (due in July).

4) Table report on operator stress until FY82.

5) ORNL will collect information on operator acceptance from relevant nuclear and non-nuclear industries (use consultants where needed) and present findings and recommendations in a paper at the Myrtle Beach conference in August.

6) FY 1982 work should be scoped so as to emphasize those areas associated with operational aids that ORNL can research without subcontractor aid (hardware, software, validation, verification, automation, control, review, need for, and availability of proper instrumentation, etc.). We should de-emphasize behavioral aspects unless we develop expertise in this area.

7) FY 1982 work plan should <u>not</u> include funding for operator/plant model development, but suggest that this awaits the results of the feasibility study.

PROGRAM TITLE: Postaccident Iodine and Tellurium Chemistry

PROGRAM MANAGER: R. P. Wichner, L. M. Toth

ACTIVITY NUMBER: ORNL # 41 89 55 13 5 (189 #B0453-1)/NRC # 60 19 01 10

TECHNICAL HIGHLIGHTS:

The general objective of the aqueous iodine laboratory study is the determination of the relative abundances of the main iodine species in a series of aqueous conditions up to temperatures of 300°C, and the determination of the effective iodine volatility under these conditions. This will be accomplished in steps which are defined principally by the temperature because the materials of containment are limited to definite temperature regimes. The major objective of the most recent work has been to obtain a container system in which aqueous iodine solutions can be studied spectrophotometrically at temperatures up to 150°C. On developing such a system, we shall identify the hydrolysis and redox products of molecular iodine and their distribution coefficients between liquid and vapor phases. It will often be possible to obtain some of these data during the course of the overall system development at temperatures up to our current objective.

We have found that stainless steel (304 L) is not a suitable container for iodine solutions at any temperature because the iodine is reduced rapidly to (presumably) iodide. This reaction so dominates the chemistry of iodine that studies in unprotected metallic containers will not produce answers to the hydrolysis chemistry that are being sought. (At 150°C iodine reacts with the stainless steel within seconds.) Attempts to coat the stainless steel with plastics such as Teflon have not proved successful because the iodine diffuses through the thin coating and loosens it by corroding the metal underneath. We have therefore ruled out coated surfaces. Two alternatives are currently under development: (1) a metal cell with a thick Teflon liner that can tolerate some degree of iodine loss through diffusion, and (2) a sealed silica insert cell (containing the iodine solution) within a steel cell that will withstand the pressures or the superheated water at 150°C and higher.

PROGRAM TITLE: PWR Blowdown Heat Transfer-Separate Effects

PROGRAM MANAGER: W. G. Craddick

ACTIVITY NUMBER: ORNL #41 89 55 10 3 (189 #B0125)/NRC 60 19 11 01

TECHNICAL HIGHLIGHTS:

Task 1: Single-Rod Loop Testing — This task has been completed.

Task 2: Analysis — Documentation of ASHUR (a heat balance code for the THTF) has been completed.

Documentation of the THTF bundle 3 in-core instrumentation, operating history, and FRS studies (such as radial dimensions and surface emissivity) has been completed.

Analysis of the ORNL Small Break LOCA Heat Transfer Test Series 2 (3.09.10I-X) is continuing. An interim report is scheduled for completion in May of 1981.

Final data tapes were shipped to the INEL data bank for the following tests:

Transient Upflow Film Boiling Test 3.06.6B
Transient Upflow Film Boiling Test 3.08.6C
Steady-State Film Boiling Test Series 3.07.9B-X

Preparation of an interim report for THTF Test 3.08.6C is continuing and approximately 80% complete. The report will discuss film boiling heat transfer in all THTF transient film boiling tests and propose a logic for predicting best estimate dispersed flow heat transfer for use in equilibrium reactor safety codes. Test 3.08.6C had a much higher flow than the other transient film boiling tests but this did not seem to significantly affect results. The conclusions of the report will be similar to those of previous interim reports.

Development of software for final analysis of the steady-state film boiling tests is continuing and approximately 50% complete.

Task 3: THTF Operations — During March, the THTF was returned to its original design, 4-in. piping configuration.

Task 4: Two-Phase Instrument Development — Work is continuing on processing mass flux spool piece data from the steady-state film boiling tests (3.07.9) using the mass flow code, AMICON. The work is 70% completed. The data is primarily for high quality (0.8-1.0) two-phase flow. The data will be used to estimate two-phase mass flux uncertainties for mass flux models using the spool piece instruments: turbine meter, drag disk, and gamma densitometer.

An uncertainty analysis for the drag disk (spool piece momentum flux measurement) based on subcooled water calibrations from four THTF tests has been completed.

An uncertainty analysis for the Rosemount test section differential pressure cells (PDE-180 series) based on low flow, subcooled data scans has also been completed. The instrument provided detailed void fraction distribution for the SBLOCA 2 Test Series.

PROGRAM TITLE: Safety Related Operator Actions

PROGRAM MANAGER: P. M. Haas

ACTIVITY NUMBER: ORNL #41 89 55 12 3 (189 #B0421-8) NRC #60 19 11 01 2

TECHNICAL HIGHLIGHTS:

Operator Actions Data Collection

Simulator data collection for BWR safety-related actions continued. The first phase of requalification training for "Utility A" has been completed, and data tapes were made of all requalification training for that utility (14 groups of operators), including some of the accident sequences selected for calibration to field data. Data collection for "Utility B" will be initiated before April 1.

Analysis of field data from PWR sites has not yet been completed. A draft report from MSU/CNS is expected in April. Field data collection at BWR sites has been initiated.

Simulator Response Characteristics

A revised draft of the report summarizing current nuclear and non-nuclear industry practice for specifying and verifying simulator response has been completed by Franklin Research Center and is being reviewed by ORNL and NRC. Work for the remainder of the project will concentrate primarily on three areas:

1) more specific information on model validation performed by the NPP simulator vendors during the simulator development process;
2) further review and assessment of non-nuclear experience and research to support the use of "training effectiveness" as the time measure of simulator fidelity;
3) review of post-TMI requirements for increased plant diagnostic information to determine effective ways of using those data for simulator response verification.

PROGRAM TITLE: Subcritical Reactivity Monitoring by the Californium-252 Source Driven Neutron Noise Method

PROGRAM MANAGER: C. W. Ricker

ACTIVITY NUMBER: ORNL #41 89 55 13 2 (189 #B0451)/NRC #60 19 11 01

TECHNICAL HIGHLIGHTS:

Calculations using the JPRKINETICS code on the initial fuel loading of Zimmer Core were performed during this reporting period for the case with the source in the center of a fully loaded core with all poison rods inserted and the central rod withdrawn. The results are being utilized to calculate measured quantities as a function of detector and source positions in order to identify detector locations that would produce meaningful experimental results.

PROGRAM TITLE: Zircaloy Fuel Cladding Creepdown Studies

PROGRAM MANAGER: D. O. Hobson

ACTIVITY NUMBER: ORNL #41 89 55 11 7 (189 #B0124)/NRC #60 19 11 04 1

TECHNICAL HIGHLIGHTS:

The final report for the subject program has been issued as NUREG/CR-1844, ORNL/NUREG-74, "Analyses of Surface Displacements of Zircaloy Fuel Cladding in the HOBBIE Creepdown Irradiation Experiments," by D. O. Hobson et al, March 1981.

The abstract from the report serves as a good description of the results:

> This report presents descriptions and results for seven of the eight in-reactor creepdown tests of Zircaloy fuel cladding. These tests, part of a joint program between the U.S. Nuclear Regulatory Commission and Energieonderzoek Centrum Nederland, were conducted to study the behavior of Zircaloy fuel cladding under conditions that approximate those found in an operating pressurized-water power reactor.
>
> We present radial surface displacements as functions of time, average diametral or circumferential strains as a function of time, and isochronal deformation surfaces. These tests were conducted at 343°C with external pressures from 13.3–18.7 MPa. Three of the specimens were subjected to stress reversal during their test runs, after which they were pressurized internally to pressures from 3.6–6.9 MPa. Fast flux [>1.0 MeV (0.16 pJ)] ranged from 3.8×10^{17} to 5.0×10^{17} n/(m^2·s).
>
> The most important conclusion to be drawn from this study involves the deformation of the cladding during testing. Contrary to similar tests conducted out of reactor, the in-reactor specimens did not deform uniformly, that is, by diametral contraction and smooth ovalization. Rather, the deformation surfaces were nonuniform — hills and valleys formed at irregular intervals. This implies that conventional concepts of creep rate and simplified modeling procedures are insufficient for predicting cladding behavior. Sufficient data have been collected in this program to supply modelers with detailed, virtually hour by hour, descriptions of the cladding surface shapes. From these data new interpretations can be derived to predict cladding behavior.

This completes the last milestone and concludes the program.

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