2223

The Technology Of Nuclear Reactor Safety

Volume 1.

Reactor Physics and Control

Edited by
T.J. Thompson
and J.G. Beckerley

Prepared under the auspices of the
Division of Technical Information,
U.S. Atomic Energy Commission

The Technology of Nuclear Reactor Safety

VOLUME 1
Reactor Physics and Control

EDITORS
T. J. Thompson
J. G. Beckerley

Prepared under the auspices of the
Division of Technical Information
U. S. Atomic Energy Commission

THE M.I.T. PRESS
Massachusetts Institute of Technology
Cambridge, Massachusetts

Library of Congress Catalog Card Number 64-24957

PRINTED IN THE UNITED STATES OF AMERICA

CHAPTER 11

Accidents and Destructive Tests

T. J. THOMPSON
Massachusetts Institute of Technology
Cambridge, Massachusetts

"What Is Past Is Prologue"

CHAPTER CONTENTS*

*Except for a few changes and additions made in proof, this chapter is based on information in the literature or known to the author prior to February 1964.

PREFACE

In reviewing the information available on the various accidents, it has appeared to the author that the accident reports from Windscale and Chalk River were particularly outstanding. They presented, in a logical manner, the facts concerning what happened and drew frank conclusions. The reports were informative enough to supply the reader with an artificial backlog of experience so that he might be forewarned of certain problems. Since the intent of this chapter is the same, the author has tried to continue the tradition of those earlier reports in an abbreviated fashion.

Because a report of an accident loses much of its value if there are no conclusions, some of the more important ones have been drawn using those from the original reports, those of other reviewers, and those of the author. In a number of cases the conclusions of this chapter augment or even differ from those of the original reports. In those cases it should be understood that the author is presenting his personal conclusions. It is hoped that the authors of earlier reports and the groups involved will understand that the sole intent is to emphasize the lessons to be learned.

1 INTRODUCTION

While there have been relatively few accidents of a serious nature involving nuclear reactors, yet each one points out certain lessons worth preserving. It is the intent of this chapter to set down the recorded facts concerning these accidents, emphasizing the causes of the accident and the lessons to be learned.

Table 3–1

REACTOR ACCIDENTS INVOLVING CRITICALITY OR REACTIVITY CHANGES

Date	Location, name	Active Fuel Coolant, Moderator	Geometry	Total Fissions	Cause (C) Quenching Mechanism (Q)	Person (P) Radiation dose (R)	Damage	Ref. in this chapter
Dec. 1949	LASL (water boiler) crit. 1944, 1950	1 kg U^{235}, $UO_2(NO_3)_2$ in 13.6 liter H_2O	Sphere, graphite reflector	$3\text{-}4\times10^{16}$	(C) Manual withdrawal of 2 control rods (Q) Expansion and rise of neutron temperature	P-1: R-2.5 r gamma	None	Sec. 3.2
12 Dec. 1952	Chalk River, Canada NRX, criticality 1947, full power May 1948	Natural uranium rods H_2O-cooled, D_2O-moderated	Rod lattice, graphite reflected	0.6×10^{20}	(C) Control rod mal-operation, safety circuit failure-complex (Q) Dump of D_2O moderator	P-none, except in clean-up; many P got small doses, highest 17r, most less than 3.9r	Core badly damaged, removed, replaced	Sec. 3.3
22 July 1954	NRTS Idaho BORAX I (transient tests 1954)	93% enriched U^{235} in U-Al plates (MTR type), H_2O-moderated	H_2O reflector, swimming pool excursion reactor	4.68×10^{18}	(C) Estimate of expected excursion low (Q) Steam void disassembly	P-none R-none	Core destroyed	Sec. 3.4
3 Oct. 1954	Hanford Production Reactor First one critical September 1944	Natural uranium rods H_2O-cooled, graphite-moderated	Process tube type—large graphite reactor	Local over-heating	(C) Water leak changed reactor pattern, short period occurred (Q) Control rod changes then scram	P-none R-none	Some fuel elements failed or were damaged	Sec. 3.5
4 Jan. 1955	Hanford – KW Reactor	. . . . same	. . . same . . .	Fuel failure, local melting	(C) Blockage of cooling water in process tube – initial start-up power decrease noted – rods withdrawn	P-none R-none (Q) Scram on over-pressure	Graphite channel removed by hole cut in shield	Sec. 3.5
Jan. 1955	Hanford Production Reactor	. . . . same	. . . same . . .	No over-power	(C) Misestimate of ρ-instruments (Q) Rod run in by operator	P-none R-none	None	Sec. 3.5
29 Nov. 1955	NRTS Idaho EBR-I Mark-II Operations in 1951	0.5 in. U^{235} rods, NaK-cooled fast reactor	Compact core Nat. U blanket	4.7×10^{17}	(C) Estimate of expected results low – earliest scram attempts not effective (Q) Shutoff by second scram; fuel bowing a factor	P-none R-minor	Core melted, little other contamination	Sec. 3.6
9 Oct. 1957	England-Windscale Operations in July 1950	Natural uranium rods air-cooled, graphite-moderated	8-sided stack of graphite $50\times50\times25$ ft 25 ft fuel channels	Graphite-uranium fire	(C) Wigner energy release, U-burning triggered by nuclear overheating (Q) Flooding with H_2O	P-none serious R-widespread radioactivity, milk over 200 mi^2 area destroyed	Severe core damage reactor not rebuilt	Sec. 3.7
18 Nov. 1958	NRTS – HTRE-3 crit. October 1958	Enriched uranium gas-cooled, solid moderator	Horizontal cylinder	Not known	(C) System on auto-control with faulty instrumentation (Q) Meltdown slump and/or scram	P-none R-some site contamination	Core melted, basic system undamaged	Sec. 3.8

18 Nov. 1958	NRTS – HTRE-3 crit. October 1958	Enriched uranium gas-cooled, solid moderator	Horizontal cylinder	Not known	(C) System on auto-control with faulty instrumentation (Q) Meltdown slump and/or scram	P-none R-some site contamination	Core melted, basic system undamaged	Sec. 3.8
24 July 1959	Santa Susanna, Cal. SRE crit. 1957 at power May 1958	2.8% U^{238} slugs in SS-clad rods Na-cooled graphite-moderated	Pseudo-cylinder, graphite reflector	2×10^{19} (in last minute)	(C) Coolant channel blockage by impurities overheating, perhaps fuel bowing (Q) Manual scram	P-none R-release was ~0.3% of core activity inventory	12 of 43 elements melted, core removed and replaced	Sec. 3.9
3 April 1960	Waltz Mill, Pa. WTR crit. 1959	93% enriched U^{235} U-Al plates-cylinder H_2O-cooled-moderated	Pseudo-cylinder, water reflected	Overheat of one element	(C) Undercooled, perhaps faulty fuel – negative auto control response (Q) Manual scram	P-none R-minor release to site	1 element melted, $10^6 to clean up damage	Sec. 3.10
3 Jan. 1961	NRTS SL–1 crit. August 1958	93% enriched U^{235} Al-U plates, boiling, H_2O-cooled-moderated	Pseudo-cylinder 5 rods, B-Al strips control	1.5×10^{18}	(C) Manual withdrawal of central control rod (Q) Expansion, boiling, core evaporation	3P-all fatal R > 800r/hr in bldg.; in recovery 14P got R > 5r	Core destroyed, vessel rose 9 ft, reactor dismantled	Sec. 3.11
5 Nov. 1962	NRTS SPERT I (destructive tests)	93% enriched U^{235} Al-U plate type H_2O-moderated	Pseudo-cylinder with transient ρ-rod, open tank		(C) Planned experiment–reactivity transient as planned, energy release effects more destructive than planned (Q) Expansion, steam void, no melting involved	P-none R-minor site contamination	Core destroyed (destruction planned on this test or next)	Sec. 3.12

Up until July 17, 1964, the method of recovering this uranium from TCE was to place 2 or 3 liters of TCE solution in a 5 in. (12.7 cm) diameter, 11 liter, critically-safe polyethylene bottle. About 6 liters of sodium carbonate solution were added. The bottle was then shaken by hand and the liquids allowed to separate. The enriched uranium, now dissolved in the sodium carbonate solution, was then removed by means of a separatory funnel.

The weight and size of the container full of liquid made the mixing difficult for the operators and soon an operator suggested another easier method. This method made use of mechanical agitation by a stirrer in a non-critically-safe sodium carbonate make-up tank. This 30 gal stainless steel tank was 18 in. in diameter with a 26 in. vertical wall and a standard dished bottom. The normal function of this tank was solely to make up sodium carbonate solution for a solvent washing operation. It was not designed for any use involving uranium solutions.

The new procedure was discussed with a supervisor before it was put into effect and was approved informally by the supervisor provided that the concentration of uranium was known. A limit of 800 ppm for use of the procedure was informally established. The new method was used by two shifts beginning on July 17, 1964, and the operator in question utilized it on July 17 as well as on July 24, the night of the accident.

Early on the morning of July 23 an evaporator failed to operate properly. It was discovered that a pipe to the evaporator was plugged with uranium nitrate crystals. Steam was used to unplug the line. When the line was unplugged, material from the evaporator system was drained into several 5 in. diameter, 11 liter bottles.

It is presumed that one of these bottles of highly concentrated U^{235} solution was responsible for the accident. Evidently, at approximately 18:05, July 24, through some mischance the operator in question emptied one of these bottles into the non-critically-safe make-up tank which already contained approximately 15 gal of aqueous sodium carbonate solution.

When nearly all the liquid had been poured into the tank the excursion occurred, solution was expelled from the vessel, and radiation alarms sounded. The operator fell backward to the floor, arose, and ran from the building to an emergency shack about 200 yds from the main building. He died at 19:20, July 26, approximately 49 hr after the incident. It has been estimated that the operator received more than 700 rem of gamma radiation and an additional large fast neutron dose.

3 ACCIDENTS AND DESTRUCTIVE TESTS INVOLVING REACTIVITY CHANGES IN REACTORS

3.1 General

This section summarizes (through 1963) the reported accidents and destructive tests involving reactivity changes in reactors. Table 3-1 outlines the principal features of all of these accidents. The last column of the table indicates the subsection of this chapter in which each accident is discussed.

In this chapter a reactor is defined as a fission

chamber, and the ion current from the principal chamber fell more rapidly than that from the gamma compensation chamber. Thus, with increasing flux the net current would appear to drop to negative values since the gamma compensation current is subtracted. The result could, and did, appear as a negative period - when actually the period was positive.

Comments, Conclusions, Recommendations

(1) There should always be at least two on-scale neutron signals preferably from different types of chambers with different types of electrical circuits—for example, a fission chamber detector utilizing a battery power supply and operating a galvanometer, and an ion chamber with its own voltage supply and amplifier-recording unit. The reactor instrumentation system should satisfy the Principle of Diversity discussed in the chapter on Sensing and Control Instrumentation, Secs. 1.4.1 and 1.4.2.

(2) Safety system circuitry should not be made a part of an experiment or of a control unit. Use of instruments for both safety and control reduces the number of independent safety circuits and thus violates the Principle of Redundancy discussed in the chapter on Sensing and Control Instrumentation, Sec. 1.4.2.

(3) Records should be kept of all changes made to any part of the system including the circuitry. Before any changes are made, a review (by someone or some group completely knowledgeable of the given reactor and the points at issue) should be carried out of the effect that the proposed change will have on the existing system. This review should also consider any previous alterations which may affect the situation.

(4) Reactor instrumentation should be designed to trip, or at least sound an alarm, on fast negative periods as well as fast positive periods.

(5) The use of automatic controls to carry out a rise to power may be somewhat questionable unless the system has been repeatedly checked and found to be completely reliable in taking appropriate action in the event an unexpected input signal is obtained. In particular, in approaching a new and higher power level slow and careful manual operation by competent operators is likely to be the more conservative approach. Especially this is true when the instrumentation is untested in the desired operating range. Manual operation using slow steps would probably have given the operator a chance to catch the problem in time. It is difficult to build judgment into an automatic startup method which is as sound and broad as that of an alert and knowledgeable human unless the situation is quite routine. (The situation here contrasts somewhat with that discussed in Sec. 3.6 where extremely rapid operations were required. Even there, the system, when automated should have been fully checked out and tested and then operated on a series of small extrapolations in performance.)

(6) Electronic circuitry and chambers should be tested under the operating conditions in which they will be used. If this is not possible, it should be recognized that they are being tested by the operations themselves and extra precautions, care, and alertness should be observed. Epler [41] says "failure to test under planned conditions must result ultimately in an unplanned test."

(7) The safety system should have adequate monitoring and alarms or scrams on such items as circuit continuity, proper high voltages, line voltages, etc.

(8) In general, it is better to utilize a smaller signal from an ion chamber further away from the neutron source with a suitable amplifier, than it is to overdrive an ion chamber and risk operating in a region where the chamber response is far from linear with increasing neutron flux.

3.9 The SRE Fuel Element Damage Accident [43-49]

The Sodium Reactor Experiment (SRE) was built at Santa Susana, Calif. to aid in the development of the sodium-cooled, graphite-moderated reactor concept for civilian power use. This 20 Mw(t) plant went critical in April of 1957 and first generated electrical power on July 12, 1957. As shown in Fig. 3-10 [43], the reactor core region is divided into hexagonal cells of zirconium-clad graphite 11 in. (27.94 cm) across the flats and 10 ft (3.05 m) high. The outer units serve as reflectors and the inner ones contain fuel in a central cylindrical tube 2.80 in (7.11 cm) ID. The fuel elements in the core in question (Fig. 3-11) [43] were made up of 7 rod clusters, each rod a 6 ft (1.83 m) column of 6 in. (15.2 cm) long uranium slugs in a 0.010 in. (0.25 mm) thick stainless steel tube. The 0.010 in. (0.25 mm) annulus was bonded by NaK and there was a helium-filled space above. The outer six rods were wrapped with a stainless steel helical wire to prevent rods from touching each other or the process channel. The flow of the sodium coolant is upward through the core and out to the heat exchanger.

"The design and construction philosophy of the system emphasized the use of conventional, commercially obtainable components wherever possible. To this end, the coolant circulating pumps are simple adaptations of hot oil pumps. The stuffing box (see Fig. 3-12) [43] was replaced by a cooled annulus around the shaft to freeze sodium and thus seal liquid sodium in the pump casing from the supporting bearings on the shaft and the drive motors. The primary and secondary systems each have a 6-in. and 2-in. pump and three of these have had erratic operating experience. . .The 6-in. 1500 gpm* pump in the primary system...was the main offender in the fuel-element damage incident that occurred in July 1959. Difficulties had been experienced from time to time with binding similar to that of the 2-in. pump, and, on two occasions, the auxilary coolant (Tetralin) in the freeze seal has leaked into the main sodium coolant stream. This was detected by identifying hydrocarbon vapor in the atmosphere above the reactor pool. The first of these occurrences,

*1500 gpm = 94.5 liter/sec

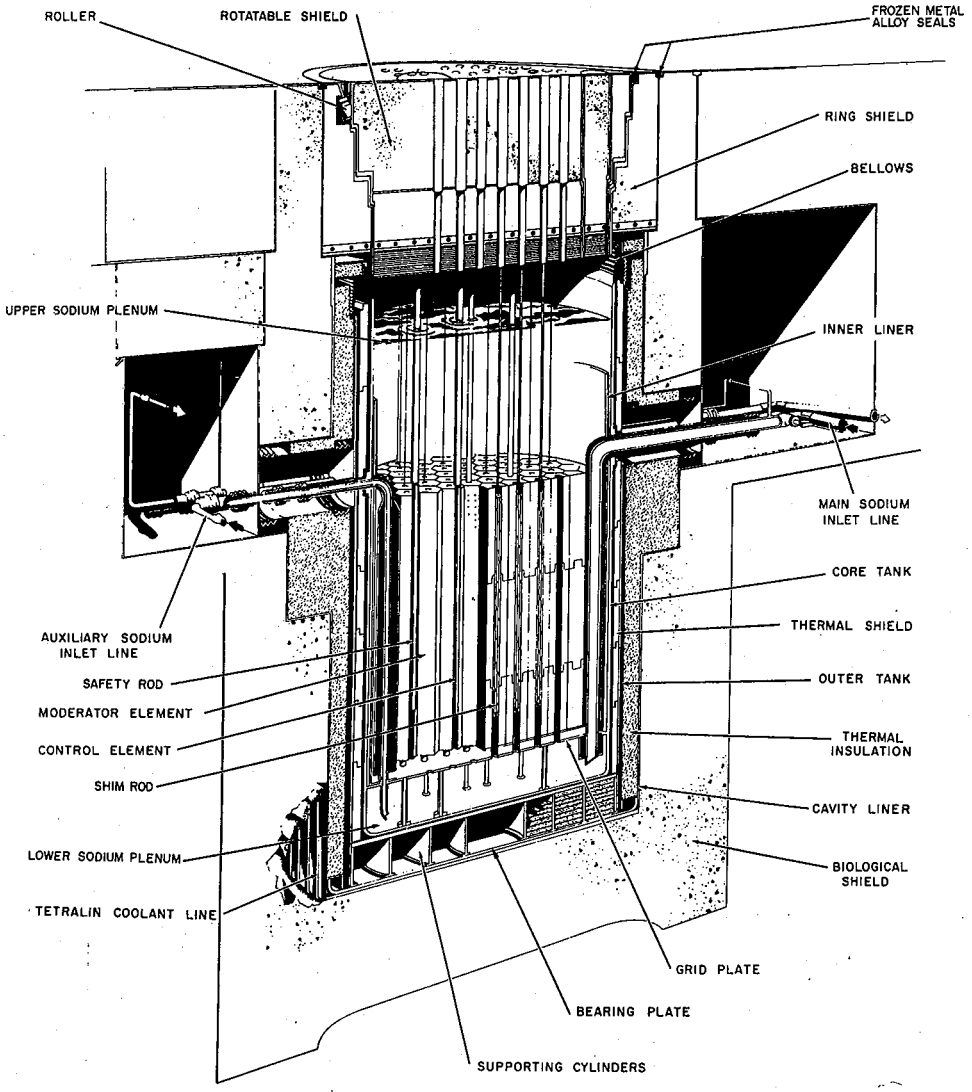

FIG. 3-10 Cross section of the SRE.

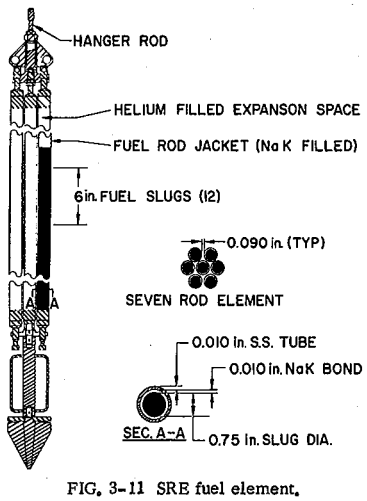

FIG. 3-11 SRE fuel element.

in April 1958, resulted in an insignificant amount of leakage before the freeze seal was repaired. This leak was caused by a pinhole in the freeze-seal casting. The second such leak occurred in May 1959 and resulted in somewhere between 2 and 10 gal of Tetralin being admitted to the primary sodium stream [44]. This leak was traced to failure of a thermocouple well. The Tetralin-cooled shaft seal was then replaced with a NaK-cooled seal arranged in such a way that two independent barriers would have to fail before mixing of NaK and the primary sodium could occur. As will be mentioned later, however, sufficient Tetralin had already been admitted to the system to create the condition that damaged the fuel assemblies [45]."

Some comments on plant performance pertinent to safety or to fission product retention taken from reference [43] are mentioned here although most of them had no direct bearing on the accident itself.

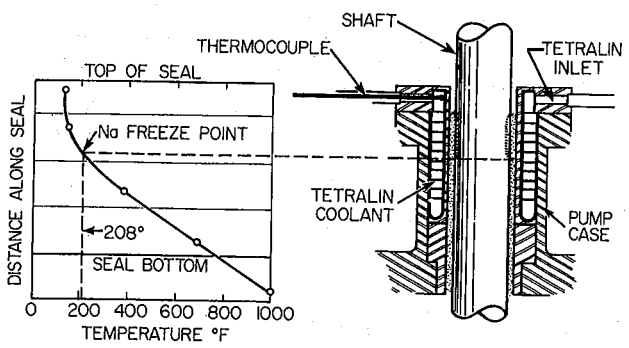

FIG. 3-12 SRE pump shaft freeze seal.

"—Unalloyed uranium metal is an unsatisfactory fuel material for a high-temperature reactor because of a tendency to swell. Fuel rods irradiated to about 1000/Mwd/ton...showed an increase in diameter of 2 to 4 mils. The...NaK annulus... had been occupied by the swelling uranium and the can slightly distended."

"Modification and maintenance of the sodium system have been accomplished with ease and safety. Piping can be cut and welded by first freezing the contained sodium. There have been no sodium fires during any operation involving cutting or welding piping containing frozen sodium. The cold traps and hot traps have performed well in removing sodium oxide; no difficulty was experienced in maintaining the oxide concentration below 10 ppm."

"The major difficulty with the instrumentation has been a series of spurious scrams caused by fluctuations in voltage from the power supply."* Reference [43] presents six pages of graphic operating and scram history for the reactor.

Since the circumstances involved in the accident extended over the entire period of time from November 29, 1958 to July 26, 1959 it seems appropriate to outline briefly the pertinent facts chronologically as is done in more detail in reference [43]. Quotations are from that reference.

Run 8 After a shutdown of about two months for repairs and modifications (involving several transfers of the sodium from the reactor to the fill tank which was known to have considerable sodium oxide), the reactor was taken to 3.6 Mw on November 29, 1958. The fuel outlet temperatures, which usually showed a spread of less than 100°F (62°C) showed much higher values (415-800°F or 259-500°C). This was attributed to high oxide content. On December 12, two elements which had been excessively hot were removed and washed. Both had black material on them. "Jiggling" by moving the element up and down one inch or less (in its position in the reactor) was found to improve heat transfer, but washing helped even more. Power was increased to 12 Mw (December 18) and 14 Mw (December 19) and the run continued to December 23. After shutdown, 15 elements were washed and more cold-trapping** was done. The run continued again from December 27 to January 29 at 20 Mw maximum power with jiggling. On January 7 a sample of the cover gas showed the presence of napthalene (and therefore Tetralin) in the system, something not suspected before—although there had been a prior leak "in June 1958."*** (It was not known if any Tetralin had entered the primary sodium system from the earlier leak caused by a crack in the bearing housing casting on the main primary pump.) The run was terminated on January 29 "because the desired exposure of 600 Mwd was attained."

Run 9 The run started on February 14 and continued until February 26. The reactor was run at 20 Mw with continuing difficulties with fuel exit temperature spread leading to shutdown and more washing and cold-trapping. "Reactor operations were resumed February 20. Examination of the records of the shim-rod positions (which was made after run 14) indicated that an increase in reactivity of 1/2% had occurred. Such an increase is expected because of the xenon decay during a shutdown of this length. However, a preliminary calculation of this effect indicated an expected increase in reactivity of 1%. It is believed that this discrepancy is due to approximations made in calculations of the xenon correction. Similar discrepancies are noted in later runs...."

"There were two reactor scrams caused by

*Frequent spurious scrams and warnings tend to dull the operators' sensitivity and to lead operators to ignore anomalous behavior. At the same time, preventing scrams is also a dangerous practice. Somewhere between these two extremes there is a reasonable level for the rate of scrams per year, but it must be judged on an individual basis, depending on the stage in the reactor operations, the type and use of the reactor, consequences of a scram, complexity of the reactor, etc.

**Cold-trapping to purify Na or NaK is discussed in the chapter on Chemical Reactions.

***The evidence would seem to indicate three Tetralin leaks - one in April, May, or June 1958 (the reports differ in the time), one in December 1958 during run 8 (no mention is made of the source of this one or its repair), and one during run 13 in May-June 1959.

excessive temperature drop across a moderator can and several scrams caused by power line transients. The reactor has a long history of scrams due to the latter effect." Run 9 "was terminated after the desired exposure of 125 Mwd was achieved." After shutdown, the fuel element in reactor channel 56 was examined. The orifice plate had a thin black deposit. The fuel element was washed and replaced in the reactor.

Runs 10, 11, 12 (March 6-7, March 16-April 6, May 14-24) These runs showed some continued improvements in the fuel exit temperature spread. "An examination of the records of shim rod position (made after run 14) shows that at the start of run 11 a loss of reactivity of 1/4% had occurred. The loss may have been due to the replacement of a thimble." In Run 11 a further series of scrams due to flow fluctuations occurred and a reactivity discrepancy of 1/3% attributed to xenon occurred again. During Run 11 the radiation level in the main sodium gallery seemed high although this was not observed until ten days after the run. At the end of this run a filter was installed in the primary system which collected considerable carbon containing material. During Run 12 a planned outlet temperature of 1065°F (574°C) was reached for 1 hour at 6 Mw power and steam was produced at 1000°F (538°C). A check after shutdown showed no measurable change in fuel dimensions.

Run 13 (May 27-June 3) Except for a sodium flow rate scram, the run was considered normal at a power of 20 Mw until 09:00, May 30. Then, several abnormalities were seen including, a slow three-day rise in inlet temperature from 545°F (285°C) to 580°F (304°C), an increase in log mean temperature difference across the intermediate heat exchanger indicating impaired heat transfer, a rise in temperature over a 20-min period from 860-945°F (460-507°C) in one fuel rod, some increases in exit fuel temperatures, a jump of 30°F (17°C) at 22:30, May 30, for moderator delta T compared to earlier 18°F (10°C) fluctuations, and one or two other temperature probe effects. In addition, "although it was not noted at the time because the reactor was on automatic control, an examination of the record of shim rod position (made after Run 14) showed that a shim rod motion corresponding to a reactivity increase of about 0.3% had occurred. This change was gradual and extended over a period of about 6 hours. Following this, the reactivity showed a steady increase of about 0.1% over the next three days of operation."

By June 2, it was obvious that the heat transfer characteristics had been impaired in the primary system and the cause was believed to be a Tetralin leak. The odor of Tetralin was detected in the pump casing of the main primary pump. The run was terminated on June 3, and after a 10 day interval to allow radioactive sodium to decay, the pump was removed. A leak was discovered in the wall of the thermocouple well of the freeze gland seal (see Fig. 3-12) where a dislodged piece of hard plating material wore through the wall as the shaft rotated.

"Seventeen fuel elements were visually examined by means of a television camera and found to be slightly dirty, but in good condition." An attempt was made to wash one element, but during the operation a pressure excursion occurred which severed the hanger (see Fig. 3-11) and lifted the shield plug out of the wash cell. It is believed that hydrocarbons from the breakdown of Tetralin could cause sodium to be trapped in the holddown tube on the hanger rod by blocking the sodium drain holes. This sodium then reacted with the wash water. "As a result of this incident, no further washing was done." It was decided to "strip" the Tetralin and organics which would volatilize by passing nitrogen gas through the sodium system. The process had been used on October 12, 1958 to remove Tetralin from the system after the first leak. The stripping began June 17 and continued until July 5. The sodium system temperature was 350°F (177°C) initially and was raised to 425°F (218°C) by the end to help the removal. In all 400,000 ft^3 (11,300 m^3) of nitrogen were used and 3 pints (1.42 liter) of Tetralin and 1500 cm^3 (91.5 $in.^3$) of napthalene crystals were removed. The system was then purged for ten hours with 4700 ft^3 (133 m^3) of helium and argon. The primary pump was reinstalled with a NaK-cooled freeze seal in place of the Tetralin-cooled seal, and the system was prepared for operation.*

Run 14 (July 12-26, 1959) The run was begun with the anticipation that the situation would be similar to that experienced in Run 8. The reactor was made critical at 06:50, July 12. At 08:35 as the reactor was slowly increasing in power to 0.5 Mw large fluctuations of 10°F (5.6°C) were noted on the moderator delta-T recorder. Normally, even at 20 Mw these were less than 5°F. The fuel exit channel temperatures started to show a spread of about 200°F (111°C). Operation continued at less than 1 Mw until 11:42 when a scram occurred due to loss of auxiliary sodium flow. Criticality was reestablished at 12:15 and operations continued at slowly increasing power levels with fuel exit channel temperatures from 510 to 770°F (265 to 410°C). Fluctuations of 30°F (17°C) in the moderator delta-T at 1.5 Mw were observed. At 15:30 reactor room air monitors showed a sharp increase in activity. The radiation level over the sodium level coil thimble in channel 7 rose to 500 mr/hr. Air filter and stack activities increased. The reactor cover gas pressure was lowered from 2 psig to less than 1 psig in an

*After the final accident during Run 14, the use of nitrogen at this point caused considerable concern about the possibility of nitriding of the stainless steel and zirconium and thus promoting fuel and moderator can failures. Tests seemed to indicate that nitriding will occur in preference to carburizing at 1200°F (649°C) in carbon-bearing sodium with a nitrogen cover gas. Apparently it will even occur after a helium purge has supposedly swept all of the nitrogen out. The nitrogen evidently is held by the carbon and calcium impurities. Measurements, coupled with the known solubility of carbon in sodium [46] showed that the system had been saturated in carbon ever since Run 8.

effort to reduce the level. By 17:00 the radiation level over core channel 7 reached 25 r/hr. Accordingly, at 17:30 power reduction began, at 20:57 the reactor was shutdown, and the sodium probe was removed from channel 7 and replaced by a shield plug.

The reactor was brought to criticality at 04:40 July 13 with exit fuel temperature scram setpoints lowered to 800°F (427°C). At 13:30 it was observed that the moderator delta-T followed a rise in the sodium outlet temperature and that the moderator temperature did not respond properly to an increase in sodium flow. It was concluded that little sodium was leaking across the grid plate for moderator coolant. At 17:28 a planned increase in reactor power from 1.6 Mw began in order to deliver heat to the electrical substation. "At the start the power level persisted in rising somewhat faster than expected even though control rods were being slowly inserted in an attempt to hold it back." (The solid curve in Fig. 3-13 [43] indicates the course of the power trace during this time.) By 18:07 with the power at 2 Mw, a negative period of about 45 sec was observed and power fell to 2.4 Mw in about 3 min. Control rod withdrawal was started, the reactor was critical at 18:11 and power rose to 3.0 Mw by 18:21. Then, as the power was increasing more rapidly, rod insertion began, but, in spite of this, power continued to rise. At about 18:24 three positive transients were observed with about 50-sec periods and at 18:25 a 7-1/2 sec period was indicated. The reactor should have started an automatic power set-back at a 10-sec period, but did not, and the operator scrammed it manually. The automatic electronic period scram did not act, as it was set for a five second period. The peak power indicated was 24 Mw.* (No particularly high temperatures were recorded during the transient.)

Later examination of the period set-back mechanism (a mechanical actuation by means of a cam in the period recorder) showed that it worked properly only if the period decreased at a slow rate, but would not operate if the period decreased rapidly.

"Recovery from the scram was made cautiously. Criticality was attained at 19:55. Approximately 2-1/2 hours after the scram, reactor power reached 2.0 Mw." Rod positions were now 52 in. (132.1 cm) out rather than 49.5 in. (125.7 cm) as before the scram, but the difference was, at the time, attributed to xenon. The rods returned to 50.5 in. (128.3 cm) by 02:00 at a power of 4.0 Mw. It was decided that the excursion had not affected the reactor adversely. Operations continued until 13:00 when a scram was caused by a short-circuit introduced into the demand circuit for the primary pump being prepared for a flow oscillation test. The reactor was made critical quite rapidly (13:11) and operations continued.

*The final report [43a] corrects this value to about 14 Mw, noting that "The 24 Mw value was obtained by a linear extrapolation of the log N recorder chart from power levels of about 2 Mw, and linear extrapolation is not valid."

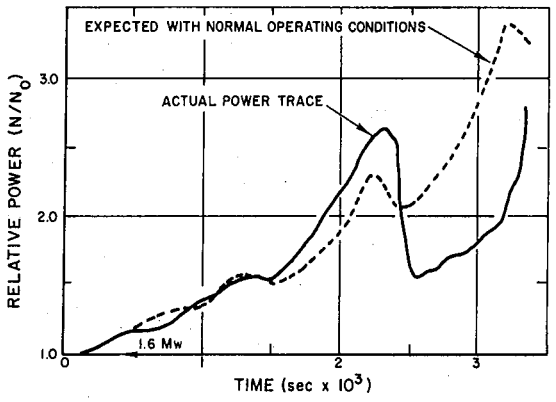

FIG. 3-13 Machine calculation I of the SRE power excursion. Zero time is 17:28 on July 13, 1959.

It was decided to pressurize and vent the reactor atmosphere once to reduce the radioactivity caused by the xenon in the cover gas. At 05:50 July 15 the pressure was reduced from 1.8 psig (1.12 atm) to 0.6 psig (1.04 atm), repressurized to 3.0 psig (1.20 atm), and then reduced to 1 psig (1.067 atm). Lowering the core pressure caused an increase of about 0.01% in reactivity and raising the pressure had the reverse effect. This is not a normal effect on this reactor. "Operation was continued at a power level of approximately 3 Mw."

A review of fuel exit temperature spread on July 15 showed that it would be useless to try "to get the Edison turbine generator 'on the line' since the maximum power level attainable would probably be less than 4 Mw." This would not permit operation at the desired high inlet temperatures while circulating through the steam generator. Alterations were made and on July 16 at 07:04 the reactor achieved criticality at a rod position showing a substantial loss of criticality since the beginning of the run. Intermittent operation continued at less than 2 Mw until July 20. Several tests on pressure effects, plugging temperatures, and sodium level variations were carried out. On July 18, the motor-generator set which supplies the vital bus stabilized power failed and operations were resumed with the unstabilized Edison supply. On July 20, the reactor power was increased to 2.5 Mw to raise loop temperatures gradually to 700°F (371°C). On July 21 at 02:10 a scram was caused by a fast period indication. (Apparently it was attributed to unstabilized power.) The reactor was critical again at 02:25. At 06:45 radioactivity in the reactor began to build up. At 09:45 flow was lost in the main secondary loop causing a scram. The secondary loop was restored to service, the vital bus put back on the repaired motor-generator set, and operations continued at power levels up to 4.5 Mw, sodium flow rates up to 1500 gpm (94.5 liter/sec), and reactor outlet temperatures to 790°F (421°C).

On July 23, it was decided to shut the reactor down and it was scheduled for 17:00 July 24. Until 09:00 July 23, reactor outlet temperatures were kept between 700 and 800°F (371 to 427°C) although a few reached the 900 to 1000°F (482 to 538°C)

)

power excursion. Zero

and vent the re-
the radioactivity
ver gas. At 05:50
ed from 1.8 psig
a), repressurized
reduced to 1 psig
pressure caused
ctivity and raising
ffect. This is not
. "Operation was
oximately 3 Mw."
rature spread on
be useless to try
ator 'on the line'
attainable would
This would not
high inlet tem-
gh the steam gen-
and on July 16 at
icality at a rod
oss of criticality
termittent opera-
Mw until July 20.
s, plugging tem-
ions were carried
erator set which
ed power failed
the unstabilized
reactor power
e loop tempera-
C). On July 21
by a fast period
attributed to un-
as critical again
n the reactor be-
lost in the main
. The secondary
he vital bus put
erator set, and
ls up to 4.5 Mw,
(94.5 liter/sec),
790°F (421°C).
shut the reactor
00 July 24. Until
peratures were
427°C) although
(482 to 538°C)

range. At 09:50 July 23, a reactor scram was caused by a fast period indication. It was attributed to an electrical transient and the reactor was critical again at 10:15.

Between 00:00 and 08:00 on July 24 it was noted (while jiggling elements to dislodge foreign material) that the elements in channels 10, 12, 35, and 76 were stuck while it was known 10 was free on July 22. A scram was caused by a fast period indication at 12:50 on July 24 and was attributed to an instrument transient and the reactor was made critical at 13:14. Accidental loss of auxiliary primary flow caused a reactor scram at 15:40. The reactor was critical again at 15:56. Cold-trapping was put back in service when the outlet temperature reached 510°F (266°C) and the primary plugging temperature gradually dropped from 455 to 350°F (230 to 177°C) within about seven hours.

On July 26 it was noted that the fuel in channels 12 and 35 were no longer stuck and 76 was somewhat free, but 10 was still stuck. The reactor was finally shut down on July 26 at 11:20 after logging 16 Mwd in Run 14. Post-run examination of the core showed that 10 of the 43 assemblies in the core had undergone severe melting of the cladding. The top and bottom halves of these ten elements were separated. The zone of failure was between one-third and two-thirds of the length of the element measured from the top. The accident showed that iodine released from the elements was very effectively retained in the sodium coolant. In fact, no activity except the noble gases was detected in the cover gases [45].

In reference [47] Fillmore has considered the transient which occurred on July 13 in detail. The general features of the transient include a slow but steady rise in power partially compensated for by control rod insertion, a sharp drop in power, followed by another short interval of slow but steady rise, and a final fast rise terminated by scram. These features are shown by the solid line in Figs. 3-13 and 3-14 [43]. On these graphs t = 0 at 17:28 hours, July 13, 1959. The dotted line of Fig. 3-13 shows the course the power trace would have been expected to take if the normal reactivity coefficients and control rod worths were in effect. These calculations were made using the AIRER IBM Code.

By introducing seven ramp and step changes in reactivity, Fillmore was able to get the agreement shown in Fig. 3-14. The general features can then be explained at least semiquantitatively. The slow but steady rise at a rate of +0.04% in a 3 minute ramp in spite of gradual control rod insertion and the negative Doppler effect is attributed to an abnormal rise of the temperature of the moderator which has a reactivity coefficient of $+1.7 \times 10^{-5}/°F$ ($+3.1 \times 10^{-5}/°C$) and perhaps also to some sodium vapor formation in partially plugged channels. (The sodium void coefficient is positive although the fuel temperature coefficient was considered to be $-1.1 \times 10^{-5}/°F$ or $-2.5 \times 10^{-5}/°C$.) The fast negative excursion is attributed to rod insertion reducing the increase in power rate and causing the collapse of sodium bubbles and void regions, thus improving coolant contact with the fuel. This cooling in turn reduced fuel temperature and caused a gain of reactivity because of the Doppler effect, and perhaps sodium void collapse, so that the control rods needed only to be withdrawn a little to again start the reactor up on a slow steady rise. This was followed by a fast transient which added +0.3% in a 5 to 10-sec ramp. It is postulated that this was caused by the more or less simultaneous voiding of about 10 partially plugged fuel elements. Study of the damaged fuel elements [48] seems to show that thermal cycling occurred at temperatures above the α-β phase transition of uranium which would lead to fuel ruptures and also steel-uranium eutectic formation. It is therefore postulated that several channels underwent one or more cycles of heating and sodium vapor formation followed by void collapse. The cycles in the various elements

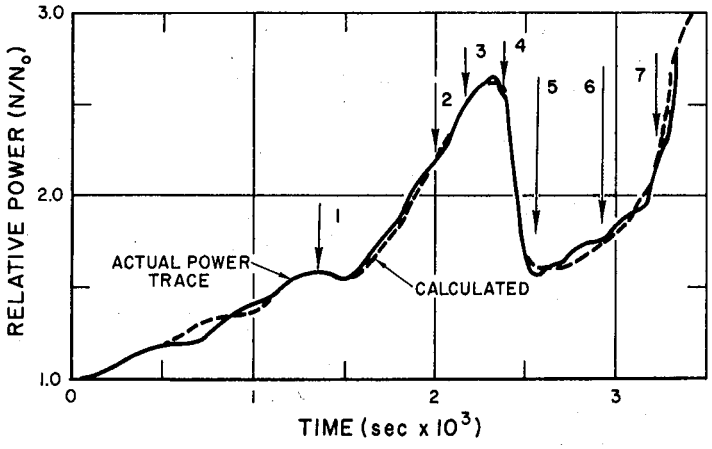

FIG. 3-14 Machine calculation III of the SRE power excursion. Numbers indicate the following sequence: (1) start ramp $\Delta\rho/\Delta t$ of $+ 0.11 \times 10^{-4}$%/sec, thus introducing total $\Delta\rho$ of + 0.0077%; (2) hold first ramp constant at + 0.007% and start second ramp $\Delta\rho/\Delta t$ of $- 0.3 \times 10^{-4}$%/sec; (3) hold first two ramps constant, start third ramp of $+ 0.1 \times 10^{-3}$/sec, thus introducing total $\Delta\rho$ of + 0.02%; (4) hold all three ramps constant (at + 0.0077%, - 0.006% and + 0.02%, respectively), introduce step $\Delta\rho = - 0.06$%; (5) introduce step $\Delta\rho = + 0.02$%; (6) introduce step $\Delta\rho = - 0.005$%; (7) introduce step $\Delta\rho = + 0.03$%. Zero time 17:28 on July 13, 1959.

are postulated as acting independently at first with periods of the order of 2 minutes. But the rise in power to 5 Mw was sufficient so that "all cycling channels were affected" and voiding in all occurred more or less simultaneously. Further, one can postulate that there was a reactivity interaction between channels and that voiding in one led to heating and voiding in others and so on.*

Comments, Conclusions, Recommendations

(1) Systems should be designed utilizing compatible components, materials, and fluids in all possible cases. In this particular case, the choice of Tetralin for cooling a bearing which could leak into the sodium system was a primary cause for the accident. If this is not possible, then "Where materials are potentially incompatible under normal service conditions, special means must be provided to separate them. Where there are compatibility changes with abnormal operating conditions, e.g. temperature, special attention is required for monitoring and controlling such conditions." [48a]

(2) At the time the choice of fuel had to be made for this reactor the knowledge of fuel performance was not so good as it is now and there were not so many choices. It was known, however, that uranium metal had an α-β phase transition with accompanying swelling at about 1220°F (660°C) and that uranium formed an eutectic with stainless steel at about 1340°F (725°C). Thus, while the use of a stainless steel-uranium metal element may have been the only choice, nevertheless phase transition and eutectic formations greatly lowered the temperature at which fuel element failure and melting occurred. Designers should always try to select materials and combinations of compatible materials that will stand up as well as possible under abnormal conditions. During the planned 1065°F (574°C) run (Run 12), the central fuel element temperatures must have been quite close to the α-β phase transition point in a number of elements.

(3) It is not good practice to use, even as temporary additives, materials or gases which may remain in the system and cause deleterious effects later. Thus, the use of nitrogen as a stripping agent for Tetralin led to a concern over nitriding later. The total cost of the time spent and the tests run was probably greater than if a chemically inert noble gas had been used in the first place.

(4) There were several instrumentation problems. The principal one involving a period setback has been described. The practice of incorporating such an important item of equipment as a period setback as a part of a recorder is questionable. Recorders require easy access for maintenance and ink-refilling, and often have very poor safety characteristics. For instance, if the slidewire breaks, the recorder goes to one extreme of its travel. This can lead to an unsafe situation, (especially if the recorder is being used to drive an automatic control unit for the reactor—as happened in one case). In addition, there was evidently much to be desired in the normal behavior of the system. Reference [43] mentions several times the fact that "the reactor has a history of spurious scrams due to apparent period transients". This problem is often encountered during the initial tests of a reactor and is often due to either improper grounding of electrical components, electrical noise from other equipment on the same power line, or power surges, or all three. Usually it can be eliminated in a short time without sacrificing period trip sensitivity. This instrumentation system had been used since 1957 and by 1959 the problem should have been solved. The fact that the operators are described as achieving criticality after scrams in time intervals of 15 to 25 minutes would seem to indicate that no thorough investigation of possible causes could have been made. In fact, if Fillmore's explanation of the fast transient effect is correct, there may be some reason to believe that some of the several period scrams which occurred in Run 14 were genuine, although "No evidence was uncovered to indicate that there were indeed genuine scrams other than during the transient at 18:25 on July 13" [48a].

McDonald and DeVan [49] give the opinion that, "the reactor instrumentation under the immediate surveillance of the operators was inadequate to indicate excessive fuel element temperatures, the blocking of coolant passages, and fission product leakage. As a result, the operators did not consider such indications (where they existed) serious enough to warrant shutting down the reactor. Since the SRE is a 'developmental facility built to investigate fuel materials' it would appear that additional instrumentation, as well as closer technical management, might have reduced the damage to the SRE core."

(5) From the evidence available in this accident it would appear that fission products other than the noble gases are retained well in sodium. This, in a sense, affords an additional safeguard unless the sodium should then become exposed to air in such a way as to become a fire hazard.

(6) The circumstances which eventually led to this accident began as early as spring, 1958, when the first Tetralin leak occurred. A second leak occurred in Run 8 on November 29, 1958, and problems continued until July 24, 1959. During that time so many difficulties were encountered that, at least in retrospect, it is quite clear that the reactor should have been shut down and the problems solved properly. Continuing to run in the face of a known Tetralin leak, repeated scrams, equipment failures, rising radioactivity releases, and unexplained transient effects is difficult to justify. Such emphasis on continued operation can and often does have serious effects on safety and can create an atmosphere leading to serious accidents. It is dangerous, as well as being false economy, to run a reactor that clearly is not functioning as it was designed to function.

*This description, if correct, illustrates the type of reactivity effect which can be very serious since the mechanism depends upon a positive internal feedback, increasing without limit, until terminated by disassembly - unless terminated first by an external means such as control rods.

r instance, if the goes to one ex- lead to an unsafe 'der is being used ; for the reactor— ldition, there was ı the normal be- ce [43] mentions he reactor has a o apparent period often encountered or and is often due ' electrical com- her equipment on r surges, or all nated in a short trip sensitivity. l been used since should have been ors are described scrams in time ould seem to in- ation of possible act, if Fillmore's effect is correct, elieve that some hich occurred in 'No evidence was re indeed genuine ısient at 18:25 on

the opinion that, er the immediate as inadequate to emperatures, the l fission product ors did not con- existed) serious wn the reactor. tal facility built ould appear that l as closer tech- uced the damage

e in this accident ducts other than in sodium. This, safeguard unless exposed to air hazard. eventually led to s spring, 1958, rred. A second ember 29, 1958, 24, 1959. During ere encountered is quite clear ı shut down and ıtinuing to run in epeated scrams, ctivity releases, ı is difficult to ed operation can ts on safety and to serious ac- as being false clearly is not nction.

In the long run, reactor economics as well as reactor safety will demand adequate continuing maintenance at all times, and this will include early shutdown and proper action whenever there is the least doubt concerning the situation. Management can and must establish this sort of attitude.

3.10 Westinghouse Test Reactor Accident [50-53]

The Westinghouse Test Reactor is a light-water-cooled and -moderated reactor utilizing highly enriched uranium fuel. The fuel element consists of three cylindrical layers surrounding a central thimble. Each layer is made up of 0.052 in. (1.32 mm) of aluminum-uranium alloy, clad with 0.0365 in. (0.927 mm) of aluminum on either side. At the center of each fuel element is a thimble which permits the insertion of a sample. Unless a special experiment is involved, the central sections are loaded with either aluminum or cobalt slugs. The use of these aluminum and cobalt slugs helps with the shimming of the reactor for control purposes.

A diagram of the core is shown in Fig. 3-15 [50]. The nine control rod positions are indicated by the black circles. Each control rod consists of an aluminum-clad cadmium cylinder with a normal fuel element follower. Thus, on the full withdrawal of a control rod from the core a regular fuel element containing 199 g of U^{235} is left in that position. In Fig. 3-15 the number before the dash indicates the top to bottom row starting at the top of the diagram, and the number after the dash indicates the position numbered from left to right in any given row. The number beneath the line indicates the number of grams of U^{235} estimated to be present at the time of the accident. Each of the fuel elements originally contained 199 g of U^{235}. The core has a 36 in. (0.914 m) active height and a main active diameter of approximately 28 in. (0.711 m) with an aluminum-water ratio of 1. At the original licensed power of 20 Mw the average thermal neutron flux in the core was approximately 5.2×10^{13} neutrons/cm^2-sec.

The fuel element that failed (Position 6-5) contained a special experiment designed to monitor fast neutron fluxes. The experiment consisted of a set of 7 hairlike nickel wires, each separately encapsulated in a quartz capsule and held in recesses of a 3/8 in. (9.52 mm) diameter aluminum rod. The rod in turn was encased in a 1/2 in. (12.7 mm) OD aluminum tube with 1/8 in. (3.18 mm) weep holes drilled through the wall at 4 in. (10.16 cm) intervals. This assembly was placed in the central thimble of the fuel element in the normal manner. A flow orifice at the bottom of the thimble section limited the flow to that required for cooling this rod. Other experiments were located in Positions 5-6, 7-3, 7-6, 8-5 and 11-1. Other fuel elements contained aluminum and cobalt, as indicated in Fig. 3-15.

The coolant flow is down through the core and up through the thermal shield. The design specifications required a flow of 15,000 gpm (950 liter/sec) at 20 Mw with a core outlet pressure of 93 psia

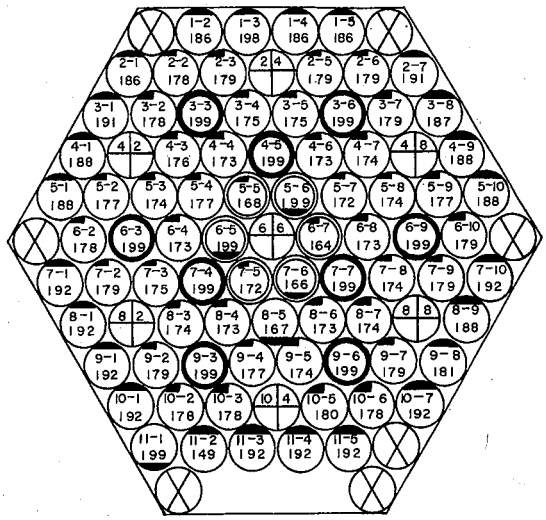

FIG. 3-15 WTR core cross section. Normal circles indicate fuel elements, heavy circles (e.g., 3-3, 3-6, etc.) indicate control rods. Circles enclosing an X are plugged, those enclosing a + are high-pressure thimbles. The contents of the irradiation volumes are denoted by darkened areas at top and bottom of individual fuel elements: black strip across top = standard V-basket with cobalt, black half-strip across top = solid aluminum V-basket, black half-strip at bottom = experiment. Upper numbers indicate position, lower numbers denote number of grams of U^{235} estimated to be present at the time of the accident. (Note: For simplification letter prefixes used in reference [50] to identify positions have been omitted in the above figure.)

(6.4 atm) and a core pressure drop of 15.5 psia (1.05 atm). The core inlet and outlet temperatures were 115 and 125°F (46.1 and 51.7°C) respectively. The normal surface heat flux under these conditions was estimated to be 1.13×10^5 Btu/hr-ft^2 (35.6 watt/cm^2) with a flow velocity of 21.3 ft/sec (6.50 m/sec). These numbers were calculated for 57 fuel assemblies in the core. At the time of the accident there were 78 fuel assemblies in the core including control rods.

The site ventilating systems, the various waste collection tanks, the evaporator-condenser tanks, and several other components all vent their noncondensable gases to a 100 ft (30.5 m) vent stack located between the Process and Reactor Service Buildings.

At the time of the accident the reactor staff was engaged in a series of tests to determine the feasibility of increasing the power gradually to 60 Mw(t). Reference [50] provides a detailed chronology of events. The quotations in the following abbreviated account are taken from there. According to the license amendment of January 8, 1960, to permit operation at a maximum of 60 Mw(t), certain restrictions were imposed. These restrictions were [50]

"1. Westinghouse shall retain the bubble formation apparatus and the special detection channel described in the application in the reactor during the power escalation program until stable operation at 60 Mw(t) power level has been established;
2. The ratio of the maximum heat flux in the reactor to the burnout heat flux shall never exceed one-half;
3. The reactor shall not be operated in such

torily limit large excursions in operating reactors of this type before fuel vaporization can occur. Then, as with the plate-type elements, the sudden vaporization and expansion of a small fraction of the fuel to burst its cladding should trigger the interaction of other melted and vaporized fuel elements and create a void in much the same fashion that happened in the three tests to date.

It is of interest to note that apparently the bursting of two pins in the SPERT oxide core test of April 1964 (Sec. 3.12) had a beneficial effect on the rate of shutdown. This seems to indicate that a material with a high vapor pressure like water in direct contact with the fissile material acts promptly to supply negative reactivity by pin bursting and boiling. Perhaps the mechanism could be used to augment or replace the Doppler effect in certain reactors. While water within the cladding would be a difficulty in an oxide-fueled power reactor with its high fuel temperature, nonetheless another material might be found which could be placed in some fuel rods to enhance their bursting and shutdown effect in order to terminate a serious transient more rapidly.

TREAT tests of fuel vaporization and bursting should provide the information to predict the uranium oxide temperatures necessary to give vapor pressures high enough to burst the fuel. (No doubt, the fission products built up during core life will have an effect on the results of these tests.) SPERT transient tests on cold cores and operating cores at temperature and pressure should provide the necessary full core transient information. Plotting the maximum fuel temperatures reached in the transients as a function of α_0 should give a curve that could then be extrapolated to find the α_0 required to give the necessary fuel temperature (really total energy deposition) to achieve the vapor pressures in the fuel to burst the cladding. This should be the energy-plateau limited transient for this type of reactor, if there is such a limit. The transient tests will also give the relation between the reactivity inserted and the total energy deposited or central fuel temperatures reached. The picture should then be complete and on this basis it may be possible to say something also about the limiting accident and about reactivity in this type of reactor. The oxide will, of course, interact with water quite differently from the metal as discussed in the chapter on Chemical Reactions. Such tests as those outlined above, particularly if they eventually lead to destructive tests, should give much insight into the importance of the metal-water reaction in the destructive transients observed to date.

Normally the cores for this type of reactor are very large. It is thus possible to deposit in them relatively large amounts of energy without serious effects. Thus it is not the total Mw-sec deposited in a transient that is important, but the energy density that is important, the Mw-sec/cm^3. Reliable central fuel temperature measurements are then probably the best indication of energy density deposited in a transient.

Other systems such as liquid-metal-cooled fast reactors may undergo similar violent voiding reactions between the coolant liquid metal and vaporized fuel. Whether the effect will result in a positive or negative reactivity change will depend on the reactor in question. Conceivably, if the void effect gives a positive reactivity change, the violent reaction described could for an instant during the interaction be made even worse by it. (These considerations are discussed in the chapter on Fast Reactor Kinetics.) These effects should be investigated in such reactors as well as in water-cooled reactors.

Even if the energy limit considered in this section should be proved not to exist, it is necessary to continue to search for such limiting processes or ways of putting limits on conceivable reactor transients. The present discussion illustrates the type of limit that is sought and the methods by which such limits could be either proved or disproved.

REFERENCES

1. H. B. Smets, "Review of Nuclear Incidents", Progress in Nuclear Energy, Series X, Vol. 3, Law and Administration, Pergamon Press, N.Y. and London, 1962. See also: H. B. Smets, "Review of the Nuclear Reactor Accidents", Vol. 1 p. 89 of Proceedings of a Symposium on Reactor Safety and Hazards Evaluation Techniques, International Atomic Energy Agency, Vienna, 1962. (Identical but limited to reactors.)
2. H. C. Paxton, "Critical assembly booby traps", Nucleonics, 16, 3(1958)80.
3a. W. R. Stratton, "A Review of Criticality Accidents", Progress in Nuclear Energy, Series IV, Vol. 3, Technology, Engineering and Safety, Pergamon Press, N.Y. and London, 1960.
3b. W. R. Stratton, "A Review of Criticality Incidents", Proceedings of a Symposium on Criticality Control in Chemical and Metallurgical Plants, OECD, Paris, 1961.
4a. H. C. Paxton (Ed.), "Fast-neutron critical assemblies and related topics", Nucl. Sci. Eng., 8(1960)523.
4b. P. H. Pitkanen, "Hazards of fast-burst facilities", Nucl. Safety, 4, 1(1962)18.
5. D. Bretin, F. Girard, and J. Charles, "L'Incident Servenu a la Pile Alize, le 15 Mars 1960", Report SEC-90, September 1960.
6. E. P. Epler, "Failure of Alize I reactor safety system", Nucl. Safety, 5, 2(1964)172.
7. D. Callihan, "Criticality Excursion of Nov. 10, 1961", USAEC Report ORNL-TM-139, Oak Ridge National Laboratory, 1962.
8. E. R. Rohrer et al., "New Criticality Excursion Machines", USAEC Report ORNL-3193, p. 168, Oak Ridge National Laboratory, 1961.
9. J. R. Buchanan, "ORNL criticality excursion", Nucl. Safety, 3, 4(1962)95.
10. D. Callihan, "Accidental nuclear excursion in Recuplex operation at Hanford, April 1962", Nucl. Safety, 4, 4(1963)136.
11. H. Paxton et al., "Nuclear Criticality Accident - LASL December 30, 1958", USAEC Report LAMS-2293, Los Alamos Scientific Laboratory, 1959.
12a. Private communication, AEC Division of Operational Safety, October 9, 1963.
12b. R. L. Kathren, W. C. Day, D. H. Denham, and J. L. Brown, "Health physics following a nuclear excursion: The LRL incident of March 26, 1963", Health Physics, 10(1964)183.
13. L. P. D. King, "Design and Description of Water Boiler Reactors", Proceedings of the First U.N. International Conference on Peaceful Uses of Atomic Energy, Geneva, 1955, Vol. 2, p. 372.
14. D. F. Hayes, "A Summary of Accidents and Incidents Involving Radiation in Atomic Energy Activities, June 1945-December 1955", USAEC Report TID-5360, 1956. (Supplements were issued in Aug. 1957, Sept. 1959, Dec. 1961.)
15. D. G. Hurst and A. G. Ward, "Canadian Research Reactors", Progress in Nuclear Energy, Series II, Reactors, Pergamon Press, N.Y. and London, 1956.
16a. W. B. Lewis, "The Accident to the NRX Reactor on December 12, 1952", Canadian Report AECL-232, Atomic Energy of Canada, Ltd., 1953.
16b. D. G. Hurst, "The Accident to the NRX Reactor. Part II", Canadian Report AECL-233, Atomic Energy of Canada, Ltd., 1953.

17. W. J. Henderson, A. C. Johnson, and P. R. Tunnicliffe, "An Investigation of Some of the Circumstances Pertinent to the Accident to the NRX Reactor of December 12, 1952", Canadian Report NEI-26, Atomic Energy of Canada, Ltd, Chalk River, 1953.
18. G. W. Hatfield, "A reactor emergency with resulting improvements", Mech. Eng., 77(1955)124.
19. J. R. Dietrich, "Experimental Investigation of the Self-Limitation of Power Driving Reactivity Transients in a Subcooled, Water-Moderated Reactor", USAEC Report ANL-5323, Argonne National Laboratory, 1954.
20. "Reactors", Nucleonics, 13, 9(1955)40.
21. J. R. Dietrich, "Experimental Determinations of the Self-Regulation and Safety of Operating Water-Moderated Reactors", Proceedings of the First U.N. International Conference on Peaceful Uses of Atomic Energy, Geneva, 1955, Vol. 13, p. 88.
22. J. R. Dietrich and D. C. Hayman, "Transient and Steady State Characteristics of a Boiling Reactor", USAEC Report ANL-5211, Argonne National Laboratory, 1954.
23. J. R. Buchanan, "Hanford reactor incidents", Nucl. Safety, 4, 1(1962)103.
24. W. E. Unbehaun, "History and Status of the EBR", USAEC Report AECD-3714, Chicago Operations Office, 1953.
25a. H. V. Lichtenberger et al., "Operation Experience and Experimental Results Obtained from a NaK-Cooled Fast Reactor", Proceedings of the First U.N. International Conference on Peaceful Uses of Atomic Energy, Geneva, 1955, Vol. 3, p. 345.
25b. H. V. Lichtenberger, W. H. Zinn, private communication, April 1964
26. F. W. Thalgott et al., "Stability Studies of the EBR-I", Proceedings of the Second U.N. International Conference on Peaceful Uses of Atomic Energy, Geneva, 1958, Vol. 12, p. 242.
27. R. R. Smith, R. G. Matlock, F. C. McGinnis, M. Novice, and F. W. Thalgott, "An Analysis of the Stability of EBR-I, Marks I to III", Proceedings of the Seminar on the Physics of Fast Reactors held at Vienna, August 3-11, 1961, International Atomic Energy Agency, 1962, Vol. III, p. 43.
28. R. O. Brittan, "Analysis of the EBR-I Core Meltdown", Proceedings of the Second U.N. International Conference on Peaceful Uses of Atomic Energy, Geneva, 1958, Vol. 12, p. 267.
29. W. H. Zinn, "A letter on EBR-I fuel meltdown", Nucleonics, 14, 6(1956)35.
30. J. H. Kittel et al., "The EBR-I Meltdown - Physical and Metallurgical Changes in the Core", USAEC Report ANL-5731, Argonne National Laboratory, 1957.
31. F. W. Thalgott, "Fast Reactor Safety: EBR-I", Proceedings of a Conference on Fast Reactors, November 20-21, 1957, USAEC Report TID-7548.
32. L. A. Mann, "EBR-I operating experience", Nucl. Safety, 3, 2(1961)68.
33. J. R. Dietrich, "EBR Oscillations", USAEC Report TID-5351, 1956 (Decl. 1957).
34. G. H. Kinchin, "The Instability of Fast Reactors", British Report RP/M83, Harwell, Berks., England, 1956.
35. H. A. Bethe, "Reactor Safety and Oscillator Tests", Report APDA-117, Atomic Power Development Associates, 1956.
36. J. C. Haire and G. O. Bright (Eds.), "Proceedings of a Conference on Reactor Kinetics held at Sun Valley, Idaho, October 12-14, 1960, Report IDO-16791, p. 309, Phillips Petroleum Co., 1962.
37. "Accident at Windscale No. 1 Pile on 10th October, 1957", Presented to Parliament by the Prime Minister by Command of Her Majesty, British Report Cmnd. 302, Her Majesty's Stationery Office, London, November 1957.
38. "Accident at Windscale No. 1 pile on October 10, 1957", Nucleonics, 15, 12(1957)43.
39. J. Metera, "On the Course and Cause of the Reactor Damage at Windscale", Nukleonika, III, 2(1958)84. Trans. from Polish, USAEC Report AEC-tr-4145, 1961.
40. H. J. Dunster, H. Howells, and W. L. Templeton, "District Surveys Following the Windscale Incident, Oct. 1957", Proceedings of the Second U. N. International Conference on Peaceful Uses of Atomic Energy, Geneva, 1958, Vol. 18, p. 296.
41. E. P. Epler, "HTRE-3 Excursion", Nucl. Safety, 3, 2(1959)57.
42. AEC Division of Compliance - private communication.
43. A. A. Jarrett, "SRE Fuel Element Damage", Report NAA-SR-4488, North American Aviation, Inc., 1959.
43a. R. L. Ashley, et al., "SRE Fuel Element Damage, Final Report" Report NAA-SR-4488 (Supplement), North American Aviation, Inc., 1961.
44. R. W. Dickinson, "Coolant block damages SRE fuel!", Nucleonics, 18, 1(1960)107.
45. R. W. Dickinson, "SRE operating experience", Nucl. Safety, 2, 3(1960)54.
46. J. G. Gratton, "Solubility of Carbon in Sodium at Elevated Temperatures", Report KAPL-1807, Knolls Atomic Power Laboratory, 1957.
47. F. A. Fillmore, "Analysis of SRE Power Excursion of July 13, 1959", Report NAA-SR-5898, North American Aviation, Inc., 1961.
48. J. L. Baliff, "Examination of Damaged SRE Elements", Report NAA-SR-4515, North American Aviation, Inc., 1961.
48a. S. Siegel, private communication, May 1, 1964.
49. W. B. McDonald and J. H. Devan, "Sodium reactor experiment incident", Nucl. Safety, 1, 3(1960)73.
50. -Staff, WTR, "Report on WTR Fuel Element Failure, April 3, 1960", Report WTR-49, Westinghouse Electric Corp., 1960.
51. R. B. Korsmeyer, "Westinghouse Testing Reactor incident", Nucl. Safety, 2, 2(1960)70.
52. R. J. Catlin, "Health Physics Aspect of the Westinghouse Testing Reactor Fuel Element Meltdown", Report WTR-55, Westinghouse Electric Corp., 1960.
53. A. N. Tardiff, "Some Aspects of the WTR and SL-1 Accidents", Proceedings of the Symposium on Reactor Safety and Hazards Evaluation Techniques Sponsored by the International Atomic Energy Agency, Vienna, May 14-18, 1962, Vol. 1, p. 43. See also: IDO19308.
54. A. Smaardy (Ed.), "Hazard Summary Report on the Argonne Low Power Reactor (SPR)", USAEC Report ANL-5744, Argonne National Laboratory, 1957.
55. D. H. Shaftman, "Zero Power Experiments on the Argonne Low Power Reactor (ALPR)", USAEC Report ANL-6078, Argonne National Laboratory, 1961.
56. D. H. Shaftman, "Proceedings of the ANPP Reactor Analysis Seminar", The Martin Col, Baltimore, Maryland, October 11, 1960.
57. C. A. Nelson, et al., "Report on the SL-1 Accident, January 3, 1961", General Manager's Board of Investigation", USAEC Information Release No. 3-326, September 24, 1962. See also: "SL-1 Recovery Operations, January 3-May 20, 1961", Report IDO-19301, (CEND 1007), Combustion Engineering, Inc., Idaho Falls, Idaho, 1961.
58. C. W. Luke and H. Cahn, "Evaluation of the Loss of Boron in the SL-1 Core I", Report CEND-1005, Combustion Engineering, Inc., 1960.
59. "Remarks by Dr. Frank K. Pittman, Director, Division of Reactor Development, USAEC", for presentation to representatives of industry, AEC Press Release, Germantown, Maryland, January 24, 1961.
60. "IDO Report on the Nuclear Incident at the SL-1 Reactor, January 3, 1961, at the National Reactor Testing Station", USAEC Report IDO-19302, Idaho Operations Office, 1962.
61. "Final Report of SL-1 Recovery Operation, May 1961 through July 1962", Report IDO-19311, General Electric Co., Aircraft Nuclear Propulsion Dept., 1962.
62. "Additional Analysis of the SL-1 Excursion - Final Report of Progress, July through October 1962", Report IDO-19313, General Electric Co., Aircraft Nuclear Propulsion Dept., 1962.
63. V. E. Schrock et al., University of California, Berkeley, Series 163, No. 2, January 1961. See also: S. J. Green, "Estimated Film Boiling Heat Transfer Coefficients at Burnout", Report WAPD-TH-132, Bettis, Atomic Power Laboratory, 1955. (OR: H. A. Johnson, V. E. Schrock, F. B. Selph, J. H. Lienhard, and Z. R. Roxztoczy, "Transient Pool Boiling of Water at Atmospheric Pressure", ASME Paper 29, Part 2, Sec. A.)
64. "SPERT Quarterly Progress Report, July-September 1958", Report IDO-16512, Phillips Petroleum Co., 1958.
65. J. Dugone and D. D. Wieland, "Fuel Plate Experience during SPERT-I Destructive Test Series with an Aluminum Clad, Plate-Type Core", Report IDO-16885, Phillips Petroleum Co., 1963.
65a. A. H. Spano and R. W. Miller, "SPERT-I Destructive Test Program Safety Analysis Report", Report IDO-16790, Phillips Petroleum Co., 1962.
66. R. W. Miller, A. Sola, and R. N. McCardell, "Report on the SPERT-I Destructive Test Program on an Aluminum Plate-Type Water-Moderated Reactor", Report IDO-16883, Phillips Petroleum Co.
67. M. R. Zeissler, "SPERT-ID-Core Data Summary", Report IDO-16886, Phillips Petroleum Co., 1963.
68. A. Sola, R. Scott, Jr., R. K. McCardell, and E. Feinauer, "Reactor Kinetic Behavior and Motion Picture Documentation of the Destructive Test", Trans. Am. Nucl. Soc., 6(1963) 137.
69. R. W. Miller, A. H. Spano, J. Dugone, D. D. Wieland, and J. E. Houghtaling, Experimental Results and Damage Effects of Destructive Test", Trans. Am. Nucl. Soc., 6(1963)138.
70a. News Item, Nucl. News, December 1963, p. 30.
70b. W. K. Ergen, "SPERT-1 'destructive' test with UO_2 fuel", Nuclear Safety, 5, 3 (1964) 231.
70c. F. Schroeder, private communication received May 19, 1964.
71. J. W. Greenwood, "Contamination of the NRU Reactor in May 1958", Canadian Report AECL-850, Atomic Energy of Canada, Ltd., 1959.
72. A. F. Rupp, "NRU reactor incident", Nucl. Safety, 3(1960)70.