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ADDENDUM TO THE SPERT III HAZARDS SUMMARY REPORT--
LOW-ENRICHMENT OXIDE CORE

J. E. Houghtaling, J. A. Norberg, and J. C. Haire

RELEASED FOR ANNOUNCEMENT
IN NUCLEAR SCIENCE ABSTRACTS

PHILLIPS
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ABSTRACT

This addendum to the Spert III Hazards Summary Report was prepared and submitted to the Atomic Energy Commission to explain the important features of a new slightly enriched oxide core, designated the E-core, and to analyze the potential problems and hazards of operating this reactor in the Spert III facility. The report takes the form of an addendum because, with the exception of the core and certain aspects of the control system, all features discussed in the previous Spert III Hazards Summary Report still apply. Pertinent mechanical characteristics of the E-core are explained in some detail, as well as changes made to the Spert III control system. Included are the nuclear characteristics of the core as determined both by calculation and extrapolation of previous experimental results. Also, the predicted kinetic behavior for the planned tests and certain accident conditions are discussed. Finally, several accident situations are considered and analyzed to obtain numerical estimates of radiological hazards. It is concluded that the E-core can be operated as proposed and remain within the safety policies of the AEC.

SUMMARY

This report is an addendum to the original Spert III Hazards Summary Report, IDO-16425, and includes a radiological safety evaluation for a slightly enriched rod-type oxide core (E-core) test program which will be initiated in the Spert III PWR facility in 1965.

The test program, which is designed to obtain information pertinent to power reactor safety and technology will be carried out on a water-moderated and -reflected core composed of about 1500 stainless-steel-clad fuel rods, each containing 38.6 g of U-235 in 4.8 percent-enriched sintered UO_2 pellets. The experimental program will include various reactivity perturbation experiments initiated over a wide range of system conditions; pressure, 0 to 2500 psig; temperature, ambient to about 550°F; coolant flow rates up to 20,000 gpm (≈ 24 ft/sec) through the core; and initial reactor power levels from the order of watts to about 30 MW.

A pre-operational analysis of the anticipated behavior of the E-core for the proposed program based on previous experience and calculations indicates that, when the core is subjected to large reactivity perturbations, reactivity feedback arising from Doppler broadening of the U-238 resonances will provide the principal mechanism for self-limitation of the excursions and that excursions with initial reactor periods as short as 2 or 3 msec may be conducted without significant core damage.

Postulated accidents which could conceivably occur during the E-core test program are evaluated and it is concluded that the safeguards provided and the experimental and operational procedures to be employed ensure that the release of significant quantities of fission products outside the primary reactor system cannot occur except in the unlikely event of a large primary system rupture leading to a loss of coolant and accompanied by fuel melting and/or breakup of a highly irradiated core. No specific mechanism can be envisioned or is postulated in the report to initiate the event, but the loss of coolant is assumed to follow an offset shear at the bottom tee of the reactor vessel. The stored thermal energy at temperature and pressure (550°F, 2500 psig) in the primary coolant, if suddenly released, would probably destroy the reactor building; and no containment is provided at Spert III because of its remote location. Thus, any fission product release from the fuel which might result from a loss-of-coolant accident must be assumed to eventually be present in the outside atmosphere. To reduce the hazards implicit in such an accident, close control of the fission product inventory and of the test conditions is required.

Upper-bound combinations of fission product inventory and meteorological conditions which are compatible with planned testing of the E-core are established and discussed in the report. Operation of the reactor is limited to a maximum steady-state power level of approximately 30 MW, a maximum nuclear energy release of 15 MW-hr in any single test, and 30 MW-hr in any 24-hour period. In addition, controlled meteorological conditions are required for all tests which can reasonably result in a total energy release greater than 2.5 MW-hr.

Detailed radiological analyses are presented for three postulated loss-of-coolant accident cases as follows: (a) a loss-of-coolant accident, with

total core melting, immediately following a 30 MW-hr test (termed the maximum referent accident); (b) a loss-of-coolant accident immediately following a 2.5 MW-hr test; and (c) a loss-of-coolant accident occurring during plant standby conditions three hours after a 30 MW-hr test. In each of the three cases, the total fission product inventory is based on the maximum core inventory for the planned test program; ie, a series of 15 MW-hr tests repeated twice daily, five days a week for four weeks.

(a) In the analysis of the first accident case, calculations are included which show that no melting will occur for any planned test, and only about 20 percent clad melting would occur during the postulated maximum referent accident. However, because of inherent uncertainties in such calculations, total core melting is assumed in the radiological analysis for this first accident case. In this analysis, meteorological conditions assumed are typical of those required for high-power tests (lapse, > 3 m/sec wind speed, wind direction 190° to 250°). The fission product release criteria of TID-14844 (1 percent solids -- 50 percent halogens -- 100 percent noble gases) are used.

The results of the analysis indicate maximum radiological doses resulting from infinite exposure at the downwind site boundary (note that the control center is not in the downwind direction) to be 0.06 rem whole body (from external γ , cloud and deposition) and 0.13 rem inhalation dose to the adult thyroid. These calculations indicate that the conditions set forth in 10 CFR 100 for the establishment of an exclusion zone are satisfied and, hence, that no undue risk of exposure of personnel accrues from this operation. For comparative purposes, the report also includes calculated doses for the maximum referent accident under inversion conditions, even though the test operations on which such an accident is predicated are not permitted.

(b) For the second accident case postulated, a radiological analysis is presented for a loss-of-coolant accident immediately following a test with a 2.5 MW-hr energy release, in order to establish the acceptability of radiological hazards arising from any accident associated with such tests. Since such tests will not necessarily be performed under favorable meteorological conditions, inversion conditions are assumed to prevail during the accident. Although fuel melting will not occur, clad rupture may be postulated to occur as a consequence of hydrodynamic forces during the accident. It is assumed that such fuel breakup is followed by sudden release of gaseous fission products from the fuel rod fill-gas spaces and subsequent slow release of additional fission products during oxidation of the hot ($> 550^{\circ}\text{F}$) UO_2 fuel. Oxidation and associated fission product release rates based on extrapolation of available experimental data are used in the analysis. Calculations indicate that the exposed fuel cools below the threshold temperature for significant oxidation reaction in slightly less than one minute; therefore, these processes are assumed to terminate after one minute.

The results of the radiological analysis for this accident case indicate that the maximum doses at the Spert control center (0.5 miles) with a 4.5-minute exposure and at the nearest site boundary (6.4 miles) with infinite exposure would result under inversion conditions and with a wind velocity of approximately 1 m/sec. At the control center these doses are 1 rem whole body and 230 rem thyroid. At the nearest site boundary, they are 0.92 rem whole body and 4.4 rem thyroid. These doses indicate that the conditions set forth in 10 CFR 100 for the establishment of an exclusion zone are satisfied

and, hence, that no undue risk of exposure of personnel accrues from this operation.

(c) In the third case, a loss-of-coolant accident is assumed to occur three hours after the last high-power (30 MW-hr) test, and an analysis similar to that of the previous case is carried out. It should be noted that approximately 96 percent of the time that the Spert III plant is at operating conditions of pressure, temperature, and flow; it is in a hot standby condition, rather than engaged in or immediately following nuclear operation. Statistically, the possibility of occurrence of a loss-of-coolant accident during such standby operation of the plant is increased by the relatively large fraction of the time spent at standby. Also, prevailing weather conditions can change, after a high-power test, from lapse to strong inversion conditions. (The estimated minimum time for such a change is three hours.) Although fuel melting would not result from such an accident, clad rupture could lead to fission product releases similar to those discussed in (b). The maximum calculated radiological doses for a 4.5-minute exposure at the control center, following an assumed plant standby accident under inversion conditions with 1 m/sec wind speed, are 1.4 rem whole body and 370 rem adult thyroid. At the nearest site boundary, 6.4 miles from the reactor, assuming essentially infinite exposure, the maximum doses calculated for this accident case are 1.3 rem whole body and 7.2 rem adult thyroid. These results indicate that the occurrence of such an accident will not result in undue hazards even though the 370 rem adult thyroid dose at the control center is slightly in excess of the 300 rem criteria set forth in 10 CFR 100 for the establishment of exclusion zones.

A number of other accidents are also discussed; however, in all cases the consequences of each are assessed to be less than those three for which the detailed radiological analyses were conducted.

The radiation hazards for the maximum referent accident of this report are also compared, in Appendix D, with the hazards arising from the accident assumed in the original Spert III Hazards Summary Report. This comparison shows that the calculational methods of the original SAR give results substantially in agreement with results obtained by the methods of this addendum, provided that the two are applied on the same basis. It also shows that the radiological doses calculated for accidents postulated for the test program and operating mode with the oxide core are factually less than those presented in the original SAR. With one exception, in fact but not in significance, they are also less than the doses for the original SAR accident postulation as recalculated with the current, more conservative techniques.

In view of the radiological hazards calculated for these various situations, the criteria set forth in 10 CFR 100 for the establishment of exclusion zones and the results of extensive testing in the recent Spert I oxide core, it is concluded that the Spert III oxide core testing program can be conducted in a manner consistent with the general policy of the AEC to protect government and contractor personnel and the general public against undue exposure to radiation and all other potential health and safety hazards which may arise in the conduct of nuclear research programs.

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I. INTRODUCTION

A slightly enriched oxide core (designated the E-core) will be assembled in the Spert III facility for kinetics testing in 1965. As a major objective of these tests, basic kinetic studies will be conducted to obtain information pertinent to power reactor safety and technology.

Previous investigations of the kinetic behavior of a slightly enriched oxide core have been conducted in the Spert I facility with initial conditions of ambient temperature ($\approx 20^{\circ}\text{C}$) and pressure, no forced coolant flow, and low initial power levels [1]. These tests, in addition to providing basic kinetic data from which a Doppler feedback model [2, 3] was developed, simulated the cold, start-up type of reactivity accident into the region of limited core damage [4, 5, 6] for initial reactor periods as low as 1.55 msec. The extension of the basic studies to environmental conditions typical of operating power reactors will be accomplished in the Spert III facility.

The primary objectives of the forthcoming Spert III test program are to obtain important additional information on the basic kinetic behavior of slightly enriched oxide cores and to investigate their dynamic response characteristics to reactivity insertions for various initial system conditions and reactor power levels. The data will yield corresponding experimental determinations of the magnitude and temperature dependence of the Doppler effect, permitting additional evaluation of the applicability of the Doppler feedback model, particularly for those excursions which are initiated from high power. As a supplement to the Spert III Hazards Summary Report, IDO-16425 [7], this addendum is intended to provide: (a) a summary of the objectives of the Spert III oxide core program, (b) a description of the operating procedures and their safety aspects, (c) selected results and analysis of related static and kinetic experiments already performed on which analysis of the expected kinetic behavior of the Spert III oxide core is based, and (d) an evaluation of the hazards associated with the performance of the projected test program.

These topics are discussed more or less sequentially in the following sections of the report. The brief facility description contained in Section II (and summarized in Appendix A) emphasizes recent additions to the facility; an outline of the experimental program including brief discussions of operating philosophy and control system design are included in Section III and Appendix B; and selected results of a pre-operational analysis are presented in Section IV. Each of these sections attempts to describe only facets of the facility or its operation that are pertinent to an evaluation of its safety.

Section V reviews a number of accidents postulated for the E-core including for each case the initiating incident, the probable results, and the indicated corrective action or response on the part of Spert personnel. The loss-of-coolant accident following sustained high-power operation is concluded to be the most hazardous accident postulated for the E-core if core meltdown is assumed.

Calculations of core meltdown (Appendix C) indicate that total core melting would not occur. Since this accident does result in the largest fission product inventory and the largest potential radiological doses, it has been termed the maximum referent accident in this report. It does not result in the maximum radiological doses, because of imposed meteorological conditions for the test operations upon which it is predicated.

Analyses of two other loss-of-coolant accidents assumed to take place with uncontrolled meteorological conditions include considerations of fission product release following clad rupture. Fuel melting is assumed not to occur in these two cases but rather the core is assumed to break up as a consequence of hydrodynamic forces and to quickly release gaseous fission products from the fuel rod fill-gas spaces. This is assumed to be followed by the slower release of additional fission products during the subsequent oxidation of the hot UO_2 fuel. Since the radiation doses calculated in Section VI were obtained by more recent calculational techniques and assumptions than those of the original Hazards Summary Report, the differences in the doses appearing in the two reports are discussed in some detail in Appendix D.

Conclusions drawn from the safety analysis of the E-core experimental program to be performed in the Spert III reactor facility are presented in Section VII.

II. DESCRIPTION OF THE REACTOR FACILITY

A brief description of the general features of the facility, with emphasis on those recently added, is presented in this section. Particulars of the reactor and equipment are discussed in detail in the Spert III Facility Report, E-Core Revision [8], the Spert III Facility Report [9], and the Spert III Hazards Summary Report [7].

1. GENERAL AREA LAYOUT

The location of the Spert III reactor building with respect to the other Spert facilities is shown in Figure 1. Remote control of the reactor and plant is provided to permit operation from the Spert control center, located about 1/2 mile from the building housing the reactor.

Fig. 1 Plan of the Spert site.

2. REACTOR VESSEL AND COOLANT SYSTEM

The reactor facility incorporates the nuclear and hydraulic features of conventional pressurized-water and boiling-water reactors. The reactor vessel and primary coolant system are designed for maximum operating pressures and temperatures of 2500 psig and 650°F, respectively. The primary system is comprised of a reactor vessel, pressurizer vessel, and two coolant loops as shown in Figure 2. Each of the primary coolant loops consists of two canned-rotor pumps operating in parallel, a heat exchanger system including temperature control valves, two flow-measuring devices, and flow-control and check valves. The total flow capability of both loops is 20,000 gpm, and the total heat removal capacity is 60 MW for 30 minutes. Coolant enters the reactor vessel at a bottom tee, flows upward through the reactor core, reverses direction, and flows downward through the thermal shields leaving the vessel near the bottom.

Fig. 2 Spart III process flow diagram.

The primary-coolant circulating pumps are the canned-rotor type, each capable of delivering 5000 gpm against a head of 328 feet of water. Heating of the primary coolant to 650°F may be accomplished in about 16 hours independent of nuclear energy input by utilizing the energy transferred to the coolant by the pumps.

Coolant flow in each loop is controlled by remotely operated gate valves, and the rate of flow is metered throughout the entire flow range by either an 8- or a 16-inch flow tube. The flow tubes were initially calibrated at a hydraulic testing laboratory prior to installation and were subsequently cross

calibrated in the system during engineering tests on the Spert III C-19/52 core. The separate flow calibrations agree to within ± 10 percent over the full range of plant operating conditions. The primary piping system is spun cast AISI, Type 304L stainless steel and is sized to permit water velocities up to 30 ft/sec with a minimum pipe surface area and attendant heat loss.

The reactor vessel is a multi-layer type with the control rod drives mounted on the fully removable bolted top head. Thermal shields comprised of five concentric cylindrical annuli of stainless steel (including the core flow skirt) with a combined metal thickness of 5-3/4 inches are provided to minimize the temperature-induced vessel-shell stresses during nuclear power operation. Because of the remote operation of the facility, the only radiation shielding provided is a 6-inch-thick lead wall (one thickness of lead brick) around the reactor vessel to reduce radiation levels near the reactor in the shutdown condition. Primary-system pressure is maintained by a steam dome in an electrically heated pressurizing vessel which is statically connected to the primary coolant system.

A recent addition to the facility is a soluble poison injection system which provides additional shutdown capability of the reactor in the event of failure of other control systems. As indicated in Figure 2, the remotely controlled poison injection system consists of a 1000 gallon tank filled with a neutron poison solution which can be injected into the primary coolant system by means of the high-pressure (3000 psi head) makeup pump. A discussion of the operational characteristics of this system is presented in Section V-2.

3. DESCRIPTION OF THE OXIDE CORE

Figure 3 shows a cross section of the Spert III oxide core (designated the E-core) and indicates the component placement in the core. The E-core is comprised of fuel rods which are 40.8 inches long: 0.466-inch OD by 0.426-inch ID, Type 348 stainless steel tubes, containing a 38.3-inch active length of 0.420-inch OD pellets of 4.80 percent enriched UO_2 . The pellet density is 10.50 g/cm³, and a 0.003-inch gas gap is provided between the outer surface of the pellets and the inner surface of the stainless steel tubes. These fuel rods were constructed for the PL-2 core [10] and have remained in storage except for a series of 38 criticality tests conducted with them at Combustion Engineering [11] in 1962.

In the E-core, the fuel rods are nominally contained in 3- x 3-inch stainless steel fuel assembly cans as indicated pictorially in Figure 4. Upper and lower grids, which hold the corresponding end boxes of the fuel assemblies, maintain the core array. With the exception of the four assemblies which surround the transient rod and the eight control rod fuel followers (which contain a 4 x 4 assembly of fuel pins), each assembly contains 25 fuel rods arranged in a 5 x 5 rectangular array with a square pitch of 0.585 inch. Each fuel assembly has two intermediate grids placed 13 and 26.5 inches above the top of the lower grid to restrict lateral fuel rod motion and the nominal 25 rod fuel assemblies are designed to permit coolant cross flow. The twelve smaller fuel assemblies are similarly constructed except that they have no provision for cross flow. The average nonmoderator-to-moderator ratio in the E-core is approximately 1.05, and the average H/U ratio is 3.5.

Fig. 3 E-core cross section (64-assembly loading).

Fig. 4 Nominal 25 fuel rod assembly.

4. CONTROL AND TRANSIENT RODS

Reactor control is provided by eight control rods driven in pairs by the four existing Spert III rod drives. As shown in Figure 5, each control rod is comprised of a poison section (consisting of a water-filled, 2.5-inch square, 1.35 weight percent B-10 stainless steel tube) and a fuel follower section (consisting of a square 4 x 4 array of fuel rods). Fixed guide tubes provide lateral support over the full travel length of each control rod. The guide tube section through the active core region is constructed of Zircaloy 2. A 5.125-inch wide cruciform transient rod, located at the center of the core, is used for initiating reactor transients. This rod, shown in Figure 6, is composed of a stainless steel upper section and a lower poison section composed of 1.35 weight percent B-10 stainless steel. For both the control and transient rods, a positioning accuracy of ± 0.01 inch is maintained to facilitate the reproducibility of experimental results.

Fig. 5 Control rod pair assembly.

Fig. 6 Transient rod assembly.

5. FACILITY MODIFICATIONS TO ACCOMMODATE THE E-CORE

Only minor modifications were required to adapt the Spert III facility for operation of the oxide core. Since the E-core will have longer fuel elements than the previous plate core, the lower grid was remounted to accept the longer core. The control and transient rod guides were extended to accommodate the longer E-core control and transient rods. Because of the greater E-core control element travel, some modification of the upper limit and top limit switches in the control rod drive unit housing was required. These modifications are discussed in detail in the modified facility report [8]. The capability of the existing control and transient rod drives to perform properly with the correspondingly heavier E-core control and transient rods has been carefully tested. The result of this examination indicates no operational problems with the existing control mechanisms.

In consideration that the facility has been successfully operated for more than four years with the previous plate core (more than 350 power excursion tests), no new operational problems are anticipated during testing of the E-core.

III. EXPERIMENTAL PROGRAM

This section presents a brief description of the experimental program for the Spert III E-core. For discussion purposes, this program has been divided into two general categories: static tests and kinetic tests. The first category includes the critical and operational core loading and the statics testing necessary to determine the basic nuclear parameters needed for operational purposes and to analyze subsequent kinetic test data. Kinetic testing of the core has been roughly subdivided into step- and ramp-induced reactivity perturbation tests and operational accident tests.

A brief discussion of administrative procedures used at Spert in the conduct of experimental programs and of the proposed reactor operating limits is also included.

1. CRITICAL CORE LOADING

This section discusses the initial critical loading of the E-core, with brief descriptions of the neutron detection instrumentation and the fuel loading sequence.

1.1 Neutron Detection Instrumentation

Several calibrated neutron pulse-counting systems will be used during the start-up of Spert III. In addition to the pulse count indicators in each system, a logarithmic count rate meter is provided on the reactor console to monitor any of the pulse channels.

Two or more boron-lined ion chambers will also be connected to strip-chart recorders and used during the core loading to provide continuous information on neutron level at high multiplications and for operation near criticality.

During the critical core loading, a neutron source will be placed in the core.

1.2 Initial Critical Core Loading Procedure

During the initial cold critical core loading, data will be obtained on the reciprocal neutron multiplication as a function of fuel loading to provide guidance for subsequent fuel loadings. The data will be used to predict the critical mass of the system prior to each loading step.

In order to facilitate in-pile instrumentation, a total of 15 "removable" fuel assembly lattice positions is provided. In these locations, fuel elements may be inserted or removed without removal of the vessel head from the reactor vessel or of the restraining hold-down grid from the core. For operating convenience, during the initial approach to criticality, the fuel loading sequence will be planned so that some of the fuel assemblies in removable positions will be taken out prior to the insertion of assemblies in new core positions. The last fuel insertions for each increased loading will then be made into the removable assembly positions with the vessel head in place. This procedure is essentially the same as that used in the initial critical loading of the Spert III C-19/52 core [12]. The calculated critical loading (Table I), including the eight control rod fuel followers, is 36 fuel assemblies.

TABLE I
CALCULATED NUCLEAR PARAMETERS
FOR THE 64-ELEMENT, SPERT III E-CORE

Critical Core	70°F[a]	550°F[a]
Total number of fuel assemblies	36	
Total mass of uranium	637.4 kg	
Total mass of U-235	30.6 kg	
Operational Core	70°F[a]	550°F[a]
Total number of fuel assemblies	64	
Total number of fuel pins	1492	
Total mass of uranium	1200.6 kg	
Total mass of U-235	57.6 kg	
Shutdown reactivity, all rod poison inserted	≈ 15.3\$	≈ 25.9\$
Shutdown reactivity, control rods inserted, transient rod poison withdrawn	≈ 11.3\$	≈ 21.9\$
Excess reactivity, all rod poison withdrawn	15.8\$	5.25
Excess reactivity at high power and temperature ^[b]		3.48\$
Excess reactivity, control rod poison withdrawn, transient rod inserted	11.70	1.26\$
Cold (70°F) excess reactivity with one control rod fully withdrawn	2.97\$	
Cold (70°F) reactivity shutdown with one control rod fully withdrawn and the transient rod poison inserted	0.88\$	
Control rods withdrawal to critical position	8.9 in.	19.5 in.
Worth of control rods near critical	1.90\$/in.	0.75\$/in.
Prompt neutron generation time	15.3 μsec	
Effective delayed neutron fraction β_{eff}	0.008	
Mean temperature coefficient (70 to 550°F)	- 2.2¢/°F	

[a] Isothermal temperature.

[b] 550°F moderator, 975°F average fuel temperature.

2. OPERATIONAL CORE LOADING

Following establishment of the critical core, additional fuel will be loaded until a core configuration is attained which provides an adequate excess reactivity for kinetic testing at desired system temperatures.

The loading of an operational core for the Spert III reactor will be essentially a continuation of the initial approach to criticality. Following each major addition of fuel, data will be obtained to ensure adequacy of the shutdown margin and to provide an estimation of the worth of the next fuel addition. The critical control rod position will be determined after each loading, and the control rods withdrawn sufficiently above the critical position to put the reactor on about a 50 sec period in order to obtain an approximate value for control rod worth.

In addition, shutdown information for a single withdrawn control rod will be obtained. The one-rod critical problem has been calculationaly investigated and the results indicate (line 13, Table I) that the cold 64-element core will be approximately 3\$ supercritical with one control rod pair fully withdrawn. Additional calculations, discussed on page 15, indicate that a smaller core (56 to 60 elements) will probably not have this undesirable property and will, in all likelihood, still meet the major design requirements of the proposed program.

The precise operational loading for the core will be determined experimentally and will be one which is at all times subcritical with one control rod pair withdrawn, provided that such a core does not substantially compromise the objectives of the experimental program.

3. STATIC TESTS

The temperature-dependent system reactivity for the E-core will be determined by a series of elevated temperature (hot) critical tests. Appropriate addition or removal of fuel will be based upon the early results of these tests such that approximately 5\$ excess reactivity is available at an isothermal system temperature of 550°F. These hot critical tests will also be utilized to determine differential and integral control rod worths, excess reactivity, and the isothermal temperature coefficient of reactivity. Subsequent static tests will determine uniform and local void coefficients, the reactivity worth of individual fuel rods, reduced prompt neutron lifetime, flux distributions, and ion chamber power calibrations.

The isothermal temperature coefficient of reactivity and the control rod worth will be obtained for system temperatures from ambient to about 600°F. Since the control rod worth is expected to be about 1.90\$/in. near the cold critical position, the measurements may be repeated a number of times to determine the control rod worth with reasonable statistical accuracy.

Pressure coefficient data will be obtained at ambient temperature with system pressures up to 2500 psig and at several elevated system temperatures and pressures. These data and the differential rod worth data will be used to obtain the reactivity change as a function of system pressure.

Flux mapping and power calibration experiments will be performed for system conditions from ambient to the upper operating limit of the core. Calorimetric power calibrations at high steady-state power levels will be obtained and compared to those obtained by flux map techniques. The calorimetric calibration results will be considered as the final values and all previous data adjusted to agree with them.

Uniform and local void coefficients will also be obtained.

Reduced prompt neutron lifetime, ℓ/β_{eff} , will be determined by a reactor noise analysis experiment [13, 14] conducted with the reactor at delayed critical (power spectral density technique) and subsequently also from step transient data.

4. STEP-INITIATED TRANSIENT TESTS

Step-initiated power excursion tests will be performed to determine the kinetic behavior of the core. A major portion of these tests will be initiated by a rapid insertion of excess reactivity into the core. For each test, the poison-bearing transient rod will be drawn into the initially just critical core, leaving the core subcritical. After withdrawal of the control rods by an increment sufficient to give a predetermined reactivity addition, the transient rod poison will be ejected rapidly from the core. The resulting power excursions will be analyzed to determine various kinetic parameters of the core.

Step-initiated transient test variables will include static pressures from 0 to 2500 psig, initial temperatures from ambient to about 550°F, coolant flow rates from 0 to 20,000 gpm through the core, and initial reactor power levels from the order of watts to about 30 MW. Selected calculational results for certain of these tests, discussed in Section IV, indicate some of the experimental results which are anticipated for the core.

In addition to the normal complement of neutron-detecting instruments used for critical and subcritical operation of the reactor, a series of ion chambers will be positioned at varying distances from the core to provide power level measurements over a range of about 10 decades. This transient reactor power instrumentation is nominally adjusted to give continuous coverage of reactor power within about two decades above and below the anticipated peak value for a given test, and backup power coverage will be maintained to at least 100 GW.

Thermocouples welded to the fuel pin cladding surfaces will yield transient fuel pin surface temperatures at representative locations in the core, and bulk water temperature measurements will be obtained at several points in the system. These fuel cladding surface thermocouples will be capable of indicating temperature rise to about 1300°C (melting temperature of stainless steel).

Pressure transducers located within and near the core will permit measurement of transient pressures developed during the tests. These transient pressure transducers will cover the pressure range from 0 to 10^3 psig (above initial system pressure). Based on data obtained from the Spert III C-core and

the recent Spert I oxide core, normal transient pressure pulses are not expected to exceed 10 psig, and pressure pulses arising from any ruptured fuel rods (see Section V-1) are not expected to exceed about 150 psig. From previous experience with the Spert III C-core, system pressure surges of the order of 100 to 200 psig are expected for the high power conditions which follow power excursions performed from initially high pressure, temperature, and power.

5. RAMP-INITIATED TRANSIENT TESTS

A series of ramp tests, in which predetermined reactivity insertions are made at various addition rates, will be conducted over much of the range of operating conditions outlined for the step-initiated transient tests. These tests will not only compare the results of ramp-type insertions to equivalent step-type reactivity additions, but will also be a good approximation to a start-up accident in which the control rods are withdrawn at some constant rate, initiating a power excursion.

6. OPERATIONAL ACCIDENT TESTS

Similarly, tests will be performed to simulate various operational-type accidents such as loss-of-flow and ramp insertion followed by rod reversal shutdown. The range of variables, as well as the tolerable severity of these tests, will be determined both by experience in working with the core and by a series of continuing analytical studies on the part of the Spert staff.

7. ADMINISTRATIVE PROCEDURES

In the execution of Spert Experimental Programs, administrative procedures are primarily used to minimize the probability of unplanned nuclear incidents and to help prevent undue hazards to Spert personnel and/or the general public. In addition, a number of automatic safety circuits and interlocks are provided in the Spert III control system and reactor facility which are designed to act on certain functions which could lead to unsafe conditions (see Appendix B for further discussion of the control system). However, close administrative control is necessary because of the nature and objectives of the experimental programs.

The basis of administrative control of each reactor facility during all phases of nuclear and nonnuclear operation is a written document called Standard Practices Manual (SPM). Included in this document is the requirement of a pretest analysis called a Test Series Proposal (TSP) for each experimental program.

The Test Series Proposal presents a brief description of the proposed tests, the objectives of the tests and expected experimental results, and any

unusual hazards associated with the proposed tests. Approval of each TSP by at least one AEC and two company safety review panels is required before testing begins.

Based on the results of evaluation of the hazards associated with the projected test program, on the objectives of the experimental program, and on the capabilities of the E-core and Spert III facility, it is expected that the reactor operating limits established for the low-enriched oxide core in Spert III will include the following: maximum steady-state power level, ≈ 30 MW; maximum nuclear energy release for a single test, ≈ 15 MW-hr; and maximum nuclear energy release in a 24-hour period, ≈ 30 MW-hr. In addition, favorable meteorological conditions as specified by the Idaho Operations Office and determined by a qualified meteorologist of the United States Weather Bureau are required for all tests which can reasonably result in a total energy release of 2.5 MW-hr or more.

IV. PRE-OPERATIONAL ANALYSIS FOR THE SPERT III OXIDE CORE

A pre-operational analysis is presented in the following section which provides an indication of the expected behavior of the E-core for the proposed experimental program. Included is a discussion of the E-core design, the calculated static core characteristics, the transient core characteristics, and the operational (high power) core characteristics. Probable experimental limits such as the maximum step reactivity insertions and steady-state power levels are discussed, and comparisons are made to previously obtained experimental results where applicable.

1. E-CORE DESIGN

In keeping with the objective of kinetic testing from initial conditions which approximate power reactor environmental conditions, the Spert III oxide core was designed for operation with coolant temperatures up to 550°F, system pressures as high as 2500 psig, and steady-state power levels to about 30 MW. Within this objective, it is necessary that the E-core have the capability to undergo short-period power excursions from the initial high-temperature, high-power condition. To meet this requirement, the following design criteria were set for the Spert III E-core.

- (1) A core loading excess reactivity of 2.5\$ is available with the primary coolant at 550°F and the reactor operating at high steady-state power [≈ 30 MW(t)]. This assures sufficient excess reactivity to permit the initiation of short-period, power-excursion tests from an initial power level of about 30 MW.
- (2) With the control rod poison inserted into the core, the transient rod poison withdrawn from the core, and the entire system at 70°F, a shutdown reactivity of at least 3.0\$
- (3) Sufficient transient rod poison reactivity worth to permit reactivity insertions greater than 2.5\$ over the anticipated range of system temperatures, pressures, and initial power levels.

Design calculations indicate that a 64-fuel-element core will provide adequate excess reactivity to accomplish the desired transient test program, but may be about 3\$ supercritical at ambient conditions with one control rod fully withdrawn from the core. The ambient temperature calculations were repeated for a 52-fuel-element core, and this core was determined to be about 4\$ subcritical with one control rod fully withdrawn. A linear interpolation of these results indicates that a 56- to 60-fuel-element core will be subcritical at ambient temperature with one control rod withdrawn and will provide about 2\$ to 4\$ less clean cold excess reactivity than the 64-element core.

Since it was not practical to perform an extensive pre-analysis for more than one configuration of the E-core, the 64-element loading was considered

for the purposes of this report (although further detailed analysis will be undertaken after the operational core is established). This choice was based primarily on the following considerations:

For calculational purposes, a reasonable maximum steady-state power level from which kinetics tests may be performed is that power level which results in hot-spot centerline fuel melting. Thermal calculations indicate that peak fuel temperatures will reach about 5000°F (threshold of melting) at a steady-state power level of about 40 MW(t) in the 64-element core, and about 35 MW(t) in the 56-element core.

The kinetic behavior of the 64-element core will be very similar to that of a slightly smaller (56- to 60-element) core in most respects, but will present the greater hazard from the maximum postulated accident if the above maximum fuel temperature limitation is applied in each case. This arises because of the higher operating power levels and correspondingly large fission product burden associated with the larger core. Therefore, the pre-operational analysis, the accident analysis, and the hazards analysis are reasonably predicated on operation of the 64-element core; and all subsequent discussion of the E-core in this report refers to the 64-element core.

A summary of the calculational results for the 64-element Spert III oxide core program is given in the following subsections. The major portion of this effort to date involves the calculations of the Doppler feedback and burst shape characteristics for a representative set of reactor operating conditions.

2. STATIC CALCULATIONS FOR THE 64-ELEMENT CORE

The static core calculations utilized nuclear calculational schemes [15-19] which are in common use. These calculations were performed for a 64-element core lattice. A cross section of this 64-element lattice, shown in Figure 3, indicates the placement of fuel elements, control rods, and the transient rod. Pertinent parameters of an individual fuel rod [10] are presented in Table II. The calculated static characteristics of nuclear parameters for the 64-element E-core were summarized earlier in Table I.

Shown in Figures 7 and 8 are the calculated axial and radial flux profiles for the cold core with control rod position as a parameter in Figure 7. These profiles clearly indicate the large flux peaking which will occur in the lower portion of the core during tests initiated from ambient temperature. Because of this, the cold core will

TABLE II
SPERT III E-CORE
FUEL ROD PARAMETERS

Pellet Density	10.50 ± 0.01 g/cm ³
Pellet Enrichment	4.80 ± 0.05 wt% U-235
Pellet Diameter	1.067 cm
Rod Diameter -- OD (Type 348 SS)	1.184 cm
Cladding Thickness	0.0508 cm
Air Gap Width	0.0076 cm
Rod Length	103.6 cm
Active Fuel Length	97.3 cm
Mass of U-235 per Fuel Rod	38.6 g

Fig. 7 Calculated axial cold core neutron flux profiles versus control rod position.

Fig. 8 Calculated radial cold core neutron flux profiles.

exhibit substantial peak-to-average temperatures (during transient testing) which are quite similar to those observed in the Spert I oxide core. Since one of the objectives of the E-core test program is to relate the kinetic properties of the two cores by use of existing analytical models, this similarity in the flux profiles of the cold Spert I and Spert III oxide cores will aid in the interpretation of test data.

The calculated ratio of maximum-to-average power density in the core is shown for various control rod positions in Figure 9, with system temperature as a parameter. These profiles indicate the expected change in the hot-spot factor as the system is taken from cold to the hot, high-power condition. The power peaking factor of approximately 2.2 for the hot (550°F) high-power (30 MW) condition is reasonably representative of that in commercial power reactors.

The computed zero-power average temperature coefficient of reactivity for

Fig. 9 Calculated ratio of maximum-to-average power density versus control rod position.

the E-core is approximately $-2.2\phi/^{\circ}\text{F}$ in the range 68 to 550°F. This temperature coefficient is principally due to moderator density changes and Doppler broadening which accompany the heating of the system.

3. TRANSIENT CORE CHARACTERISTICS

The dynamic reactor response presented below was computed using a Doppler feedback model developed at Spert [2, 3]. This kinetics model, which gave excellent results when applied to short-period tests in the recent Spert I slightly enriched oxide core, is expected to reasonably predict the dynamic characteristics of the E-core over a wide range of variables.

Figure 10 shows several computed power burst shapes with varying initial reactivity insertions for the initially cold core at ambient conditions. For each of these bursts, the step reactivity input (taken as a 20\$/sec ramp) began at time $t = 0$, when the power level was 1 W. Because of the comparable prompt Doppler coefficients of reactivity, the maximum power for each of these bursts is similar to the corresponding values obtained in the recent Spert I oxide core tests. In Figure 11 is shown the power burst for the 3\$ step from Figure 10, together with energy release, hot-spot fuel temperature rise, and reactivity insertion. For this severe transient, in which the initial period is about 1.7 msec and which represents an upper limit of anticipated step initiated power excursions, the calculated hot-spot fuel temperature reaches the point of incipient melting approximately 1 sec after the start of the test.

Fig. 10 Several calculated power burst shapes with varying reactivity insertions for the initially cold (78°F) system.

Fig. 11 A calculated 1.7-msec-period burst showing power, energy, and fuel temperature for the initially cold system.

The small secondary power peak appearing in Figure 11 occurs because only about 2.20\$ of the 3\$ external reactivity insertion has been accomplished at the time of the first power peak. The sustained high runout power level, together with the large power peaking factor for this case, lead to the rapid onset of fuel melting following such a severe test.

Figure 12 shows the period dependence of the computed initial peak power and the energy to the time of the first power peak for the Spert III E-core, together with the corresponding experimental results from the Spert I oxide core. For the long-period regime ($\alpha \lesssim 50 \text{ sec}^{-1}$) the difference in the curves arises from the necessary assumption of coolant flow in the Spert III case. In the absence of this flow, the peak power and corresponding energy release values for the E-core are expected to lie slightly below those obtained from the Spert I oxide core for all reactor periods, as they do for the short-period portion ($\alpha_0 \geq 50 \text{ sec}^{-1}$) of Figure 12. For the short-period regime, the differences between the calculated E-core response and that obtained in the Spert I oxide core reflect the dependence of the Doppler shutdown characteristic on certain core design parameters. For example, due to the difference in H/U ratios, the resonance escape probabilities are (respectively) 0.86 and 0.84 for the Spert I and Spert III oxide cores. This results in a Doppler feedback coefficient in the E-core which is slightly larger than that in the Spert I oxide core. Another result of the smaller H/U ratio in the Spert III core is the shorter prompt neutron generation time (lifetime) of $15.3 \mu\text{sec}$ as compared to $26.3 \mu\text{sec}$ for the Spert I core. Because of this relatively shorter lifetime, correspondingly less reactivity must be compensated to quench a power burst of given initial period. Finally, the peak-to-average heat source in the Spert III cold core is about 20 percent greater than that in the Spert I core, which further increases the Doppler coefficient.

Fig. 12 Calculated peak power and burst energy to peak power versus initial reciprocal period for the initially cold (78°F) Spert I and Spert III oxide cores.

4. OPERATIONAL (HIGH POWER) CORE CHARACTERISTICS

The nature of the Doppler feedback mechanism is such that the energy release required to affect a given reactivity compensation is dependent upon the initial temperature of the system. In consideration of this, the following reactor conditions have been examined: (a) the initially cold (70°F isothermal) low-power condition, (b) the hot (550°F isothermal) low-power condition, and (c) the hot high-power (550°F moderator and 1-, 10-, and 30-MW conditions). The calculated reactor response for case (a) was presented in Section 3, page 18. Additional calculational results presented below indicate the relative kinetic behavior to be expected for conditions (b) and (c).

Because of the relatively smaller Doppler resonance broadening, a given energy addition in the initially hot core (hot fuel) results in less reactivity compensation than the same energy addition in the initially cold core (cold fuel). In particular, a given Doppler reactivity compensation in the initially hot low-power core requires about 1.4 times more energy release (average fuel temperature rise) than in the initially cold core; and the initially hot high-power core requires about 2.5 times more energy release to compensate the same reactivity (average fuel temperature rise) than does the initially cold core.

However, the maximum fuel temperature rise for a given energy release is also dependent upon the initial reactor condition. As indicated in Figure 9, the maximum-to-average heat source is significantly decreased by an increase in the initial system temperature. This decrease in maximum-to-average heat source appears to be almost proportional to the increase in energy release required to effect a given reactivity compensation with increasing temperature; ie, the maximum-to-average heat source is about reciprocally proportional to the corresponding energy release required for a given reactivity compensation.

The calculations indicate that, because of this mutually compensating behavior, the maximum fuel temperature rise required to compensate a given reactivity insertion will vary by less than 20 percent over the full range of initial temperature and power conditions.

As discussed earlier, thermal calculations indicate that peak fuel temperatures will reach about 5000°F (threshold of melting) at a steady-state power level of 40 MW(t). The resulting considerations of fuel melting place a practical limit on the maximum steady-state power level from which kinetics testing can be performed as well as the type and severity of such tests.

In summary, the calculations indicate that transient tests conducted from the low-power initial condition in the Spert III E-core will not result in significant damage to the facility for initial periods as short as about 2 msec if the reactor is promptly scrammed after the first power peak. For the case of short-period transient tests from high initial reactor power and temperature which approach the limits of the E-core capability, sudden fuel melting and cladding stress can occur and lead to damage of the core. The permissible range of variables as well as the tolerable severity of high temperature and initial power tests will be determined both by experience in working with the core (experimentally) and by continuing analytical studies on the part of the Spert staff.

V. ACCIDENT ANALYSIS

This section presents a review of those accidents which could conceivably occur during the conduct of the experimental program with the Spert III E-core. In general, the accidents postulated may be categorized as originating from a failure of some component in the system. However, some could also occur as a consequence of procedural violations. In the analysis of each accident type, attention is given to the initiating incident, the safety implications of the incident, and the indicated corrective action or responses on the part of Spert personnel.

The following discussion proceeds generally in the order of increasing potential severity of the accidents, beginning with the loss-of-flow case which is not believed to comprise a serious safety hazard for the E-core, and ending with three specific cases of the loss-of-coolant accident, one of which, because of its high fission product inventory and the potentially high radiological dose source which such inventory represents, has been termed the maximum referent accident for the E-core. It does not, however, lead to the maximum radiological dose hazard since it is predicated on high-power operation of the reactor and, hence, can only occur under controlled meteorological conditions.

Implicit in the assumption of the loss-of-coolant accident is the failure of the procedural and mechanical devices which aid in the prevention, control, and/or minimization of other unplanned operation. Also uncontrolled transport of fission products is assumed to result from such an accident. For these reasons, fuel meltdown calculations were performed (Appendix C) to determine the conditions under which this maximum referent accident could occur. The results of these calculations indicate that for planned operation of the core, under no conditions of loss of coolant will Spert III E-core fuel melting occur.

1. LOSS-OF-FLOW ACCIDENT

The extreme case of the loss-of-flow accident involves the temporary loss of part or all coolant pumping power during nuclear operation at high-power level. In general, the rapid decrease in coolant flow results in an inability of the coolant to remove all the heat generated by the reactor. Therefore, the system temperature increases, and there exists the possibility of fuel clad melting if the reactor power is not rapidly decreased. The accompanying steam formation can cause overpressurization of the primary system, release of coolant by relief mechanisms, and, with the escaping coolant, the release to the atmosphere of fission products from the melted fuel.

In order to ensure a rapid decrease in the reactor power in the event of an accidental loss of flow, two scram circuits have been added to the Spert III reactor protective system to preclude damage from such occurrences. One channel senses coolant flow directly (ΔP circuit); the other senses it indirectly by monitoring electrical power delivered to the main coolant pumps (ΔV circuit).

The first of these (ΔP) senses the output from a differential pressure transmitter which monitors the pressure drop across the reactor vessel. A fast-response chart recorder is driven by this output signal, and any pressure

drop less than the preselected value encoded in the recorder will cause a reactor scram.

The second loss-of-flow channel (ΔV) will scram the reactor on a loss of electrical power to the primary coolant pumps. This circuit is in addition to the normal electrical power failure scram system and is connected to the normal scram circuit system during all nuclear operation with flow unless deliberately bypassed for some specific reason, such as a planned loss-of-flow test.

Calculations based on the loss-of-coolant accident indicate that the core will not melt, even in the absence of coolant, after tests in which the reactor power is held at 30 MW for 30 minutes. Since stagnant coolant will be present, no hazard to the facility, operating personnel, or the general public would be expected to follow a loss-of-flow incident because of decay heat generation.

2. STUCK-ROD ACCIDENT

The precise operational loading for the Spert III E-core will be experimentally determined such that the core is at all times subcritical with one control rod pair withdrawn, provided this core does not substantially compromise the objectives of the experimental program. However, it is possible for more than one control rod pair to become stuck in the withdrawn or partially withdrawn position, leading to a supercritical condition at certain system temperatures.

In the event that more than one control rod should stick, the reactor will first be made subcritical by the insertion of all operable control rods (including the transient rod) and/or by heating the system by means of the primary circulating pumps and allowing the temperature coefficient to render the reactor subcritical.

With the system subcritical, a condition which can be maintained for several days by operating the plant at high temperature and pressure, a full assessment of the specific problem will be made; and details of an appropriate shutdown procedure will be established.

A reasonable method of reactor shutdown in this situation is by injection of a neutron poison into the system, and a soluble poison (boric acid) injection system has been installed in Spert III for such emergency conditions in which other control mechanisms fail. The poison injection is accomplished by use of the primary system makeup pump. The system is shown schematically in Figure 2, and the details discussed in the modified facility report [8]. Calculations indicate, as shown in Figure 13, that a boric acid concentration of 5 g/l will be required to make the reactor 2\$ subcritical (with all control rods withdrawn) at room temperature. Using the existing makeup pump to add the poison solution to the system, less than 2 hours will be required to bring the boric acid concentration to 5 g/l, as shown in Figure 14. If for some reason the poison

Fig. 13 Compensated reactivity versus boric acid concentration.

Fig. 14 Injection time versus boric acid concentration.

injection system should fail to operate, shutdown of the reactor can also be accomplished by removal of moderator from the reactor by operation of the blowoff and/or blowdown valves. The exact manner in which this blowdown operation will be accomplished will depend on the status of the reactor and will of necessity be determined at such time as the problem occurs. This mode of shutdown is less desirable because of possible thermal stress problems associated with rapid removal of moderator.

The success of these systems is partially dependent upon the electrical power supply in that they both require commercial power. During the past four years, partial or total power outages have occurred at Spert as shown in Table III [20]. Since the frequency of power failure and outage time is small, the probability of an electrical failure during a test which necessitates an emergency shutdown is considered to be acceptably low.

3. CONTROL-ROD WITHDRAWAL ACCIDENT

Because the test program for the E-core requires many operations under system conditions in which more than 15\$ excess reactivity can be rapidly

TABLE III

UNPLANNED ELECTRICAL POWER OUTAGES AT THE SPERT SITE

<u>Date</u>	<u>Duration of Outage</u>
28 January 1965	11 minutes
30 August 1964	60 minutes
13 August 1963	17 minutes
4 August 1963	6 hours, 20 minutes
29 July 1962	10 seconds
13 July 1962	2 seconds
2 September 1961	10 seconds
1 September 1961	20 minutes

inserted by the sudden withdrawal of all control rods from the core, a power level scram device has been installed to prevent the destructive power conditions implicit in such an accident. An analysis of the calculated behavior of the core in the presence of this device indicates that it provides full protection from the hazards resulting from the control rod withdrawal accident. A discussion of the accident, its safety implications, and the power level scram is given in the following paragraphs.

For the maximum control rod withdrawal speed of 17.4 in./min in the E-core, corresponding reactivity insertion rates are very similar to those experienced in the previous C-core, namely about 60 ϕ /sec for the initially cold core condition. As a result of this large reactivity insertion rate, an extended fast control rod withdrawal by operator error or control system malfunction could result in damage to or destruction of the core.

In order to assess the specific dangers arising from a control rod withdrawal accident, an analysis has been performed for several initial system conditions. These conditions correspond to: (a) the initially cold (78°F) low-power; (b) the hot (550°F), low-power core; and (c) the hot (550°F moderator) core with initial power levels of 1, 10, and 30 MW(t). For each case, a coolant flow rate of 20,000 gpm and the maximum withdrawal rate were assumed, which constitute the most pessimistic conditions.

The reactor power and temperature histories arising from each of the five initial conditions are plotted in Figures 15 and 16, respectively. In each case, as seen in Figure 15, a reactor power excursion appears immediately after initiation of the control rod withdrawal. For the 10 and 30 MW cases (labeled D and E in Figure 15), the calculations indicate a monotonic power rise to $\approx$ 150 MW during the first three minutes. Although significant fuel melting and damage to the core would occur before this new steady-state power level should be reached, core damage is not indicated during the first 10 to 15 seconds after initiation of the withdrawal, as shown in Figure 16. The possibility that a control rod withdrawal accident from the initial high-power condition could remain undetected for this length of time is considered to be extremely improbable. This is because the performance of such tests requires the immediate attention of at least six persons in the reactor control room, including two or more qualified reactor operators. Each person in the control room observes one or more preassigned reactor variables and has ready access to a reactor scram device (frequently the scram devices are hand-held by the observers) which is to be used in the event of any (pre-determined) abnormality of the reactor variable.

For the cold (1-W initial power) and hot, low-power (1 W and 1 MW) cases, labeled A, B, and C, respectively, in Figure 15, a definite power peak occurs. Subsequent to this initial power excursion which is relatively mild (by comparison with equivalent step-initiated behavior) in every case, the reactor power rises steadily with increasing reactivity insertion. In particular, for the initially cold case, Figure 16 indicates that fuel melting would begin within 8 seconds after initiation of the withdrawal and would be extensive within 20 to 30 seconds.

Fig. 15 Calculated power behavior during several assumed control rod withdrawal accidents.

Fig. 16 Calculated thermal behavior during several assumed control rod withdrawal accidents.

Because of the potential serious consequences of this accident, a power level scram device has been installed in the facility (see Appendix B). This device consists of two independent reactor power monitors in a circuit, either of which will initiate reactor scram (completed scram time, < 0.5 sec) for any detected power level above a preset value. Since the smallest anticipated reactor power peak for the rod withdrawal accident is above 50 MW, the trip point for the power level scram will be set between 1 to 40 MW. This scram mode is operative during all nuclear operation (when the control console is energized) except when the transient test timer is running or when deliberately bypassed with a special key switch. With these two exceptions, which permit power excursions and high-power operation required by the experimental program, full protection is provided from the control rod withdrawal accident.

4. COLD-WATER ACCIDENT

Because of the intrinsic large negative temperature coefficient of reactivity of the E-core, precautions must be taken to avoid a sudden large decrease of the primary system coolant-moderator temperature when operating the reactor at or near critical with the plant at high temperature and pressure. For these conditions, the cold-water accident results in behavior very similar to that arising from the control rod withdrawal case, and must be considered similarly hazardous. This problem was recognized and discussed in connection with the similar operation of the Spert III C-core; and although the nature of the problem has not changed (from the C- to the E-core), its consequences are less serious though still hazardous for the E-core. To protect the reactor facility from this type of operational accident, primary reliance is placed on administrative safeguards and safe operational procedures. As an added protection, a " ΔT scram system" (see Appendix B) has been installed in the facility and used extensively with the C-core. This system is believed to provide full protection against the hazards arising from a cold-water accident in the E-core. In view of the importance of this accident for the E-core, however, the cold-water accident analysis is presented in the following paragraphs.

Three ways by which a cold-water accident from a sudden drop in the primary coolant temperature can possibly occur are (a) start-up of cold loop, (b) malfunction of the heat exchanger temperature control system, and (c) evaporative cooling of the coolant-moderator caused by a break in the primary system.

4.1 Cold Loop Start-Up

The potential nuclear hazard associated with the start-up of a cold loop may be minimized by administrative control. Prior to high temperature nuclear operation in which flow is required, both primary coolant loops will be circulated until temperature equilibrium is established, thereby eliminating a cold loop. Equal loop flow rates are not, however, necessary.

The possibility of modifying the primary system to eliminate the establishment of a large temperature difference between the coolant in a down loop and that in the operating loop has been investigated. This could be accomplished

by cutting bleed holes in the four primary system check valve gates to permit a continuous small circulation rate in the down loop. However, for the experimental program to be conducted with the E-core, the investigation showed that the necessary modifications are not warranted and that the present method of operation essentially eliminates the possibility of any serious cold-water accident.

4.2 Heat Exchanger Temperature Control Malfunction

It is conceivable that a malfunction of the primary system, heat exchanger, temperature control circuits could cause the coolant temperature to drop, causing a reactivity insertion to the reactor. For example, failure of an electrical circuit in the heat exchanger temperature control system could cause the 16-inch bypass valve to close and the 10-inch heat exchanger valve to open. Figure 17 shows the experimentally measured valve positions versus time for the above condition assuming that the bypass valve was initially 100 percent open and the heat exchanger valve was initially 100 percent closed. With these data plus experimental and calculated thermal-hydraulic operational characteristics of the heat exchangers, calculations have been made showing the mixed mean coolant-moderator temperature leaving the heat exchanger system as a function of time after the temperature control malfunction occurs.

Fig. 17 Spert III heat exchanger valve positions versus time during a heat exchanger, control system failure.

The assumptions made for these calculations are the following:

- (1) The water initially in the heat exchanger and 10-inch heat exchanger piping is at ambient temperature ($\approx 70^{\circ}\text{F}$) and 2500-psig pressure.
- (2) There is complete mixing of the water in the primary circulating pumps and no heat added to the water by the primary pumps.
- (3) There is no mixing of water between the two loops at the reactor vessel bottom tee, and the reactor experiences the excess reactivity caused by the full temperature drop in the core. This assumption is conservative since it requires that only water from the cold loop passes unmixed through the core.
- (4) The decrease in the moderator temperature for a given time increment is added stepwise to the entire reactor core. The

resulting reactivity insertion is calculated from the computed isothermal temperature coefficient of reactivity (based on the average temperature decrease for the given time increment) multiplied by the temperature decrease for the time increment. Two-second time increments are used for the calculations.

- (5) No reactor feedback effects are taken into account. The calculations were stopped either after one complete flow cycle through the system or after the heat exchanger valves had completed their operation, whichever time was less. This is conservative, since the reactor feedback will lower the reactivity.

Figure 18 shows the calculated coolant-moderator temperature versus time, and Figure 19 shows the associated integrated reactivity insertion versus time for initial primary system conditions of 500°F temperature, 2500-psig pressure, and 1000- and 20,000-gpm flow rates. The time (X) for the water to travel from the temperature control valves to the reactor core is directly dependent upon the system flow rate; ie, for 10,000 gpm/loop, $X \approx 6$ sec. As can be noted from Figure 19, the initial average reactivity insertion rate is quite low: less than 10 ϕ /sec for 500°F and 1000-gpm system flow, and less than 20 ϕ /sec for 500°F and 20,000-gpm system flow.

The reactivity rates are conservatively predicted because of the assumption that the fuel and moderator temperatures are isothermally lowered, without feedback heating, as the water temperature in the core is lowered. Because of the relatively long time constant for escape of heat from the fuel rods, the

Fig. 18 Calculated coolant-moderator temperature versus time for a heat exchanger, control system failure.

Fig. 19 Calculated reactivity insertion versus time for a heat exchanger, control system failure.

effective reactivity insertion will be smaller than that shown in Figure 19. Also, a prompt power rise associated with the decreasing moderator-coolant temperature in the core will provide a rapid fuel temperature rise to further limit the severity of the incident. From this it appears that at least 13 seconds would be available in which to scram the reactor after initiation of such a cold-water accident before fuel damage would occur.

To further protect the reactor facility from the type of operational accident postulated, a " ΔT scram system" has been installed since publication of the original Hazards Summary Report. This " ΔT scram system", intended as a backup to existing operating and administrative procedures, was used during about 100 transient tests of the previous C-core; during which time, it operated efficiently and reliably. The system consists of a fast response temperature sensing device located in the primary coolant stream, downstream of the heat exchanger mixing tee. The output signal from the sensor is fed to an adjustable down-scale, setpoint recorder located at the control center in the Spert III reactor control room. The recorder is connected to the reactor control rod scram circuit such that when the setpoint is reached, the reactor control rods are scrammed; and an audible and visual alarm is energized on the reactor annunciator panel. The setpoint is continuously adjustable in the range 0 to 20°F below an adjustable reference temperature. The reference temperature setting is, adjusted on the recorder after the primary coolant-moderator temperature equilibrium has been established just prior to and during nuclear operations. The time response of the instrument loop is about 1.0 second, and the time for the control rods to scram is less than 0.5 second. The shortest time for water to travel from the location of the temperature detector to the reactor is about 6 seconds (10,000 gpm/loop flow rate). Thus, in the event of a heat exchanger malfunction which results in a coolant-moderator temperature drop that exceeds the setpoint on the controller, ample time is available for the " ΔT scram system" to scram the reactor before cold water reaches the core.

4.3 Evaporative Cooling From Depressurization

The problems arising from a loss of primary system integrity will be considered separately, according to the conditions at the time of the break in the primary system. Should a large break occur near the end of an extended high-power nuclear test, partial melting of the core could result. This possibility is discussed in Section 6 (loss-of-coolant accident), and the lesser question of moderator cooling due to sudden depressurization which is not followed by core melting will be discussed here. The reactivity insertion caused by moderator cooling resulting from sudden depressurization of the primary system is, of course, dependent upon the rate of depressurization and the temperature at which it occurs.

It is possible to have an electrical or mechanical malfunction of the primary system blowoff and/or blowdown valves which could conceivably cause evaporative cooling. Calculations have been made which show that the maximum heat dissipation rate with both valves open is approximately 50 MW, based on the assumption that the system is at 500°F, the reactor is at or near critical, and both valves stick open at the same instant and remain open for an indefinite period. During the initial depressurization, until the

system reaches the saturation pressure, the cooling effect would take place in the pressurizer vessel and would not affect the coolant-moderator temperature in the reactor vessel. However, after the saturation pressure has been reached, further loss of water could result in evaporative cooling of the reactor coolant-moderator. If the same rate of heat dissipation is assumed, ie, 50 MW, it can be shown that during the initial coolant-moderator cooling in the reactor core, the reactivity insertion rate would be no greater than $\approx 5 \phi/\text{sec}$ for an initial system temperature of 500°F.

An operational accident of this type would be detected not only by the reactor power level rise but also by two plant alarms: the "pressurizer level low" alarm and the "primary pump suction pressure low" alarm. For this condition, the "pressurizer level low" alarm would be tripped approximately 10 seconds after the valves stick open. However, in addition to the evaporative cooling caused by the water loss through the stuck valves, when the pressurizer level drops to ≈ 1.5 feet above the heaters, the makeup pump automatically turns on and pumps ≈ 5 gpm of 70°F, 2500-psig water into the primary system near the reactor vessel tee. At a system flow rate of 1000 gpm, the cooling effect of the makeup water addition is to decrease the water temperature by about 0.05°F/sec at 500°F. This is a negligible reactivity contribution for the first several seconds. At higher flow rates, the initial cooling effect from makeup pump operation is even less. From this conservative calculation and the previous experience with ramp reactivity insertions in Spert III, it is concluded that at least five minutes would be available to scram the reactor before core damage would occur.

5. FAILURE OF FUEL RODS DURING AN EXCURSION

This section discusses the fuel rod rupture problem, its implications, and consequences. Available experimental evidence indicates that any severe consequences of such an accident are improbable. A planned continuing investigation of the problem at Spert and the means by which the probability of such an accident will be minimized are also discussed.

Although no tests are planned for the E-core which purposely lead to widespread destructive effects, the rupture of fuel rods during nuclear operation must be admitted as a possibility. The recent experiences with both the Spert I and the Western New York Nuclear Research Center research reactor [21] oxide cores indicate that the probability of appreciable damage to the E-core from such failures is small. However, it is recognized that the maximum range of operating conditions of the E-core is more severe than in either of these systems. Because of this and the uncertainties associated with postulated damage thresholds, it is difficult at this time to precisely define the ultimate safe operating conditions for the E-core.

Because of repeated thermal cycling of the fuel during testing of the E-core, cracking and flaking of the sintered fuel are expected to occur. It is possible for fuel chips to be displaced into the pellet-cladding gas gap, resulting in local pressure and/or thermal stresses which could lead to cladding failure. Recent thermal cycling experience with prototype PBF fuel rods [22]

indicates that fuel pellet cracking may be predominantly radial, with little spalling, and does not progress after the first few cycles.

If such a clad failure should occur, regardless of cause, immediate detection of the failure might be difficult. Since the primary means for detecting such a failure is by observation of the fission products in the system, immediate detection would, therefore, depend on the quantity of fission products released. If the cladding failure should remain undetected, subsequent waterlogging of the fuel rod would probably lead to its rupture during later operation of the facility.

For the case of fuel rod ruptures occurring during a reactor power excursion performed from initial conditions of sufficiently high power, pressure, and temperature, there is not sufficient information available to ensure that a cascade effect would not follow, leading to rupture of additional fuel rods, overpressurization of the primary system, and release of fission products to the atmosphere. However, this sequence would result in hazards significantly less than those following the loss-of-coolant with core meltdown, discussed in part 6 of this section, if the fission product inventory is small at the time of the accident. Since the maximum fission product inventory in the E-core will remain relatively small (for example, less than 500 curies of iodines) during all low initial-power tests, the radiation hazards associated with this accident are expected to be slight until high initial-power testing begins. Additional in-pile subassembly test data will be analyzed to define more clearly the hazards implicit in the fuel rod rupture accident prior to initiation of high initial-power tests.

That fuel similar to the E-core rods can sustain near-melting temperatures during short-period power excursion tests was shown by a series of Western New York Nuclear Research Center tests [21] in late 1964. In these tests, the fuel temperatures reached about 4000°F during each of 10 excursions; and the zirconium clad temperatures appear to have exceeded 1600°F. All fuel rod clad remained intact during these tests. This experience indicates that power excursion tests which do not result in fuel or clad melting will not result in fuel rupture, provided the fuel cladding is initially intact.

Based on recent experience with the Spert I oxide core, the probability of propagation of fuel rod rupture during power excursion tests appears to be small. The fuel rods comprising the E-core and the Spert I oxide core are similar, but are composed, respectively, of sintered UO_2 pellets and compressed UO_2 powder. During two recent ambient temperature and pressure short-period tests ($\tau_0 \approx 2.2$ msec, $\tau_0 \approx 1.5$ msec) [4, 5, 6], defective fuel rods in the Spert I core were ruptured, expelling the hot powdered UO_2 into the water moderator. These fuel rod ruptures resulted in rapid void formation which caused earlier self-limitation of the power excursions than would have otherwise occurred. Accompanying pressure surges were, in each case, less than 150 psig. It is probable that rupture of a fuel rod in the E-core during a power excursion test would have much the same effect, but might shut the reactor down less rapidly because the sintered E-core fuel fragments would offer a smaller effective heat transfer surface to the water.

In order to minimize the hazards implicit in the fuel rod rupture accident, the philosophy of the E-core test program will be to proceed from mild to the more severe tests in a cautious manner. Before each new and

more severe series of tests, a detailed examination of the safety aspects of the new tests will be made, based on analysis of available experimental data and theoretical models. Previous to high-initial-power testing of the E-core, a series of in-pile tests of E-core fuel will be performed in the Capsule Driver Core. These tests are expected to provide data directly applicable to the detailed analysis of the safety implications from rupture of fuel rods in the E-core.

To detect any fuel failure in Spert III, the analysis of primary system coolant samples for fission product nuclides is regularly performed on a routine basis. Additional water samples are analyzed after every test in which the possibility of fuel failure is believed to be significant. A visual fuel rod inspection program will also be maintained during the experimental program. In these inspections, selected hot-spot fuel rods will be removed from the core and checked for bowing, swelling, and discoloration. Additional visual inspections will be performed following any unusual temperature, pressure, or power behavior which may occur. However, the proximity of fuel rod failure cannot be precisely determined by nondestructive examination of the fuel. Therefore, one or more fuel rods will also be removed from the E-core for destructive examination at appropriate intervals during the test program.

If such a fuel rod failure or failures should occur in Spert III, the primary coolant will be drained into the hot-waste storage tank or directly into the leaching pond, depending on how much contamination it should contain. The system will then be cleaned and refilled; the damaged fuel rod or rods will then be replaced; and operations continued. No hazard to the facility or to personnel is envisioned for such an accident in any planned test.

6. LOSS-OF-COOLANT ACCIDENT

Rupture of the Spert III primary system is considered to be extremely unlikely; however, its occurrence cannot absolutely be established as impossible. It is conceivable that the stored thermal energy in the moderator-coolant could be released to the reactor building environment by a sudden, large primary system break; and if such occurred, it would probably destroy the conventional building housing the reactor system. In the event of such a large system break, it is also conceivable that the hydrodynamic forces associated with such a break could eject fuel rods from the reactor vessel, damage their clad, and release fission products. The loss of coolant which would accompany any major primary system rupture could result in exposure of the core, and since the E-core will undergo an estimated total energy release of about 20 MW-days during its test lifetime, the possibility of melting of fuel by core afterheat and the subsequent fission product release to the atmosphere must also be considered in a complete accident analysis. The specific test program envisioned and the operating limits for the reactor system must also enter into system accident analyses.

All of these various factors have been incorporated in the loss-of-coolant accident analysis performed for the low enrichment oxide core in Spert III. This analysis has been carried out by considering the following three specific

cases of the loss-of-coolant accident. The first consists of an assumed loss of coolant immediately following an extended high-power test series. Analysis has shown that fuel melting under such conditions would not occur; however, in view of the uncertainties which arise in the absence of conclusive experimental demonstrations, it has been assumed for the purpose of the accident analysis that melting did occur. This particular loss-of-coolant accident, because of its fission product inventory, has been termed the maximum referent accident. The second loss-of-coolant accident case discussed is an assumed loss of coolant immediately following a test with a 2.5 MW-hour energy release. Tests with such energy releases are somewhat less severe than the most energetic test planned in the experimental program, and have a fission product inventory and potential radiological dose which would not require any special meteorological control. The third loss-of-coolant accident case discussed is an assumed loss of coolant with the reactor system at temperature and pressure three hours following a high-power test, at which time meteorological conditions established for the test may have changed to something much less favorable for radiological hazards considerations.

It is assumed for all three loss-of-coolant accident cases that an offset shear at the bottom vessel tee results from an unspecified cause. No specific mechanism for initiating the loss-of-coolant accident is discussed for any of the three cases since none can be credibly envisioned.

6.1 Loss of Coolant Immediately Following a High-Power Test

Calculations have been made to estimate the power levels at which fuel or clad melting might occur following a loss-of-coolant accident at the end of 1/2 hour of steady-state power operation. These calculations (outlined in Appendix C) show that neither fuel nor clad melting should be expected for a loss-of-coolant accident following a 1/2-hour, steady-state power run of 30 MW and that only about 20 percent of the core cladding would melt as a result of such an accident following a 1/2-hour, steady-state power run of 60 MW. (Note that 60 MW for 1/2 hour is the maximum steady-state capability of the Spert III plant, and about 40 MW is the maximum expected capability of the E-core.)

The gamma afterheat source will rapidly approach its asymptotic value after several weeks of testing in Spert III assuming two high-power tests a day, five days a week for four weeks. Therefore, additional meltdown calculations were performed, based on the realistic estimate of two transient tests a day, five days a week, each test to be performed at a steady-state power level of 30 MW for 1/2 hour. These calculations also indicate no core melting should occur for a loss-of-coolant accident following the last test at the end of the fourth week of testing.

A radiological analysis of this loss-of-coolant accident case is presented in detail in Section VI. The analysis includes assessment of the hazards associated with a series of steady-state power operations of 30 MW for 1/2 hour repeated twice daily, five days a week for four weeks, followed by an assumed full core meltdown as a result of a postulated loss-of-coolant accident. In view of the results of the analysis, it has been concluded that high-power tests such as those immediately preceeding the postulated accident would be performed only under controlled meteorological conditions.

6.2 Loss of Coolant Immediately Following a Low-Power Test

A 2.5 MW-hr limit has been established as the maximum energy release permitted in a given test without the requirement of meteorological control. This limit is such that the maximum fission product inventory available for release, in the event of an accident following such a test, would not result in unacceptable radiological hazards.

In Section VI, results are presented in detail for an analysis of the hazards associated with a loss-of-coolant accident postulated to occur immediately following a low-power, 5 MW test. The 5 MW steady-state power has been established as that power level which will ensure conformity with the 2.5 MW-hr energy limit. Included in that discussion are the detailed assumptions predicated for the accident which, in general, is envisioned as an offset shear of the bottom vessel tee and subsequent expulsion and breakup of the reactor core such that fission products are released.

Since the accident is postulated to occur during nuclear operation, no personnel would be within 1/3 mile of the facility, and emergency activities at the control center would be dictated by the specific circumstances at the time. As shown in the radiological analysis, no undue hazards would exist from the occurrence of this accident; however, it is expected that evacuation of the control center would be required.

6.3 Loss of Coolant During Standby Plant Operation

The third loss-of-coolant situation postulated is one which could occur while the reactor was shut down between tests and while the plant was being maintained at operational temperature and pressure conditions. Since this situation prevails approximately 96 percent of the total time that the reactor system is at operational pressure, temperature, and flow conditions, the relative probability of an accident associated with it must be correspondingly increased.

Details of the radiological analysis performed for this accident postulation are presented in Section VI together with the specific detailed assumptions reflected in the analysis.

It is shown from the results of the analysis: first, that the radiological dose levels at the control center following such an accident would not constitute an undue hazard since the minimum time for degradation of the meteorological conditions specified for the test operations is about three hours and, second, that the dose levels at the site boundary are equally acceptable. Consequently, no unusual corrective action is necessitated by this accident postulation other than evacuation of the control center.

VI. RADIOLOGICAL HAZARDS EVALUATION FOR THE OXIDE CORE

This radiological safety analysis for the Spert III E-core is comprised of assessments of (a) a loss-of-coolant accident, with assumed total core meltdown immediately following a high-power test performed under favorable meteorological conditions; (b) a loss-of-coolant accident immediately following a low-power test performed under adverse meteorological conditions; and (c) a loss-of-coolant accident during standby plant operation under adverse meteorological conditions.

Case (a), the loss-of-coolant accident immediately following a high-power test, has been termed the maximum referent accident in this report since it involves the largest fission product inventory and, consequently, the largest potential radiological hazard. In view of the meteorological conditions imposed for high-power operation, it does not result in the highest radiological doses.

For each of the three accident cases, the analysis includes: some general discussion of the particular accident, the calculational assumptions, the calculated radiological hazards associated with the accident, and a discussion of the safety implications of the postulated accident and resultant potential doses. Also included at the outset is a brief survey of the calculational methods used to determine the various radiological dose values.

1. RADIOLOGICAL CALCULATIONS

The IBM-7040 Curie RSAC^[23] computer program and modified Sutton^[24] diffusion equations were used to compute the radiation exposures to personnel. The RSAC program, given the power history, release conditions, and prevailing meteorological conditions, computes radiation doses from 235 distinct fission product nuclides. For the calculations, the meteorological conditions shown in Table IV were assumed, with their associated diffusion parameters.

TABLE IV
SUTTON'S DIFFUSION PARAMETERS
FOR SELECTED METEOROLOGICAL CONDITIONS

<u>Variable</u>	<u>Inversion Condition</u>	<u>Lapse Condition</u>	<u>Lapse Condition</u>
Stability Index, n	0.5	0.2	0.2
Diffusion Parameter C_y, C_z	0.03	0.38	0.35
Wind Velocity, $u(\text{m/sec})$	1	3	7

1.1 Inhalation Dose

Inhalation dose to various organs of the body was obtained by the solution of Equation (1) utilizing the IBM-7040 Curie RSAC computer program. The elements that contribute to these doses are shown in Table V.

$$\frac{dA_k(t)}{dt} = -\lambda_b A_k(t) - \lambda_r A_k(t) + B_R \cdot C_F \cdot \frac{X}{Q} \cdot L(t) \cdot N_k(t) \quad (1)$$

where

$A_k(t)$ = number of atoms of the kth nuclide in the organ at any time (t)

λ_b = biological decay factor (sec^{-1})

λ_r = radioactive decay factor (sec^{-1})

B_R = breathing rate ($3.47 \times 10^{-4} \text{ m}^3/\text{sec}$)

$L(t)$ = leak rate function (sec^{-1})

C_F = fraction of inhaled material which reaches the critical organ

$N_k(t)$ = decay fission product activity at any time, (t)

X = airborne concentration (curies/meter³)

Q = fission product activity (curies/sec).

Integration of Equation (1) for any individual organ and any selected isotope yields the dose to the organ from that isotope.

Depletion of the radioactive cloud by deposition of fission products on the ground was neglected, resulting in increased dosages in every case. The calculations assumed appropriate (instantaneous or effectively constant) fission product releases at ground level, and doses were calculated along the center-line path of the cloud.

1.2 Cloud Gamma Dose

For the puff release, cloud gamma doses were calculated from Equation (2)[25].

$$\text{TID} = \frac{2\dot{Q}C_f}{\pi C_y C_z u X^{2-n}} e^{\left[\frac{-11^2}{C_z^2 X^{2-n}} \right]} \quad (2)$$

where

TID = total integrated dose (rem)

$\dot{Q}$ = source strength (curies)

u = wind velocity (m/sec)

X = distance downwind (m)

n = Sutton stability index

TABLE V
INHALATION DOSE CONTRIBUTING ISOTOPES

<u>Isotope</u>	<u>Total Body</u>	<u>Thyroid</u>	<u>Muscle</u>	<u>Lung</u>	<u>Liver</u>	<u>Kidney</u>	<u>Bone</u>	<u>Testes</u>
Ga	X			X	X	X		
As	X			X	X	X		
Br	X			X				
Rb	X		X	X	X			
Sr	X			X		X	X	
Y	X			X			X	
Zr	X			X	X		X	
Nb	X			X	X	X	X	
Mo	X			X	X	X		
Tc	X			X	X	X	X	
Ru	X			X		X	X	
Rh	X			X	X	X	X	
Pd	X			X	X	X		
Ag	X			X	X	X	X	
Cd	X			X	X	X		
In	X	X		X	X	X	X	
Sn	X	X		X	X		X	
Sb	X	X		X	X		X	
Te	X	X		X	X	X	X	X
I	X	X		X				
Cs	X			X	X	X	X	
Ba	X		X	X	X	X	X	
La	X			X	X		X	
Co	X		X	X	X	X	X	
Pr	X			X	X	X	X	
Nd	X			X	X	X	X	
Pm	X			X	X	X	X	
Sm	X			X	X	X	X	
Fu	X			X	X	X	X	
Gd	X			X	X		X	

C_y, C_z = Sutton diffusion coefficients

h = height of release (m)

C_f = conversion factor ($0.26 \frac{\text{rem m}^3}{\text{sec Curie}}$).

For the continuous release, cloud gamma doses were calculated by integration of Equation (2) over time as Equation (2a).

$$\text{TID} = \frac{2C_f}{2u C_y C_z X^{2-n}} e^{\left[\frac{h^2}{C_z^2 X^{2-n}} \right]} \int_{t_1}^{t_2} \dot{Q}' dt \quad (2a)$$

where

$\dot{Q}'$ = released source strength (curies/sec)

and other terms are described above.

The source strengths $\dot{Q}$ and $\dot{Q}'$ were obtained from the IBM-7040 RSAC code.

1.3 Deposition Gamma Dose

The relationship shown in Equation (3) was used to calculate the deposition gamma dose.

$$D = \frac{2}{\pi u C_y C_z X^{2-n}} \int_{t_1 \geq \frac{X}{u}}^{t_2} \int_{\frac{X}{u}}^{\frac{X}{u} + T} V_g L_r^2(t) dt dt \quad (3)$$

where

D = deposition gamma dose (rem) (= 0.01 m/sec, all cases)

$Q(t)$ = source strength (curie-sec)

V_g = deposition velocity (= 0.01 m/sec, all cases)

L_R = leak rate

C_f = conversion factor ($2.78 \times 10^{-3} \frac{\text{rem m}^2}{\text{sec-curie}}$).

For the case of a short release time (as compared to total exposure time), this reduces to Equation (3a)[26]

$$D = \frac{2Q V_g C_f}{\pi u C_y C_z X^{2-n}} \quad (3a)$$

which was applied for all cases of instantaneous fission product release. To obtain deposition gamma doses at the Spert Control Center for the accident

during standby plant operation, Equation (3) was used to obtain 4.5 min (evacuation time for Spert Control Center) deposition exposure results. For all other deposition dose calculations, however, it was assumed that the subject remained in the contaminated area for an infinite time after passage of the cloud, which significantly increased these resulting doses. All source strengths were obtained from the IBM-7040 RSAC computer program for fission product inventory.

1.4 Direct Radiation at Spert Control Center

The infinite time direct radiation dose received from fission products at the reactor building was calculated using Equation (4)[27].

$$D = \frac{C Q E_{\gamma} \mu_a B e^{-\mu d}}{4\pi d^2} \quad (4)$$

where

D = direct radiation dose

Q = source strength (curie-sec)

E_{γ} = average gamma energy (0.7 MeV/disintegration)

μ_a = energy absorption coefficients (m^{-1})

B = dose buildup factor

d = source to detector distance (m)

μ = linear absorption coefficient (m^{-1})

C = conversion factor to obtain dose.

Attenuation was considered to be due to air only.

2. HAZARDS -- LOSS OF COOLANT IMMEDIATELY FOLLOWING A HIGH-POWER TEST

Because of the uncertainties inherent in meltdown calculational techniques, it was deemed advisable to assess the consequences of the worst conceivable situation associated with a loss-of-coolant accident. Therefore, a radiological safety analysis was made for the case in which a complete core meltdown is assumed to follow the loss of coolant. For this worst case, the test immediately preceding the accident was assumed to release 30 MW-hr. (twice the maximum energy release to be permitted in any single test), and all exposures were calculated for an instantaneous release of radioactive fission products following the accident.

The assumption of "instantaneous" release in the form of a cloud was consistent with the expectation that such an occurrence would lead to damage of the Spert III reactor building by overpressurization during the coolant release.

The Spert III reactor has no containment structure. The assumption could be very conservative if the building which houses the reactor facility should remain intact during fission product release. The use of the instantaneous release assumption in the calculations, while not descriptive of the release situation associated with fuel meltdown following a loss of coolant, is conservative since it results in an overestimate of fission products in the radiological source term. The nuclear energy release from the test exceeds by a factor of two the maximum energy release permitted under the proposed reactor operating limits. Thus, the analysis of this accident case yields results which constitute an upper bound to the radiological doses which would accrue from any fuel meltdown occurring under favorable meteorological conditions and following a high-power test. In addition to the results obtained for lapse conditions, which is the primary objective of the analysis, doses have also been calculated for inversion conditions to illustrate the need for and degree of meteorological control required.

2.1 Assumptions

In order to present a realistic approach to the accident conditions, the analysis is based on the following assumptions, which represent the maximum high-power test history expected for the E-core.

- (1) The core has undergone two high-power tests [see assumption (2) below] per day, five days per week, for a total of 39 tests previous to the one which involves the accident. Thus, a total of 40 nuclear tests contributes to the fission product inventory at the time of the accident. This assumption, together with assumption (2), is consistent with the probable maximum power history for the core and leads to a fission product inventory which approaches the asymptotic value for an infinite number of test runs prior to the postulated accident.
- (2) Each test previous to the final one involved operation of the facility at 30 MW nuclear power for 1/2 hour. The final test, which is assumed to have been the second test on the fifth day of the fourth week of testing, involved operation of the facility at 30 MW nuclear power for one hour (double the allowable operating limit).
- (3) In the loss-of-coolant accident, which took place immediately following the last test, the entire core was melted, resulting in the release of 1 percent of the solid fission products, 50 percent of the halogens, and 100 percent of the noble gases instantaneously in the form of a radioactive cloud. The above release fractions were adopted in conformity with the guidelines set down in 10 CFR 100 and TID-14844.
- (4) The direct radiation hazard to personnel at the Spert control center, 1/2 mile from the reactor, was determined by assuming the source to be from 100 percent of the fission products at the reactor building above ground level.

2.2 Radiological Hazards

Tables VI through VIII show the inhalation doses to various organs of the body for two wind speeds under lapse conditions and one wind speed under strong

inversion conditions. These results show that, as may be expected, thyroid and bone receive significantly larger radiation doses than any other organ of the body. For all three meteorological conditions, thyroid doses exceed 100 rem at distances up to about a 1/2 mile, the distance between the Spert III reactor and the control center. In the case of inversion conditions, thyroid doses exceed 100 rem at a distance 10^5 meters (about 62 miles) downwind. The potential inhalation doses illustrate the need for meteorological control to ensure both lapse and favorable wind direction for tests of which the inventories would approach those of the case analyzed.

Table IX shows the cloud gamma dose as a function of distance from Spert III for each of the selected meteorological conditions. For distances greater than a 1/2 mile (800 m) downwind, all lapse doses are substantially less than 25 rem. The magnitude of the potential cloud gamma dose calculated for an inversion condition illustrates the need for the lapse requirement.

TABLE VI

ADULT INHALATION DOSES FOR 3 m/sec
WIND SPEED AND LAPSE ATMOSPHERIC CONDITIONS
(Loss of Coolant Following High-Power Test)

Distance from Spert III Meters	Total Body (rem)	Thyroid (rem)	Testes (rem)	Muscle (rem)	Lung (rem)	Liver (rem)	Kidney (rem)	Bone (rem)
100	1.35×10^1	6.11×10^3	2.90×10^{-1}	1.02×10^{-2}	1.20×10^2	4.79×10^0	6.24×10^0	3.42×10^1
500	7.45×10^{-1}	3.35×10^2	1.61×10^{-2}	6.18×10^{-4}	6.62×10^0	2.65×10^{-1}	3.47×10^{-1}	1.92×10^0
1,000	2.13×10^{-1}	9.58×10^1	4.64×10^{-3}	1.90×10^{-4}	1.90×10^0	7.61×10^{-2}	1.00×10^{-1}	5.57×10^{-1}
5,000	1.14×10^{-2}	5.06×10^0	2.57×10^{-4}	1.15×10^{-5}	1.02×10^{-1}	4.22×10^{-3}	5.61×10^{-3}	3.22×10^{-2}
10,000	3.14×10^{-3}	1.39×10^0	7.35×10^{-5}	3.30×10^{-6}	2.84×10^{-2}	1.21×10^{-3}	1.60×10^{-3}	9.42×10^{-3}
50,000	1.44×10^{-4}	6.26×10^{-2}	3.95×10^{-6}	1.82×10^{-7}	1.34×10^{-3}	6.58×10^{-5}	8.27×10^{-5}	5.10×10^{-4}
100,000	3.62×10^{-5}	1.56×10^{-2}	1.09×10^{-6}	5.21×10^{-8}	3.45×10^{-4}	1.86×10^{-5}	2.24×10^{-5}	1.43×10^{-4}

TABLE VII

ADULT INHALATION DOSES FOR 7 m/sec
WIND SPEED AND LAPSE ATMOSPHERIC CONDITIONS
(Loss of Coolant Following High-Power Test)

Distance from Spert III Meters	Total Body (rem)	Thyroid (rem)	Testes (rem)	Muscle (rem)	Lung (rem)	Liver (rem)	Kidney (rem)	Bone (rem)
100	6.84×10^0	3.09×10^3	1.46×10^{-1}	5.06×10^{-3}	6.06×10^1	2.42×10^0	3.15×10^0	1.72×10^1
500	3.77×10^{-1}	1.70×10^2	8.09×10^{-3}	2.94×10^{-4}	3.35×10^0	1.34×10^{-1}	1.74×10^{-1}	9.59×10^{-1}
1,000	1.08×10^{-1}	4.87×10^1	2.33×10^{-3}	8.85×10^{-5}	9.61×10^{-1}	3.84×10^{-2}	5.03×10^{-2}	2.77×10^{-1}
5,000	5.89×10^{-3}	2.63×10^0	1.30×10^{-4}	5.64×10^{-6}	5.25×10^{-2}	2.13×10^{-3}	2.82×10^{-3}	1.58×10^{-2}
10,000	1.66×10^{-3}	7.39×10^{-1}	3.72×10^{-5}	1.66×10^{-6}	1.49×10^{-2}	6.12×10^{-4}	8.14×10^{-4}	4.65×10^{-3}
50,000	8.16×10^{-5}	3.57×10^{-2}	2.03×10^{-6}	9.20×10^{-8}	7.45×10^{-4}	3.36×10^{-5}	4.37×10^{-5}	2.62×10^{-4}
100,000	2.14×10^{-5}	9.31×10^{-3}	5.76×10^{-7}	2.64×10^{-8}	1.98×10^{-4}	9.56×10^{-6}	1.21×10^{-5}	7.43×10^{-5}

TABLE VIII

ADULT INHALATION DOSES FOR 1 m/sec
WIND SPEED AND INVERSION^[a] ATMOSPHERIC CONDITIONS
(Loss of Coolant Following High-Power Test)

Distance from Spert III Meters	Total Body (rem)	Thyroid (rem)	Testes (rem)	Muscle (rem)	Lung (rem)	Liver (rem)	Kidney (rem)	Bone (rem)
100	2.59×10^4	1.17×10^7	5.57×10^2	2.06×10^1	2.30×10^5	9.18×10^3	1.20×10^5	6.61×10^4
500	2.29×10^3	1.03×10^6	5.03×10^1	2.13×10^0	2.04×10^4	8.14×10^2	1.09×10^3	6.07×10^3
1,000	8.01×10^2	3.58×10^5	1.78×10^1	7.87×10^{-1}	7.16×10^3	2.92×10^2	3.88×10^2	2.19×10^3
5,000	6.55×10^1	2.88×10^4	1.58×10^0	7.12×10^{-2}	5.95×10^2	2.60×10^1	3.42×10^1	2.04×10^2
10,000	2.14×10^1	9.35×10^3	5.53×10^{-1}	2.52×10^{-2}	1.97×10^2	9.15×10^0	1.18×10^1	7.14×10^1
50,000	1.41×10^0	5.97×10^2	5.45×10^{-2}	2.24×10^{-3}	1.37×10^1	7.91×10^{-1}	9.30×10^{-1}	6.03×10^0
100,000	4.08×10^{-1}	1.64×10^2	1.44×10^{-2}	7.85×10^{-4}	4.10×10^0	2.70×10^{-1}	3.06×10^{-1}	2.07×10^0

[a] Inversion doses are calculated to illustrate need for meteorological control. Test operations leading to this dose potential are not permitted.

TABLE IX

CLOUD GAMMA DOSES FOR THREE WIND
SPEEDS AND TWO ATMOSPHERIC CONDITIONS
(Loss of Coolant Following High-Power Test)

Distance from Spert III Meters	Dose (1 m/sec) Inversion ^[a] (rem)	Dose (3 m/sec) Lapse (rem)	Dose (7 m/sec) Lapse (rem)
100	1.15×10^6	9.21×10^2	6.45×10^2
500	5.59×10^4	2.71×10^1	1.90×10^1
1,000	1.49×10^4	5.96×10^0	4.19×10^0
5,000	5.98×10^2	1.73×10^{-1}	1.23×10^{-1}
10,000	1.23×10^2	3.40×10^{-2}	3.20×10^{-2}
50,000	3.11×10^0	1.00×10^{-3}	1.00×10^{-3}
100,000	5.50×10^{-1}	Negligible	Negligible

[a] Inversion doses are calculated to illustrate need for meteorological control. Test operations leading to this dose potential are not permitted.

Deposition gamma doses are shown in Table X for each of the selected weather conditions. Again, lapse deposition doses are substantially less than 100 rem 1/2 mile downwind and illustrate the desirability of both lapse and favorable wind direction control.

TABLE X

DEPOSITION GAMMA DOSES FOR THREE WIND
SPEEDS AND TWO ATMOSPHERIC CONDITIONS
(Loss of Coolant Following High-Power Test)

Distance from Spert III Meters	Dose (1 m/sec) Inversion ^[a] (rem)	Dose (3 m/sec) Lapse (rem)	Dose (7 m/sec) Lapse (rem)
100	6.12×10^6	3.20×10^3	1.61×10^3
500	5.36×10^5	1.76×10^2	8.86×10^1
1,000	1.88×10^5	5.02×10^1	2.55×10^1
5,000	1.47×10^4	2.63×10^0	1.38×10^0
10,000	4.48×10^3	6.68×10^{-1}	3.88×10^{-1}
50,000	1.32×10^2	2.65×10^{-2}	1.81×10^{-2}
100,000	1.74×10^1	Negligible	Negligible

[a] Inversion doses are calculated to illustrate need for meteorological control. Test operations leading to this dose potential are not permitted.

The calculated direct radiation dose to personnel at the Spert Control Center was determined to be 1.35 rem following the accident.

Table XI shows the radiation doses that may be expected at the site boundary for three southwest wind directions as a function of wind speed for each of the three assumed atmospheric conditions. Note that the two lapse conditions with these wind directions correspond to meteorological controlled conditions; and that for these lapse directions, all corresponding site boundary doses are substantially less than 25 rem.

2.3 Safety Implications

The results calculated for the inversion conditions (Tables VI through XI) indicate the necessity for meteorological control for those test conditions which could produce a fission product inventory such that its release would lead to potential doses of this magnitude.

The results calculated for lapse conditions are summarized in Table XI. In all cases, the doses at the site boundary under lapse conditions are much less than those set forth in 10 CFR 100 to define an exclusion zone. In addition, they are less than those calculated in the approved Spert III Hazards Summary Report.

TABLE XI

SUMMARY OF RADIATION DOSES AT THE SITE
BOUNDARY FOR THREE SOUTHWEST WIND DIRECTIONS
(Loss of Coolant Following High-Power Test)

Organ Effected	Wind Speed (m/sec)	Atmospheric Condition	Dose at 190° (rem)	Dose at 220° (rem)	Dose at 250° (rem)
External Gamma (Cloud + Deposition)	1	I[a]	1.3×10^2	2.2×10^2	3.8×10^2
	3	L[b]	2.4×10^{-2}	3.7×10^{-2}	6.2×10^{-2}
	7	L	1.8×10^{-2}	2.6×10^{-2}	4.1×10^{-2}
Adult Thyroid Inhalation	1	I[a]	5.7×10^2	8.1×10^2	1.2×10^3
	3	L	5.7×10^{-2}	8.2×10^{-2}	1.3×10^{-1}
	7	L	3.0×10^{-2}	4.6×10^{-2}	7.2×10^{-2}

[a] Inversion doses are calculated to illustrate need for meteorological control. Test operations leading to this dose potential are not permitted.

[b] L denotes lapse conditions.

3. HAZARDS -- LOSS OF COOLANT IMMEDIATELY FOLLOWING A LOW-POWER TEST

A limit of 2.5 MW-hr on nuclear energy release has been established such that tests having lesser releases can be performed with no meteorological control requirements. To ensure that this 2.5 MW-hr is indeed a sufficiently conservative limit, an analysis of a loss of coolant following test operations involving energy releases of 2.5 MW-hr has been made. The meltdown analysis presented in Appendix C clearly establishes that a loss-of-coolant accident in this system following tests with 2.5 MW-hr energy releases will not lead to core meltdown. However, when the system is at certain conditions of pressure and temperature, clad rupture as a consequence of hydrodynamic forces during the postulated loss of coolant cannot be completely ruled out. Consequently, the analysis has been made on the postulate that clad and fuel rupture results in an immediate release of gaseous fission products from the fuel rod fill-gas spaces followed by slow release of additional fission products during subsequent oxidation of the hot UO_2 fuel.

The assumption of an "instantaneous" partial inventory puff release was made for this accident since, as in the more severe meltdown accident, thermal energy stored in the primary system could destroy the reactor building as well as eject fuel from the primary vessel.

3.1 Assumptions

The analysis is based on the following assumptions:

- (1) The fission product inventory in the core corresponds to that obtained from assumptions (1) and (2) of Section VI-2.1, page 42, except that the final test, which is assumed to have been the second test on the fifth day of the fourth week of

testing, involved operation of the facility at 5 MW nuclear power for 1/2 hour (ie, a 2-1/2 MW-hr energy release).

- (2) By some unspecified means, an offset shear of the entire bottom tee of the reactor vessel, or equivalent damage, occurs.
- (3) The entire reactor core is expelled from the pressure vessel by hydrodynamic action, and 100 percent of the fuel is exposed to the atmosphere (without melting) by clad cracking.
- (4) 2×10^{-4} percent of all noble gases and 7×10^{-4} percent of all available halogens are "instantaneously" released from the fill-gas spaces of the fuel rods into the atmosphere. These release fractions are based on extrapolation, to the conditions of Spert III power operation, of recent ORNL data^[28] in which the diffusion of fission products from UO₂ into a helium atmosphere was measured for several UO₂ temperatures. The data were extrapolated as T⁻¹ to 600°C, the assumed fuel surface temperatures during previous high-power operation, to obtain appropriate release fractions.
- (5) The exposed (initially 620°F average temperature) fuel oxidizes to U₃O₈ at a rate of 0.3 percent/min and, of the oxidized fuel, 12 percent of the halogens, 4.5 percent of the noble gases, and 0.3 percent of the solids are released during a one-minute interval. These oxidation and release rates are based on extrapolation, to the conditions of the assumed accident, of recent ORNL data^[29] in which oxidation rates and corresponding fission product release rates are tabulated for UO₂ heated in air. The data are based on five-hour heating cycles at constant temperature; however, for the postulated loss-of-coolant accident, thermodynamic calculations show that the exposed fuel cools quite rapidly. In less than one minute, the exposed fuel is calculated to cool below the threshold temperature for significant oxidation reaction (270°C)^[30]; and within about thirty minutes the fuel temperature is below 100°C.

3.2 Radiological Hazards

In order to obtain information indicative of the maximum potential doses in the downwind direction, a strong inversion and a 1 m/sec wind speed were specified as the meteorological conditions for the two calculations performed. These two calculations yielded doses at the Spert control center (about 1/2 mile from the reactor) for an exposure time of 4.5 min and at the nearest site boundary (about 6.4 miles from the reactor) where the subject was assumed to remain in the path of the cloud for 70 years, ie, infinite exposure.

At the control center, the time of exposure can be controlled by personnel evacuation; the 4.5 min exposure time is in conformity with the results of evacuation tests performed at Spert. Since the reactor is continuously monitored by plant engineers at the control center when the system is at the pressure and temperature required for the accident to occur, evacuation of the control center would be completed before arrival of the radioactive cloud (assumed to be moving at 1 m/sec) following the postulated accident. Therefore, the 4.5 min control center (1/2 mile) personnel exposure time is extremely conservative.

The nearest site boundary doses are also overly conservative since the subject was assumed to remain in the exposure area for essentially an infinite time after the accident, and no credit was taken for depletion of the radioactive cloud by deposition of fission products on the ground. Both of these assumptions significantly increase the resulting computed radiation doses.

Table XII lists the calculated radiation doses for the two components of the assumed accident: the small puff release from noble gases and halogens in

TABLE XII
RADIATION DOSES (REM) FOR LOSS-OF-COOLANT
ACCIDENT IMMEDIATELY FOLLOWING A LOW-POWER TEST

	<u>PUFF RELEASE</u>	
	<u>1/2 Mile</u>	<u>6.4 Mile[a]</u>
Cloud Gamma	3.0×10^{-2}	4.0×10^{-4}
Deposition Gamma	3.3×10^{-1}	7.0×10^{-3}
<u>Inhalation</u>		
Total Body	1.0×10^{-2}	≈ 0
Thyroid	6.8×10^0	4.0×10^{-2}
Lung	9.0×10^{-2}	≈ 0
<u>CONTINUOUS RELEASE</u>		
	<u>1/2 Mile[b]</u>	<u>6.4 Mile[a]</u>
Cloud Gamma	6.3×10^{-1}	3.0×10^{-1}
Deposition Gamma	3.6×10^{-2}	6.2×10^{-1}
<u>Inhalation</u>		
Total Body	5.2×10^{-1}	1.1×10^{-2}
Thyroid	2.2×10^2	4.4×10^0
Testes	1.7×10^{-2}	3.6×10^{-4}
Muscle	7.1×10^{-4}	1.6×10^{-5}
Lung	5.1×10^0	1.1×10^{-1}
Liver	3.3×10^{-1}	7.2×10^{-3}
Kidney	3.2×10^{-1}	6.9×10^{-3}
Bone	2.3×10^0	4.9×10^{-2}

[a] Nearest site boundary.

[b] Computed dose for 4.5 minute initial exposure after arrival of the cloud.

the fuel gas-gap spaces, and the steady release during subsequent oxidation of exposed fuel. The puff release exposures are essentially negligible at all distances greater than a few hundred yards, but the calculated doses at the control center from the oxidizing UO_2 are sufficiently high that evacuation of the downwind area to a distance of approximately five miles would be required in the event of such an accident. An additional factor of conservatism in the results calculated for continuous release doses is the assumption of 100 percent fuel exposure as a result of clad and fuel breakup. These results would be reduced in direct proportion to reduction of the 100 percent exposure.

3.3 Safety Implications

The magnitude of the radiological doses presented in Table XII, together with the degree of conservatism inherent in the calculations and the demonstrated evacuation potential of the Spert area, establish the acceptability of hazard for tests having an expected energy release not to exceed 2-1/2 MW-hrs regardless of meteorological conditions.

4. HAZARDS -- LOSS OF COOLANT DURING STANDBY OPERATION

In the preceding analyses it has been shown that, under the operating and administrative procedures planned for testing of the E-core, the hazards associated with a loss-of-coolant accident postulated to occur immediately after any nuclear test are acceptable. However, approximately 96 percent of the time that the Spert III plant is at operating conditions of pressure, temperature, and flow, it is in a standby condition rather than engaged in or immediately following nuclear operation. Statistically, the possibility of occurrence of a loss-of-coolant accident during such standby operation of the plant is increased by the relatively large fraction of the time spent at standby. Also, prevailing weather conditions can change, after a high-power test, from lapse to strong inversion conditions (the estimated minimum time for such a change is about three hours).

In view of these facts, an analysis has been performed to assess the hazards associated with a loss of coolant and attendant expulsion and breakup of the core which could occur under uncontrolled meteorological conditions following the completion of a series of high-power tests.

4.1 Assumptions

The analysis is based on the following assumptions:

- (1) The fission product inventory in the core corresponds to that obtained from assumptions (1) and (2) of Section VI-2.1, page 42, except that the last test (30 MW for 1 hour) was completed with the plant intact. Three hours have elapsed, however, since completion of that test, and meteorological conditions have changed to strong inversion with a wind speed of 1 m/sec. The plant remains on standby at 550°F and 2500 psig, nonnuclear operation.
- (2) The accident, fuel exposure, oxidation and release rates discussed in assumptions (2) through (5) of Section VI-3.1, above, are essentially unchanged. Radiological dose calculations were

TABLE XIII

RADIATION DOSES (REM) FOR LOSS-OF-COOLANT
ACCIDENT DURING STANDBY PLANT OPERATION

	<u>PUFF RELEASE</u>	
	<u>1/2 Mile</u>	<u>6.4 Mile[a]</u>
Cloud Gamma	3.0×10^{-2}	4.0×10^{-4}
Deposition Gamma	3.3×10^{-1}	7.0×10^{-3}
<u>Inhalation</u>		
Total Body	1.0×10^{-2}	≈ 0
Thyroid	6.8×10^0	4.0×10^{-2}
Lung	9.0×10^{-2}	≈ 0
<u>CONTINUOUS RELEASE</u>		
	<u>1/2 Mile[b]</u>	<u>6.4 Mile[a]</u>
Cloud Gamma	9.6×10^{-1}	4.4×10^{-1}
Deposition Gamma	5.4×10^{-2}	9.0×10^{-1}
<u>Inhalation</u>		
Total Body	7.9×10^{-1}	1.6×10^{-2}
Thyroid	3.6×10^2	7.2×10^0
Testes	2.05×10^{-2}	4.6×10^{-4}
Muscle	7.4×10^{-4}	1.7×10^{-5}
Lung	7.4×10^0	1.5×10^{-1}
Liver	3.6×10^{-1}	8.2×10^{-3}
Kidney	3.7×10^{-1}	8.3×10^{-3}
Bone	2.5×10^0	5.7×10^{-2}

[a] Nearest site boundary.

[b] Computed dose for 4.5 minute initial exposure after arrival of the cloud.

performed for the same cases as in Section VI-3, and the same degree of conservatism is inherent in the results.

4.2 Radiological Hazards

Table XIII lists the calculated radiation doses for the two components of the assumed accident: the small puff release from noble gases and halogens

in the fuel gas-gap spaces, and the steady release during subsequent oxidation of exposed fuel. The puff release exposures are essentially negligible at all distances greater than a few hundred yards, but the potential doses at the control center from the oxidizing UO_2 dictate that evacuation of the control center and the downwind area to a distance of about 5 miles be planned in the event of such an accident.

4.3 Safety Implications

The radiological doses presented in Table XIII illustrate that a loss-of-coolant accident following high-power testing, at the earliest time that weather conditions could become unfavorable, would not lead to an unacceptable radiological hazard.

5. CONCLUSIONS

The radiological analyses have shown that, within the administrative and operational safeguards specified for planned testing of the Spert III E-core, any postulated loss-of-coolant accident would result in acceptable radiological hazards which are in all cases less than those established in the approved Spert III Hazards Summary Report. In addition, all of the dose levels at the nearest site boundary are less than those specified in 10 CFR 100 for establishment of an exclusion area.

The analyses consisted of assessing the hazards associated with maximized core fission product inventory and total release fractions, and establishing that an accident with such a system would result in acceptable radiological hazards. The assumption of these extreme conditions of fission product inventory and release fractions constituted a gross overestimate of severity of the situation since the pre-accident operations postulated exceed the proposed operating limits and also since the analytical calculations indicate that core melting will not occur even for the accident postulated. Even with such over-conservative assumptions on the severity of the situation, the radiological hazards calculated did not exceed those established in 10 CFR 100 for the establishment of an exclusion zone and, hence, are deemed acceptable. It then follows that a loss-of-coolant accident, immediately following any planned test performed under controlled meteorological conditions would result in no greater hazard.

In addition, a postulated loss-of-coolant accident immediately following the maximum steady-state power test to be permitted under uncontrolled meteorological conditions was shown to result in an acceptable radiological hazard. It then follows from this and the previous analysis that any loss-of-coolant accident, occurring immediately after any planned test, would not result in unacceptable hazards.

The case of rapid change from lapse to strong inversion meteorological conditions following a high-power test was also analyzed. For this case, a loss-of-coolant accident was considered to occur three hours after completion of a high-power test in which fission product inventory was maximized. It was shown that such an accident would result in acceptable radiological hazards, and it follows from the three cases considered that a loss-of-coolant accident

occurring at any time during or after any planned test would not result in unacceptable radiological hazards.

On this basis it is concluded that the oxide core described herein can be operated for the conduct of this test program without exceeding the limits established by AEC policy.

VII. CONCLUSIONS

As a supplement to the Spert III Hazards Summary Report, this addendum has discussed recent changes in the Spert III facility and presented an outline of the Spert III oxide core program, an analysis of the static and kinetic characteristics of the core, and an evaluation of the hazards associated with the test program.

The Spert III oxide core testing program will represent an important phase of the Spert experimental reactor safety program. As described in this report, all testing will be conducted in accordance with the standard Spert procedural controls for reactor testing. All nuclear operation will be conducted remotely, with no personnel closer than about 1/2 mile away from the reactor.

The accident analysis indicates that the loss-of-coolant accident is potentially the most hazardous accident for the oxide core, particularly if melting of the core with substantial release of fission products to the atmosphere should follow such an accident. Meltdown calculations have been made using realistic estimates of the nuclear power history for the E-core experimental program and conservative (safe side) assumptions with respect to the requirements for obtaining fuel melting. These calculations show that fuel melting will not occur during a loss-of-coolant accident following any planned test. However, in order to evaluate the worst conceivable consequence of this accident, a radiological safety analysis was performed for an assumed total core meltdown initiated by a postulated loss-of-coolant accident. Since this accident involved the largest fission product inventory in the core and consequently the largest potential radiological hazard, it has been termed the maximum referent accident. However, because of the meteorological restrictions employed, it does not lead to the maximum dose levels. A realistic estimate of the probable maximum nuclear power history for the oxide core was used in these calculations to establish the fission product source.

Two additional loss-of-coolant accidents were analyzed to determine the radiological hazards in the absence of core melting, but under adverse meteorological conditions. The first was assumed to occur immediately following a low-power test having a 2.5 MW-hr energy release and for which controlled meteorological conditions would not be required. The second was assumed to occur three hours after an extended high-power test, at which time the weather was assumed to have changed from lapse to strong inversion conditions and the plant was in nonnuclear standby operation. In both of these analyses the fission product inventory assumed prior to the specific test and accident was the same as that postulated for the maximum referent accident.

The calculated radiological doses arising from these three postulated accidents establish that, for any credible accident which may arise during the planned program, associated radiological hazards will be less than those given in the original Spert III Hazards Summary Report. And, in addition, the calculated doses at the nearest site boundary for each of the three accidents postulated in this report are less than the dose levels cited in 10 CFR 100 for the establishment of an exclusion zone.

A number of other accidents were also discussed; however, in all cases the consequences of each were assessed to be less than those three for which the detailed radiological analyses were conducted.

The radiation hazards for the maximum referent accident of this report were also compared, in Appendix D, with the hazards arising from the accident assumed in the original Hazards Summary Report. This comparison shows that, with the exception of cloud gamma doses calculated for inversion conditions, the calculational methods of the original SAR give results substantially in agreement with results obtained by the methods of this addendum, provided that the two are applied on the same basis. For cloud gamma doses with inversion conditions, the calculational methods of this report give results significantly more conservative than those of the original SAR at times less than about 10^4 sec after release of the cloud. The result of this added conservatism leads to the requirement of controlled meteorological conditions for those operations which could otherwise be postulated to lead to the excessive cloud gamma doses. The use of more recently accepted calculational constants and the inclusion of deposition gamma doses in this addendum also increase the calculated total doses.

In view of the radiological hazards calculated for these various situations, the criteria set forth in 10 CFR 100 for the establishment of exclusion zones and the results of extensive testing in the recent Spert I oxide core, it is concluded that the Spert III oxide core testing program can be conducted in a manner consistent with the general policy of the AEC to protect government and contractor personnel and the general public against undue exposure to radiation and all other potential health and safety hazards which may arise in the conduct of nuclear research programs.

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APPENDIX A

DESIGN DATA SUMMARY

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APPENDIX A

DESIGN DATA SUMMARY

A summary of the characteristics of the reactor and major plant equipment is tabulated in this section. The nuclear characteristics of the reactor as presented represent those used for nuclear design of the core. The numbers quoted are calculated values and have not been verified by experiment. The core has been designed for transient testing at the maximum reactor operating conditions, ie, high steady-state nuclear power at 2500 psig and 550°F. A fuel loading sufficient to provide an excess reactivity of about 3.48\$ for experimental purposes under these conditions was selected for the design criterion.

The information included here, such as core loading, core composition, core volume, and heat-transfer area, is based upon a 64-fuel-assembly core. It is important that the reader realize that this loading is not necessarily representative of the core loading which will be used throughout the experimental program. The core loading for a particular series of experimental investigations can only be determined by experiment and will be dependent upon the particular reactor parameters which are under investigation.

1. GENERAL REACTOR DESIGN DATA

Type -- pressurized-water experimental reactor suitable for boiling or pressurized water experiments
Function -- experimental, reactor transient-behavior, and safety studies
Moderator -- H₂O
Coolant -- H₂O
Neutron energy -- thermal
Core type -- heterogeneous, rod type
Heat Removal -- Plant, 60 MW (limited)
Heat Power -- Core, 40 MW (max)
Power density (max at 40 MW) -- 308 kW/liter of UO₂
Design pressure and temperature (max) -- 2500 psig at 650°F

2. REACTOR-COMPONENT DESIGN DATA

Core -- (Operational Loading)
Configuration -- cylinder (approximately)
Diameter -- ≈ 30 in.
Active height -- 38.3 in.
Volume -- 2.21×10^4 in.³
Composition
Water -- 6.7 wt%
Stainless steel -- 22.5 wt%
UO₂ -- 70.8 wt%

U -- 62.4 wt%
U-235 -- 3.0 wt%
Fuel load, U-235 -- 57.4 kg
Number of fuel assemblies, all types -- 64

Fuel Assemblies

Type -- rectangular, rod type
Number (Operational Loading)
25 rod assembly -- 52
16 rod assembly -- 4
control rod assembly -- 8
Overall dimensions
25 rod assembly -- 2.975 x 2.975 x 52-3/4 in.
16 rod assembly -- 2.476 x 2.476 x 52-3/4 in.
control rod fuel section -- 2.496 x 2.496 x 45-41/64 in.
Fuel content
25 rod assembly -- 962.5 g U-235
16 rod assembly -- 616 g U-235
control rod assembly -- 616 g U-235
Pitch -- 0.585 in. (square)
Flow area (inside) / assembly
25 rod assembly -- 4.29 in.²
16 rod assembly -- 3.16 in.²
control rod assembly -- 2.80 in.²
Heat-transfer area (active fuel length)
25 rod assembly -- 1402 in.²
16 rod assembly -- 897 in.²
control rod assembly -- 897 in.²

Fuel Rods

Type -- cylindrical
Materials
Fuel tube -- stainless steel, Type 348
Pellets -- 4.8 percent enriched sintered UO₂
Fuel pellet dimensions
Long pellet -- 0.420 in. OD x 0.766 in.
Short pellet -- 0.420 in. OD x 0.511 in.
Rod dimensions
Diameter -- 0.466 in. OD
Wall thickness -- 0.020 in.
Overall length -- 40.8 in. (excluding end plugs)
Active length -- 38.3 in.
U-235/rod -- 38.6 g
Gas gap -- 0.003 in. helium

Control Rods

Type -- rectangular: upper-section is absorber material; lower section is a fuel subassembly
Number -- 8
Composition
Absorber section -- 1.35 wt% boron-10 in Type 18-8 stainless steel; 0.186-in.-thick hollow square box

Fuel section -- see fuel assembly section on the preceding page
Overall dimensions

Absorber section (nom) -- 2.496 x 2.496 x 45-31/32 in.

Fuel section (nom) -- 2.496 x 2.496 x 45-41/64 in.

Travel -- 45 in. maximum

Average scram time -- 0.350 sec

Withdrawal rates

Fast -- 17.4 in./min

Intermediate -- 11.5 in./min

Slow -- 5.64 in./min

Transient Rod

Type -- cruciform: lower-section absorber material, upper-section AISI
Type 347 stainless steel

Number -- 1

Composition (absorber section) -- 1.35 wt% boron-10 in 18-8 stainless steel

Dimensions

Thickness of blades -- 3/16 in.

Blade width -- 5.125 in.

Absorber section length -- 38 in.

Travel -- 45 in. max.

Average drop time -- 0.2 sec

Control Rod Drives

Type -- Acme nut and screw; air pressure maintains rod in contact with screw
and scrams rods

Number -- 5

Motor type -- 480 V, 3 ϕ , 3 speed, constant torque

Motor rating -- 1 hp

Reactor Vessel

Construction

Shell -- layer type

Top head -- forged, full-opening flanged hemispherical

Bottom head -- layer type

Inside diameter -- 48 in.

Shell thickness (including clad) -- 3.25 in.

Head thickness -- 3.5 in.

Materials

Outer shellplate -- VMS 1146 plates 1/4 in. thick

Inner shellplate -- ASTM 225 Gr B with a 1/8-in.-thick 304L stainless
steel clad

Top head -- VMS-1146 manually clad with Type 309 Cb stainless
steel

Design pressure

Working pressure -- 2500 psig

Inst. shock pressure -- 3500 psig

Design temperature -- 700°F

Overall length, includes head -- $\approx$ 19 ft, 11-1/8 in.

Maximum diameter of head flange -- 5 ft, 7 in.

Number of thermal shields -- 5 (including reactor flow skirt)

Total thickness of shields -- 5.75 in.

Average length of shields -- 45.5 in.
Inside diameter of inner shield -- 32 in.
Weight of vessel

Shell	55,784 lb
Head	10,606 lb
Bottom tee	5,580 lb
Shield	11,430 lb
Total	83,400 lb

Insulation -- 4 in. of foamglass

3. THERMAL DESIGN DATA -- 64-ASSEMBLY CORE

Coolant -- deionized water
Heat power (max) -- 40 MW
Coolant flow (max) -- 20,000 gpm
Coolant passes through core -- one
Velocity in core -- (max flow) -- 23.6 ft/sec
Heat transfer area -- 580.9 ft²
Average heat flux (core) -- 2.35×10^5 Btu/hr/ft² at 40 MW

Primary Pumps

Type -- single stage, single suction, double volute, canned rotor
Number -- 4
Capacity, ea. -- 5000 gpm
Net differential head -- 328 ft of H₂O at 650°F, 2500 psig
Motor
 Rating -- 500 hp
 Type -- 480 V, 3Ø, 60 cycle
 rpm -- 3550
Weight, ea. -- 8500 lb
Material -- stainless steel, Type 304L

Heat Exchangers

Type -- U-tube, evaporative
Number -- 2
Heat transfer capacity, ea. -- 30 MW
Tube side fluid -- primary coolant, deionized H₂O
Shell side fluid -- deionized H₂O
Design pressure
 Tube side -- 2500 psig
 Shell side -- 75 psig
Design temperature
 Tube side -- 668°F
 Shell side -- 320°F
Heat-transfer surface -- 1940 ft²
Number of tubes -- 367 U's
Tube size, OD -- 5/8 in., 16 BWG
Tube straight length -- 191 in.
Tube pitch -- triangular
Pressure drop -- 10 psi

Materials

Tubes -- stainless steel, Type 304L
Tube sheet -- A-266-5S, CL 2, clad with stainless steel, Type 304L
Channel -- A-266-5S, CL 2, clad with stainless steel, Type 304L
Channel cover -- A-266-5S, CL 2, clad with stainless steel, Type 304L
Shell -- A-212, Gr A

Weight

Full -- 43,200 lb
Empty -- 23,400 lb
Tube bundle -- 11,600 lb

Pressurizer Vessel

Number -- 1

Design -- Section VIII ASME Boiler and Pressure Vessel Code

Design Pressure -- 2750 psig

Design Temperature -- 700°F

Dimensions

overall height -- 15 ft, 2-11/16 in.

overall diameter -- 3 ft, 3.3 in.

Volume -- $\approx 83.0 \text{ ft}^3$

Shell thickness -- 3.15 in.

Head thickness -- 2.9 in.

Materials

Backing -- A-212, Gr B, FBX

Clad -- nickel 200 (ASTM B-162), 0.100 in.

Heaters -- electric immersion, 480 V, 3Ø

Number -- 16

Capacity, ea. -- 12 kW

Total capacity -- 192 kW

Weight (empty) -- $\approx 26,000 \text{ lb}$

Primary System Piping

Type -- centrifugally-cast stainless steel as per ASTM-A-362-52T, modified.
Wall thickness

10 to 16 in. -- schedule 160

8 in. -- schedule 140

Material -- stainless steel, Type 304L

Insulation -- 3 in. of 85 percent Magnesite

Makeup Pump

Number -- 1

Type -- triplex, direct flow, vertical reciprocating pump

Piston diameter -- 7/8 in.

Stroke -- 2-1/2 in.

Capacity rating (max)

gpm -- 5

Head -- 3000 psi

Motor

Type -- 480 V, 3Ø, 60 cycle, gear motor

hp -- 10

rpm -- 280

4. SECONDARY-COOLANT SYSTEM

Pumps

Type -- single stage, single suction, vertical split case, centrifugal pump

Number -- 2

Capacity, ea. -- 250 gpm

Total discharge head -- 200 ft of H₂O at 70°F

Motor

Rating -- 20 hp

Type -- 480 V, 3Ø, 60 cycle

rpm -- 3500

Weight -- 800 lb

Materials -- stainless steel, Type 316

Piping

All piping to the upstream side of the heat-exchanger level-control valves

Material -- stainless steel

Type -- seamless or welded

Wall thickness -- schedule 40

All other piping

Material -- carbon steel

Type -- seamless

Wall thickness -- schedule 40

5. AUXILIARY EQUIPMENT

Water-treatment System

Softner

Type -- sodium zeolite

Capacity/cycle -- 50,000 gal

Flow rate (max) -- 90 gpm

Deionizer

Type -- mixed bed

Capacity/cycle -- 20,000 gal

Flow rate (max) -- 100 gpm

Effluent water resistivity -- 1.0 + megohm-cm

Deionized-water storage tank

Construction -- carbon steel, rubber-lined, shallow-dished heads

Capacity -- 1200 gal

Outside diameter -- 108 in.

Length straight section -- 300 in.

Raw-Water Primary Pump Cooling System

Cooling tower

Type -- double flow, forced-draft evaporative

Water flow rate -- 300 gpm

Fan motor -- 5 hp, 480 V, 3Ø

Overall dimensions

Length -- 16 ft, 0 in.

Width -- 8 ft, 6 in.

Cooling Water Pump

Type -- close-coupled, vertical turbine pump

Number -- 1

Capacity -- 300 gpm

Total disc. head -- 60 ft H₂O

Motor

Type -- 480 V, 3Ø

Rating -- 7-1/2 hp

rpm -- 1770

Poison Shutdown System

Poison -- boric acid solution

Addition method -- reactor primary system "makeup" pump

Reactivity compensated by poison addition -- 17.80\$

Boric acid concentration required -- 4.97 g/l

Time for total addition -- 107 minutes

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CONTROL SYSTEM DESIGN

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APPENDIX B

CONTROL SYSTEM DESIGN

1. INTRODUCTION

This appendix constitutes a brief review of the philosophy and features of the Spert III reactor control system with emphasis on the changes that have been made since the previous facility [9] and hazards reports [7].

From a general viewpoint, the primary requirements for the control system are that no hazard to personnel shall arise from control system operation and that risks to reactor equipment shall be minimized. The control system must provide proper manipulation of the control units and must furnish information on the operation performed. It also must indicate equipment failures or improper operation and be designed in such a manner that any component failure which constitutes loss of control will automatically shut the reactor down.

2. GENERAL DESIGN FEATURES

Except for the control rod itself, the control system for the Spert III E-core is essentially the same as was used in the C-core. The same mechanical design criteria that were used for the Spert III C-core will be used for the Spert III E-core. The extensive testing program performed on the C-core indicated no need for changes in the mechanical design criteria. A mechanical stress analysis was made on the existing C-core control and transient rod drive system that ascertained their ability to satisfactorily operate with the slightly heavier control rod units of the E-core. No major changes have been made to these drive systems, which are described in detail in the Spert III modified facility report [8]. A brief discussion of its salient design features, from the safety viewpoint, is given below.

The control rod drive typically consists of a cylinder, a piston, a shock absorber, a drive screw, and a 3-speed induction motor. No direct connection exists between the piston, which is directly joined to the control rod poison section, and the drive screw and motor. Air pressure on the bottom side of the piston maintains the piston in contact with the drive screw, thus permitting precise positioning of the control rod with the drive motor and screw. Air pressure is also maintained on the top side of the piston by means of an air reservoir. Scram is accomplished by quickly releasing the air on the bottom of the piston. The top air reservoir is of sufficient capacity to drive the control rod into the core and hold the rod in the down or safe position against any available reactor vessel pressure.

Two transient test sequence timers having ranges of 0 to 30 and 0 to 300 seconds are included in the system for precise programming of experiments. Each timer has seven available circuits. The timer program is started by operation of a spring-return key switch and may be stopped by use of a push-button

switch on the console. After the transient rod poison has been ejected from the core to initiate a power excursion, however, the timers cannot be stopped before completion of the sequence which includes a programmed scram.

Limit switches are provided which stop the drive screw travel at the lower limit and top limit position. A similar switch stops any insertion of the drive below the one and one-half inch withdrawn position unless the drive motor is in slow speed. This is used to protect the drive from overrunning the lower limit position. Due to the increased length of the "E" type fuel assembly over the "C" type, the upper limit switch was removed from the drive unit housing. The top limit switch will now serve as both the top limit and the previous upper limit switch.

A spring-locked latch is provided to hold down each control rod piston once it is seated, thus preventing withdrawal of any control rod from the core without operator action. A key switch must be operated to unlatch each rod and permit its withdrawal. The transient rod drive is similar to the control rod drives except that a latch is provided to secure the rod piston and, hence, the rod itself to the rod drive screw until unlocked by a key switch. No latch is provided to hold the transient rod at its seat position.

Control circuitry provides that loss of contact between any screw and its piston will scram that control rod and initiate a fast rundown of all control rod drives.

An automatic regulating system is provided to maintain sufficient scram air on the rod pistons for any available reactor vessel pressure. Failure of this system causes an automatic fast rundown of the control rod drives. Two dump valves (connected in parallel) are provided on each rod drive for scrambling. The action of either valve is sufficient for a scram, and both valves are tested prior to nuclear operation. Failure of any scram component or fast rundown circuit to operate satisfactorily when tested requires that no further operation be permitted until repairs have been made. Normal fail-safe practices have been followed in all circuits and plant designs.

A standby gasoline-electric power source for the drive system is in operation and is "on the line" at all times during nuclear operation. A bus transfer unit provides switchover to the standby source in the event of commercial power failure. Failure of either power source will shut the reactor down automatically. In the improbable event of simultaneous failure of both power sources, the lock-down latches and the scram air supply are each sufficient to hold down the scrambled control rods. The rod drives would, of course, remain withdrawn until restoration of power. Terminals are provided at the control console to permit a continuity check of all the rod seat switches in the absence of power. Scram air would probably remain adequate to hold down the rods for a period of hours without power to the compressor. In the event that commercial power is not restored in a few minutes, an attempt will be made to repair the standby source. However, restoration of commercial power is necessary to shut down the plant in a normal manner.

Since an accidental drop of the withdrawn transient rod could initiate an unplanned power excursion, a mechanical latch mechanism secures the rod piston to the drive screw. This mechanism is released by a key switch but must be latched in order to perform any operation upon the system other than

scram, control rod insertion, or programmed initiation of a transient. As an added precaution, no top air may be supplied to the transient rod except as a part of the arming sequence. Adjustment of the top air pressure is manual but may only be made with the latch locked. In the event that the top pressure is sufficient to override the hold air, the resultant thrust will mechanically prevent the latch from unlocking.

3. CONTROL ROD SCRAM RELAY SYSTEM MODIFICATIONS

A complete description of the control and transient rod relay system is presented in the modified facility report [8]. Several additions to the circuitry resulting from operating experience that did not exist at the writing of the original SAR [7] are briefly discussed in this section.

Because of the long cable runs (≈ 3000 feet) between the reactor building and the control center and the nature of the Spert III control system, significant capacitive coupling exists between various system elements and ground. Therefore, it was possible that control system relays may have failed to respond to the operator demand if a particular shorted condition should have arisen between one side of a relay and ground. To prevent behavior of this type, a modification was made to the control wiring which consists of a low-impedance grounding mechanism. Voltage and current relays were provided, which monitor the voltage difference between the control system and ground as well as the common ground current, that, in the event of any abnormality in this voltage or current, initiate scram of the reactor and fast rundown of the control rods. The system, used during kinetic testing of the C-core, provided protection against unsafe situations which could occur as a result of a single failure to ground.

Because of the potentially serious consequences of an unplanned control rod withdrawal accident (discussed earlier in this report), a power level scram has been installed in the facility. This system consists of two independent channels of linear power information, each channel having its own electronics and ion chamber. Each channel operates an adjustable trip circuit set to trip at a predetermined reactor power level.

These trip circuits are connected to the reactor scram circuit so that they are in continuous operation except when the transient test timer is running (during a power excursion test) or when bypassed with a special key switch. The proper functioning of each circuit will be determined by the reactor operator prior to each nuclear operation of the facility.

With the Spert III plant operating at high temperature and pressure, it is conceivable that a malfunction of the primary system heat exchanger temperature control circuit could cause the operation of the temperature control valves such that the coolant-moderator temperature would drop, causing a positive reactivity insertion to the reactor. To protect the reactor facility from this type of operational accident or any other cold or water addition accidents, a ΔT scram system has been added since publication of the original Spert III Facility Report. The ΔT scram system consists of a fast response temperature sensing device located in each loop of the primary coolant stream, downstream from the heat exchanger mixing tee. The output signal from the

sensor is fed to an adjustable down-scale setpoint recorder located in the Spert III control room. The setpoint is connected to the reactor control rod scram circuit such that the reactor is scrammed when the setpoint is reached. The recorder's reference temperature indicator is adjustable over the full range of temperature obtainable at Spert III. The setpoint is usually set about 10°F below the reference temperature indicator after the primary coolant-moderator temperature equilibrium has been established and before nuclear operations begin. Hence, if a slug of water which was cooler than the primary coolant temperature by 10°F or more were to enter the primary coolant stream, the reactor would be scrammed.

At any time during which the reactor is operating at an appreciable power level, the potential exists for a loss-of-flow accident through a reduction in pumping power, an accidental primary flow valve closure, or other means of losing primary flow. In order to protect the reactor core from fuel damage resulting from this type of accident, a loss-of-flow scram circuit was installed in the reactor scram system. This system consists of two basic circuits. The primary circuit is a differential pressure (ΔP) scram circuit. A pressure instrument loop measures the ΔP across the reactor vessel. The signal from the loop is recorded on an instrument with a variable setpoint. If the ΔP drops to the setpoint, a reactor scram is initiated. The second circuit monitors the individual primary pump voltage by relays connected across one phase of a three phase power input to each pump. These relays control four series contactors (one for each pump) in the scram circuit. If the voltage is lost to any pump, the relay will de-energize and initiate a scram.

To permit testing of the scram circuits, zero flow operations, or intentional loss-of-flow tests, a keyswitch was installed on the reactor console to bypass both circuits.

Each of the additions to the reactor scram circuit discussed in the preceding paragraphs has a corresponding alarm signal, and the ΔT scram has an alarm signal connected to the annunciator panel.

4. SAFETY FEATURES

Table XIV lists possible drive-system malfunctions and/or faulty operations and the corrective actions designed to counteract these faults.

TABLE XIV

SPERT III CONTROL-ROD-DRIVE SAFETY BACKUPS

Malfunction	Corrective Action
Loss of commercial electric power only	Automatic scram and switching to emergency power. Automatic fast rundown by emergency power.
Loss of emergency electric power only	Automatic fast rundown with commercial power.
Loss of all electric power	Automatic scram. Automatic latch of pistons. Scram air hold down pistons.
Loss of piston contact	Automatic scram of that rod. Automatic fast rundown of all drives.
Loss of air above pistons	Automatic fast rundown of all drives.
Loss of all electric power, coincident with loss of all air	Acme self-locking screw prevents further rod rise and maintains status quota. Motor brake backs up self-locking Acme screw.
Failure of shock absorber to assume extended position	Interlock prevents raising of drives more than 6 inches.
Failure of primary drive-rod seal	Second set of seals becomes operative. Warning light on console signifies that primary seal has failed.
Breakage of mechanical-drive parts above Acme screw (motor couplings, broken gears, etc)	Self-locking Acme screw prevents rise of rods. Manual air scram possible.
Failure of air system to follow reactor pressure	Warning light on console over or under pressure of the scram air. Automatic fast rundown.
Failure of operator to observe top and lower limit of rod travel	Top and lower limit switches cut power from motor.
Failure to obtain contact on all pistons	Permission to withdraw rods is automatically denied. All contact switches must be "on" to have motor power.
Attempt by operator to unlatch piston with a net upward driving force on piston (no piston contact)	Friction on the latch will prevent its withdrawal.
Attempt by operator to drive rods at top speed to "rock bottom"	1-1/2-inch down auto-stop switch denies use of all speeds inward except slow speed. Lower limit switch cuts power from motor.
Various faults in control wiring	Continuity circuits necessary to turn on main power are provided.
Loss of air in air reservoir or low reservoir pressure	Warning light on console indicates low pressure and reactor should be shut down at earliest time which does not abort.
Cold water slug in primary coolant system	ΔT scram circuit scrams reactor if slug is 10°F or cooler than primary coolant temperature.
Loss-of-flow of primary coolant	Reactor scram by loss-of-flow scram circuitry.
Shorting to ground of any point in the control system	Reactor scram and fast rundown of control rods.

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APPENDIX C

SAMPLE MELTDOWN CALCULATIONS

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APPENDIX C

SAMPLE MELTDOWN CALCULATIONS

In the calculations made to determine core melting for a loss-of-coolant accident following high power operation of the E-core, several basic and important approximations and assumptions were used. In general, the method of Lottes [31] was employed.

Radiation was assumed to be the primary mode of heat loss from the reactor core following an instantaneous loss of coolant. For each case, the calculations were performed with the heat source arising from only one half-hour reactor power history. Within the accuracy of the calculations, this is a good approximation since the fission product afterheat decays to approximately one percent of its initial value in the first 12 hours after shutdown. Cooling by conduction and convection was neglected even though this could contribute significantly to the cooling mechanism. Neglecting the conductive and convective heat transfer mechanisms maximizes the heat in the core available for melting and, thus, results in a conservative estimate of fuel melting. To estimate the amount of core cooling during the time after the loss of coolant, the following assumptions were made:

- (1) The heat radiated by each element was assumed to have been completely absorbed in the adjacent outer annulus, proceeding from the core center outward to the outer tank wall.
- (2) The tank wall was treated as a heat sink at the average bulk coolant temperature.
- (3) The calculational model was established as a series of concentric opaque, infinitely long cylinders.
- (4) Fission product afterheat was assumed to be spatially distributed in the same manner as the radial neutron flux during operation.
- (5) Initial fuel rod temperatures in each concentric cylinder were assumed to be the average temperature of that annulus (ie, the volume averaged temperature of all SS and UO₂ in the annulus) that existed during steady-state power operation. Heat removal by the coolant during blowdown was neglected.

Initial calculations employed a Way-Wigner type fission product heat source, but subsequent refinements in the calculations utilized the IBM-7040 Fission Product Inventory Code [32]. This code is based on the constants and methods outlined by W. D. Owings [33] and Perkin and King [34]. The radial heat flux was approximated by the radial core power density determined from the neutron flux distribution obtained during the nuclear core calculations.

Two power and operating time histories (a 15 MW-hr and a 30 MW-hr test) were considered in order to establish approximate limits for operation which can result in melting of the core from the loss-of-coolant accident.

The following sections present an outline of the calculational technique.

1. CORE MODEL

The fuel rods of the core are replaced by four concentric, opaque cylinders.

2. RADIATIVE HEAT LOSS EQUATIONS

$$q_{1 \rightarrow 2} = 0.171 \epsilon_{1 \rightarrow 2} A_1 \frac{T_1^4}{100} - \frac{T_2^4}{100} = C_1 Q \quad (C-1)$$

where

$q_{1 \rightarrow 2}$ = heat radiated from surface 1 to surface 2, Btu/hr

Q = shutdown power, Btu/hr

A_1 = area of radiating surface 1, ft^2

T_1, T_2 = absolute temperature of radiating surface 1 and 2, respectively, $^{\circ}\text{R}$

$\epsilon_{1 \rightarrow 2}$ = emissivity

C_1 = fraction of power generated inside r_1 .

Similarly:

$$q_{2 \rightarrow 3} = 0.171 \epsilon_{2 \rightarrow 3} A_2 \frac{T_2^4}{100} - \frac{T_3^4}{100} = C_2 Q \quad (C-2)$$

$$q_{3 \rightarrow 4} = 0.171 \epsilon_{3 \rightarrow 4} A_3 \frac{T_3^4}{100} - \frac{T_4^4}{100} = C_3 Q \quad (C-3)$$

$$q_{4 \rightarrow 5} = 0.171 \epsilon_{4 \rightarrow 5} A_4 \frac{T_4^4}{100} - \frac{T_5^4}{100} = C_4 Q \quad (C-4)$$

Combining Equations (C-1) through (C-4) gives Q in terms of $\frac{T_1^4}{100}$ and $\frac{T_5^4}{100}$ and, hence, a numerical value for Q , the shutdown power. Physically, Q represents the time at which radiation heat loss from the core equals the rate of heat production in the core. Q is then plotted on a graph of the afterheat as a function of time. Integration of the afterheat curve to the time of Q gives the total heat for heat-up and meltdown of the fuel rods.

In addition to the above analysis, an iterative-type radiative heat loss calculation was performed for the time during which the core heats up. That is, to the time when Q is reached.

This calculation indicates that for the case of 30 MW for 1/2 hour (15 MW-hr, the maximum test planned for the E-core), approximately 193,000 Btu of afterheat is available for total core heat-up and melting. Of this total, about 64,000 Btu of heat is available for the heat-up and melting of the inner 25 percent of the core, based on a flux weighting factor of 0.334 for this part of the core (0.261 for the next 25 percent, 0.199 for the next 25 percent, and 0.206 for the last 25 percent). The heat loss by radiation from the inner annulus during heat-up to time Q is determined from the iterative heat loss calculation to be about 12,000 Btu. The resultant available heat to effect melting in the inner annulus is approximately 52,000 Btu.

The total heat required to cause the inner 25 percent of the core clad to melt is:

Heat up of UO_2 to MP of stainless steel	= 87,000 Btu
Heat up of the stainless steel to its MP	= 54,000 Btu
Heat of fusion for stainless steel	= <u>38,000</u> Btu
Total	179,000 Btu

Nuclear calculations show that the vertical peak-to-average power generation during high-temperature, high-power operation is about 1.35; and that in the hottest (central) annulus, the radial peak-to-average power generation is about 1.1. Therefore, a hot-spot factor of about 1.5 exists in the central core annulus. Applying this factor to the meltdown calculations, ie, 179,000 Btu/1.5 = 119,000 Btu to account for the higher initial temperature in the hot spot and 52,000 Btu x 1.5 = 78,000 Btu to account for the higher heat generation rate

in the hot spot, an estimate can be made of the probability of clad melting in this region. This approximation indicates there is not enough afterheat generation, even in the localized hot spot, to cause clad melting.

These calculations were repeated with the same methods and assumptions, but based on a previous power run at 60 MW for 1/2 hour (30 MW-hr, twice the maximum planned test for the E-core). This power history was selected since it is the maximum capability of the Spert III plant. The results indicate that less than 20 percent of the total core cladding would melt following a loss-of-coolant accident from the 60 MW power condition.

APPENDIX D

COMPARISON OF RADIOLOGICAL HAZARD CALCULATIONS IN THIS ADDENDUM AND IN THE ORIGINAL HAZARDS SUMMARY REPORT

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APPENDIX D

COMPARISON OF RADIOLOGICAL HAZARD CALCULATIONS IN THIS ADDENDUM AND IN THE ORIGINAL HAZARDS SUMMARY REPORT

The calculated radiation doses resulting from the maximum referent accident postulated in this report are, in certain cases, apparently larger than the doses calculated for the maximum accident postulated in the original SAR (IDO-16425) [7]. An examination of the respective postulated maximum accidents for the two cases leads to the question of why some of the doses calculated for the addendum accident are larger than those shown in IDO-16425, since the postulated maximum accident in the original SAR appears to be more severe than the maximum referent accident of the E-core addendum. The reasons for these differences are discussed in this appendix with respect to the assumptions, calculational techniques, and levels of sophistication of the radiological hazards calculations in the two cases.

In order to obtain a comparison of the hazards associated with the accident postulated in the original SAR with the hazards arising from the maximum referent accident postulated in this addendum, the calculational method used in the addendum, which constitutes a more comprehensive analysis and uses more recent data, has also been applied to obtain a reevaluation of the radiation doses resulting from the reactor power history postulated in the original SAR. The radiological calculations in the original SAR used calculational techniques and nomograms current at that time. In this addendum report, the results have been obtained from a detailed computer calculation using Radiological Safety Analysis Code (RSAC). RSAC considers 30 distinct isotopes which contribute inhalation doses to seven internal organs plus total body exposure; whereas in the original SAR, the contribution to two internal organs from 15 isotopes was considered and total body exposure was neglected. Also, some modification of the specific values of Sutton's diffusion parameters is reflected in the newer calculations.

The doses originally calculated and published in IDO-16425 are reproduced in Table XV, and the results of the RSAC recalculation for the same case are presented in Table XVI. In addition, selected calculational results of the radiological hazards from the maximum referent accident of the present core are presented in Table XVII. Examination of Tables XVI and XVII reveals that, using present computational techniques, the radiological hazards for the maximum referent accident of the E-core (based on a realistic estimate of probable reactor power history) are less than those arising from the power history and accident postulated in the original SAR (where the same techniques, constants, and assumptions were used in each case).

Comparison of Tables XV and XVI, however, indicates significant differences in the radiation hazards resulting from the techniques and constants employed in IDO-16425 and in this addendum. These differences and their implications are considered in some detail in the following paragraphs.

TABLE XV

APPROXIMATE DOSES FROM RADIOACTIVE
CLOUD FOR MAXIMUM POSSIBLE ACCIDENT
(Reproduced from Spert III Hazards Summary Report)

Distance from Origin (miles)	Average Conditions (3 m/sec Lapse)	Inversion Conditions (1 m/sec Inversion)
External Gamma Dose (rads)		
0.5	6.0×10^2	5.2×10^3
5	2.2×10^0	7.0×10^1
15	2.1×10^{-1}	1.1×10^1
30	≈ 0	3.0×10^0
Internal Thyroid Inhalation Dose (rads)		
0.5	4.9×10^2	1.3×10^5
5	8.2×10^0	3.0×10^3
15	1.0×10^0	5.0×10^2
30	3.0×10^{-1}	1.4×10^2
Internal Bone Inhalation Dose (rads)		
0.5	7.1×10^3	1.8×10^6
5	1.3×10^2	5.8×10^4
15	1.9×10^1	1.1×10^4
30	5.5×10^0	4.0×10^3

TABLE XVI

APPROXIMATE DOSES FROM RADIOACTIVE
CLOUD FOR MAXIMUM POSSIBLE ACCIDENT
(Using the power history from the Spert III Hazards
Summary Report but computed with the RSAC Program)

Distance from Origin (miles)	Average Conditions (3 m/sec Lapse)	Inversion Conditions[a] (1 m/sec Inversion)
<u>External Gamma Dose (rem)</u>		
0.5	1.2×10^2	4.0×10^5
5	1.5×10^0	8.2×10^3
15	1.8×10^{-1}	7.7×10^2
30	2.8×10^{-2}	1.8×10^2
<u>Internal Thyroid Inhalation Dose (rem)</u>		
0.5	1.6×10^2	5.8×10^5
5	2.4×10^0	1.6×10^4
15	3.1×10^{-1}	2.6×10^3
30	7.9×10^{-2}	6.5×10^2
<u>Internal Bone Inhalation Dose (rem)</u>		
0.5	4.1×10^0	1.7×10^4
5	7.5×10^{-2}	5.4×10^2
15	1.1×10^{-2}	1.0×10^2
30	3.0×10^{-3}	3.6×10^1

[a] Inversion doses are calculated for comparative purposes only. Test operations leading to these dose potentials are not permitted.

TABLE XVII

APPROXIMATE DOSES FROM RADIOACTIVE
CLOUD FOR MAXIMUM REFERENT ACCIDENT

Distance from Origin (miles)	Average Conditions (3 m/sec Lapse)	Inversion Conditions ^[a] (1 m/sec Inversion)
	External Gamma Dose (rem)	
0.5	8.4×10^1	2.8×10^5
5	1.1×10^0	6.1×10^3
15	1.2×10^{-1}	7.3×10^2
30	2.2×10^{-2}	1.2×10^2
	Internal Thyroid Inhalation Dose (rem)	
0.5	1.4×10^2	4.9×10^5
5	2.1×10^0	1.3×10^4
15	2.5×10^{-1}	2.1×10^3
30	5.1×10^{-2}	5.2×10^2
	Internal Bone Inhalation Dose (rem)	
0.5	8.1×10^{-1}	3.0×10^3
5	1.3×10^{-2}	1.0×10^2
15	1.8×10^{-3}	1.9×10^1
30	5.5×10^{-4}	6.6×10^0

[a] Inversion doses are calculated for comparative purposes only. Test operations leading to these dose potentials are not permitted.

1. FISSION PRODUCT INVENTORY RELEASE

The total fission product sources used to compute the doses of Tables XV and XVI are given in Figures 20 and 21. Since the sources for the original SAR were calculated from the Way-Wigner formula (AECU-3066)[26] without fractionation, while the RSAC calculations were based on fractionation of the total fission product inventory (1 percent solids, 50 percent halogens, and 100 percent of noble gases), the differences in these sources are reasonable. Note, however, that the more detailed RSAC calculated doses do not decay as a simple exponential in time. To obtain a measure of the fractionation correction, RSAC-calculated fission product sources were also obtained with no fractionation. These sources appear as the broken lines in Figure 20 and 21. For the time interval 10^3 sec to about 10^5 sec, the Way-Wigner sources are in good agreement with the corresponding RSAC sources, but are larger at times shortly after and at times long after release.

This substantial agreement between the fission product sources obtained from RSAC and Way-Wigner calculations indicates that the large differences

Fig. 20 Long-lived fission product sources for the Spert III accident, computed by two methods.

Fig. 21 Short-lived fission products for the Spert III accident, computed by two methods.

in the doses of Tables XV and XVI cannot arise from the respective fission product sources alone.

The relative effects of the various approximations and assumptions used in the two (original SAR and RSAC) calculations are summarized in Table XVIII. In this table, the first item gives the multipliers required to correct the unfractionated fission product sources for the fractionation assumed in this addendum. These correction factors (multipliers) are intended to permit the modification of each value of Table XV (with the exception of external gamma doses) for comparison with Table XVI.

2. EXTERNAL GAMMA DOSES

Table XIX presents the external gamma doses from Tables XV and XVI. For the RSAC-computed case, these have been broken into direct cloud gamma and deposition gamma doses which are computed separately. The deposition gamma doses, which comprise the predominant term in every case, were computed on the conservative basis that prior depletion of the cloud by deposition was neglected and that the subject remained in the specific contaminated area for an infinite time after passage of the cloud. In the original SAR, external gamma doses were based on Holland's nomograms^[26] which are for cloud gamma dose only. Therefore, only the cloud gamma doses are compared here for the two cases.

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TABLE XIX

EXTERNAL GAMMA (CLOUD AND DEPOSITION) DOSES (REM)
COMPUTED FROM HOLLAND'S NOMOGRAMS AND FROM RSAC

Distance from Origin (miles)	External (RSAC)		Deposition (RSAC)		Cloud (RSAC)		Cloud (Original SAR)	
	Lapse (3 m/sec)	Inversion (1 m/sec)	Lapse (3 m/sec)	Inversion (1 m/sec)	Lapse (3 m/sec)	Inversion (1 m/sec)	Lapse (3 m/sec)	Inversion (1 m/sec)
0.5	1.2×10^2	4.0×10^5	1.04×10^2	3.7×10^5	1.6×10^1	3.3×10^4	6.0×10^2	5.2×10^3
5	1.5×10^0	8.2×10^3	1.5×10^0	7.9×10^3	6.9×10^{-2}	2.5×10^2	2.2×10^0	7.0×10^1
15	1.8×10^{-1}	7.6×10^2	1.7×10^{-1}	7.4×10^2	4.0×10^{-3}	2.0×10^1	2.1×10^{-1}	1.1×10^1
30	2.8×10^{-2}	1.8×10^2	2.8×10^{-2}	1.8×10^2	≈ 0	3.6×10^0	≈ 0	3.0×10^0

The major difference in the cloud gamma doses can be attributed to the position-dependent attenuation factor, $G(r)$, that is utilized in the nomogram calculations [26]. This attenuation factor appears in equation form as

$$D_G = A \int_0^\infty \int_0^\infty \int_{-\infty}^\infty \int_{-\infty}^\infty X(x, y, z, t) G(r) dx dy dz dt$$

where $X(x, y, z, t)$ is a Sutton-type "concentration factor", and $G(r)$ is a position-dependent "attenuation factor" for the cloud. In this formulation,

$$G(r) = \frac{B_r e^{-\mu r}}{r^2}$$

where r is the radius vector from the observer to the cloud and B_r is a multiple-scattering buildup factor which is also a function of r . Doses calculated by current methods (RSAC) use the total integrated dose equation [26]

$$TID_{cloud} = \int_t X(x, y, z, t) dt$$

which gives the maximum dose an observer can receive as a result of the cloud passage. For the case where y and z are zero, this reduces to

$$TID_{cloud} = \frac{2}{\pi C u (\frac{r}{ut})^{2-n}}$$

Another difference in the cloud gamma doses computed for the original SAR and for this addendum can be resolved by examination of the diffusion constants used in the two cases. Table XX compares these external cloud gamma doses, all recomputed with Holland's nomograms but using the diffusion constants of the present addendum [columns labeled (1)] and the diffusion constants of the original SAR [columns labeled (2)]. Although this comparison does not indicate greatly changed doses, it does show that presently accepted diffusion constants lead to more conservative inversion doses and less conservative lapse doses for the selected wind speeds.

In Table XXI, the recomputed doses from the original SAR columns labeled (1) in Table XX have been further modified to account for the difference in the Way-Wigner and RSAC sources (Figures 20 and 21), including the fractionation assumed by RSAC. With the exception of the position-dependent attenuation factor, $G(r)$, these doses (labeled "nomogram results") permit a direct comparison of radiological hazards obtained from Holland's nomograms and from RSAC calculations. To facilitate this comparison, the RSAC-computed doses from columns 5 and 6 of Table XIX are also presented in Table XXI. The remaining differences in the respective doses are seen to be quite large for the inversion case at small distances, but diminish rapidly with distance so that they compare reasonably well at 10 to 15 miles. The respective lapse doses are in substantial agreement at all distances. These remaining time-dependent differences in the doses arise from the position-dependent factor $G(r)$ that is implicit in the nomogram calculations.

TABLE XX

EXTERNAL CLOUD GAMMA DOSE COMPARISON

Distance from Origin (miles)	Lapse Dose -- rem (3 m/sec)		Inversion Dose -- rem (1 m/sec)	
	Diffusion Constants		Diffusion Constants	
	Present Addendum[1]	Original SAR[2]	Present Addendum[1]	Original SAR[2]
0.5	2.5×10^2	6.0×10^2	6.8×10^3 ^[a]	5.2×10^3
5	5.2×10^{-1}	2.2×10^0	1.2×10^2	7.0×10^1
15	4.0×10^{-2}	2.1×10^{-1}	2.0×10^1	1.1×10^1
30	≈ 0	≈ 0	7.0×10^0	3.0×10^0

[a] Inversion doses are calculated for comparative purposes only. Test operations leading to these dose potentials are not permitted.

TABLE XXI

COMPARISON OF ADJUSTED ORIGINAL EAR CLOUD GAMMA DOSES

Distance from Origin (miles)	Cloud Gamma Doses (rem)			
	Lapse (3 m/sec)		Inversion (1 m/sec)	
	Nomogram Results	RSAC Results	Nomogram Results	RSAC Results
0.5	7.4×10^0	1.6×10^1	4.5×10^2	3.3×10^4
5	2.4×10^{-1}	6.9×10^{-2}	2.2×10^1	2.5×10^2
15	$\approx 5.0 \times 10^{-2}$	$\approx 4.0 \times 10^{-3}$	3.8×10^0	2.0×10^1
30	≈ 0	≈ 0	1.2×10^0	3.6×10^0

3. INTERNAL THYROID AND BONE INHALATION DOSES

Other than the fission product sources and fractionation of the sources, which have been discussed, the major difference in the thyroid and bone inhalation doses computed for the original SAR and for this addendum arise from the different diffusion and other constants used in the two cases. The manner in which the doses of the original SAR are modified by use of the constants of this addendum is given in Table XVIII. These currently accepted specific values of Sutton's diffusion parameters and the radiological and biological constants generally reflect the use of more recent data which are believed to represent improved values.

For purposes of comparison, the 0.5 mile doses of Table XV have been multiplied by the appropriate overall correction from Table XVIII and compared with the corresponding values from Table XVI. This comparison (Table XXII) shows that, on a common bases, the methods of the original SAR lead to inhalation doses which are in good agreement with the RSAC-computed doses used in this addendum.

TABLE XXII

INHALATION DOSE COMPARISON AT ONE-HALF MILE

Lapse Condition (3 m/sec -- rem)				Inversion Condition ^[c] (1 m/sec -- rem)			
Thyroid		Bone		Thyroid		Bone	
A[a]	B[b]	A[a]	B[b]	A[a]	B[b]	A[a]	B[b]
2.26×10^2	1.6×10^2	2.02×10^1	4.1×10^0	8.98×10^5	5.8×10^5	3.26×10^4	1.7×10^4

[a] Original SAR (with diffusion, radiological, and biological constants of RSAC) calculation.

[b] RSAC calculation.

[c] Inversion doses are calculated for comparative purposes only. Test operations leading to these dose potentials will not be permitted.

4. CONCLUSIONS

The above comparisons show that, with the exception of cloud gamma doses, the use of Holland's nomograms (as in the original SAR) gives results substantially in agreement with results obtained by the methods of this addendum (RSAC), provided that the two are applied on the same basis. Because of the position-dependent factor $[G(r)]$ implicit in the nomogram calculations but not used in the RSAC results, the RSAC-computed inversion cloud gamma doses are significantly larger than the respective nomogram results at short distances. However, the differences in these doses also diminish rapidly with distance. Deposition gamma radiation included in the RSAC but neglected in the nomogram calculations, contributes significantly to the RSAC-computed cloud gamma doses in every case.

The preceding section has compared the calculational techniques used to obtain the radiation doses shown in the original SAR and in this addendum report. The reasons for the large discrepancies between the radiological hazards (doses) presented for the maximum referent accident of this addendum and those calculated for the original SAR were shown to be directly related to the respective calculational methods used to obtain the doses. When the

radiation doses arising from the respective maximum referent accidents were computed by the same calculational technique, it was shown that the accident postulated in the original SAR would result in the more serious radiological hazards.

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