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STATUS REPORT OF CONFINEMENT STUDIES
DOUGLAS UNITED NUCLEAR OPERATED REACTORS

53250

Compiled by
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November 30, 1965

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TABLE OF CONTENTS

	<u>Page</u>
INTRODUCTION	1
SUMMARY AND CONCLUSIONS	3
Status of Analytical Studies	3
Maximum Credible Accident	3
Consideration of More Serious Accidents	5
Evaluation of Present Confinement System	5
Modification and Additions to the Confinement System	6
Conclusions	7
PRESENT FACILITIES DESCRIPTIONS	9
Present Confinement Facilities	9
K Reactor Block Description	9
B, C, and D Reactors Block Description	11
General	12
ANALYTICAL PROGRAM	13
Scope of Study	13
Single Supply Header Rupture	13
Loss of Coolant to Half of the Headers	15
Total Loss of Coolant	17
Details of Analysis	20
Fission Product Inventory Computation	22
Reactor Kinetics Computation	22
Fuel Temperature Rise and Fission Product Release Computation	23
Dose Computation	24
Results of the Calculations to Date	25
Fuel Release Assumptions	25
Release Quantity	26
Release Paths	27
Dose at Boundaries from an Atmospheric Release for Existing Confinement	28
Release to the Columbia River	31
Effect of Future Operating Conditions	32
EXPERIMENTAL PROGRAMS	34
Five-Year Research and Development Program	34
Summary of F Reactor Confinement Tests	35

DECLASSIFIED

FACILITY IMPROVEMENT DESCRIPTION	38
Relationship of Calculations to Facility Improvements	38
Accident Preventive Measures - Earthquake Resistance	38
Pumps and Piping	40
Buildings	42
Improvement in Ability to Cope with Accidents	43
Confinement System Improvements	43
Extension of the Fog Spray System	43
Exhaust Ventilation Fan Diesel Drives	45
Confinement Filter System	45
Miscellaneous Improvements	46
Confinement Aids	46
Emergency Liquid Waste Disposal	47
Temporary Containment in Reactor Shell	48
Effluent Line Trapping	50
Major Fission Product Release to the Building	50
Downcomer Strengthening	51
Reactor Flooding	51
Low Pressure Gas Storage	53
High Pressure Gas Storage	54
Foam Generation	55
Other Methods of Noble Gas Retention	56
Recirculation Facilities	56
Other Types of Confinement or Containment	57
Building Sealing	57
High Stack	59
Containment Dome	61
Composite Solutions for Specific Incidents	62
General Improvements	62
Loss of Coolant to Single Crossheader	63
Loss of Coolant to Alternate Crossheaders at K Reactors	64
Total Loss of Process Coolant	65
Compatible Solution for any Postulated Incident	65
APPENDIX A	A-1
Seismological Characteristics of the Hanford Region	
APPENDIX B	B-1
Details of Computations Used in the Confinement Study	
REFERENCES	R-1

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STATUS REPORT OF CONFINEMENT STUDIES
DOUGLAS UNITED NUCLEAR OPERATED REACTORS

INTRODUCTION

Confinement studies on the Hanford Production reactors were started in March at the request of the Richland Operations Office.⁽¹⁾ The Advisory Committee on Reactor Safeguards indicated to the Division of Production staff that the production reactors should be operated under the same basic safety criteria that are presently applied to licensed reactors. It also recommended that studies be undertaken at an accelerated pace to determine means by which this objective could be accomplished. A planning meeting was held with Division of Production staff on April 14 to 15.

The confinement studies were then undertaken with the following objectives established:

- Review and reestablish if necessary, the maximum credible accident for the production reactors operated by Douglas United Nuclear (DUN).
- Evaluate the present confinement systems under the Code of Federal Regulation Title 10, Part 100, Reactor Site Criteria,⁽²⁾ for the maximum credible accident and an accident corresponding to the failure of reactor fuel containing 50% of the reactor's fission product inventory.
- Establish improvements or additions in the confinement system needed to more nearly comply with 10 CFR Part 100 radiation dose criteria for existing boundaries of the exclusion area and the low population zone for the maximum credible accident.
- List facility additions and changes required to comply with 10 CFR Part 100 criteria for more serious accidents than the maximum credible accident. Include cost information for all capital items.

For this report, the maximum credible accident is defined as that credible accident that would cause the maximum quantity of fission products to be released from the reactor.

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The philosophy upon which the Reactor Site Criteria is based was indicated in the Statement of Considerations used by the AEC in proposing rule making for 10 CFR Part 100. ⁽³⁾

"The basic objectives which it is believed can be achieved under the criteria set forth in the proposed guides are:

- a. Serious injury to individuals off-site should be avoided if an unlikely but still credible accident should occur.
- b. Even if a more serious accident (not normally considered credible) should occur, the number of people killed should not be catastrophic.
- c. The exposure of large numbers of people in terms of total population dose should be low."

This philosophy suggests two accident cases should be reviewed: the credible accident causing the greatest fission product release and the next more serious accident. Although not defined, the more serious accident might be that resulting from one additional component failure or procedural error over the combination considered credible. The more serious accident would be evaluated against less restrictive dose considerations than would the maximum credible accident. The objectives, listed above for the confinement studies, match the apparent philosophy used as a basis for 10 CFR Part 100 except for the last item which implies the confinement system should be adequate to maintain doses within the criteria for more serious accidents.

This report presents the interim status of the confinement studies in meeting the objectives aforementioned and lists the improvements under evaluation which might be used to bring the reactors more nearly in compliance with 10 CFR Part 100 criteria for the maximum credible accident. Some of the potential improvements, which will be considered in completing the study for accidents more serious than the maximum credible accident, are indicated.

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SUMMARY AND CONCLUSIONS

Status of Analytical Studies

The analytical studies have progressed considerably toward defining the detailed sequence of events in various loss-of-coolant accidents for the Hanford once-through cooled reactors. One finding is that the fuel would not reach the melting point of uranium in the single crossheader accident because of heat loss to adjacent cooled channels. The significance of this finding cannot be determined presently because key experimental data are not available to define the amounts of fission products released at uranium temperatures below the melting point. For this reason, in the analysis of the single crossheader failure the conservative assumption still used is that the fission products released are the same as if the fuel melted. Pertinent information regarding fission product release following cladding and uranium melting will be provided by the nuclear safety studies being initiated under the Five-Year Hanford Research and Development Program. ⁽⁴⁾

Tests recently made in a full-size reactor confinement system showed that in-service testing of the confinement system filters is feasible. Charcoal iodine collectors were shown to have the designed removal efficiency for elemental iodine after 5 yr in service. In addition, fog spray and fire fighting foam were tested as confinement aids.

Maximum Credible Accident

To judge compliance with 10 CFR Part 100, the maximum credible accident must be defined. Historically, the maximum credible accident for the DUN operated production reactors has been designated as the loss of coolant to a single crossheader; the cause of such an accident can be identified without recourse to such severe initiating events as earthquake, bomb blast, etc. In addition, the Hanford philosophy of reactor safety has been the provision of reliable methods of cooling fuel in all accident situations short of severance of the primary coolant piping or destruction of the reactor building.

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As part of the studies reported here, the maximum credible accident has again been reviewed, and also accident events of more serious consequences.

The only natural event believed to lead to a more serious accident is an earthquake. The earthquake history and geology of the Hanford area have been the subjects of past and present studies. As a result of one of these studies, F. Neuman gave an opinion that an earthquake of MM-6 intensity on the Modified Mercalli scale represented a sound engineering design value for the Hanford area.⁽⁵⁾ Because of the severe consequences of a major fission product release, protection of the coolant system for an MM-7 intensity earthquake is considered to be warranted. Additional earthquake protection was considered in a previous study.⁽⁶⁾

An earthquake of MM-6 intensity could potentially cause loss of electrical power to the primary coolant pumps, loss of steam to the secondary pumps, and failures in the last-ditch coolant system at one reactor (D) which could lead to total loss of coolant to that reactor. In addition, an earthquake of this intensity might also cause the high tanks adjacent to both B and D Reactors to fall, possibly resulting in damage to the reactor buildings and the coolant systems. Therefore, if an earthquake of this intensity is to be considered credible for the Hanford area, then a total loss of coolant to D Reactor and a loss-of-coolant accident more severe than the single crossheader rupture at both B and D Reactors should be considered credible. At the C and two K Reactors, the last-ditch coolant systems are more resistant to earthquake damage, and the single crossheader rupture can still be considered the maximum credible accident.

If any improvements at all are to be made to the reactors or confinement systems, the parts of the reactor buildings and coolant systems vulnerable to earthquakes should be strengthened. The cost for strengthening the last-ditch piping at D Reactor, the penthouses surrounding the safety rods and certain critical items at the other reactors to withstand an MM-7 intensity earthquake is estimated at about \$3,000,000. This estimate also

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includes the diesel-driven last ditch pumps at B and D Areas and diesel drives for the confinement exhaust fans at B, C, and D Reactors. These improvements would then add credence to reaffirming the single cross-header rupture as the maximum credible accident for all five reactors.

Consideration of More Serious Accidents

Consideration of those accidents that might lead to melting of fuel containing 50% of the reactor fission product inventory also led to consideration of the 100% meltdown accident. For example, the postulated set of events required to cause a 50% melt in a K Reactor would cause loss of coolant to all of the fuel in the smaller reactors. For this reason, total loss of coolant accidents have been included in this study.

It must be remembered that in the past, the circumstances leading to these more severe accidents were not considered credible; therefore, these accidents have never been reviewed in any detail. The 50 and 100% meltdown accidents have now been studied and their consequences evaluated. An appreciable part of this study effort has been directed toward developing adequate analytical tools to provide meaningful solutions to the complex problems involved.

Evaluation of Present Confinement System

Reanalysis of the single crossheader accident has shown that a larger portion of the fission products than previously considered may pass through the effluent pipes to the open retention basins where the noble gases and part of the other fission products would be released to the atmosphere.^(7, 8) The solid fission products remaining in the water would partially settle out in the retention basin. The halogens would be expected to dissolve in the water and then be discharged directly to the Columbia River since the existing effluent systems are without means for cribbing. If one assumes that half of the fission products that are released from the fuel do pass through the effluent system, the doses at the present exclusion and low population zone boundaries would be well within the 10 CFR Part 100 criteria. However, the release to the river could result in potential thyroid

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dose to persons drinking the contaminated river water in the three cities nearest Hanford. The dose would exceed the 300 rem used as a basis for 10 CFR Part 100. Unlike an atmospheric release, the path and speed of the Columbia River is known, and time would be available to take emergency action to restrict the use of the contaminated water in the cities downstream of Hanford.⁽⁹⁾

Analysis of accidents involving release of 50% or more of a reactor fission product inventory indicates that the release through the effluent piping might be substantial. This path would release a greater quantity of fission products to the atmosphere than the path through the confinement filter system. Since this path cannot be ruled out in the existing reactor and confinement systems, it is used as the basis of dose calculation in this report. The calculations show that the release would cause doses exceeding the 10 CFR Part 100 criteria at both the exclusion and low-population zone boundaries. Even if the confinement filter system could withstand the heat load, a release through it would result in doses still exceeding the criteria because the majority of the dose from an atmospheric release is due to the noble gases.

Modifications and Additions to the Confinement System:

Modification of the existing effluent systems in each of the reactor areas would give capability to divert and store a majority of the effluent coolant containing fission products from a single-crossheader accident. A rough cost estimate for these modifications for all five reactors would be \$2,000,000. These modifications would be expected to reduce the release to the river to a level below that which would exceed a 300 rem dose to the thyroid from drinking the river water at the point of nearest public access. Further study is needed to determine just what reduction factor might be attained.

The present confinement system, put in place by Project CGI-791 in 1959, is based on the concept of a filtered release from a single crossheader-type accident. Thus far, efforts to identify modifications or

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additions of the present system that would accommodate a major fuel meltdown accident have met with limited success. This document describes a number of techniques for enhancing confinement ranging from minor extensions of the present fog spray system to near-complete containment. These investigations have pointed out a number of areas where insufficient study or information exists for making sound judgments at this time regarding some of these techniques.

Although there is serious question whether complete containment could be accomplished, significant improvements over the existing confinement could be obtained for major release accidents. These modifications would be expensive, however, and the benefit realized is not believed to warrant the expenditure. Minor improvements to the present confinement system, such as providing storage capability, improving the backup on the exhaust fan drives, and extending the fog spray would be beneficial and provided at reasonable costs.

A summary of the tentative improvements to the confinement systems is given in Table I. These are grouped in packages according to the accident involved. The cost estimates shown should be considered as approximate, since they were made without definition of specific details.

Conclusions

In summary, the present conclusions drawn in the study are as follows:

- Methods of obtaining compliance with 10 CFR 100 for a major meltdown type accident are apparent at this time.
- The reactor effluent is an alternate release path, but more study of methods of accommodating reactor effluent containing fission products is needed.
- Minor modifications to the existing confinements should be made to enhance the capability in handling a single crossheader incident.
- Upgrading the earthquake resistance of the smaller reactors is warranted.

Until these studies have been completed, it seems inappropriate to make specific recommendations regarding Federal Regulation Compliance for accidents more serious than the single-crossheader failure.

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TABLE I

SUMMARY OF TENTATIVE CONFINEMENT SYSTEM IMPROVEMENTS

Protection Improvement Packages ^(a)	Package Cost ^(b) (In thousands)	Dose at Low Population Zone Boundary from Atmospheric Release ^(c)	
		Existing System	Improved System
1. <u>General</u> (Earthquake & Misc.)	\$ 3,000		<1 rem
1. & 2. <u>Loss of Coolant</u> <u>to One Crossheader</u> (Reduce river release)	\$ 5,000	<1 rem	<25 rem
1. & 3. <u>Loss of Coolant to</u> <u>Half of K</u> (Total loss at B, C, & D)	\$10,600	75 rem	<25 rem
1. & 4. <u>Total Loss of Coolant</u> <u>All Five Reactors</u>	\$10,900	230 (for one K) 75 (for small)	<25 rem <25 rem
1. & 5. <u>For All Above Incidents</u>	\$12,000	230 (for one K) 75 (for small)	<25 rem <25 rem

(a) See page 62 for package descriptions. Each package includes protection for incident at all reactors.

(b) Costs for packages include cost of general package.

(c) Total body dose.

PRESENT FACILITIES DESCRIPTIONS

Present Confinement Facilities

The present confinement system at each production reactor consists of the reactor building divided into ventilation zones with progressively lower pressures in the increasingly contaminated areas, a fog spray on the reactor discharge face, exhaust fans, particulate filters, charcoal halogen collectors, and an exhaust chimney. The system is instrumented to perform either automatically or manually (Figure 1).

The system is designed to remove approximately 99.95% of the particulate fission products, 95% of the halogens, and 0% of the noble gases for fission products released within the building for a credible accident. No provision is made for hold-up or collection of noble gas fission products.

K Reactor Block Description

Since the reactors themselves may be effective confiners, the following is a description of the reactor block. The K Reactors have a graphite moderator stack of loosely keyed blocks nominally 3-3/4 in. square by 48 in. long. The stack is 41 ft high, 41 ft wide, and 33-1/2 ft long. The bottom graphite layer rests on a surface of cast iron blocks (thermal shield) which in turn rests on a massive concrete foundation.

The thermal shield is a layer of cast iron, 10 in. thick, surrounding the graphite stack. The shield is built of blocks overlapping each other at the edges allowing no crack to pass straight through. During normal operation, cooling is provided for the top, bottom, and side thermal shields by the flow of water through 3/4 in. stainless steel cooling tubes imbedded in the blocks. The front and rear shields are cooled by the water flowing through the process tubes piercing the shields.

The biological shield consists of a high density concrete structure enclosed in structural steel framework and 1 in. thick steel skin. The shield surrounds the thermal shield except on the bottom. The shield consists of various aggregates and, consequently, varying density concrete

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FIGURE 1

Reactor Confinement Facility Layout

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packed into individual steel crates. The thickness is 63 in. at the front and sides, 45 in. at the rear, and 83 in. at the top. The biological shield is cooled at the inner surfaces by cooling water tubes of 3/4 in. stainless steel.

The steel framework of the top shield is imbedded in the side shields, giving a monolithic structure of the top and side shields. The front and rear shields can move relative to the side shields although keyed to side shields. Joints allowing thermal expansion of the shielding crates are covered with 1/16 in. thick stainless steel omega seals. The vertical and horizontal joints between the inlet and outlet shields and the top and side shield assemblies are covered with 1/8 in. thick neoprene seals. Each process tube has a gas seal at the front and rear penetrations.

B, C, and D Reactors Block Description

The B, C, and D Reactors are generally similar to the K's but vary in details. The graphite stack is 36 ft wide, 36 ft high, and 28 ft long.

The thermal shield consists of cast iron 8 1/8 in. thick on the top, 8 in. on the sides, 10 in. front and rear, and 10 1/4 in. on the bottom. Cooling tubes are provided in the top, bottom, and side shields.

The biological shield consists of alternate layers of steel and masonite 52 in. thick on all sides and the top; the bottom rests on concrete. C Reactor is an exception in that its top shield is 84 in. thick, heavy, aggregate concrete. The shields at B and D Reactors are not provided with cooling tubes. C Reactor has cooling tubes in the top and side shields' inner surfaces.

The entire block is covered by a welded steel plate box. Neoprene seals and metal omega seals are utilized for component movement. Each process tube has a gas seal at the front and rear penetrations.

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General

Each reactor type has penetrations for horizontal control rods, vertical safety rods, process tubes, experimental test holes on the sides, and monitoring penetrations. Gas seals are provided in each case.

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ANALYTICAL PROGRAM

Scope of Study

It has always been accepted at Hanford that the most serious incident for DUN production reactors would be the rapid and total loss of coolant during equilibrium operation. This would be an autocatalytic accident in that voiding introduces the largest nuclear transient in the reactors as presently designed. Furthermore, the loss of coolant incident, despite proper functioning of the vertical safety rods, would lead to fuel cladding melting and, in the worst cases, melting of uranium. Initially, only the study of 50% melting was considered, but actually three loss-of-coolant accidents have been studied. They are:

- A single supply header rupture
- Loss of coolant to half of the supply headers
- Complete loss of coolant to a reactor,

thereby providing a range of potential fission product releases from half of one header to the full reactor inventory.

Single Supply Header Rupture

The single supply header rupture accident has been discussed in the Hazards Summary Reports for the production reactors^(4, 5) as being the only accident where a single pipe break could cause loss of coolant to more than one process channel. The header break would scram the reactor by a trip from one of the individual tube pressure monitors on the affected process tubes. This case has been reviewed and is still considered to be the maximum credible accident if an earthquake of MM-6 intensity is not considered credible. A restatement of this accident is given in the next paragraph.

The break is assumed to occur at one end of the supply header on the process tube side of the check valve. See Figure 2 for a schematic of the coolant piping and process tubes. The resulting pressure gradient from the supply riser side of the header to the break would cause front-to-rear

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FIGURE 2
Single Header Rupture Schematic

coolant flow in the process tubes nearest the riser, no coolant flow where the front header pressure equaled the rear header pressure (normally in the range of 30 psig), and reverse coolant flow from the rear header for the tubes nearest the break. The tubes near both the riser and open ends of the header would have less than normal operating flow, but the flow would be adequate to prevent melting of the aluminum cladding in some of these tubes. There would be tubes on both sides of the tube at zero pressure differential in which the pressure differential would not be adequate to provide enough coolant flow and prevent melting of the fuel cladding.

Loss of Coolant to Half of the Headers

Loss of coolant to half of the crossheaders is the accident that most nearly approaches the study objective where fuel containing 50% of the reactor fission product inventory fails.

Loss of coolant to half of the headers is possible only at the K Reactors (though it is not judged credible) where four supply risers are connected in pairs to alternate headers. Figure 3 shows the K-Reactor coolant supply system. The accident to cause this condition in the K Reactors would require all of the following:

- A pipe break in the critical location (Point A on schematic) where two of the risers are tied together by the connection to the diesel, last-ditch system.
- Procedural failure from leaving open a flow-balance valve (marked B on the schematic) connecting together the two halves of the primary coolant system (this valve is ~~shut~~ shut for all reactor operating conditions).
- Either a failure of one of two check valves (C or D on schematic) to prevent backflow from one of the high pumps to the break or one of the two high head pumps (E or F on schematic) is out of service.

This accident was not discussed in the Hazards Summary Reports because it is not judged to be a credible combination of circumstances. There would not be adequate coolant flow to the process tubes supplied by half of the

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K Reactor Coolant Supply Schematic

headers because all but one of the six high lift pumps' flow would go to the pipe break. The remaining tubes would have adequate cooling during and after the reactor shutdown to prevent cladding melting.

Total Loss of Coolant

Total loss of coolant at the K Reactors would involve either failure by rupture of three large pipes (A, G, and H on schematic) or complete failure of the primary, secondary, and last-ditch diesel coolant supply systems. An earthquake severe enough to cause rupture of three separate pipes would most likely also rupture the building shell and the reactor. The piping at the K Reactors is designed for Zone II earthquake resistance. Since an earthquake exceeding this intensity is not considered credible, the combinations of events to cause complete loss of coolant are not considered credible (see Appendix A).

A total loss-of-coolant accident at the B or D Reactors would require either a procedural failure to keep the flow balance valve (A on schematic in Figure 4) between the two ends of the common header closed during operation, combined with a single pipe rupture in the common header or a riser (B on schematic); or pipe ruptures in both ends of the common header or both risers. Neither of these combinations of events is judged credible. However, if an earthquake of MM-6 intensity should occur, total loss of coolant might result at D Reactor due to electrical power failure to the primary coolant system pumps, failure of the steam supply to the secondary coolant pumps, and failures in the pipes in the last-ditch coolant system. In addition, earthquake-damaged high tanks at the B and D Reactors might cause loss of coolant to more tubes than are supplied by one crossheader. If an earthquake of MM-6 intensity is considered credible, then total loss of coolant at B and D Reactors should be considered credible. Earthquake protection measures are discussed later.

At C Reactor, a total loss-of-coolant accident would require a major pipe rupture (Point A on schematic in Figure 5) on the reactor side of the main check valves, combined with procedural failure to keep the

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FIGURE 4

B and D Reactor Coolant Supply Schematic

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Primary and Secondary Pumps

FIGURE 5

C Reactor Coolant Supply Schematic

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valve (B on schematic) between the two halves of the coolant system closed during operation. The combination of events to cause this accident is also not considered to be credible. The coolant systems would not all be expected to fail because of an earthquake.

Details of Analysis

In evaluating the real consequences of a reactor accident, a more accurate description can be made by using analysis techniques and test data not available for previous analyses. Many of the previous accident analysis calculations were made with simplified mathematical relationships which were needed to make hand calculations feasible. These simplifying assumptions were made conservative to compensate for the inexactness of the calculations. As computers have become more generally available, more sophisticated calculations can be made. This study is making use of several computer programs developed since the writing of Hazards Summary Reports, ^(4, 5) and a heat transfer computer program specifically developed for this study.

All calculations in a reactor accident study must be based on either experimentally proven relationships or assumed relationships. The computer programs used in this study closely describe the known relationships, but assumed relationships must be made in the programs because certain experimental data are not available. Most of the assumptions used in making the calculations are described in the detailed discussion of the individual programs in Appendix B. The assumptions used in coupling the programs to obtain evaluations of the accident cases are discussed with the results on page 25.

The elements of the calculation model for the confinement study and their interrelations are shown in Figure 6. The diagram shows the initiating event feeding input data into both the reactor kinetics and the heat transfer programs. The kinetics program gives the basis for any change in fission product inventory and feeds the nuclear transient heat generation into the heat transfer program. Input on heat generated by fission products

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FIGURE 6

Elements of Calculation Model

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is also fed as input to the heat transfer calculation. The output of the heat transfer calculation is the time sequence of uranium melting that is fed to the fission product inventory program. This program then calculates the fission products released from the reactor and heat in fission products. The confinement system must handle the reactor release and modify the building release to meet dose criteria such as is in 10 CFR, Part 100. Dose calculation is based on building release and meteorological data.

A brief discussion of the computations used in making this confinement study is given in the following pages. Appendix B includes a more detailed discussion of these computations.

Fission Product Inventory Computation

The fission product inventory is calculated for this study by using the Radio-Isotope Buildup and Decay Computer Program called RIBD. The RIBD program is used to compute the inventory in a reactor for any given operating history and to compute the decay of these fission products following reactor shutdown. The Program is capable of handling input data for 450 fission product nuclides. It can be used to compute activity level in curies, concentrations in gram-atoms, heat generation rates from both gamma and beta decay and integrated energy released following shutdown of a reactor. This information can be presented for each element, selected groups of elements, or all the elements present.

Reactor Kinetics Computation

The older Hanford production reactors are overmoderated such that loss of the coolant water, which is a moderator, results in increasing the reactivity. In cases where the coolant is lost suddenly, a nuclear excursion will occur. A reactor kinetics computer program called CLUMSY I is being used in this study to compute the nuclear excursion resulting from loss-of-coolant accidents. The CLUMSY I program simulates loss of coolant to any part or all of a reactor.⁽¹⁰⁾ It also simulates the reactivity effects of the safety rods descending into the reactor after the reactor is scrammed.

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The input to the program used for the single crossheader accident analysis assumed the process tubes that lost coolant all contain fuel at goal exposure and were all operating at the maximum assumed tube power for the reactor.

The process tube row, the last to be affected by the safety rods, is near the bottom of the reactor but still in the central zone. A tube row at maximum tube power in this region is used to compute the fuel temperature rise during the excursion. The average fuel temperature rise in this row is calculated to be increased by a factor of 1.3 at shut down.

In computing the nuclear excursion for the total loss-of-coolant accident, 1 sec is assumed to elapse between accident initiation and the start of coolant boilout. Support for this assumption is included with the CLUMSY I discussion in Appendix B. In this analysis, the maximum temperature rise would also occur near the bottom of the reactor, and the average fuel temperature would increase by a factor of 1.75 during the excursion.

Fuel Temperature Rise and Fission Product Release Computation

Following reactor shutdown, heat is generated by fission product radioactive decay. The decay heating rate is approximately 7% of the power generated by fission during operation. A computer program called ALOC (Accidental Loss of Coolant) was developed to predict the fuel temperature rise following a loss-of-coolant accident. A previous study in 1959 used a similar program,⁽¹¹⁾ but that program is no longer available and the computed fuel temperature transient was not recorded.

The ALOC program computes the heat transferred to the moderator graphite and the temperature rise of the fuel caused by the remaining heat from the fission product radioactive decay. For the total loss-of-coolant accident, an insulated tube and graphite cell is assumed. In the K Reactor case, the computed time to reach the melting point of uranium was 2.7 hr from fission product decay heating alone. This would be reduced by the stored graphite energy, assuming it is all released, to about 2 hr. Heating

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from metal-water reactions are not included because it is judged these would not affect the time to melting even though they might add heat after melting of the uranium. Uranium oxidation due to reaction with steam may cause more rapid release of fission products. Further experimental work is needed in this area. In the single-header accident analysis, a heat sink was assumed to account for the heat transfer to the adjacent rows of cooled tubes. This resulted in the computation indicating the uranium would not reach the melting temperature. For the K-Reactor accident in which coolant is lost to every other header, a conservative heat sink is more difficult to establish. Since no method for calculating the loss to the heat sink has been developed for this accident, the temperature rise used for the affected fuel is arbitrarily assumed to be the same as for the total loss-of-coolant accident.

The time for the fuel temperature to reach the uranium melting point in the smaller reactors is longer because of the lower fuel power levels. The time to melt for the total loss-of-coolant accident in the B or D Reactor was computed at about 5.7 hr, with the effect of the stored energy in the graphite included in the calculation. The single crossheader accident analysis with a heat sink has not been made for B and D Reactors, but the heat sink would keep the fuel temperature lower than at a K Reactor because of the slower heating rate and longer time available for heat to flow to the sink.

Dose Computation

The whole body dose received by an individual located at any point of interest is determined from a calculation of the exposure received as a cloud of gamma ray emitting fission products passes that point. The 10 CFR Part 100, Reactor Site Criteria⁽²⁾ defines the exclusion boundary as the location where an individual would receive a maximum of 25 rem total body dose or 300 rem to the thyroid in the first 2 hr of exposure. For this confinement study, the doses calculated are for the present site exclusion boundary at 5.5 mi and the low population zone boundary at 21 mi which is three-quarters of the distance to the population center at Richland (approximately 25,000 residents).

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The dose calculation uses a mathematical model that includes factors for plume geometry, dilution in the reactor building, ground deposition, fission product decay and change in gamma energy. The model uses atmospheric conditions chosen to maximize the total body dose at the low population zone boundary. These conditions include a wind speed of 1 m/sec. When the same conditions are used for the exclusion boundary dose calculation, the fission product cloud would travel 4.4 mi in the first 2 hr after the release is initiated. This results in a small dose at the 5.5 mi actual site boundary in 2 hr.

Because of the confinement system particulate filters and halogen collectors and the scavenging of these fission products when passed through water, the total body dose is judged relatively greater than the thyroid dose for the maximum credible accident. For the more severe accidents, the total body dose exceeds the 10 CFR Part 100 criteria when the present confinement system is considered. Thyroid doses have not yet been calculated, but they will be calculated for the final report. The total body dose calculation has been a satisfactory tool for judging the present system and potential changes or additions to the system. The results of the atmospheric release dose calculations are given on page 28.

Results of the Calculations to Date

Fuel Release Assumptions

It was assumed that all of the fission products available for release from each fuel element would be released at the start of uranium melting, but none before melting. This assumption was made because of lack of fission product release data for metallic fuel at comparable exposures. Experiments reported by Buddery and Scott showed that the noble gas release from metallic uranium below 850 °C depended upon the exposure. (12) The sample with the lowest exposure used in their tests was above the level of 0.1% uranium burnup expected in the production reactors. This test showed that less than one-half of 1% of the Kr^{85} was released during one heating cycle to 850 °C for uranium at 0.15% burnup. Another report by

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the same authors gave the total Kr⁸⁵ released below the melting point for slow heating rates, but only two data points were given.⁽¹³⁾ These data indicate up to 50% of the noble gases might be released for the expected very slow heating rate to the uranium melting point; however, the release rate would increase as the temperature increases and, even if the rate increase were linear, the majority of the release would be near the melting point. The time from initial to complete uranium melting for the hottest fuel piece is about 15 min.

In the single crossheader failure analysis, credit cannot yet be taken for the fact that no melting is expected because knowledge is not available on the fission product release rate for the computed heating cycle. Therefore, to be conservative, the assumptions given in the Hazards Summary Reviews must be used. The Hazards Summary Reviews assumption for the single header was that fuel elements containing 50% of the fission product inventory supplied by the stricken header would be melted at the time computed for an insulated maximum powered fuel element. The computed time is about 2 hr for a K Reactor fuel element. These are the numbers used in computing doses for the single crossheader accident in this report.

For the analysis of half or total loss-of-coolant accidents, the release is spread over time according to the initial melting time for each element in the reactor. This more nearly represents the release pattern where a wider spread of tube powers would be involved. (See Figure B-6 in Appendix B.) In these accidents, if some means for capturing and holding the noble gases is required to approach the 10 CFR Part 100 criteria, the prolonged release is more conservative because it dictates the largest storage volume when the gases are mixed with flowing building air.

Release Quantity

The quantity of fission products released for a single crossheader accident would be approximately the same for the B, D, and K Reactors because the two tube rows supplied by one header in B and D Reactors contain the same quantity of fission products as the one row supplied by one

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header in a K Reactor. At C Reactor, only one row of process tubes is supplied by one header so the release there would be smaller. Flow loss to half of the headers in a K Reactor would mean inadequate flow to fuel elements containing half of the reactor inventory. Total loss of coolant would, of course, mean melting in all of the reactor fuel.

Release Paths

The release for a single crossheader accident might be expected to be split with approximately 50% going through the header break and 50% going out through the effluent system due to the pressure differential both from front-to-rear and rear-to-front across the tubes attached to the stricken header. This quantity would be reduced by any fission products entering the reactor through melted aluminum tubes in B, D, and C Reactors. Because the exact reduction and holdup introduced are not known, no credit is given for this path in the analysis.

In the half and total loss-of-coolant accidents, the fission products would either pass into the effluent piping or go into the reactor graphite stack. There would be an outlet temperature surge and a corresponding pressure surge that might break the effluent piping. Therefore, the effluent would be released either into the building or out through the effluent piping to the retention basins or to both with a partial break. In these cases, however, all the release is assumed through the effluent path.

The release factors assumed for fission products released in the reactors with the present confinement system are given in Table II.

From Table II, it can be seen that all of the noble gases in the fraction of inventory contained in melted uranium will be released through either path. Also, when the release is through the effluent pipes with water flowing, it is assumed 80% of the volatile solids, particulates, and iodine released from the fuel would be captured in the water. Since the single-pass cooled reactors currently have no effective cribs for the effluent water, this 80%

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would be released into the river. A summary of the release factors for halogens, used in computing boundary dose for the three accident cases, are given in Table III. This table also indicates the release paths used in the dose computations.

Dose at Boundaries from an Atmospheric Release for Existing Confinement

The doses at the exclusion and low population zone boundaries were computed using the fission product inventory affected and the release paths indicated in Table III for the three accident cases considered. The release factors for the fission product groups from Table II were also used. Table IV gives the calculated doses at the existing boundaries and at Richland. The population center distance of $1\frac{1}{3}$ times the distance to the low population zone boundary sets the low population boundary at three-fourths of the distance to Richland (population approximately 25,000). The distance from the reactors to Richland is 28 mi so the low population zone boundary is set at 21 mi.

The single-crossheader failure dose calculations assume half of the released fission products pass through the effluent pipes with the factors shown in Table II used for this path. The remaining fission products released are assumed to pass through the supply crossheader break into the building with no credit for reduction because of contact with flowing water in the crossheader. The dose calculations for the more severe accidents assume the effluent coolant path which will result in the maximum amount of fission products being released.

According to 10 CFR Part 100, the total body dose criteria are 25 rem at the exclusion zone boundary assuming exposure in the first 2 hr following the start of the release and 25 rem at the low population zone boundary for exposure during the entire period of fission product cloud passage. Comparing the doses in Table IV with the 10 CFR Part 100 criteria shows that only the doses from a release due to a single crossheader failure would be within the criteria with the present confinement system.

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TABLE II
ASSUMED ATMOSPHERIC RELEASE FACTORS FOR FISSION PRODUCTS

<u>Fission Product Group</u>	<u>Released(a) from Fuel</u>	<u>Not Plated Out</u>	<u>Not Absorbed in Coolant</u>	<u>Passing Filter or Charcoal</u>	<u>Total Release Factor</u>
<u>Path Through Confinement System Only</u>					
Noble Gases	1.00	1.0	NA ^(b)	1.0000	1.0
Halogens	0.50	0.5	NA	0.0500	1.2×10^{-2}
Volatile Solids ^(c)	0.50	0.3	NA	0.0005	7.5×10^{-5}
All Remaining Fission Products	0.01	0.3	NA	0.0005	1.5×10^{-6}
<u>Path Through Effluent Pipes Only With Partial Water Flow</u>					
Noble Gases	1.00	1.0	1.0	1.0	1.0
Halogens	0.50	1.0	0.2	1.0	0.1
Volatile Solids	0.50	1.0	0.2	1.0	0.1
All Remaining Fission Products	0.01	1.0	0.2	1.0	2×10^{-3}

TABLE III
ASSUMED ATMOSPHERIC RELEASE FACTORS
FOR HALOGENS PRESENT CONFINEMENT SYSTEM

<u>Path</u>	<u>Inventory Affected</u>	<u>Released from Fuel</u>	<u>Path Fraction</u>	<u>Absorption or Plateout</u>	<u>Filter and Charcoal</u>	<u>Total</u>
<u>Case I. Single Header:</u>						
Building	0.01 K	0.5	0.5	0.2	NA ^(b)	5×10^{-4}
	0.02 B					1×10^{-3}
Effluent	0.01 K	0.5	0.5	0.5	0.05	6.25×10^{-5}
	0.02 B					1.25×10^{-4}
<u>Case II. 50% Loss of Coolant (K Only):</u>						
Effluent	0.5	0.5	1.0	0.2	NA	5×10^{-2}
<u>Case III. Total Loss of Coolant:^(d)</u>						
Effluent	1.0	0.5	1.0	0.5 ^(e)	NA	0.25

(a) Release factors indicate multiplication factor to apply to fission products passing these points in the system.

(b) Not applicable.

(c) Includes cesium, tellurium, and selenium.

(d) Assumes fuel does not burn.

(e) No water in effluent pipe but credit taken for plateout on piping.

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TABLE IV
TOTAL BODY DOSE FROM ATMOSPHERIC RELEASE
EXISTING CONFINEMENT SYSTEM

<u>Existing Points of Interest</u>	<u>Distance, Miles</u>	<u>Total Body Dose in rems</u>		
		<u>Postulated Accidents</u>		
		<u>Case I^(a)</u>	<u>Case II</u>	<u>Case III</u>
Exclusion Zone Boundary	5.5	< 10 ^(b)	< 25 ^(b)	< 25 ^(b)
Low Population Zone Boundary	21 (3/4 x 28)	< 1	74	230
Richland	28	< 1	40	100

(a) See Table III for identification of cases with accident and release path.

(b) Estimated Dose. Meteorological conditions chosen to maximize dose at low population zone boundary include 1 m/sec wind speed. At this speed, the cloud of fission products would travel only 4.4 mi in first 2 hr and thus not reach the boundary.

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The more serious accidents would require confinement system modifications to accommodate the resulting fission products released.

Release to the Columbia River

For the accident where coolant is lost to tubes supplied by a single crossheader, a portion of the released fission products from the damaged fuel elements would enter the Columbia River from the existing effluent system which is without means for storage. Considering the potential melt in the zirconium tubes of a K Reactor, the release path would be both through the rear effluent coolant piping system and to the break in the front crossheader. In both paths, the gases and other fission products would be thoroughly water washed. Because of the small amount of surface area within the tubes, no credit is taken for plate-out. It is assumed that about 80% of the iodines released from the uranium would be dissolved in the cooling water. This water could take one of three paths toward the Columbia River: the rear effluent piping system feeds into the 107 tanks which empty into the river via the outfall line; the water from the front face leaves by the "C" work platform pit to enter the ground percolation crib; or leaves via the work area floor drain and flows directly to the river via the outfall line. Conservatively, it is assumed that all of the water-born fission products from a K Reactor would go directly to the Columbia River. This is definitely so for the three older reactors. It is assumed melting of uranium metal would begin after 2 hr and fuel containing half of the fission product inventory in the tubes supplied by one header would melt. This would be 1% of a K Reactor's fission product inventory. The controlling radionuclide in the river water would probably be I^{131} .⁽¹⁴⁾ The melted uranium release fraction for halogens is 50% (Table II).

The amount of I^{131} entering the Columbia River during the first half hour after the onset of melting would amount to 2×10^{-5} % of the iodine inventory. Using a minimum river flow rate of 36,000 cfs, the peak concentrations of I^{131} that would occur at F Area, the nearest point of public access, and at Richland (the first point drawing water from the Columbia for municipal use) is 0.13 $\mu\text{Ci/ml}$ and 0.11 $\mu\text{Ci/ml}$, respectively. These

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concentration estimates are based on dye tests by J. P. Corley, Pacific Northwest Laboratory (to be documented at a later date).

It is estimated that the travel time from the release first entering the river (about 3 hr after shutdown) until the leading edge reaches F Area will be 5.5 hr with 17 to 20 hr required to reach Richland.⁽¹⁵⁾ These concentrations could cause an estimated thyroid dose to a child (2 g thyroid and 1 liter consumption) that would exceed 2000 rem.

A release of this magnitude would initiate the Columbia River Emergency Plan.⁽¹⁴⁾ The time available while the contamination is traveling to a point of potential exposure would permit activation of emergency procedures based on river monitoring. These emergency procedures should minimize the probability of exposure exceeding the dose criterion of 100 CFR Part 100. Procedures are in place for notification of downstream users of river water. A preliminary time estimate that the river downstream might be unfit for use is 3 weeks.

Effect of Future Operating Conditions

Current studies on changes in reactor operating conditions for diversification include increasing the power density of the fuel and modernization of the small reactors by enlarging the process channels to allow irradiation of mixed products.

It has been calculated with ALOC that the time to melt a maximum powered K Reactor element for a 50% increase in tube power would be about 1 hr; for a 100% increase, the time would be about 3 min. These are compared to 2 hr at present powers. The fission product inventory in the reactor would remain essentially the same because these cases are based on no increase in total reactor power and in summation of fuel columns exposure.

The fuel pieces used in the enlarged process channels of the modernized small reactors would be of a rod-in-tube design. When used with alternate products in adjacent channels, they would have a lower power density but a higher tube power than the present fuel. The ALOC heat transfer code

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is not now programmed to handle this geometry, and no calculations have been made. However, it is estimated that the inner uranium rod might start to melt in 5 to 10 min, but the outer uranium tube would probably not melt as soon as the current reactor fuel pieces. Each process tube would contain more fission products than at present, but the total in the reactor would be essentially the same as it is currently in these reactors.

The enlarged process channels in the small reactors would make them under-moderated, (instead of over-moderated) such that a loss-of-water accident would have a negative reactivity effect, thus contributing to their nuclear safety. Since the nuclear excursion due to loss-of-coolant accidents in the existing over-moderated reactors has little effect on the melting time, this change would not be expected to change the uranium melting time appreciably, but would provide an inherent shutdown mechanism independent of external scram devices. This mechanism could be an important safety feature.

The general effect of these contemplated changes on the effectiveness of the confinement systems would be an increase in the release and the boundary doses. This increase would be caused by earlier melting of the uranium and earlier release of the fission products. It has not been determined whether the 10 CFR Part 100 dose criteria would be exceeded for these changes or not. If provision is made to capture the initial part of the release for protection from the more severe loss-of-coolant accidents, the boundary doses would be the same with the higher powered fuel as with the current loadings.

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EXPERIMENTAL PROGRAMS

Five-Year Research and Development Program

Evaluation of the consequences of reactor accidents has still required the use of many conservative assumptions because of lack of technical information. A 5 yr research and development program has been jointly developed at Hanford by the Irradiation Processing, New Production Reactor, and Chemical Processing Departments working with Pacific Northwest Laboratory. This program will be continued by Douglas United Nuclear.

Under the nuclear safety section of this program, experiments and studies are planned to investigate the many aspects of fission product release accidents.⁽⁴⁾ Specific areas to be investigated are:

- Steam and water flow patterns in reactors during the course of an accident.
- Release of biologically significant fission products from reactors during melting of the fuel.
- Chemical form of released fission products (particularly iodine).
- Methods of retaining noble gases.
- Local meteorological studies.
- Ground disposal of radioactive liquid wastes.
- Control of plutonium and fission product particulates and aerosols resulting from fires or violent chemical reactions.

The first experimental work planned for the production reactors is to characterize the fission product release from aluminum clad fuel over the predicted heating cycle. This work is needed to help define the time sequence of events and the quantity of fission products that the confinement system must handle. Another problem area, which needs early resolution, is an experimental evaluation of the solubility of halogens and scavenging of volatiles and solids when fission products are passed through water. Experiments on the effectiveness of confinement aids such as fog spray and foam in removing fission products and organic compounds of iodine are also planned.

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Long-range plans are to continue the investigation of fission product release and potential confinement system improvements. In general, the order of investigation might be rearranged as the importance of each item is better defined.

Early tests could indicate inexpensive methods of confining the fission products are available, and the more expensive approaches such as containment are not required by the production reactors at Hanford. The results could also have potential benefit for others in the nuclear industry.

Summary of F Reactor Confinement Tests

The permanent shutdown of the F plutonium production reactor at Hanford provided the opportunity to perform full-scale tests on the confinement system used to collect the radioactive materials from the exhaust air stream. This confinement filter system was in service nearly 5 yr. The objectives of tests were to measure the efficiency of the current system and to test some potential methods of improvement.

The production reactors confinement system utilizes negative pressure and filtration to control the release of radioactive particulates and halogens to the atmosphere. Currently, the filter system is a bank of filters followed by a bank of coconut shell charcoal. The air is pushed through the filter system by electric exhaust fans, with steam-driven exhaust fans as emergency backup. Besides the filter system, a fog spray system in the reactor discharge work space is activated when the concentration of I^{131} exceeds $1 \mu\text{Ci}/\text{ft}^3$ in the exhaust air. For the tests, a special fog spray was installed in the exhaust air duct downstream of the filters, and a foam generator (designed for fire fighting) was installed in the exhaust 200 ft stack base.

The F Reactor filter system efficiency was measured by three methods: dioctylphthalate (DOP), uranine, and condensation nuclei. The test results show that the nuclei counter would be satisfactory for monitoring the integrity of the absolute filters continually. Although more testing is required to obtain the optimum aerosol for use with nuclei counter, it

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may be possible to use nuclei found in the air. The uranine dye method also promises to be satisfactory for measuring filter efficiency with the confinement system in service. However, laboratory measurement must be made of the concentration of uranine particles on the filter samples used which would mean the uranine method could be used only for periodic measurements.⁽¹⁷⁾ The F Reactor overall particulate filter efficiency measured was 99.6%.

In an iodine deposition test, radioactive I^{128} was released into the discharge area supply air, and the cleanup of the air stream caused by deposition on the reactor surfaces and in the duct was measured. The results varied with concentration of iodine released. The lowest plate-out factor of 50% was obtained when the iodine concentration in the air was low. A test with a higher iodine concentration gave a plate-out factor of 86% on the same surfaces and duct. The difference in these two plate-out factors is probably because of the organic iodide forms present.

In another test, approximately 100 Ci of I^{128} were released into the discharge area supply air with the fog spray operating. The fog spray reduced the concentration in the air exhausted from the discharge area by a factor of 500. This indicates the fog spray system has potential for greater removal factors than have been used in analyses of confinement systems.

A test was also performed to determine the iodine removal efficiency of the confinement system filter and a supplemental fog spray. The iodine was released at a location selected to minimize plate-out and to permit the maximum amount of the iodine to reach the filter system. Approximately 100 Ci of I^{128} was released into the exhaust air stream. The efficiency of the absolute filter for iodine removal was 88.2%. The absolute filter plus charcoal collector removed 99.45% of the iodine. When a supplemental fog spray was used downstream of the absolute filter and the charcoal collector, the overall efficiency was 99.94% for iodine removal. The relative humidity during this test was less than 30% at the charcoal

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collector. This test indicated an efficiency of 95.3% for the charcoal iodine collector, confirming the original design figure of 95%.⁽¹⁸⁾

The effectiveness of encapsulation of the reactor exhaust air in fire fighting foam was also tested. Sixteen Ci of A^{41} and 35 Ci of I^{128} were released into the reactor building air. The air was exhausted from the reactor without filtration, encapsulated in foam at the base of the 200 ft exhaust stack, and released to the atmosphere from the top of the stack. Most of 500,000 ft³ of foam formed in a 10 min test then settled back to the ground and released the radioactive argon and iodine to the environment over a 2 hr period. The incentive here is that a slow release rate would allow a wind direction variability to increase the dispersion of the noble gases. The foam method of fission product entrainment has been demonstrated by Silverman and Yoder.⁽¹⁹⁾ While foam techniques for firefighting purposes are well established, the foaming agent is not designed as a containment aid. However, a foam given longer bubble life with chemical additives to possibly improve its capability to hold organic compounds might have application in a confinement system.

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FACILITIES IMPROVEMENT DESCRIPTION

This section describes the different approaches studied for solution of the various problems associated with serious reactor incidents. It should be noted that these are not detailed studies and that the costs shown are only preliminary estimates.

Relationship of Calculations to Facility Improvements

As a part of the overall study, calculations and detailed analysis of postulated incidents need to be completed before final recommended facilities can be determined. Various loss-of-water incidents have been formulated and the behavior of the reactor fuel postulated. The mechanism and rate of the fission product discharge must be established based on the fuel behavior, thus requiring experimental work with irradiated fuel.

Any proposed facilities would be designed to prevent release to the environs of excessive amounts of radioactive material. Specific considerations are the magnitude of the total release, time of release, rate of release, constituents of the release, and points of release. These items are dependent on the fuel behavior and release mode.

Accident Preventive Measures - Earthquake Protection

In considering reactor accidents, it is concluded that the most probable means of incurring serious damage to the reactor areas would be from a severe earthquake; earthquake problems have received attention in recent years. Both F. Neumann, a recognized seismologist, and the U. S. Coast and Geodetic Survey have studied the earthquake potential of the Hanford Area.⁽⁵⁾ Fortunately, Hanford is located in an area of low seismic activity, and its geological makeup is such that a large amount of damping occurs with any seismic activity originating outside the immediate area. It was Neumann's opinion that an earthquake of intensity MM-6, on the Modified Mercalli scale (corresponding to a ground acceleration of about 0.1 g), represented a sound engineering design value for this area. The U. S. Coast and Geodetic Survey map placed Eastern Washington in an

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earthquake area of Zone 2. Zone 2 corresponds to earthquake intensities up through MM-7, or a ground acceleration of 0.17 g. An additional study was made of the earthquake resistance of the Hanford production reactors by Holmes and Narver, Incorporated.⁽²⁰⁾ The basis for this study was a ground acceleration of 0.33 g, corresponding to the maximum acceleration known at that time in United States experience and recorded at El Centro, California, in 1940. This acceleration is equivalent to an intensity of MM-8.

Although Neumann's recommendations appear valid, conservatism is warranted by the very serious potential consequences of a loss-of-water accident. Tentatively, a value of MM-7, corresponding to a ground acceleration of 0.17 g, will be utilized as a design basis for strengthening existing structures. Because of the extensive work in upgrading all parts of a reactor plant, the philosophy of providing a single earthquake resistant coolant system will be adopted. Therefore, only the last ditch coolant system and critical items in the reactor buildings will be strengthened; no provisions specifically for personnel protection are included.

Similarly, because of the serious potential consequences of a severe earthquake, an additional knowledgeable opinion of the earthquake potential of the Hanford area will be obtained, and, if necessary, the design criteria modified.

The original production reactor areas do not have significant resistance to earthquakes as evidenced the unreinforced concrete block walls used as filler panels in steel framed buildings, the precast concrete roof slabs which provide no diaphragm action, and high tanks designed for wind loads only. The original power plants, steam lines, sub-stations, pumping stations, and pipe lines are all vulnerable to severe earthquakes. Reactor areas constructed later (i. e., C and particularly the K Areas) have greater resistance to seismic loadings.

In the B and D Areas, both the electrical and emergency steam power from existing facilities are discounted for pumping because of the vulnerability of the sub-stations and power plants in these areas. In the

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K Area, emergency electrical power from the oil fired boilers in the 165 Buildings is assumed to be available for pumping.

The modifications required to make an entire reactor area resistant to earthquakes of moderate intensity would be prohibitive in cost; however, the cost of assuring an emergency flow of raw river water to the reactors in each area is not beyond reason. The modifications in the following paragraphs are the minimum changes considered necessary to supply raw river water to the reactors in case of an earthquake of MM-7 on the Modified Mercalli Scale (or 0.17 g in terms of horizontal ground acceleration). This list of modifications may not be conclusive but contains the major items. Items of smaller importance may develop as the problem is studied in greater detail. Total costs for this program at the B, D, and K Areas are estimated at \$2,300,000, including engineering costs and contingency. (B Area includes B and C Reactors.)

Pumps and Piping

Emergency raw water to the B and C Reactors is currently supplied from the 181-B river pumps through the 182-B diesel-driven coolant backup pumps and then to the reactors through existing coolant backup lines. The 183 filter plants and the 190 storage tanks are bypassed, so their vulnerability can be neglected.

Modifications would include replacement of the steam turbine drives on the two emergency pumps at the 181-B river pump house with diesel drives complete with fuel storage facilities. The long slender pump shaft casings require lateral support to the building, which complicates their removal; so lateral support is advised on only the emergency pumps. Most of the valves in this building are of cast iron, and replacement of those on the emergency pumps with steel valves is necessary for greater impact resistances and reliability. Some valves in the header system may also require replacement with steel to isolate the emergency pumps. Pipe penetrations at the building walls need to be examined and modified to provide for differential movement between the pipe and the building. Pipe

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supports within the 181-B Building are probably adequate and no modifications are required. Buried lines between 181-B and 182-B are steel and require no modification although stress conditions at anchor points should be examined.

Four diesel driven emergency pumps are in the 182-B pump house; necessary modifications consist of replacement of cast iron valves with steel for greater strength and providing lateral supports for critical piping and providing clearance at wall penetrations for differential movement between piping and the building. The buried lines to the B and C Reactors were recently installed using concrete cylinder pipe and are considered adequate for the assumed earthquake conditions.

Piping within the B and C Reactor buildings will have to be analyzed to determine if additional lateral support is required. In addition, all cast iron components would be replaced with steel.

The emergency raw water supply to D Reactor is essentially the same as in B Area. Therefore, piping and pump modifications at the 181-D river pump house would involve the installation of two diesel pump drives complete with fuel storage facilities, the same lateral bracing of pump shaft casings, replacement of cast iron valves, and modification of wall penetrations for differential movement.

At the 182-D pump house, there are no diesel driven emergency pumps; therefore, two diesel pump drives complete with fuel storage would be installed in addition to the other modifications, which are replacement of cast iron valves, lateral support of critical piping, and improved wall penetrations. The buried line from 182-D to the D Reactor is of cast iron and not reliable for strong earthquakes; a new steel line would be installed.

Piping within the D Reactor building would require the same modifications as B; i.e., lateral support of piping as required and replacement of cast iron fittings with steel.

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Modifications to K Area would be mainly at the 181 river pump houses. Emergency pump shaft casings need lateral support and all cast iron valves would be replaced with steel valves. Piping within the reactor buildings was designed to meet earthquake requirements. Buried lines in the area are steel, but building penetrations may need revision.

Buildings

Proposed building modifications stated herein are to improve only the pump houses, reactor building, and high tanks or chimneys adjacent to reactor buildings. Damage to other buildings would not affect the flow of emergency raw water to the reactor and therefore can be tolerated.

The 181-B pump house was enlarged with a new addition to accommodate C Reactor. This addition was not tied into the original construction, and the two sections would tend to destroy each other in a severe earthquake, thus requiring that the two sections be joined. In addition, the exterior concrete block wall filler panels provide little shear resistance, and columns would become overstressed under seismic loading. Therefore, the concrete block walls would be replaced with reinforced concrete or perhaps transite walls adequately tied into the floor, columns, and roof slab.

Columns in the 182-B pump house are also weak and should be strengthened by adding reinforced concrete shear walls between the pump room and pipe gallery. The 187-B high tanks adjacent to the reactor building are designed for wind loads only. It may be possible to reinforce them for an earthquake intensity of 0.17 g ground acceleration; however, complete replacement with new construction would probably be more economical. The 116-B ventilation chimney adjacent to the reactor building is adequate, except cracking in the unreinforced concrete foundation may occur. Prevention of this cracking and improved stability by adding reinforced concrete in the footing is not considered necessary.

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The upper part of the 105-B Reactor building should be modified because of the potential debris damage to the reactors; also this structure is needed for confinement. Necessary modifications would be replacement of the concrete block walls above the top of the reactor with insulated metal siding, strengthening the steel frame with wall bracing and lateral bracing in the winch level floor, and replacement of the precast concrete slab roof deck with metal decking to provide diaphragm action.

A small but very important item at all reactors is the replacement of the existing emergency power battery racks with racks of an earthquake resistant design.

Modifications to buildings in D Area would be essentially the same as B Area. The 181-D river pump house should be strengthened by replacing the exterior concrete block walls. The 182-D pump house would require the same shear walls as 182-B, and the 105-D Reactor building would need the same modifications as 105-B. Replacement of the 187-D high tanks would be required also.

No modifications to buildings in K Area are anticipated as these were designed for a bomb blast loading and are well braced. Materials used for construction of these buildings are better suited for earthquake resistance.

Improvement in Ability to Cope with Accidents

Confinement System Improvements

A number of potential improvements to the Hanford confinement systems would increase the basic capability or reliability for any of the incidents postulated. These improvements would not alter the current methods of confinement, nor would they provide means for capturing or delaying the release of noble gases.

Extension of the Fog Spray System

When the confinement facilities were designed and installed, the fog spray was restricted to the discharge area enclosure where the probability

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of a uranium fire from a charge-discharge accident was the greatest. The purpose of the fog spray was to absorb halogen vapors, settle out airborne particulate matter, wash down contaminated surfaces, cool exposed fuel elements, and condense any steam formed.

Considering the possibility of fission products being released in the work area or even the complete loss of coolant, it appears desirable to extend the fog spray systems to the reactor block and the work area. This would accomplish two purposes: first, any fission products escaping from the reactor core into the building would be exposed to scrubbing action; second, the fog spray system water on the reactor skin would provide some cooling to the reactor gas seals.

In addition, the work area system would be very effective for a charging face accident in which irradiated fuel might be ejected from the reactor.

To increase the reliability of the fog spray system, the last ditch coolant system would be utilized as the supply source. Approximately 85 to 100 fog spray nozzles would be located over the work area, top-of-unit, experimental levels, and at the small reactors in the inner rod rooms. The inner rod room at the K Reactors would not be included since the gas seals are located on the outer wall away from the reactor block and would not be exposed to a significant amount of heat.

Operation of the extended fog spray system will be partially automatic and manual. The rear face system will continue to be automatically activated, while the new system in the inner rod rooms, experimental levels, and particularly the top-of-unit and work areas would require manual activation. In general, manual activation would be utilized in the areas where other equipment such as electric motors or instrumentation is vulnerable to water damage.

The cost of this system is estimated at about \$80,000 per reactor or a total of \$400,000.

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Exhaust Ventilation Fan Diesel Drives

The smaller reactors (B, C, and D) have steam turbine drives on the emergency exhaust ventilation fans. Since the overhead steam lines and the power house boilers are vulnerable to damage by earthquakes and other external conditions, self-contained diesel drives are proposed. The drives would be rated at 150 hp units for B and D Reactors and 175 hp for C Reactor. They would be designed for unattended, automatic operation. A 4000 gal fuel tank would be provided to assure 2 weeks continuous operation. During incidents from single element melting to the defined maximum credible incident, both exhaust fans would operate to clear out the building as rapidly as possible. For more serious accidents, only one fan would be operated so the system could operate longer. The costs of new diesel units for the three reactors are estimated at approximately \$30,000 per reactor or a total cost of about \$100,000.

The emergency exhaust fan drives for the K Reactors were designed originally as diesel units since steam as a backup power source was not available.

Confinement Filter System

The filtration systems provided for the production reactor confinement systems at Hanford have adequate capacity to retain the particulate and halogen fission products which could be released from the maximum credible accident (failure of a single crossheader). Air flow through the filters would remove the fission product heat. Some relatively minor improvements are suggested below.

The present operational criterion for filter dust loading is to utilize the filters until dust loadings on the filters cause a reduction in the 105 Building ventilation flow rate. The final dust loadings may then approximate a pressure drop of 5 in. of water across the filters. If the humidity of the exhaust air approaches saturation when operating with comparatively loaded filters, then mud would tend to form on the filters. Mud would

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continue forming, increasing the pressure drop across the filters, until the filters would become completely plugged. Since exhaust air will approach saturation during an accident when the fog spray system is on, the particulate filters should be routinely replaced at a lower filter pressure drop, possibly 3 to 4 in. of water.

A deficiency in the existing filter systems is the inability for in-place testing. As described earlier, both the uranine dye and nuclei counter methods of testing appear adequate and it is expected that one of these will be adopted.

Some improvements have been made in the design of charcoal filters as halogen collectors; more efficient types of charcoal have become available, and there are indications that some chemical additives are helpful. Since 1959, when the confinement facilities were installed at Hanford, there have also been improvements in the design of absolute filters. These improvements are the type which would be incorporated with normal filter replacement.

Miscellaneous Improvements

Operating experience with the existing confinement facilities has disclosed minor deficiencies such as the need to reduce the maintenance required on the halogen monitors, to assure the adequacy of filtered water piping suffering from tubercule growths, and to revise electrical supplies to ensure power for major power outages. Certain emergency procedures and equipment need study to determine if they are necessary, and if the intended purpose (such as breathing air systems and exhaust damper automatic closure to 90%) will be served. The inflatable seals on the confinement filters have not experienced the expected life and require evaluation.

Confinement Aids

This section describes potential confinement features not presently incorporated into the existing confinement systems. While many separate problems were studied, several different approaches were investigated for

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some. Part of these concepts can be utilized directly, and others will require further research and development to establish practicality. In a later section of this document, combinations of confinement aids are discussed for specific accidents.

Emergency Liquid Waste Disposal

Two separate, potential problems of contaminated liquid disposal exist; however, their solution is the same. One problem is associated with the proposed extension of the fog spray system, and the second is concerned with significant fission product release through the reactor effluent.

The existing fog spray systems are located entirely in the rear face (discharge area) enclosures and drain directly to the metal storage basin in the reactor building. Due to the construction of the buildings, no other method of drainage is practical. However, expansion of the system to include the reactor block and work area presents a different drainage problem. About 3000 gpm would accumulate in the C work platform pit and 105 Building basement. Curbing would be provided in the work area, rod rooms, and experimental levels to direct the fog spray water to drains. The drains from these locations would be routed to a basement sump for pumping to an emergency waste facility. The diesel driven pump would be automatically controlled by liquid level in the sump. The sump would require a valved drain to the process sewer for periodic use during normal operations. It might be possible to utilize a gravity drain system to the waste facility. A gravity drain system would eliminate equipment and instrumentation, and most importantly it would increase the reliability of the system as an engineered safeguard.

It is also assumed that gross contamination of the reactor effluent stream would be handled similarly. As soon as it were determined that such an accident had occurred, the reactor effluent would be diverted immediately to the emergency waste facility.

Design of this facility would closely follow that of the proposed N Reactor emergency waste facility. It would be essentially an open

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concrete tank located below grade for gravity drainage of the effluent from the reactor. The emergency effluent line would have a water seal at the waste facility and would be vented back to some gas storage system. A berm would surround the waste facility for shielding purposes. The tank would be sized to receive the anticipated quantity of contaminated water from the assumed accident. If for some unexpected reason, additional space were required, overflow stand pipes would be provided to carry the excess water to a drain field located immediately below the concrete tank.

The two main parameters upon which the drain field design is based are the infiltration rate and the allowable throughput of contaminated water. The earth between the tank and the water table would filter out particulates and remove water soluble radionuclides by adsorption on the clay minerals in the soil. A critical part of the design is a water distribution facility so that the entire drain area would be uniformly watered. Pending experimental work on the specific soil conditions at each emergency waste facility site, preliminary estimates of drain field sizes have been based on work done by Pacific Northwest Laboratory for N Reactor.

The emergency waste facility size is dependent upon the design accident. The cost is estimated at \$500,000 to \$1,000,000 for each area.

Temporary Containment in Reactor Shell

The existing confinement facilities have no provisions for capturing noble gases; however, the amount released during the maximum credible accident does not result in exceeding the dose criteria for 10 CFR 100. In considering more severe accidents, the release of noble gases results in exceeding the dose criteria of 10 CFR 100. The only method of solution appearing to have any practicality is that of capturing and holding the noble gases to allow for radioactive decay.

To accomplish this, traps would be installed in the effluent lines near the reactor buildings as described in effluent line trapping, p. 50. Isolation valves would be installed in the gas system near the inlet and

and outlet of the reactor. Then a vacuum would be drawn on the downcomer vent and the discharge pumped to an underground storage tank. Activation of this system would be manual. A 250 hp diesel driven compressor, rated at 500 cfm and 250 psig discharge, would pump residual steam and fission product gases to a buried 75,000 ft³, 250 psig tank. A return line would be routed to the exhaust ventilation duct upstream of the confinement filter building for subsequent disposal of the contaminated gases.

Shortly after loss of coolant, failure of fuel cladding and central zone aluminum tubes would occur. However, with the reactor piping intact (at least from the inlet crossheaders through the effluent lines) and a water seal installed near the base of the downcomer, the fission products would still be confined. As long as pressure in the piping system and reactor block was maintained low or slightly negative, and the many gas seals function, the fission products would be contained. With the reactor shields insulating the rubber gas seals from the hot graphite and fuel, it is anticipated that the seals would not fail for a period of 10 to 20 hr. Most of the seals are neoprene or silicone rubber having short term (<24 hr) service up to about 250 and 500 °F, respectively.

It is anticipated that if the gases were collected in such a manner for a period of about 10 hr, the rest of the noble gases could be safely released to the atmosphere.

A serious drawback, however, is that there is no provision for the ultimate disposal of the reactor and fuel. Presently, it is believed that without coolant the ultimate fate of a stricken reactor would be a fire; hence, reactor flooding (described later) is considered.

As shown earlier the fission products generate decay heat at substantial rates for long periods, and some parts of the fuel and graphite are at temperatures in excess of 1000 °C in a very short period of time. It is expected that eventually the rubber gas seals would fail and the shield would crack open. Oxygen entering the reactor block through these openings

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would immediately react with any hot graphite or fuel it contacts. If this occurred, then essentially 100% of the reactor inventory of fission products would be released to the building.

Effluent Line Trapping

With loss of coolant, the initial fission product release path is out through the effluent system. A 1 to 2 psi water seal could be installed on the effluent lines near the reactor buildings.

In addition to the effluent traps, the existing effluent line vents would be routed back to the reactor confinement system.

The costs are estimated at approximately \$150,000 for the five reactors.

The Dresser coupling joints in the present effluent lines at all reactors are badly deteriorated and leak universally. While this is acceptable under normal operating conditions, repairs may be required for those sections of lines to be used as part of the confinement system. Repairs are estimated at about \$250,000.

Major Fission Product Release to the Building

If the building release is large, decay heat might destroy the confinement filters and release the fission products to the environs. The problem is aggravated when reduced emergency air flow rates are considered. Reduction of the filter heat load could be accomplished by: increasing the filter area, filter cooling, or reducing the fission product load in the ventilation air.

Increasing the number of confinement filters is an expensive modification, which would reduce the air flow per filter and its cooling ability. Since increased air flow would interfere with capturing the noble gases, the alternate is the use of water sprays or heat exchangers, and neither appears practical. The remaining solution is the reduction of the fission product load in the ventilation air. The F confinement tests show that a well-designed fog spray system in the exhaust tunnels might remove a sufficient

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fraction of the fission products. To protect the filters from excess water, moisture separators would be installed immediately upstream of the filters, and the fog spray water would drain to the waste facility. The cost of such a system is estimated at about \$75,000 per reactor.

Downcomer Strengthening

In considering substantial or complete loss of reactor coolant, the approach sections to the downcomers appear vulnerable to failure caused by the transient boiling surge. If the downcomers did fail, a severe building release of fission product would occur. Since large volume storage would be required to retain the noble gases if they were mixed with building air, continuity of the downcomer may be beneficial. In addition, filter system overloading might occur.

The most direct and economical solution to prevent a problem from this source would be to strengthen the downcomers and approach sections. Specific solutions for each reactor have not been studied as yet; however, based on some recent work in the K Reactor downcomers, the estimated cost is about \$50,000 for each reactor.

Reactor Flooding

As mentioned before, a reactor fire is the expected ultimate result of complete loss of coolant to an existing Hanford production reactor during equilibrium operation. If coolant is permanently lost, the decision could be made to deliberately flood the reactor. Water contacting hot graphite would flash to steam with some of it probably undergoing a water-gas reaction with additional hot graphite. The excess steam and any water-gas reaction products would have to be vented. The water-gas reaction is strongly endothermic, however, it would not be expected to occur over a long period of time because the rate of reaction is small at temperatures below 800 °C.

Flooding at the smaller reactors represents a problem because of their shield construction. The shields are essentially free-standing structures and cannot withstand much internal pressure. Therefore, these

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reactors would be flooded by distributing relatively small quantities of water at the top of the stack and permitting the eventual excess water to drain from the gas system pressure relief point. Then, the water would be routed to the emergency waste facility line. The distribution system would probably consist of several small lines installed in the top reflector region of the graphite where exact alignment during drilling operations would not be critical.

The K Reactor shields are stronger and could withstand the hydrostatic pressure created by filling the reactor with water. Isolation valves would be installed in the gas system piping near the reactor inlet and outlet points. Water could be introduced either at the bottom of the reactor through the gas system or through a distribution system in the top reflector region.

To vent the reactor blocks, ducts would be installed in the top shields and would lead from both the inlet and outlet gas plenums. The vented gases and excess steam would contain fission products and, therefore, could not be released directly to the environs. These gases would be routed to a quench tank or basin located near the reactor. The uncondensables and the water-gas reaction products would be collected in a low pressure holding tank over the quench basin. Whether the stored gases will have to be cooled because of the decay energy will be determined in later studies. After a sufficient decay period, these gases would be processed through the confinement filters and released.

The pipe ruptures, which are assumed to have created this total loss of coolant incident, would also rob all flow from the last ditch coolant system. Therefore, an independent source of coolant would be required. This system would supply coolant for the fog spray, stack flooding, and the quench tank cooling requirements of about 9,000 gpm total. This could be accomplished either by wells near the reactor or an independent pump and line from the river.

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Analytical and experimental work is required before this system can be properly evaluated. The reactor would be flooded as rapidly as possible, consistent with the ability to vent the stack and control the internal pressure. The possibility of a steam-uranium reaction is recognized, and the probability and extent of that reaction requires study.

Only a preliminary estimate can be given at this time. The reactor vents, quench tank, water supply, and gas storage are estimated at about \$700,000 per reactor.

Low Pressure Gas Storage

As previously mentioned, the only feasible method of controlling a noble gas release appears to be collecting and holding for a period to allow for radioactive decay. During the initial stages of this study, consideration was given to collecting and holding the building exhaust air, assumed to contain the gaseous fission products. Later it was determined that the reactor effluent system represented the primary release path, and that storage of building air was unnecessary. However, discussion of this scheme is included for information purposes.

One scheme investigated was the utilization of the emergency ventilation fans for pumping the discharge air into low pressure (atmospheric) storage. This appears to have some potential. The exhaust air would be collected for a period of several hours after an incident until the noble gas activity in the building air is low enough that release to the atmosphere would not endanger anyone outside the low population zone boundary. Use of the emergency exhaust fans reduces the requirement for additional operating equipment, thereby lowering cost and increasing reliability.

To make this scheme practical, the emergency exhaust air flow must be reduced to an absolute minimum. A major effort would be made to seal the reactor buildings so that the negative pressure in the confinement zone around the reactor block could be maintained even with a substantial reduction in the exhaust air flow rate. It is assumed that the exhaust air flows could be reduced to the range of 5,000 to 10,000 cfm. The effort

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required to achieve the desired degree of sealing will be substantial and will approach that for the 0.3 psig building seal (described later).

The low pressure storage containers are assumed to be large plastic bags inflating with an inch or less of water pressure. These bags would have no structure over them, and being exposed to the elements, would have to be replaced periodically due to deterioration. A system sized for 10 hr operating period at 10,000 cfm would utilize four units 50 ft in diameter by approximately 1200 ft in length with a volumetric capacity of about 10 million ft³. After a suitable holding period, the stored gases would be reprocessed through the filter system.

To reduce leakage from the storage units, they would be filled to only about 75% capacity. This would allow expansion of contained gases caused by the decay energy of fission products; also, external cooling of the gas bags may be required. Total leakage of the stored noble gases is estimated to be less than 0.5% per day. Deterioration of the plastic bag material is not expected to be a problem as the total dose is expected to be 100,000 to 200,000 R.

The estimated costs for the low pressure storage containers and the associated piping are about \$150,000 per reactor. In addition, the costs for sealing the building to reduce the exhaust air flow rate would approach \$1,000,000 per reactor.

A more conventional, low pressure, storage system, used by some utility companies, is a floating steel tank. Another type of low pressure, storage tank utilizes a steel shell with an internal plastic membrane. These storage schemes are considered uneconomical for such large requirements. Storage costs are estimated in the range of \$0.40 to \$0.70/ft³ or \$3,500,000 to \$6,000,000 per reactor. Building sealing would also be required.

High Pressure Gas Storage

In addition to the somewhat novel, low pressure, gas storage scheme, a system of compressing the gases into steel storage tanks was

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investigated. Two systems briefly studied were the collection of exhaust ventilation air at the rate of 10,000 cfm and the collection of the evolved gases from the effluent line at the rate of 500 cfm. Storage of building air would be unnecessary if the downcomer is strengthened so the smaller system could be utilized to collect gases from the effluent lines or from temporary containment in the reactor shell. Both systems would compress the gases into 250 psig steel tanks.

A 10,000 cfm, 250 psig compressor would utilize a 4000 hp diesel drive and would require a 3000 gal fuel storage capacity for a 10 hr operating period. A 1,000,000 ft³, 250 psig storage tank would be provided; cooling of the gas would be required after compression. A preliminary cost estimate for this system is about \$3,500,000 per reactor. An additional \$1,000,000 would be required for building sealing.

The 500 cfm, 250 psig compressor utilizes a 200 hp diesel drive and would require a 150 gal storage capacity for a 10 hr operating period. The compressed gases would be cooled before entering the storage tank. This system is estimated to cost \$250,000 per reactor.

Foam Generation

There is a possibility that high capacity foam generators similar to those used in fire fighting could be utilized to generate foam with the exhaust air containing fission products. The foam would act as the containment "vessel."

As described previously, qualitative tests were conducted at F Reactor after shutdown. It is possible to generate and expel foam from a 200 ft stack at the rate of 50,000 cfm. The foam returns to the ground quickly.

Unknowns regarding the foaming scheme are:

- The percentage of exhaust air contained in the foam

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- Means to create a long lasting foam (fire fighting foam is short lived in extremely hot weather)
- Ability to retain noble gases.

Further research and development are required to evaluate this concept.

Other Methods of Noble Gas Retention

Other methods of capturing noble gases, briefly investigated, include cryogenics, kerosene absorption, deep bed charcoal filtration, and chemical reaction. While these may be technically feasible, their associated costs are very large because of the quantities of materials to be processed and the dilution of the fission products. Further consideration of these is not anticipated.

Recirculation Facilities

A unique problem exists for the loss of water to alternate cross-headers at the K Reactors. If significant fuel damage occurs, fission products would be released through the effluent system over a long period. This release would require a tremendously large, emergency waste facility to handle the diverted water.

Recirculating facilities could be installed so that after the waste storage was filled (several days) this system could be manually actuated. The required reactor flow would be about 5000 gpm at that time. The system would consist of a diesel driven pump and heat exchangers located away from the reactor building, piping to the reactor, and a diesel driven river pump to provide coolant for the heat exchanger. Fuel storage facilities would be provided to allow 30 days unattended operation. Costs for this system are estimated at \$300,000 per reactor.

DECLASSIFIED

Other Types of Confinement or Containment

Building Sealing

The small production reactors, B, C, and D, have walls of concrete block panels between steel frames and roofs of concrete tiles covered with a built-up tar and gravel surface.

The buildings can be classified as light, non-airtight, industrial buildings except for the reinforced concrete portions.

Past studies have indicated a limit of 0.1 psi internal pressure that could be withstood with the present structures without modifications. Securing the roof decks, siding and concrete block walls could achieve a new limit of 0.3 psi.

The schemes discussed in this section, Building Sealing, Containment, and Tall Stacks were investigated under the assumption that the primary release path would be to the reactor building. Determination that the primary release path would actually be through the effluent system has reduced this section to an academic consideration.

All of the reactor buildings have reinforced concrete foundations, below grade walls and above grade shielding. The K Reactor walls are corrugated transite siding on a steel frame. The roof decks are poured gypsum or light concrete slabs covered with roofing paper, tar and gravel.

Three means of containment have been considered:

- Sealing the present building and modifying to withstand 0.3 psi.
- Rebuilding the superstructure and sealing to withstand 1.0 psi.
- Constructing a dome over the facility of approximately 400 ft diameter capable of 2 psi internal pressure.

The dome would have a volume of 10,000,000 ft³ versus 800,000 ft³ for the building schemes and be capable of containing an energy release of about 15 times the building schemes.

The optimum containment scheme for the postulated incidents for utilization and cost has not been determined.

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All containment schemes would require means for removing gases and vapors to a holding system of low pressure gas bags or tanks. The exact method would depend on the postulated incident and the calculated quantity of material to be stored. Fog spray systems would also be utilized. Costs for these items are not included below.

The three cases of containment are presented for comparison:

(1) 0.3 psi Building Sealing

Required building modifications include a seal coating on the light construction walls and installation of air locks and gas-tight doors. Penetrations such as the supply ventilation ducts and gas tunnels would require automatic closures, and the penetrations such as pipe sleeves, conduits, and process pipe tunnels would need sealing by caulking.

The building is complex architecturally and has a large number of potential leaks; therefore, 100% airtightness for positive internal pressures is not practical. However, if these modifications are made with extreme care, the leakage rate is estimated to be about 9000 cfm at a 0.3 psig differential pressure, or 1,300,000 ft³ per 24 hr if pressurized for that period. It is probable that an MM-7 intensity earthquake would cause leaking of the building seals.

The capital investment for the 0.3 psi case is estimated at about \$1 200,000 for each reactor, with a reactor outage requirement of approximately 1 month.

(2) 1.0 psi Building Sealing

The K Reactors have secondary shielding walls around the reactor that can withstand 1 psi internal pressure if tied together at the top. Present structural steel frames over the containment zone would be replaced with a frame and walls to withstand 1 psi. Building sealing would be necessary including air locks, special closures on penetrations for conduits, air ducts, etc. A thick coating (approximately 1/8 in.) of special paint would be utilized on all interior surfaces of the containment wall to seal cracks.

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Cost would be approximately \$4,300,000 for each K Reactor with an outage time of 8 mo required.

Leakage is estimated at 100% per day at 1.0 psi. This is equivalent to 800,000 ft³ per 24 hr if pressurized for 24 hr. An MM-7 earthquake might the damage the building seal.

The smaller reactors would require more work than the K's and a corresponding (but as yet unestimated) high cost to obtain a building seal capable of 1.0 psi internal pressure.

High Stack

The current use of the term "tall stacks" is applied to stacks extending 600 to 800 ft above grade. The potential for greater dispersion at higher elevations and the increased vertical distance between the surrounding population and the centerline of a plume of contaminants may significantly reduce exposure, particularly in the first few miles. The extent of the potential beneficial effect cannot be predicted with confidence. The potential use of tall stacks for the Hanford production reactors has had only cursory examination. The following reservations are only expressed qualitatively until further information is available.

(1) The mathematical models developed for describing the dispersion of contaminants are limited to the lower 200 to 300 ft of the atmosphere. The validity of these models for releases at the higher elevations is doubtful. If the models are valid, the values of the diffusion coefficients are unknown. The dispersion of a plume released at 600 to 800 ft above grade cannot be predicted with confidence.

There are a number of tall stacks in the United States that have been in use for several years. The type of service differs greatly, however, in that these stacks have a chronic near-continuous release of materials. It is not possible to analyze the experience from such stacks for potential application to postulated reactor accident conditions

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involving relatively short term releases of large quantities of radioactive material. There are no experimental data for short term releases at these high elevations that could be used to test the validity of present atmospheric dispersion models.

(2) A capping inversion frequently exists at approximately the elevation of the adjacent mountains in many climatic regions. This stable layer would serve to limit dilution in the vertical direction from a tall stack. To achieve significant exposure reduction, a stack that penetrates the stable layer in the atmosphere may be required. The Hanford terrain with the Saddle Mountains nearby might require a much taller stack to achieve greater dispersion. A local research program could determine the importance of this effect.

(3) Although increased wind speed tends to disperse contaminants quicker, it also decreases the transport time of contaminants to potential offsite exposure areas. This increased speed of contaminant movement at higher elevations might more than offset the asset of the quicker dispersion.

For various postulated accidents, noble gas radionuclides contribute a substantial portion of the predicted whole body dose. Rapid transport in the atmosphere to populated areas could increase the predicted whole body dose significantly because of less radioactive decay (and associated change in gamma energy).

(4) Potentially, release of contaminants from a tall stack under accident conditions could result in greater deposition of the contaminants at distances 10's of miles and decreasing deposition at shorter distances as compared to conventional 200 to 300 ft stacks. The area (and distance) adversely affected could be increased.

Depending upon the contaminant and the quantity released to the atmosphere, it may be better to release the material at or near the ground. An example would be a release of a few hundred curies of I^{131} . This

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material tends to deposit on vegetation and ground surfaces; thus the amount transported offsite and the subsequent exposure would be significantly reduced.

Recent literature has described a tall tower that with modifications could be utilized for a ventilation stack.⁽²¹⁾ The 3000 ft height is the approximate elevation required to penetrate the possible capping inversion caused by the Saddle Mountains. The same unanswered questions regarding the 800 ft stack remain. The stack could be designed to withstand a severe earthquake.

The estimated costs of various height stacks are:

800 ft x 20 ft diam	\$ 4,000,000
1,500 ft x 20 ft diam	\$ 8,000,000
3,000 ft x 20 ft diam	\$20,000,000

One stack would be located in each area; i.e., B, D, and K. Depending on the postulated accident, additional construction such as water effluent cribs, reactor stack flooding, and fog spray extensions may also be necessary.

2.0 psi Containment Dome

A steel dome surrounding the 105 Building and ground sealing would be the basic features of the containment dome. Capital costs would be approximately \$7,300,000, and the required reactor outage, approximately 4 mo. Leakage rate for this 2.0 psi system is estimated at 10% per day at 2.0 differential pressure. This would be equivalent to 0.1 of the structure volume or 1,000,000 ft³ per 24 hr if pressurized for that period.

The containment dome would be approximately 400 ft in diameter and about 120 ft above the ground. The maximum internal design pressure would be limited by the dead weight of the foundation required to prevent uplift. The upper portion of the structure would be an ellipsoidal or spherical shaped dome of welded steel plate over a steel framework.

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The lower portion would be a cylindrical ring of reinforced concrete, partly below grade to utilize the weight of the earth on the foundation preventing uplift. The majority of air leakage would occur through the bottom of the container. Leakage through the earth could be prevented by a plastic type membrane, under the surface, that is fused by heat at the joints and attached to the container and building foundations (either mechanically or with adhesive). The dome could be designed to withstand earthquakes.

Normal ventilation of the dome container could be accomplished by locating the dome intake on the opposite side of the building supply fans so the air would be circulated within the dome before reaching the fans. Thus, the temperature would be kept near ambient to prevent condensation. Some insulation on the steel surfaces would also be necessary.

Composite Solutions for Specific Incidents

Many individual improvements, applicable to various accidents, have been described in this document. Although it is not practical to do all of them, workable combinations of improvements are given later for the specific postulated incidents. At the end of this section Table V summarizes the proposed overall systems for each postulated incident. Table VI summarizes the other confinement systems and components studied and indicates the reasons they were not proposed for use.

No containment or building sealing proposal is included because the evaluation of the incidents has shown the problem as one of liquid effluent control rather than gaseous release within the building.

A group of general improvements are proposed regardless of the incident considered.

General Improvements

The general improvements are based primarily on earthquake considerations. The last ditch coolant systems and critical items in the reactor

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buildings would be strengthened to withstand an earthquake intensity of MM-7.

In addition to the earthquake improvements, the steam turbine drives on the emergency exhaust ventilation fans would be replaced with diesel units at B, C, and D Reactors.

Confinement filter improvements would be incorporated with normal filter replacement.

The costs of these improvements are estimated at about \$3,000,000.

Loss of Coolant to Single Crossheader

The probable release paths of fission products from the crossheader incident would be into the work area and out through the effluent system. Extension of the fog spray system and utilization of the last ditch coolant as the water source would increase the capability of the confinement system to adequately handle the release to the building.

As previously indicated, the total quantity of radionuclides released into the reactor cooling water and drains leading to the Columbia River would require either storage of the water or initiation of the Columbia River Emergency Plan. Effluent basin and tank facilities could be made available for temporary diversion and storage for radioactive decay. Since all the 107 basins and tanks currently leak, one such structure would be set aside for each reactor or area and designated for emergency use only. These tanks and basins would be restored to a leak tight condition, and suitable valving would be provided to permit rapid diversion of contaminated effluents into these structures. After a sufficient decay period, the liquid would probably be cribbed and the storage tank or basin buried to dispose of the remaining fission products. Each K Reactor could use two of its 10 million gallon 107 effluent tanks, leaving the third empty. On loss of coolant to a crossheader, the effluent would be immediately diverted to the empty 107 tank while the reactor flows would be reduced to a practical

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minimum. The fog spray water would be routed to the effluent system. While the third tank was filling, the first two would be drained, and one of them would be available for an additional 10 million gallons of storage. At B, C, and D Reactors, similar solutions could be attained. The B and D or DR 107 basin, could be held empty on standby service for diversion of effluent from a stricken reactor. The basins and tanks at each area will hold more than 24 hr of reduced reactor flow, which should contain all of the significant water borne fission products. The effluent line vents would be routed back to the confinement filters.

In addition, provisions have to be made for reducing the reactor flows to a minimum as rapidly as practical. This would require either operating personnel or reliable instrumentation. Remotely operated equipment from a central location (such as the Central Fire Station) would be a possible solution. In any case, personnel action for shutting down unnecessary operating equipment would be required prior to evacuation of the area. No costs are available for the required instrumentation.

It is recognized that some fission product release would undoubtedly escape to the river before the effluent stream was diverted; however, this would probably be small.

Future experimental work with irradiated fuel may indicate much lower fission product releases than have been assumed for this incident.

The preliminary estimate for these facilities is \$2,000,000.

Loss of Coolant to Alternate Crossheaders at K Reactors

This more serious incident would release about 50% of the reactor inventory of fission products through the primary path of the effluent system. Although the events required to incur the accident are not considered credible, a course of action which would satisfactorily reduce the consequences is indicated.

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The downcomer approach sections would be strengthened to prevent rupture during the steam surge after the sudden coolant flow decrease. Extension of the fog spray system, though not required for this incident, would be used during lesser, more probable accidents.

Because of the gross amounts of fission products involved, tanks or reservoirs located near the river bank and subject to leakage could not be utilized. Therefore, an emergency waste facility would be provided. The facilities would be similar in design to that proposed for N Reactor and would have a storage capacity of about 12 million gal of liquid waste. This is equivalent to about 48 hr reduced flow to a single riser, assuming the fog spray system is not in operation. The primary release path would be through the effluent system; the fog spray system would not be actuated except by high activity indication in the building exhaust air. To retain the noble gases from the waste facility and the effluent line would be routed to a central location and stored. These gases would be collected during the first 10 hr of the incident when essentially all of the noble gases would be released.

Low volume, recirculation facilities would be proposed. The waste facility would be filled in about 48 hr. If access to the area can be gained, the recirculating facilities would be activated. This loop would be connected to both C and D risers, and the proper valving would depend on the incident. Otherwise, the decision could be made to overflow the waste facility, utilizing the drain field below it. Based on data from N Area soil, the drain field could take an additional 180 million gal or about 60 days reduced flow.

The preliminary cost estimated for the overall facilities is \$2,800,000.

If the incident that results in the loss of coolant to alternate cross-headers at the K Reactors occurred at B, C, or D Reactors, a total loss of process coolant would occur because of differences in the respective,

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primary piping systems. It is not possible to experience loss of coolant to alternate crossheaders at B, C, and D Reactors. Protection against this accident is described below and a preliminary cost estimate for B, C, and D Reactors is \$4,800,000.

Total Loss of Process Coolant

Since the primary release path for this incident is also through the effluent system, extension of the fog spray system and strengthening of the downcomer approach sections would be done. The emergency waste facility would be sized for storage of fog spray water because there would be no source of process coolant. The fog spray would operate continuously for about 24 hr and intermittently thereafter to keep the building wet. If additional storage was needed, the overflow and drain field capacity would be used. This emergency waste facility would be only about one-half the size described for the alternate crossheader case. Reactor flooding facilities would be required for this incident.

Certain personnel action prior to evacuation would be required to assure an orderly plant shutdown and minimize equipment damage. Remotely operated equipment and instrumentation would permit monitoring of the incident at a safe distance.

The preliminary estimate for the overall system is \$7,900,000.

Compatible Solution for any Postulated Incident

The facilities described for the most serious accidents, loss of coolant to alternate crossheaders and total loss of coolant, would reduce the consequences of those specific accidents and the single crossheader accident to acceptable levels. However, the solutions for the more serious accidents do not encompass one another because of the large amount of liquid waste created by the remaining remaining reactor coolant flow in one case and the requirement for stack flooding facilities in the other. A solution compatible with all postulated incidents would include the facilities

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described above for total loss of coolant at all reactors, plus the recirculating systems and larger liquid waste facility at the K Reactors discussed on pages 64 and 65. The preliminary estimate for these overall facilities is \$9,000,000.

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TABLE V
CONFINEMENT SYSTEM IMPROVEMENT SUMMARY

Improvement	Feasible	Benefit	Cost in Thousands	Remarks
1. General Improvements				
System	Yes		\$3,000	These general improvements would be implemented in addition to the systems listed below for any specific loss of coolant incident.
Earthquake resistance		Elimination of most probable cause of total loss of coolant		
Extension of fog spray systems		Reduction of building contamination resulting from a charge face incident		
Diesel drives on emergency exhaust fans at B, C, and D		Increased reliability		
Miscellaneous instrumentation and control improvements		Increased reliability		
2. Loss of Coolant to Single Crossheader				
System (a)	Yes	Prevent major contamination of river	2,000 ^(a)	The atmospheric release from this incident is adequately controlled by existing confinement system, but the estimated river contamination would require initiation of the Columbia River Emergency Plan.
Modification of effluent basins and valving to provide storage		Prevent uncontrolled release of fission products to river		
Remote instrumentation, controls, and shutdown equipment		Provide for orderly plant shutdown and monitoring of incident		
3. Loss of Coolant to Alternate Crossheaders at K. Total Loss of Coolant at B, C, D				
System (a)	Yes	Atmospheric dose reduction ~3 and prevention of river contamination	7,600 ^(a)	Based on 1) storage with subsequent processing and controlled release of gaseous fission products, and 2) permanent storage of reactor effluent and contaminated fog spray water.
Liquid waste facility - all		Prevention of major river contamination		
Downcomer strengthening - all				
Recirculating facilities - K				
Remote instrumentation, controls, and shutdown equipment - all				
Gas storage - all				
Reactor flooding systems - B, C, D		Atmospheric dose reduction from 75 rem at LPZ boundary to < 25 rem		
Independent water source - B, C, D				
Reactor block venting - B, C, D				
Quench tank - B, C, D				
Effluent traps - all				
4. Total Loss of Coolant - All Reactors				
System (a)	Yes	Atmospheric dose reduction ~9 and prevention of river contamination	7,900 ^(a)	Based on 1) storage with subsequent processing and controlled release of gaseous fission products, and 2) permanent storage of contaminated fog spray water.
Liquid waste storage		Prevention of major river contamination		
Downcomer strengthening				
Remote instrumentation, controls, and shutdown equipment				
Reactor flooding system				
Independent water source		Atmospheric dose reduction from 230 rem at LPZ boundary to ~ 25 rem		
Reactor block venting				
Quench tank				
Gas storage				
Effluent traps				
5. Facilities for All Above Incidents				
System (a)	Yes	Atmospheric dose reduction ~9 and prevention of river contamination	9,000 ^(a)	Based on 1) storage with subsequent processing and controlled release of gaseous fission products, and 2) permanent storage of contaminated fog spray water.
Liquid waste facility		Prevention of major river contamination		
Downcomer strengthening				
Remote instrumentation, controls, and shutdown equipment				
Reactor flooding system				
Independent water source		Atmospheric dose reduction from 230 rem at LPZ boundary to ~ 25 rem		
Reactor block venting				
Quench tank				
Gas storage				
Effluent traps				
Recirculating facilities (K only)				

(a) Would also require General Improvements at an additional cost of \$3,000,000.

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TABLE VI
CONFINEMENT SCHEMES AND COMPONENTS NOT INCLUDED IN PROPOSED SYSTEMS

<u>Improvement</u>	<u>Feasible</u>	<u>Benefit</u>	<u>Cost/Reactor in Thousands</u>	<u>Remarks</u>
<u>Containment</u>				
Sealed Reactor Building: 0.3 psig 1.0 psig	No	Controlled release	\$1,000 4,000	Leakage rates excessive and does not control river contamination.
Containment Dome	No	Controlled, delayed release	7,500	Leakage rate high and does not control river contamination.
Temporary Containment in Reactor Shell	No	Provide storage for gaseous fission products and temporary holdup of all fission products	300	Provides no reactor coolant, bulk of fission products would ultimately escape reactor block.
<u>Storage</u>				
Low Pressure Storage of Building Exhaust Air: Plastic bags + building seal Steel tanks + building seal	No	Temporary storage of noble gases	1,300 4,500	Costly and does not protect against major release path of the more severe incidents. Low pressure storage on a small scale is included in the facilities to protect against accidents more severe than loss of coolant to a single cross-header.
High Pressure Storage of Building Exhaust Air	No	No advantage over low pressure storage		
Foam Generation	No	Temporary encapsulation and delayed release of noble gases	100	
Chemical Reaction	No	Isolation of noble gases	--	Uneconomical because of dilute gases
Cryogenic Application	No	Separation of noble gases	--	Uneconomical because of dilute gases
<u>Delayed Release</u>				
Deep Bed Charcoal Filters	No	Delayed release of noble gases	--	Holdup of noble gases too short to be practical.
<u>Dispersion</u>				
Tall Stack 800 ft 3000 ft	Indeterminate	Controlled release	4,000 20,000	Requires research to demonstrate feasibility in this specific region. Uneconomical and does not control release to the river.

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APPENDIX ASEISMOLOGICAL CHARACTERISTICS OF THE HANFORD REGION

Seismology is largely a science in which the physical effects of naturally occurring earthquakes must be observed and interpreted. Earthquakes cannot be routinely simulated. Seismology, therefore, is largely historical, and the possible effects of earthquakes of a given magnitude in a given area cannot be well predicted. Moreover, knowledge of the structures of the earth beneath about 10 km depth, where most of the damaging earthquakes originate, currently can be determined only from the earthquakes themselves. The seismic characteristics of a region, then, can generally be determined only after a long history of earthquakes.

About eight recording seismographs have been installed in the Pacific Northwest, most of them within the last 20 yr. They, together with the very recently constructed Blue Mountain Seismological Observatory near Baker, Oregon, now place Hanford in an excellent position to learn more of the very minor earthquakes that constitute much of the microseismic activity of most regions. Thus, as recently as July 26, 1965, the epicenters of two very minor quakes were pinpointed about 20 mi southwest of Richland. So minor were the quakes that no one reported feeling them and they appeared to be noted only by seismographs and by reactor galvanometers at Hanford. Neither the cause nor depth of focus could be estimated.

The overall stability of the Hanford region is generally well recognized by the lack of damaging earthquakes. This is most likely attributable to its position midway between the tectonically more active areas in the Cascade Range to the west, the Rocky Mountains to the east, the Okanogan Highlands to the north, and the Blue Mountains to the south. All those areas tend to show a somewhat greater earthquake incidence than the Columbia River plateau area in which Hanford is centered. The great thickness of the high density basalts beneath Hanford, the mode of their accumulation, and the mechanics of the development of the basin are significant. Figure A-1, Geologic Cross Sections, Columbia River Basalt

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Plateau, shows that the basalts are at least 10,000 ft thick, as indicated by the Standard Oil Company of California, Rattlesnake No. 1, test well; they may be 15,000 ft thick. The mountain ridges such as the Rattlesnake Hills, Horse Heaven Hills, and Saddle Mountains that extend into the basin from the Cascade Range to the west repeatedly have been demonstrated to be folded rather than faulted as are the Basin and Range mountains of Nevada with which they first were equated. Necessary adjustment for relief of stresses that accumulate both in the mountain ridges and the basin in general result in slight slippage on the many joints characteristic of basalts and by movement along the many beds of sediments between basalt flows (Figure A-2, The Columbia River Basalt Series). The emission of the flows also appears to have occurred dominantly, if not wholly, in the Blue Mountain area of uplift well to the south and southeast, marginally to the plateau. Epicenters or potential epicenters of damaging-intensity earthquakes beneath Hanford are not evident.

The major potential threat of earthquakes at Hanford lies from those originating elsewhere in Washington, dominantly from the Puget Sound area and to a lesser extent from the Walla Walla area (see Figure A-1). Many quakes originating elsewhere (roughly 30) have been recorded as detected at Hanford by reactor instrumentation. These include the Hebgen Lake (West Yellowstone) shock and about seven of its aftershocks; the Anchorage, Alaska, quake; and the Puget Sound quake of April 29, 1965. Historically, these and early quakes show that ground motion up to only Modified Mercalli 4 intensity appears the rule.

It is important to note, moreover, that intensities at Hanford are less than the intensities noted at towns within 50 mi distance, including in some cases even the Tri-Cities. The late Frank Neumann, former Chief Seismologist of the U. S. Coast and Geodetic Survey and later of the University of Washington, attributed this to the more favorable foundation conditions at Hanford than elsewhere.⁽²²⁾ Thus, at distances of 160 mi from the epicenter of the Olympia, Washington, earthquake of April 13, 1949, MM intensities ranged from slightly less than 3 to slightly more than 7.

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A-4

APPENDIX A

DUN-200

FIGURE A-2

The Columbia River Basalt Series

Hanford, at that distance, experienced MM-3 to 4 intensities, the near minimum at that distance. Basically, the thick sequence of sediments (see Figure A-2) overlying the basalt filter and absorb most of the earthquake waves. Exploration work by Geophysical Service, Incorporated, of Dallas, Texas, in 1963 reiterated the filtering and absorption,⁽²³⁾ as did several documented large-scale quarry blasts.⁽²⁴⁾ Large-sized charges of explosives are necessary to transmit suitable energy even for meaningful records.

The potential for damage to structures exists, but appears to require earthquakes originating close at hand emitting waves that are amplified as they pass through the sediments. Such amplification can take place with waves with a period of about 0.2 sec. Waves of that period normally are filtered out and only the longer period waves are propagated forward. If buildings with similar periods of vibration lie in the wave path, a coupling relationship could conceivably result in damage from a low-intensity earthquake. This becomes increasingly possible with highly complex facilities such as exist at Hanford.

For building code purposes, the entire area east of the Cascades in Washington is in Zone 2 of the Seismic Probability Map. Professor Neumann, an author of the map, says however:⁽²²⁾

"It is felt that local geology provides a better guide to the maximum lateral earthquake forces that may be exerted on structures than this more or less arbitrary scheme of zoning. In other words the zone figures are based broadly on the worst that has been experienced (mostly on poorly consolidated soils) and that these figures could be revised downward as the quality of the foundation material improves."

A few comparisons are desirable. The Walla Walla quake of 1935 was MM-7 at the epicenter, MM-4 at Pasco; the Olympia quake of 1936 was MM-7 at the epicenter, MM-1-3 at Pasco; the Olympia quake of 1949 was slightly above MM-8 at the epicenter, MM-3-4 at Hanford (MM-3-5 estimated at Pasco and Richland); the Hebgen Lake quake of 1959 was MM-7 at the epicenter and MM-2-3 at Hanford; the Cougar, Washington quake of

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1961 was MM-6 at the epicenter and MM-1-3 at Richland but less than MM-2 at Hanford; the Alaska quake of March 27, 1964, was only MM-2-3 at Hanford; and the Puget Sound quake of April 29, 1965, was only MM-3-4 at Hanford.

Studies of the response of reactor instruments to earthquakes show that different reactors respond differently to the same earthquake, and that each reactor responds differently to earthquakes of similar intensity. These differences are believed real and probably reflect different foundation conditions, and building design and construction as well as somewhat different characteristics of each earthquake. A set of strong motion seismographs or accelographs have been ordered to measure the ground motion from earthquakes. One will be installed on basalt bedrock, one on the alluvial materials near a building of interest, and the third in the building. So installed, they will provide a measure of the coupling relationship and possible amplification of ground motion with actual earthquakes.

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APPENDIX B

DETAILS OF COMPUTATIONS USED IN THE CONFINEMENT STUDY

Fission Product Inventory

Radio-Isotope Buildup and Decay (RIBD) is the name given to a Hanford-developed computer program for the calculation of in-reactor fission product inventories and decay heat generation rates.

When used for fission product calculations, RIBD includes a library containing pertinent nuclear data for 450 fission product nuclides including both ground states and isomeric states where they are known to exist, whether formed directly from fissions of U^{235} , U^{238} , and Pu^{239} or subsequent to fission as a result of neutron capture and/or radioactive decay.

RIBD organizes the various members described by the library into a grid with the various linkages established from the chain branching data, yield data, and neutron capture cross sections with their branching ratios. Radioactive decay includes not only the simple member-to-member cascade but also the more complex forms where branching may partially or completely skip one or two intervening members. A sample of a portion of a grid is shown in Figure B-1. Upper boxes of a pair, where they exist, refer to isomeric states; lower boxes refer to ground states. Y25 and Y49 refer to variable direct yield rates from U^{235} and Pu^{239} , respectively. Horizontal transitions between boxes are by neutron capture; vertical transitions are by beta decay and/or isomeric transition. Ground states are allowed to have up to seven source paths, isomeric states up to five source paths. Since time factors and neutron flux levels exert competing influences on each other, inventories of nuclides are affected to varying degrees by the reactor mode of operation. Consequently, RIBD was designed to follow the operating history of the fuel in detail as closely as the user desires through the entire residence period including power level changes, shutdowns, and outages.

DECLASSIFIED

B-2

DUN-200

Element
Atomic
Number

FIGURE B-1

Sample Reaction Grid Used in RIBD

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The output report from the program includes six optional combinations of activity levels (expressed in curies), concentrations (expressed in gram-atoms), heat generation rates (both beta and gamma), and integrated energy releases between time points for any set of points (up to 90) following shutdown that the user desires. Complete information is presented for each nuclide arranged element-by-element with summary sheets of elemental subtotals and grand total. A class summary is also written for noble gases, halogens, volatile solids, and all remaining fission products. Heat generation from the decay of U^{239} and Np^{239} (if applicable) are also detailed and included in the totals as is the fission energy liberated during the scram transient. In addition to the mode of operation summarized above, RIBD is also used as a subprogram to generate fission product inventories for program TYPHON to be described in the dose computation section of this Appendix.

The fission product library presently in use with RIBD has been scanned and updated five times during the past 2 yr, the latest revisions having been made in May, 1965. The fission yield data were obtained from the work of Yu A. Zysin; A. A. Lbov, and L. I. Sel'chenkov.⁽²⁵⁾ These data were compiled from over 200 literature sources and appear to favor data presented by Catcoff.⁽²⁶⁾ Beta and gamma energies released during isotope or isomeric decay along with their branching ratios were developed from the decay schemes presented by Landolt-Bornstein.⁽²⁷⁾ Neutron capture cross sections and their branching ratios were taken from Goldman.⁽²⁸⁾ Half lives were obtained by cross checking all of the above sources.

Kinetics Calculations for Loss-of-Coolant Accidents

Nuclear excursions resulting from loss-of-coolant flow and subsequent boilout which causes a compounding reactivity gain are simulated with the reactor kinetics code CLUMSY I.⁽¹⁰⁾ This code solves the time-dependent, one-energy group diffusion equation in one space dimension by

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a finite difference technique on the neutron leakage term, either in the x, y, or z direction for cartesian coordinates or r direction for cylindrical coordinates. As many as 50 nodes may be used. The calculations of the transient neutron concentration includes the following as input for each node: up to seven delayed neutron groups, linear and nonlinear (tabular) reactivity changes, and fuel and coolant temperature coefficients of reactivity. In addition to these feedback effects, there are tables for each node which exist solely for the boilout accident simulation. These tables provide variable feedback as a function of integrated tube power (kW-sec) for reactivity effects and for decreasing heat transfer as the coolant is boiled out. The bases for these functions (as well as the functions themselves) were obtained from HW-52340. (29)

Since the code treats each nodal calculation separately, the boilout function may be applied to one or all nodes. Therefore, the simulation of coolant loss to part or all of the reactor is possible.

For all coolant loss cases, the reactor model uses rectangular geometry. The reactor model and the nodal points are joined in the z (vertical) direction. Envisioning the model, one should imagine the reactor as n horizontal slabs, with fixed neutron leakage in the x and y directions. The leakage effects in the z direction are calculated as previously noted. This model allows the loss of coolant to one or more rows of process tubes, and further allows a reactivity simulation of the vertical safety rod system descending into the reactor. Thus, at a given time for a typical transient, the top of the reactor may be shutting down due to partial insertion of the safety system, while the bottom of the reactor may still be supercritical. Figure B-2 provides a graphic representation of the reactor for calculational purposes.

Two cases of interest are discussed to display types of results: loss of coolant while at equilibrium power to (1) two rows of process tubes (one crossheader feeds two rows of process tubes in the small reactors and one row

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FIGURE B-2
Reactor Model Used in CLUMSY I

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in the K Reactors), and (2) the whole reactor. In the first case, boiling is assumed to occur instantaneously, since the crossheaders join directly to a row of process tubes. In the second case, a slower time-to-boiling of 1 sec is assumed. The boilout functions (reactivity and heat transfer) are started at the time of initial boiling, and they are driven by the integrated tube power. As coolant is boiled out, there is an increase in reactivity which causes the power to increase. The variables then are interdependent, and an escalation effect is produced.

The assumption that boilout starts in the tube at 1 sec after the incident that causes total loss of flow to the reactor is based on the following:

The boilout would not occur before water flow is practically stopped. Investigation of the mechanism which might cause sudden loss of flow indicates the only natural event which has any potential for causing simultaneous rupture of the two or more of the major supply pipes adjacent to the reactor would be an earthquake. Rupture of two pipes at B, C, and D Reactors and three pipes at the K Reactors would be required for total loss of coolant at any one of the reactors provided the valves between the two sides of the primary systems are kept closed during operation. Analyzing an earthquake and attempting to determine what would be involved in order to have a pipe failure occur in less than 1 sec will show the conservation of the time assumed. An analysis of the risers and main coolant supply lines from the pump manifold for the B and D Reactors was made by Holmes and Narver, Incorporated.⁽²⁰⁾ Their report indicates that even using as the basis the most violent motion registered instrumentally prior to their study, the El Centro, California, 1940 earthquake,⁽⁵⁾ the pipes would be torsionally stressed to about 3200 psi. This stress is less than 20% of the torsional fatigue limit for carbon steel piping. These pipes have a natural period of about 1 sec which would mean that failure in less than one cycle, 1 sec, would require stresses well above the fatigue limit. The fact that this would have to occur in two or more pipes makes the probability of occurrence even more remote. The K Reactors are judged to have greater resistance to earthquake caused pipe failure in the critical region near the reactor. Other factors which would also tend to add to the time to failure are: time from the first earthquake tremor reaching the reactor scram level of MM-4 to a tremor of sufficient magnitude to break pipe, time for the pipe to fall away from the other side of break and cease to pressurize reactor, and inertial flow decay time.

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For the case of coolant loss to a crossheader, the K Reactors and small reactors are discussed as one case since the results are the same for both. Maximum expected tube operating conditions and fuel exposure are assumed for all the tubes losing coolant. The power excursion for this case is relatively mild and is essentially localized to the rows of process tubes fed by the affected crossheaders. The fuel temperatures in the adjoining rows do not rise appreciably, and boiling occurs only in the affected tubes. The average fuel temperature in the affected tubes increased by a factor of 1.3 during the 2 sec excursion. The failing crossheader causing the largest nuclear excursion would be located just above the bottom enrichment ring where the power density is high and the safety system reaches last.

The accident conditions assumed for the case of total loss of coolant apply to a K Reactor (results for the small reactors are very similar) operating at 5000 MW and at an average exposure of 600 MWd/t (which is well above the average for the current charge-discharge scheme). The coolant is assumed to reach the boiling point 1 sec after inlet pressure loss caused by the initiating event. The reactor safety system is scrambled by a low pressure trip from one of the sensitive pressure gages on each process tube.

The excursion for this case in the K Reactors results in a local (bottom of reactor) power increase of a factor of 3.5. The fuel temperature in that region increased by a factor of 1.75 by shutdown. There was a factor of 1.5 power increase in the center of the reactor, and almost none in the top of the reactor (where the safety system reaches in about 1 sec).

The resulting fuel temperature transients for three locations in the reactor (top E-ring, middle, bottom E-ring) are shown in Figure B-3.

Heat Transfer

The generation of significant heat continues after reactor scrams due to the radioactive decay of the fission products. Just following cessation

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B-8

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FIGURE B-3

Spatial Dependence of Fuel Temperature
for Total Coolant Loss Accident

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of fission power, the fission products generate approximately 7% of the equilibrium operating power. If this heat is not continually removed by some means, the fuel will eventually melt. The time after coolant loss at which the fuel will melt is of importance in confinement and containment work. Therefore, the ALOC (Accidental Loss of Coolant) computer program was developed to predict the time required for a fuel element to melt following a loss of coolant.⁽³⁰⁾ A study on the temperature transient following a loss of coolant was performed in 1959 by Ekern, Muraoka, and Zaloudek for the old solid fuel formerly used.⁽¹¹⁾ Since that time the reactor fuel has been changed to an internally and externally cooled design designated as I&E fuel. Furthermore, improved fission product heat generation rate data are now available. The ALOC code calculates the temperature transient in an I&E fuel element and uses data from the RIBD computer program for heat generation from fission products. No depletion was assumed.

Immediately following a flow loss to the operating reactor, the coolant remaining in the process tube will begin to increase in temperature until coolant boiling occurs, and the reactor scram will be initiated in less than 0.2 sec (maximum Panellit response time allowed by Process Standards) after the flow loss. After boiling starts in the process tube, the generated steam will expel the remaining water out of the tube. The rate at which this process proceeds depends on the tube power and the time required for the reactor to scram.

The starting point for ALOC computer code calculations is the time at which the process tube has been voided of coolant. The time required for coolant boilout was experimentally investigated by Pound and Busselman⁽²⁹⁾ for the old solid fuel type. Since data for I&E type fuel are not currently available, the data for the old solid type fuel are used. This is considered a fair assumption since despite the increase in the surface area available for heat transfer there is more water in the process tube and the average fuel element temperature is lower with the I&E type fuel.

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The mathematical model used to simulate the fuel temperature transient consists of dividing the physical system into circular nodes and using a forward finite difference technique. In the model, only radial heat transfer is considered, assuming that axial and angular heat flow is negligible. The physical system is broken into 15 radial nodes as listed below.

Node Number

1	Inner fuel jacket
2-5	Uranium fuel
6	Outer fuel jacket
7	Process tube
8-11	Graphite tube block
12-15	Graphite filler block

The general mathematical equation describing the heat transfer in each node is:

$$\frac{dT_i}{d\zeta} = \frac{1}{\rho_i C_i A_i} \left(\frac{T_{i+1} - T_i}{R_i} + \frac{T_i - T_{i-1}}{R_{i-1}} + H_i \right)$$

where, A_i = cross sectional area of node i

C_i = specific heat of material in node i

H_i = heat generation rate in node i

R_i = thermal resistance between nodes i and $i+1$

T_i = temperature of node i

ρ_i = density of material in node i

ζ = time.

Temperature dependent material properties are used in the computer code. To investigate the uranium melting time as a function of the axial fuel position, the equilibrium power for a piece of fuel at the point of interest is used assuming the axial power distribution is a cosine.

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It is assumed that for a total loss of coolant the outermost node (node 15) is completely insulated. However, for a loss of coolant to a single crossheader, the graphite temperature is assumed to increase to 950 °C and then remain at 950 °C due to cooling by the process tubes above and below the tubes of the stricken crossheader. For the loss of coolant to half of the headers, the heat sink effect has not yet been calculated, so the insulated case is used. When uranium and aluminum are in contact at temperatures greater than 630 °C and the heating is slow, a diffusion process can occur and an aluminum uranium alloy UAl_3 may form. UAl_3 melts at a temperature greater than the melting temperature of uranium. The ALOC code allows the choice of assumptions of whether the aluminum jacket will melt or form an aluminum-uranium alloy. If the assumption that the aluminum jacket melts is selected, the ALOC code assumes that the molten jacket slumps down and occupies a portion of the annulus between the fuel and the process tube. The aluminum process tubes, as in B, D, and C Reactors, and the fuel elements are assumed to remain centered after the tube melts. This is a conservative assumption for these reactors since after the tube melts, the fuel would fall making contact with the graphite tube block causing a decreased thermal resistance between the fuel and the graphite. In the K Reactors, which have Zircaloy-2 tubes, this assumption is less conservative, but even these tubes would be expected to sag and contact the graphite at temperatures approaching the melting point of uranium. The assumed sources of heat following an accident are: (1) the heat generation due to delayed neutron fissioning during the scram which is obtained from output of the CLUMSY computer code, ⁽¹⁰⁾ and (2) the heat generation from fission products which is obtained from the RIBD computer code. The program does not account for the reduction in heat generation in the fuel as fission products are released. When the reactor is at equilibrium operation, approximately 95% of the heat generated is released in the fuel, while during shutdown only about 89% of the heat is generated in the fuel. ⁽³¹⁾ To be conservative, the ALOC codes assumes 95% of the heat is always released in the fuel.

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The effect of the possible release of graphite stored energy is to raise the graphite temperature and shorten the time to uranium melting. If in the B and D Reactors it is released prior to melting, the temperature rise would be about 50 °C. In the K and C Reactors, the energy would give a maximum temperature rise of about 30 °C. The possible heat generated by water-aluminum and steam-uranium reactions are neglected. Experimental investigations have shown that the reactivity of water on molten aluminum is negligible up to 1170 °C.⁽³²⁾ Therefore, this reaction should have essentially no effect on the time to uranium fuel melting. The kinetic rate data on the steam-uranium reaction are very scattered and have a variation of over an order of magnitude.⁽³³⁾ It was calculated that the heat of reaction for a single fuel (using the highest kinetic rate data for temperatures below the melting temperature of uranium) is about 900 Btu/hr. The heat generation in a center fuel element in a 2000 kW K Reactor tube at the time of scram is 2.57×10^5 Btu/hr and at 1 hr after a scram is about 2.57×10^3 Btu/hr. Neglecting the possible heat of the steam-uranium reaction is not considered to cause a serious error, because appreciable quantities of steam would not be available in the total loss of coolant accident; and for the other accidents steam would be available, because there would be a sufficient pressure differential across the tubes, to provide more than enough cooling to offset the heating from this reaction.

The ALOC code has been used to investigate several different coolant loss cases for the K Reactors. Figure B-4 is a plot of the temperature transient for the hottest fuel element in a maximum powered tube with no heat loss. The graphite temperature transient has also been included as an aid to the following discussion.

During the first few seconds while the reactor is scrambling, the heat generation is quite high and consequently, the fuel temperature increases quite rapidly up to the melting point of the aluminum jacket. The thermal resistance between the fuel and the graphite is greatly reduced following the melting and slumping of the fuel jacket causing a decreasing rate of fuel

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FIGURE B-4
Temperature Transients Following a Loss of Coolant
for a 2000 kW K Reactor Process Tube

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temperature rise after the fuel and tube temperatures increase beyond the graphite temperature. Heat is transferred from the fuel to the graphite at a rate increasing with the fuel to graphite temperature difference resulting in a greatly decreased rate of fuel temperature rise above 1000 °C.

It was found that for a single crossheader accident, the uranium does not melt. The time required for the hottest piece of uranium in a process tube to melt, following a total loss of coolant, as a function of the power is shown in Figure B-5. If it is assumed that the aluminum fuel jacket alloys with the uranium, the fuel in a 2000 kW tube at K Reactor was found to melt in 5 to 6 min. However, very little UAl alloy formation should be expected due to the rapid rate of temperature rise through the alloying minimum temperature of 630 to 660 °C, the melting point of aluminum, where the aluminum would slump to the bottom of the channel. In melting tests with unirradiated elements, the molten aluminum cladding was observed to slump to the bottom of the channel. ⁽³⁴⁾

Figure B-6 is a plot of the fractional release of fission products computed for the K Reactors following a total loss of coolant as a function of time. To define a rate of release of fission products for this curve, it is being assumed that all the fission products in an element that are going to be released are released at the time the uranium melts. The reasoning behind this includes the possibility that some of the fission product gases may escape at the time the fuel jacket melts or fails. Sufficient data do not exist for the proper treatment of this effect on the rate of release of fission products. Experimental boilout tests and fuel melting tests are needed to better define the time and rate of release of fission products following a loss of coolant.

The program has been used to investigate a few cases for the old reactors, and the melting times were found to be significantly longer than for the K Reactors. The elapsed time to uranium melting after the total loss-of-coolant accident is computed to be 5.7 hr with total graphite energy release assumed.

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FIGURE B-5
Time for Melting
After Coolant Loss at K Reactor

Fraction of Fission Products

FIGURE B-6

Fraction of Fission Products Available for Release
from Melted Uranium Versus Time After Total Coolant Loss
(K Reactor at Current Administrative Power Level Limit)

Dose Calculation

The penetrating total body dose from a potential atmospheric release of fission products is determined from the centerline air concentrations at ground level, of gamma emitting radionuclides time-integrated over the time of cloud passage. A model is used to calculate total body dose. (35-37) The model takes into consideration radionuclide fractionation, radioactive decay during atmospheric transport, dilution of source term, and wind variability, in addition to the other parameters normally included in atmospheric release models. Due to its complexities, the model has been programmed for analysis by electronic digital computers. The program is named TYPHON. For the sake of simplicity, the following mathematical description applies to only that part of the model related to one group of parent radionuclides; e.g., either noble gases, halogens, volatile solids or all remaining fission products:

$$D_{TB} = 0.26 (\Lambda Q)_0 \exp(\Lambda T) \sum_{i=1} \left\{ F(T, \tau)_i \left(\frac{E}{Q} \right) \Delta \tau_i \right.$$

$$\left. \int_{T+i-1}^{T+i} \left[\frac{\Lambda}{E_Y} \exp(-\lambda t) \right]_i \exp(-\lambda t) d\tau' \right\}$$

where:

D_{TB} = total body penetrating dose in rems from one group of parent nuclides

$(\Lambda Q)_0$ = initial leak rate of parent nuclides in curies per unit time at some reference time after shutdown

Q = sum of parent radionuclides in curies

Λ = fractional volumetric leak rate per unit time

T = atmospheric transport time in unit time

$F(T, \tau)_i$ = finite cloud size correction factor as a function of transport time and duration of exposure through time interval i (dimensionless)

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$(\frac{E}{Q}) \Delta \tau_i$ = normalized time-integrated air concentration at ground level, corrected for cloud depletion by ground deposition, during the i^{th} time interval in μC_i -sec/cm³ per curie emitted

$\Delta \tau_i$ = $\tau_i - \tau_{i-1}$, the i^{th} time interval in unit time

τ_i = duration of exposure or emission in unit time

$\hat{E}_\gamma$ = effective gamma energy of parent j in MeV per disintegration

λ = radioactive decay constant of parent j per unit time

t = time after shutdown in unit time

The RIBD computer program furnishes the fission product inventory in units of curies and the effective gamma energy in MeV per disintegration, for each radionuclide present.

The following parameters are required input data for the TYPHON code: (1) inventory of fission products released in four categories, noble gases (radioisotopes of xenon and krypton), halogens (radioisotopes of bromine and iodine), volatile solids (radioisotopes of cesium, tellurium, and selenium), A.R.F.P. (all remaining fission product radionuclides); (2) the fractional leak rate constant for the building; (3) the height above ground level at which the release originates; (4) wind speed at the height of release; (5) the ratio of the wind speed at the height of release to that at ground level; (6) atmospheric stability parameters which determine the plume size and shape; (7) wind variability factors for the specific atmospheric conditions and release times, ⁽³⁸⁾ and (8) ground deposition coefficients for each of the categories of fission products.

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REFERENCES

1. A. T. Gifford to A. B. Greninger. Unpublished Data, Richland Operations Office, March 18, 1965. (Personal Communication - "Reactor Safety - Code of Federal Regulations," OR:JPT).
2. Title 10, Code of Federal Regulations, Part 100, Reactor Site Criteria.
3. Federal Register, Power and Test Reactors, Notice of Proposed Rule Making, 26(28), pp. 1224-1226. February 11, 1961.
4. M. C. Leverett and R. E. Trumble Richland Integrated Process Development Program, Nuclear Safety Program, RL-GEN-27, Sup. 13, Revision 1. July 30, 1965.
5. F. Neumann. Final Report on Seismological Investigations at the Hanford Works, GEH-24659. November 10, 1959.
6. W. J. Tupper. Budget Study Report FY-1963, Plant Acquisition and Construction Budget, HW-68347. January 30, 1961.
7. General Electric Company, IPD. Hazards Summary Report, Safety Analysis and Hazards Evaluation, Six Oldest Hanford Production Reactors, HW-74094, Vol. I. June 21, 1963.
8. General Electric Company, IPD. Hazards Summary Report, Safety Analysis and Hazards Evaluation, Hanford K Production Reactors, HW-74095, Vol. I. June 28, 1963.
9. AEC Manual - RL Chapter 06R1. Procedure for Plant-Engendered Emergencies. December 9, 1963.
10. R. E. Tiller and C. A. Oster. CLUMSY I - A Time and Space Dependent Reactor Kinetics Code for the IBM-7090 Computer, HW-77221. August 1, 1963.
11. W. P. Ekern, J. Muraoka, and F. R. Zaloudek. An Analysis of the Time Sequence of Events Following Complete Loss of Coolant to a Hanford Reactor, HW-62494. November 11, 1959.
12. K. T. Scott and J. H. Buddery. "The Release of Krypton-85 From Irradiated Uranium on Thermal Cycling Through the Phase Changes," J. of Nucl. Mater. 5, pp. 94-100. November 1, 1962.
13. J. H. Buddery and K. T. Scott. "A Study of Melting of Irradiated Uranium," J. Nucl. Mater., vol. 5, pp. 81-93. November 1, 1962.

DECLASSIFIED

14. K. R. Heid, editor. Unpublished Data, Pacific Northwest Laboratory, Richland, BNWL-MA-11.
15. J. K. Soldet. Columbia River Flow Time Calculations, HW-58312 REV. July 1, 1962.
16. L. R. Wallis. Emergency Procedure Manual, HW-74702 REV 1. November 9, 1964.
17. D. Wisehart. Field Method Using Uranine to Measure Penetration of High Efficiency Filter, HW-79303. October, 1963.
18. Arthur D. Little, Inc. Halogen Collector Test Program, HW-65587. March, 1960.
19. L. Silverman and R. E. Yoder. "Foam Suppression of Radioactive Iodine and Particulates," Eighth Annual Air Cleaning Conference, TID-7677, p. 297. Division of Technical Information Extension. October, 1963.
20. Final Report - Earthquake Resistance of the Existing Hanford Production Reactors, HN-146. Holmes & Narver, Inc. June, 1960.
21. A. Amirikian. "Cylindrical Shell Segments," Modern Welded Structures, Vol. 1, James F. Lincoln Arc Welding Foundation. May, 1963.
22. F. Neumann. Seismological Investigations at the Hanford Area, HW-63832. November 2, 1959.
23. R. E. Brown and J. R. Raymond. Geophysical Seismic Evaluation Study at Hanford, BNWL-47. Pacific Northwest Laboratory. December, 1964.
24. R. T. Jaske. Large Scale Quarry Blasting on the Hanford Reservation, HW-79614. January 15, 1964.
25. Y. A. Zysin, A. A. Lbov, and L. I. Sel'chenkov. Fission Product Yields and Their Mass Distribution, Consultants Bureau, New York, N.Y. 1964.
26. S. Catcoff. Nucleonics, vol. 18, no. 11, p. 20. 1960.
27. H. H. Landolt and R. Bornstein. Energy Levels of Nuclei, A-5 to A-257, Springer Verlag, Berlin. 1961.
28. D. T. Goldman. Chart of the Nuclides, Seventh Addition. June, 1964.
29. D. C. Pound and G. J. Busselman. Process Tube Boilout Experiments, HW-52340. August, 1957.

DECLASSIFIED

30. D. E. Bush. Unpublished Data, General Electric Company, Richland.
31. R. Nilson and R. H. Meichle. Fractional Heat Generation in Hanford Reactors After Shutdown, HW-69628. May 17, 1961.
32. M. H. Higgins and R. D. Schultz. The Reaction of Metals with Water and Oxidizing Gases at High Temperatures, IDO-28000. Idaho Operations Office. April 30, 1959.
33. G. E. Zima. Pyrophoricity of Uranium in Reactor Environments, HW-62442 RD. October 26, 1959.
34. I&E Fuel Melt Experiment, HW-69058. (Color movie film)
35. E. C. Watson and G. C. Gamertsfelder. Environmental Radioactive Contamination as a Factor in Nuclear Plant Siting Criteria, HW-SA-2809. March 29, 1963.
36. E. C. Watson, R. E. Trumble, and R. L. Junkins. Estimated Distances for Exclusion Area and Low Population Zone Boundaries: N Reactor, HW-80643. February 3, 1964.
37. E. C. Watson and N. G. Wittenbrock. Environmental Exposure as a Function at Containment Vessel Leakage Rate, BNSA-9. Pacific Northwest Laboratory. March 25, 1965.
38. J. J. Fuquay and C. L. Simpson. Development of an Atmospheric Transport Model, BNSA-123. Pacific Northwest Laboratory. March 5, 1965.

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