OCT 21 1963

325

ORNL-3361
UC-80 - Reactor Technology
TID-4500 (22nd ed.)

MASTER

ACTIVITY RELEASE FROM THE N.S. SAVANNAH
IN THE MAXIMUM CREDIBLE ACCIDENT

T. D. Anderson
J. R. Buchanan
W. B. Cottrell
M. H. Fontana
O. H. Klepper
H. C. McCurdy

OAK RIDGE NATIONAL LABORATORY

operated by

UNION CARBIDE CORPORATION

for the

U.S. ATOMIC ENERGY COMMISSION

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

Printed in USA. Price: \$2.50 Available from the
Office of Technical Services
U. S. Department of Commerce
Washington 25, D. C.

LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

- A. Makes any warranty or representation, expressed or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or
- B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

Contract No. W-7405-eng-26

Reactor Division

ACTIVITY RELEASE FROM THE N.S. SAVANNAH IN THE
MAXIMUM CREDIBLE ACCIDENT

T. D. Anderson	M. H. Fontana
J. R. Buchanan	O. H. Klepper
W. B. Cottrell	H. C. McCurdy

Date Issued

OCT 16 1963

OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee
operated by
UNION CARBIDE CORPORATION
for the
U.S. ATOMIC ENERGY COMMISSION

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

CONTENTS

	<u>Page</u>
FOREWORD	vii
ABSTRACT	ix
1. INTRODUCTION	1
2. POWER REQUIREMENTS IN HARBOR AREAS	4
Power Requirements in New York and Seattle	5
Power Requirements for Generalized Cases	5
3. EFFECT OF OPERATING HISTORY ON TIME FOR FUEL- ELEMENT FAILURE	7
Analytical Method for Estimating Failure Times	7
Relative Failure Times for Various Cases	9
Time Dependence of Failure	9
4. FISSION-PRODUCT DECAY	12
Calculation of Fission-Product Inventory for Various Cases	12
Results of Fission-Product Inventory Calculations	14
5. RELEASE FROM CONTAINMENT SYSTEMS	19
Effects of System Holdup	20
Combined Effects of Decay and Holdup	26
6. EFFECT OF RELEASE MECHANISMS ON EXPOSURE	26
Assumptions Employed in Exposure Calculations	32
Calculational Modes	32
Fission-Product Release to Containment System	33
Filter Efficiency	33
Atmospheric Dilution Near the Ship	33
Ventilation System Dilution and Source Elevation	34
Calculational Method	34
Atmospheric Dispersion	34
Submersion Exposure	34
Thyroid Exposure	36
Results of Exposure Calculations	37

7. DISCUSSION	44
Implications of Exposure Calculations	44
Integrated Submersion Exposure	47
Integrated Thyroid Exposure	48
Exclusion Distances	49
Justification of Mechanisms Evaluated	49
Power History	49
Core Failure Sequence	50
Fission-Product Decay	52
Holdup Effect of Reactor Compartment	53
Other Pertinent Factors	54
Ability to Remove Ship from Dock After Accident in the Time Required	54
Performance of the Filters and Compartment Ventilation Systems	55
Implications of Total Population Exposure	56
Implications of Limited Time in Port	57
Proximity and Resulting Exposure of Passengers and Visitors Abroad Ship	57
Effect of Radioactive Cloud Size on Exposure	58
8. CONCLUSIONS	58
APPENDIX A. DIFFERENCES BETWEEN N.S. SAVANNAH AND STATIONARY POWER PLANTS	65
The "Population Center Distance"	65
Characteristics of the Area Surrounding the Ship Berth	66
Proximity of Some Members of the Public to the Reactor	66
Effect of Operating Power History	67
Removal of Ship to a Remote Anchorage	67
Release to the Hydrosphere	68
Accidents as a Consequence of Plant Mobility	68
Electrical Power	69
Evacuation of the Ship at Sea	69
Time in Port	70
Ship Compartmentalization	70
Collision Protection	71

APPENDIX B. FISSION-GAS RELEASE DURING OPERATION	72
Discussion of the Conductivity Integral	73
Determination of the Pellet Surface Temperature	74
Use of the Conductivity Integral	76
Calculation of Fission-Gas Release	78
Pressure Buildup and Cladding Stress in Fuel Rod	81
Fuel Element Failure Criterion	87
APPENDIX C. COMPARISON OF ACTUAL AND SYNTHESIZED DECAY CHAINS	89
APPENDIX D. BUILDUP OF ACTIVITY IN REACTOR COMPARTMENT	92
APPENDIX E. POPULATION EXPOSURE AROUND NEW YORK HARBOR	94
APPENDIX F. EFFECT OF CLOUD SIZE ON SUBMERSION EXPOSURE	104
REFERENCES	106

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

FOREWORD

During the late spring and summer of 1962 an intensive study was made of the hazards associated with the operation of the N.S. SAVANNAH in harbor areas. Based on the findings of this study by government and contractor personnel, tentative criteria were developed for port operation that received approval of the regulatory authorities for the interim operating period of the N.S. SAVANNAH. The pertinent calculations of the ORNL personnel who participated in the study are presented in this document.

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

ABSTRACT

The release of fission products that would occur following the maximum credible accident aboard the N.S. SAVANNAH has been examined. Four significantly different, but realistic, operating histories were considered. The rate of release of noble gases and of iodine isotopes as a function of time after the accident was determined for each operating history and for both normal and emergency reactor-compartment ventilation systems. The influence of radioactive decay and of the time delay in release and transport of activity through the containment system was investigated. Most of the results are expressed in terms of activity release and resultant individual exposures, although some consideration is given to population exposures and to the interpretation of these results in the light of stationary reactor site criteria.

ACTIVITY RELEASE FROM THE N.S. SAVANNAH IN THE
MAXIMUM CREDIBLE ACCIDENT

T. D. Anderson	M. H. Fontana
J. R. Buchanan	O. H. Klepper
W. B. Cottrell	H. C. McCurdy

1. INTRODUCTION

The activity released in the maximum credible accident of any nuclear installation is the starting point for site evaluation. In the case of stationary power reactors the common siting practices, as they have evolved under the supervision of the Atomic Energy Commission over the last ten years, were recently formalized in a criteria guide, which was issued as part of the Code of Federal Regulations.¹ These regulations specifically excepted nuclear ships. Separate interim criteria² were developed by N.S. SAVANNAH project personnel to provide rational bases for port and berth evaluations during the initial operating period.

Of the many differences between stationary reactors and ship reactors (see Appendix A), the point of greatest concern in port evaluations is the proximity of the large number of people usually found around major port areas. A comparable condition is precluded in the siting of stationary power reactors by the somewhat arbitrary city distance criterion. Although port criteria cannot be identical with stationary reactor site criteria, it is essential that the two provide comparable safeguards for the general public, and it is necessary that the required exclusion areas around ships be compatible with areas available at major port berths. In order to permit the assessment of port site requirements the activity release from the N.S. SAVANNAH in the maximum credible accident has been evaluated in considerable detail.

The N.S. SAVANNAH power plant* is a pressurized-water reactor system designed for a continuous power output of 69 Mw (thermal). The fuel is slightly enriched uranium (approximately 4.4% U²³⁵) in the form of sintered uranium dioxide pellets encapsulated in 1/2-in.-OD stainless steel

*See ref. 3 for detailed description of the N.S. SAVANNAH reactor and propulsion system.

tubes with a fueled length of 66 in. The core contains a total of 5248 of these fuel elements in 32 fuel element assemblies, which are arranged in a square array, with the corner assemblies omitted, to form roughly a 5-ft cube. Heat removal from the core is provided by two primary coolant loops having two pumps and one steam generator in each loop. The system normally operates at a pressure of 1750 psia and an average coolant temperature of 508°F.

The entire primary system is contained in a shielded steel containment vessel that is designed to withstand the pressure and temperature excursion of the maximum credible accident. Piping lines that penetrate the containment vessel are provided with isolation valves which are closed upon an increase in containment pressure. The maximum permissible leakage rate of the containment vessel has been specified as 1.5% of the contained gases per day,² and periodic leak tests are conducted to assure that the specified leakage rate is not exceeded. The containment vessel is located within a closed compartment of the ship that is designated as the "reactor compartment." This compartment meets normal watertightness requirements of regulatory agencies, but it is not designed to the leakage standards usually associated with containment vessels and is not periodically leak tested. The compartment is maintained at negative pressure, however, by either of two independent ventilation systems, and the exhaust air is passed through high-efficiency particulate matter and iodine removal filters before being released to the environment. Thus, the reactor is provided with two containment systems in series that serve both to delay and to reduce the release of activity following an accident.

The maximum credible accident of the N.S. SAVANNAH reactor has been established³ as a large rupture of the primary coolant system. In previous analyses^{4,5} of the consequences of this accident, it was assumed that (1) the accident occurred after long-term operation of the reactor at maximum power, (2) all fuel elements would fail at the onset of the accident and all releasable activity would be immediately dispersed throughout the containment vessel, (3) the released activity would leak from the containment vessel at the specified maximum leakage rate and would pass to the environment with no further time delay, and (4) the activities would not be depleted by radioactive decay. Similar assumptions are

commonly employed in analyses of stationary power plants where a lesser degree of conservatism is either difficult to justify or there is little incentive for doing so.

In the case of the N.S. SAVANNAH, however, there is incentive for a more realistic treatment of release mechanisms, particularly those which influence the rate of activity release during the period immediately following the accident. The time-dependence of the activity release is an important consideration in assessing the inherent mobility of the ship as a potential means for reducing the consequences of an accident after it occurs.

Other studies² have indicated that with proper planning and preparation, it would be physically feasible to evacuate unnecessary personnel from the ship and to move the ship from a dock position to an isolated anchorage within a period of 24 hr and, probably, within 2 hr. The actual feasibility of carrying out such a move would, of course, be dependent on assurance that radiation exposure to tug crews, essential ship personnel, and the general public could be held to acceptable levels. The anchorage to which the stricken vessel would be moved would be determined for each port as a part of the port evaluation studies.

The purpose of the present work is to investigate the effects of power history, time delay in release and transport of activity through the containment system, and radioactive decay of fission products on the rate at which activity would be released from the N.S. SAVANNAH in the maximum credible accident. The rate of release of noble gases and of iodine isotopes as a function of time after the accident were estimated for four cases of power history and for alternate reactor compartment ventilation systems* and compared with a "reference release rate" estimated by the previous simplified treatment. Also, to illustrate the effect of

*Note that the reactor compartment ventilation systems considered in this report are those which existed on the N.S. SAVANNAH during its initial port visits. In early 1963, following an interim period of sea and port operations, the reactor compartment ventilation systems were extensively modified. The modified arrangement is believed to provide increased reliability and simplicity of operation. Particulate and iodine filtration equipment remained unchanged.

the more detailed treatment of factors influencing release, exposures were estimated for several of the cases of interest and compared with exposures for the reference case.

2. POWER REQUIREMENTS IN HARBOR AREAS

The methodology of evaluating hazards arising from nuclear reactor accidents has been developed using stationary power plants as a basis. Stationary water-cooled reactors have almost universally been associated with an accident in which all releasable fission products are released instantaneously to the containment vessel and start leaking at the maximum rate to the atmosphere. The reasons for this treatment of activity release are associated with the type of reactor, mode of operation, and type of accident considered. In this context, the maximum credible accident for stationary, water-cooled, power reactors is usually a loss-of-coolant accident preceded by a long history of operation at 100% power. A short time after the loss of all cooling, a high-power-density reactor core will fail, and most of the releasable fission products will be dispersed throughout the containment system.

Although the N.S. SAVANNAH reactor design is similar to that of some stationary reactors, there are significant differences between ship- and land-based plants that should be considered in evaluating the consequences of an accident. As pointed out in Appendix A, perhaps the biggest single difference lies in the fact that a ship is inherently mobile, and therefore in the event of an accident the radioactive source could be moved out of a heavily populated area within a few hours. Furthermore, ship operation while in populated port areas is characterized by reduced power. This normal reduction in power is necessary because of the marine traffic, restricted area for maneuvering, and shallow water. A history of reduced power operation has two effects on the release of activity. First, the total amount of the fission products is decreased, and, second, the rate of core failure is slowed because of the decreased decay heat. This decrease and delay in fission-product release allows fuller exploitation of ship mobility to safeguard against the release of dangerous quantities of radioactive material near the populated areas. It is important,

therefore, to consider in more detail the power requirements of harbor operation.

Power Requirements in New York and Seattle

Upon approaching a port area the speed and therefore power of ocean-going vessels are reduced, as mentioned above. The operating pattern while in the port area usually precludes sustained output above 50% of rated power. After docking, the power is further reduced to that required for the hotel load and is maintained at this level for the duration of the period at dockside. In the case of the N.S. SAVANNAH, the hotel load is 15% of full power.

As examples of in-port operation, the approximate power requirements⁶ for the harbors of New York and Seattle are shown in Figs. 1 and 2. Although each harbor has its own peculiar characteristics, the operating requirements given in Figs. 1 and 2 are probably typical.

Power Requirements for Generalized Cases

The following cases of operating history are defined for purposes of further evaluation in subsequent sections of this report:

Case 1

Operating sequence prior to maximum credible accident	100% power for infinite time
Situation	Occurrence of accident preceded by full-power operation for a long time; this is the power history on which previous hazards evaluations were based

Case 2

Operating sequence	100% power for infinite time; 50% power for 5 hr
Situation	This might represent a typical operating mode for port entry

Case 3

Operating sequence	100% power for infinite time; zero power for 5 hr
Situation	Port entry under other than reactor power

Fig. 1. Reactor Operation While Entering New York.

Fig. 2. Reactor Operation While Entering Seattle.

Case 4

Operating sequence	100% power for infinite time; 50% power for 5 hr; 15% power for 12 hr
Situation	Typical port entry followed by base-load operation for 12 hr at dockside

Case 5

Operating sequence	100% power for infinite time; 50% power for 5 hr; zero power for 12 hr
Situation	Typical port entry followed by restricted base-load operation

3. EFFECT OF OPERATING HISTORY ON TIME FOR FUEL-ELEMENT FAILURE

The time available between the occurrence of the maximum credible accident and fuel-element failure depends on the reactor power at the time of the accident and the reactor operating history. Previous failure analyses were made by the Babcock & Wilcox Company for a number of reactor conditions,⁷ but a sequence of reactor operations that might be representative of in-port conditions was not explicitly covered. The case of essentially infinite-time operation at 100% power preceding the accident was considered for most of the previous hazards studies. As indicated in Section 2, reactor operation while entering and within a port is normally restricted to less than full-power. It was of interest therefore to determine the influence of reduced-power operation on the time available between the accident and fuel-element failure.

Analytical Method for Estimating Failure Times

It must, of course, be recognized that the conditions prior to and after an accident such as the maximum credible accident are not well defined. It is therefore necessary to make a number of assumptions that may have little connection with reality but which, hopefully, will set an upper limit on the consequences of the accident. In order to go beyond this "upper limit" analysis and to consider the effects of certain

detailed changes, as in the present analysis, it is important to realize that the relative results are much more significant than absolute results. With this in mind, the approach taken in this analysis was to estimate the time for cladding failure for a given case in terms of the time for the same failure for a reference case (case 1) and to apply the relative results to the information given previously by Babcock & Wilcox for case 1. In this way the time to failure for each case was estimated on a basis consistent with the previous analysis.

First, the cladding temperature, T_1 , for case 1 was expressed as some function $G_1(t)$ of time after the mca:

$$T_1 = G_1(t) .$$

For another case, say case i, the cladding temperature is given by

$$T_i = G_i(t) .$$

If the criterion for cladding failure is some predetermined temperature, T_F , and the times for cladding failure are designated t_1 and t_i for case 1 and case i, respectively, these failure times can be determined from the equations

$$G_1(t_1) = T_F$$

and

$$G_i(t_i) = T_F .$$

The relationship between t_1 and t_i is then given by

$$G_1(t_1) = G_i(t_i) . \quad (1)$$

In obtaining the G-functions, two major assumptions were made: (1) immediately following the accident the outer surface of the cladding is thermally insulated, and (2) the cladding and UO_2 temperatures are the same at all times following the accident.

In view of the desire to maintain consistency with the Babcock & Wilcox analysis, the first assumption is questionable when it is considered that the Babcock & Wilcox analysis did not assume an insulated fuel element until 30 sec after the accident. Nevertheless, in their analysis the heat transfer coefficient decreased quite rapidly in the

first 2 sec. Thus, assumption 1 is probably a fair approximation. To further substantiate the first assumption, the method developed for this analysis was used to predict the time to a given temperature for three cases previously analyzed by Babcock & Wilcox. The maximum difference in predicted time was less than 15% for all test cases. The second assumption is of little consequence except for short times, since temperature equilibrium in the insulated element is established in less than 2 min.

Relative Failure Times for Various Cases

Equation 1 was used to determine the time to fuel-element failure for a particular power history in terms of the failure time for case 1. The results of this analysis are shown in Fig. 3. It should be recalled that the criterion for cladding failure is some predetermined temperature. However, up to this point, it has not been necessary to specify the failure temperature. Thus Fig. 3 is valid for any fuel-element failure that is uniquely determined by temperature.

Time Dependence of Failure

As indicated in Appendix B, it appears that fuel-element-cladding failure will not occur until the cladding temperature is very near the melting point. Once the cladding of a fuel element reaches the melting point, approximately 2% of the gaseous fission products will be released instantaneously. The other 98% will be retained in the UO_2 fuel bodies. If the UO_2 continues to increase in temperature, a sizeable fraction of the volatile fission products retained in the UO_2 will be released upon attaining grain-growth temperature ($\sim 3200^\circ\text{F}$). If the fuel temperature continues to increase, some diffusion of the remaining fission products will occur until, finally, when the UO_2 temperature reaches the melting point, almost all the volatile material will have been released from the fuel.

It can be seen from the above discussion that a number of mechanisms are involved in the release of activity from the core, and a proper

Fig. 3. Effect of Operating History on Time for Fuel-Element Failure After MCA.

analysis of the time dependence of release would take these mechanisms into account. The uncertainties involved in such a detailed analysis are great, however, and therefore for the present work the only mechanism considered was cladding failure by melting. It was assumed that once any part of the cladding of a fuel element reached the melting temperature, all fission products that would ultimately be released were released.

Although not realistic, this assumed model for fission-product release gives conservative results from the standpoint of the hazards analysis.

According to this model the fraction of a given fission product released after an elapsed time, t , following the accident is given by

$$F(t) = f E(t) ,$$

where $F(t)$ is the amount of a given fission product released from the core as a gas up to time, t , expressed as a fraction of the total quantity of the fission product in existence at time, t , $E(t)$ is the fraction of the fissioned fuel exposed by cladding failure up to time t , and f is the fraction of a fission product associated with the exposed fuel that is released as a gas. It should be noted that $F(t)$ may be different for each fission product because f is characteristic of the particular fission product being considered. On the other hand, the quantity $E(t)$ is common to all fission products.

For case 1, the quantity $E_1(t)$ was calculated by Babcock & Wilcox.⁷ By using their results for $E_1(t)$, the information given in Fig. 3 could be applied to obtain fuel-element failure curves for any of the power history cases considered; this was done, and the results are shown in Fig. 4.

Fig. 4. Fuel Exposed by Cladding Melting.

4. FISSION-PRODUCT DECAY

The power requirements for the cases of interest (cases 1, 2, 4, and 5)* were defined in Section 2, and the fuel-element failure times for these cases were evaluated in Section 3. The amounts of noble gas and iodine available during the accident are also of interest because the exposure to one or the other of these fission products would be controlling following the accident. Whether it would be the iodine or the noble gases that would be controlling would depend largely on the iodine removal efficiency of the filter in the compartment ventilation system.

Calculation of Fission-Product Inventory for Various Cases

The source of fission products as a function of irradiation time, decay time, and neutron flux was calculated by solving the following differential equations:

$$\frac{dN_i}{dt} = \Sigma_f \phi Y_i^{\text{ind}} + kN_{i-1} \lambda_{i-1} - N_i \lambda_i - N_i \phi \sigma_i \quad (2)$$

and

$$\frac{dN_{i-1}}{dt} = \Sigma_f \phi Y_{i-1}^{\text{cum}} - \lambda_{i-1} N_{i-1} - \sigma_{i-1} \phi N_{i-1} \quad (3)$$

When $t = 0$, $N_i = N_i^0$, and $N_{i-1} = N_{i-1}^0$, solving for N_i yields:

$$N_i = N_i^0 e^{\lambda_{\text{eff}}^i t} + \Sigma_f \phi \left(\frac{Y_i^{\text{ind}}}{\lambda_{\text{eff}}^i} + \frac{kY_{i-1}^{\text{cum}} \lambda_{i-1}}{\lambda_{\text{eff}}^{i-1} \lambda_{\text{eff}}^i} \right) \left(1 - e^{-\lambda_{\text{eff}}^i t} \right) + \frac{\lambda_{i-1}}{\lambda_{\text{eff}}^i - \lambda_{\text{eff}}^{i-1}} \left(kN_{i-1}^0 - \frac{\Sigma_f \phi kY_{i-1}^{\text{cum}}}{\lambda_{\text{eff}}^{i-1}} \right) \left(e^{-\lambda_{\text{eff}}^{i-1} t} - e^{-\lambda_{\text{eff}}^i t} \right), \quad (4)$$

*Case 3 was not considered because of the similarity of the core failure curve in that case to that of case 4 and the lack of interest in the operational mode that it implied.

where

- N_i = number of atoms of nuclide i per unit volume of core (cm^{-3}),
- N_i^0 = initial value of N_i (cm^{-3}),
- $\lambda_{\text{eff}}^i = \lambda_i + \sigma_i \phi$ = effective removal constant (sec^{-1}),
- λ = decay constant (sec^{-1}),
- σ_i = neutron absorption cross section of nuclide i (cm^2),
- t = time (sec),
- Σ_f = macroscopic fission cross section of fissionable nuclide (cm^{-1}),
- ϕ = neutron flux (neutrons/ $\text{cm}^2 \cdot \text{sec}$),
- Y_i^{ind} = independent fission yield of nuclide i (number of atoms of nuclide i per fission),
- k = branching ratio, that is, fraction of nuclide $i-1$ decays that yield nuclide i ,
- Y_{i-1}^{cum} = cumulative fission yield of parent isotope.

Equation (4) can be used for decay calculations by setting the neutron flux equal to zero. It is to be noted that Eq. (4) is strictly applicable for a nuclide with one precursor and removal by both radioactive decay and neutron capture. No provision was made for buildup of nuclide N_i or N_{i-1} by neutron capture of the nuclide of the same atomic number in the chain of next lower atomic mass. This latter restriction is not important for the nuclides of interest here. For the time periods considered the restriction of one precursor introduces only a small error, because, in general, the N_{i-2} nuclides have very short half-lives compared with the half-lives of the nuclides of interest. Therefore the cumulative yield is used for the N_{i-1} nuclide, assuming that it is instantly generated by fission.

Equation 4 was solved by means of the ORSFT compiler for the IBM-7090 computer. The program was written so that after a time period at a given power the calculated values of nuclides N_i and N_{i-1} were stored so that these values could be used as the initial values, N_i^0 and N_{i-1}^0 , in the following time period of the same case.

The calculated values of N_i and N_{i-1} for the various time periods were converted to the total core inventory (in curies) by the relation:

$$Q_i = \frac{\lambda_i N_i}{3.7 \times 10^{10}} V, \quad (5)$$

where Q_i is the total core inventory of nuclide i (curies) and V is the core volume (cm^3). The fission products studied were the important gamma-emitting noble gases and the important iodine isotopes. Some approximations were made to fit the decay-chain structure to the requirements of Eq. (4). The true chains and the synthesized decay chains that were used in the calculations are shown in Appendix C.

Results of Fission-Product Inventory Calculations

The core inventory in curies of the various isotopes at the time of the accident was calculated as indicated. It is shown in Table 1 for cases 1, 2, 4, and 5. The noble gas energy release rates are shown in Fig. 5 for the various cases. In each case, the number of curies of each isotope was multiplied by the average gamma energy of that isotope and the resulting energy release rates of the seven rare-gas nuclides were added. This procedure was repeated for various times after shutdown, and

Table 1. Core Krypton, Xenon, and Iodine Inventory
At Time of Accident

Nuclide	Half-Life	Average Gamma Energy (Mev)	Inventory at Time of Accident (curies)			
			Case 1	Case 2	Case 4	Case 5
Kr ^{85m}	4.36h	0.181	8.67×10^5	6.31×10^5	2.04×10^5	9.30×10^4
Kr ⁸⁷	78m	0.56	1.53×10^6	8.19×10^5	2.30×10^5	1.38×10^3
Kr ⁸⁸	2.77h	2.07	2.13×10^6	1.38×10^6	3.72×10^5	6.84×10^4
Xe ^{133m}	2.3d	0.233	9.02×10^4	7.00×10^4	8.73×10^4	8.69×10^4
Xe ¹³³	5.27d	0.081	3.67×10^6	3.66×10^6	3.61×10^6	3.61×10^6
Xe ¹³⁵	15.6m	0.52	1.02×10^6	8.28×10^5	3.51×10^5	2.43×10^5
Xe ¹³⁵	9.13h	0.268	1.75×10^6	1.88×10^6	1.60×10^6	1.60×10^6
I ¹³¹	8.05d	(a)	1.67×10^6	1.65×10^6	1.60×10^6	1.58×10^6
I ¹³²	2.4d	(a)	2.54×10^6	2.52×10^6	2.33×10^6	2.30×10^6
I ¹³³	20.8h	(a)	3.98×10^6	3.74×10^6	2.74×10^6	2.56×10^6
I ¹³⁵	6.68h	(a)	3.41×10^6	2.72×10^6	1.14×10^6	7.80×10^5

^aThyroid dose is the important consideration.

Fig. 5. Noble Gas Gamma Energy Release Rate Relative to Initial Rate for Case 1.

the results were expressed as fractions of the value obtained for case 1 at the time of the accident. The results were gamma-energy weighted functions that were directly useful in submersion exposure calculations:

$$\text{Relative release rate} = \frac{\sum_{i=1}^n E_{\gamma}^i Q_i(t) \text{ (case } n\text{)}}{\sum_{i=1}^n E_{\gamma}^i Q_i(0) \text{ (case 1)}} , \quad (6)$$

where

E_{γ}^i = the average gamma photon energy of nuclide i (Mev),

n = number of nuclides,

$Q_i(t)$ = curies of nuclide i at time t after the accident,

$Q_i(0)$ = curies of nuclide i at the time of the accident.

The initial values of the gamma-energy-weighted release rate are given in Table 2.

Table 2. Initial Noble Gas Energy Release Rates for the Various Cases

Case	Initial Energy Release Rate (Mev·curies)	Initial Noble Gas Release Rate as Fraction of Case 1 Rate [Eq. (6)]
1	6.74×10^6	1.000
2	4.68×10^6	0.694
4	1.859×10^6	0.276
5	1.026×10^6	0.152

The variations of iodine activity as a function of time after the accident are shown in Figs. 6 through 9. Table 3 gives the relative initial inventory of the iodine isotopes for the various cases.

Table 3. Initial Value of Iodine Isotopes for Various Cases as a Function of Initial Value for Case 1

Nuclide	Half-Life	Initial Iodine Release Rate as Fraction of Case 1 Rate		
		Case 2	Case 4	Case 5
I^{131}	8.05d	0.982	0.958	0.946
I^{132}	2.4h	0.992	0.917	0.906
I^{133}	20.8h	0.940	0.688	0.643
I^{135}	6.68h	0.798	0.334	0.228

Fig. 6. Radioiodine Inventory in N.S. SAVANNAH Core - Case 1.

Fig. 7. Radioiodine Inventory in N.S. SAVANNAH Core - Case 2.

Fig. 8. Radioiodine Inventory in N.S. SAVANNAH Core - Case 4.

Fig. 9. Radioiodine Inventory in N.S. SAVANNAH Core - Case 5.

The data presented in this analysis provided the "source term" for the activity available in the core. This term was modified by the fuel-element rupture rate, the containment and filter holdup effects, and meteorological dispersion factors to arrive at dose rates external to the ship.

5. RELEASE FROM CONTAINMENT SYSTEMS

The calculated rate of release of activity from the containment system establishes the exposure potential (which has a direct bearing on the siting requirements) of a nuclear reactor. Previous siting evaluation of the N.S. SAVANNAH made use of many simplified and conservative assumptions concerning the release of fission products to the atmosphere in the event of the maximum credible accident. In view of the fact that application of these greatly simplified assumptions would in some instances indicate exposures that are not compatible with the berthing requirements of a nuclear ship, the present analysis presents a more detailed evaluation of activity release from the N.S. SAVANNAH containment system.

The following conservative assumptions concerning the release of fission products from the reactor system to the atmosphere had previously been employed.⁴

1. The reactor would operate for an essentially infinite time at full power up to the occurrence of the accident (power history case 1 as defined in sec. 2).
2. The core would fail immediately following the maximum credible accident and release all volatile fission products to the containment vessel.
3. Immediately following the accident the containment system would start to leak activity to the atmosphere at a rate determined by the tightness of the containment vessel.
4. Depletion of the fission products by decay would not be significant.

The leak rate of fission products to the atmosphere as calculated on the basis of the above assumptions will hereafter be designated as

the "reference release rate"; it is given by

$$\text{Reference release rate} = l f N_0 ,$$

where l is the leak-rate constant for the containment vessel (fraction of enclosed material released per unit time), f is the fraction of a given radioactive material that is released to the containment vessel as a gas, and N_0 is the total quantity of a given fission product in the core after operation at full power for a long time. In contrast to this simplified calculation, the present analysis accounts for holdup effects provided by the core and reactor compartment, decay of the fission products, and actual cases of power history, as presented in Section 2.

Effects of System Holdup

A schematic diagram of the three-barrier containment model is shown in Fig. 10. As may be seen, an isotope may enter an enclosure by leakage from a preceding enclosure and by decay of a precursor isotope. An isotope may leave an enclosure by leakage and may decay to a daughter isotope. The rate processes involved in the analysis are illustrated in Fig. 11. The symbols used in Fig. 11 are:

$R(t)$ = rate of escape of given isotope from reactor core,

$F(t)$ = fraction of total quantity of isotope that has escaped from the core,

$Q(t)$ = quantity of isotope held up in first containment barrier,

$U(t)$ = quantity of isotope held up in second containment barrier,

$Q_p(t)$ = quantity of precursor isotope held up in first containment barrier,

$U_p(t)$ = quantity of precursor isotope held up in second containment barrier,

l = leak-rate constant for first containment barrier,

m = leak-rate constant for second containment barrier,

λ = decay constant for isotope being considered,

λ_p = decay constant for precursor isotope,

$N(t)$ = total quantity of isotope existing at time t ,

$N_p(t)$ = total quantity of precursor isotope existing at time t .

Fig. 10. Schematic Diagram of Three-Barrier Containment System.

The general problem of a three-barrier system was considered in ref. 8. As shown in ref. 8, if all precursor isotopes move through the system in the same manner as the isotope of interest or if there are no precursors, it is possible to separate the problems of mobility and decay. For this special case the leak rate to the atmosphere is given by

Fig. 11. Schematic Diagram of Containment Rate Processes.

$$m U(t) = \frac{ml N(t)}{(m-l)} \left[m e^{-mt} \int_0^t e^{m\tau} F(\tau) d\tau - l e^{-lt} \int_0^t e^{l\tau} F(\tau) d\tau \right], \quad (7)$$

where τ is the variable of integration.

The case of a reactor system with a single containment vessel, which, upon initiation of an accident, releases some fraction, f , of a given fission product from the core was considered. It was assumed that leakage from the containment would be small enough so that little depletion would occur because of the leakage. For this simple case the leakage to the atmosphere is given by $lf N(t)$, and the leak rate given by Eq. (7) expressed as a fraction of the simple, single-containment case* is

*It should be noted that a distinction is made between the "single-containment" case and the "reference" case that was defined above. The reference release rate is based on case 1 power history and no decay, whereas the single-containment release rate is not limited to a particular power history, and decay is taken into account.

$$\frac{m U(t)}{lf N(t)} = \frac{m}{f(m-l)} \left[m e^{-mt} \int_0^t e^{m\tau} F(\tau) d\tau - l e^{-lt} \int_0^t e^{l\tau} F(\tau) d\tau \right]. \quad (8)$$

In order to calculate the relative release rate from Eq. (8), it is necessary to evaluate $F(t)$. As indicated in Section 3, $F(t)$ is given by

$$F(t) = f E(t),$$

where, as before, f is the fraction of a fission product associated with exposed fuel that is released from the core as a gas and $E(t)$ is the fraction of the fissioned fuel exposed.

Figure 4 gives $E(t)$ for five different power histories, and the general shape of the E -curves suggests an empirical fit of the form

$$E(t) = 0 \quad t \leq t_0$$

$$E(t) = 1 - e^{-\mu(t-t_0)} \quad t > t_0, \quad (9)$$

where t_0 is the time elapsed from start of accident to initial fuel-element failure and μ is the core release-rate parameter.

If the failure curves are approximated by Eq. (9),

$$F(t) = 0 \quad t \leq t_0$$

$$F(t) = \left[1 - e^{-\mu(t-t_0)} \right] f \quad t > t_0. \quad (10)$$

By using Eq. (10) for $F(t)$, the relative leak rate given by Eq. (8) becomes:

$$L(\theta) = \frac{\kappa}{\kappa - 1} \left[\frac{\epsilon}{\kappa - \epsilon} e^{-\kappa\theta} - \frac{\epsilon}{1 - \epsilon} e^{-\theta} + \left(\frac{1}{1 - \epsilon} - \frac{\kappa}{\kappa - \epsilon} \right) e^{-\epsilon\theta} \right], \quad (11)$$

where

$$L(\theta) = \frac{m U(t)}{lf N(t)},$$

$$\theta = l(t - t_0),$$

$$K = \frac{m}{l},$$

$$\epsilon = \frac{\mu}{l}.$$

The relative leak rate $L(\theta)$, as given by Eq. (11), was determined for four different cases of power history and for both the normal and emergency reactor compartment ventilation systems. The results are shown in Figs. 12 and 13.

The information necessary to calculate $L(\theta)$ is given in Table 4. To obtain the reactor compartment leak-rate constant m , it was assumed that only half of the compartment volume was exhausted by the ventilation system; thus the values of m and therefore the leak rates are overestimated. The values of μ were obtained by fitting the empirical expression for $E(t)$ to the curves of Fig. 4.

A special case of this generalized study of the effect of system holdup is shown in Appendix D in which the compartment concentration (and hence release rate) as a function of time is given as a function of the

Table 4. Information Used to Calculate Reactor Compartment Leakage

	Case 1	Case 2	Case 4	Case 5
Compartment volume ventilated, ft ³	3×10^4	3×10^4	3×10^4	3×10^4
Compartment ventilation rate, ft ³ /min				
Normal system	4000	4000	4000	4000
Emergency system ^a	200	200	200	200
m (day ⁻¹)				
Normal system	192	192	192	192
Emergency system	9.6	9.6	9.6	9.6
l (day ⁻¹)	0.015	0.015	0.015	0.015
μ (day ⁻¹)	74.91	67.1	34.88	30.41
t_0 (sec)	500	1090	2450	3600

^aThe 1800-cfm dilution air flow is not included.

Fig. 12. Release Rate of Gaseous Fission Products from N.S. SAVANNAH Reactor Compartment, Emergency Ventilation.

Fig. 13. Release Rate of Gaseous Fission Products from N.S. SAVANNAH Reactor Compartment, Normal Ventilation.

ratio of the compartment volume to the ventilation rate. In this analysis the fission products were assumed to be released instantaneously from the fuel.

Combined Effects of Decay and Holdup

The decay of the noble gases and iodine isotopes was considered in Section 4. To obtain the release rate of radioactivity to the environment it was necessary to combine the decay information of Section 4 with the reactor compartment relative release rates given in Figs. 12 and 13. The combined effects of holdup and decay were best illustrated by expressing the activity release rate as a fraction of the reference release rate:

$$\text{Fraction of reference release rate} = \frac{\int_0^t N(t) L(\theta) d\theta}{\int_0^t N_0 d\theta}.$$

The release rates of noble gas activity are shown in Figs. 14 and 15. The release rates of the iodine isotopes are given by Figs. 16 through 23.

6. EFFECT OF RELEASE MECHANISMS ON EXPOSURE

The preceding sections cover various effects that would reduce the amount of radioactivity released during the accident. In this section the personnel exposures corresponding to the reduced activity releases are presented. The four reactor operating histories, cases 1, 2, 4, and 5, as defined in Section 2, were considered. The submersion exposure rate at 500 ft from the reactor was computed for each of these power histories during the first 24 hr following the accident, and a curve was prepared to permit interpolation to other distances. Several different sets of assumptions affecting the reactor compartment ventilation, fuel failure mechanism, and fission-product decay were made in each of these cases.

The calculations involved are tedious and time consuming, and therefore only the power history of case 2 was investigated further, since case 2 represents the situation to be expected during most port entries.

Fig. 14. Release Rate of Noble Gas Activity from N.S. SAVANNAH Reactor Compartment, Emergency Ventilation.

Fig. 15. Release Rate of Noble Gas Activity from N.S. SAVANNAH Reactor Compartment, Normal Ventilation.

Fig. 16. Release Rate of I^{131} from N.S. SAVANNAH Reactor Compartment, Emergency Ventilation.

Fig. 17. Release Rate of I^{131} from N.S. SAVANNAH Reactor Compartment, Normal Ventilation.

Fig. 18. Release Rate of I^{132} from N.S. SAVANNAH Reactor Compartment, Emergency Ventilation.

Fig. 19. Release Rate of I^{132} from N.S. SAVANNAH Reactor Compartment, Normal Ventilation.

Fig. 20. Release Rate of I^{133} from N.S. SAVANNAH Reactor Compartment, Emergency Ventilation.

Fig. 21. Release Rate of I^{133} from N.S. SAVANNAH Reactor Compartment, Normal Ventilation.

Fig. 22. Release Rate of I^{135} from N.S. SAVANNAH Reactor Compartment, Emergency Ventilation.

Fig. 23. Release Rate of I^{135} from N.S. SAVANNAH Reactor Compartment, Normal Ventilation.

The total integrated exposures at the site boundary and at the tug were calculated for this case, again under several different assumptions. It is believed that even though only a limited number of conditions was studied, sufficient information was developed to indicate that significant reductions in exposure result if the calculations are based on more realistic ground rules.

The reactor containment vessel of the N.S. SAVANNAH plant is enclosed by a secondary containment system that incorporates a filtered air exhaust system. The filtering efficiency for particulate fission products is expected to be at least 99.9%. For this reason and because the release fraction of particulate fission products from the core is estimated as only 1%, it is evident that exposure to volatile fission products would predominate.⁴

Assumptions Employed in Exposure Calculations

Sections 3, 4, and 5 showed that more realistic assumptions concerning power history, fuel element failure, fission-product decay, and fission-gas holdup in the reactor compartment can reduce the estimates of activity release. Calculations to show what effect each of these individual considerations has on exposures for all values of some of the parameters involved would be voluminous. Therefore the following calculational modes and conditions were selected to illustrate some of the more interesting combinations of effects. Each of these modes may be considered for any power history.

Calculational Modes

Mode A. Continuous equilibrium activity release from the containment system was assumed commencing at the instant of the accident. Fission-product decay after the accident, the time required for the fuel elements to heat up and fail, and the holdup time provided by the reactor compartment were ignored. This mode, when considered in conjunction with the power history of case 1, is the "reference case" that is typical of the conventional exposure calculations made for stationary reactors.

Mode B. Mode B was identical to mode A, except that the decay of fission products between the instant of the accident and the time they entered the atmosphere was included. This provision for decay is somewhat conservative, since it ignores the decay during the time required for the activity to travel from the ship to the exposed personnel.

Mode C. In addition to fission-product decay as in mode B, the time required for fuel elements to fail and the holdup of the ventilation system were factored into the calculation for mode C. Of the various modes considered, mode C was thought to be closest to the conditions at a real accident. Where applicable, the exposure calculations were made for both the normal ventilation system and for the emergency ventilation system.

Fission-Product Release to Containment System

The following release fractions recommended for stationary reactors⁹ were used to calculate the amounts of fission products released from the reactor to the containment shell during the accident:

Noble gases	100%
Halogens	50%
Solids	1%

An additional 50% plate-out fraction for halogens in the containment shell was included, as proposed by the AEC.⁹

Filter Efficiency

Even though shipboard tests have shown somewhat higher filtering efficiencies for halogens, the following filter efficiencies were assumed in accordance with the conservative approach of these calculations:

Normal ventilation system	90%
Emergency ventilation system	99%

No filtering was assumed for the noble gases.

Atmospheric Dilution Near the Ship

Because mobility after an accident is one of the important safety characteristics of a ship propulsion reactor, the exposure to the crew

of a tug engaged in moving the ship was investigated. Analytical methods for calculating atmospheric dilution, such as Sutton's,¹⁰ are not directly applicable to the short distance between the ship and tug. A dilution factor of 1000 was therefore assumed between the stack release point of the normal ventilation system and the tug. This value was based on a dilution factor of 100 between the stack release point and the ship's deck, as used in the Camden study,⁴ plus an additional dilution factor of 10 between the ship's deck and the tug. The corresponding dilution factor for the emergency ventilation system, which discharges from the top of the house, was taken as 100.

Ventilation System Dilution and Source Elevation

The flow rates for the normal and emergency ventilation systems are 4000 and 2000 cfm, respectively. Of the 2000-cfm emergency ventilation flow, 200 cfm is drawn from the reactor compartment and the remaining 1800 cfm is diluent flow from the outside air. The concentration of fission products at the tug was calculated assuming uniform mixing of activity in the ventilation system air stream. The estimate of the concentration at the site boundary and beyond, on the other hand, assumed point source release at ground level. Ground-level release was assumed because some people might be inside buildings at about the same elevation as the actual elevated release point. Point source release was postulated because the flow rates of the ship's ventilation system are relatively small and have a negligible effect on the airborne fission-product concentration at the site boundary and points beyond.¹¹

Calculational Method

Atmospheric Dispersion

The atmospheric dispersion of radioactivity from a ground-level release was calculated from Sutton's atmospheric dispersion equations.¹⁰

Submersion Exposure

Previous calculations⁴ showed that the noble gases would be the main contributors to external exposure. The external exposure rate, D

(in rem/hr), from submersion within a large cloud of radioactive material was estimated from the following expression, taken from ref. 4:

$$D = 9.4 \times 10^{-4} QE \frac{X}{Q},$$

where

Q = (noble gas inventory at the time of the accident) $\times$ (fraction released from the core) $\times$ (fraction leaked per second from the containment vessel),

E = effective gamma energy of noble gases at the time of the accident

= 0.61 Mev,

$\frac{X}{Q}$ = dispersion, sec/m³,

and the value 9.4×10^{-4} is the conversion factor needed to give the dose rate in rem per hour. Submersion exposures close to the source will be overestimated by this equation because of the effect of the size of the finite cloud. All submersion exposures presented in this report are based on the assumption of an infinite cloud. The effect that consideration of the size of the finite cloud would have on the estimated exposures is discussed in Appendix F.

Mode A. For mode A activity release, Q is the source strength of the activity released to the atmosphere in microcuries per second. The value of Q is constant.

Mode B. For mode B, the dose is given by

$$D_{\text{mode B}} = D_{\text{mode A}} \times (\text{fraction of reference energy release rate}).$$

The fractions of reference energy release rate were taken from Fig. 5, and the exposure dose rates during the 24-hr period following the accident were computed.

Mode C. Similarly, for mode C, the dose is given by

$$D_{\text{mode C}} = D_{\text{mode A}} \times (\text{fraction of reference release rate}).$$

The fractions of reference release rate were taken from Figs. 14 and 15. Again, the exposure dose rates during the 24-hr period following the accident were computed.

Thyroid Exposure

Previous calculations⁴ showed that the iodine isotopes would be the main contributors to thyroid exposure. The integrated thyroid dose (TID_{thy}) from iodine inhalation is given by

$$TID_{thy} = D BR \frac{\bar{X}}{\bar{Q}} \int_0^t Q dt ,$$

which was adapted from the work of Cottrell et al.⁴ In this expression

Q = (iodine inventory at time of accident) $\times$ (fraction released from core) $\times$ (fraction not plated out) $\times$ (fraction leaked per second from the containment vessel) $\times$ (fraction passed by the iodine filter),

t = time elapsed after accident, sec,

D = the exposure from 1 μ c of activity (values for the isotopes of interest are listed in Appendix E),

BR = breathing rate, taken as 500 cm^3/sec ,

$\frac{\bar{X}}{\bar{Q}}$ = dispersion, sec/m^3 .

For Mode A, Q , the strength of the source released to the atmosphere (in curies/sec), is constant, so

$$\int_0^t Q dt = Qt .$$

Thyroid exposures were not calculated for Mode B. The effect of fission-product decay is expected to be minor during the first few hours after the accident because of the relatively long half-lives of I^{131} and I^{133} , the main contributors.

For Mode C,

$$TID_{thy, mode C} = TID_{thy, mode A} \frac{1}{t} \int_0^t \left(\frac{\text{fraction of reference release rate}}{\text{release rate}} \right) dt .$$

The integral was evaluated numerically from Figs. 16 through 23 for the four iodine isotopes of interest.

Results of Exposure Calculations

Results of the submersion exposure calculations are shown in Figs. 24 through 30 for both the normal and the emergency ventilation systems. Submersion exposure dose rates at 500 ft are shown in Figs. 24 through 27 for cases 1, 2, 4, and 5 during the 24-hr period after the accident. The individual total integrated dose in 24 hr after the accident is shown in Fig. 28 for case 2 at 500 ft. Figures 29 and 30 show the individual total integrated dose in 24 hr after the accident for case 2 at the tug.

Results of the thyroid exposure calculations are shown in Figs. 31 through 34 for both the normal and the emergency ventilation systems. The individual total integrated dose to the thyroid at 500 ft in 24 hr after the accident is shown in Figs. 31 and 32, while Figs. 33 and 34 contain this information for the tug. If values of the downwind doses are desired at distances other than 500 ft, the 500-ft exposure should be multiplied by the distance factor, as shown in Fig. 35.

Inasmuch as the distance at which an individual would receive 25 rem in 2 hr following the accident is one of the cornerstones of the criteria for both nuclear ships and stationary reactors, it was instructive to determine this distance for several of the cases and modes of interest to ship operation. The results are presented in Table 5.

For mode C there was a difference in the exposure depending upon which ventilation system was in use. No such difference showed up in the calculations for modes A and B, as both of these modes assumed ground-level source release with no holdup and effectively excluded the differences between the two systems.

Table 5. Exclusion Radii Based on a Whole-Body Submersion Dose of 25 rem in 2 hr at Site Boundary

Assumptions	Exclusion Radius (ft)
Case 1, mode A, the "reference case"	475
Case 2, mode A	410
Case 2, mode B	360
Case 2, mode C	
Normal ventilation	245
Emergency ventilation	105

Fig. 24. Submersion Exposure Rate at 500 ft, Case 1.

Fig. 25. Submersion Exposure Rate at 500 ft, Case 2.

Fig. 26. Submersion Exposure Rate at 500 ft, Case 4.

Fig. 27. Submersion Exposure Rate at 500 ft, Case 5.

UNCLASSIFIED
ORNL-LR-DWG 77580

Fig. 28. Total Integrated Submersion Exposure Dose at 500 ft, Case 2.

UNCLASSIFIED
ORNL-LR-DWG 77581

Fig. 29. Total Integrated Submersion Exposure Dose at Tug, Case 2, Normal Ventilation.

UNCLASSIFIED
ORNL-LR-DWG 77582

Fig. 30. Total Submersion Exposure Dose at Tug, Case 2, Emergency Ventilation.

Fig. 31. Total Integrated Thyroid Exposure Dose at 500 ft, Case 2, Normal Ventilation.

Fig. 32. Total Integrated Thyroid Exposure Dose at 500 ft, Case 2, Emergency Ventilation.

7. DISCUSSION

Implications of Exposure Calculations

Various mechanisms have been evaluated, as mentioned above, and shown to have a significant effect on the activity release. Since the alternate release mechanisms investigated had somewhat interdependent effects on exposure, these effects were grouped in the three modes discussed in Section 6. These modes were then compared with a reference case and with each other to show trends. Since integrated exposures are mainly of interest for time-dependent activity releases, exposure rates were calculated only for whole-body submersion at one location (500 ft), as shown in Figs. 24 and 27. The effect of decay was of importance for all power

Fig. 33. Total Integrated Thyroid Exposure Dose at Tug, Case 2, Normal Ventilation.

Fig. 34. Total Integrated Thyroid Exposure Dose at Tug, Case 2, Emergency Ventilation.

Fig. 35. Relative Exposure Along Plume Centerline During Ground Release.

histories considered but was greatest where no reduction in reactor power prior to the accident was assumed; for this case the exposure rate was reduced by a factor of about 1.0 within one day after the accident. It was further noted that consideration of both the delay in cladding failure and the ventilation system holdup, as illustrated by the difference between modes B and C, affected exposure rates the most for the shorter time periods after the accident. For example, in case 2 this effect vanished altogether within 2 hr with the normal ventilation system operating and 12 hr with the emergency ventilation system operating.

The reduction in dose rate afforded by the emergency ventilation system was greater and prevailed for longer periods than the corresponding effect of the normal ventilation system in all cases. This was to be expected because the scavenging flow rate of the emergency system is low and a longer time is required for the activity in the reactor compartment to approach equilibrium.

Integrated Submersion Exposure

Submersion exposures at 500 ft are shown in Fig. 28 for case 2, which is for a power history representative of a typical port entry. Exposures for the reference case are included for comparison. As a measure of the time available to evacuate areas adjacent to the site, the exposure times required to receive a 25-rem whole-body dose at 500 ft can be compared for the various modes considered. If the previous power history is taken into account, the time required to receive 25 rem increases from about 2 hr (the reference case) to 2.6 hr (case 2, mode A). If, in addition, decay of the fission-product activity is considered (case 2, mode B), this time increases to 3.6 hr. Further, if the ventilation system holdup and the time needed for the cladding to fail are considered, about 5.4 and 10 hr, respectively, would elapse with the normal and emergency ventilation systems. The latter time is five times as long as the time computed for the reference case.

Equivalent reductions were predicted in the submersion exposures aboard the tug, as shown in Figs. 29 and 30. It may be noted that the exposures calculated for the emergency ventilation system are considerably larger than those obtained for the normal ventilation system. This

is the result of the atmospheric dilutions assumed between the ship and the tug, as discussed in Section 6. According to this model, the atmospheric concentration of fission products varies inversely with the ventilation system flow rate. On the other hand, a large flow rate from the reactor compartment will decrease the holdup afforded by the compartment, as shown in Fig. 28. A solution to these contradictory requirements may be provided by a ventilation system that draws a rather small amount of air from the reactor compartment (perhaps a flow just sufficient to maintain a negative pressure) and then dilutes this stream with a large volume of relatively clean air drawn from the atmosphere.

Integrated Thyroid Exposure

The use of more realistic assumptions has less effect on thyroid exposures than on submersion exposures. As a measure of the time available to evacuate areas adjacent to the site it is of interest to compare the exposure times required to receive 300 rem to the thyroid at 500 ft. (This exposure represents a hazard comparable to a 25-rem whole-body dose.)

The normal ventilation condition was considered first, and it was found, as shown in Fig. 31, that under the reference case conditions a 300-rem exposure would be received in 0.7 hr. Taking into account the case 2 power history increases this time only to about 0.8 hr. This is because the operating time at reduced power (5 hr) is short compared with the half-lives of the more important isotopes of iodine. If decay, normal ventilation system holdup, and the time needed for the cladding to fail are considered, the exposure time required to receive 300 rem to the thyroid at 500 ft becomes 1.4 hr, or twice the time computed for the reference case. At the tug, the times for the reference case, case 2 mode A, and case 2 mode C were found to be 15, 15.8, and 20.4 hr, respectively. These values were obtained from Fig. 33.

Turning to the emergency ventilation calculation, it should be recalled that an iodine filter efficiency of 99% was assumed, a factor of 10 higher than that of the normal system filters. As a result, the exposures at 500 ft calculated for the emergency system are about a factor of 10 lower than corresponding exposures obtained for the normal ventilation system. It is shown in Fig. 32 that, under the reference case

conditions, 300 rem would be received at 500 ft in 6.5 hr. The corresponding numbers for case 2 mode A, and case 2 mode C are 7 and 11.1 hr, respectively, increases of 7% and 70% over the reference case. The times required to receive 300 rem to the thyroid at the tug showed similar increases over the reference case, as shown in Fig. 34.

Exclusion Distances

A further illustration of the effects of release mechanisms on exposures is shown in Table 5, Section 6, which lists the exclusion radii needed to limit submersion exposure following the accident to 25 rem in 2 hr. While the exclusion radius for the reference case is 475 ft, consideration of case 2 power history and mode C release mechanisms reduces this distance to 245 ft and 105 ft for the normal and emergency ventilation systems, respectively. It should be noted that Sutton's atmospheric diffusion model¹⁰ probably is not valid for these short distances (since the concentration approaches infinity as the distance approaches zero); however, it is the model recommended in the AEC's site criteria and is used here in lieu of a better analytical model.

Justification of Mechanisms Evaluated

The calculations described above demonstrated that several mechanisms operate to delay and reduce the activity release from the N.S. SAVANNAH containment system following the maximum credible accident. Exposures calculated on the basis of these mechanisms were substantially less than exposures calculated for the same accident but employing standard techniques. Inasmuch as the calculational procedures employed here have not been generally accepted by the licensing and regulating authorities, a discussion of the following specific issues is appropriate: (1) power history, (2) core failure sequence, (3) decay of fission products, and (4) holdup effect of reactor compartment.

Power History

The operating power history of a reactor immediately prior to the maximum credible accident would have a significant effect on both the

time to core failure and the activity available for release. The latter effect would be most significant in cases where the noble-gas exposures were controlling. For nuclear ships there is some justification for assuming a power level less than full power for purposes of the accident analysis. This justification is based on the inherent operating characteristics of ships in and near harbors where the accident would have the greatest consequences.

In the analysis given here, a power requirement of 50% power for 5 hr was considered to be representative of typical port entry operations. This power requirement was based on a review of the anticipated power requirements for the harbors of New York and Seattle. Nevertheless, it is recognized that there may be some ports for which the above assumptions concerning power history would not be conservative. It is suggested, however, that, if necessary, an administratively imposed reduction to 50% power for 5 hr prior to docking would not place a severe limitation on ship operation.

Core Failure Sequence

Most hazards analyses are based on instantaneous core meltdown as a convenient, conservative assumption, even though in reality such meltdown would be impossible to attain. If it is assumed that the maximum credible accident in the N.S. SAVANNAH is, indeed, the loss-of-coolant accident, the time to fission-product release is a function of the pre-accident temperature of the fuel, the afterheat within the fuel, the physical properties of the fuel, and such heat-removal mechanisms as might exist. For the calculations described here no credit was taken for heat transfer, and all releasable fission products were assumed to be released from the fuel when any portion of the cladding reached the cladding melting temperature. Nevertheless, calculations which were performed on this basis indicate that, depending upon the prior operation, 10 min to 1 hr is required before the initiation of cladding melting, and considerably longer times are required for the melting of a significant fraction of the core. Thus, in the event of such an accident, the initial exposures, that is, the exposures during the first 10 min to 1 hr after the accident, will be virtually nil. In order to assure that such

time intervals are available, however, other accident sequences by which a core meltdown may occur more rapidly, for example, a nuclear excursion, must be excluded.

As is well known, the long time constant of UO_2 fuel elements in combination with the Doppler coefficient of low-enrichment systems provides a very effective inherent safety mechanism for reactors of the N.S. SAVANNAH type. This has been demonstrated by both theoretical and experimental investigations. For example, startup accident analyses have been performed by General Electric¹² for three types of light-water reactors. The three reactors were characterized by their fuel elements, as follows: (1) uranium-aluminum alloy, highly enriched, in aluminum-clad fuel plates; uranium-zirconium alloy, highly enriched, in zirconium-clad fuel plates; and UO_2 , slightly enriched, in zirconium-clad rods. The results of the analyses indicated that, even with an initial reactor period of 4.34 msec, the UO_2 fuel elements remained well below the melting temperature of the zirconium cladding. With the same initial period, the other two cores suffered extensive fuel-element melting.

Experimental evidence of the effectiveness of the Doppler coefficient in preventing damage to UO_2 fuel elements is also available. SPERT I tests were conducted on a core made up of fuel rods from the N.S. SAVANNAH critical experiment.¹³ Self-limiting power-excursion tests, initiated by step insertions of reactivity, were performed over a wide range of initial asymptotic reactor periods from 30 sec to 3.2 msec. Power excursions with asymptotic periods as short as 3.2 msec were successfully quenched by the Doppler effect without core damage. The implication of the results of these tests is summarized well in ref. 13, as follows:

"The fact that the core was able to sustain without damage a 3.2-msec period self-limiting power excursion, which requires a step reactivity insertion in excess of $\$2$, implies the following:

"(a) In order to obtain a minimum period of 3.2 msec in a ramp-induced, self-limiting power excursion, an estimated ramp rate of about 8 $\$/\text{sec}$ is required for a sufficient time to inject at least $\$2$ reactivity into the system. This ramp rate is perhaps a factor of 50 or more greater than that obtainable in a rod withdrawal accident from most control rod drive systems presently in use. The 8 $\$/\text{sec}$ rate also may

cover any reasonable fuel loading accident in an oxide reactor.

"(b) Whereas no damage occurred in the oxide core for a 3.2-msec step test, the same test in a plate-type, water-moderated reactor would in all likelihood result in substantial fuel plate melting and in mechanical damage by transient steam pressures. That damage occurs in the one case and not in the other appears to be a consequence of the rapidity with which heat in the plate-type core is transferred from the meat to the cladding, and thence (after the cladding surface temperature has risen substantially above the saturation point), transferred quickly to the water. The sudden violent boiling that occurs gives rise to the destructive transient pressures, and, as a result of the decreased plate surface heat transfer coefficient, to the subsequent increase in surface temperature up to the melting point. On the other hand, an oxide fuel rod core with an adequate Doppler coefficient need not depend on heat transfer to effect shutdown, and its low heat transfer rate is advantageous in the event of a severe power excursion in helping to maintain the fuel rod cladding temperature near the saturation point and in suppressing rapid steam formation and consequently violent pressures. The implication of these results is that a rod-type oxide core may be safer than a plate-type core for a research reactor, and also may provide the basis for an efficient burst facility."

Based on the demonstrated effectiveness of the Doppler effect in preventing core damage during a nuclear excursion, it appears that there is good justification for excluding this type of accident in considering possible core failure sequences.

Fission-Product Decay

The decay of fission products following a maximum credible accident determines both the afterheat in the reactor and the amount of any given isotope in existence. Inasmuch as reactor accidents are commonly evaluated by assuming equilibrium full-power operation, the release processes are such that iodine (or, in some instances, strontium) exposures are controlling. There has been little incentive to consider radioactive decay processes in such evaluations. In the case of a mobile reactor such as the N.S. SAVANNAH, both these situations may be altered to such an extent that decay processes could have a significant effect on the

resulting exposures and, hence, siting considerations. The new situation for the N.S. SAVANNAH arises from two independent factors: (1) the power level immediately prior to docking may be substantially less than full power for a significant fraction of a day, and (2) the containment system filters are sufficiently effective in restricting the release of iodine and particulate matter so that the submersion dose from noble gases is controlling. The effect of fission-product decay in the former case is obvious, that is, it gives longer times to core melting following reduced power operation. The effect of decay on the submersion exposure is also significant, inasmuch as most noble gases have relatively short half-lives.

It should be noted that, since the noble gases have a more rapid decay rate than the halogens, situations could exist in which the noble gases were initially controlling but subsequently decayed and the iodine became controlling. It is also pertinent to observe that while effective filters for the noble gases may be employed, such filters are much more cumbersome than filters of comparable efficiency for halogens and particulate matter, so the submersion exposure from noble gases represents a convenient design point.

Holdup Effect of Reactor Compartment

The volume of the reactor compartment is sufficiently large that, if uniform mixing of the containment vessel leakage is assumed, the effective holdup time in the compartment would be significant. This holdup effect is especially important for emergency ventilation because of the relatively low exhaust rate. Of course, uniform mixing throughout the whole containment volume would not be a good assumption because of the partial separation of the compartment into a lower and an upper void. For this reason the calculations given here were based on the assumption that only half of the compartment volume was available for mixing; uniform mixing in this volume was assumed. Actually the degree of mixing is impossible to establish without experimental evidence. One complicating feature of the mixing problem is that the degree of mixing probably depends to a large extent on the location of the leak from the containment vessel. Containment vessel leakage is likely to be distributed

among the numerous penetrations, most of which are located in the lower void. Since the normal ventilation system exhausts from the lower void, the degree of conservatism associated with the mixing assumption may be questionable when the normal system is operating. The emergency system exhausts from the upper void, and therefore it is felt that the mixing assumption is a reasonable representation of the system performance.

Other Pertinent Factors

In addition to the factors discussed above, there are several others that have a significant bearing on the hazards evaluation of nuclear ship operation in harbors. While the factors discussed here have not been specifically analyzed in this report, it is important to understand their relationship to the overall safety evaluation. Such factors include:

1. ability to remove ship from dock after the maximum credible accident,
2. performance of filters and compartment ventilation systems,
3. implications of total population exposure,
4. proximity and resulting exposure of passengers and visitors aboard or near ship, and
5. effect of radioactive cloud size on submersion exposure.

Ability to Remove Ship from Dock After Accident in the Time Required

The question of the feasibility of ship removal was considered only insofar as the resulting exposures to the remaining ship's crew and tugboat personnel were concerned. The problems of ability to evacuate visitors aboard ship or at dockside, availability of tugboats, emergency planning to effect the removal of the ship, etc., are covered by the Port Analysis Report that is prepared for each port to be visited (see refs. 14-18 for such reports on the ports of Savannah, Norfolk, Panama Canal, Seattle, and San Francisco). The exposures calculated herein are all centerline downwind values that could be reduced orders of magnitude by the simple expedient of stationing crew members and positioning the tugboat upwind of the source. Even the centerline downwind submersion exposures (see Figs. 29 and 30) are, however, within the acceptable emergency exposure limits for many hours, those from the normal ventilation system

with its elevated discharge being a factor of ≤ 20 lower than those from the emergency system. The thyroid exposure at the tugs (Figs. 33 and 34) would be considerably greater than the submersion exposure regardless of which ventilation system was employed, since the assumed efficiencies of the iodine filters in the normal and emergency systems have been conservatively set for the interim operating period at 90 and 99%, respectively. Since the criterion for thyroid exposure is 300 rem as opposed to 25 rem for whole-body submersion, it is apparent that the submersion exposure is controlling in most cases; however, thyroid exposures of this magnitude should and, in this case, can be avoided by providing shipboard and tugboat personnel with respiratory protection, such as face masks with charcoal canisters, when operating under such emergencies. Normal practice would also require that, where possible, emergency operating personnel would remain upwind of the source.

Performance of the Filters and Compartment Ventilation Systems

Heretofore most reactor containment systems have consisted of a single containment envelope. For boiling- and pressurized-water reactors, the envelope was commonly a steel pressure shell. The N.S. SAVANNAH containment system consists not only of such a steel shell but also the ventilated reactor compartment within which the steel container is located. Hence it is essential that the compartment integrity be monitored with the same care as that for the steel shell and also that the efficiency of the compartment discharge filters and the operability of the ventilation system be demonstrated and continuously maintained in a manner comparable to that required to assure the continued integrity of the steel shell. More specifically, no significant activity may be permitted to leave the reactor compartment except through the normal or emergency ventilation systems, which are provided with iodine and particulate matter filters.

In the present study iodine filter efficiencies of at least 90 and 99% were assumed for the normal and emergency ventilation systems, respectively. Actually, the efficiencies of these two filters have been repeatedly checked and shown to be greater than the prescribed minimum values. It may be anticipated that, following the review of the initial operation of the N.S. SAVANNAH, the assumed efficiencies for the filters

in both of these systems will be increased. This could have an important bearing on the dock facilities at which the N.S. SAVANNAH would be permitted to berth. Once the iodine and particulate matter had been effectively attenuated by filtration, the noble gases would be controlling and, if a further reduction in accident exposures was desired, it would be necessary to provide for attenuation of the noble gas activity. Removal of the noble gases by filtration might also be considered, although before such a scheme would be practical it would be necessary to reduce the volume by, for example, the installation of an annular shell immediately outside the steel containment shell that would be maintained at a slight negative pressure by the low flow in question.

Implications of Total Population Exposure

The data presented here indicate the effect of various processes and mechanisms on activity release and hence upon exposures from the N.S. SAVANNAH maximum credible accident. It was shown in Table 5, Section 6, that these processes permit the exclusion radius to be reduced in the cases cited from 475 to 105 ft, with similar reductions in the controlled-zone boundary. It appears from these studies that the distance requirements for nuclear ship berths should be attainable within existing regulations, if, where necessary, a case could be made for the validity of one or more of the mechanisms described here. In any event, because of the proximity of population centers to most important ship berthing areas, it is generally impossible for the berthing areas to fulfill the population center distance stipulation of the Reactor Site Criteria.¹ In lieu of this stipulation, a total population exposure criterion of 2×10^6 man-rem has been proposed,⁴ and, as indicated in Appendix E, it would be possible for a nuclear ship such as the N.S. SAVANNAH to fulfill this criterion if the maximum credible accident were to occur in the New York Harbor. It is true, of course, that the site boundary exposures and the total population exposures would not be independent, so for a given site (or berth) when one exposure increases (or decreases) the other also increases. Furthermore the establishment of a 24-hr limit (because of the assumed removal of the ship) in the determination of the controlled-zone boundary has the effect of reducing the area that can be subjected

to a high (up to 25 rem) exposure, with the result (as indicated by the population exposures for New York Harbor given in Appendix E) that even for ports with high population densities the total population exposure (of persons outside the controlled zone) does not approach the 2×10^6 man-rem criteria in 24 hr. The concept of the total population exposure would thus appear to have more meaning as applied to stationary reactor site evaluations because the low population distance is so great that large numbers of people may be concentrated immediately beyond. Even in berth evaluation, however, the concept is useful, since it serves as an index to measure the number of persons who may be admitted to the immediate vicinity of the ship. In the case of Savannah Harbor, people in the immediate vicinity would account for 383,000 man-rem, or over 70% of the total estimated population exposure.

Implications of Limited Time in Port

The fact that a nuclear vessel is in any one port only a small percentage of the time has a direct bearing on the probability of the occurrence of the accident in that port; however, this probability is only one of many that could be associated with the occurrence of the maximum credible accident at any place or time, none of which are included in hazards calculations. Hazards evaluation technology will have to become much more refined than it is at the present before probability factors can be meaningfully evaluated in such an analysis.

Proximity and Resulting Exposure of Passengers and Visitors Aboard Ship

The existence of members of the public, passengers and visitors, either aboard or at dockside constitutes an exposure potential that has no significant counterpart in a stationary power plant. The tentative criteria on port operation of the N.S. SAVANNAH proposed by the Maritime Administration suggested that such persons be evacuated before they had received more than a 25-rem whole-body dose or a 300-rem thyroid exposure.² If these exposure criteria are accepted as reasonable, the real question is whether it is feasible to carry out the evacuation of these persons before they would receive an excessive exposure. Since the delay

in activity release due to the time to cladding melting and the holdup in the reactor compartment defer the time to maximum activity release for several hours (see Figs. 14 through 23), the required evacuation is feasible, even if carried out in up to 2 hr with the evacuated persons downwind (see Figs. 28 to 34). Furthermore, if the actual finite cloud size were considered, a substantial reduction would result in the submersion exposures previously calculated on the basis of a uniform concentration equivalent to the Sutton¹⁰ centerline downwind concentration (see the following section). Actually, emergency evacuation procedures would be such that times in downwind concentrations would be minimized, and the evacuation should be effected in much less than 2 hr.

Effect of Radioactive Cloud Size on Exposure

Whereas most submersion exposure calculations have assumed an infinite radioactive cloud of uniform concentration, the data in Appendix E indicate that, at distances close to the source, submersion exposures will be overestimated by from one to two orders of magnitude. This is of particular importance to nuclear installations such as the N.S. SAVANNAH for which the submersion exposure is controlling and which require some operating personnel to remain close by following an accident. The exposures to persons being evacuated may also be substantially less than would otherwise be calculated.

8. CONCLUSIONS

In estimating the activity released from the N.S. SAVANNAH in the maximum credible accident, there is reasonable justification for including the influence of reduced power operation during port entry, delay in transport of containment vessel leakage through the reactor compartment, and radioactive decay following the accident. The combined effects of these factors, when treated in a conservative manner, reduce calculated exposures by approximately an order of magnitude for the situations of interest. Although this reduction is not large compared with that afforded by the primary attenuating mechanisms, that is, containment vessel holdup, iodine filtration, and atmospheric dilution, it

may be important in assessing the feasibility of evacuating personnel from the immediate vicinity of the ship and of moving the stricken ship to a remote anchorage.

A reduction in power during port entry would affect both the time to core failure and the activity available for release. A power history consisting of continuous full-power operation followed by 5 hr of operation at 50% of full power (case 2) gives a reasonable approximation of the fission-product inventory at the time of docking under typical port entry conditions. This reduction decreases the noble gas (gamma energy) inventory to about two-thirds of the full-power value and increases the time between loss-of-coolant and initial cladding failure by at least several minutes. The delay in cladding failure has a significant effect on the activity release rate at short times. At times longer than about 1 hr, however, the conservative heat transfer model used indicates that the cladding of most of the rods would have failed with either power regime.

Radioactive decay following the accident would be an important factor in alleviating the noble gas exposure hazard. On the other hand, the radioiodine release rate would not be appreciably reduced by decay because of the relatively long half-lives of the important isotopes I^{131} and I^{133} .

It is desirable and practical to provide particulate matter and iodine filtration of the air exhausted from the reactor compartment so that the controlling exposure would be that from noble gases. This condition is usually a convenient design point in view of the difficulty of removing noble gases from air streams. An iodine removal efficiency of 99% or better is needed for this purpose.

The reactor compartment would delay the release of activity to the environment and permit particulate matter and iodine filtration to be accomplished prior to release. The compartment delay is significant if it can be assumed that there will be uniform mixing of containment vessel leakage with a large fraction of the compartment volume. The reduction in exposure rate would be most pronounced during the period when concentrations in the compartment were building up to equilibrium values,

but there would be a corresponding reduction in accumulated exposures at longer times.

Maximum benefit from reactor compartment delay would be obtained if (1) the compartment exhaust rate were the minimum required to maintain the compartment at negative pressure, (2) there were assurance of uniform mixing within the compartment or that flow patterns in the compartment were such that containment vessel leakage would not pass directly to the exhaust without a comparable time delay, (3) a large flow of diluting air were drawn from the atmosphere to reduce the concentration of fission products near the discharge point, and (4) the discharge point were elevated to provide additional atmospheric dilution before exposure of nearby personnel.

Considerations of reduced power during port entry, radioactive decay, and delay in release appreciably reduce the calculated exposures in the maximum credible accident. If it is assumed that the emergency ventilation system is operating, the present calculations indicate the following for case 2 with respect to the reference case:

1. The 2-hr submersion dose at the tug or at 500 ft downwind is reduced by a factor of about 10, the major contribution being delay in the reactor compartment.
2. The 24-hr submersion dose at 500 ft downwind is reduced by a factor of about 7, with radioactive decay being the principal contributor.
3. The 2-hr thyroid dose is reduced by a factor of about 3.5, and the 24-hr thyroid dose by a factor of 1.5.
4. The exposure time for receiving a 25-rem whole-body dose is increased by a factor of about 5 to about 10 hr.
5. The exclusion radius (for 25 rem in 2 hr) is reduced by a factor of 4 to about 100 ft.
6. The exposure rate is lowest immediately after the accident, reaches a maximum in about 4 hr, and then gradually decreases, reaching half the maximum value at about 16 hr after the accident.

If the normal ventilation system is operating, the exposure reductions are less pronounced. This is due to the higher flow rate and consequent shorter time delay in the reactor compartment and to the fact

that thyroid exposures are controlling as a result of the comparatively low iodine-removal efficiency assumed for this system.

The calculated exposures are sufficiently low and the calculations retain sufficient conservatism that there is good assurance that personnel will not receive an excessive dose while the ship is being moved to a remote anchorage. The use of respiratory protection and of radiation monitoring to assist in positioning personnel engaged in the move would be advisable and would further reduce exposures.

PAGES 62 to 64
WERE INTENTIONALLY
LEFT BLANK

Appendix A

DIFFERENCES BETWEEN N.S. SAVANNAH AND
STATIONARY POWER PLANTS

The differences¹⁹ that have some bearing on the relative safety of nuclear ships and most stationary power plants have been identified. The N.S. SAVANNAH power plant is either possessed of or provided with these features by virtue of the marine environment, with the result that it has several significant differences from a stationary reactor of comparable size. The design and operation of a nuclear vessel, as well as safeguards analysis and evaluation, should take into account each of these differences, insofar as possible. Their probable effects on the health and safety of the general public are discussed here.

The "Population Center Distance"

Facilities that meet the needs of large ships exist only at large ports. Therefore if nuclear vessels could not be berthed in major ports, their commercial usefulness would be seriously curtailed. On the other hand, while there are obvious economic advantages to the location of stationary power reactors near population centers, such considerations are not (at the present time, at least) overriding. Hence the recently approved Reactor Site Criteria¹ established certain minimum distances by which a power reactor should be separated from population centers. Such separation distances are incompatible with nuclear vessels. In order to compensate for the proximity to population centers, the N.S. SAVANNAH not only is provided with an additional containment barrier, but also it can be removed from the dock to an isolated anchorage. Environmental analyses of various ports have shown how effective the N.S. SAVANNAH containment is expected to be. The maximum individual exposure at the site boundary and the total population exposure in the event of an accident would be well below the values considered acceptable for stationary power plants, even if relatively long removal times were considered.

Characteristics of the Area Surrounding the Ship Berth

In addition to the proximity to population centers, ship berths may be expected to be characterized by relatively small exclusion areas and high population densities in the immediate vicinity of the exclusion, or controlled, area boundary. Since nuclear ships will be expected to utilize existing berthing facilities, insofar as possible, the "effective" exclusion area that could be associated with any berth would only be that which might "happen" to exist. Other things being equal, it would be desirable to select the berth which afforded the greatest exclusion area. The N.S. SAVANNAH is designed, however, so that in the event of the maximum credible accident, the exposures at comparable distances from the reactor are orders of magnitude less than those in most stationary power plants. Furthermore, should the exposure at the site boundary tend to exceed the specified limit because of the small size of the exclusion area, the potential exposure could be reduced by restricting the operating history of the ship prior to berthing.²⁰⁻²⁴

Proximity of Some Members of the Public to the Reactor

The crew of a nuclear ship is the counterpart of the operating staff of a stationary power plant. As far as routine exposures and dockside accidents are concerned, the ship's crew belongs in the category of radiation workers and would be subject to occupational exposure limits as defined by the ICRP.²⁵ Similarly, passengers on nuclear ships are presumably regarded, as far as routine exposures are concerned, in the same light as "members of the public living in the neighborhood of controlled areas." Furthermore, when at dockside it should be possible to evacuate passengers so that no passenger would receive more than 25 rem to the whole body or 300 rem to the thyroid in the event of an accident. On the other hand, when at sea, the passengers will be subject to much the same risks as the crew, just as the passengers and crew share the same risks in a conventional ship. It should be noted that the Reactor Site Criteria¹ make no specific requirements for operating personnel safety in the event of an accident. Furthermore, visitors to a stationary

nuclear facility assume some of the risks of the operating personnel. Somewhat similarly, passengers on a nuclear vessel forego their status as members of the public at large and assume some of the risks of the operating personnel. On the other hand, the safety features of the N.S. SAVANNAH, most significantly the multiple containment provisions, provide a degree of protection of the crew, as well as the passengers, which is not required or generally found at stationary reactors, and which should be capable of limiting the resulting exposure to less than 25-rem whole body or 300-rem thyroid.

Effect of Operating Power History

Stationary nuclear power plants normally operate at or near design power as much of the time as possible. The operating power on a nuclear ship will be reduced to about 50% of normal operating power as it approaches a harbor and will be further reduced to the hotel load of about 15% immediately after berthing. Thus the magnitude of the potential activity release and the time before the activity is released are both improved with time following the reduction of power prior to entering a port. Should some particular situation require an even further reduction in potential activity release, this could be accomplished by further reductions in power for longer times.

Removal of Ship to a Remote Anchorage

In the siting of stationary power reactors the zone outside the exclusion area must be so sparsely populated that it can be evacuated in such a manner that no one will receive exposures in excess of 25-rem whole body. The evacuation time is 2 hr at the inner boundary of the zone and infinity at its outer perimeter. One of the biggest single safety advantages of a nuclear ship power plant over its stationary counterpart lies in the fact that the ship is mobile. Thus, in the event of an accident, or in anticipation of an accident in which fuel element melting is expected, the ship may be removed from its berth to a predetermined remote anchorage.

Release to the Hydrosphere

A ship is always subject to accidents that conceivably could result in its being sunk. Thus, should the maximum credible accident and the ship sinking occur simultaneously, the activity would be released to the water. From the standpoint of the dose to the surrounding population, this would undoubtedly be preferable to the uncontrolled release of activity to the atmosphere, since, as was shown in the Camden site study,²⁶ the resulting activity concentration in the Delaware River at Camden would barely exceed maximum permissible values if only the steel containment shell was effective in limiting the release of activity. Values of the hydrospheric dispersion parameters for various harbors are given in the respective site reports. The subject is not discussed further here because the hazards are less significant than those from atmospheric dispersion of the same activity.

Accidents as a Consequence of Plant Mobility

Accidents caused either by the motion of a nuclear vessel or other ships have no reasonable counterpart in stationary reactor operation. Such accidents would include power plant failures induced by ship motion, vibrational fatigue, storms, ship collisions, and groundings. Safety factors to preclude the possibility of such failures were factored into the design of the plant. While it cannot be concluded that such mechanisms cannot contribute to an accident, insofar as possible the design minimizes the possibility. It is regarded as incredible that these mechanisms can act simultaneously to cause the maximum credible accident and rupture a containment barrier in port. Furthermore, while the possibility of ship accidents increases in harbor waters because of the traffic density, the probability of accidents of sufficient severity to release activity actually decreases as the ship enters the harbor and approaches its berth because of the restriction on permissible speeds in harbor areas. Specifically, the N.S. SAVANNAH is designed²⁷ so that no credible accident in harbor waters could either (1) produce the maximum credible accident or (2) rupture the pressure containment vessel,

although such an accident is conceivable when the ship is on the high seas.

Electrical Power

In a stationary power plant, reliable power may be obtained from lines to a number of grid systems, in addition to an emergency station. Aboard ship, however, no external sources are available and emergency power must be provided from self-contained power sources. The continued operation of the blowers in the ventilation system is of course dependent upon the reliability of the electrical power supply. Emergency power is available to these blowers from the emergency diesel system, as well as from the auxiliary diesel generators.

In addition to the normal power supply from the turbine-generators, the two auxiliary diesel generators can supply 1500 kw on a normal basis or more than 1800 kw for periods of at least 2 hr through either or both of the two main power buses. The 300-kw emergency diesel generator can supply the reactor emergency cooling system through separate buses and would be automatically switched to furnish power to all vital loads, such as reactor compartment ventilation, primary pumps, etc., if the normal or auxiliary power sources failed. Like the auxiliary diesel generators, it is rated for 125% output for a minimum of 2 hr. Control power is supplied at all times from a battery-backed floating bus. Twin sets of motor-driven inverters and alternators, either of which can supply the total bus load, assure uninterrupted battery charge maintenance and availability of control power.

Evacuation of the Ship at Sea

Abandoning the ship at sea is not necessarily feasible or less hazardous than radiation exposure. Emergencies on the high seas are conceivable in which it might be considered less hazardous to remain aboard ship even in the face of certain radiation exposure. This would be less applicable to ships at berths or even in harbor waters. Some

conditions, such as storms, might impair the ease with which the passengers and crew could be removed.

Time in Port

A ship would visit many ports and would be expected to be in any one port only a small fraction of the time. In a stationary power plant the surrounding population is subjected to the full potential of an accident approximately 90% of the time. On the other hand, the port population is not subject to any exposure potential when the ship is not in port.

Ship Compartmentalization

Considerations of seaworthiness and safety against sinking dictated that the containment vessel be housed in a watertight compartment that is adequately reinforced to maintain the integrity of the ship's hull under all sea conditions. This compartment is provided with two independent ventilation systems, each containing provisions for removal and entrapment of exhausted airborne radioactivity, and is maintained at a slight negative pressure with respect to the atmosphere. One of the ventilation systems is operated from the ship's emergency electrical supply, and thus effective activity removal is independent of the operation of the ship's power plant. The compartment constitutes a complete second containment shell surrounding the inner vessel.

Since the reactor compartment is watertight, it would be possible in the event of a primary piping rupture to flood this compartment partially in order to cool the containment vessel and reduce the pressure inside at an increased rate, thereby minimizing fission-product leakage to the atmosphere. This procedure would have an added advantage in that some of the leaking activity would be adsorbed in the water; however, the compartment ventilation system would eventually be rendered inoperable. An inexhaustable supply of coolant is assured because cooling external to the containment vessel can utilize the adjacent sea

water. Flooding of the reactor compartment could be easily accomplished by a fire boat, if available.

Collision Protection

To maximize the protection afforded the primary containment vessel against penetration by collision, the hull of the ship in the vicinity of the reactor was strengthened²⁷ by increasing the thickness of deck plates and by placing a layered steel and redwood collision barrier immediately outboard of the reactor compartment. This additional reinforcement not only renders the containment vessel invulnerable to penetration by all existing merchant ships at harbor speeds but also substantially improves the resistance of the reactor compartment to failure.

Appendix B

FISSION-GAS RELEASE DURING OPERATION

An analysis was made of the pressure buildup in the fuel rods because of fission-product gases leaving the UO_2 -fueled matrix during the lifetime of the core. The purpose of the analysis was to determine the temperature at which the cladding could be expected to fail in a temperature transient following a loss-of-coolant accident. The release of fission gases from the fuel into the volume provided by the open pores of the UO_2 pellets and the free volume of the fuel element was calculated based on recently obtained data for the pellet temperature distribution and rare-gas release. A degree of conservatism was maintained by using the most reasonable pessimistic data available. It was found that 2% of the rare-gas inventory would be released during the lifetime of the core in the highest flux region at the midplane of the central rod. The conservative simplifying assumption was therefore made that 2% release could be expected from all the fuel elements in the core.

Since a high fraction of release has been found to occur in UO_2 undergoing grain growth, the extent of grain growth in the N.S. SAVANNAH fuel was estimated by comparison with suitably correlated experimental data. It was found that grain growth would not occur in the peak power region of the core, which has a peak-to-average power ratio²⁸ of 3.75 during 69-Mw operation.* Power operation in excess of 21% above 69 Mw would be required to initiate grain growth. Without grain growth, reasonably accurate estimates of the rare-gas release in the relatively low-density (91 to 93% theoretical) pellets used in core 1 of the N.S. SAVANNAH reactor could be obtained by use of the Booth model.^{29,30} In this model, the UO_2 compact is assumed to be composed of microspheres that communicate with the atmosphere through open pores.

The pressure buildup in the fuel sheath was calculated using 2% rare gas, iodine, cesium, and rubidium release added to the partial-pressure contribution of the initial 1 atm of helium fill gas. The volume filled

*The axial ratio is 1.5; the radial, 2.0; and the local, 1.25.

by this gas mixture that contributes to the stress on the cladding was taken as the open porosity of the UO_2 plus the free volume between the pellets and the stainless steel cladding with end caps in place.

The tangential cladding stress caused by the internal pressure build-up was compared with data obtained in stress-rupture experiments performed at ORNL, and it was concluded that the stress was sufficiently low that the fuel elements would not fail until the cladding temperature approached 2550°F , which is the melting point of stainless steel.

Discussion of the Conductivity Integral

In order to estimate the amount of fission-gas release during operation, it was necessary to investigate the existence of grain growth because of the high fractional release reported in regions of grain growth. For this study, experimental data were scanned to find the lowest heat rate that would yield grain growth, and the result was compared with the heat rate in the N.S. SAVANNAH reactor. The parameter used for the comparison was

$$\int_{T_s}^{T_{gg}} K(T) dT, \quad (\text{B.1})$$

where

- K = thermal conductivity of the UO_2 ,
- T_s = surface temperature of pellet,
- T_{gg} = temperature of structural change in UO_2 , that is, grain growth, in this case.

This parameter bears the following relationship to the linear heat rate in a solid, cylindrical pellet:³¹

$$\int_{T_s}^{T_c} K(T) dT = \frac{W}{4\pi}, \quad (\text{B.2})$$

where

- T_c = central temperature,
- W = linear heat rate of the fuel element (w/cm).

Since parameter (B.1) does not require detailed knowledge of the thermal conductivity and is independent of the pellet radius, it is useful for comparison of experimental results. In order to determine the existence of grain growth in the N.S. SAVANNAH reactor fuel, it was necessary only to show that the following inequality was satisfied:

$$\int_{T_{sr}}^{T_{cr}} K(T) dT < \int_{T_{se}}^{T_{gg}} K(T) dT + \int_{T_{sr}}^{T_{se}} K(T) dT, \quad (B.3)$$

where

T_{sr} = pellet surface temperatures in the reactor under study,

T_{cr} = pellet central temperature in the reactor,

T_{gg} = grain-growth temperature,

T_{se} = pellet surface temperature measured experimentally.

The value of the conductivity integral is

$$\int_{T_{sr}}^{T_{cr}} K(T) dT = 23.4 \text{ w/cm}$$

at 69 Mw for the midpoint of the central fuel element, which has a peak-to-average flux ratio of 2 radially, 1.5 axially, and 1.25 locally, yielding a peak-to-average ratio of 3.75.

Determination of the Pellet Surface Temperature

In order to compare the calculated value with the experimental value, the surface temperature of the pellet must be known. This is dependent on the heat flux, cladding thickness, temperature difference between the cladding outer surface and the cooling water, temperature of the water cooling the exterior of the cladding, and the interface thermal conductance between the pellet and the cladding. The following values were used for this analysis:

Peak heat flux*	243,000 Btu/hr·ft ²
Cladding thickness	0.035 in.

*Derived from 69-Mw operation and a peak-to-average flux ratio of 3.75.

Water temperature ²⁸	509°F
Cladding to water temperature difference	11°F
Thermal conductivity ³² of type 304 stainless steel at 520°F	10.7 Btu/hr·ft ²
Interface thermal conductance	1000 Btu/hr·ft ² ·°F
	0.567 w/cm ² ·°C

The interface thermal conductance is difficult to evaluate analytically because of the fracturing of the UO₂ during irradiation and the subsequent uncertainty in the gap thickness to be used in thermal conductivity calculations. Experiments have been performed, as reported in ref. 31, for correlating gap conductance deduced from observations of grain growth and melting of UO₂ samples in metal sheaths. Some of these data are presented in Fig. B.1 as a function of the ratio of diametral

Fig. B.1. Effect of Clearance on Gap Conductance (taken from ref. 34).

clearance to the pellet diameter, as assembled. For an assembled clearance ratio of 0.01, approximately the value for the N.S. SAVANNAH fuel, it can be seen from Fig. B.1 that a sheath-to-fuel thermal conductance of $0.567 \text{ w/cm}^2 \cdot ^\circ\text{C}$ ($1000 \text{ Btu/hr} \cdot \text{ft}^2 \cdot ^\circ\text{F}$) lies in the low region of the range of data from experiments with stainless steel-clad fuel. Based on these values, the surface temperature of the peak power pellet was found to be 432°C , or 809°F .

Use of the Conductivity Integral

Although $\int_{T_s}^{T_{gg}} K(T) dT$ is a function of the linear heat rate alone, evaluation of one of the limits of integration when the other is known requires knowledge of the thermal conductivity, $K(T)$. The central pellet temperature is shown in Fig. B.2 as a function of the conductivity integral³³ for thermal conductivity data³⁴⁻³⁶ applicable when the bottom limit of the integral is 100°C . It is apparent from Fig. B.2 that for a given linear heat rate the temperature of the upper limit of integration increases as the thermal conductivity decreases. Although the Kingery^{34,35} data are probably the most reliable, the Hedge and Fieldhouse³⁶ data were used here because of the extra conservatism imparted to the analysis; that is, if grain growth were not found using the Hedge and Fieldhouse data, it would not be found using any of the other data.

Thus the inequality of Eq. (B.3) would be

$$\int_{432^\circ\text{C}}^{T_{cr}} K(T) dT < \int_{T_{se}}^{T_{gg}} K(T) dT + \int_{432^\circ\text{C}}^{T_{se}} K(T) dT . \quad (\text{B.3a})$$

The last term can be found from Fig. B.2 by breaking it into two parts, as follows:

$$\int_{432^\circ\text{C}}^{T_{se}} K(T) dT = \int_{100^\circ\text{C}}^{T_{se}} K(T) dT - \int_{100^\circ\text{C}}^{432^\circ\text{C}} K(T) dT . \quad (\text{B.4})$$

The experimental data³¹ for $\int_{T_{se}}^{T_{gg}} K(T) dT$ for six experiments with helium fill gas in the gap clearance and 21 experiments with argon fill gas were scanned to find the lowest value of the integral that yielded

UNCLASSIFIED
ORNL-DWG 63-2352

Fig. B.2. Central UO_2 Pellet Temperature as a Function of the Thermal Conductivity Integral Based on the Thermal Conductivity Data of References 35 and 36.

grain growth within the pellet. This value was found to be 23.8 w/cm . The surface temperatures in this experiment was deduced to be 650 $^{\circ}\text{C}$ using Hedge and Fieldhouse data.³⁶ Substituting the values obtained from Fig. B.2 into Eqs. (B.3) and (B.4) yielded

$$\int_{432^{\circ}\text{C}}^{T_{\text{cr}}} K(T) dT = 23.8 + 21 - 16 = 28.8 \text{ w/cm} .$$

Since

$$\int_{432^{\circ}\text{C}}^{T_{\text{cr}}} K(T) dT = 23.4 \text{ w/cm}$$

for the pellet in the highest flux region of the N.S. SAVANNAH reactor, the inequality of Eq. (B.3) is satisfied and grain growth does not occur.

Calculation of Fission-Gas Release

Having established that the high rates of release associated with grain growth do not occur under normal operating conditions, an estimate was made of the amount of fission gases that would diffuse out of the UO_2 into volumes within the fuel rod, since these gases would contribute to the internal pressure and would be released immediately upon failure of the cladding. For the relatively low-density pellets of core I (91 to 93% of theoretical density), the Booth³⁰ model should give reasonable estimates.

The Booth model is based on the assumption that the compacted and sintered UO_2 pellet can be idealized as a compact of many microspheres with the voids between interconnected and communicating with the free volume within the cladding. The radius of these spheres is taken as:

$$a = \frac{3d}{S}, \quad (\text{B.5})$$

where

a = radius of equivalent microsphere, cm,

d = fraction of theoretical density,

S = total surface area per unit volume (cm^2/cm^3).

Thus, mathematical solutions of Fick's second law,³⁷ with appropriate boundary conditions should represent the rate of diffusion of fission products from UO_2 .

The diffusion equation for a fission product produced at a constant rate at a constant temperature without radioactive decay is

$$\frac{\partial C}{\partial t} = D \left(\frac{\partial^2 C}{\partial r^2} + \frac{2}{r} \frac{\partial C}{\partial r} \right) + B, \quad (\text{B.6})$$

where

C = concentration of fission products as a function of radius and time,

r = radial position in sphere,

t = time,

D = diffusion coefficient,

B = rate of production.

The solution for the fraction, f , of the given fission product released in time, t , is³⁸

$$f = 1 - \frac{6a^2}{90Dt} + \frac{6}{\pi^4 Dt} \sum_{n=1}^{\infty} \frac{1}{n^4} \exp(-n^2 \pi^2 Dta^2) . \quad (\text{B.7})$$

For $f < 0.57$, corresponding to values of the ratio

$$\frac{\pi^2 Dt}{a^2} \leq 1 ,$$

and the value of f is given by the approximation

$$f = 4 \left(\frac{Dt}{\pi a^2} \right)^{1/2} - \frac{3Dt}{2a^2} . \quad (\text{B.8})$$

Again, the second term is negligible for values of

$$\frac{\pi^2 Dt}{a^2} < 0.01 .$$

For nonisothermal conditions, the pellet must be subdivided into radial zones of constant temperature and Eq. (B.8) must be applied for each zone. This is necessary because of the strong temperature dependence of the diffusion coefficient D , as expressed by

$$D = D_{\text{ref}}(T) \exp(-Q/RT) , \quad (\text{B.9})$$

where

- D = the diffusion coefficient,
- $D_{\text{ref}}(T)$ = the diffusion coefficient at a reference temperature,
- Q = activation energy,
- R = gas constant,
- T = temperature, degrees absolute.

The procedure, while tedious, has been performed by Robertson and his co-workers, and their results are presented in Fig. B.3, which was taken from ref. 39. The data were obtained for a pellet surface temperature of 400°C and present fractional release of xenon as a function of

UNCLASSIFIED
ORNL-DWG 63-2353

Fig. B.3. Fractional Release of Xenon in a Cylindrical UO_2 Fuel Element as a Function of Thermal Rating, Time, and Diffusion Rate (Taken from ref. 39, Courtesy, American Institute of Mining, Metallurgical, and Petroleum Engineers, Inc.).

$D'_{1400^{\circ}\text{C}} t$ and $\int_{400^{\circ}\text{C}}^{T_e} K(T) dT$, where $D'_{1400^{\circ}\text{C}}$ is the value of D/a^2 at 1400°C , and t is the time of irradiation. The data of Fig. B.3 were based on an activation energy of 46,000 cal/g·mole, compared with the more commonly accepted value of 71,000 cal/g·mole.³¹ This difference had the effect of overestimating the release at temperatures below 1400°C , and therefore gave a degree of conservatism to the data of Fig. B.3 when applied to the

N.S. SAVANNAH reactor. The conductivity integral of the reactor was adjusted to the 400°C surface temperature condition for use with the data of Fig. B.3 by the following relation:

$$\int_{400^{\circ}\text{C}}^{T_c} K(T) dT = \int_{432^{\circ}\text{C}}^{T_c} K(T) dT + \int_{400^{\circ}\text{C}}^{432^{\circ}\text{C}} K(T) dT, \quad (\text{B.10})$$

where the last term can be found from Fig. B.2 as follows:

$$\int_{400^{\circ}\text{C}}^{432^{\circ}\text{C}} K(T) dT = \int_{100^{\circ}\text{C}}^{432^{\circ}\text{C}} K(T) dT - \int_{100^{\circ}\text{C}}^{400^{\circ}\text{C}} K(T) dT : \quad (\text{B.11})$$

Based on the conservative data of Hedge and Fieldhouse,³⁶ this adds 1.2 w/cm to the N.S. SAVANNAH reactor value of 23.4 w/cm, yielding

$$\int_{400^{\circ}\text{C}}^{T_c} K(T) dT = 24.6 \text{ w/cm}$$

for use with the data of Fig. B.3. This procedure for correction to a 400°C surface temperature yields the correct central temperature but underestimates the surface temperature. Since the central portions of the pellet release by far the largest amounts of fission gas, this discrepancy is negligible.

The value of $D'_{1400^{\circ}\text{C}}$ used was $2.8 \times 10^{-11} \text{ sec}^{-1}$ (ref. 40), as determined experimentally for N.S. SAVANNAH fuel, and t was $6.52 \times 10^7 \text{ sec}$, based on full-power operation at 69 Mw for a core life of 52,000 Mwd. Substituting these values and reading from Fig. B.3 yields a release value of approximately 2% for the peak power pellet in the reactor. The release could have been mapped as a function of core power distribution, but for this core it was assumed that the whole core released the same fraction as the peak power pellet, thus giving a further degree of conservatism in the results.

Pressure Buildup and Cladding Stress in Fuel Rod

The fission gases that are released to open pores and thence to the free volumes within the fuel sheath contribute to the internal pressure. In calculating the steady-state pressure, it is necessary to include the

effect of the higher temperature within the open pores in the pellets by a relationship of the form

$$p = \frac{nR}{\int_{V_i} \frac{dV_i}{T_i}} \quad (\text{B.12})$$

where

p = pressure,

n = number of moles of fission gas released,

R = gas content,

V_i = volume of gas experiencing temperature T_i .

The temperature becomes essentially isothermal within a short time after the loss-of-coolant accident because of the leveling of the temperature profile caused by the decrease in heat generation and the removal rate. Thus it can be assumed that all the gas is at a uniform temperature at any given time after the accident.

The total pressure is the sum of the partial pressures of each nuclide:

$$p_t = \sum_{i=1}^n p_i, \quad (\text{B.13})$$

and

$$p_i = \frac{n_i RT}{V} \quad (\text{B.14})$$

where

n_i = number of moles of element i ,

V = open pore volume plus free volume.

The open pore volume was found to be approximately 7% of the pellet volume from Fig. B.4. Thus the total open porosity per rod would be 10.1 cm³, as found from

$$V = 0.07 \left[\frac{m}{(0.91) \rho_{td}} \right] \frac{1}{N}, \quad (\text{B.15})$$

UNCLASSIFIED
ORNL-DWG 63-2354

Fig. B.4. Porosity as a Function of Density for ADU Oxide Pressed to 65% of Theoretical Density and Sintered in Hydrogen (Taken from J. Belle and B. Lustman, pp. 442-515 in "Fuel Elements Conference, Paris," TID-7546, Book 2, March 1958).

where

$$\begin{aligned}
 m &= \text{mass of the fuel in the core} = 7.555 \times 10^6 \text{ g}, \\
 \rho_{td} &= \text{theoretical density, } \sim 10.97 \text{ g/cm}^3, \\
 N &= \text{number of rods in the core} = 5248.
 \end{aligned}$$

The free volume was taken from values given from ref. 41 for test fuel elements with the free volume increased by the clearance area times the length difference between the core fuel elements and the test element (27.5 in.), which yielded an additional free volume of 5.55 cm³. Thus the total free volume per rod was taken as 15.65 cm³ per rod.

The internal pressure in the highest power pin would therefore be

$$p_{\text{tot}} = p_{\text{he}} + p_{\text{fp}}, \quad (\text{B.16})$$

with

$$p_{\text{he}} = p_{\text{fill}} \frac{T}{T_{\text{fill}}}, \quad (\text{B.17})$$

and

$$P_{fp} = \frac{\sum_{i=1}^n Y_i F f}{6.02 \times 10^{23} V_{tot}} \frac{RT}{}, \quad (B.18)$$

where

- P_{he} = partial pressure of helium fill gas, atm,
 P_{fp} = summed partial pressures of long-lived and stable fission gases, atm
 P_{fill} = pressure of fill gas before operation, atm,
 T_{fill} = temperature of fill gas when rod was filled, °K,
 T = temperature of fuel element during transient, °K,
 R = gas constant, atm·cm³/g·mole,
 Y_i = yield of isotope i, atoms per fission,
 F = number of fissions during lifetime of core in central pin,
 f = fraction of long-lived and stable isotopes released.

The following values were used in evaluating Eqs. (B.16), (B.17), and (B.18):

P_{fill} , atm	1
T_{fill} , °K	293
F^*	$2.805 \times 10^{22} \times 2$
f	0.02
$Y^{**} (Kr^{83})$	0.005577
$Y (Kr^{84})$	0.0102
$Y (Kr^{85}, 10.5 \text{ y})$	0.003
$Y (Kr^{86})$	0.0207
Total Kr	0.0395
$Y (Xe^{131})$	0.0293
$Y (Xe^{132})$	0.0438
$Y (Xe^{134})$	0.0806
$Y (Xe^{135})$	0.063
$Y (Xe^{136})$	0.0646
Total Xe (less Xe^{135})	0.2183
Total Xe (with 80% Xe^{135})	0.0504
Total Xe	0.2687
Total gases	0.3082

*The number of fissions for the central pin was taken as 2 times that of the average pin based on the radial peak-to-average flux distribution.

**Fractional yield values taken from refs. 42 and 43.

Y (Rb ⁸⁵ , stable)	0.012
Y (Rb ⁸⁷ , 6.2×10^{10} y)	0.027
Total Rb	0.039
Y (I ¹²⁷)	0.0025
Y (I ¹²⁹)	0.0100
Total I	0.0125
Y (Cs ¹³³)	0.065
Y (Cs ¹³⁵)	0.062
Y (Cs ¹³⁷)	0.059
Total Cs	0.186

The contributions of iodine, rubidium, and cesium were considered at temperatures above their respective boiling points. The results are shown in Fig. B.5 converted to units of pressure and temperature.

The cladding stress as a function of temperature was determined by taking the results derived above and presented in Fig. B.5 and using them

Fig. B.5. Internal Pressure in Peak Power Fuel Pin vs Temperature Based on 2% Rare Gas Release and 2% Cs, I, Rb Release, with End of Core Life at 52,000 Mwd of Operation at 69 Mw.

to calculate the tangential cladding stress, as follows:

$$\sigma_t = \frac{\Delta P d}{2t}, \quad (\text{B.19})$$

where

σ_t = tangential cladding stress, psi,

ΔP = pressure differential across cladding ($p_{\text{internal}} - p_{\text{ambient}}$), psi

d = inside diameter of cladding, 0.430 in.,

t = thickness of cladding, 0.035 in.

Substitution of the above values into Eq. (B.19) yielded

$$\sigma_t = 6.15 \Delta P. \quad (\text{B.19a})$$

Computations were made using the internal pressures given in Fig. B.5 and an assumed ambient pressure of 14.7 psia. The results are shown in Fig. B.6.

Fig. B.6. Tangential Cladding Stress in Peak Power Fuel Pin at 14.7-psia Ambient Pressure from 2% Release of Xe, Kr, $I^{127} + I^{129}$, Cs, and Rb, with End of Core Life at 52,000 Mwd of Operation at 69 Mw.

Fuel Element Failure Criterion

The tangential stress vs time to rupture, as determined in experiments performed at the Oak Ridge National Laboratory with type 304 stainless steel,⁴⁴ is shown in Fig. B.7. The highest temperature data available show that at 2200°F and a cladding stress of 1100 psi (obtained from Fig. B.6), the time to failure is 2880 sec. Since the cladding temperature would be rising at the rate of 2°F/sec, it would attain a temperature of 2200°F within 1000 sec after the accident. A temperature of 2550°F would be reached in an additional 175 sec. It is apparent that the cladding would not fail before it reached 2200°F, but it might fail at any time after that. It can be concluded that the use of the original Babcock & Wilcox data for the fraction of core fuel elements reaching 2550°F as a function of time does not overestimate the time to failure for any given fuel element by more than 175 sec. Since this period is small compared with the time values in other parts of this analysis, the cladding was considered to fail upon reaching a temperature of 2550°F. This analysis permitted the use in the present hazard calculation of the original

Fig. B.7. Stress Rupture Properties of Type 304 Stainless Steel Tubing.

Babcock & Wilcox data presented in Fig. B.8, which shows the time required to attain 2550°F at increasing core radii.

Fig. B.8. Time Variation of Radial Temperature Distribution in Reactor Following Loss-of-Coolant Accident.

Appendix C

COMPARISON OF ACTUAL AND SYNTHESIZED DECAY CHAINS^{43,45}

*The numbers under the nuclides refer to cumulative yield in per cent.

In the Xe^{135} case the delay of 15.6 m Xe^{135m} is negligible compared with 6.68 h I^{135} , and therefore the branching ratio from I^{135} to Xe^{135} was taken as one. Also, the simplification to the one precursor model for the I^{131} concentration in the first hour or so after the beginning of the transient results in an overestimate of the I^{131} concentration. Further, Xe^{133m} and Xe^{133} are overestimated slightly in the first several hours after the accident because of the effects of the 52 m Te^{133m} . In calculating I^{133} , it is assumed that the 52 m Te^{133m} is the parent, and the other effects are treated as direct fission yields. As may be seen, no great errors are introduced by these simplifications for time periods in excess of 1 or 2 hr, and errors in the affected nuclides in the shorter time periods are on the conservative (overestimated) side.

Appendix D

BUILDUP OF ACTIVITY IN REACTOR COMPARTMENT

The concentration of activity (neglecting decay) in the reactor compartment approaches an equilibrium value in a manner uniquely determined by the compartment volume and, assuming uniform mixing, its ventilation. The time at which the compartment will be at any given fraction of the equilibrium concentration will be independent of the activity inleakage, although, of course, the absolute value will not. The differential equation for the concentration is

$$W \, dQ = aR \, dt - QV \, dt ,$$

where

W = compartment volume, ft^3 ,

Q = activity concentration, $\mu\text{c}/\text{ft}^3$,

a = activity in containment vessel, curies,

R = leakage rate, $\%/ \text{day}$,

V = ventilation rate, ft^3/min ,

t = time, min.

This equation was differentiated and the results were expressed in terms of the percentage of the equilibrium concentration. The data thus obtained are plotted in Fig. D.1 to show the effect of reactor compartment holdup.

UNCLASSIFIED
ORNL-DWG 63-2355

Fig. D.1. Reactor Compartment Holdup Effect as a Function of Compartment Volume and Ventilation Flow Rate.

Appendix E

POPULATION EXPOSURE AROUND NEW YORK HARBOR

Population exposures resulting from atmospheric dispersion of radioactivity from ground-level release during inversion conditions following a maximum credible accident on the N.S. SAVANNAH were calculated for the harbor of New York City. The operating history prior to the accident was assumed to be that of case 2, that is, 100% power for infinite time followed by 50% power for 5 hr. A meltdown incident was evaluated under each of two conditions of ventilation of the reactor compartment, that is, normal and emergency. These and other controlling conditions during the accident are described below:

Iodine released - 25%, with 50% of remainder held by fuel and 50% deposited in the containment vessel

Noble gases released - 100%

Solids - Since the fuel was assumed to hold 99% of the solids and the effluent from the reactor compartment is filtered, the contribution of solid fission products was low enough to be neglected in calculations of submersion doses.

Primary container leakage - 1.5% per day

Compartment ventilation system exhaust rates

Normal system - 4000 cfm

Emergency system - 2000 cfm, 1800 cfm of this dilution air

Ventilation filtration efficiencies for iodine

Normal system - 90%

Emergency system - 99%

Release - ground level and point source, as in Section 6

It was assumed that the N.S. SAVANNAH was berthed in New York Harbor at either pier 84 or pier 86 in the Hudson River. If the wind carried radioactivity in the most direct path to the public on Manhattan Island, the distance traveled to the nearest uncontrolled area would be about 650 ft. For the purposes of determining total population exposures, this was the distance used. Overall, the plume was assumed to travel in a north-easterly direction so that it would remain over densely populated areas. It would travel over Harlem, the Bronx, and along the Connecticut

coast through Bridgeport. A total downwind distance of 60 miles was considered in estimating the population exposures.

Results of the incident evaluations are presented in Figs. E.1 through E.3. The first figure gives total integrated submersion doses during the first day after the accident. The different modes of activity release are described below:

Fig. E.1. Total Integrated Submersion Dose for Case 2 at New York.

UNCLASSIFIED
ORNL-DWG 63-2357

Fig. E.2. Total Integrated Thyroid Dose for Case 2 with Normal Ventilation at New York.

Mode A is based on continuous equilibrium activity release from the containment system commencing at the instant of the accident. Fission-product decay after the accident, the time required for the fuel elements to heat up and fail, and the holdup time provided by the reactor compartment were ignored.

Fig. E.3. Total Integrated Thyroid Dose for Case 2 with Emergency Ventilation at New York.

Mode B is identical to mode A except that the decay of fission products between the instant of the accident and the time they were released from the compartment to enter the atmosphere was included. This assumption is still somewhat conservative, since it ignores the decay during the time required for the activity to travel from the ship to the exposed personnel.

Mode C includes, in addition to fission-product decay as in mode B, the time required for the fuel elements to fail and the holdup of the ventilation system. Of the various modes considered, this one is thought to be closest to real accident conditions. Mode C calculations were made for both the normal ventilation system and for the emergency system.

Exposure information available from the Figs. E.1, E.2, and E.3 are assembled in Table E.1 for time intervals of 2, 4, 8, and 24 hr after the accident.

Calculation Method

The atmospheric dispersion of radioactivity from ground-level release was calculated from Sutton's equation:

$$\frac{X}{Q} = \frac{2}{\pi C_y C_z \bar{U} x^{2-n}} \exp \left(- \frac{y^2}{C_y^2 x^{2-n}} \right).$$

The meanings of the symbols are given in Table E.2. The release was assumed to take place during a nighttime inversion, the period of poorest atmospheric dilution. The corresponding atmospheric parameters are listed in Table E.3. Isopleths for these conditions are shown in Figs. E.4 and E.5. The fission-product inventory considered was that given in Table 1 of Section 4.

Submersion Exposures

The individual total integrated submersion dose in rems was estimated from the following expression, which was adapted from report ORNL-2867:⁴

$$\text{TID} = 9.4 \times 10^{-4} \frac{X}{Q} \int_0^t E_\gamma Q \, dt.$$

For mode A, E_γ is the effective gamma energy in Mev per disintegration, Q is the source strength in microcuries, and t is the exposure time in hours. The E_γ value was taken as 0.61 Mev, since the noble gas activity predominates under the assumed release conditions. The integral $\int_0^t E_\gamma Q \, dt$ was assumed to equal $E_\gamma Q t$, since the source strength and effective gamma strength were assumed to be constant. The individual dose for any given geographical area was determined by taking the appropriate X/Q factor from Figs. E.4 and E.5 and multiplying it by the amount of activity released in microcuries determined from the product of $9.4 \times 10^{-4} E_\gamma Q t$. The lowest individual exposure considered was

Table E.1. Total Integrated Doses for Case 2 at New York

Time After Accident (hr)	Submersion Doses (man rem)			Thyroid Doses (man rem)			
	No Decay, Holdup, or Cladding Rupture Delay	Decay, Holdup, and Clad- ding Rupture Considered		Normal Ventilation		Emergency Ventilation	
		Normal Ventilation	Emergency Ventilation	No Decay, Holdup, or Cladding Rupture Delay	Decay, Holdup, and Cladding Rupture Considered	No Decay, Holdup, or Cladding Rupture Delay	Decay, Holdup, and Cladding Rupture Delay Considered
2	6,800	3,200	950	64,000	36,000	31,000	8,000
4	13,700	7,200	2,900	128,000	90,000	64,000	27,000
8	27,500	12,700	8,000	256,000	209,000	128,000	74,000
24	82,500	22,800	17,800	768,000	580,000	385,000	262,000

Table E.2. Nomenclature

Symbol	Definition	Units
C_y, C_z	Diffusion coefficients in the y and z planes, respectively	$m^{n/2}$
n	Parameter associated with meteorological stability	Dimensionless
Q	Source strength as total or rate	
	1. Instantaneous source strength, total	curies
	2. Continuous point source emission rate	curies/sec
t	Time	sec
$\bar{U}$	Mean wind velocity	m/sec
x	Distance directly downwind from source	m
y	Distance horizontal distance normal to x	m
z	Vertical distance from ground level (as to center of radioactive cloud or to source, etc.)	m
X	Concentration	curies/ m^3 or curies· m^{-3} ·sec ⁻¹

Table E.3. Atmospheric Parameters for Nighttime Inversion

Site	New York City
Windspeed, $\bar{U}$ (m/sec)	1.64
Diffusion coefficient, C_y	0.19
Diffusion coefficient, C_z	0.09
Stability parameter, n	0.5

Fig. E.4. Isopleths for Typical Inversion Conditions at New York.

Fig. E.5. Isopleths for Typical Inversion Conditions at New York.

0.005 rem. Population submersion exposures were obtained by multiplying the individual exposure by the population density for each given geographical area.

For mode B,

$$\text{TID}_{\text{mode B}} = \text{TID}_{\text{mode A}} \frac{1}{t} \int_0^t \left(\frac{\text{fraction of reference}}{\text{energy release rate}} \right) dt .$$

The fractions of reference energy release rate were obtained from Fig. 5, where the fraction of reference energy release rate is

$$\frac{[\sum E_{\gamma} Q(t)]}{[\sum E_{\gamma} Q(0)]} .$$

Exposures during the first 24 hr following the accident were computed.

For mode C,

$$\text{TID}_{\text{mode C}} = \text{TID}_{\text{mode A}} \frac{1}{t} \int_0^t \left(\frac{\text{fraction of reference}}{\text{release rate}} \right) dt .$$

The fraction of the reference release rate was obtained from Figs. 14 and 15. Again, exposures during the first 24 hr following the accident were computed.

Thyroid Exposure

The total integrated dose per person due to inhalation of any given isotope during a period t is given by:

$$\text{TID} = D \text{ BR } \frac{X}{Q} \int_0^t Q dt ,$$

as adapted from report ORNL-2867.⁴

For mode A, the continuous source strength, Q , released to the atmosphere in curies per second is constant, so $\int_0^t Q dt = Qt$. Values of Q for iodine were taken from Table 1 and adjusted to account for filtering, containment leak rate, plateout, and retention in the fuel; t is the exposure time in seconds; and D is the exposure from the intake of

1 μc of iodine. Values for D were taken from ref. 46 and revised according to the report of committee II on permissible dose for internal radiation.⁴⁷ Values of D for the isotopes important to the thyroid are listed in Table E.4. The term BR is the breathing rate, which was taken as 500 cm^3/sec . Values of X/Q were chosen, as before, for a particular geographical area. Population density information was then applied to yield total thyroid exposures to individuals.

For mode B, thyroid exposures were not calculated. The effect of fission-product decay is expected to be minor during the first few hours after the accident because of the relatively long half-lives of I^{131} and I^{133} , the main contributors.

Table E.4. Thyroid
Exposure from Intake
of 1 μc of Isotope

Isotope	D (mrem)
I^{131}	1484
I^{132}	53.5
I^{133}	399
I^{135}	124

For mode C, the thyroid exposures were

$$\text{TID}_{\text{thy mode B}} = \text{TID}_{\text{thy mode A}} \frac{1}{t} \int_0^t \left(\frac{\text{fraction of reference}}{\text{release rate}} \right) dt .$$

The integral was evaluated numerically from Figs. 16 through 23 for the four iodine isotopes of interest.

Appendix F

EFFECT OF CLOUD SIZE ON SUBMERSION EXPOSURE

The adequacy of the site and the safety of a nuclear installation are determined by the exposures calculated to result as a consequence of various accidents. These exposure calculations involve three major components: direct, inhalation, and submersion. Inasmuch as the submersion exposure is usually at least an order of magnitude lower than the inhalation exposure, the submersion exposure is usually calculated on the presumption of an infinite or semi-infinite "cloud" of uniform concentration. This conservative assumption is reasonable and convenient to make because the resulting exposures are not controlling and because the exact solution to the problem employing a finite cloud with a concentration gradient is difficult mathematically.

In the case of the N.S. SAVANNAH the filtration of the released gases from the outer container reduces the level of the inhalation exposure to the extent that the submersion exposure becomes controlling. In this situation it becomes important to make a more accurate estimate of the submersion exposures.

The determination of the submersion exposure from a finite cloud is discussed in Meteorology and Atomic Energy.¹⁰ Although the derivation is not given, the solution is presented in the form of two nomograms, one for a power excursion release and another for steady-state fission-product release. The ratio of the infinite submersion exposure to the finite submersion exposure as a function of distance from the source was determined from the latter nomogram for meteorological conditions which correspond to the N.S. SAVANNAH maximum credible accident.

The relationship thus determined is plotted in Fig. F.1. The conditions that were assumed in determining this information from the nomograms in Meteorology and Atomic Energy were the following:

Wind speed	1 m/sec
C_x	$0.07 \text{ m}^{n/2}$
C_y	$0.4 \text{ m}^{n/2}$
n	0.5 dimensionless

UNCLASSIFIED
ORNL-DWG 63-2359

Fig. F.1. Ratio of Dose from Infinite Cloud to Dose from Finite Cloud vs Distance from Source.

As may be seen from Fig. F.1, at distances close to the source the use of the actual finite cloud results in calculated exposures much less than those calculated for the infinite cloud. Even as far from the source as 1000 ft (315 m) the finite cloud dose is lower by a factor of approximately 10.

REFERENCES

1. Code of Federal Regulations, Title 10, Part 100, Reactor Site Criteria, Federal Register, 27(71): 3509-3511 (April 12, 1962).
2. "Port Operation in the Matter of the N.S. SAVANNAH," AEC-MARAD Joint Group, 970/5053, Revised, August 1, 1962.
3. Babcock & Wilcox Company, "Nuclear Merchant Ship Reactor Final Safeguards Report," Vol. 1, USAEC Report BAW-1164, 1961.
4. W. B. Cottrell, L. A. Mann, E. L. Parker, and G. D. Schmidt, "Environmental Analysis of the N.S. SAVANNAH Operation at Camden," Rev. 1, USAEC Report ORNL-2867, Oak Ridge National Laboratory, January 24, 1961.
5. "Site Report: York River, Virginia," Maritime Administration and USAEC, May 1960.
6. R. O. Mehann, States Marine Line, Inc., personal communication to T. D. Anderson, Oak Ridge National Laboratory, May 29, 1962.
7. G. R. Thomas, "Nuclear Merchant Ship Reactor Final Safeguards Report Power Plant Accidents," USAEC Report BAW-1164, Vol. VII, Babcock & Wilcox Company, March 1960.
8. T. D. Anderson, "Time-Dependent Release of Gaseous Material from a Three-Barrier Containment System," USAEC Report ORNL-TM-397, Oak Ridge National Laboratory, November 19, 1962.
9. J. DiNunno et al., "Calculation of Distance Factors for Power Test Reactor Sites," USAEC Report TID-14844, March 23, 1962.
10. "Meteorology and Atomic Energy," US Department of Commerce and USAEC Report AECU-3066, July 1955.
11. "Meteorological Factors in the Appraisal and Control of Acute Exposures to Stack Effluents," Proceedings of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy, Geneva, 1958, Vol. 18, United Nations, New York, 1958.
12. J. I. Owens, "Metal-Water Reactors: II, An Evaluation of Seven Nuclear Excursions in Light Water Reactors," USAEC Report GEAP-3178, General Electric Company, June 15, 1959.

13. A. H. Spano et al., "Self-Limiting Power Excursion Tests of a Water Moderated Low-Enrichment UO₂ Core in SPERT I," USAEC Report IDO-16751, Phillips Petroleum Company, Feb. 28, 1962.
14. "Port Analysis Report, Savannah, Ga.," prepared by the AEC-Maritime Administration, Aug. 7, 1962, submitted for the Public Hearing on the interim sea and port operation of the N.S. SAVANNAH.
15. "Port Analysis Report, Norfolk, Va.," prepared by the AEC-Maritime Administration, Aug. 15, 1962, No. 970/4592, submitted for the Public Hearing on the interim sea and port operation of the N.S. SAVANNAH.
16. "Port Analysis Report, Panama Canal," prepared by the AEC-Maritime Administration, Aug. 22, 1962, No. 970/4603, submitted for the Public Hearing on the interim sea and port operation of the N.S. SAVANNAH.
17. "Port Analysis Report, Seattle, Washington," prepared by the AEC-Maritime Administration, August 1962, No. 970/4715, submitted for the Public Hearing on the interim sea and port operation of the N.S. SAVANNAH.
18. "Port Analysis Report, San Francisco," prepared by the AEC-Maritime Administration, September 1962, submitted for the Public Hearing on the interim sea and port operation of the N.S. SAVANNAH.
19. W. B. Cottrell, L. A. Mann, F. L. Parker, and G. D. Schmidt, "Accidental Activity Release from Nuclear Ships," paper in Nuclear Ship Propulsion, International Atomic Energy Agency, 1961.
20. "Site Report: New York Harbor," Maritime Administration and USAEC, May 1962.
21. "Site Report: Savannah, Georgia," Maritime Administration and USAEC, May 1962.
22. "Site Report: Seattle, Washington," Maritime Administration and USAEC, May 1962.
23. "Site Report: Panama Canal," Maritime Administration and USAEC, May 1962.
24. "Site Report: Galveston, Texas," Maritime Administration and USAEC, April 1961.

25. "Recommendations of the International Commission on Radiological Protection," adopted September 9, 1958, Pergamon Press, New York.
26. W. B. Cottrell, L. A. Mann, F. L. Parker, and G. D. Schmidt, "Environmental Analysis of the N.S. SAVANNAH Operation at Camden," Rev. 1, USAEC Report ORNL-2867, Oak Ridge National Laboratory, January 24, 1961, Figs. 4.20-4.23.
27. "Nuclear Merchant Ship Reactor Final Safeguards Report," Vol. VIII, USAEC Report BAW-1164, The Babcock & Wilcox Company, 1961.
28. "Nuclear Merchant Ship Reactor Project, Preliminary Safeguards Report," Vol. I, USAEC Report BAW-1117, Revised, The Babcock & Wilcox Company.
29. W. B. Cottrell et al., "Fission Product Release from UO_2 ," USAEC Report ORNL-2935, Oak Ridge National Laboratory, Sept. 12, 1960.
30. A. H. Booth, "A Method for Calculating Fission Gas Diffusion from UO_2 Fuel and Its Application to the X-2-f Loop Test," Canadian Report CRDC-721, Sept. 1957.
31. J. Belle (Ed.), Uranium Dioxide: Properties and Nuclear Application, Naval Reactors, Division of Reactor Development USAEC, available from Supt. of Documents, USGPO, 1961.
32. W. H. McAdams, Heat Transmission, 3rd ed., McGraw Hill, New York, 1954.
33. Personal communication, T. D. Anderson, Oak Ridge National Laboratory, to M. H. Fontana, Oak Ridge National Laboratory, June 5, 1962.
34. W. D. Kingery et al., "Thermal Conductivity: X. Data for Several Pure Oxide Materials Corrected to Zero Porosity," J. Am. Ceram. Soc., 37: 107 (February 1954).
35. W. D. Kingery, "Thermal Conductivity: XIV. Conductivity of Multicomponent System," J. Am. Ceram. Soc., 42: 617 (December 1959).
36. J. C. Hedge and I. B. Fieldhouse, "Measurement of Thermal Conductivity of Uranium Oxide," AECU-3381 (September 20, 1956).
37. J. Crank, The Mathematics of Diffusion, Claredon Press, Oxford (1956).

38. A. H. Booth and G. T. Rymer, "Determination of the Diffusion Constant of Fission Xenon in UO_2 Crystals and Sintered Compacts," Canadian Report CRDC-720 (August 1958).
39. J. A. L. Robertson et al., "Irradiation Behavior of UO_2 Fuel Elements," Nuclear Metallurgy, 6: 45-70 (1959).
40. Personal communication, L. D. Schaffer, Oak Ridge National Laboratory, to Preston P. Eddy, DRD, USAEC, June 23, 1960, Subject: D' Measurements in UO_2 .
41. D. L. Zimmerman, "Irradiation and Postirradiation Examination of N.S. SAVANNAH Test Fuel Element SL-a," CEAP-3342, General Electric Co., January 28, 1960.
42. W. B. Lewis, "Behavior of Fission-Gases in UO_2 Fuel," DL-45, Chalk River, Ontario, November 1961.
43. J. O. Blomeke and Mary F. Todd, "Uranium-235 Fission-Product Production as a Function of Thermal Neutron Flux, Irradiation Time and Decay Time. I. Atomic Concentrations and Gross Total," Part I, Vol. 1, USAEC Report ORNL-2127, Oak Ridge National Laboratory, December 1958.
44. J. R. Weir, Jr., Oak Ridge National Laboratory, as reported by T. H. Row, "EGCR Fuel Failures in Depressurization Accidents," unpublished report, April 16, 1962.
45. Seymour Katcoff, "Fission Product Yields from Neutron-Induced Fission," Nucleonics, pp. 201-208, November 1960.
46. T. S. Burnett, Nucl. Sci. Eng., Vol. 2, 1957.
47. Health Phys., 3(3), 1960.

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

ORNL-3361
 UC-80 - Reactor Technology
 TID-4500 (22nd ed.)

Internal Distribution

1. R. G. Affel	22. G. W. Keilholtz
2. T. D. Anderson	23. O. H. Klepper
3. C. A. Baumann	24. H. G. MacPherson
4. S. E. Beall	25. H. C. McCurdy
5. J. R. Buchanan	26. J. R. McWherter
6. T. J. Burnett	27. E. C. Miller
7. C. D. Cagle	28. F. L. Parker
8-12. W. B. Cottrell	29. G. W. Parker
13. J. A. Cox	30. J. J. Pinajian
14. F. L. Culler	31. A. W. Savolainen
15. D. M. Davis	32. R. P. Shields
16. W. K. Ergen	33. M. J. Skinner
17. B. R. Fish	34. G. M. Watson
18. M. H. Fontana	35-37. Central Research Library
19. D. R. Gilfillan	38-40. Document Reference Library
20. A. A. Grau	41-62. Laboratory Records
21. C. E. Guthrie	63. Laboratory Records (RC)

External Distribution

- 64. Research and Development Division, ORO
- 65-66. Reactor Division, ORO
- 67-70. Division of Reactor Development (Maritime Reactors), AEC,
Washington (Attention: J. E. Robb)
- 71. Division of Reactor Development (Nuclear Safety), AEC, Washington
(Attention: Merson Booth)
- 72. George G. Sharp, Inc., New York (Attention: D. L. Gorman)
- 73-74. States Marine Lines, Inc., Galveston, Texas, c/o Todd Shipyard
Corp. (Attention: R. O. Mehann)
- 75-76. Todd Shipyard Corp., Nuclear Division (Attention: B. W. Winchell)
- 77-78. Babcock & Wilcox Company, Atomic Energy Division, Lynchburg, Va.
(Attention: W. F. Long)
- 79. Babcock & Wilcox Company, Atomic Energy Division, Lynchburg, Va.
(Attention: John H. MacMillan)
- 80-81. Division of Reactor Development (Army Reactors), AEC, Washington
(Attention: Col. R. B. Burlin)
- 82-83. Nuclear Power Field Office, Ft. Belvoir, Virginia (Attention:
L. J. Misenheimer)
- 84-85. Martin-Marietta Company, Nuclear Division (Attention:
J. F. O'Brien)
- 86. Commandant (MMT) U.S. Coast Guard, 1300 E. Street, N.W.,
Washington, D.C.
- 87. Ebasco Services, Inc., New York (Attention: W. C. Stamm)

- 88. Zeluine Levine, The Martin-Marietta Co., Nuclear Division
Baltimore, Md.
- 89. R. E. Wascher, The Babcock & Wilcox Company, Atomic Energy
Division, Lynchburg, Va.
- 90-91. R. P. Denise, N. S. Savannah Technical Staff, Box 1750,
Galveston, Texas
- 92-704. Given distribution as shown in TID-4500 (21st ed.) under
Reactor Technology (75 copies - OTS)